JANUARY 1999

REVISION 0C

NAC-MPC SAFETY ANALYSIS REPORT

for the

NAC Multi-Purpose Canister System

Docket No. 72-1025

9901220254 990109
PDR ADOCK 05000029
Y PDR

NAC INTERNATIONAL

655 Engineering Drive Norcross, Georgia 30092 Phone 770-447-1144 Fax 770-447-1797

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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 [redacted]. The NAC-MPC system consists of a transportable storage canister, vertical concrete cask, and a transfer cask. [redacted]

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 (DOE) 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 [redacted].

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 [redacted].

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. [redacted]

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).
[redacted] 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 [redacted] basket.
Contents	Up to 36 pressurized-water reactor (PWR) fuel assemblies [redacted] in the transportable storage canister.
Canister Basket	The structure placed in the transportable storage canister to support the fuel assemblies (fuel basket) [redacted].
- 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 [redacted] 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.

Table 1-1 Terminology (Continued)

- 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.
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.
Structural Lid	The secondary confinement boundary for the canister. The structural lid provides the lifting point for the loaded canister.
Quick Disconnect	The quick-disconnect valved nipple used in the vent and drain ports to facilitate operations.

Table 1-1 Terminology (Continued)

[redacted]

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	A component having the same external dimension as a standard fuel assembly that ensures geometry control and confinement of Yankee Class spent fuel having cladding defects. The Reconfigured Fuel Assembly can contain a maximum of 64 full length fuel rods.
Failed Fuel	Individual Yankee Class spent fuel rods having cladding defects identified by inspection or testing; or rod sections having gross cladding defects.
Vertical Concrete [redacted] Cask [redacted]	A [redacted] concrete [redacted] closed at the top end by a shield plug and lid that [redacted] the [redacted] canister [redacted] during storage. The [redacted] 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.

[redacted] [redacted]

- 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.

- Transfer Cask 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 & 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 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 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 design-basis earthquake is specified in Section 2.2.3 in accordance with 10 CFR 72.102 criteria.
	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)]		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.
	The DCSS must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under all credible normal, off-normal, and accident conditions. [10 CFR 72.124(a) and 72.236(c)]		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 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)]		The maximum allowable cladding temperatures are specified in Tables 2.1-1 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.
	Storage systems must be designed to allow ready retrieval of spent fuel waste for further processing or disposal. [10 CFR 72.122(l)]		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)]	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.
	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)]		Section 10.4 presents the necessary minimum site boundary distances from an array of loaded storage systems to meet the controlled area dose limits.
	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)]		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 DCSS must be designed to 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.
	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)]		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.
	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)]		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 a. 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 systems 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)]	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.
	Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal. [10 CFR 72.122(l)]		The procedures include methods for retrieving the spent fuel after storage for off-site transport or for return to the spent fuel pool.
	The DCSS must be designed to minimize the quantity of radioactive waste generated. [10 CFR 72.24(f) and 72.128(a)(5)]		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 applicant must describe equipment and processes proposed to maintain control of radioactive effluents. [10 CFR 72.24(l)(2)]		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.
	To the extent practicable, the DCSS must be designed to facilitate decontamination. [10 CFR 72.236(i)]		The canister is designed to facilitate decontamination, as described in Section 2.3.5.3.
	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)]		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(l) 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**
I. 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 1, 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**
1. 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 [however, 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 st[illegible]e 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	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 72.24(d)	Contents of Application: Margins of Safety / Mitigation of Accident Consequences	
	10 CFR 72.122(h)(1)	Overall Requirements: Confinement Barriers and Systems	The temperatures of the system are maintained within allowable limits, and do not preclude retrieval of spent fuel from the system.
	10 CFR 72.122(l)	Overall Requirements: Retrievability	
	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	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(g)	Specific Requirements for Spent Fuel Storage Cask Approval: Minimum 20-year Lifetime	
	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-21118)	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, with a detailed description of the shielding features of the system 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	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 [illegible]lity evaluation of the storage cask for the most reactive [illegible]edible 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, no release of radionuclides to the environment is credible.

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.

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 3 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 7.1.1.5, 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.
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 Section 8.2, 8.3 specify the steps necessary for retrieval of the spent fuel from the system for further processing or disposal.

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

Chapter 8 – Operating Procedures		
Area	**Regulatory Requirement**	**Description of Compliance**
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.

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 2.[illegible] and [illegible], 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.[illegible], 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	The description of the radiation protection and ALARA considerations of the system are provided in Section 10.1.
	10 CFR 72.24(e)	Contents of Application: ALARA Features	
	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 doses 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	At 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 Lim[illegible] r 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 [illegible] der credible conditions (i.e., k_{eff} equal to or less than 0.95). At [illegible] d 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 [illegible] 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 72.28]:	
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.

[redacted]

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) [redacted]. 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 [redacted] fuel [redacted]. The NAC-MPC has been designed and analyzed for a [redacted]-year life.

The principal components of the NAC-MPC system are the canister, the vertical concrete [redacted] 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 Yankee Class spent fuel assemblies [redacted]. The fuel [redacted] is initially loaded into a canister containing [redacted] basket. Figure 1.1-2 depicts the canister and the spent fuel basket. [redacted]

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

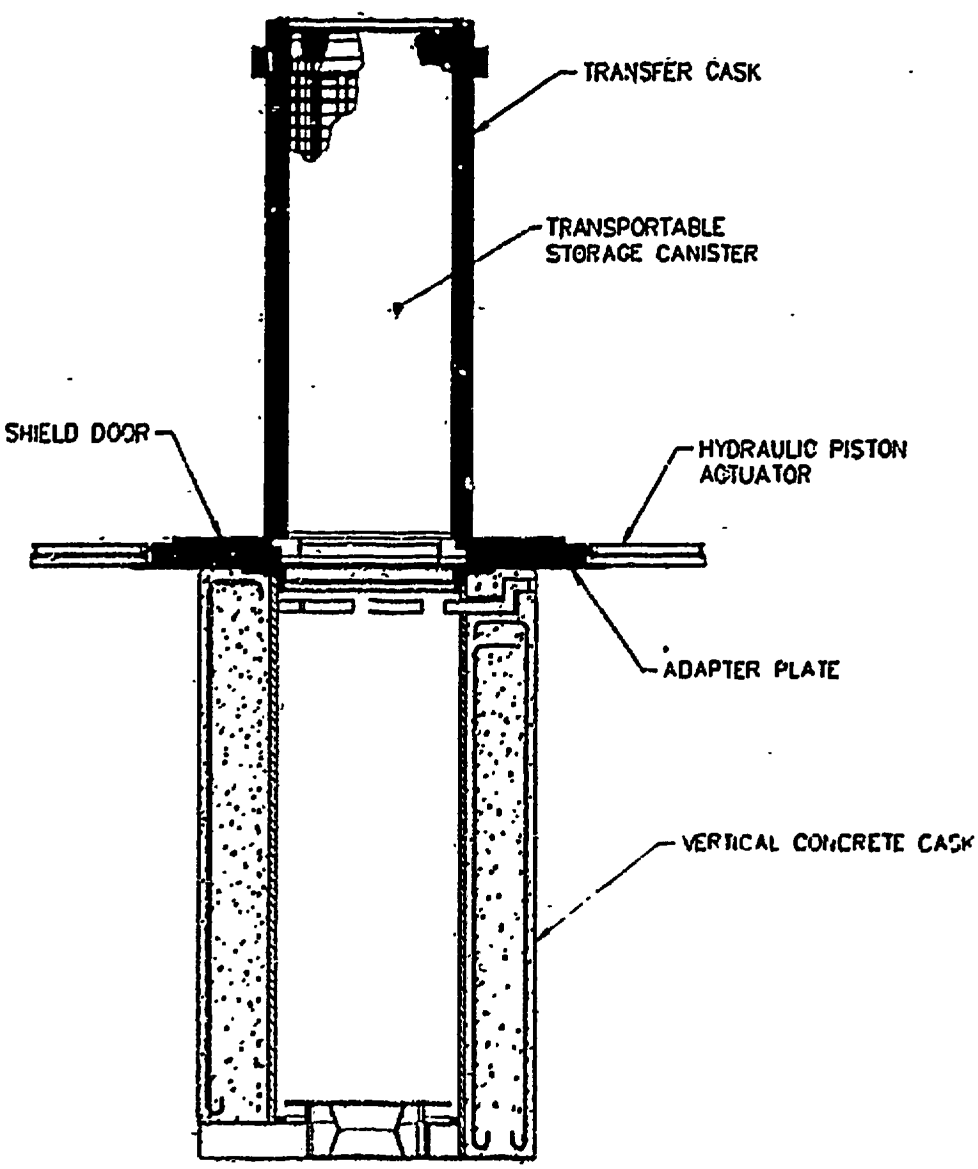
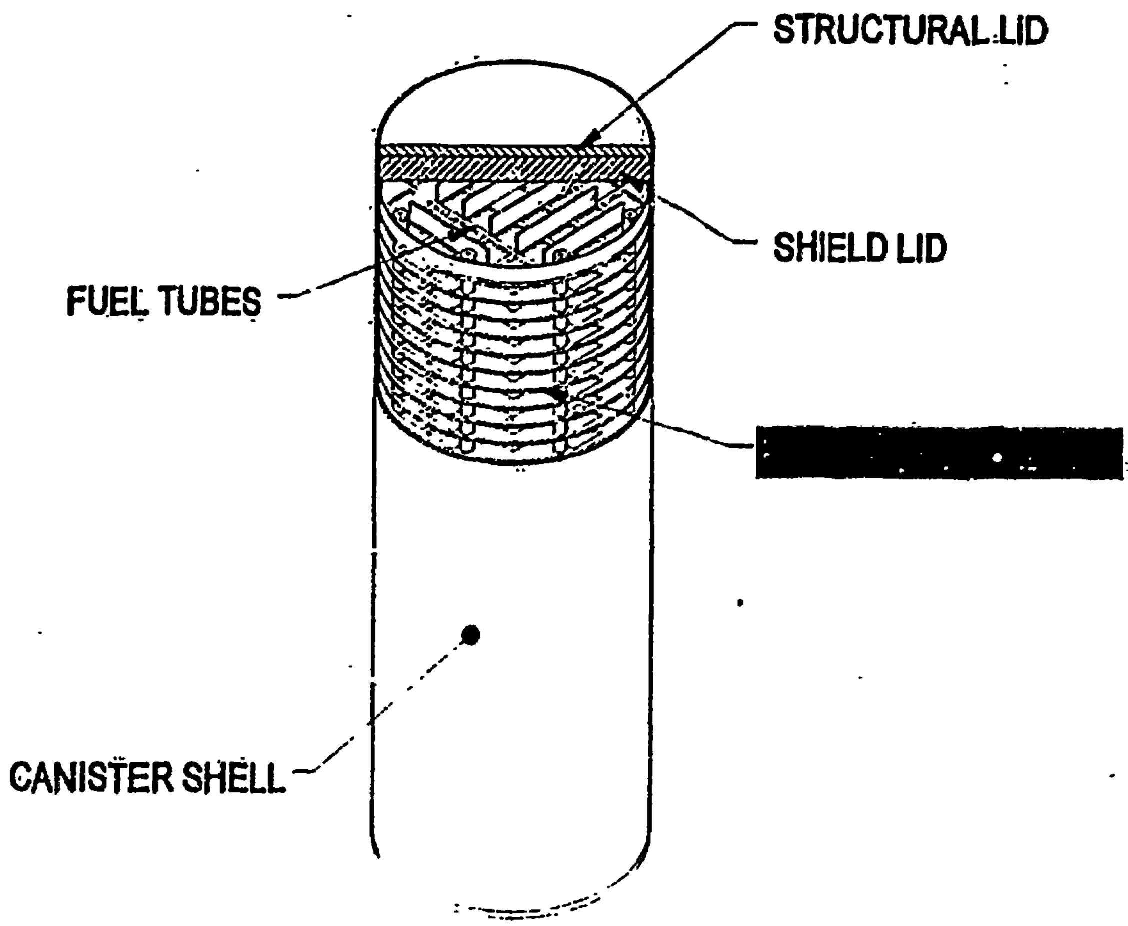


Figure 1.1-2 Transportable Storage Canister Showing the Spent Fuel Basket



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1.2 The NAC-MPC System

The NAC-MPC system provides long-term storage [redacted] and subsequent transport [redacted] 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);
- Alignment plates 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). Transport conditions established the design basis load conditions for the canister, except for canister lifting. 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 [redacted] a basket that accommodates up to 36 Yankee Class [redacted] fuel assemblies. [redacted]

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 [redacted]. 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 [redacted] 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. [redacted]

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 [redacted] 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 from 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.

[redacted]

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 with 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 film 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, [redacted].

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. [redacted]

1.2.1.5 Transport Cask

The transportable storage canister is designed to be transported in the NAC-STC. The canister [redacted] is positioned in the NAC-STC cavity with two axial spacers. The spacers are required because the [redacted] 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 [redacted] 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 [redacted]; 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 [redacted] lower the cask to the bottom of the pool.
- Load the selected spent fuel assemblies [redacted] 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 welds. 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 [redacted] fuel assemblies [redacted]. The Yankee Class fuel consists of fuel assemblies manufactured by Westinghouse, United Nuclear, Exxon, and Combustion Engineering. The assemblies vary in initial enrichment from [redacted] to 4.94 w/o ^{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 [redacted] fuels are presented in Table 1.2-6. [redacted]

A canister may contain one or more Reconfigured Fuel Assemblies designed to confine Yankee Class spent fuel rods, or portions thereof, which are classified as failed, and to maintain the geometric configuration of the rods. It is designed to confine failed fuel during all storage and transport conditions. Since there is no significant remaining "gap activity" in the failed rods, pressure retention is not a concern. 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 [redacted].

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 [redacted] fuel assembly. These effects are evaluated in the appropriate chapters that follow.

[redacted]

[redacted]

Figure 1.2-1 Vertical Concrete Storage Cask

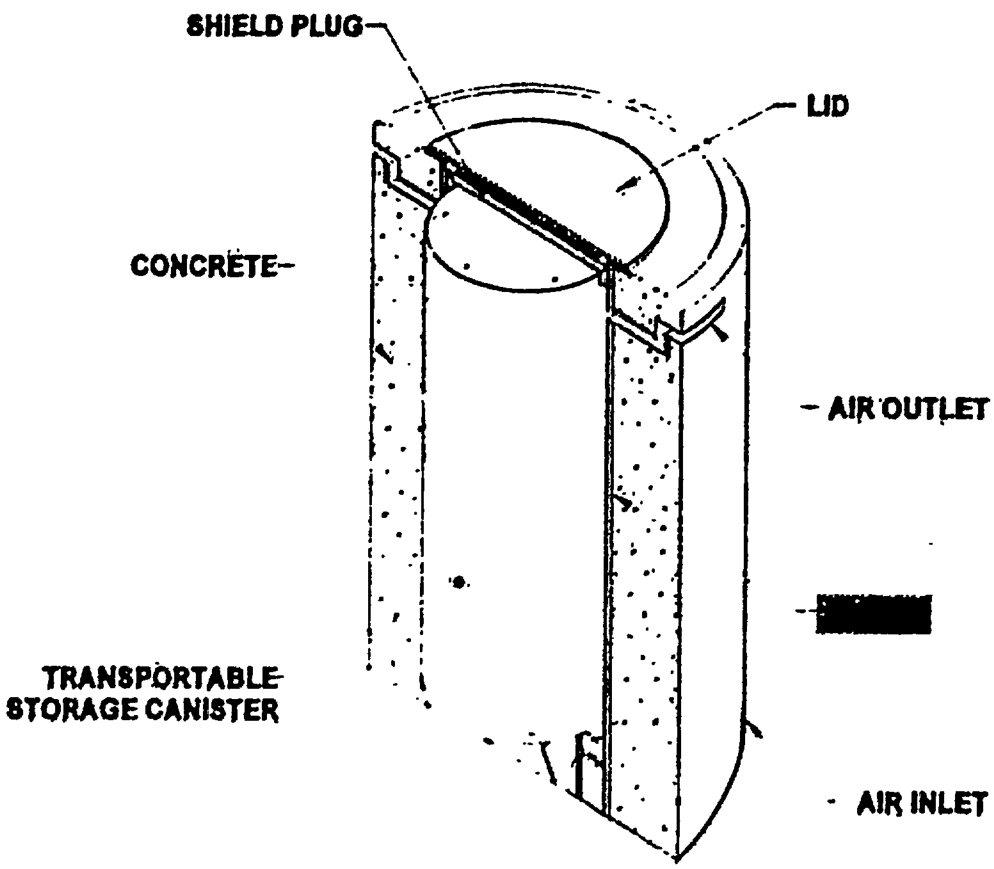


Figure 1.2-2 Transfer Cask

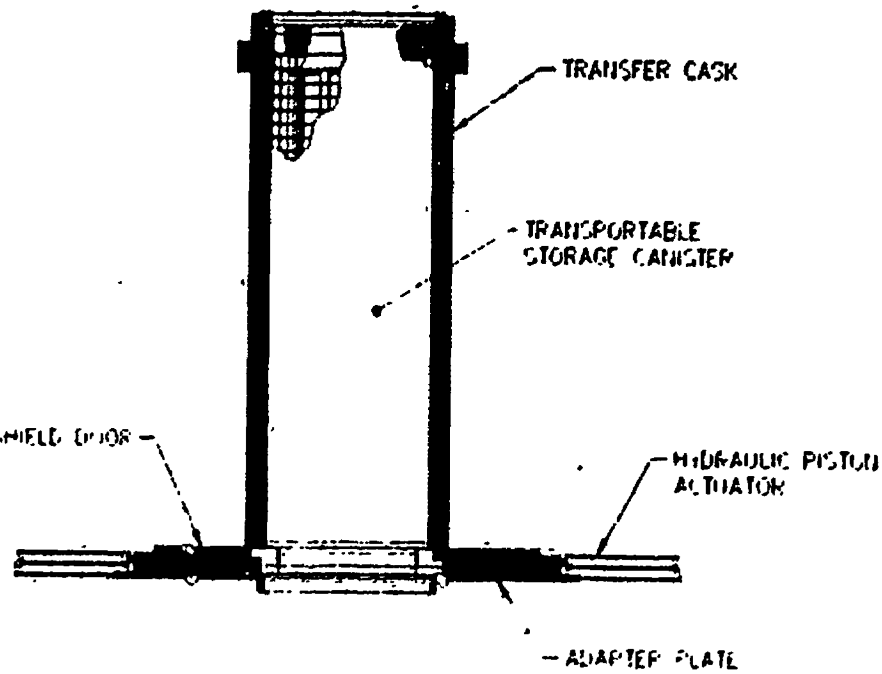


Figure 1.2-3 NAC-STC Transport Configuration

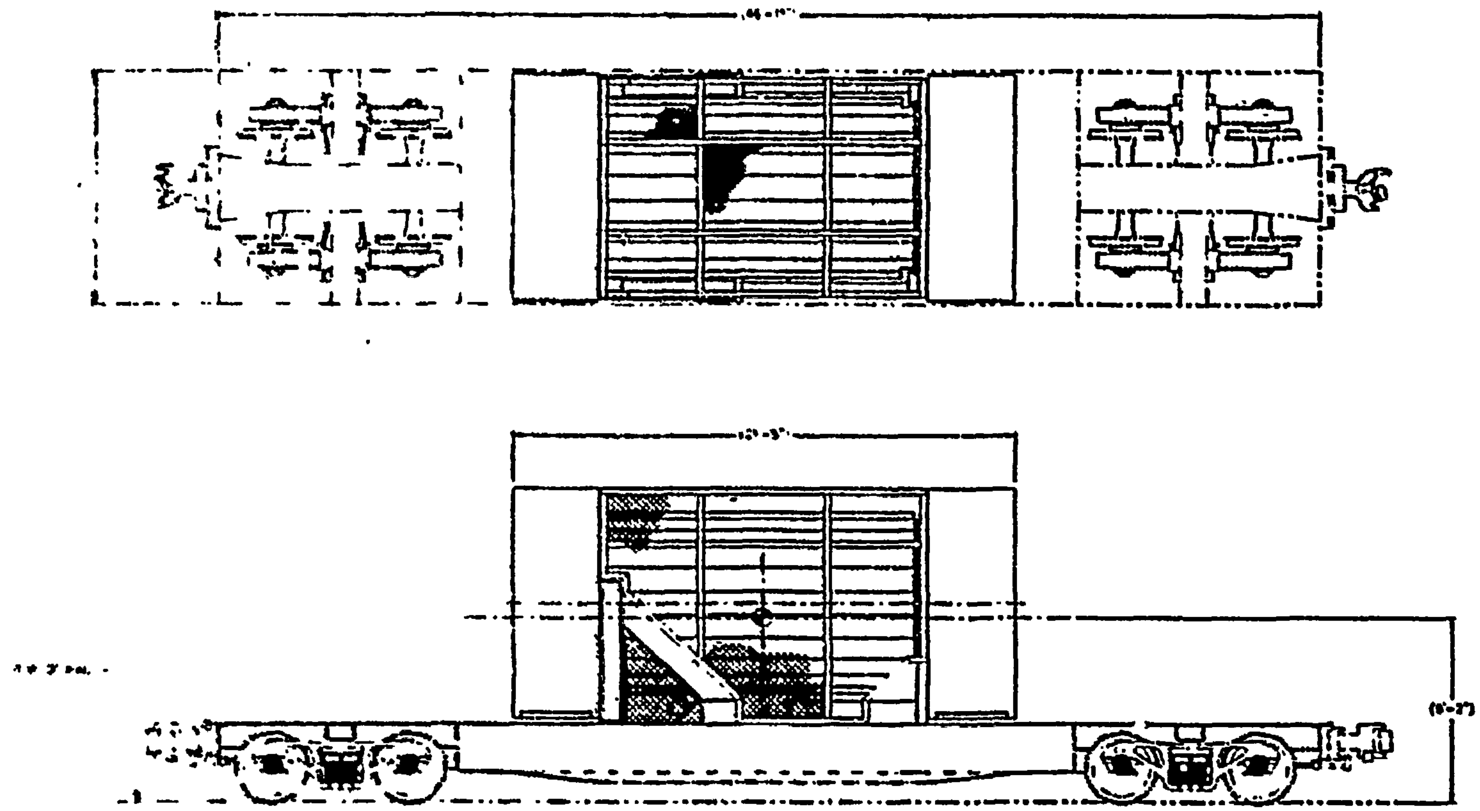
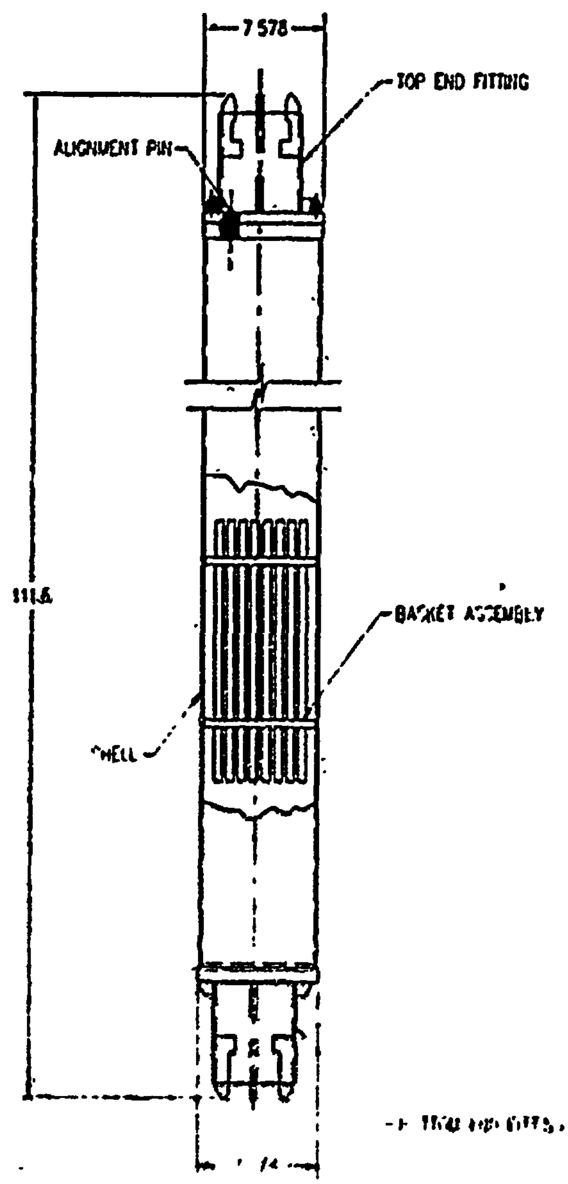
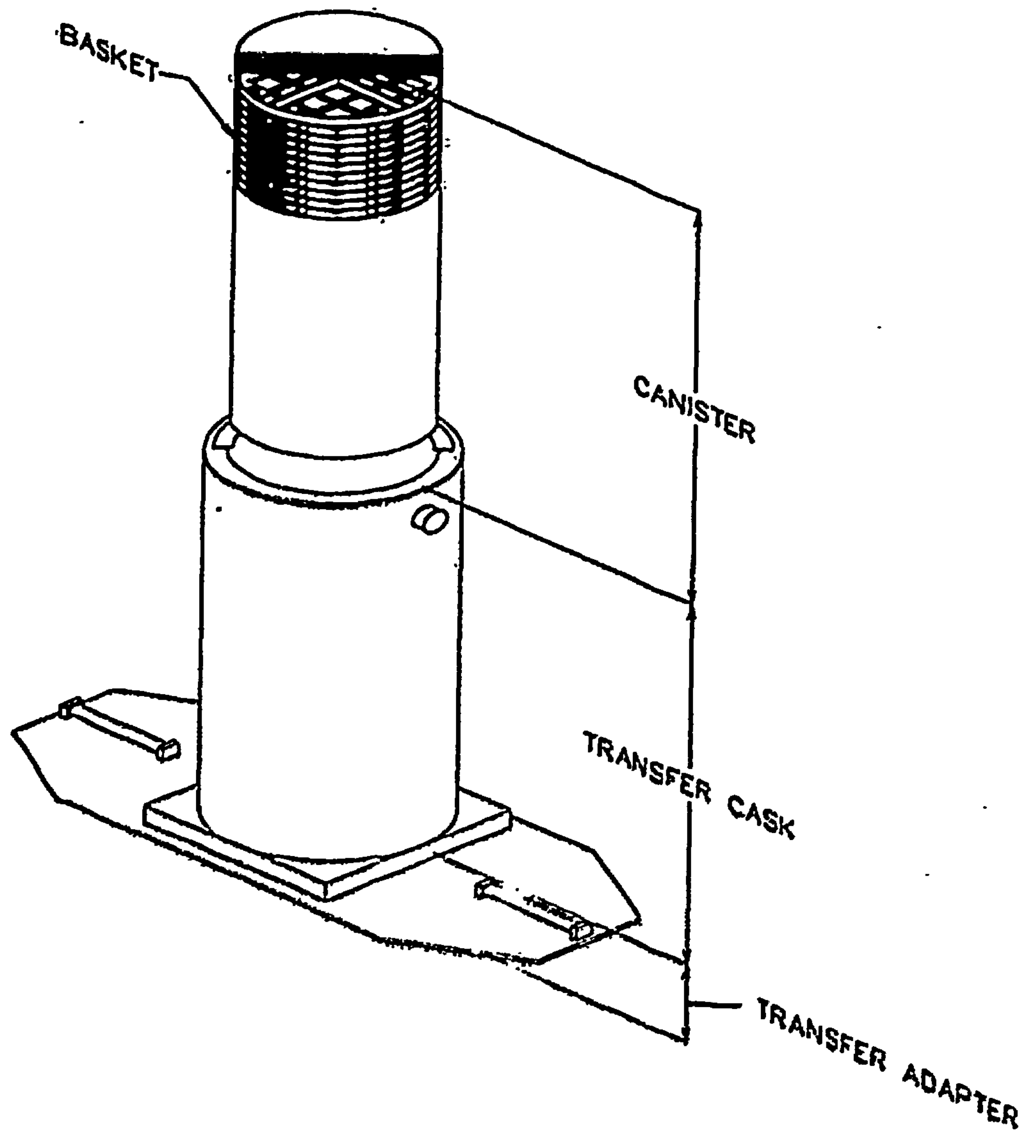


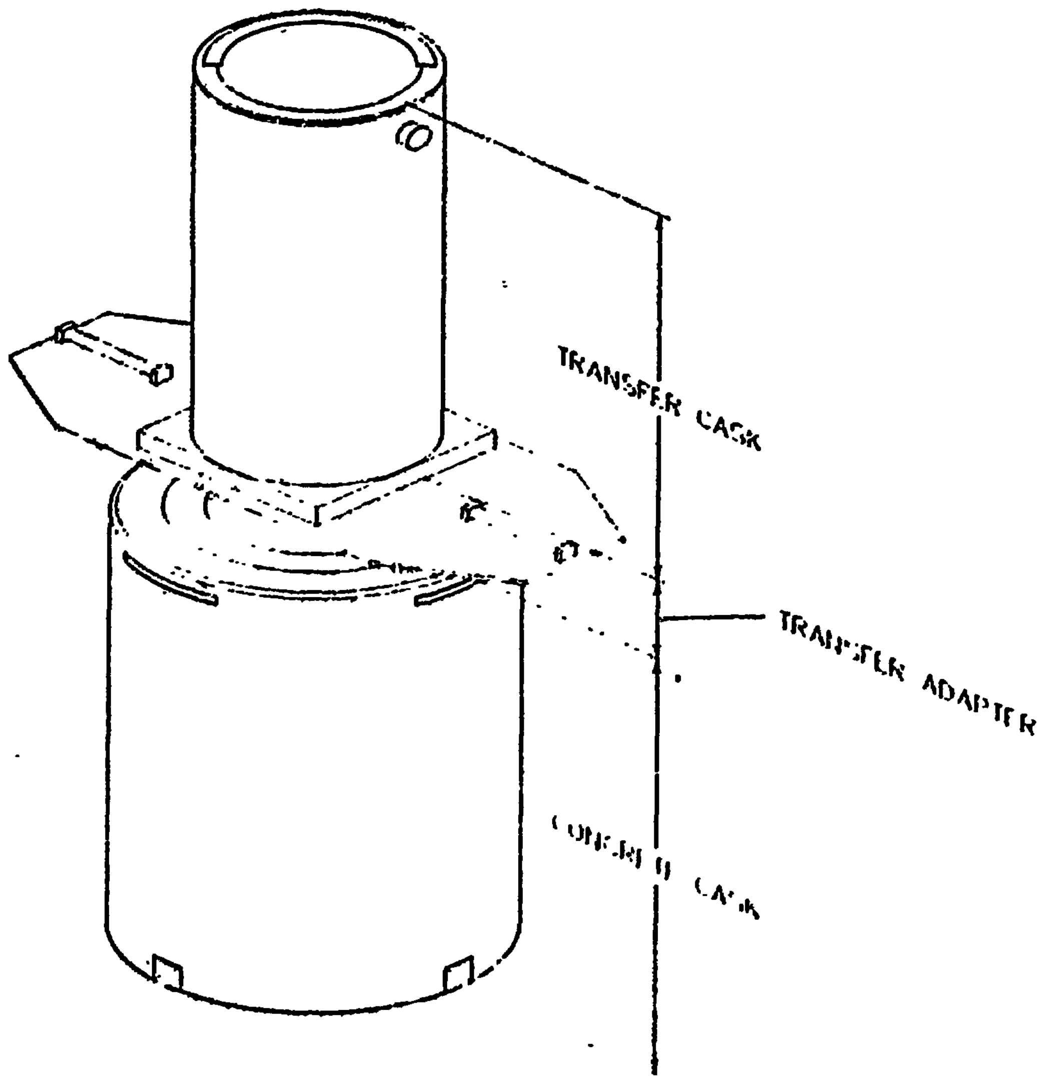
Figure 1.2-[redacted] Reconfigured Fuel Assembly



[redacted]



[redacted]



[redacted]

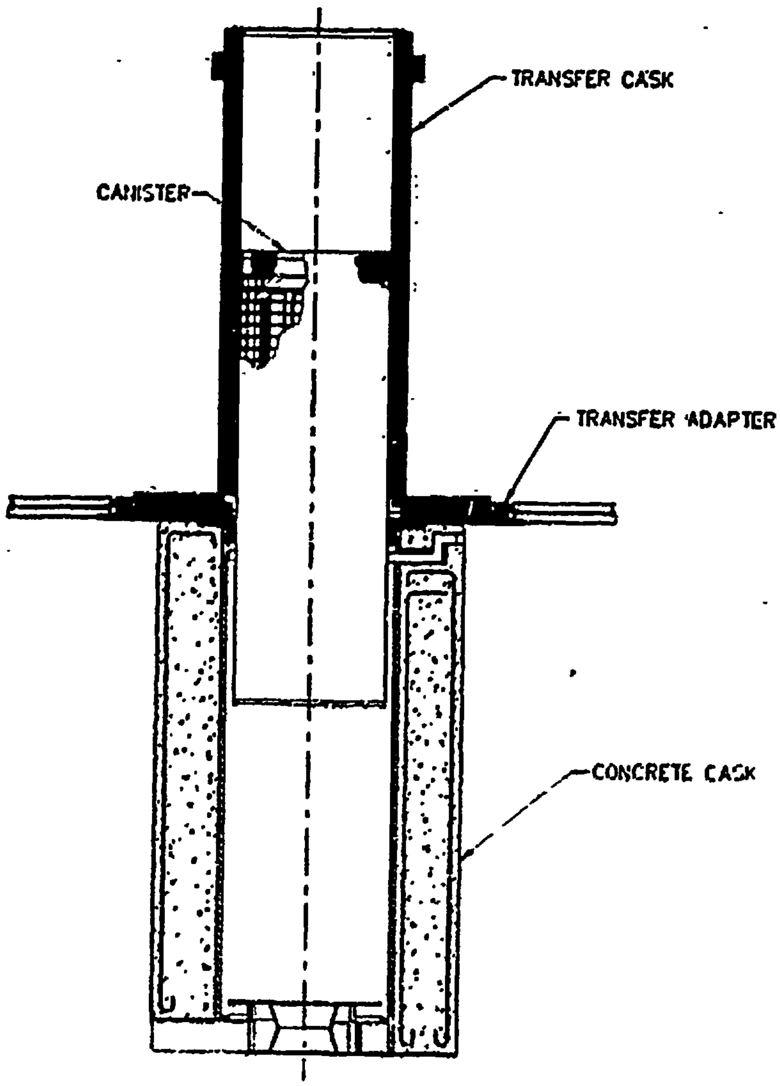


Table 1.2-1 Major Physical Design Parameters for the Transportable Storage Canister [redacted]

Parameter	Value	
	Transportable Storage Canister	[redacted]
Outside Diameter	70.64 in.	[redacted]
Length	122.5 in.	[redacted]
Capacity	36 Yankee Class [redacted] fuel assemblies [redacted]	[redacted]
Weight	54,730 lbs. (nominal) w/ fuel [redacted]	[redacted]
Maximum heat load	12.5 kW (fuel) [redacted]	[redacted]
Maximum fuel [redacted] temperature [redacted]	[redacted]	[redacted]
Internal atmosphere	Helium	[redacted]

Table 1.2-2 Transportable Storage Canister [redacted] 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 accord[illegible]ith 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 [redacted]
- All welds specified to be dye 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 [redacted] ASME Section V, Article 5, with acceptance [redacted] 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 [redacted] 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 [redacted] the specifications and drawings.

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

Parameter	Value
Height	[redacted] in.
Outside diameter	128 in.
Shielding (side wall) Concrete thickness Steel thickness	 21 in. 3.50 in.
Radiation dose rate ([redacted]): Side surface Top surface Air inlet/ outlet vents	 ≤ 50 mrem/hr ≤ [redacted] 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.
- Minimum 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 [redacted]	≤ 200 mrem/hr

Table 1.2-6 NAC-MPC Design Basis Fuel Characteristics

Parameter	Yankee Class [redacted] Fuel[1,2] [redacted]		
	United Nuclear Type A	Combustion Type A	Westinghouse Type B
Number of Assemblies per Canister	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
[redacted]	[redacted]	[redacted]	[redacted]
Maximum Burnup, MWD/MTU	32,000	36,000	[redacted]
Maximum Assembly Decay Heat, kW	0.347	0.347	[redacted]
Maximum Decay Heat, kW	[redacted]	12.5	[redacted]
Minimum Cool Time, yr	[redacted]	[redacted]	[redacted]

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. [redacted]

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

[redacted]

[redacted]

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

[redacted]

[redacted]

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

FIGURE WITHHELD UNDER 10 CFR 2.390

1.5 Licence 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-[redacted]	0	1
455-856	Name Plate [redacted]	0	1
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
455-859	Assembly, Transfer Adapter, MPC[redacted]	[redacted]	3
455-860	Assembly, Transfer Cask (TFR), MPC [redacted]	[redacted]	4
455-861	Weldment, Structure, Vertical Concrete Cask (VCC), MPC [redacted]	[redacted]	2
455-862	Loaded Vertical Concrete Cask (VCC), MPC [redacted]	[redacted]	[redacted]
455-863	Lid, Vertical Concrete Cask (VCC), MPC[redacted]	[redacted]	[redacted]
455-864	Shield Plug, Vertical Concrete Cask (VCC), MPC[redacted]	[redacted]	1
455-866	Reinforcing B[illegible] [redacted] Concrete Placement, Vertical Concrete Cask (VCC), MPC[redacted]	0	3
455-870	Canister Shell, MPC[redacted]	[redacted]	1
455-871	Details, Canister, MPC[redacted]	[redacted]	2
455-872	Assembly, [redacted] Storage Canister (TSC), MPC[redacted]	[redacted]	1
455-873	Assembly, Drain Tube, Canister, MPC[redacted]	[redacted]	1
455-881	PWR Fuel Tube, Captivated BORAL, MPC[redacted]	[redacted]	1
455-887	[redacted]		
455-888	[redacted]		
455-891	Bottom Weldment, Fuel Basket, MPC[redacted]	0	1
455-892	Top Weldment, Fuel Basket, MPC[redacted]	1	1
455-893	Support Disk and Misc. Basket Details, MPC[redacted]	[redacted]	1
455-894	Heat Transfer Disk, Fuel Basket, MPC[redacted]	[redacted]	1
455-895	Fuel Basket Assembly, MPC[redacted]	[redacted]	1

[redacted] [redacted]

[redacted]		[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

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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 [redacted] fuel [redacted].

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
Transfer Cask	ANSI N14.6 and NUREG-0612
Design Weight: Canister Assembly w/fuel [redacted] Transfer Cask Vertical Concrete Cask	 54,730 lbs. 80,743 lbs. 151,364 lbs.
Thermal: Maximum Temperature, [redacted] [redacted]ture, [redacted] Cladding Ambient Temperature Range Average Annual Ambient Temperature Cone Temperature: Normal Conditions Off-Normal/Accident Conditions	 340°C for 10-yr. Cooled 380°C for 5-yr. Cooled 570°C Off-Normal/Accident/Transfer 340°C [redacted] 430°C [redacted] -40° to 125°F 75°F ≤ 150°F; ≤ 200°F local ≤ 350°F local/ surface
[redacted] 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. (max)
Concrete Cask Top Lid Contact Dose Rate	< 55 mrem/hr. (max)
Concrete Cask Air Inlet/Outlet	< 100 mrem/hr. (max)
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, [redacted] 3.9 wt % ^{235}U Exxon 16 x 16, [redacted] 4.0 wt % ^{235}U
Fuel Cladding	Stainless Steel - Westinghouse Zircaloy - All others
Spent Fuel Capacity (May include one or more Reconfigured Fuel Assemblies)	36 United [redacted] 36 Combustion [redacted] 36 Exxon [redacted] 34 Westing[redacted] 36 Fuel [redacted] 20,600 [redacted]
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 0.107 kW

1. Based on the design basis, Combustion Engineering fuel at 36,000 MWD/MTU [redacted] limited to 24,000 MWD/MTU and 10 years minimum cool time. The [redacted] is 32,000 MWD/MTU.

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2.1 Spent Fuel [redacted] To Be Stored

The NAC-MPC has been designed to safely store up to 36 Yankee Class spent fuel assemblies[redacted]. 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 pins. 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 [redacted] parameters of each fuel design type. [redacted]

[redacted]

2.1.1 Bounding Fuel Evaluation

The criticality evaluations [redacted] that the United Nuclear Type A 16 x 16 fuel assembly is the most reactive [redacted]

[redacted] shielding evaluations [redacted] that the Combustion Engineering Type A has the largest [redacted] 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 [redacted] years after reactor discharge. [redacted] Combustion Engineering [redacted] evaluated for a source term based on an initial [redacted] enrichment of [redacted], a maximum burnup of 36,000 megawatt days per metric ton of uranium, and a minimum cool time of [redacted] years after reactor discharge [redacted]

[redacted]

2.1.[redacted] Reconfigured Fuel Assembly

One or more transportable storage canisters may hold Reconfigured Fuel Assemblies containing spent fuel rods classified as failed fuel. The Reconfigured Fuel Assembly may consist of up to 64 rod segments or whole rods having cladding defects. 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. [redacted]. The [redacted] hole allows the draining and drying of the individual tubes during routine closing of the canister. The [redacted] precludes the release of gross particles to the canister. The effects of the container of failed fuel are evaluated in the appropriate sections. The structural, thermal, shielding, [redacted], and criticality effects of the Reconfigured Fuel Assembly are bounded by those of an intact fuel assembly.

[redacted]

[redacted] corrosion, stress corrosion cracking, localized corrosion, mechanical [redacted]-based failure mechanisms. The EPRI report is based on [redacted] steel cladding, including 304, modified 348, 348 and 348H, which [redacted]

[redacted] limit is conservatively [redacted] NAC-MPC [redacted] steel and Zircaloy-clad fuel. This [redacted] than the [redacted] by the EPRI report for [redacted]. The short-term [redacted] limit for stainless steel-clad fuel is 430°C for the NAC-MPC System. This [redacted] also conservatively used for Zircaloy-clad fuel.

Table 2.1-1 Yankee Class Fuel Parameters

	Combustion Engineering	Combustion Engineering	Exxon	Exxon	Exxon	Exxon	Westinghouse	Westinghouse	United Nuclear	United Nuclear
Assembly Configuration	Type A	Type B	Type A	Type B	Type A	Type B	Type A	Type B	Type A	Type B
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. % [redacted]U										
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	4.94	4.94	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Initial Fuel Weight (KgUO2/Assembly)	264.8	261.1	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
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
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 [illegible] years [redacted] MWD/MTU and minimum enrichment of 3.5 wt%. The maximum decay heat [redacted].

Table 2.1-1 Yankee Class Fuel Parameters (Continued)

	Combustion Engineering	Combustion Engineering	Exxon	Exxon	Exxon	Exxon	Westinghouse	Westinghouse	United Nuclear	United Nuclear
Guide Bar Configuration										
Guide Bar Material	Zircaloy-4	Zircaloy-4	SS 304L	SS 304L	Zircaloy	Zircaloy	[illegible]	[illegible]	N/A†	N/A†
Guide Bar Width (cm)	1.1	1.1	1.1	1.1	1.1	1.1	[illegible]	[illegible]	N/A†	N/A†
Guide Bar Length (cm)	245.2	245.2	244.6	244.6	244.6	244.6	[illegible]	[illegible]	N/A†	N/A†
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	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	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	[illegible]	6.70	6.70	6.70	6.70	6.70	6.70	17.02§	17.02§
Bottom Nozzle Weight (kg)	9.10	9.10	5.18	5.18	5.18	5.18	5.20	5.20	13.21§	13.21§
Upper Plenum Spring Weight (g)	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	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‡	Zirc-4/Inconel 625‡	SS 304L	SS 304L	Zircaloy-4	Zircaloy-4	N/A	N/A	Inconel 718	Inconel 718
Grid Spacer Weight (g)	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	618.4	N/A	N/A	0	0
Number of Grid Spacers	[illegible]	6	6	6	6	6	N/A	N/A	6	6

† United Nuclear assemblies fabricated from Zircaloy incorporate displacement rods [illegible] inches long and 0.365-inch in outside diameter

‡ Five grid spacers are Zircaloy 4. The bottom spacer is Inconel 625

§ Estimated weight

Table 2.1-2 Yankee Class Reconfigured Fuel Assembly Parameters

	CE	Exxon	Exxon	West.	United Nuclear
Parameter	Type A/B	Type A/B	Type A/B	Type A/B	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	235.9975	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.

2.2 Design Criteria for Environmental Conditions and Natural Phenomena

The design criteria defined in this [redacted] 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 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 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 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 and ANSI 57.9 for the remainder of the structure. The criteria are:

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

The normal, long-term design temperature was selected to model the expected average ambient to bound most annual average temperatures seen by a cask over its lifetime. 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 assuming half-blockage of the air inlets and the complete blockage of the air inlets and outlets. 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
Transitional 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_a	E_s			
6	Accident	D	L		T_a			A	
7	Accident	D	L		T_a				F
8	Accident	D	L		T_a		W_t		

Load Combinations are from ANSI 57.9 and ACI 349.

D	=	Dead Load	T_a	=	Off-Normal or Accident Temperature
L	=	Live Load	E_s	=	Design Basis Earthquake
W	=	Wind	W_t	=	Tornado/Tornado Missile
T_o	=	Normal Temperature	A	=	Drop/Impact
F	=	Flood			

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	$P_m \leq S_m$ $P_L + P_b \leq 1.5\ S_m$ $P_L + P_b + Q \leq 3S_m$
	[redacted] Lifting Devices ANSI N14.6 and NUREG 0612	Redundant load path: combined shear or max. tensile stress $\leq S_u/5$ [redacted] $S_y/3$
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 [redacted]) $P_L + P_b < 1.8\ S_m$ 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[illegible] 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 and final pass. 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 [illegible] 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 [redacted]. The design maximum dose rate at the top of the canister structural lid, with supplemental shielding, is 200 mrem/hr [redacted] 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	[illegible]	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	3	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	[redacted]	3	Location Nut	Operations	B
				2	Bottom	Structural, [redacted]	A
				1	Shell	Structural, [redacted]	A
455-871	1	Details, Canister	[redacted]	8	Key	Operations	C
				7	Port Cover (Two)	[redacted] 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
[redacted]		[redacted]					
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
[redacted]	1	[redacted]	1	[redacted]	[redacted]	[redacted]	[redacted]

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

Drawing Number	Sh	Title	Rev	Item No.	Component	Function	Safety Class
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
				[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]			
				[redacted]	[redacted]	[redacted]	[redacted]

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. [redacted]

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. [redacted] 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.

[redacted] [redacted]

	[redacted]				
	[redacted]			[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	-	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	-	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	4.22E-02	[redacted]	[redacted]	[redacted]

[redacted] of spent fuel

[redacted]

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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 [redacted] spent fuel [redacted]. 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 [redacted] is designed to hold up to 36 Yankee Class fuel assemblies. [redacted] 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 contains 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. [redacted]

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.)

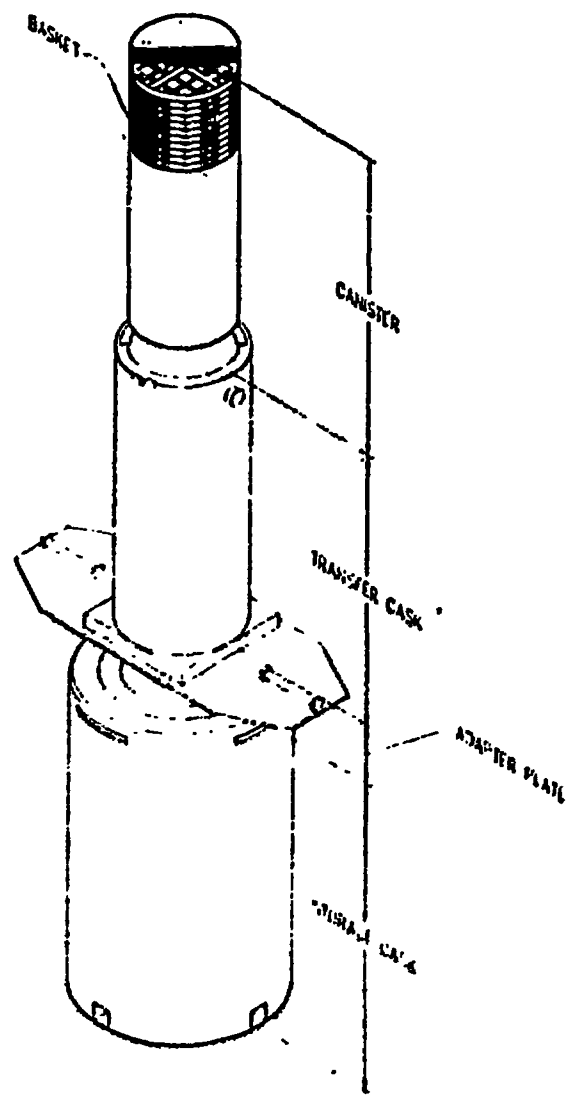
All other NAC-MPC system components shown on the drawings presented in Section 1.5 are either nonstructural or not classified as important to safety. They are appropriately included as loads in the evaluation of the components listed above.

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 [illegible].

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-[illegible]), 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 [illegible]). 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	[redacted]	[redacted]
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)	63,270	61.2
Canister (loaded with fuel with lid)	54,720	65.0
Concrete Cask (empty, with shield plug, and without lid)	148,526	80.5
Concrete Cask and Canister (loaded with fuel with lids)	206,091	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)	143,913	63
Transfer Cask and Canister (loaded with fu[illegible] with lids)	135,473	64.1
Water in Canister ([illegible])	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 [redacted]. 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

	Temperature (°F)							
Property (units)	-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/in³)	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 I, Design Data, Property Code 2110
8 Hanford, Volume I, Design Data, Property Code 1304

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

	Temperature (°F)							
Property (units)	-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	19.1	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/in³)	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 3104

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

	Temperature (°F)							
Property (units)	-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/in³)	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

	Temperature (°F)							
Property (units)	-40	-20	+70	+200	+300	+400	+500	+650
Ultimate Strength[1,2] (ksi)	135.0	135.0	135.0	135.0	195.0	131.4	128.5	125.7[2]
Yield Strength[1] (ksi)	105.0	105.0	105.0	97.1	93.0	89.5	87.0	83.6
Design Stress Intensity[3] (ksi)	45.0	45.0	45.0	45.0	45.0	43.8	42.8	41.9
Modulus of Elasticity[1] (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[6] @ 10 cycles (ksi)	401.8	401.8	396.2	385.4	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	[illegible]	5.90E-6	5.91E-5	5.91E-6	5.93E-6
Poisson's Ratio[8]	0.291							
Density[9] (lbm/in³)	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

	Temperature (°F)							
Property (units)	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-[redacted], 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

	Temperature (°F)			
Property (units)	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-[illegible], Table 1, 2, 3, 4, 5.

Table 3.3-7 Mechanical Properties of A500 Carbon Steel

	Temperature (°F)							
Property (units)	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-[redacted], 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 [redacted] B Low Alloy Steel

	Temperature (°F)							
Property (units)	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/in³)	0.284							

[1] Cases of ASME Boiler and Pressure Vessel Code, Case N-71-[redacted], 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

	Temperature (°F)							
Property (units)	70	100	200	300	400	500	600	[redacted]
Ultimate Strength[1,2] (ksi)	42.0	40.7	38.2	31.5	17.2	6.7	3.4	[redacted]
Yield Strength (ksi)[1,2]	35.0	33.9	32.2	26.9	14.0	5.3	2.5	[redacted]
Design Stress[1](ksi)	10.5	10.5	10.5	8.4	4.4	N/A	N/A	[redacted]
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]	[redacted]
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]	[redacted]
Poisson's Ratio[7]	0.33							
Density[8] (lbm/in³)	0.098							

[1] ASME Code, Section II, Part D, [redacted]

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

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

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

% Value @ Room Temperature

Temp°F	100	200	300	400	500	600	[redacted]
S_u	97	91	75	41	16	8	[redacted]
S_y	97	92	77	40	15	7	[redacted]

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

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

[5] MIL-HDBK-5F, Figure 3.6.2.2.4

[6] MIL-HDBK-5F, 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] (lb/ft³)	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)	5.5×10^{-6}					
Modulus of Elasticity[1] (ksi)	---	3.64×10^{6}	3.38×10^{6}	3.09×10^{6}	3.73×10^{6}	3.43×10^{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)	← [redacted] →			

1 GESC Product Data.

[redacted] [redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	4[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]							

1 [redacted]
2 Tabulated [redacted]
3 [redacted]
4 [redacted]
5 [redacted]
6 ASME Code, [redacted] Table TE-1.
7 [redacted] "Standard Handbook for Mechanical [redacted]

[redacted]

[redacted]	[redacted]					
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]					
[redacted]	[redacted]					

[redacted]

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 [redacted] potential [redacted] 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. [redacted] and spent [redacted] loading procedures [redacted].

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 [redacted] during any phase of normal, off-normal, or accident condition loading, unloading, handling, or storage operations. ~~Therefore, only the potential aluminum reaction with~~ [redacted]

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.

[redacted]

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

3.4.1.2.2 Nonferrous Metals

[redacted] or aluminum flame-sprayed components are [redacted] further concluded that it is possible, at higher [redacted] aluminum/water reaction to produce a hydrogen [redacted] approaches or exceeds the Lower Flammability Limit [redacted] NRC Inspection Reports No. 50-266/96005 and 50-301/96005 [redacted] Beach Nuclear Plant concluded that hydrogen generation by [redacted] to other sources.

Thus, it is reasonable to conclude that small amounts of [redacted] may be produced during NAC-MPC canister loading or [redacted] 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 [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted] or diluting any combustible gas concentrations.

The vacuum pump shall exhaust to a system or area [redacted] If hydrogen [redacted] detected at the port coverplates, [redacted] used to flush combustible gases from the port. Once [redacted] further likelihood of a combustible gas burn because [redacted] combustible gas. Once welding of the shield lid has been [redacted] vacuum dried and back-filled with helium.

No hydrogen is expected to be detected prior to, or during, [redacted] in the shield lid remains open from the time that the loaded [redacted] fuel pool until the time that the vent port coverplate is ready to be [redacted] 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

[redacted]

[redacted]

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 [illegible] Siloxane coating. E-Series epoxy enamel is approved for use in Coating Service [illegible] areas of [illegible] plants, which include the spent fuel pool and reactor containment [illegible]. These are chemically

[redacted]

3.4.1.3 General Effects of Identified Reactions

[redacted] potential chemical, galvanic, or other [redacted] been identified for the NAC-MPC system. [redacted] condition; such as the generation of flammable or explosive quantities of combustible gases [redacted] result [redacted] during any phase of canister operations for normal, off-normal, or accident conditions [redacted]

3.4.1.4 Adequacy of the Canister Operating Procedures

Based on this evaluation, which resulted in [redacted] identified [redacted], 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.

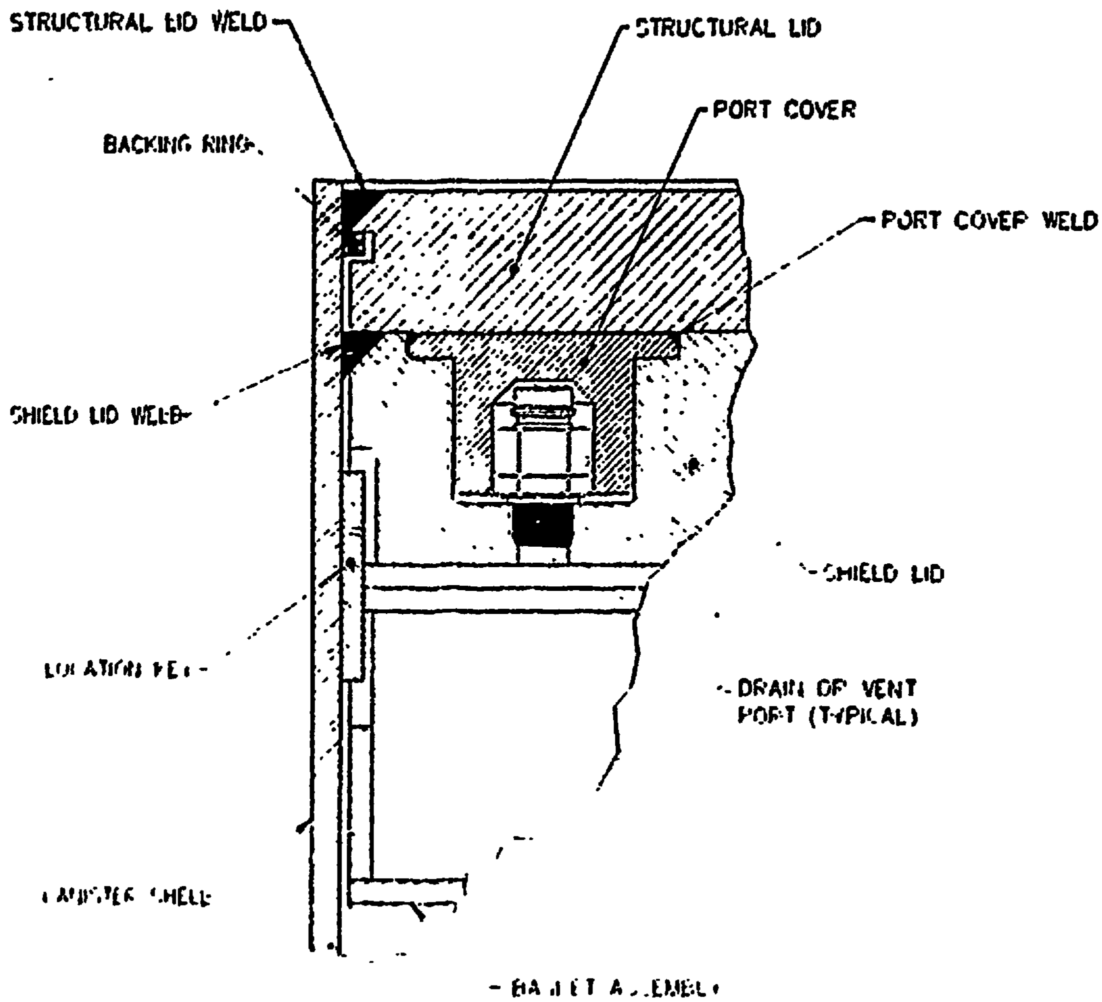
3.4.1.5 Effects of Reaction Products

[redacted] potential chemical, galvanic, or other [redacted] been identified for the NAC-MPC. [redacted]. The overall integrity of the canister and the structural integrity and retrievability of the spent fuel [redacted] not adversely affected for [redacted] 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 [redacted].

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

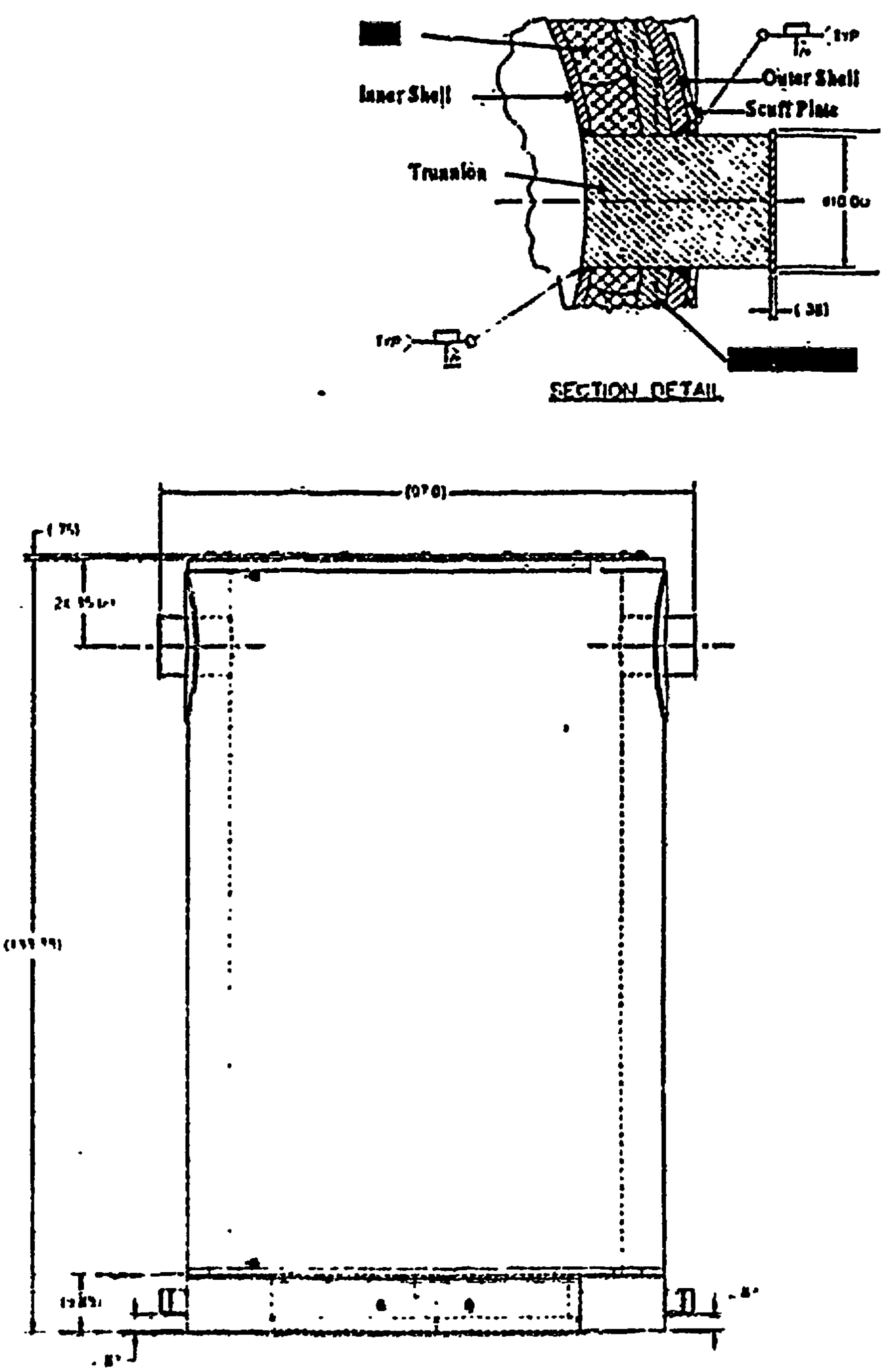
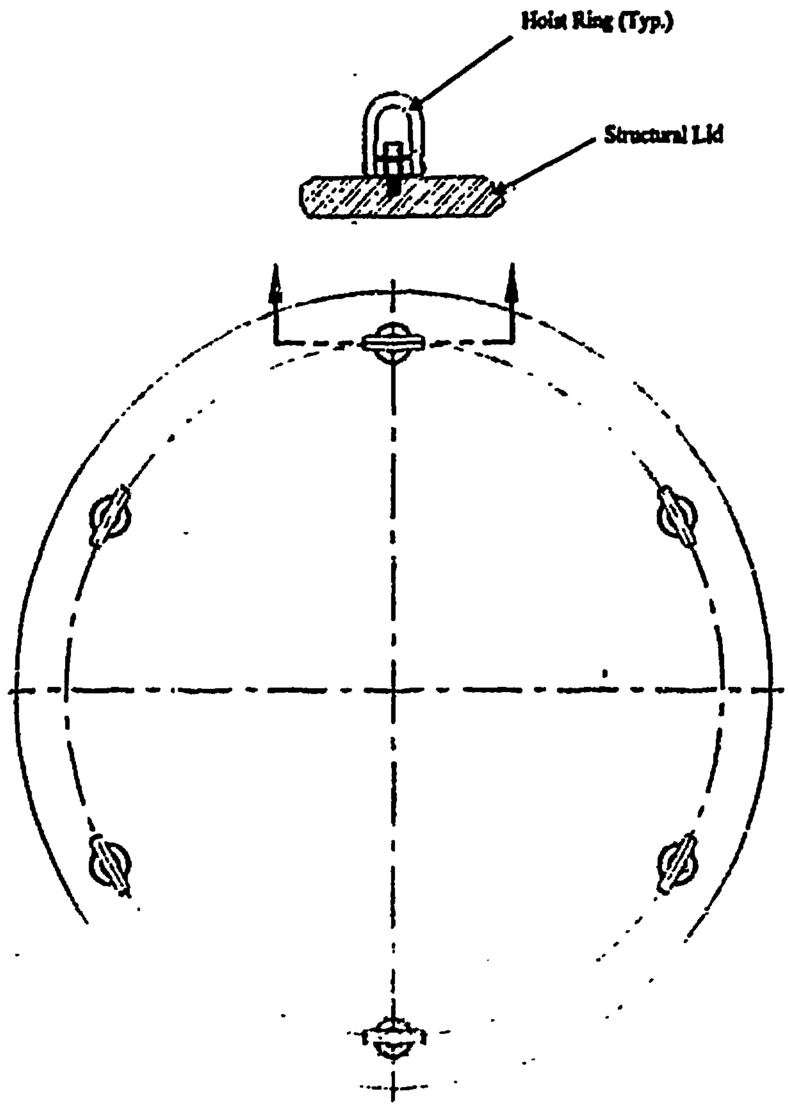


Figure 3.4.3-2 Canister Hoist Ring Design



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 [redacted] one [redacted] 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 x 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:

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

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

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

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

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

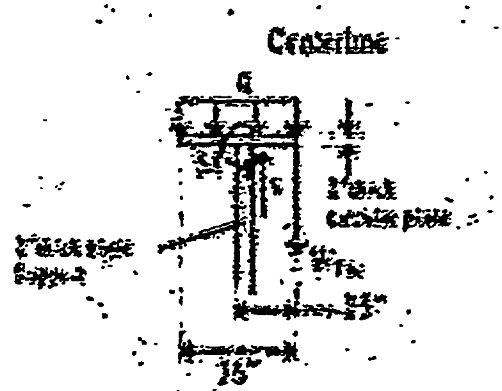
Therefore, the margin of safety is:

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

<u>Pedestal - Horizontal Plate</u>

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:

$$M = (LF)(q)(A)(P_r) = (1.1)(14.0)\left(\frac{\pi(36^2 - 25^2)}{4}\right)(5.5) = 44{,}670 \text{ in-lb.}$$

where,

A = the area of the pedestal from the plate support to the edge of the circular plate.
P_r = 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:

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

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

$$M.S. = \frac{23{,}760}{\text{[redacted]}} - 1 = +\text{[redacted]}$$

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 Fy = 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),

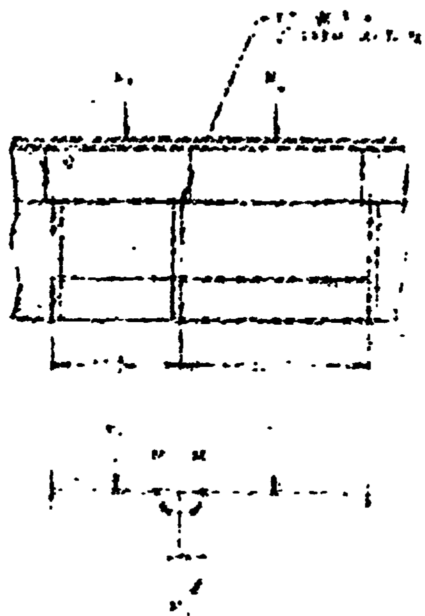
L = 6.0 in., height of pedestal ring (unbraced),

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:

$$P_t = \text{one-fourth of the total load} = (54{,}730 + 2{,}313)/4 = 14{,}261 \text{ lbs.} \approx 14{,}300 \text{ lbs.}$$



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

$$P_1 = 14{,}300 \times (15.35/37.7) = 5{,}822 \text{ lbs.}$$

$$P_2 = 14{,}300 \times (22.35/37.7) = 8{,}478 \text{ lbs.}$$

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.

$$M_1 = \frac{1.1(P_1L_1)}{8} = \frac{(1.1)(5{,}822)(15.35)}{8} = 12{,}288 \text{ in-lb}$$

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

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

The maximum bending stress f_b is computed as

$$f_b = \frac{6M_3}{bt^2} = \frac{(6)(13,756)}{(2)(1.5)^2} = 18,355 \text{ psi}$$

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

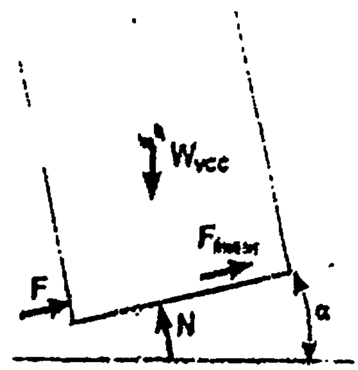
The storage cask is supported by air pads in each of 4 quadrants during transport. The layout of the air pads (four 48 in. × 48 in. square pads) must clear the air inlet locations by approximately 3 inches to allow for hydraulic jack access.

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 %):

$$F = \frac{54{,}730 \times 1.1}{3} = 20{,}068 \text{ lb}$$

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 [illegible]. 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 (in²)
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 (in²)
A_n	= shear area of lid threads (in²)
$D_s min$	= 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:

$A_t =$ [redacted]

$A_t =$ [redacted]

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

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

$L_e =$ [redacted] 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:

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

$A_s =$ [redacted] in^2

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

$A_n =$ [redacted] in^2

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

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

$= 2.048$

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.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted] [redacted]

[redacted] [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

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. [redacted]

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:

$$FS = \frac{S_y}{S_{INT}} = \frac{20{,}300\,psi}{3{,}753\,psi} = 540 > 3$$

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

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

Figure 3.4.3.2-1 Canister Lift Finite Element Model

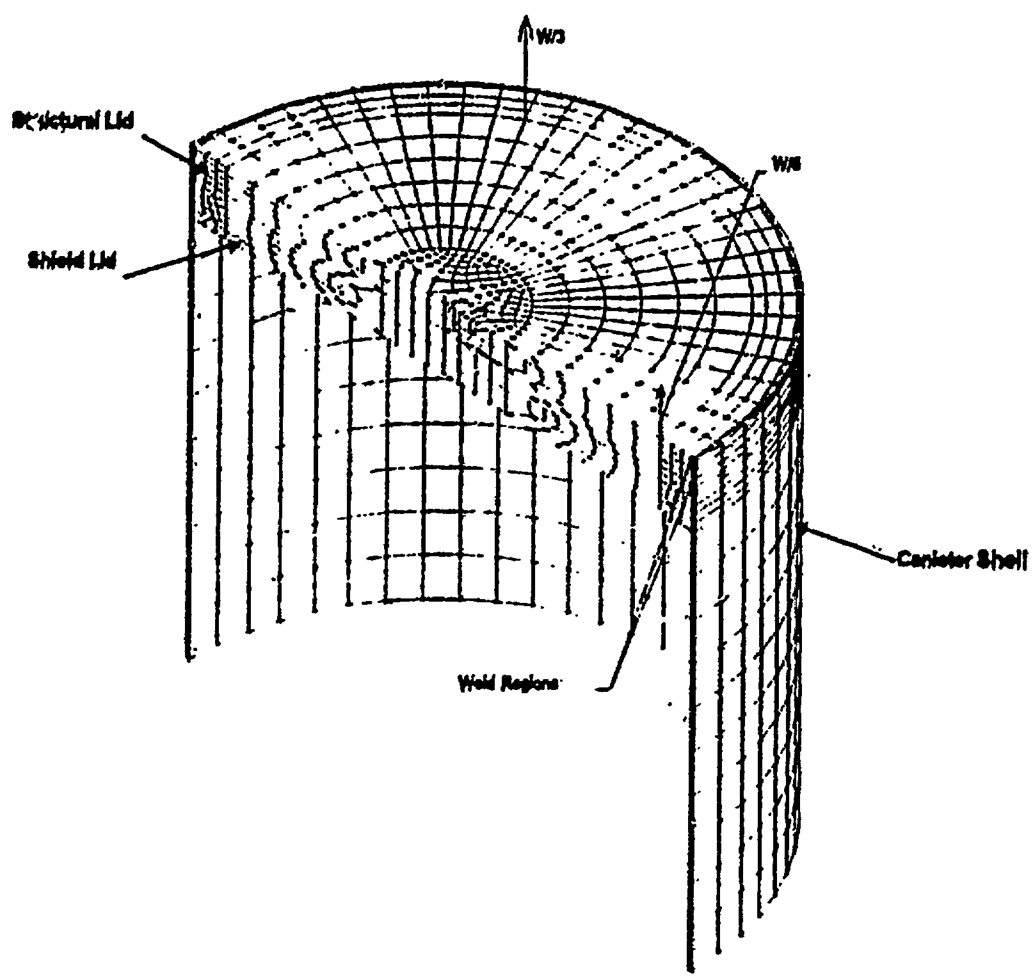
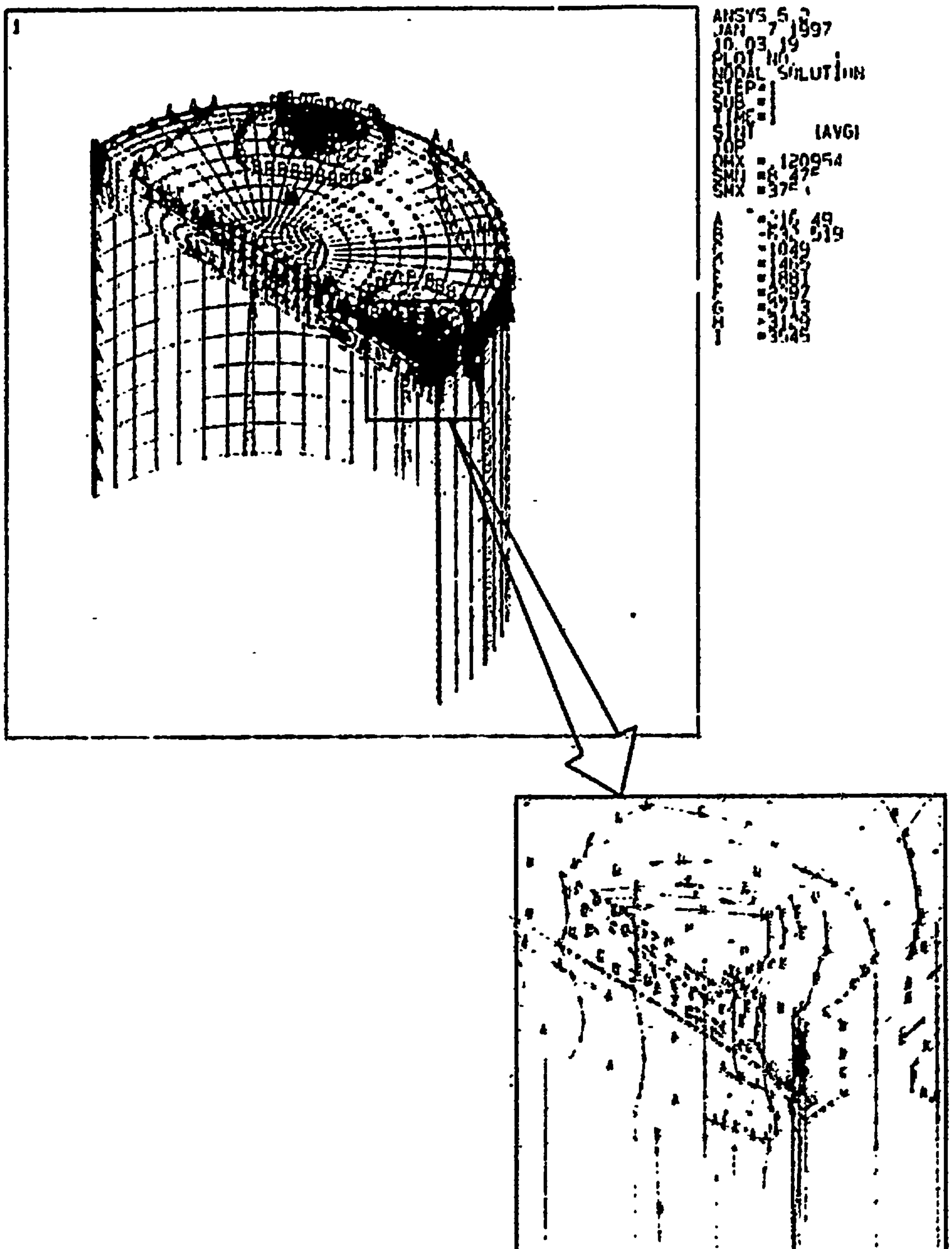


Figure 3.4.3.2-2 Canister Lift Model Stresses Intensity Contours (psi)



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, [redacted], are relieved by slight local material yielding and the stress design factors are not applicable.

The localized stresses occur at the interface of the [redacted] meridional direction lengths of the high stress areas [redacted] shell and outer shell, respectively. In accordance with [redacted] 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.

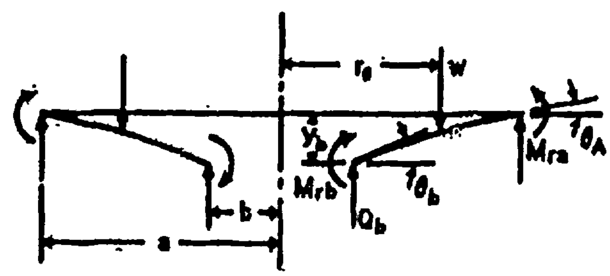
3.4.3.3.2 Retaining Ring and Bolts

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. [redacted]

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):

$w_t = 81{,}000$ 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 = [redacted]

Radial length of applied load:

$L_r = 2\pi r_o$

$L_r = 221.8$ in

Applied unit load:

$$w = \frac{w_t}{L_r}$$

$w = 401.7$ psi

The shear modulus is:

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

$= 1.08 \times 10^7$ psi

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

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

$D = 1.101 \times 10^6$ 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:

$\tau_b = 0$ psi

$\tau_a = 482.3$ psi

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

$\sigma_r\,(b) = 0$ psi

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

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

$\sigma_t\,(b) = 828.0$ psi

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

$$\sigma_{3b} = \left(\frac{\sigma_r(b) + \sigma_t(b)}{2}\right) - \sqrt{\tau_b^2 + \left(\frac{\sigma_r(b) - \sigma_t(b)}{2}\right)^2}$$

$\sigma_{3b} = -828.0$ psi

$\sigma_{3b} = 0$ psi

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

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

$SI_b = 828.0$ psi

The maximum stress intensity occurs at the outer radius of the retaining ring. For [redacted] conditions, the allowable stress [redacted] is equal to the lesser of [redacted] S_m and 1.[redacted] S_u [redacted]. For ASTM A588, the allowable stress [redacted] at 300°F is [redacted] ksi. The calculated stress [redacted] of 15.24 ksi is less than the allowable stress [redacted] and the margin of safety is +[redacted].

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Retaining Ring Bolts

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

$$F_F = \frac{wt}{Nb} \quad = \text{[redacted]} \text{ lbs.}$$

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

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

$$F_M = \text{[redacted]} \text{ lbs.}$$

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:

[redacted]

[redacted]. The bolt tensile stress is:

[redacted]

For [redacted] conditions, the allowable stress for primary membrane stress in a bolt is $2S_m$. [redacted] bolts is [redacted] ksi at 200°F. The margin of safety for the bolts is +[redacted].

[redacted]

[redacted]

For the top plate (ASTM 516) at a temperature of 200°F, [redacted] 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:

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

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

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

where

$F_T = F = F_F + F_M$

Conservatively, the shear allowable for normal conditions is used.

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

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).

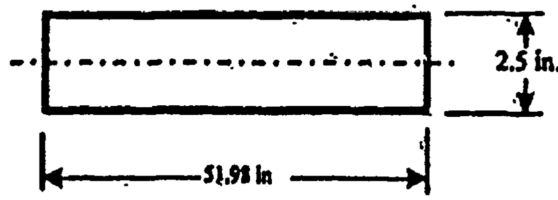
Yield Strength (S_y) = 31.9 ksi at 300° F (ASTM A-350)
Ultimate Strength (S_u) = 70.0 ksi at 300°F (ASTM A-350)

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

= $L \times t = 129.95\ in^2$,

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:

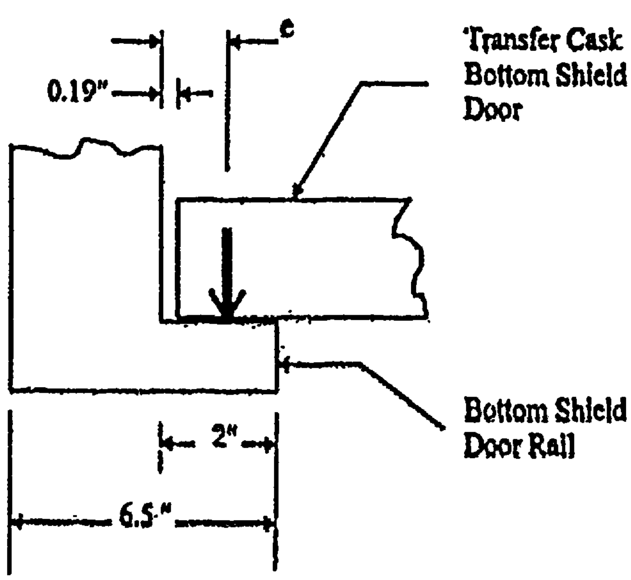
$$\sigma_b = \frac{M}{S} = \frac{90.8}{54.15} = 1.68\ ksi,$$

where,

M = moment at the bottom section,

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

= 90.8 in-kips,



and,

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

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

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

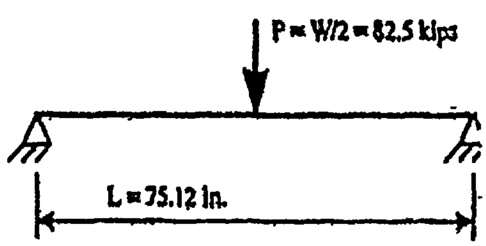
Per ANSI N14.6, Section 4.2.1, shear stress or maximum tensile stress are to be compared with material yield and ultimate strength. The resulting factors of safety are:

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

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

Stress Evaluation for the Shield Doors

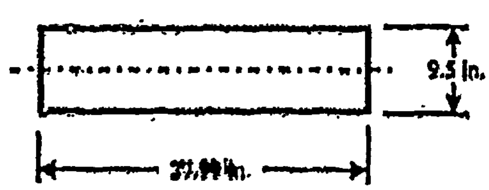
The shield doors consist of two 9.5-inch thick plates that rest on top of the rails. Stresses can be approximated by modeling the shield door as a simply supported beam with a concentrated load at the center.



The shear stress in each shield door is:

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

$$= 0.31 \text{ ksi}$$



(Shield Door Cross Section)

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

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

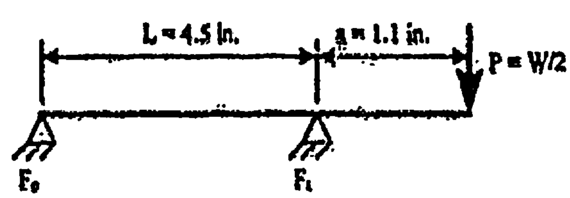
$$\varphi_u = \frac{70}{3.68} = 19.0 > 10 \qquad \text{(For ultimate strength criteria)}$$

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.

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

$$= 20.17 \text{ kips}$$



$$F_1 = \frac{165}{2} + 20.17 = 102.67 \text{ kips}$$

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

$$= \frac{102.67}{56.25 \times 0.75} = 2.4 \ldots$$

The factors of safety are

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

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

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

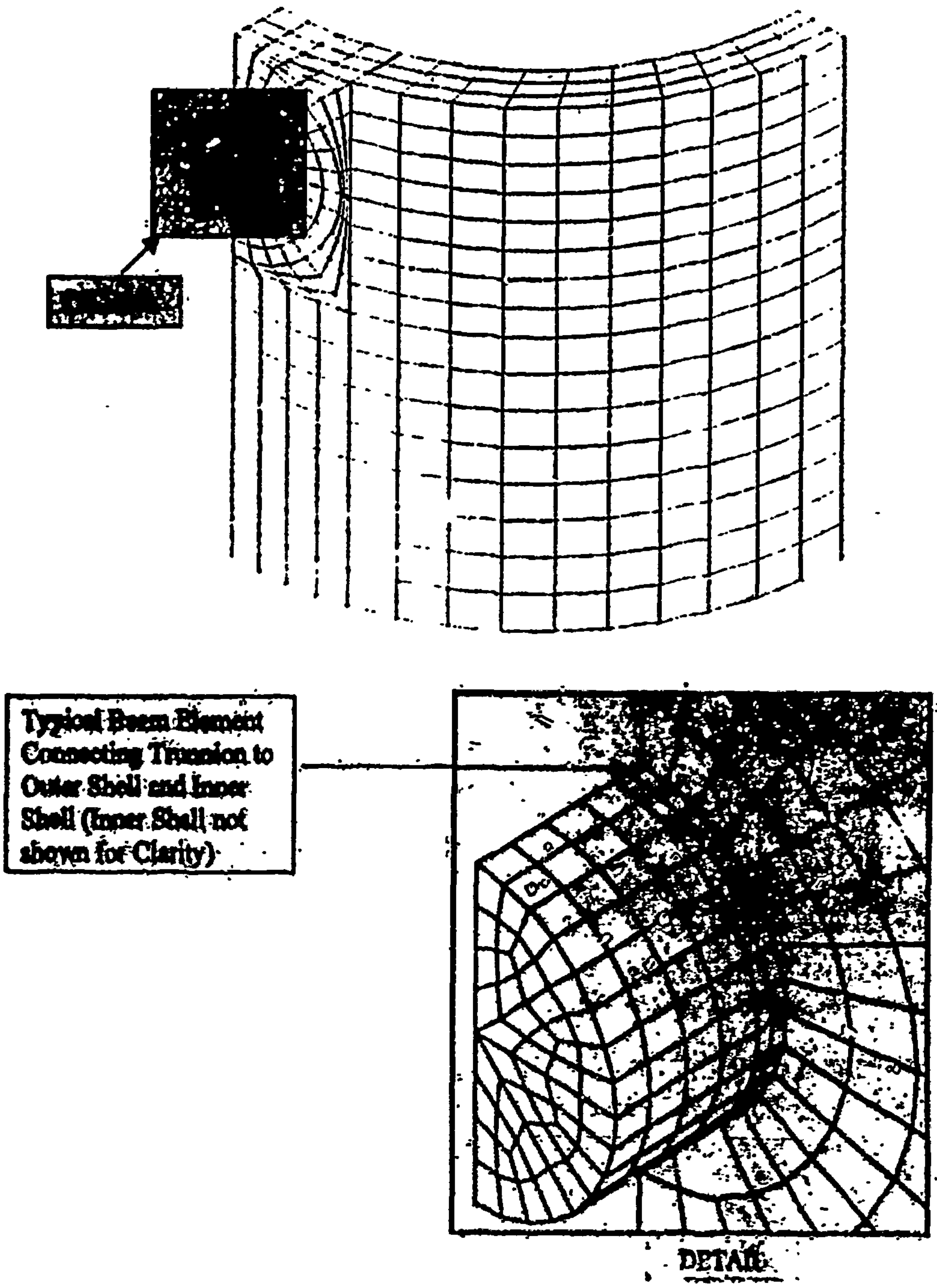


Figure 3.4.3.3-2 Node Locations for Transfer Cask Outer Shell Adjacent To Trunnion

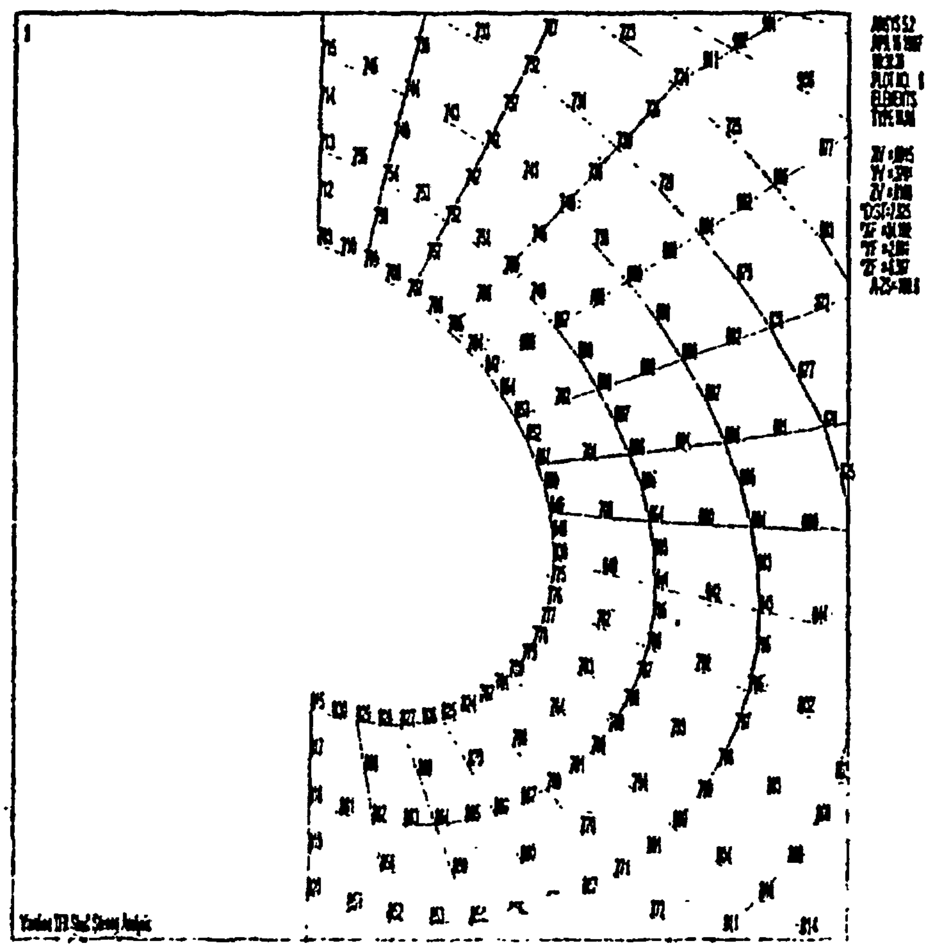


Figure 3.4.3.3-3 Node Locations for Transfer Cask Inner Shell Adjacent To Trunnion

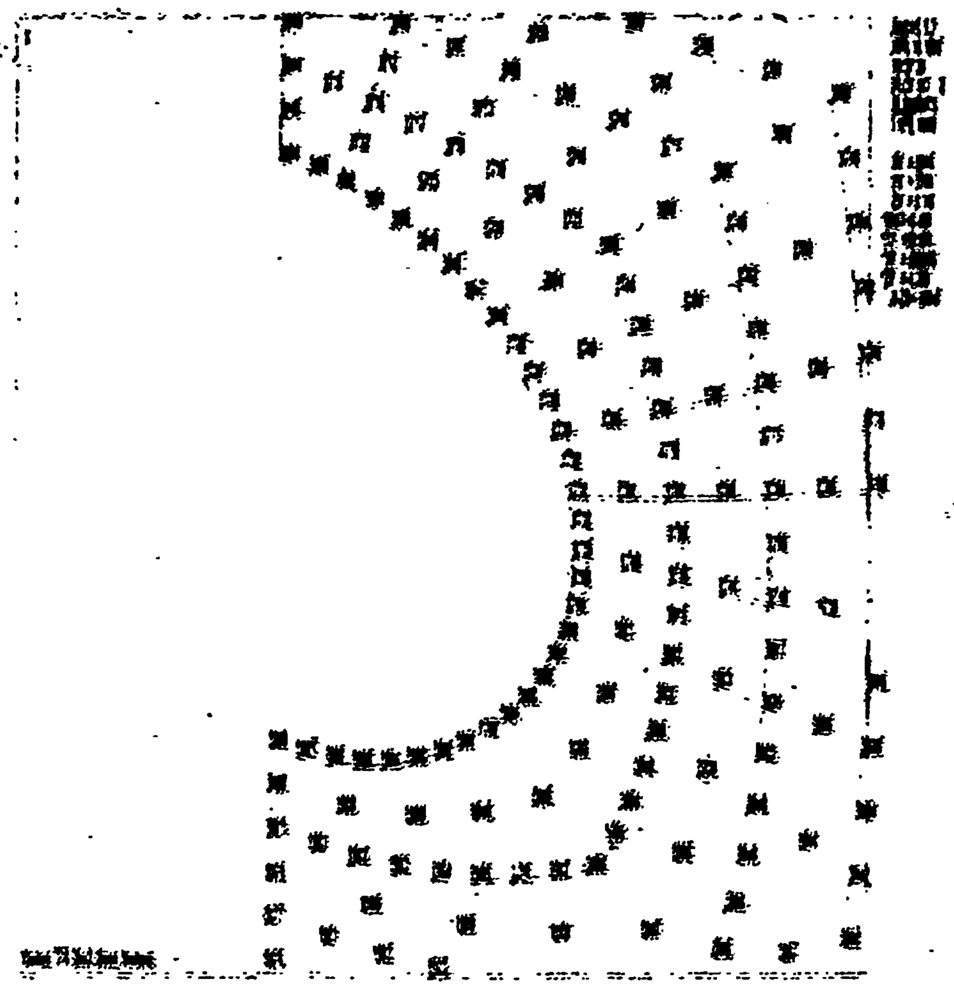


Figure 3.4.3.3-4 Stress Contours for Transfer Cask Outer Shell

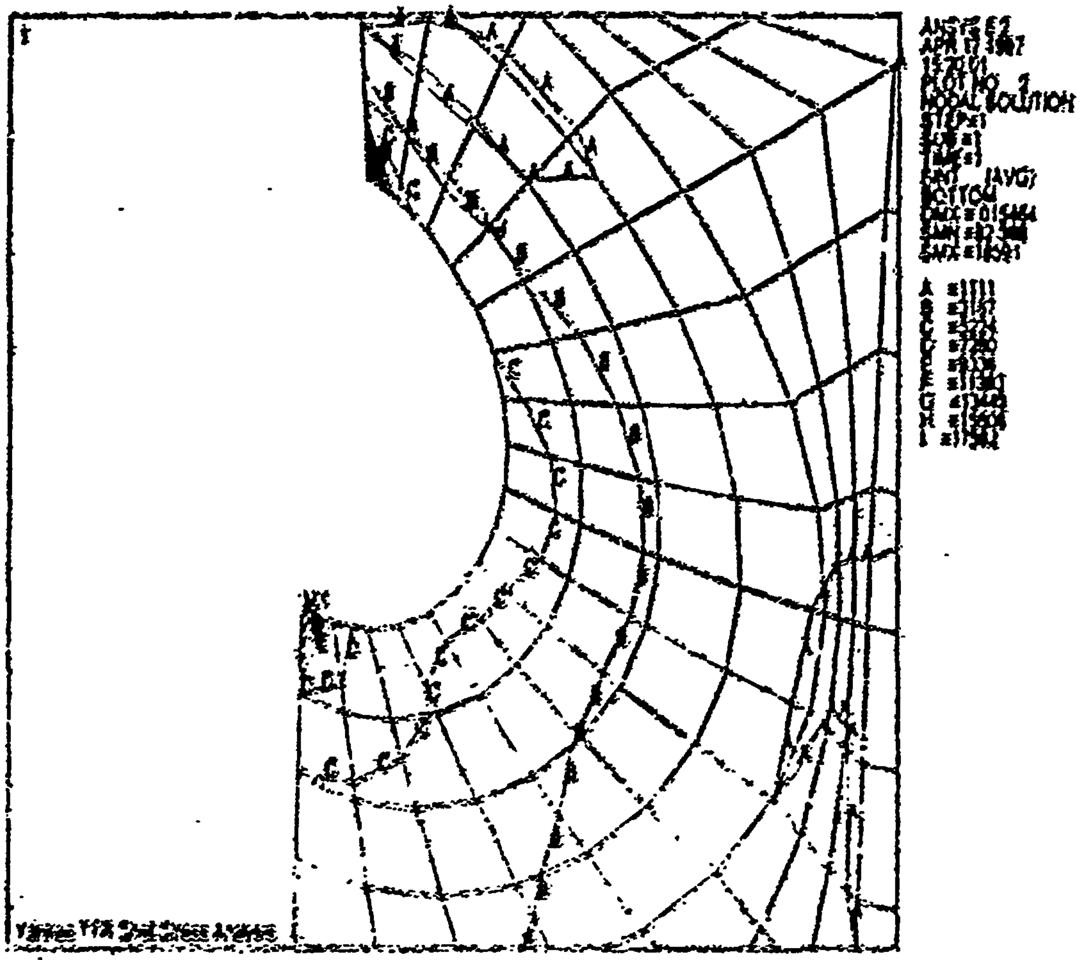


Figure 3.4.3.3-5 Stress Contours for Transfer Cask Inner Shell

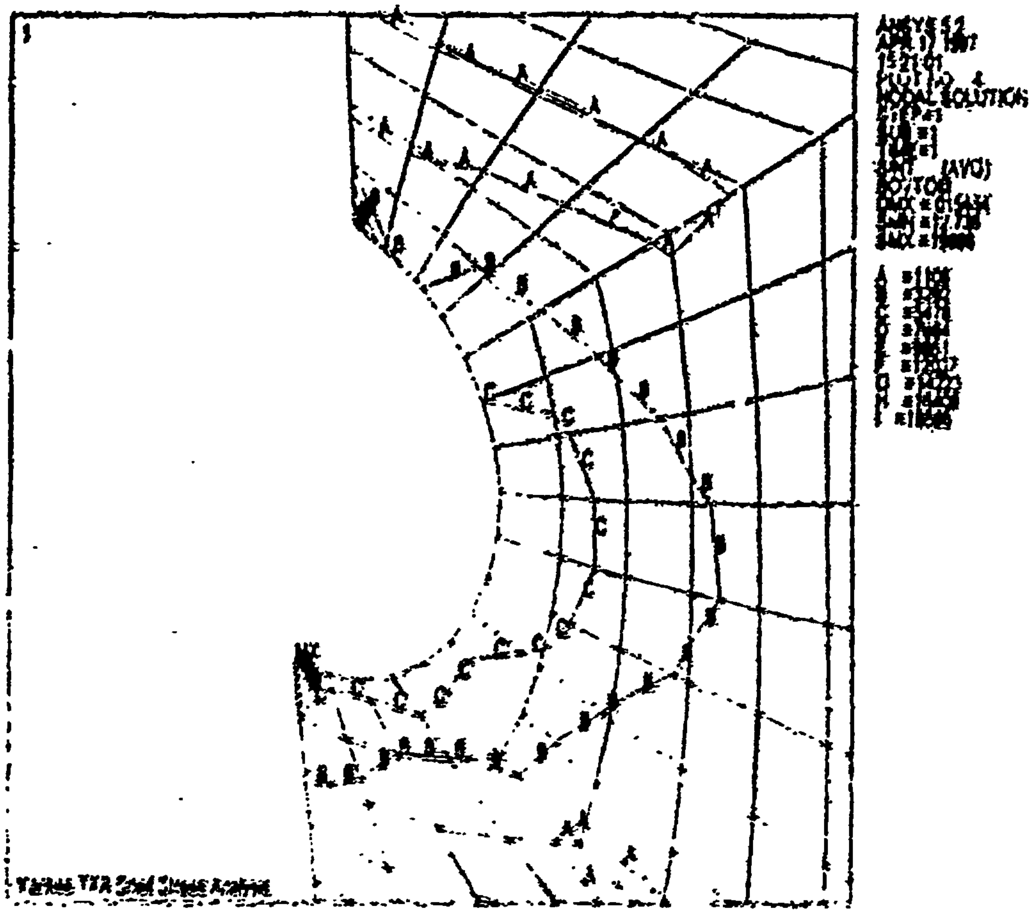


Table 3.4.3.3-1 Top 30 Stresses for Transfer Cask Outer Shell

	Principal Stresses (psi)			Nodal S.I.	F.S. on Yield[2]	F.S. on Ultimate[2]
Node[1]	S1	S2	S3	(psi)	(S_y/S.I.)	(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
329	8595.7	4797.9	-20.588	8516.2[3]	N/A	N/A
818	7296.6	1839.7	-7.8552	7304.4[3]	N/A	N/A
862	6536.6	2753.7	-9.4784	6546	7.0	10.7
638	5595.9	-19.633	-848.4	6444.3	7.1	10.9
776	4756.4	314.83	-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	1052.5	-1275.6	-4516.8	5569.1	8.2	12.6
767	5246.3	3215.3	-188.53	5434.8	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	0.68016	5135	8.9	13.6
820	5118.7	2375.9	-1.6525	5120.3	8.9	13.7
893	30.529	-567.88	-4905	4935.5	9.2	14.2
852	4928.5	2243.8	-2.5875	4931.1	9.3	14.3
769	4816.4	1032.3	-1.2375	4817.7	9.5	14.5
647	505.55	-3770.9	-4172.3	4677.9	9.7	15.0
641	2895.5	1.2087	-1776.8	4672.3	9.8	15.0
694	2169.8	4.4159	-2469.8	4639.6	9.8	15.1
901	32.924	-363.76	-4599.6	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	2431.9	-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	-1.7619	-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	-302.37	5317.6	8.6	13.2
1838	1675.8	22.011	-3579.3	5255.1	8.7	13.3
1645	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	-3981.1	4198.1	10.9	16.7
1854	1463.1	30.789	-2681.4	4144.5	11.0	16.9
1806	3183.2	68.412	-943.41	4126.6	11.1	17.0
1768	1787.2	-0.4604	-2229.5	4016.7	11.4	17.4
1648	3076.7	1254.1	-849.56	3926.2	11.6	17.8
1832	3613.4	1174.9	-9.7208	3643.1	12.5	19.2
1738	2162.3	1.6241	-1473.5	3635.8	12.5	19.3

Notes

1 See Figure 3.4.3.3-1 for node locations.
2 S_y = 45,650 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. [redacted]

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:

	Maximum ΔT (°F)		
Condition	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 follow:

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

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

$$= 80 \text{ psi}$$

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:

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

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

$$= 106 \text{ psi}$$

where,

area = $\pi \times D \times t_w$ = 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 [redacted] internal pressure load, [redacted] using the ANSYS model and temperature distribution and restraints described in Section 3.4.4.1.1. [redacted] 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:

$$\Delta P = 17.93 - 11.32 = 6.61 \text{ psi} < 11.3 \text{ psi}.$$

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:

$$\Delta T = \frac{S_a}{2E\alpha} = 58°F,$$

where,

S_a = 28,200 psi, the value obtained from the fatigue curve for service cycles < 10^6,

E = 27×10^6 psi, modulus of elasticity at 300 °F,

α = 9×10^{-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

[redacted]

NB-3226 requires that P_m not exceed $0.9S_y$ at the test temperature. [redacted] over water pressure test meets this condition, the stress [redacted] due to the normal conditions design pressure of 11.5 psig [redacted] for the 15 psig test pressure. The canister material is [redacted] 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:

$$(P_m)_{test} = (15/11.5) \times (4.60 \text{ ksi}) = 6.0 \text{ ksi, which is} < .9S_y = 18.3 \text{ ksi } (.9 \times 20.3 \text{ ksi})$$

Thus, the criteria is met.

NB-3226 requires that for $P_m \leq 0.67S$, 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 also evaluated in addition to dead load and handling load. The results of the weldment structural 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:

$$\text{Area} = (7.8 + 2 \times 0.048)^2 - 7.8^2$$
$$= 1.507 \text{ in}^2$$

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.

$$\text{Margin of Safety} = 17{,}300/57 - 1$$
$$= +\text{Large}$$

Figure 3.4.4.1-1 Canister ANSYS Finite Element Model

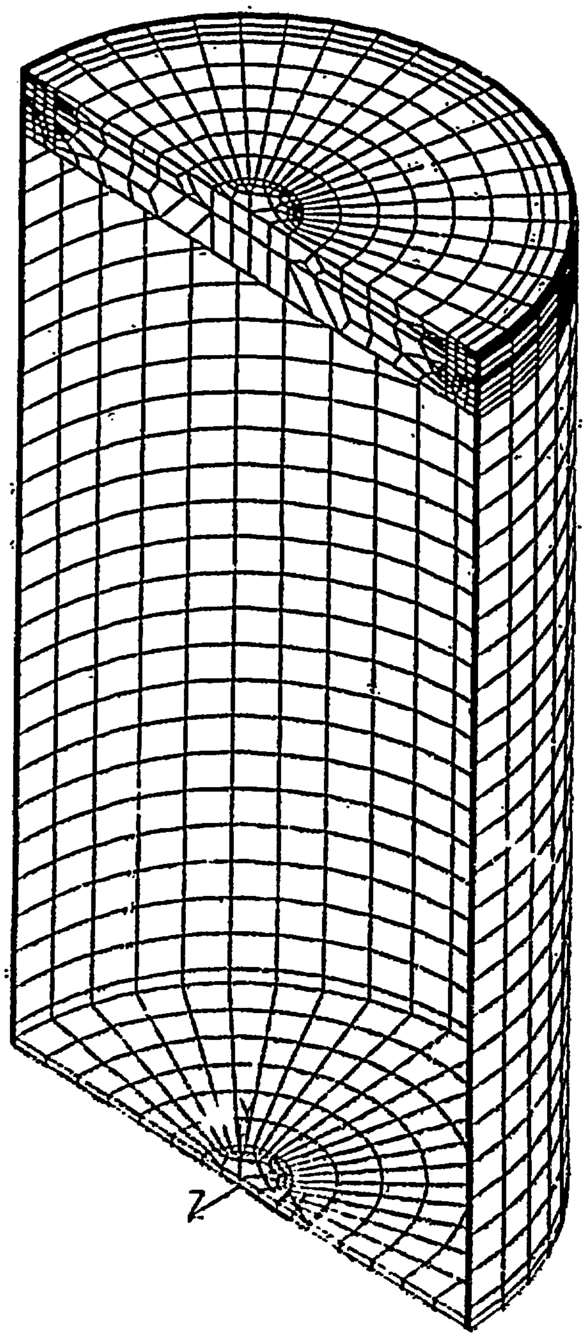


Figure 3.4.4.1-2 Weld Regions of Canister ANSYS Finite Element Model at Structural and Shield Lids

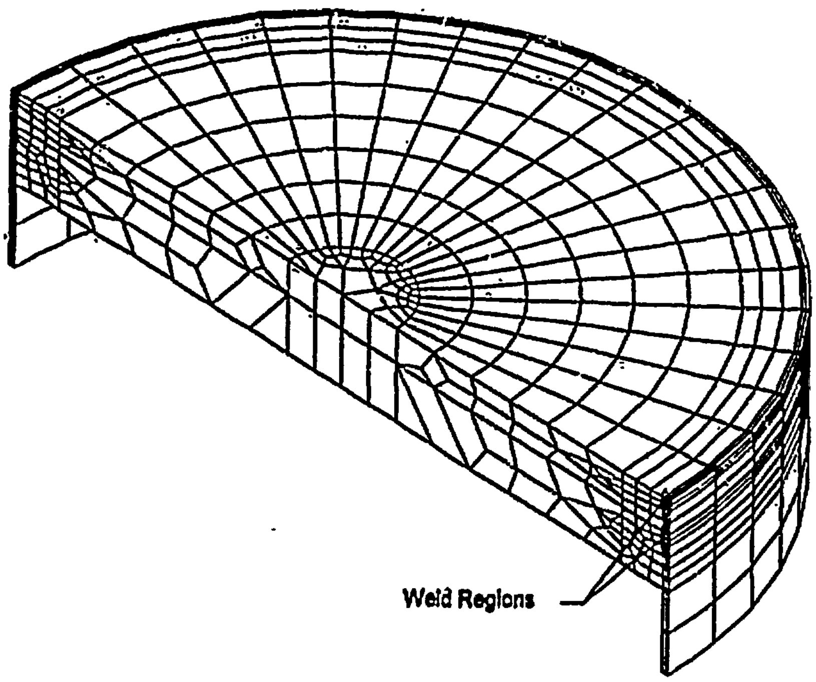


Figure 3.4.4.1-3 Bottom Plate of the Canister ANSYS Finite Element Model

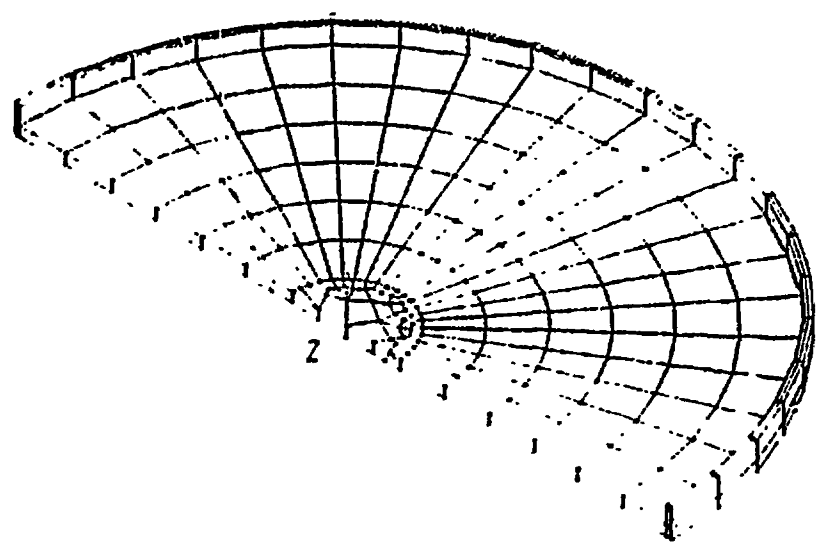
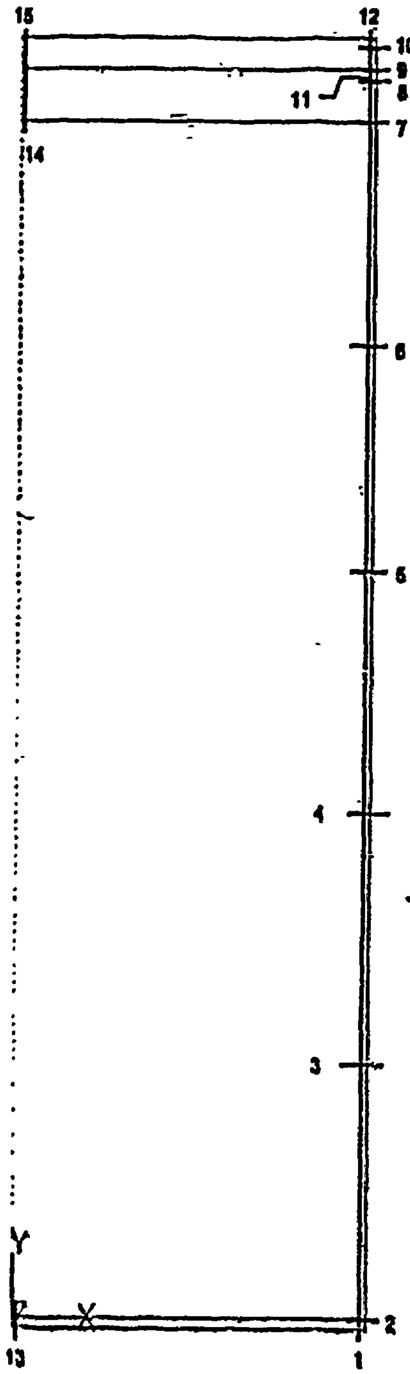
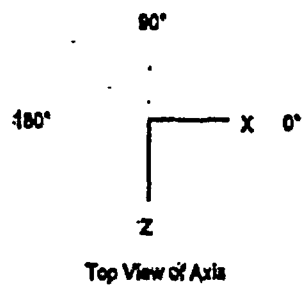


Figure 3.4.4.1-4 Locations for Section Stresses in the Canister ANSYS Finite Element Model





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

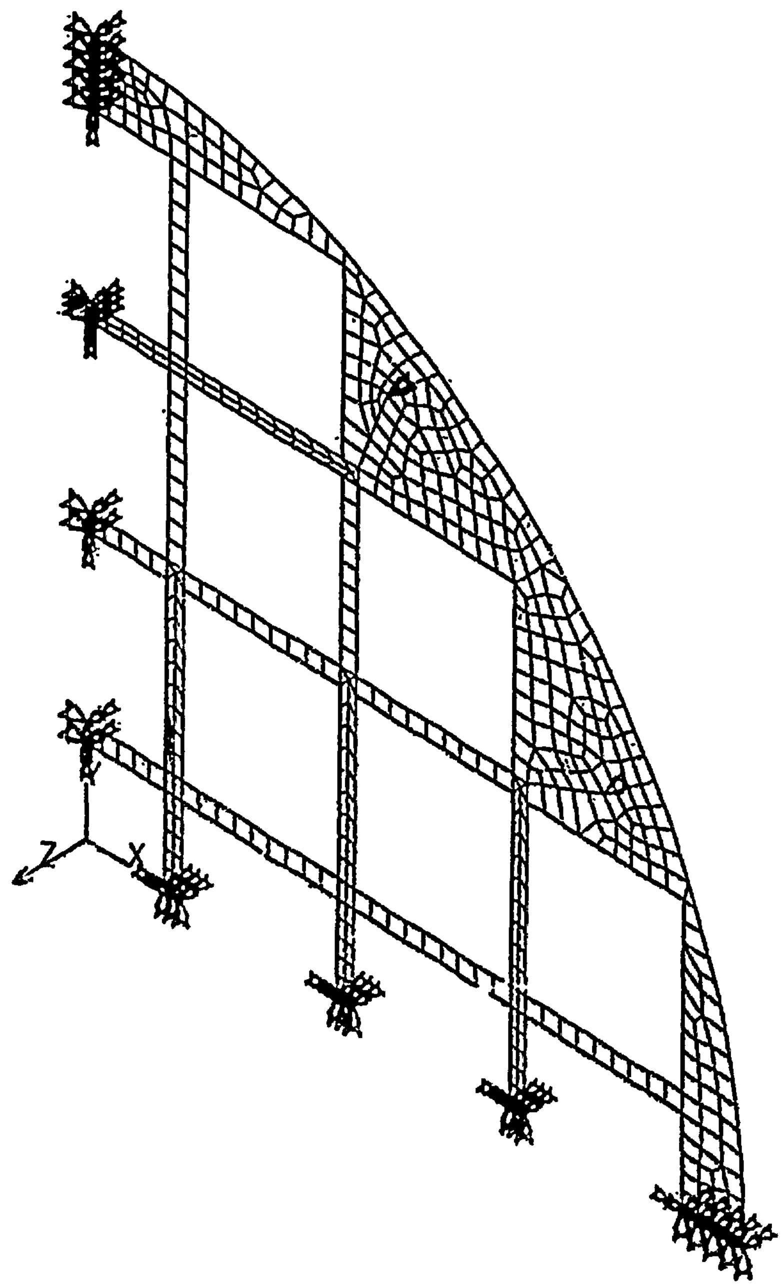


Figure 3.4.4.1-6 Fuel Basket Top Weldment ANSYS Finite Element Model

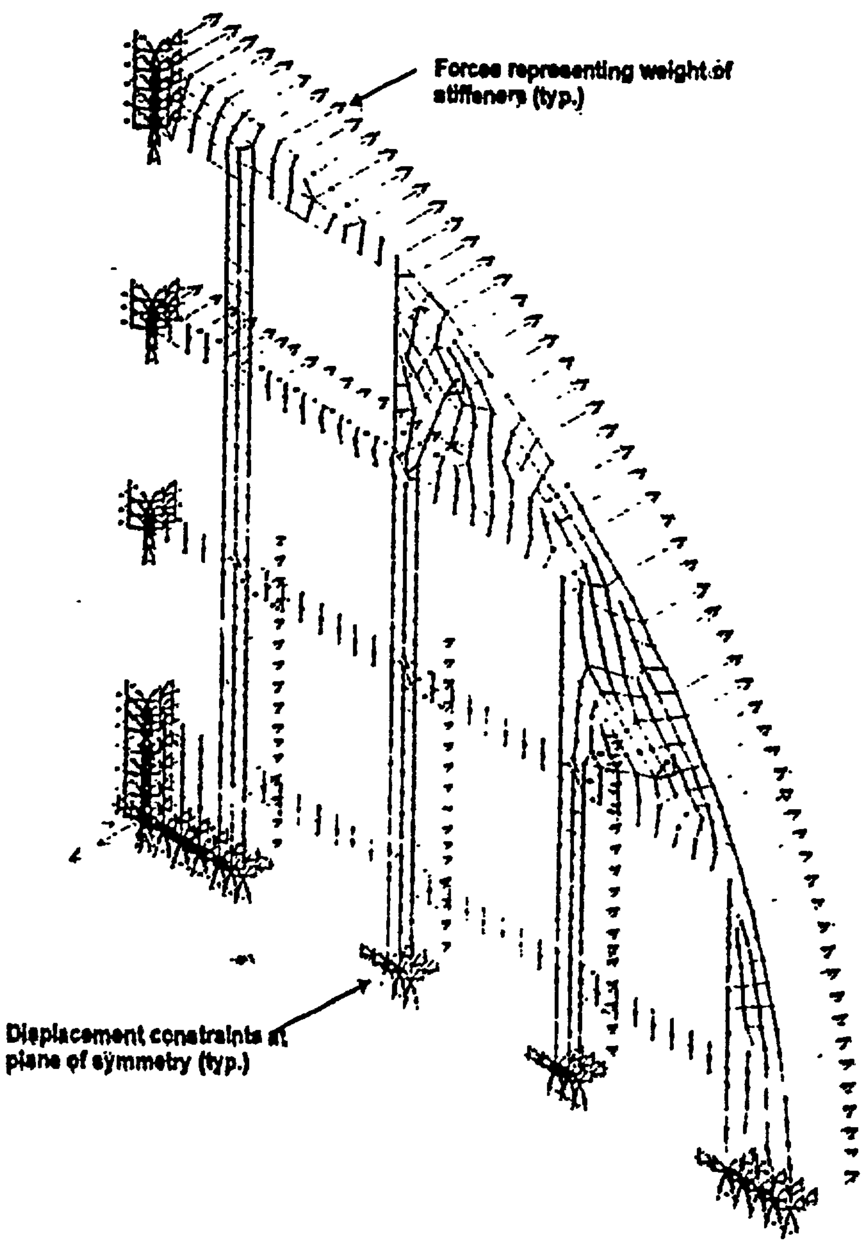


Figure 3.4.4.1-7 Fuel Basket Bottom Weldment ANSYS Finite Element Model

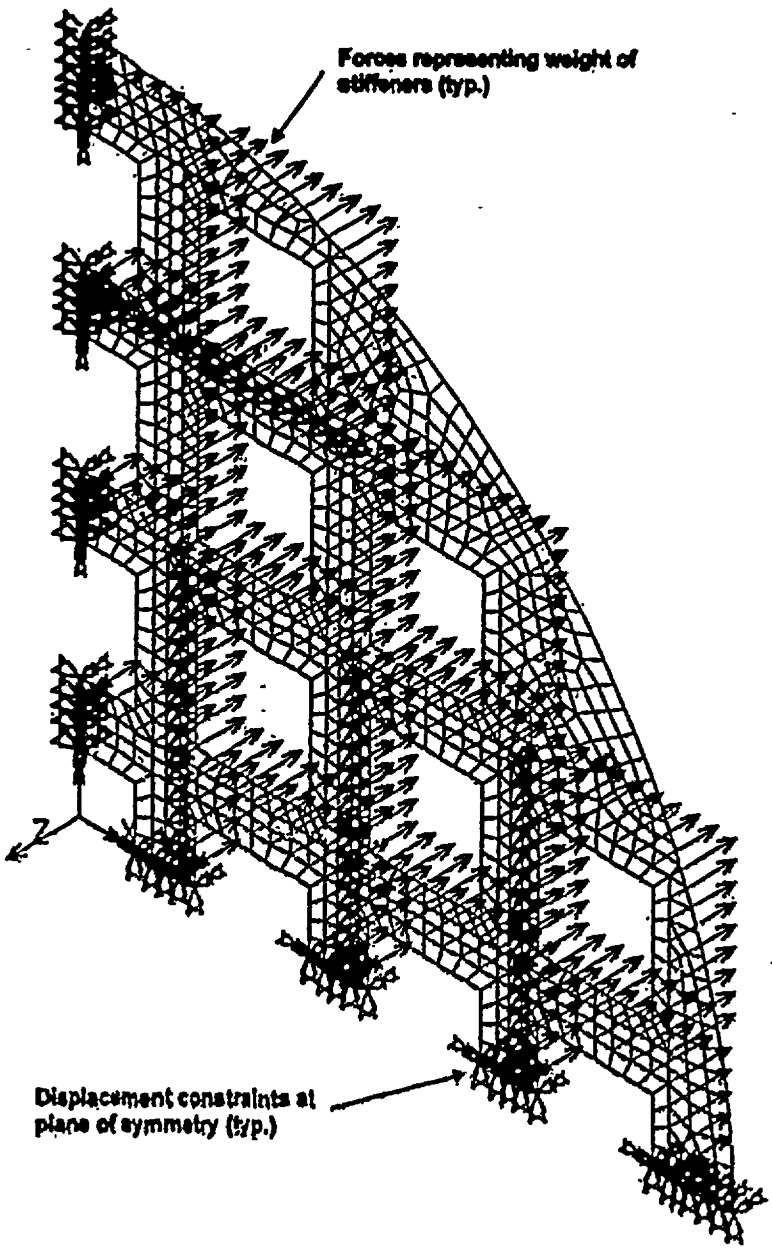


Figure 3.4.4.1-8 Fuel Tube Configuration

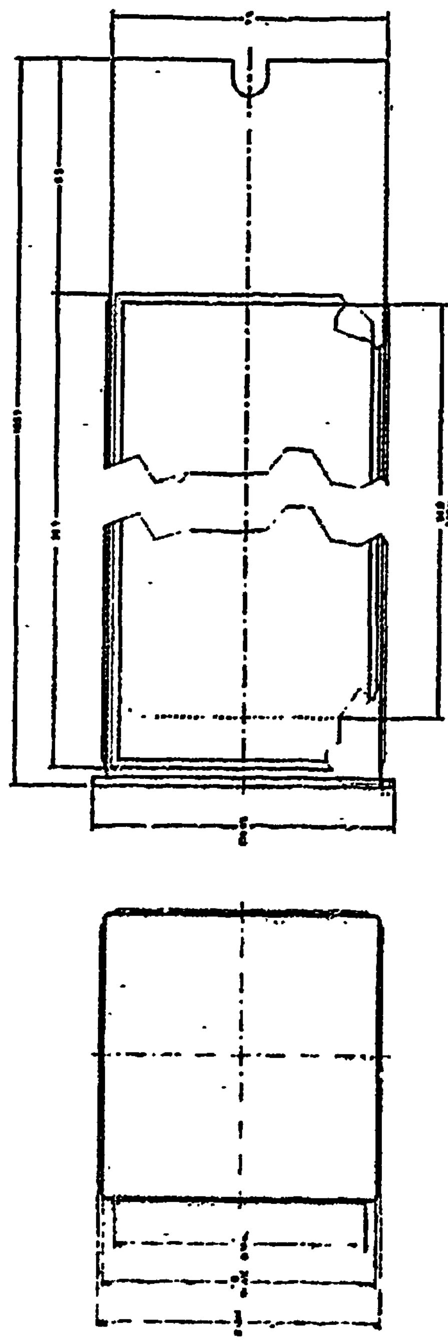


Table 3.4.4.1-1 Summary of Maximum Canister Thermal Stresses (ksi)

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	-0.2	13.29
11	-4.7	-5.1	-2.7	-0.7	<0.1	0.1	2.90
12	-4.1	1.6	<0.1	-2.8	-0.1	-0.3	6.12
13	-19.4	-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 extraction.

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 2.4.4.1-4 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.[†]	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	0.5	<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	0.15
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

† See Figure 3.4.4.1-1 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	1.1	10.6	-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	4.36
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[3]	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	0.7	14.50	19.42
4	< 0.1	0.8	0.6	< 0.1	< 0.1	< 0.1	0.	14.50	18.03
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	12.53
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	16.70	13.09
13	1.4	0.1	1.3	-1.5	-4.5	< 0.1	9.47	16.70	0.76
14[2]	-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.

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	25.05	14.10
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.

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	50.10	6.17
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.

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	---
		$P_m + P_b$	953	$1.5S_m$	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	$= \pi (D^2 - ID^2)/4 = 7{,}059.2 \text{ in.}^2$

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:

σ_y = W/A = -19.2 psi (compressive)

3.4.4.2.3 Thermal Load

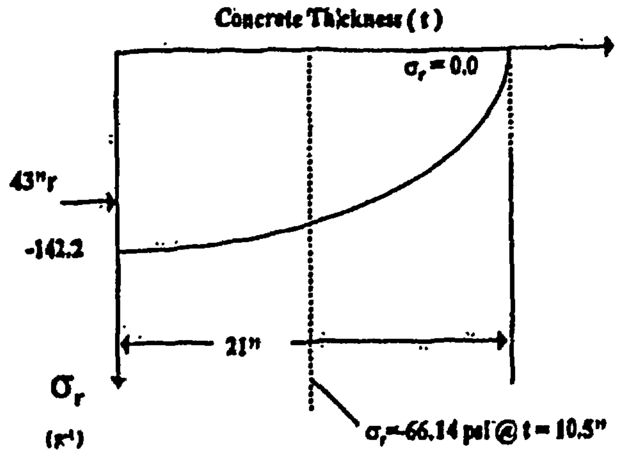
An axisymmetric finite element model consisting of [redacted] elements for the steel liner and concrete shell was developed to calculate the thermal stresses in the concrete (see Fig. 3.4.4.2-1). The nodes at the steel liner/concrete interface are coincident and are connected analytically by coupling 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,

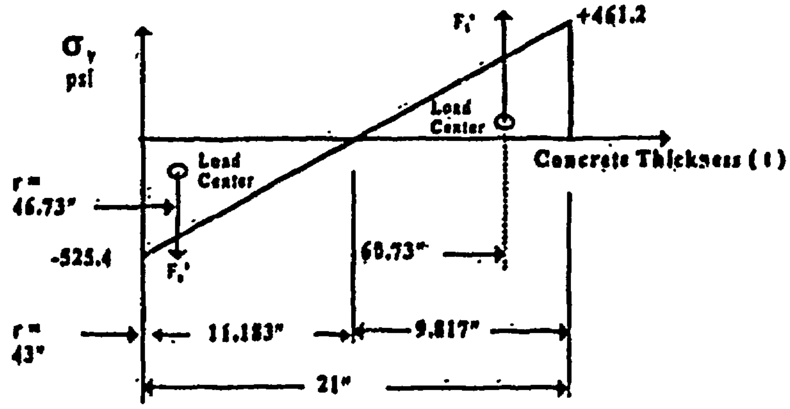
$$\phi = 0.70$$

$$f_c' = 4{,}000 \text{ psi}$$

$$\sigma_{bearing} = \phi \cdot f_c' = (0.70)(4{,}000) = 2{,}800 \text{ psi}$$

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 below.



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:

$$F_{v\,applied} = F_v' / 56 = 15{,}423 \text{ lbf per vertical reinforcing bar.}$$

Using a 1.275 load factor for normal operating loads:

$$1.275\, F_{v\,applied} = 1.275(15{,}423) = 19{,}664 \text{ lbf}$$

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 [redacted] Section 9.2)

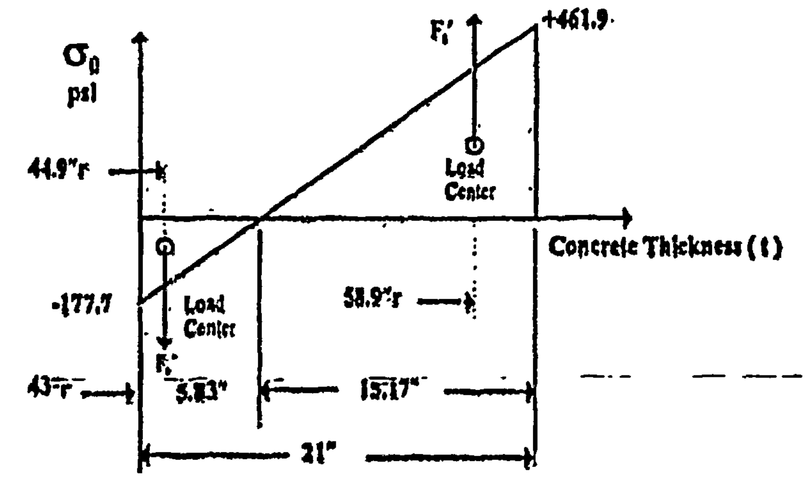
The reinforcing bar tensile load capacity is:

$F_\theta = (\sigma_{\theta\,allowable})(A_{\#6}) = (54{,}000)\,(.44) = 23{,}760$ 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:

$$\text{M.S.} = \frac{23{,}760}{19{,}664} - 1 = +\,0.21$$

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:

$$F_{\theta\,\text{applied}} = F_t' \times 4 = 14{,}014 \text{ lbs.}$$

Using a 1.275 load factor for normal operating loads

$$1.275\, F_{\theta\,\text{applied}} = 1.275(14{,}014) = 17{,}868 \text{ lbf}$$

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:

$$\text{M.S.} = \frac{23{,}760}{17{,}868} - 1 = +0.33$$

Figure 3.4.4.2-1 Concrete Cask Axisymmetric Thermal Stress Model

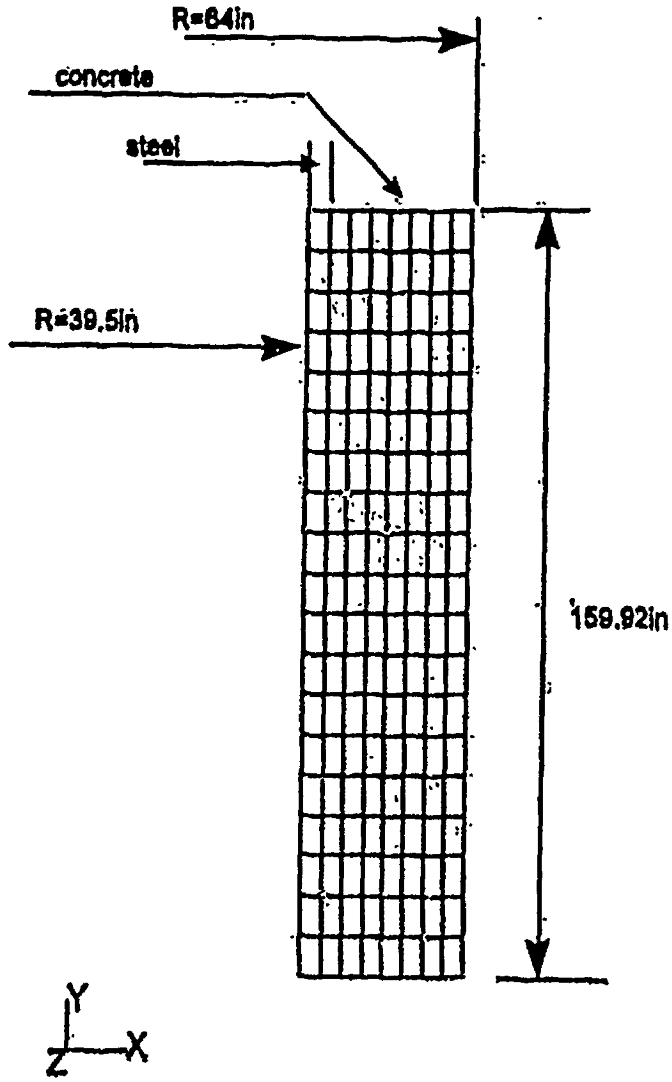


Table 3.4.4.2-1 Stress Summary for Concrete Cask Load Combinations

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_4 (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	[redacted] psi	+[redacted]
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) = [redacted] 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 [redacted]

[redacted]

[redacted]

[redacted] Margin of Safeties are:

Stress Type	Analysis Stress (ksi)	0.8 x Allowable Stress (ksi)	[redacted]
P_m	1.19	13.36	[redacted]
$P_m + P_b$	1.66	20.04	[redacted]
$P + Q$	6.99	40.08	[redacted]

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.

[redacted] used in this evaluation correspond to [redacted]

[redacted] for the circumferential and axial directions [redacted] associated with flaws oriented in the [redacted] The maximum stress reported for these components is 1.8 [redacted] used for the critical flaw evaluation, the value of 10 [redacted] 1.8 ksi. The 360-degree flaw employed for the circumferential [redacted] bounding with respect to any partial flaw in the weld, [redacted] horizontal directions. Therefore, using a minimum [redacted] acceptable, since it is less than the very conservatively determined [redacted]

3.7 References

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[redacted]

[redacted]

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9.1.1 Keeler and Long E-Series Epoxy Enamel

March, 1995 | 8111-1

HEADQUARTERS:
P. O. Box [illegible]
[illegible] Echo Lake Road
Watertown, CT [illegible]
Tel [illegible]
Fax [illegible]

PROTECTIVE COATING SYSTEMS FOR NUCLEAR POWER PLANTS

INTRODUCTION

[illegible]

[illegible]

HISTORY

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

NUCLEAR COATINGS

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	2.5 - 10.0 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	2.0 - 12.0 mils DFT
Finish	No. E-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 6.0 mils DFT
System S-11		
Primer/Finish	No. 6548/7107 EPOXY WHITE PRIMER	6.0 - 10.0 mils DFT
System S-12		
Primer/Finish	No. 4500 EPOXY SELF-PRIMING SURFACING ENAMEL	6.0 - 10.0 mils DFT
System S-14 (FLOORS ONLY)		
Finish	No. 3000 EPOXY SELF LEVELING FLOOR COATING	10.0 - 32.0 mils DFT
System S-15		
Primer	No. 6548/7107 EPOXY WHITE PRIMER	3.5 - 6.0 mils DFT
Finish	No. [illegible]	3.5 - 6.0 mils DFT
CONCRETE COATING SYSTEMS		
System XL-4		
Curing Compound/Sealer	No. 6130 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 6548 S EPOXY SURFACER	Fill - 30.0 mils DFT
Finish	No. E-1 SERIES EPOXY ENAMEL	2.5 - 6.0 mils DFT
System XL-6		
Curing Compound/Sealer	No. 6130 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 6548 S EPOXY SURFACER	Fill - 30.0 mils DFT
Finish	No. E-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 6.0 mils DFT
System XL-8		
Curing Compound/Sealer	No. 6130 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. 6548/7107 EPOXY WHITE PRIMER	2.0 - 10.0 mils DFT
Finish	No. E-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 6.0 mils DFT
System XL-10		
Curing Compound/Sealer	No. 6130 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer	No. [illegible] EPOXY SURFACER	Fill - 30.0 mils DFT
Finish	No. E-1 SERIES EPOXY HI-BUILD ENAMEL	3.0 - 6.0 mils DFT
System XL-12		
Curing Compound/Sealer	No. 6130 EPOXY CLEAR CURING COMPOUND	0.5 - 1.75 mils DFT
Surfacer/Finish	No. 4500 EPOXY SELF-PRIMING SURFACING ENAMEL	10.0 - 30.0 mils DFT
System XL-14 (FLOORS ONLY)		
Primer/Sealer	No. 6130 EPOXY CLEAR PRIMER/SEALER	1.5 - 2.5 mils DFT
Finish	No. 3000 EPOXY SELF LEVELING FLOOR COATING	30.0 - 35.0 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. [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	Steel	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	Concrete	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

[illegible]

KEELER & LONG

E.340

HEADQUARTERS:
P. O. Box 460
856 Echo Lake Road
Watertown, CT 06795
Tel (860) 274-3701
Fax (860) 274-8387

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	63% ± 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.

P. O. Box 460, 844 Echo Lake Road
Watertown, CT 06795
Tel: (860) 274-6791 Fax: (860) 274-5437

[illegible]

3.8.2 Ameron PSX 738 Siloxane Coating

Ameron

PSX® 738
Engineered Siloxane® coating
Patent No. 5,119,[illegible]

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 (25%)	up to 200°F
DI Water	up to 200°F
Sodium [illegible]	Ambient
Ethylene glycol	Ambient
Acetone	Ambient
Methylene chloride	Ambient
Xylene	Ambient
Mineral spirits	Ambient
JP-4 fuel	up to 110°F
Petroleum ether	Ambient
Kerosene	up to 140°F
Transmission fluid	Ambient
Acetic acid (glacial)	Ambient
[illegible]	up to 122°F
Sulfuric acid (25%)	Ambient
Nitric acid (1%)	Ambient
Hydrochloric acid [illegible]	Ambient

[illegible]

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)	
[illegible]		
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 21-5	750	400
Flash point (SETA)	°F	°C
liquid	55	13
Amercoat 935	208	98
Amercoat 101	145	63
Amercoat 13	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 21-5	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.	3 hours	
Environmental conditions		
Temperature	°F	°C
air	45 to 120	7 to 49
surface	See [illegible]	
Relative humidity during application and initial drying		
minimum	40%	
maximum	Avoid condensation	

[illegible]

Formerly Amercoat 738

[illegible]

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	[illegible]	Service Temperature °F/°C [illegible]	[illegible]	[illegible]	[illegible]
Steel	15/101	1	8	10	10	10	15
	15/101	2	NR	10	10	14	20
	935	1	NR	8	8	13	15
Stainless	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 [illegible] in California.

Warranty

[illegible]

Limitation of Liability

[illegible]

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	34	72

NR = Not Recommended

**Note – at [illegible] DFT Drying and Cure times will increase by [illegible].

***Note – Thinned material at [illegible] will increase cure times.

Thinners
Amercoat 15 (up to 100°F surface temperature)
Amercoat 101 (up to 140°F surface temperature)
Amercoat 935 (140° - 300°F surface temperature)
DO NOT use Amercoat 935 below 140°F surface temps. [illegible]

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	[illegible] months from manufacturing date	
powder	2 years from shipment date	

[illegible]

Ameron.
Protective Coatings Systems

201 North Berry Street, Brea, California [illegible] [illegible] The Netherlands

[illegible]

Ameron

PSX® 738

Engineered Siloxane® coating

Patent No. 4,113,[illegible]

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

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.

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: 8 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 [illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Steel	15/101	1	8	10	10	10	15
	15/101	2	NR	10	10	14	20
	925	1	NR	8	8	12	12
Dimetcote	15/101	1	—	—	—	10	13
	925	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	90
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	60
thinned**	10	15	25	60	NR
@ 75% R.H.					
unthinned	3	6	10	25	40
thinned**	6	12	15	36	72

NR = Not Recommended

*Note – at 12 mils total DFT Drying and Cure times will increase by 20%. **Note – Thinned material at 8 mils will increase cure times.

6 Thin with recommended thinner as required for workability, see below.

Thinners

Amercoat 15 (up to 100°F surface temperatures)
Amercoat 101 (up to 140°F surface temperatures)
Amercoat 925 (140° - 200°F surface temperatures)
DO NOT use Amercoat 925 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.

Ameron
Protective Coatings
Systems

201 North Berry Street Brea, California 92821 [illegible] 4191 [illegible] The Netherlands

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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 [illegible] 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	LONG-TERM	SHORT-TERM	REFERENCE
Concrete	150(B)/200(L)	350	ACI 349
[redacted] Fuel Cladding	[redacted]	[redacted]	PNL-6189 PNL-4835
[redacted]	[redacted]	[redacted]	[redacted]
Aluminum Disk	[redacted]	[redacted]	MIL-HDBK-5G
NS-4-FR	[redacted]	[redacted]	[redacted]
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	7[illegible]0	ASME Code Case N-71-[redacted]
ASTM A36 Carbon Steel	700	700	ASME Code Case N-71-[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

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.

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.

[redacted]

Table 4.1-4 Summary of Thermal Evaluation for NAC-MPC Storage System

Design Condition	Concrete	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Allowable Long-Term	150(Bulk) 200(Local)	[illegible]	[illegible]	[illegible]	600	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Short-Term	350	[illegible]	700	800	600	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Long-Term Conditions Normal (75°F Ambient)	133(Bulk) 165(Local)	[illegible]	[illegible]	N/A	N/A	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Short-Term Conditions Off-Normal Half Inlets Blocked (75°F Ambient)	168	565	529	N/A	N/A	[illegible]	192	[illegible]	[illegible]	[illegible]
Off-Normal Severe Heat (100°F Ambient)	196	587	532	N/A	N/A	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Off-Normal Severe Cold [illegible]	5	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Accident [illegible]	215	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	N/A	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	N/A	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

1. Concrete [illegible]
2. [illegible]
3. [illegible]
4. [illegible]
5. [illegible]

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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[3]	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

	Temperature (°F)[3]			
Property (units)	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 Faumeister.

3 Maximum service temperature: 600°F (based on [illegible] 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[1] (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 3.6.2.1).

2. Recommendeu 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
Specific Heat[1] (Btu/lbm-°F)	1.24	1.24	1.24	1.24
Density[1] (lbm/in^3)	4.83E-6	3.70E-6	3.01E-6	2.52E-6

[1] Kreith.

Table 4.2-6 Thermal Properties of Dry Air

	Temperature (°F)			
Property (units)	100	300	500	700
Conductivity[1] (Btu/hr-in-°F)	0.00128	0.00161	0.00193	0.00223
Density (lbm/ft^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) 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.

[redacted]

Table 4.2-8 Thermal Properties of ASTM A 36, ASTM A 588 [redacted] Carbon Steel

Property (units)	Temperature (°F) 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.
[redacted]

Table 4.2-9 Thermal Properties of Zircaloy and Zircaloy-4 Cladding

	Temperature (°F)			
Property (units)	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[1]	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

	Temperature (°F)[3]	
Property (units)	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. All models are generated by the ANSYS program.

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):

$$\left.\begin{aligned} Nu &= 0.13(Gr\cdot Pr)^{1/3} \\ h_c &= Nu\cdot k_f / H_{VCC} \end{aligned}\right\} \quad \text{for } Gr > 10^9$$

Top surface (Incropera):

$$\left.\begin{aligned} Nu &= 0.15Ra^{1/3} \\ h_c &= Nu\cdot k_f / L \end{aligned}\right\} \quad \text{for } Ra > 10^7$$

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 $\varepsilon_1 = 0.9$ 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 $\varepsilon_1 = 0.8$ for a carbon steel surface is used, and a view factor (F_{12}) = 1 is applied.

[redacted]

[redacted]

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.)

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

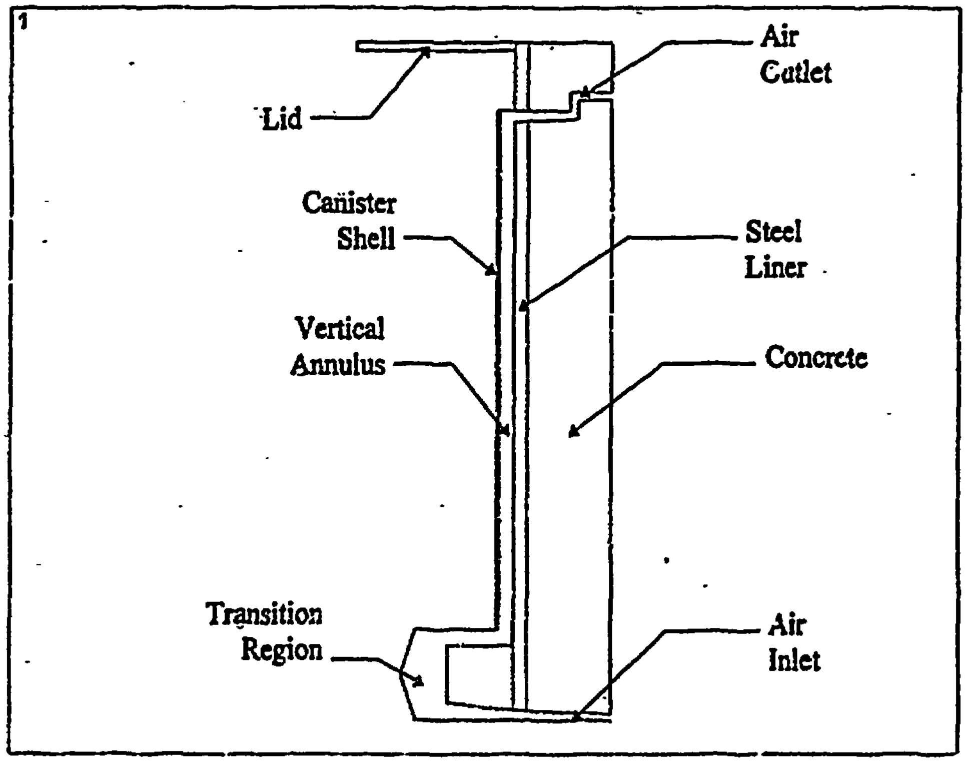


Figure 4.4.1.1-2 Two-Dimensional Axisymmetric Air Flow and Concrete Cask Finite Element Model

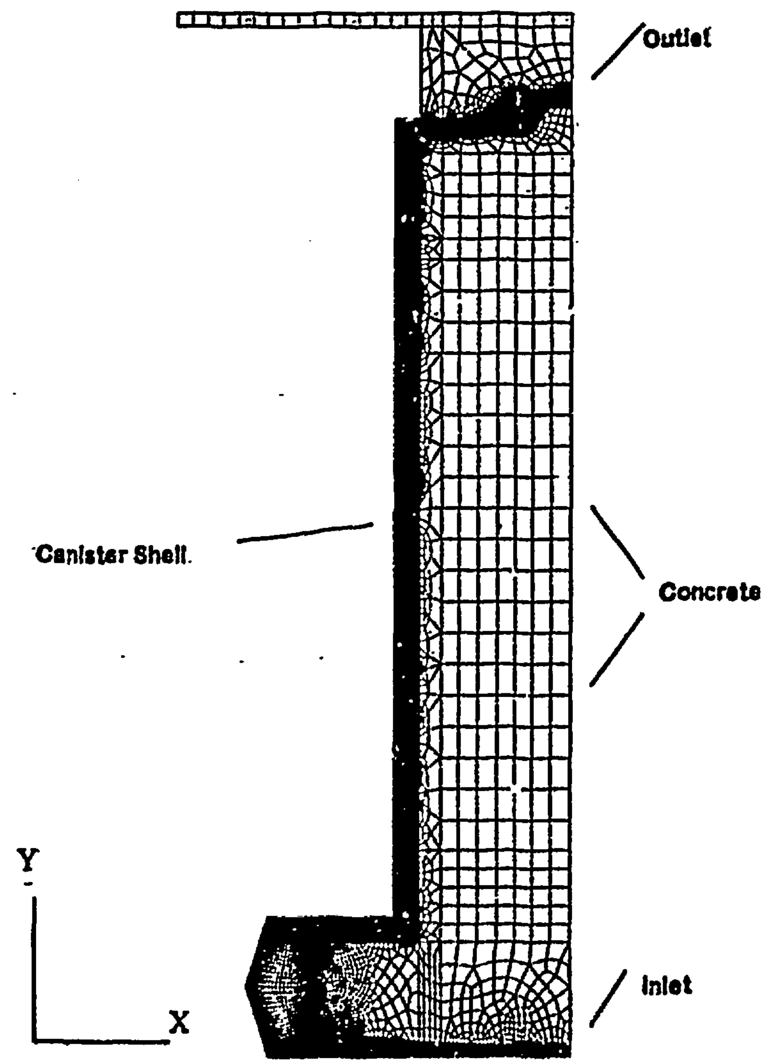


Figure 4.4.1.1-3 Axial Power Distribution for Yankee Class Fuel

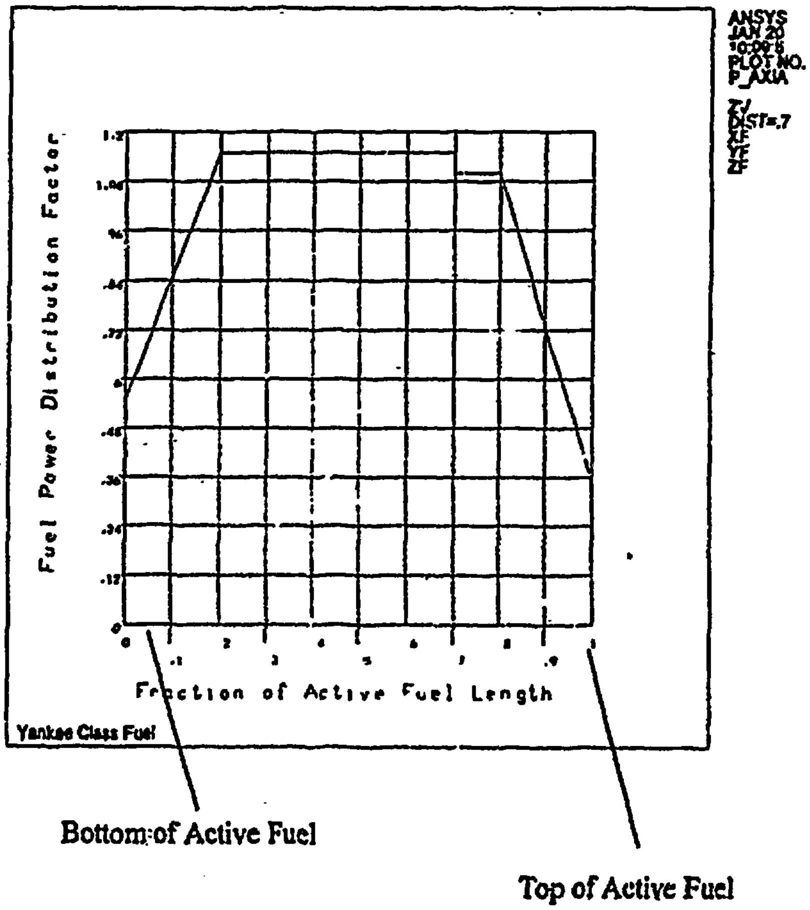


Figure 4.4.1.1-4 Convergence Process of Mass Flow Rate

Mass in: mass flow rate at the air inlets, kg/s

Mass out: mass flow rate at the air outlets, kg/s

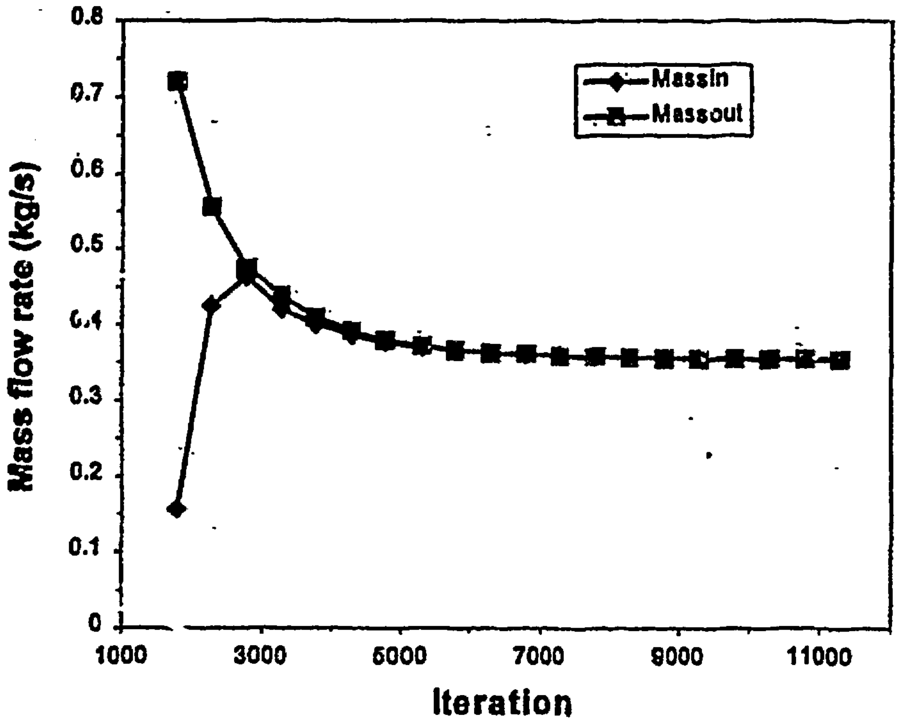


Table 4.4.1.1-1 Comparison of Numerical Results Using Different Element Sizes and Number of Elements

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. [redacted]

[redacted] 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:

$$q_r = \sigma \varepsilon_{eff} A F \left(T_i^4 - T_j^4\right)$$

where

σ = the Stefan-Boltzman constant
= 1.19×10^{-11} Btu/hr-in^2-°R^4
ε[redacted] = [redacted] 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-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

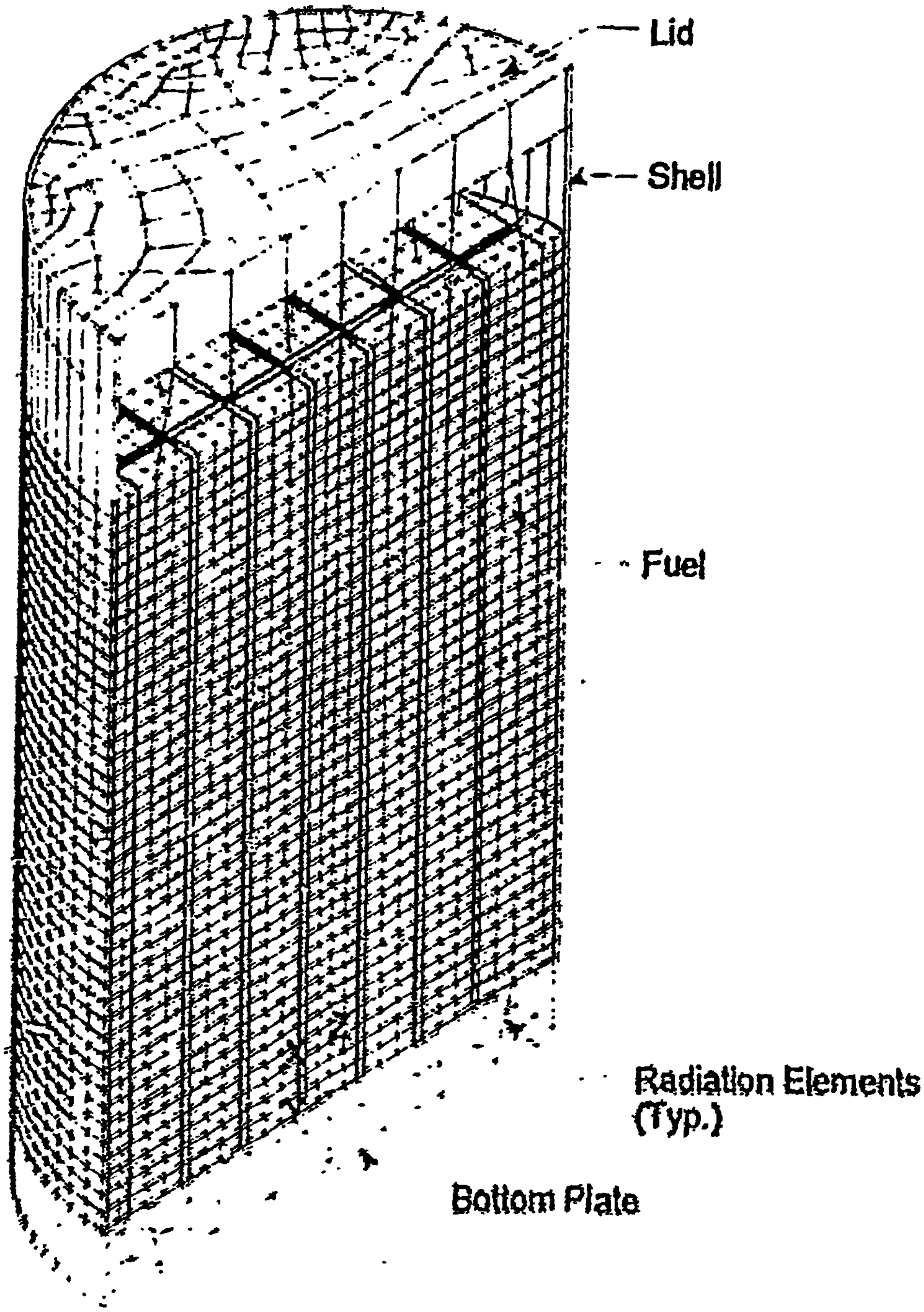
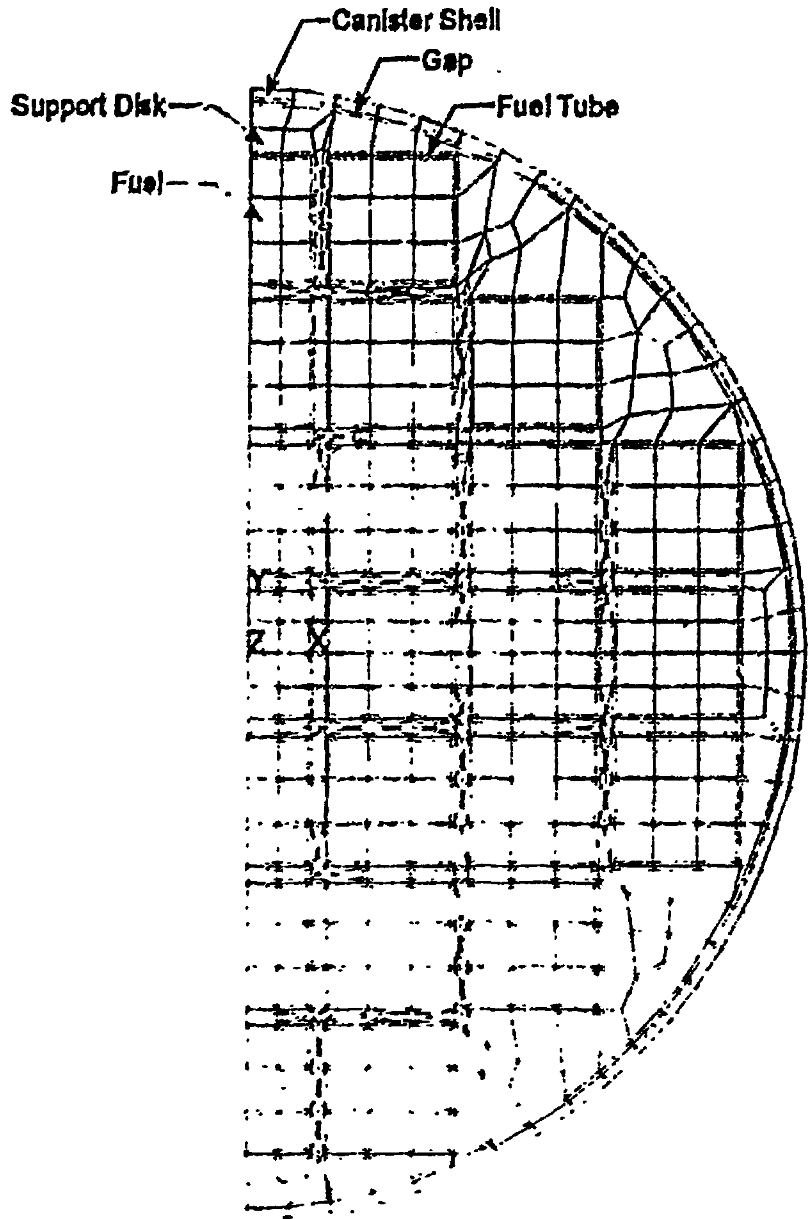


Figure 4.4.1.2-2 Three-Dimensional Canister Model - Cross-Section



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.

$$T_c = T_s + 0.29468 \, (Q a^2 / K_{eff})$$

where T_c = the temperature at the center of the fuel (°F)
T_s = the temperature applied to the exterior of the fuel (°F)
Q = volumetric heat generation rate (Btu/hr-in³)
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.

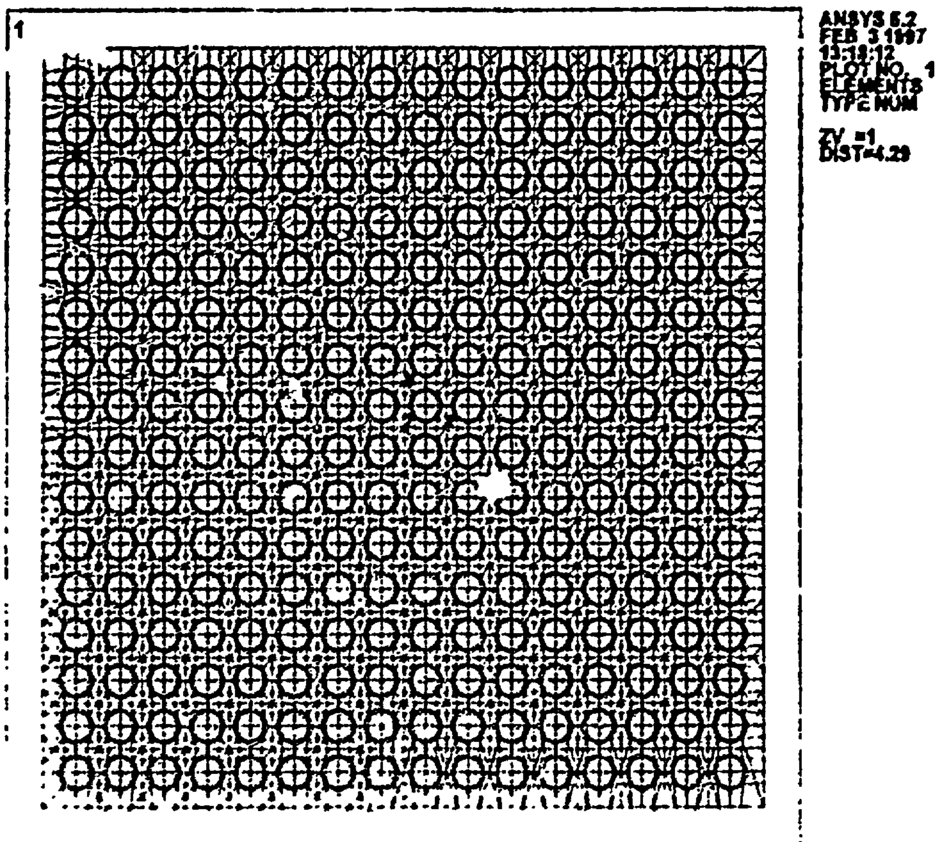
Conductivity (Btu/hr-in-°F)

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

$$Kyy = \frac{\sum K_i t_i}{T}$$

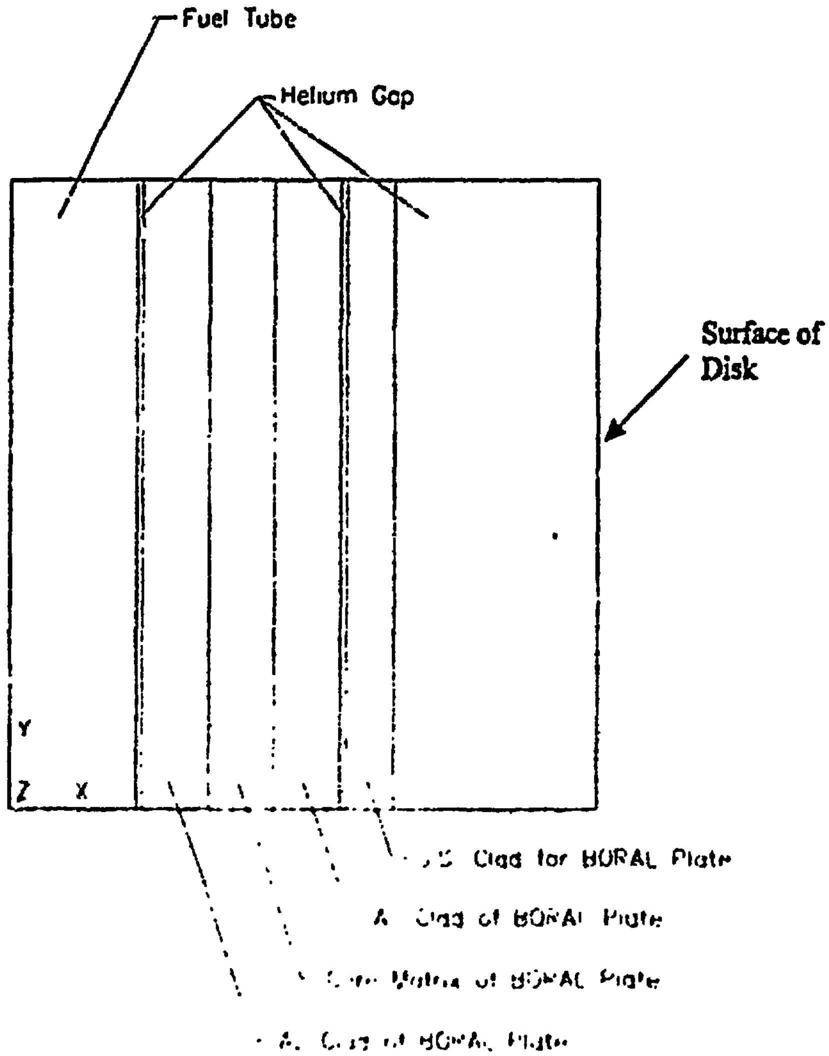
where

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:

$$q_r = \sigma \epsilon_{eff} A F (T_s^4 - T_t^4)$$

where

σ = the Stefan-Boltzman constant
= 1.19×10^{-11} Btu/hr-in^2-°R^4
ε = 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-in^2)

The q_r can be rewritten as:

$$q_r = \sigma\varepsilon AF(T_i^2 + T_j^2)(T_i + T_j)(T_i - T_j)$$

By combining both expressions:

$$Q_t = (\sigma\varepsilon F(T_i^2 + T_j^2)(T_i + T_j) + h_c)A(T_i - T_j)$$

or

$$Q_t = h_{eff} A(T_i - T_j)$$

where

$$h_{eff} = \sigma\epsilon F(T_i^2 + T_j^2)(T_i+T_j) + h_c$$

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.

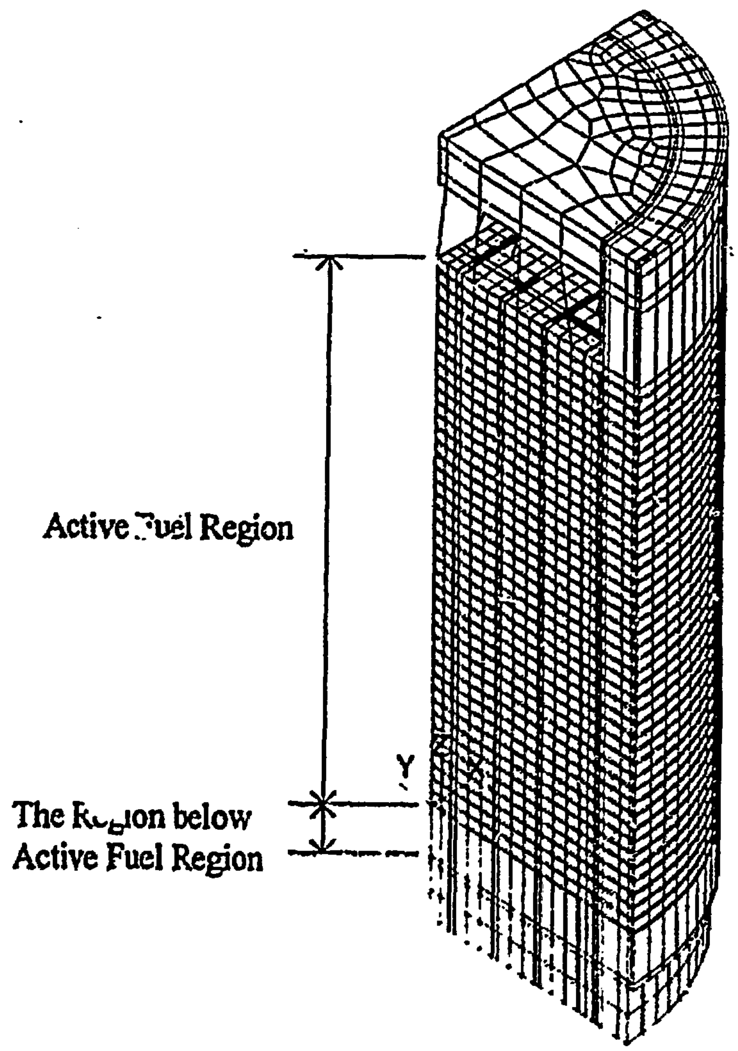
The convection heat transfer coefficient is determined by the following expression, which is established by empirical correlation for vertical plate and cylinders and horizontal plates (Kreith):

$$h_c = 0.0015\ \Delta T^{1/3}\ (\text{Btu/hr-in}^2\text{-}^\circ\text{F})$$

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 ϵ_1 and ϵ_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.

$$\text{Heat generation rate} = Q/V = 0.6595 \text{ Btu/hr-in}^3$$

where;

$$Q = \text{heat rate per pin (unit height)} = (0.0016)(3413)(1.15)/(91) = 0.069 \text{ Btu/hr}$$

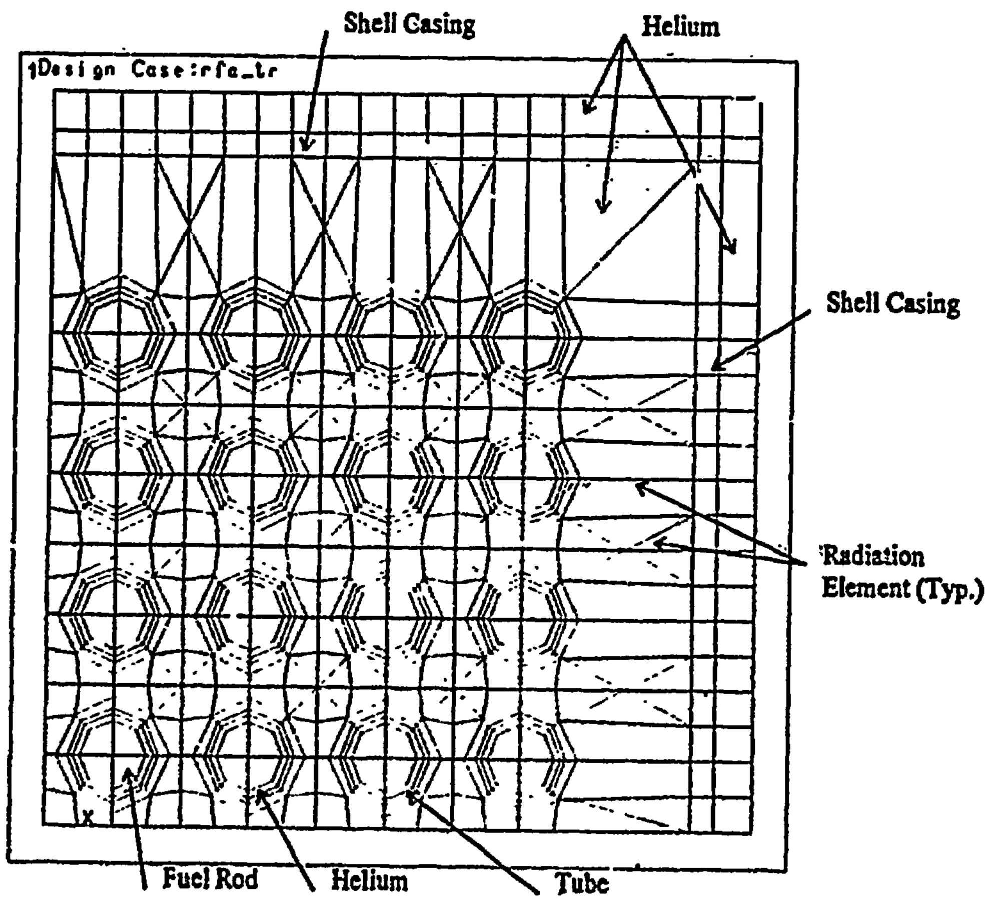
$$V = \text{volume of pin (unit height)} = \pi\, 0.365^2/4 = 0.1046 \text{ inch}^3$$

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 [redacted] 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 [redacted] (0.347 kW).

Design Condition	Maximum [redacted] Tube[redacted] Temperature (°F)
Normal	540
Off-Normal and Accident	580
Transfer	715

[redacted]

Figure 4.4.1.[redacted]-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

Figure 4.4.3-1 shows the temperature distribution of the concrete storage 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 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 transient history of maximum temperatures of the fuel region in the canister for the transfer conditions (canister containing water for [redacted] hours, vacuum for 10 hours [redacted]) is shown in Figure 4.4.3-5. Table 4.4.3-1 shows the maximum component temperatures for the normal condition of storage. The maximum component temperatures for the transfer conditions, [redacted] condition with helium inside the canister and the transient condition of vacuum [redacted] are shown in Tables 4.4.3-2 and 4.4.3-3, respectively. The maximum component temperatures for the reconfigured fuel assembly are shown in Table 4.4.3-4. Temperature distributions for the off-normal and accident conditions are presented in Sections 11.1 and 11.2.

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 storage cask. The air in the center of the annulus exhibits a much lower temperature gradient, indicating cooler air. The higher temperature at the storage 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.

For the transient transfer condition, the maximum temperature of the water is [redacted]°F at the end of 20 hours based on an initial water temperature of 100°F.

Figure 4.4.3-1 Temperature Distribution (°F) for Normal Storage

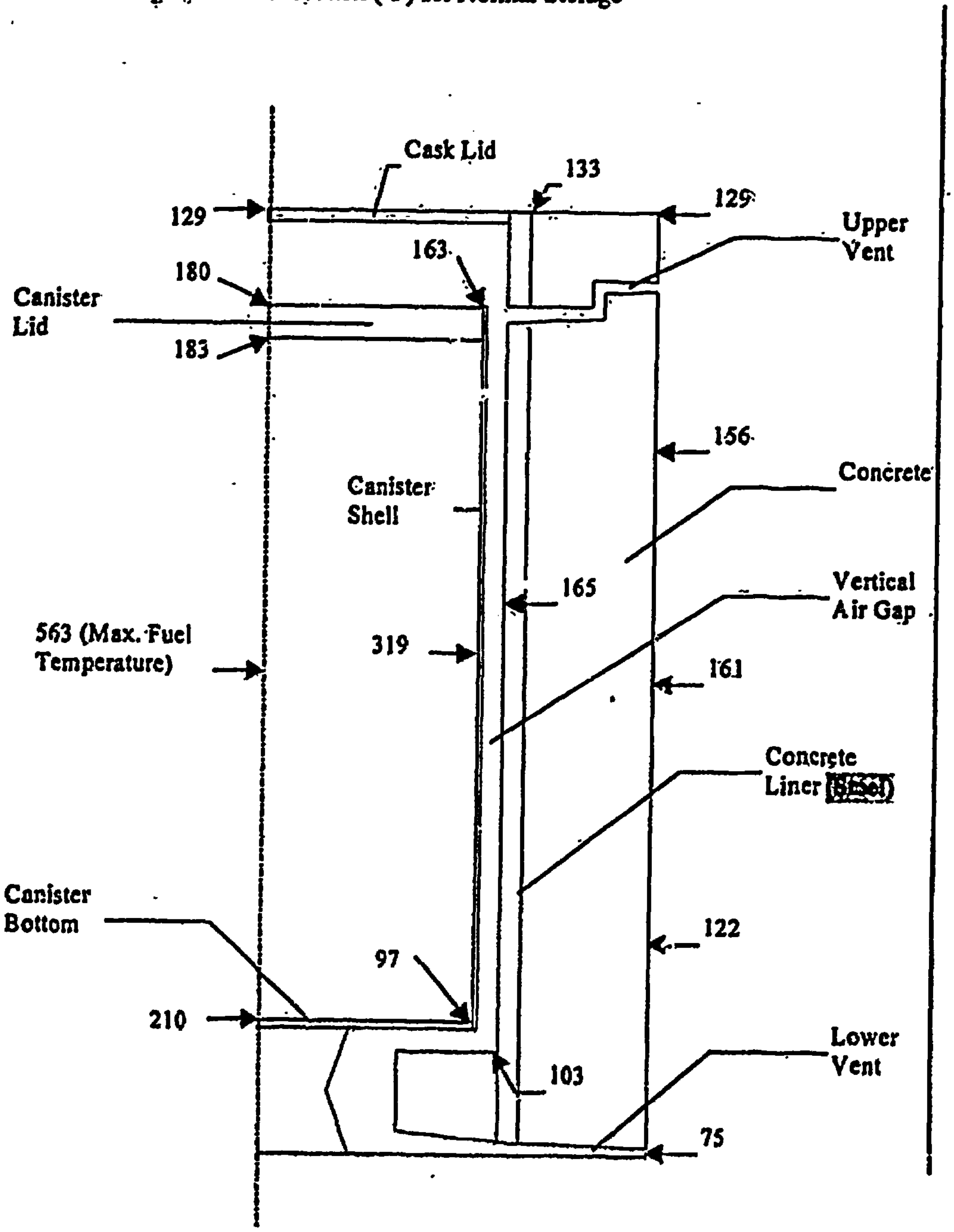


Figure 4.4.3-2 Air Flow Pattern in the Storage Cask in Normal Storage

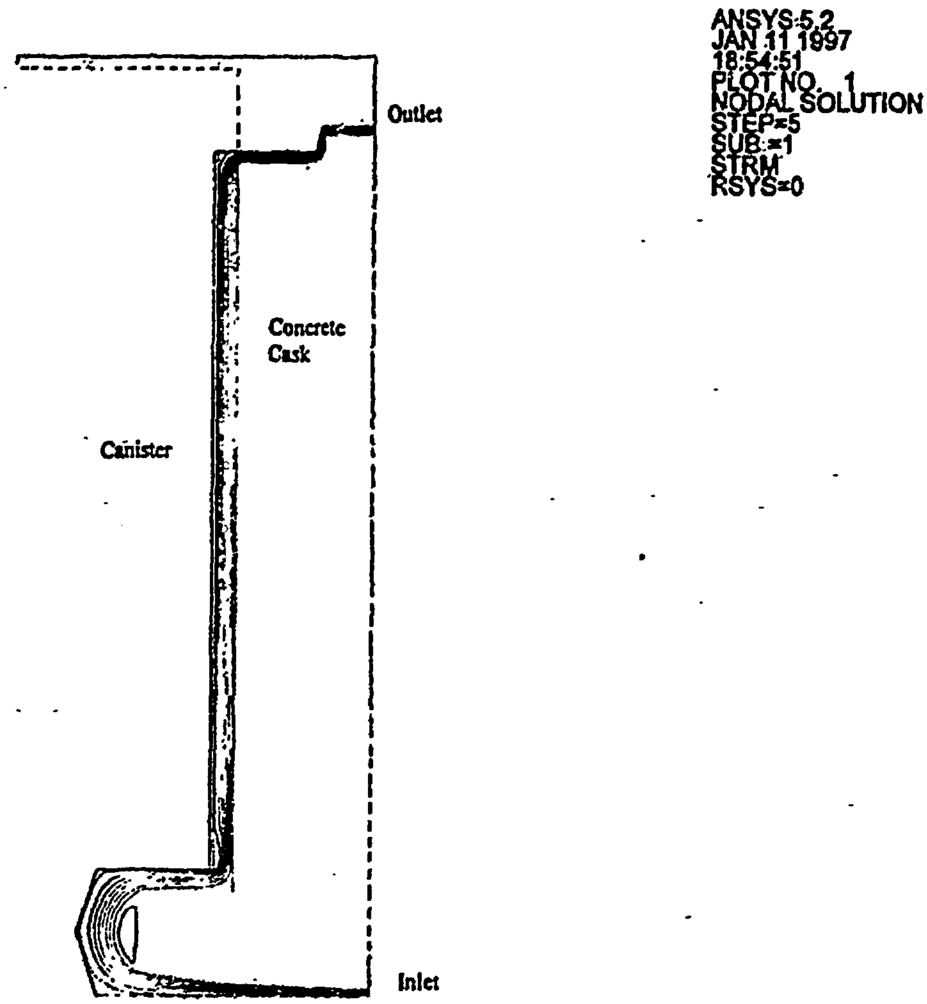
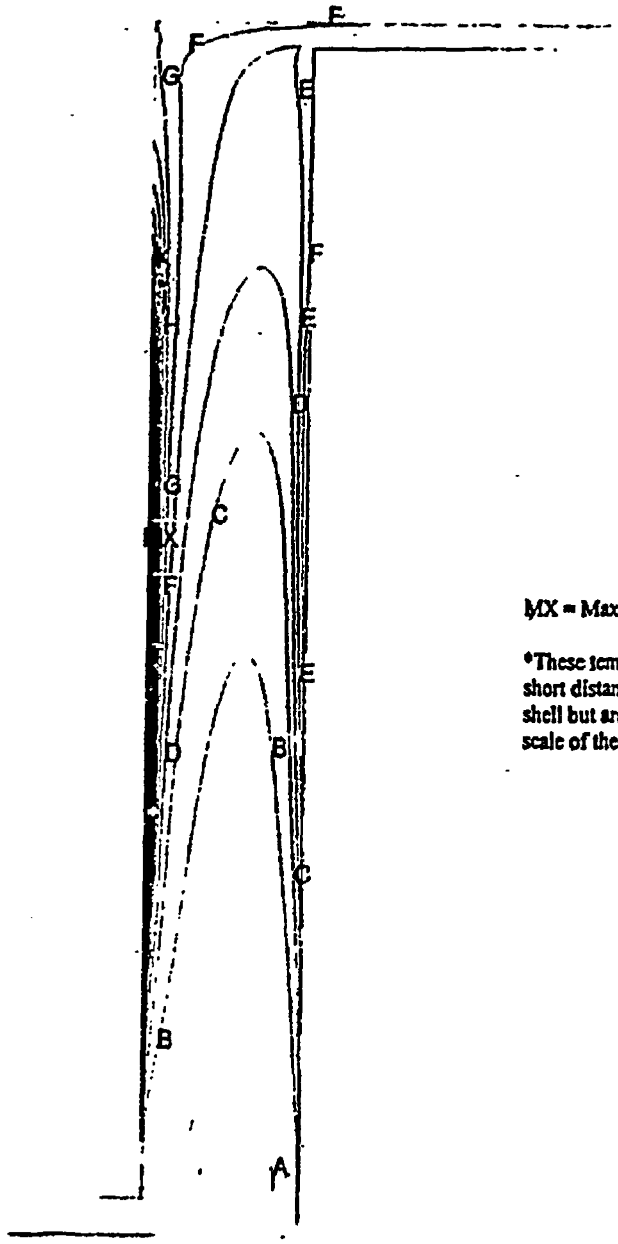


Figure 4.4.3-3 Air Temperature Field in the Storage Cask During the Normal Storage Condition

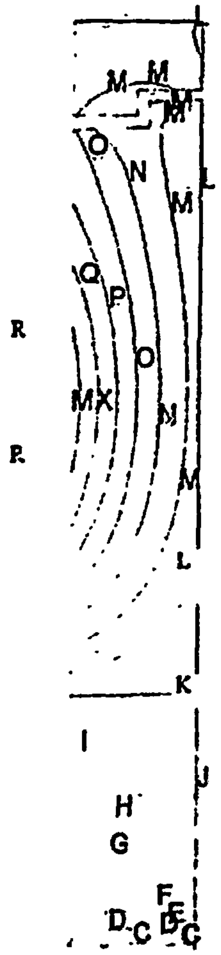


°F	
MX=	315
A=	79
B=	93
C=	106
D=	120
E=	133
F=	147
G=	160
H=	174
I=	187
J=	201
K=	214
L=	228
*M=	241
*N=	255
*O=	268
*P=	282
*Q=	295
*R=	309

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

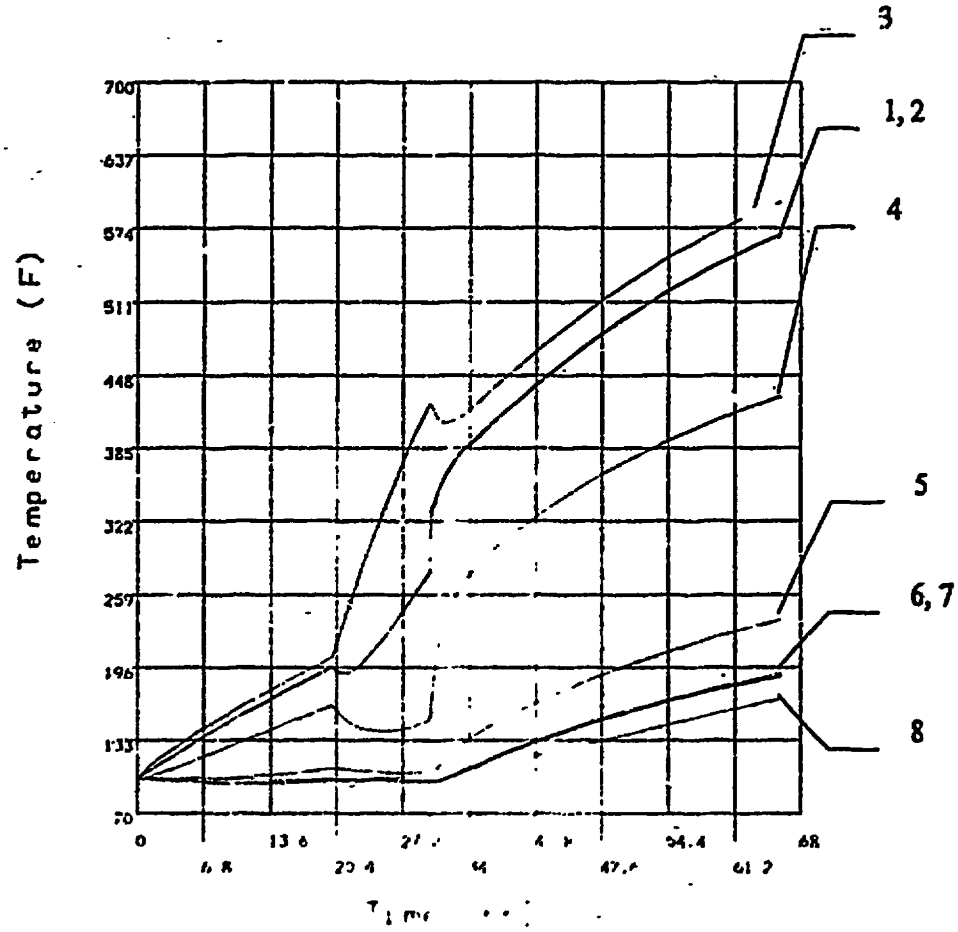


°F
MX= 165
*A= 77
*B= 82
C= 87
D= 92
E= 97
F= 102
G= 107
H= 112
I= 117
J= 122
K= 127
L= 132
M= 137
N= 142
O= 147
P= 152
Q= 157
R= 162

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 4.4.2-5 History of Maximum Component Temperatures for 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	[redacted]
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 [redacted] Transfer Condition [redacted]

Component	Maximum Temperature (°F)	Allowable Temperature (°F)
Fuel	[redacted]	[redacted]
Lead	[redacted]	600
Neutron Shield	[redacted]	[redacted]
Aluminum Disk	[redacted]	[redacted]
Support Disk	[redacted]	800
Canister	[redacted]	800
Transfer Cask Shells	[redacted]	700

[redacted]

Table 4.4.3-3 Maximum Component Temperatures for the [redacted] Transfer Condition [redacted]

Component	Maximum Temperature (°F)[redacted]	Allowable Temperature (°F)
Fuel	[redacted]	[redacted]
Lead	[redacted]	600
Neutron Shield	[redacted]	[redacted]
Aluminum Disk	[redacted]	[redacted]
Support Disk	[redacted]	800
Canister	[redacted]	800
Transfer Cask Shells	[redacted]	700

[redacted]

Table 4.4.3-4 Maximum Component Temperatures for the Reconfigured Fuel Assembly

Design Condition	Maximum Temperatures (°F)			
	[redacted]	[redacted]	[redacted] Tube [redacted]	Fuel [redacted]
Normal	[redacted]	543	563	563
Off-Normal and Accident Conditions	[redacted]	583	602	602
Transfer	[redacted]	718	734	734

[redacted]

[redacted] Fuel cladding allowable temperatures:

Normal conditions: 644°F

Off-normal, Accident and Transfer conditions: [redacted]°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 ' 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:

$$N = \frac{Pv}{RT}$$

$$N = \frac{\left\{(315\ \text{psig} + 14.7) \times \frac{1\ \text{atm}}{14.7\ \text{psia}}\right\} \times 83.605\ \frac{\ell}{\text{Cask}}}{0.0821\ \frac{\text{atm}\ \ell}{\text{Mole K}} \times 293\ \text{K}} = 77.95\ \frac{\text{Moles of Rod Fill Gas}}{\text{Cask}}$$

The number of moles of fission gas per assembly is:

Isotope	Atomic Weight (gram/mole)	Mass (gram)	Number of Moles
KR	83	10.9	[illegible]
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.

$$V^{TSC}_{Free\ Gas\ Volume} = V_{Canister} - \left(\frac{\left(M_{TSC\ Shield\ Lid} + M_{TSC\ Structural\ Lid}\right)}{\rho_{Steel}} + V^{TSC}_{Basket} + V_{Fuel} \right)$$

$$V_{Canister} = \pi \frac{d^2}{4}\left(L_{Canister} - L_{TSC\ Bottom\ Plate}\right) = \pi \times \frac{(69.39\ \text{inches})^2}{4} \times (122.50\ \text{inches} - 1.0\ \text{inch})$$

$$V_{Canister} = 459{,}472.93\ \text{inches}^3$$

$$V^{TSC}_{Basket} = \frac{\left(M^{TSC}_{Basket} - \left(M_{BORAL} + M_{Aluminum} + M_{Support\ Disks}\right)\right)}{\rho_{Steel}} + \frac{M_{BORAL}}{\rho_{BORAL}} + \frac{M_{Aluminum}}{\rho_{Aluminum}} + \frac{M_{Support\ Disk}}{\rho_{17\text{-}4\ PH}}$$

$$V^{TSC}_{Basket} = \frac{(9{,}530\ \text{lb} - (694.81\ \text{lb} + 810\ \text{lb} + 3{,}720\ \text{lb}))}{0.288\ \frac{\text{lb}}{\text{in}^3}} + \frac{694.81\ \text{lb}}{0.095\ \frac{\text{lb}}{\text{in}^3}} + \frac{810\ \text{lb}}{0.098\ \frac{\text{lb}}{\text{in}^3}} + \frac{3{,}720\ \text{lb}}{0.282\ \frac{\text{lb}}{\text{in}^3}}$$

$$V^{TSC}_{Basket} = 43{,}719.16\ \text{inches}^3$$

$$V^{TSC}_{Free\ Gas\ Volume} = 459{,}472.93 - \left(\frac{(5{,}390\ \text{lb} + 3{,}230\ \text{lb})}{0.288\ \frac{\text{lb}}{\text{in}^3}} + 43{,}719.16\ \text{inches}^3 + 88{,}171.78\ \text{inches}^3 \right)$$

$$V^{TSC}_{Free\ Gas\ Volume} = 297{,}651.43\ \frac{\text{inches}^3}{\text{Cask}}$$

$$V^{TSC}_{Free\ Gas\ Volume} = 297{,}651.43\ \frac{\text{inches}^3}{\text{Cask}} \times \frac{1\ \ell}{61.02\ \text{inches}^3} = 4{,}877.93\ \frac{\ell}{\text{Cask}}$$

$$N = \frac{1\ \text{atm} \times 4{,}877.93\ \frac{\ell}{\text{Cask}}}{0.0821\ \frac{\text{atm}\ \ell}{\text{Mole}\ K} \times 339\ K} = 175.26\ \frac{\text{Moles of TSC Backfill Gas}}{\text{Cask}}$$

The maximum normal operating pressure (MNOP) in the canister is calculated using the ideal gas law where:

$$N = N_{TIC\ Back\text{-}Fill} + 0.03\left(N_{Rod\ Back\text{-}Fill}\right) + 0.3(0.03)\left(N_{Fission\ Gas}\right)$$

$$N = \text{[redacted]}\ \frac{\text{Moles}}{\text{Cask}} + 0.03\left(77.95\ \frac{\text{Moles}}{\text{Cask}}\right) + 0.3(0.03)\left(\text{[redacted]}\ \frac{\text{Moles}}{\text{Cask}}\right)$$

$$N = \text{[redacted]}\ \frac{\text{Moles}}{\text{Cask}}$$

Therefore, the maximum normal operating condition canister internal pressure is:

$$P = \frac{\left(\text{[redacted]}\ \frac{\text{Moles}}{\text{Cask}}\right) \times \left(0.0821\ \frac{\text{atm}\ \ell}{\text{mole K}}\right) \times 505.37\ \text{K}}{\left(4{,}877.93\ \frac{\ell}{\text{Cask}}\right)} = 1.54\ \text{atm} = \text{[redacted]}\ \text{psia} = \text{[redacted]}\ \text{psig}$$

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.

[redacted]

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 [redacted] 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.

[redacted]

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 [redacted] shielding analysis is not required for the reconfigured fuel assembly.

The transfer cask has a multiwall radial shield [redacted] 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 [redacted] operations. This temporary shielding is removed prior to storage.

The [redacted] 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.

[redacted]

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, [redacted] 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. [redacted] the new code sequence is [redacted] the user to obtain surface [redacted] tallies on the cask surfaces other than the cask [redacted] accept user-defined surface detectors instead of the [redacted] SCALE 4.3 SAS4. Each surface can be defined by [redacted] location and extent. Each surface may be broken into [redacted] were tested against the SCALE 4.3 manual and NAC test cases. [redacted] 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 [redacted] transfer cask [redacted] with a wet and dry canister cavity are shown in Table 5.1-[redacted] The maximum dose rates with design-basis fuel and the canister cavity wet during shield lid welding operations are 210.2 (0.8%), [redacted] (1.1%) and [redacted] (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 ([redacted]%) and [redacted] ([redacted]%) 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	[illegible]
	gamma	47.0	11.5	67.0	21.5	4.0	[illegible]
	Total	47.3 (0.4%)	54.0 (4.9%)	99.0 (5.4%)	21.8 (1.3%)	15.4 ([illegible]%)	[illegible] ([illegible]%)
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	97.0	39.8	388.8	19.0
	Total	210.2 (0.8%)	188.7 (1.1%)	97.2 (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	[illegible]	[illegible]	26.1	[illegible]
	Total	413.4 (1.5%)	358.9 (2.6%)	[illegible] ([illegible]%)	103.4 (0.6%)	39.6 (4.5%)	[illegible] ([illegible] %)

[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 (Hermann) 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-[redacted].

The NAC-MPC design basis fuel source terms are shown Table 5.2-[redacted]. Source strengths are defined for five source regions: active fuel, upper end fitting, upper plenum, lower end fitting and lower plenum. [redacted] 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.

[redacted]

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 ^{55}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 is 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. [redacted] on SAS2H calculations of the neutron source [redacted], a 4.2 power dependence of the neutron source on burnup, i.e., neutron source ~ $B^{4.2}$ [redacted]. This yields a 1.80 peaking facto-.for neutrons.

Figure 5.2-1 Yankee Class Combustion Engineering Design Basis Fuel Assembly Array

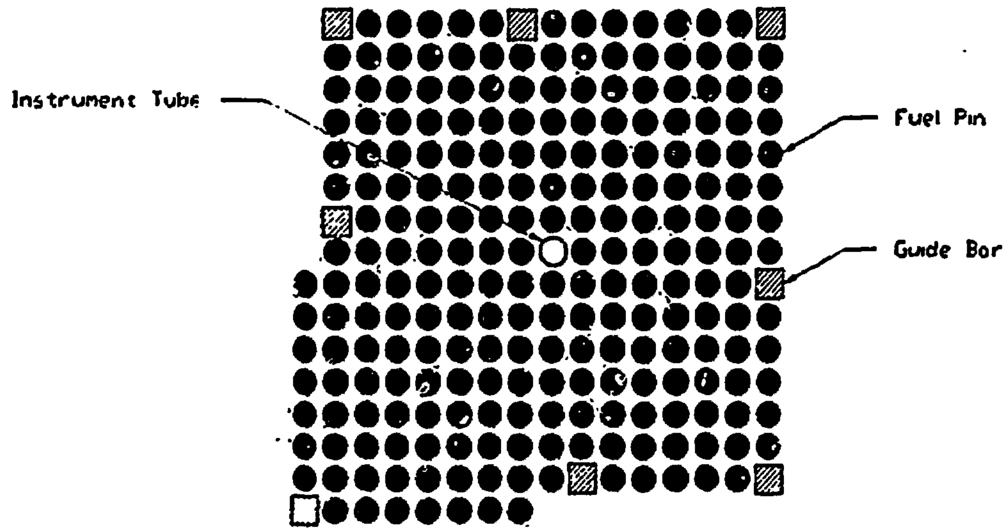
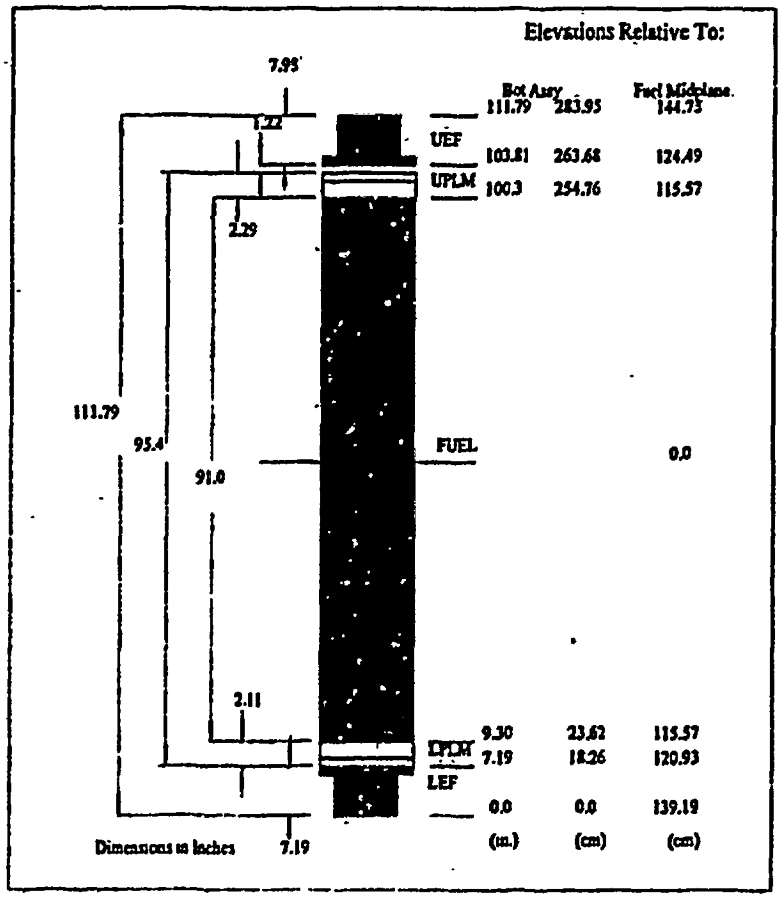
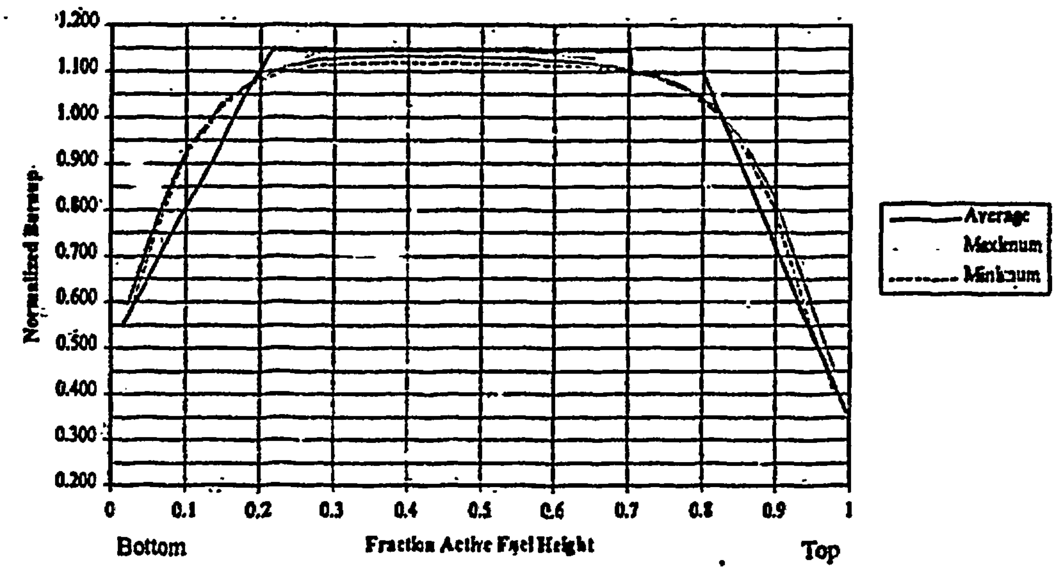


Figure 5.2-2 Yankee Class Combustion Engineering Design Basis Fuel Assembly Source Regions and Elevations



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 [redacted]



Condition: 30,000 - 36,000 [redacted]

Figure 5.2-4 NAC-MPC Design Basis Neutron and Gamma Source Axial Profiles

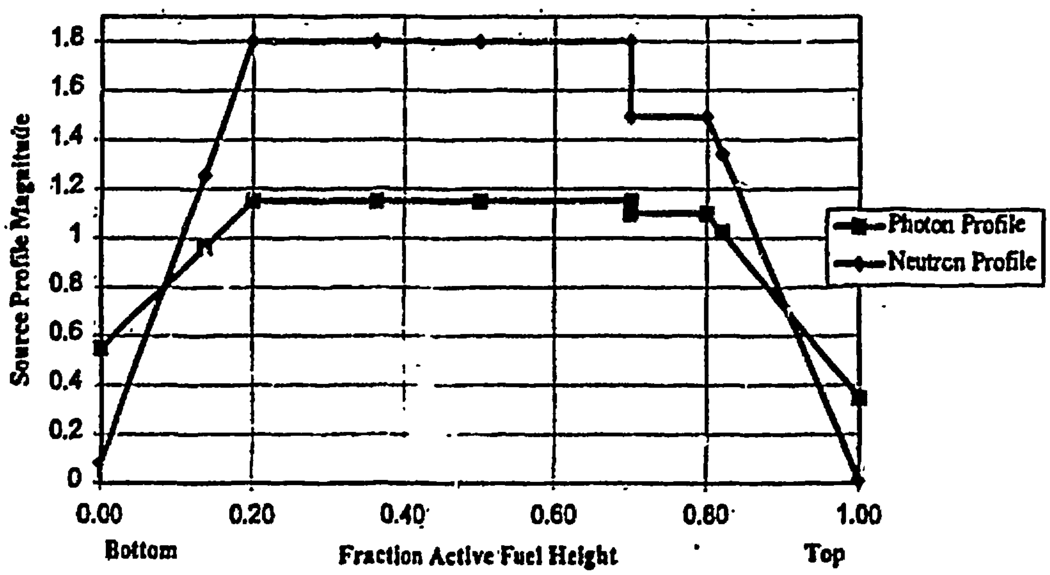


Table.5.2-1 Yankee Class Fuel Assembly Physical Parameters

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	[redacted]	[redacted]	4.94	4.94	4.00	4.00
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	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 [redacted]

[redacted]	[redacted]
[redacted]	[redacted]
Guide Bar Configuration	[redacted]
Guide Bar Material	[redacted]
Guide Bar Width (inches)	[redacted]
Guide Bar Length (inches)	[redacted]
Guide Bar Shape	[redacted]
Number Per Assembly	[redacted]
Instrument Tube Configuration	[redacted]
Instrument Tube ID (inches)	[redacted]
Instrument Tube OD (inches)	[redacted]
Instrument Tube Length (inches)	[redacted]
Number Per Assembly	[redacted]
Instrument Tube Material	[redacted]
Hardware Configuration	[redacted]
Top Nozzle Material	[redacted]
Bottom Nozzle Material	[redacted]
Upper Plenum Spring Material	[redacted]
Top Nozzle Length (inches)	[redacted]
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 [redacted]
Lower Plenum Mass (kg)	1.73
Incore Grid Spacers	Zircaloy-4

Notes: 1. For simplicity, all Inconel and steel are modeled as [redacted]
2. Includes Inconel grid and lower plenum spacer.
3. Combustion Engineering fuel may be loaded at a cool [redacted]
burnup of 32,000 MWD/MTU and a minimum [redacted]

Table 5.2-[redacted] 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-[redacted] NAC-MPC Design Basis Fuel Source Terms [redacted]

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, [illegible] Years Cooled, and 36,000 MWD/MTU Burnup.

Table 5.2-5 NAC-MPC Design Basis Fuel Gamma Source Spectra [redacted]

Group	[redacted]	
	[redacted]	[redacted]
1	3.7701E+04	[redacted]
2	1.7759E+05	[redacted]
3	9.0547E+05	[redacted]
4	2.2566E+06	[redacted]
5	6.2676E+08	[redacted]
6	5.1211E+09	[redacted]
7	1.0789E+11	[redacted]
8	9.9933E+10	[redacted]
9	4.8070E+12	[redacted]
10	3.4718E+13	[redacted]
11	6.3503E+13	[redacted]
12	8.2333E+14	[redacted]
13	1.1897E+14	[redacted]
14	1.7831E+13	[redacted]
15	2.8386E+13	[redacted]
16	1.0201E+14	[redacted]
17	1.3136E+14	[redacted]
18	4.5899E+14	[redacted]
Total	1.7842E+15	[redacted]

1. 36,000 MWD/MTU and 8 years cool time.

Table 5.2-[redacted] NAC-MPC Design Basis Fuel Neutron Source Spectra [redacted]

Group	Neutrons/sec
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. 35,000 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. [redacted] 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 zones, the active fuel, the plenum and the end fittings [illegible]. 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, 455-891, [illegible] and [illegible] (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 resulting 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 [redacted] 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, [illegible]125 inches ([illegible] 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.

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 [illegible] 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

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 volume[illegible] 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 [illegible] top fuel, top plenum and top end fitting regions [illegible] elevations with respect to the fuel midplane of [illegible] 45.50, 49.01, and 56.99 inches (115.57, [illegible]4.49 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 [illegible] 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 ar[illegible] [illegible]ial 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

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 [illegible] fuel assembly and shield densities are provided in Table 5.3-3.

Figure 5.3-1 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	9.30[illegible]E+04
LPLM	9.9462E+04	2.73[illegible]
$FUEL_{bot}$	5.8[illegible]E+05	[illegible]
$FUEL_{mid}$	2.9[illegible]E+06	[illegible]
$FUEL_{top}$	7.70[illegible]E+05	2.11[illegible]
UPLM	1.6562E+05	[illegible]
UEF	3.7654E+05	[illegible]

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	[illegible]	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	7.73	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.

[redacted] 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

[redacted] 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	6.102E-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

[redacted] 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	0.0701	^{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	4.04[illegible]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 [redacted]. 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 term 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, [redacted]. 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 [redacted] 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, [redacted] the dose rate results from three-dimensional analysis are reported.

[redacted]

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. [redacted]

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 [redacted] 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 [redacted] 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 [redacted] computed to be 25.1 mrem/hr [redacted], 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.

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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 (5.9%) and 194.7 (0.3 %) mrem/hr, respectively, and the peak and average dose rates at 1 meter are 56.7 (5.6%) 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

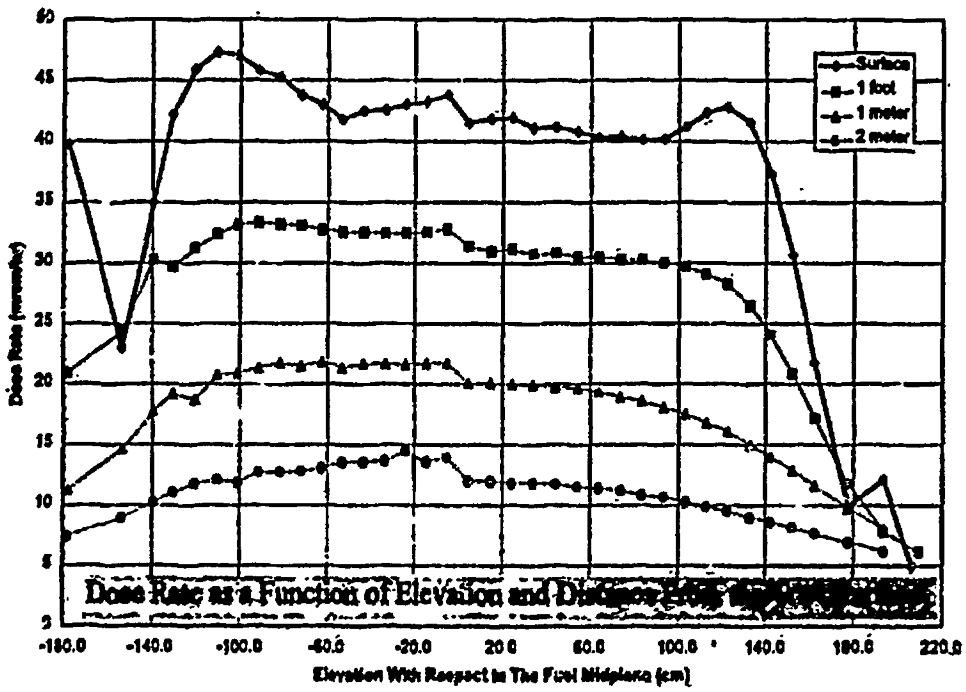


Figure 5.4-2 Storage Cask Three-Dimensional Model Surface Dose Rate Profile

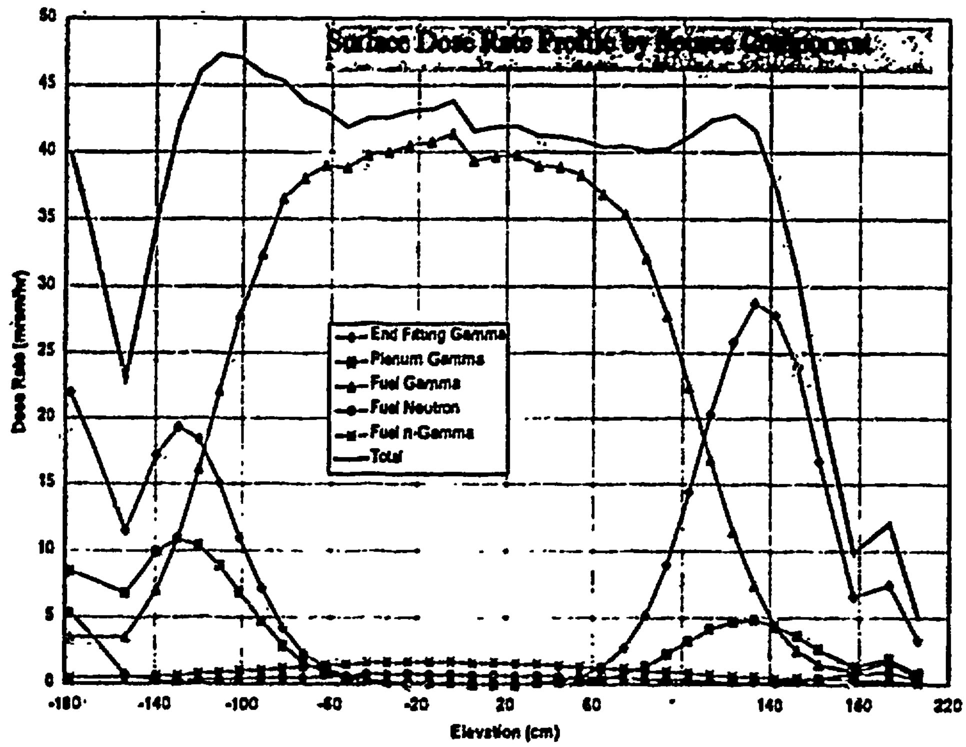


Figure 5.4-3 Storage Cask Top Outlet Vent Dose Rate Profile

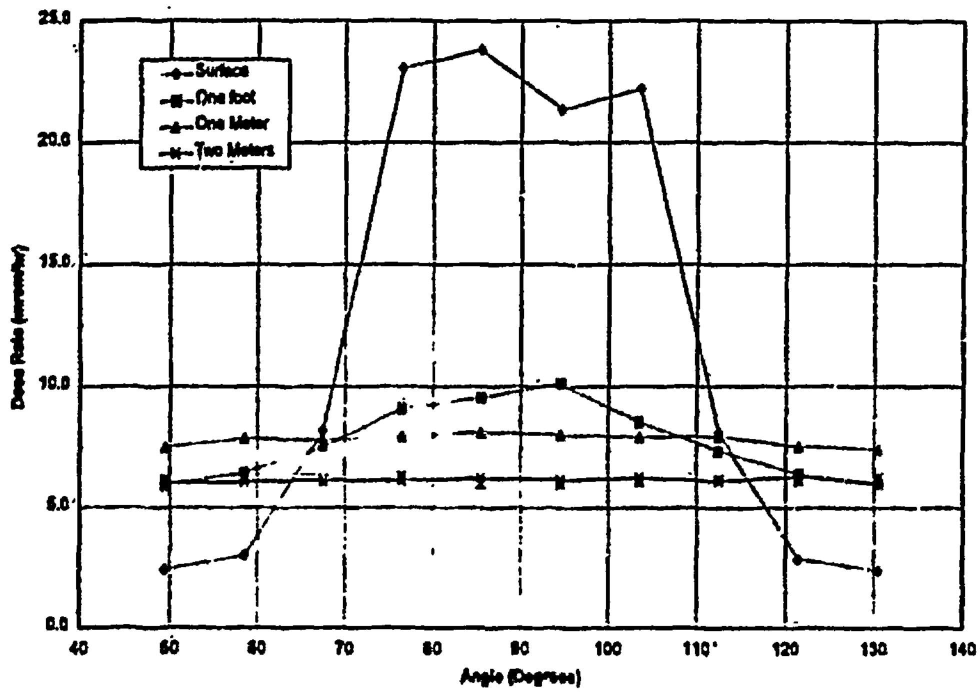


Figure 5.4-4 Storage Cask Bottom Inlet Vent Dose Rate Profile

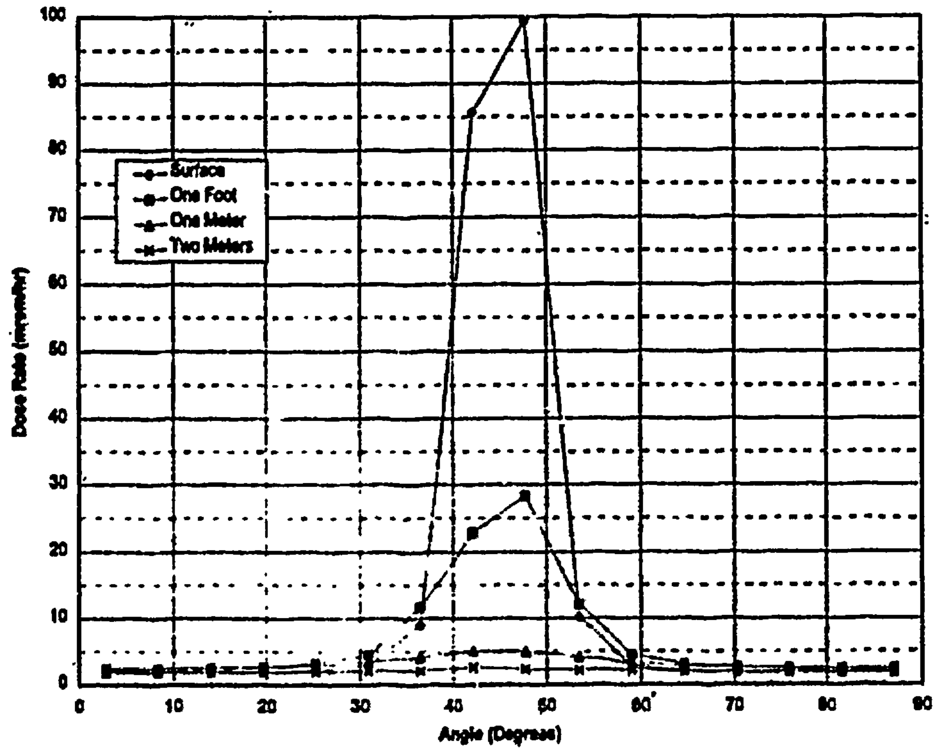


Figure 5.4-5 Storage Cask Dose Rate Profile at Stand Cutout Elevation

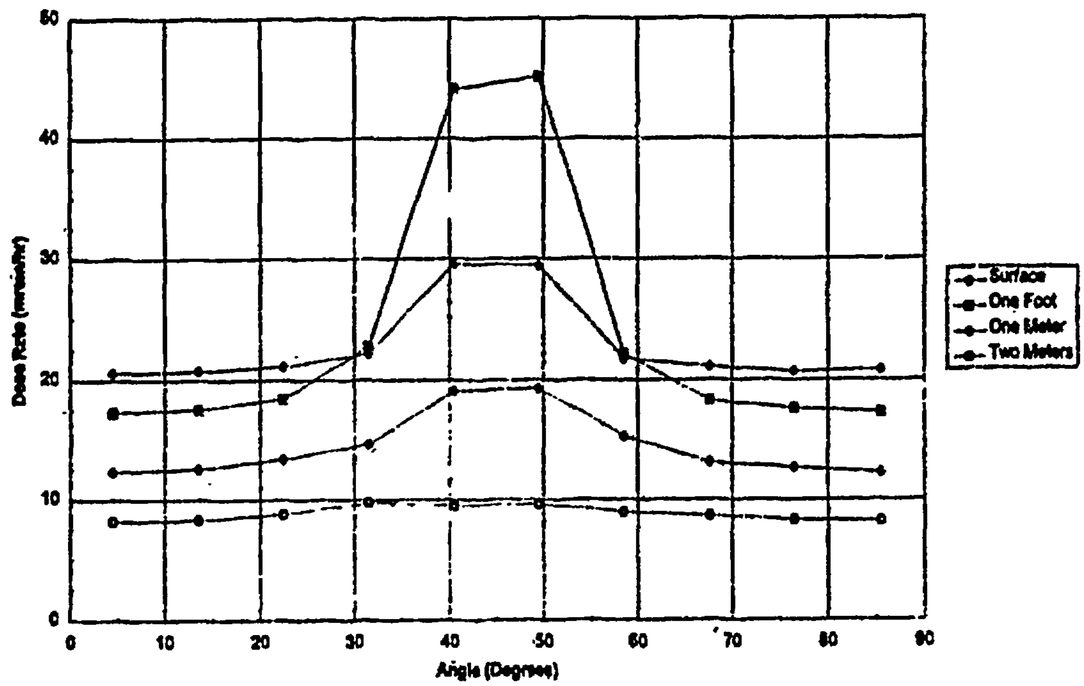


Figure 5.4-6 Storage Cask Top Dose Rate Profile as a Function of Radius from the Centerline and Distance from Surface

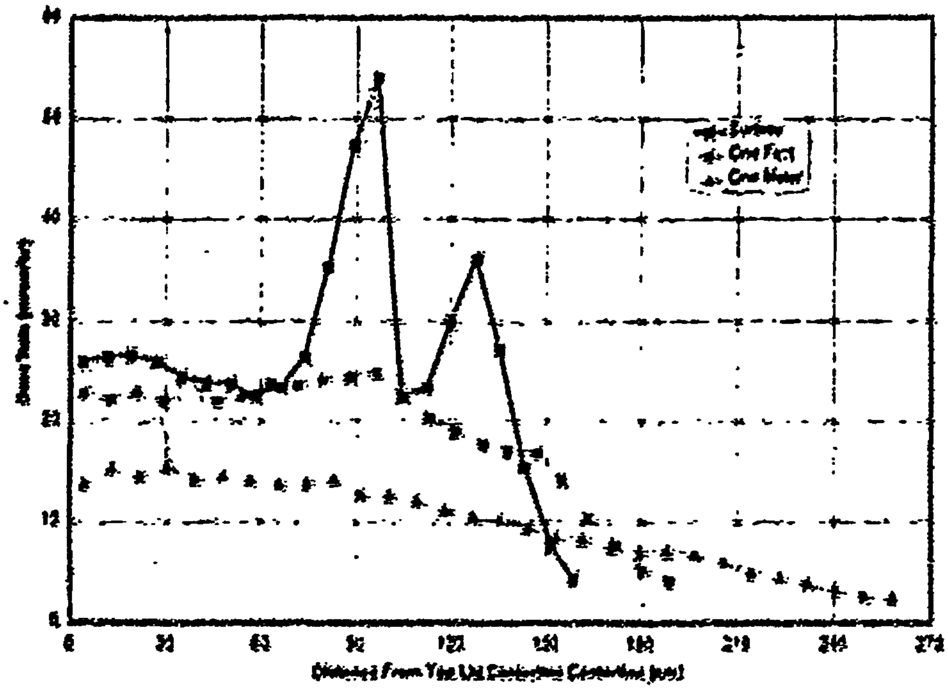


Figure 5.4-7 Storage Cask Top Surface Dose Rate Profile by Source Component

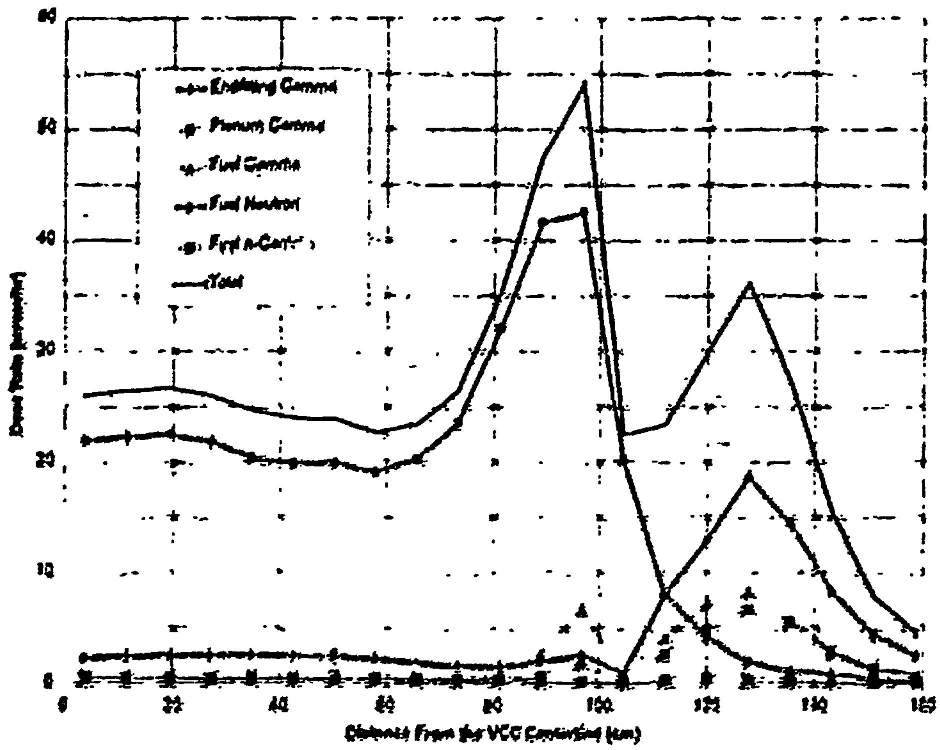


Figure 5.4-8 Transfer Cask Side Dose Rate Profile as a Function of Elevation and Distance, Wet Cavity

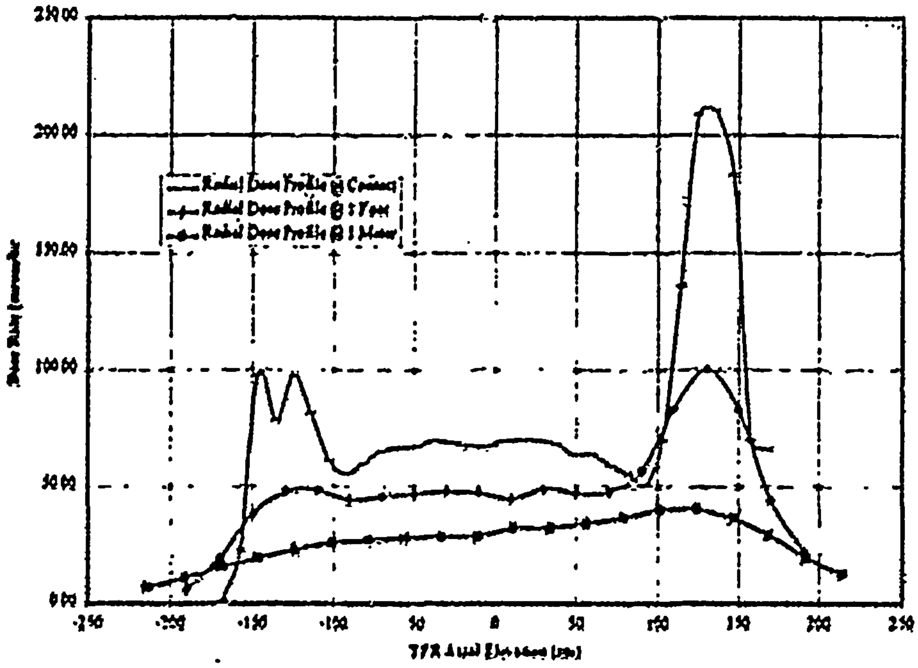
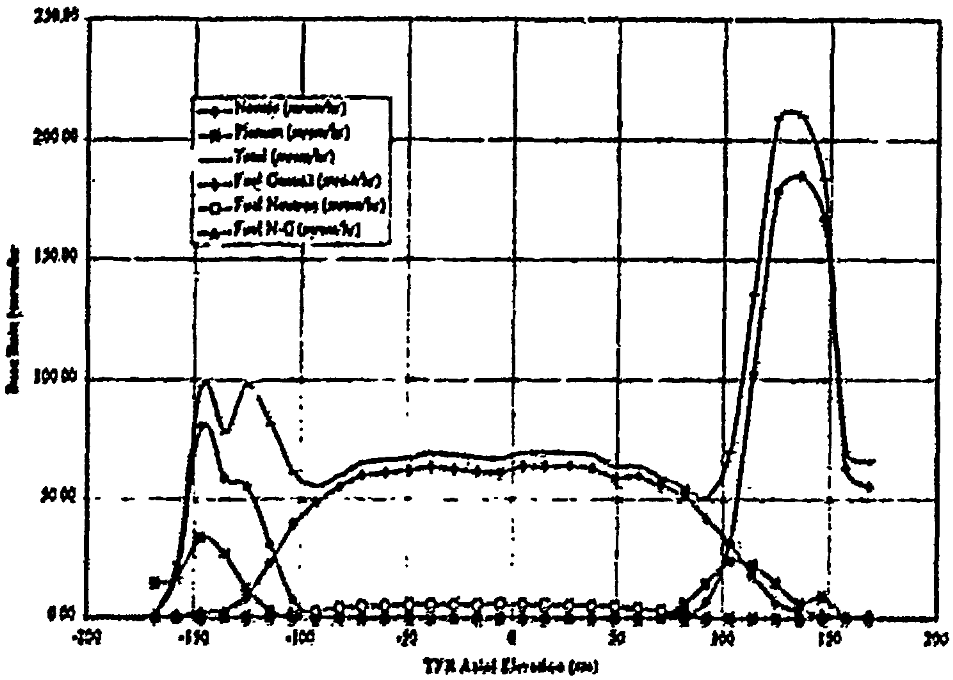


Figure 5.4-9 Transfer Cask Side Dose Rate Profile by Source Component, Wet Cavity



Note: Peak in nozzle dose at 133 cm is due to draining 56 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

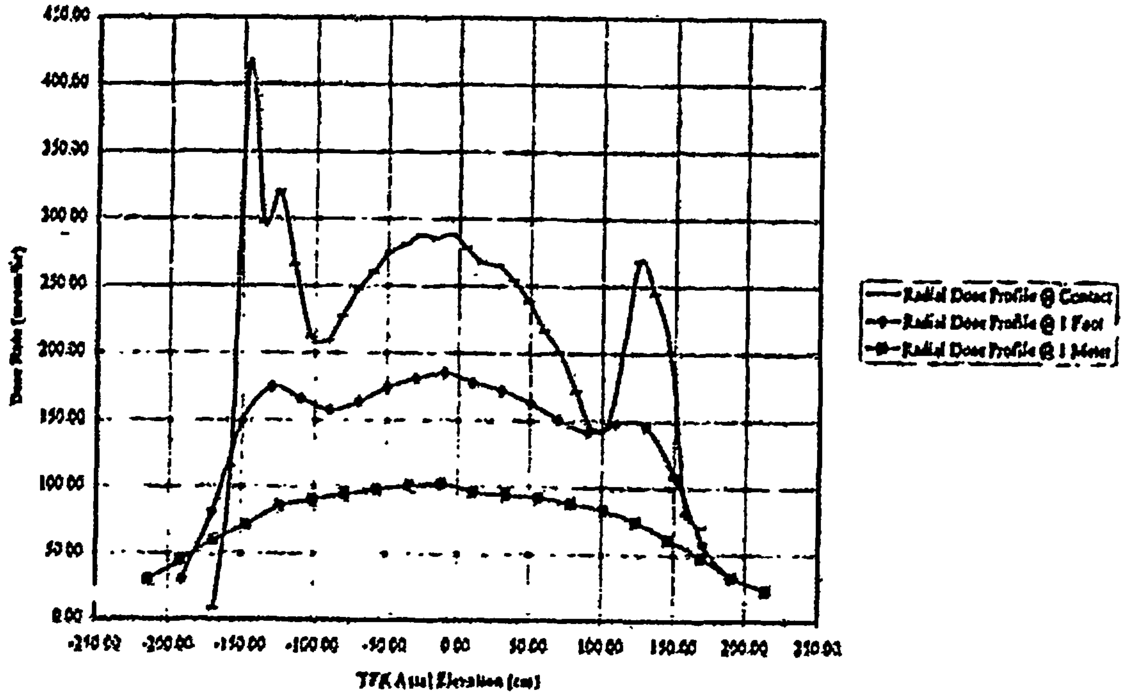


Figure 5.4-11 Transfer Cask Side Dose Rate Profile by Source Component, Dry Cavity

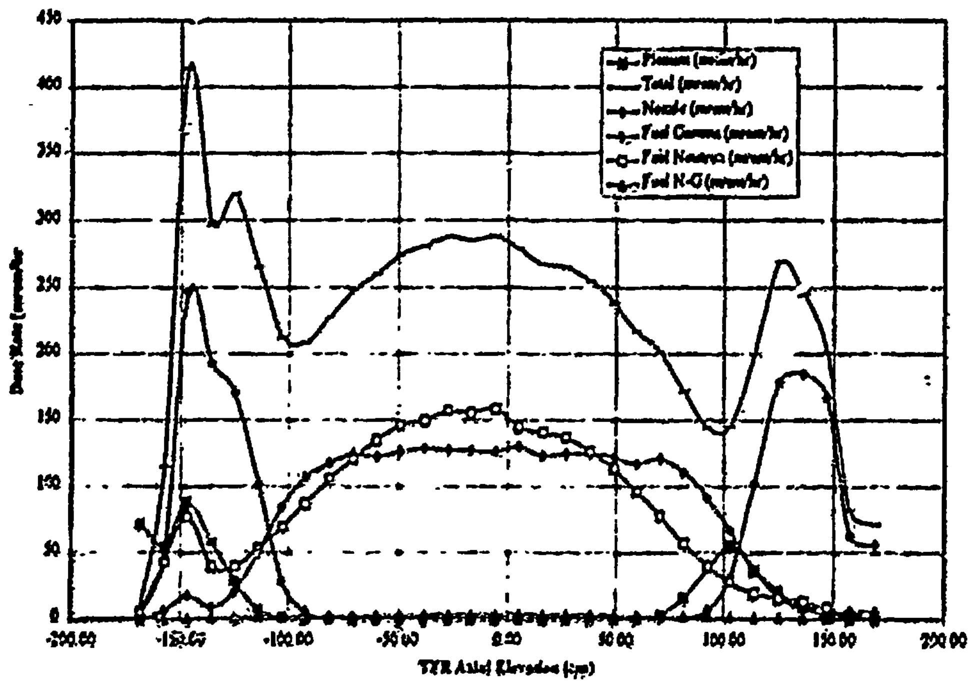


Figure 5.4-12 Transfer Cask Top Surface Dose Rate as a Function of Radius and Distance from Surface, Shield Lid, Structural Lid, and Temporary Shield On, Dry Cavity

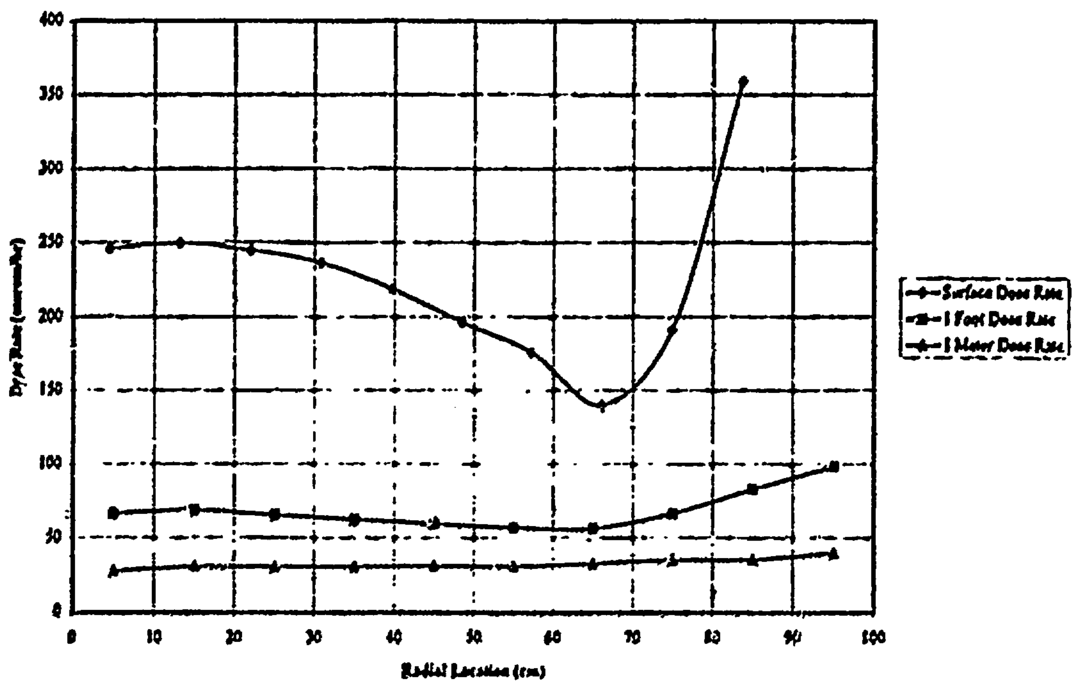


Figure 5.4-13 Transfer Cask Top Surface Dose Rate by Source Component, Shield Lid, Structural Lid, and Temporary Shield On, Dry Cavity

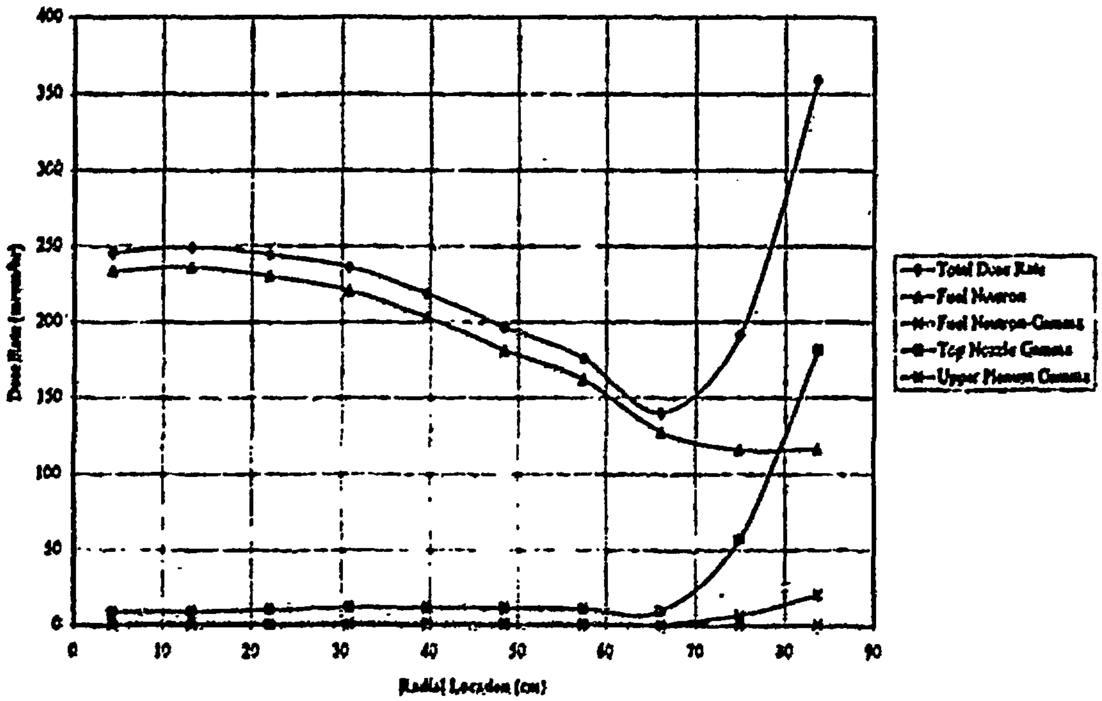


Figure 5.4-14 Transfer Cask Bottom Surface Dose Rate, Wet Cavity

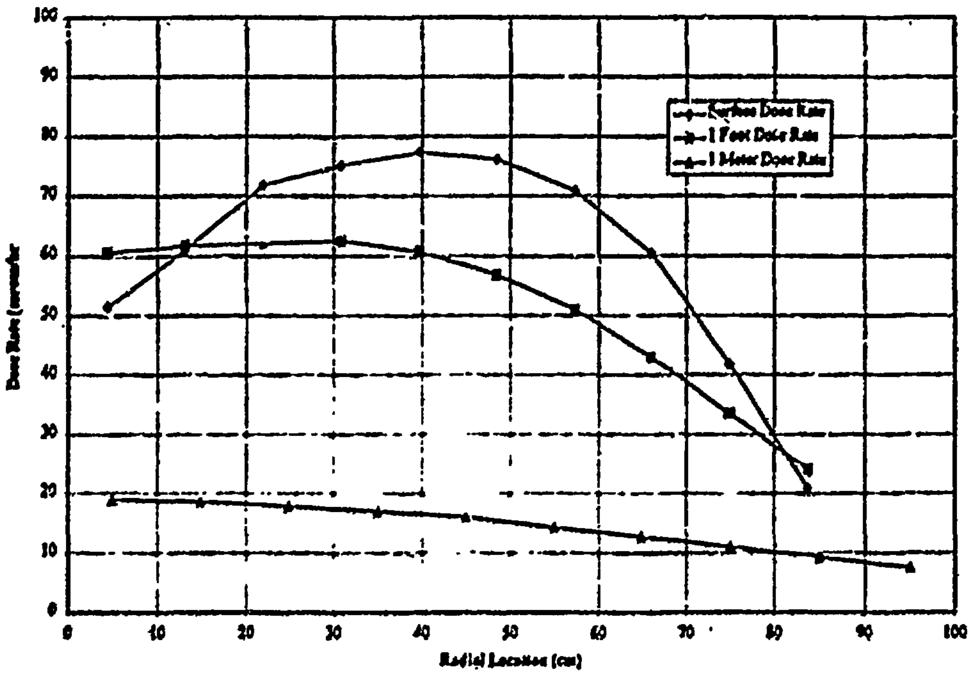


Figure 5.4-15 Transfer Cask Bottom Surface Dose Rate, Dry Cavity

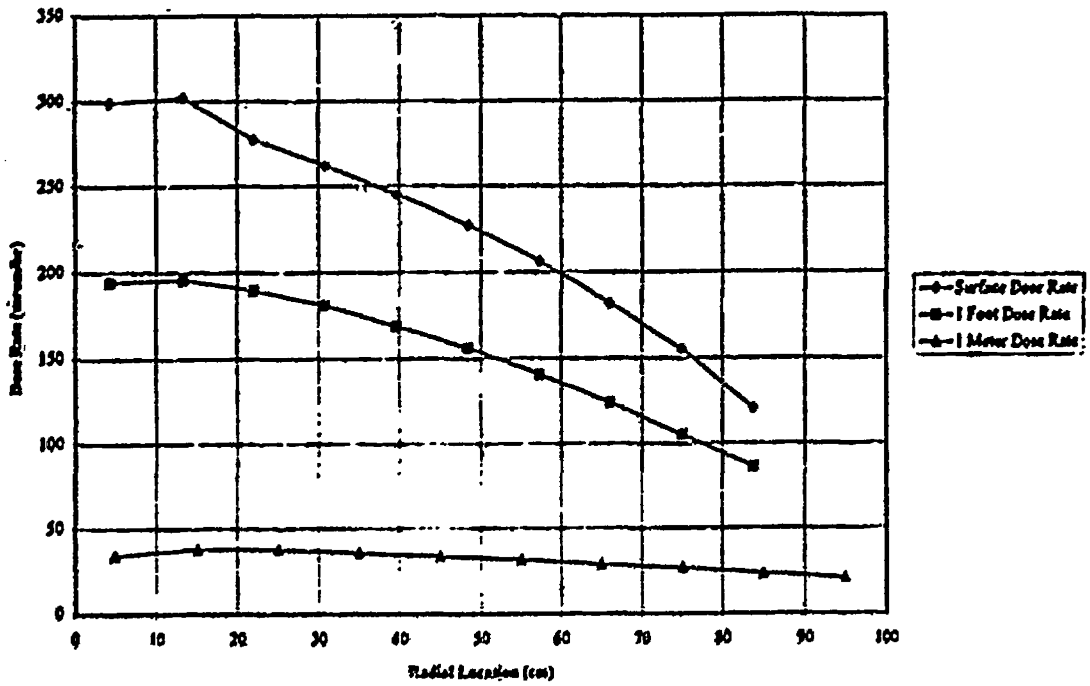


Table 5.4-1 ANSI Standard Neutron Flux-To-Dose Rate Factors

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)/(γ/Cm²/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-[illegible] 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/cm²)	Total Axial Neutron Group Flux (n/sec/cm²)
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 cm².
2. Total axial source is total flux multiplied by cask axial surface area of 8.301 x 10^4 cm².

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6.0 CRITICALITY EVALUATION

6.1 Discussion and Results

This chapter demonstrates that the NAC-MPC storage system containing 36 Yankee Class fuel assemblies is subcritical in accordance with the requirements of 10 CFR 72.124(a), 10 CFR 72.236(c) and Chapter 6 of NUREG-1536. These requirements are interpreted to mean that the effective neutron multiplication factor of the NAC-MPC system is less than 0.95, including biases and uncertainties under normal, off-normal and accident conditions.

The NAC-MPC storage system comprises a transportable storage canister (canister), a transfer cask and a vertical concrete cask (storage cask). The canister comprises a stainless steel canister and a basket. The basket comprises 36 fuel tubes held in place with stainless steel support disks and tie rods. The transfer cask containing the canister and basket is loaded underwater in the spent fuel pool. Once loaded with fuel, the canister is drained, dried, inerted, and welded shut. The transfer cask is then used to transfer the canister to the storage cask where it is stored until transported off site.

Under normal conditions, such as loading in a spent fuel pool, moderator (water) is present in the canister while it is in the transfer cask. Also, during draining and drying operations, moderator is present and its density will vary. Thus, the criticality evaluation of the transfer cask includes a variation in moderator density and a determination of optimum moderator density. Off-normal and accident conditions are bounded by assuming the most reactive mechanical basket configuration, as well as moderator intrusion into the fuel cladding (100% fuel failure).

Under normal conditions, moderator is not present in the canister while it is in the storage cask. However, access to the environment is possible via the air inlets in the storage cask and the convective heat transfer annulus between the canister and the storage cask steel liner. This access provides paths for moderator intrusion during a flood. Under off-normal conditions, moderator intrusion into the convective heat transfer annulus is evaluated. Under accident conditions, it is hypothetically assumed that the canister confinement fails, and moderator intrusion into the canister and into the fuel cladding (100% fuel failure) is evaluated. This is a highly conservative assumption, since, as shown in Chapters 3 and 11, there are no design basis normal, off-normal or accident conditions that result in the failure of the canister confinement boundary that would allow the intrusion of water.

Criticality control in the NAC-MPC canister basket is achieved using a flux trap principle. Each of the basket tubes in the canister basket is surrounded by four BORAL sheets with a core areal density of 0.01g $^{10}B/cm^2$ (minimum), which are held in place by stainless steel cladding. The spacing of the basket tubes is maintained by the stainless steel support disks. These disks provide water gap spacings between tubes of 0.875, 0.810, or 0.750 inches, depending on the position of the fuel tube in the basket. When the canister is flooded with water, fast neutrons leaking from the fuel assemblies are thermalized in the water gaps and are absorbed in the BORAL sheets before causing a fission in an adjacent fuel assembly. This criticality control can accommodate up to 36 Yankee Class Zircaloy-clad assemblies with an initial enrichment of 4.0 wt % ^{235}U or 36 Yankee Class stainless steel-clad assemblies with an initial enrichment of 4.94 wt % ^{235}U.

The criticality evaluation of the NAC-MPC is performed with the SCALE 4.3 (ORNL) Criticality Safety Analysis Sequence (CSAS)(Landers). This sequence includes KENO-Va (Petrie) Monte Carlo analysis to determine the effective neutron multiplication factor (k_{eff}). The 27 group ENDF/B-IV neutron library (Jordan) is used in all calculations. CSAS with the 27 group library is benchmarked by comparison to 63 critical experiments relevant to Light Water Reactor (LWR) fuel in storage and transport casks.

Criticality evaluations are performed for both the transfer and storage casks under normal, off-normal and accident conditions. Considerations are given to the most reactive fuel assembly type, worst case mechanical basket configuration and variations in moderator density. The maximum effective neutron multiplication factor with bias and uncertainties for the transfer cask is 0.9021 under either normal, off-normal or accident conditions. The maximum multiplication factor with bias and uncertainties for the storage cask is 0.4503 under normal dry storage conditions and 0.9018 under the hypothetical accident conditions involving full moderator intrusion. These values reflect the following conservative conditions:

1. No fuel burnup (fresh fuel assumption).
2. No fission product build up as a poison,
3. 36 Yankee Class fuel assemblies of the most reactive type.
4. UO_2 fuel density at 95% of theoretical.
5. No dissolved boron in the spent fuel pool water (water temperature 293°K).
6. 75% of [redacted] ^{10}B loading in the BORAL plates.
7. Infinite cask array.
8. No axial leakage.
9. A most reactive mechanical configuration involving: [redacted] the fuel tubes [redacted] moved toward the center of the basket, maximum fuel tube opening, minimum disk opening, maximum disk thickness and closely packed disk openings.
10. Moderator intrusion into the fuel rod clad/pellet gap under accident conditions.

Analysis of simultaneous moderator density variation inside and outside either the transfer or storage casks shows a monotonic decrease in reactivity with decreasing moderator density. Thus, the full moderator density condition bounds any off-normal or accident situation. Analysis of moderator intrusion into the storage cask heat transfer annulus with the dry canister shows a slight decrease in reactivity from the completely dry condition.

The NAC-MPC storage system containing 36 Yankee Class fuel assemblies of the most reactive type in the most reactive configuration is well below the 0.95 [redacted] criticality safety limit, including all biases and uncertainties under normal, off-normal and accident conditions.

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6.2 Package Fuel Loading

The NAC-MPC storage system can safely transfer and store 36 Yankee Class fuel assemblies. [redacted] Loading [redacted] is [redacted] by [redacted] the top weldment, [redacted] The various Yankee Class assemblies to be transferred and stored in the NAC-MPC are presented in Table 6.2-1. Five vendor categories, each with two fuel rod configurations (types), are available within the Yankee Class. These are: Combustion Engineering (CE) 16 x 16 Type A and Type B, two [redacted] Exxon 16 x 16 Type A and Type B, United Nuclear 16 x 16 Type A and Type B, and Westinghouse 18 x 18 stainless steel clad Type A and Type B. See Figures 6.2-1, 6.2-2 and 6.2-3 for the fuel array configurations. The Combustion Engineering manufactured fuel has the same fuel rod arrays for Types A and B as the Exxon fuel. The Type A and Type B fuel array configurations allow a cruciform control rod to be inserted between assemblies during core operation. The most reactive Yankee Class fuel assembly is the United Nuclear, Type A, 16 x 16 fuel assembly with 4.0 wt % ^{235}U initial enrichment. This fuel assembly type bounds all of the Yankee Class fuel assemblies, including the Westinghouse stainless steel clad fuel with a 4.94 wt % ^{235}U initial enrichment. The United Nuclear Type A fuel assembly with 4.0 wt % ^{235}U initial enrichment is the design basis fuel assembly used in NAC-MPC storage system criticality evaluations. To preclude any potential increase in reactivity due to empty fuel rod positions in the spent fuel assembly, any fuel rods removed from the assembly lattice must be replaced with solid filler rods fabricated from Zircaloy or Type 304 stainless steel.

A canister may contain one or more reconfigured fuel assemblies. The reconfigured fuel assembly is designed to confine the Yankee Class spent fuel rods, or portions thereof, which are classified as failed fuel and to maintain the geometric configuration of those fuel rods. This assembly can accept up to 64 full length spent fuel rods in an eight by eight array of tubes.

The reconfigured fuel assembly consists of a shell (square tube with end fittings), a basket assembly and 64 fuel tubes. Reconfigured fuel assembly parameters are presented in Table 6.2-2. The external dimensions of the shell are the same as those of a standard Yankee Class 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 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

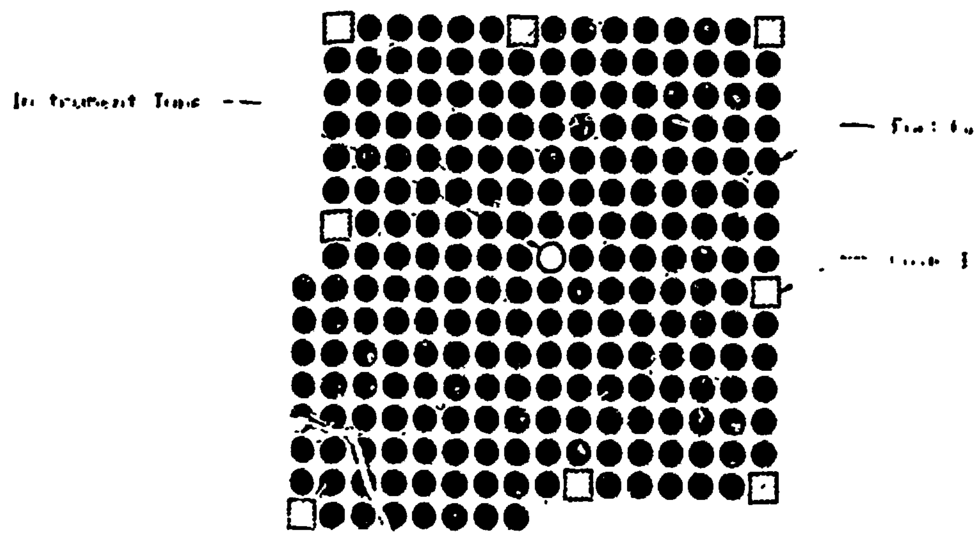
with drilled holes to permit draining, drying and inerting with helium. The shell has holes in the top and bottom fittings to permit draining, drying and inerting of the assembly.

The total number of full length pins 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 256 rods). Consequently, the reactivity of the reconfigured fuel assembly, even with the most reactive fuel rods, is less than the design basis fuel assembly used in criticality evaluations.

[redacted]

Figure 6.2-1 Yankee Class Type A and Type B Exxon and CE Fuel Assembly Arrays

Type A



Type B

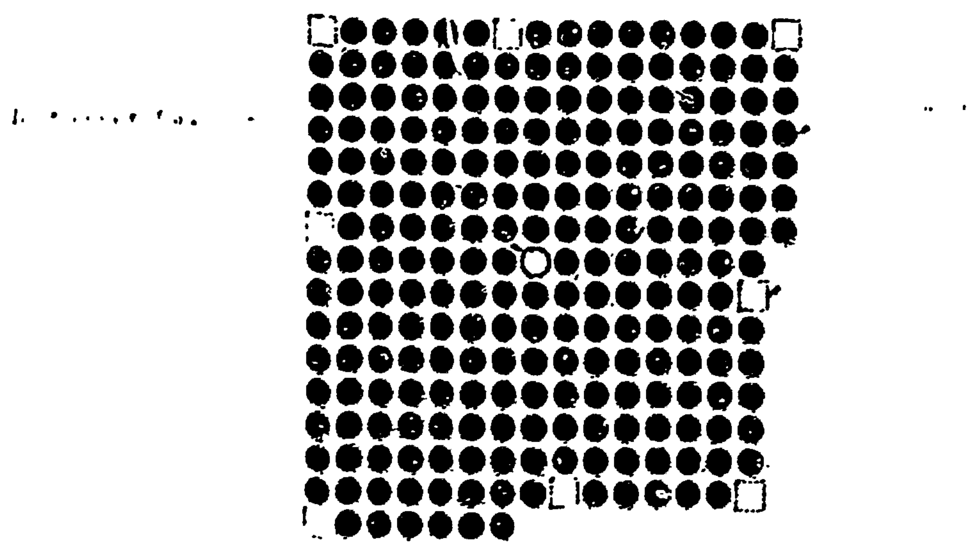
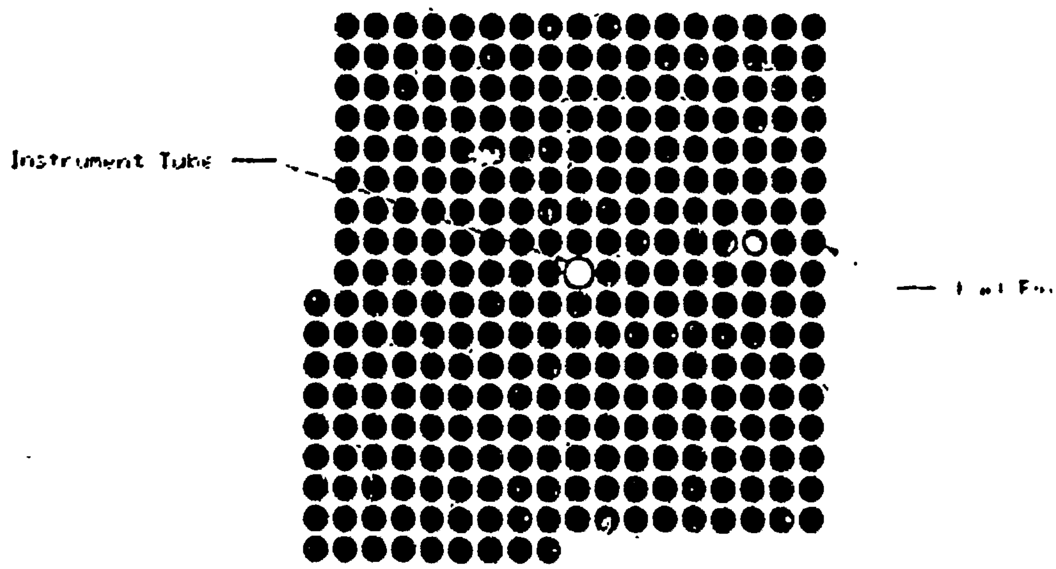


Figure 6.2-2 Yankee Class Type A and Type B Westinghouse Fuel Assembly Arrays

Type A



Type B

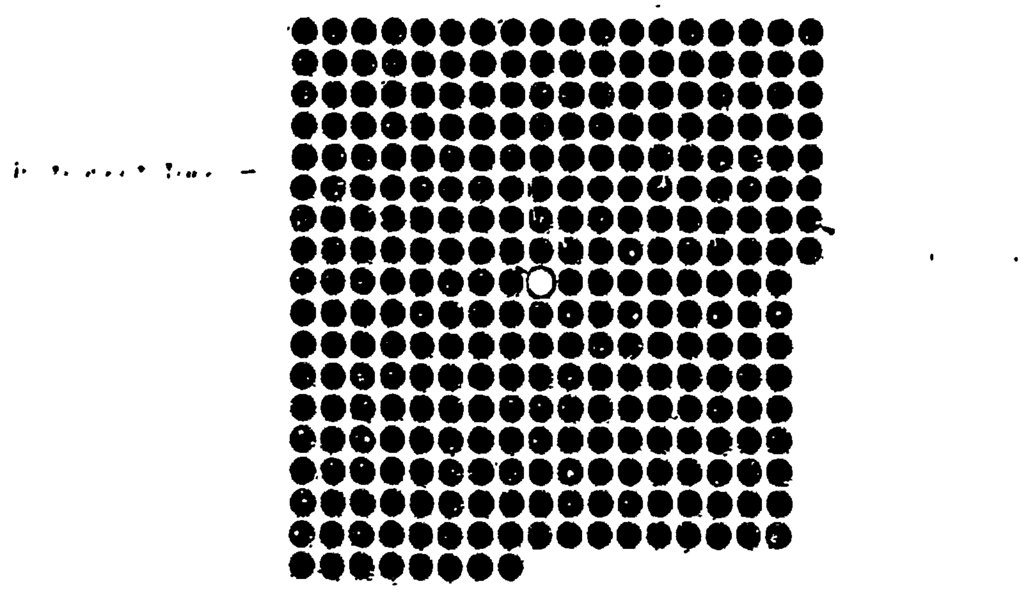
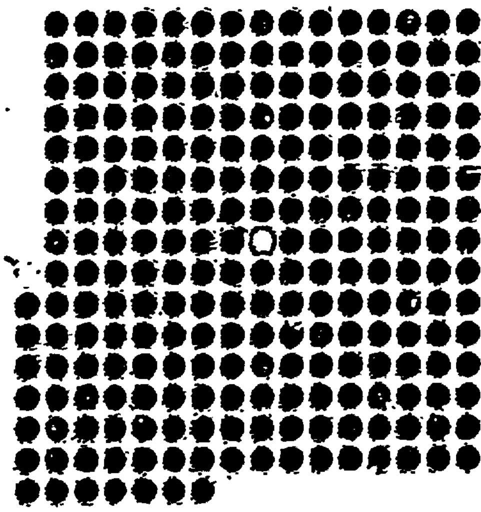


Figure 6.2-5 Yankee Class Type A and Type B United Nuclear Fuel Assembly Arrays

Type A



Type B

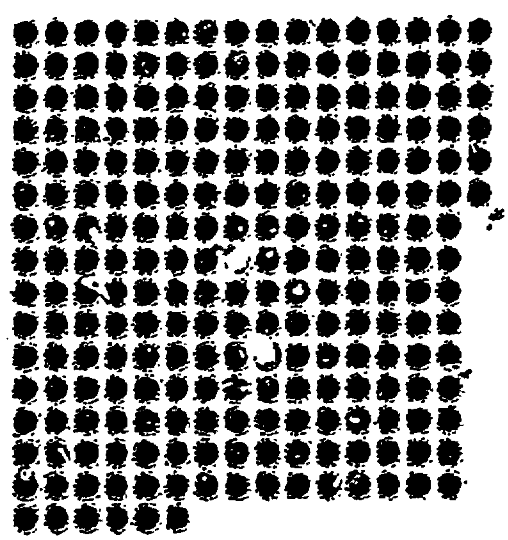


Table 6.2-1 Yankee Class Fuel Assembly Parameters

Parameter	CE Type A	CE Type B	Exxon Type A	Exxon Type B	Exxon Type [illegible]	Exxon Type [illegible]	West. Type A	West. Type B	United Nuclear Type A	United Nuclear Type B
Assembly Configuration	•	•	•	•	•	•	•	•	•	•
Assembly Array	16 x 16	16 x 16	16 x 16	16 x 16	16 x 16	16 x 16	[illegible]	[illegible]	[illegible]	[illegible]
Max. Enrichment (wt % ^{235}U)	3.90	3.93	4.00	4.00	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Max. MTU[a]	0.2394	[illegible]	0.2394	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Fuel Rod Configuration	•	•	•	•	•	•	•	•	•	•
Fuel Rod Pitch (cm)	1.1989	1.1989	1.1989	1.1989	1.1989	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Active Fuel Length (cm)	231.1400	231.1400	231.1400	231.1400	231.1400	231.1400	[illegible]	[illegible]	[illegible]	[illegible]
Rod OD (cm)	0.9271	0.9271	0.9271	0.9271	0.9271	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Clad ID (cm)	0.8052	[illegible]	0.8052	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Pellet OD (cm)	0.7887	0.7887	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Diametral Gap (cm)	0.0165	0.0165	0.0165	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Rods per Assembly	231	230	231	[illegible]	231	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Fuel Material	UO_2	UO_2	UO_2	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Clad Material	Zircaloy	Zircaloy	Zircaloy	[illegible]	Zircaloy	Zircaloy	[illegible]	[illegible]	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	[illegible]	[illegible]
Number Per Assembly	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	[illegible]	[illegible]
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)	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Instrument Tube OD (cm)	1.1481	1.1481	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
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

[a] Maximum MTU based on 95% of UO_2 theoretical density for the fuel pellet stack [illegible]

Table 6.2-2 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.

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6.3 Criticality Model Specification

This section describes the KENO-Va models used in the criticality evaluation of the NAC-MPC. The geometric representation of the design basis fuel assembly, the NAC-MPC basket and the associated transfer and storage cask shield regions are described. The material densities and nuclide concentrations are also specified.

6.3.1 Description of Calculational Models

Three models are used in the NAC-MPC storage system criticality evaluations: a fuel/basket model, a transfer cask model and a storage cask model. The fuel/basket model is a single NAC-MPC basket cell containing a fuel assembly. This model is used to determine the most reactive fuel assembly type and to evaluate mechanical perturbations of the fuel and basket. The transfer cask model is an explicit representation of the NAC-MPC basket and canister containing 36 design basis fuel assemblies surrounded by the transfer cask radial shields. This model is used to evaluate the transfer cask reactivity during loading, draining and drying operations. The storage cask model is an explicit representation of the NAC-MPC basket and canister containing up to 36 design basis fuel assemblies, surrounded by the storage cask radial shields. This model is used to evaluate the storage cask during normal, off-normal and accident storage situations.

The fuel/basket model comprises a single basket cell with periodic axial and reflective radial boundary conditions. This simulates an infinite array of fuel/basket cells. The model geometry is divided into four vertical layers an aluminum heat transfer layer; a flux trap water layer, a steel support disk layer; and a top flux trap water layer. In each of these layers, the fuel assembly array, fuel tube, BORAL sheets, flux-trap water gap, steel, or aluminum disk web are modeled. Figure 6.3-1 shows a sketch of the NAC-MPC fuel/basket model.

Both of the storage and transfer cask criticality models are derived from a radial slice of the basket at the central (heat transfer) region. This section is the most reactive region, due to the number of steel support disks and aluminum heat transfer disks displacing water in the flux trap gap. Each model is a stack of four slices containing one aluminum disk, two identical water regions and one steel support disk region (stacked aluminum, water, steel, water). The basket is modeled in each slice and contains a design basis fuel assembly in each of 36 basket tubes with

BORAL sheets. Both cask models explicitly arrange the fuel assemblies and basket tubes in the most reactive configuration consistent with the mechanical tolerances specified by the design drawings. In all models, the fuel assemblies, basket tubes, BORAL sheets and water gaps are explicitly represented. There are no homogenizations. Each cask slice includes the cask radial shield regions surrounded by an outer CUBOID. The four slices are stacked into the KENO-Va GLOBAL UNIT.

Periodic boundary conditions are imposed on the top and bottom of the GLOBAL UNIT to simulate infinite axial extent. Reflecting boundary conditions are imposed on the sides of the GLOBAL UNIT simulating an infinite number of casks in the X-Y plane. Moderator density is varied both in the cask cavity regions normally filled with water and in the exterior CUBOID. Figures 6.3-2 and 6.3-3 show the transfer cask and storage cask criticality models, respectively.

6.3.2 Package Regional Densities

The SCALE 4.3 standard composition library (Petrie) default densities and isotopic distributions are used unless otherwise indicated. The densities used in the KENO-Va criticality analyses are.

Material	Density (g/cc)
UO_2 at 95% TD	0.412
Zircaloy	6.56
Type 348 Stainless Steel	7.92 (non-standard, Westinghouse fuel clad)
H_2O	0.9982
Type 304 Stainless Steel	7.92
Lead	11.344
Aluminum	2.702
BORAL (core)	2.623 (non-standard)
NS-4-FR	1.63 ~~(non-standard)~~
Concrete	2.243 (based on 140 lb/ft^3 design spec. minimum)
Carbon Steel	7.821

6.3.2.1 Fuel Region

Fuel rod densities are:

Material	Element	Density (atoms/barn-cm)
UO_2 (at 3.9 wt %)	^{235}U	9.271×10^{-4}
	^{238}U	2.231×10^{-2}
	O	4.646×10^{-2}
UO_2 (at 4.0 wt %)	^{235}U	9.406×10^{-4}
	^{238}U	2.229×10^{-2}
	O	4.646×10^{-2}
UO_2 (at 4.9 wt %)	^{235}U	1.162×10^{-4}
	^{238}U	2.207×10^{-2}
	O	4.646×10^{-2}
Zircaloy	Zr	4.331×10^{-2}
Stainless Steel 348	Fe	5.529×10^{-2}
	Cr	1.743×10^{-2}
	Ni	1.057×10^{-2}
	C	3.180×10^{-4}
	Mn	1.736×10^{-3}
	Si	1.698×10^{-3}
	P	6.159×10^{-5}
	S	4.463×10^{-5}

6.3.2.2 Cask Material

Cask material densities are:

Material	Element	Density (atoms/barn-cm)
Boral Core	^{10}B	7.098×10^{-3}
	^{11}B	3.925×10^{-2}
	C	1.220×10^{-2}
	Al	3.358×10^{-2}
Aluminum	Al	6.03×10^{-2}
Stainless Steel 304	Cr	1.743×10^{-2}
	Fe	5.936×10^{-2}
	Ni	7.721×10^{-3}
	Mn	1.736×10^{-3}
Lead	Pb	3.297×10^{-2}
NS-4-FR	H	5.841×10^{-2}
	O	2.607×10^{-2}
	C	2.265×10^{-2}
	N	1.401×10^{-3}
	Al	7.781×10^{-3}
	^{11}B	3.565×10^{-4}
	^{10}B	9.798×10^{-5}
Concrete	O	4.494×10^{-2}
	Si	1.621×10^{-2}
	H	1.340×10^{-2}
	Na	1.704×10^{-3}
	Ca	1.483×10^{-3}
	Fe	3.386×10^{-4}
	Al	1.702×10^{-3}
Carbon Steel	Fe	8.350×10^{-2}
	C	3.925×10^{-3}

6.3.2.3 Water Reflector Densities

The material densities for the water inside and outside the storage cask under normal conditions are:

Material	Element	Density (atoms/barn-cm)
H_2O	H	6.677×10^{-2}
	O	3.338×10^{-2}

Water density is varied using the volume fraction (VF) parameter on the SCALE 4.3 material information processor card. This acts as a simple multiplier on the above densities.

Figure 6.3-1 NAC-MPC KENO-Va Fuel/Basket Model

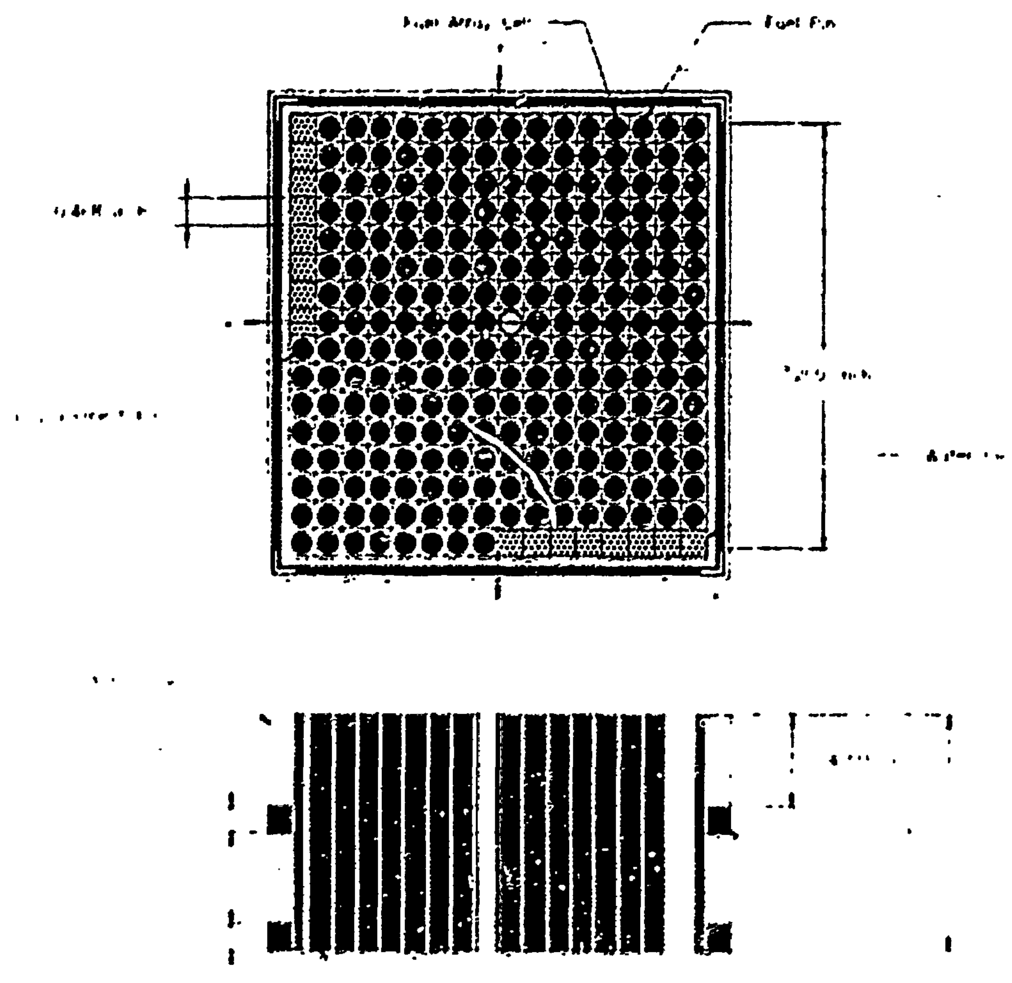
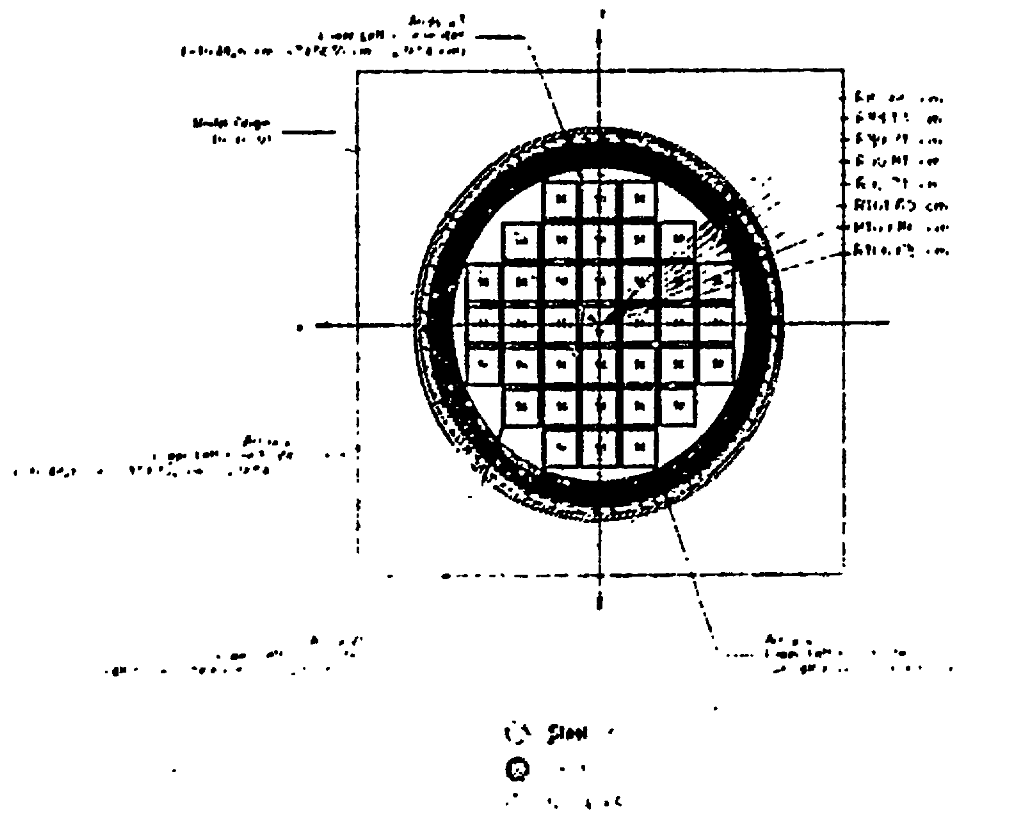
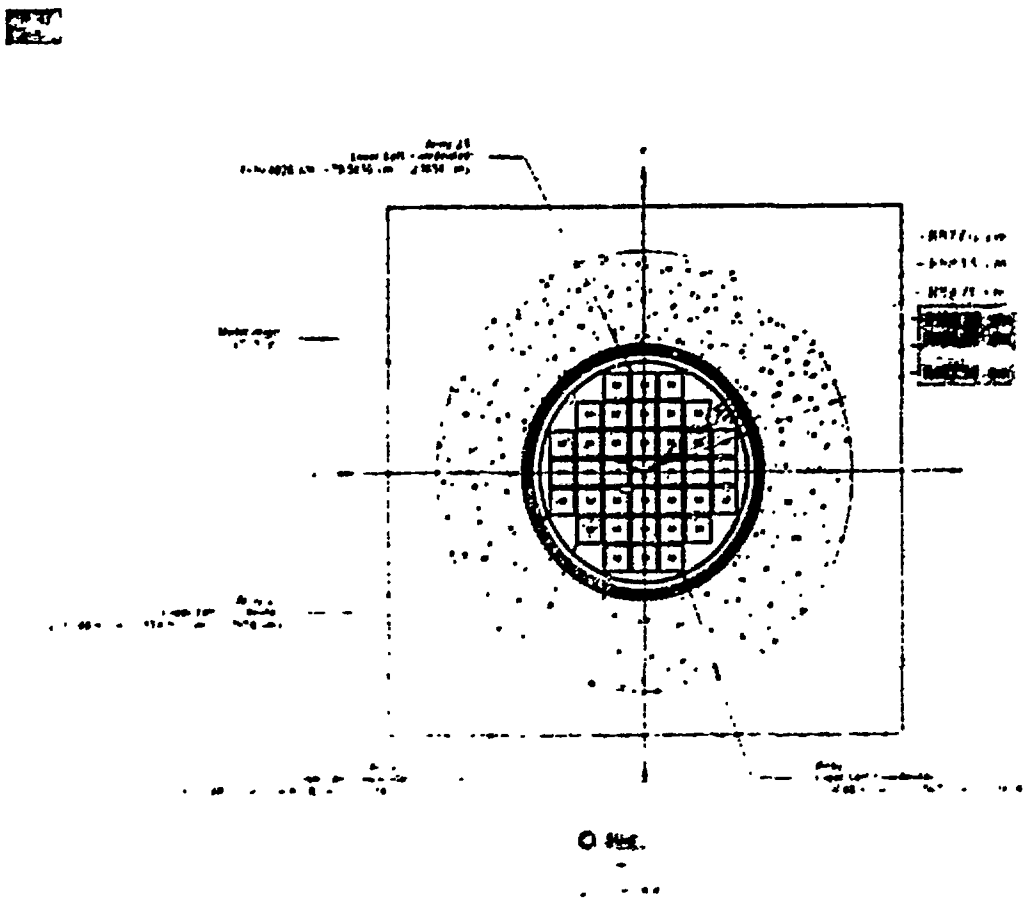


Figure 6.3-2 NAC-MPC KENO-Va Transfer Cask Model



Note: This model represents a slice between a support disk and a heat transfer disk. The center fuel tube position is filled with unit 51. This unit cell describes the vacant fuel tube . As previously stated, the criticality analysis considers 36 Yankee Class fuel assemblies, assuming the center tube is empty.

Figure 6.3-3 NAC-MPC KENO-Va Storage Cask Model



Note: This model represents a slice between a support disk and a heat transfer disk. The center fuel tube position is filled with unit 51. This unit cell describes the vacant fuel tube. As previously stated, the criticality analysis considers 36 Yankee Class fuel assemblies, assuming the center tube is empty.

6.4 Criticality Calculation

This section demonstrates that the criticality analysis of the NAC-MPC is sufficient to satisfy the requirements of 10 CFR 72.124(a), 10 CFR 72.236(c) and Chapter 6 of NUREG 1536. The calculational method is described. Criticality calculations are performed to determine: the most reactive Yankee Class fuel assembly type; the most reactive mechanical configuration in the NAC-MPC basket; and the most reactive moderator density under normal, off-normal and accident conditions.

6.4.1 Calculational Method

The criticality evaluation of the NAC-MPC is performed with the SCALE 4.3 (ORNL) Criticality Safety Analysis Sequence (CSAS) (Landers) for the PC. CSAS includes: the SCALE Material Information Processor (Bucholz), BONAMI (Greene), NITAWL-II (Westfall), and KENO-Va (Petrie). The Material Information Processor generates number densities for standard compositions, prepares geometry data for resonance self-shielding, and creates data input files for the cross section processing codes. The BONAMI and NITAWL-II codes are used to prepare a resonance-corrected cross section library in AMPX working format. The KENO-Va code calculates the model k_{eff} using Monte Carlo techniques. The 27 group ENDF/B-IV neutron cross section library (Jordan) is used in this calculation. The NAC-MPC KENO-Va models are described in further detail below.

6.4.2 Fuel Loading Optimization

The fuel loading is optimized in the NAC-MPC cask criticality models by using. 1) fresh fuel, 2) the most reactive Yankee Class fuel assembly type, 3) the highest possible fuel stack density (95 % of theoretical) and 4) the most reactive basket configuration. The cask models represent fully loaded baskets with 36 design basis fuel assemblies. The models use reflecting boundary conditions on the sides and periodic boundary conditions on the top and bottom. These boundary conditions simulate an infinite array of casks of infinite axial extent.

6.4.3 Criticality Results

This section establishes the most reactive Yankee Class fuel and the most reactive configuration of the fuel within the canister basket. These results are used to calculate the effective neutron multiplication factor for the transfer cask and storage cask assuming full moderation.

6.4.3.1 Most Reactive Assembly

Using the fuel/basket model of the NAC-MPC basket, each of the Yankee Class fuel assembly vendor categories shown in Table 6.2-1 is evaluated. Each particular fuel rod array is explicitly modeled. This includes the Westinghouse 18 x 18 Types A and B at 4.94 wt % ^{235}U, United Nuclear 16 x 16 Types A and B at 4.0 wt % ^{235}U, Exxon 16 x 16 Types A and B with steel guide bars and instrument tube at 4.0 wt % ^{235}U, and CE Type A and B at 3.90 wt % ^{235}U, as well as the reconfigured fuel assembly with the most reactive fuel rods. Note, the Exxon 16 x 16 Types A and B with Zircaloy guide bars and instrument tube is identical to the CE configuration. In order to standardize the comparison, each assembly is evaluated with the fuel UO_2 at 95% theoretical density.

Table 6.4-1 shows the multiplication factor for each Yankee Class fuel type in the NAC-MPC basket. This table shows that either the United Nuclear Type A or Type B has the highest multiplication factor of the Yankee Class fuel assembly vendor categories. Table 6.4-1 also shows that it is difficult to resolve the difference between Type A and Type B fuel assemblies. There is only one fuel rod difference in loading. However, since the United Nuclear Type A has the highest UO_2 mass, this fuel rod array is chosen as the most reactive design basis fuel assembly for the NAC-MPC This design basis fuel assembly is used in all subsequent transfer and storage cask evaluations.

6.4.3.2 Most Reactive Mechanical Configuration

Using the fuel/basket model with the design basis fuel assembly, an evaluation of the effect of different NAC-MPC basket perturbations is made This criticality analysis determines the most reactive basket mechanical configuration by altering the nominal fuel/basket model with the design basis assembly and comparing the perturbed k_{eff} to the nominal result. If Δk_{eff} ($k_{perturbed}$ - $k_{nominal}$) is positive, the tolerance causes an increase in reactivity. Conversely, if Δk_{eff} is negative, the tolerance causes a decrease in reactivity. Two sets of perturbations are assessed in this

evaluation of the criticality control: fabrication tolerances and component movement within the basket.

Four major fabrication tolerances are evaluated: the fuel tube opening, the disk opening, the disk thickness and the disk opening placement. Modifications to the nominal fuel/basket model dimensions are made based on the basket and fuel tube tolerances given on the NAC-MPC drawings provided in Chapter 1.0. The tolerance analysis results are shown in Table 6.4-2. Table 6.4-2 shows that the most reactive set of basket tolerances are maximum fuel tube opening, minimum disk opening, maximum disk thickness and minimum (close packed) disk opening placement.

Increasing the fuel tube opening brings more moderator into the gap between the assembly and the tube lowering the efficiency of the BORAL sheets, hence increasing the reactivity of the system. Minimizing the disk opening and maximizing the disk thickness removes water from the flux trap, consequently increasing k_{eff}. Finally, decreasing the web thickness, decreases the flux trap size and also moves assemblies closer together producing an increase in k_{eff}. With respect to fabrication tolerances, this is the most reactive configuration.

Two major component movements within the basket are evaluated: the assembly within the tube and the tube within the basket. Unique to this package is the Yankee Class diagonally symmetric fuel assembly. Consequently, movement toward the three corners must be evaluated as opposed to one corner for a fully symmetric assembly. This assembly produces five movement perturbations: fuel tube movement to the upper right corner, the upper left corner, the lower left corner and side to side. Table 6.4-3 shows the assembly movement analysis results. These results show that the most reactive assembly position is centered within the basket tube. This centering provides the most optimum moderating water gap within the tube.

Similar to the fuel assembly movement analysis, five possible fuel tube movements are evaluated: the upper right corne , the upper left corner, the lower left corner and side to side. Mirror and periodic boundary conditions on the sides of the model are evaluated. Table 6.4-4 shows the tube movement evaluations. These results indicate that the most reactive fuel tube location is shifted to the right side of the tube with mirrored boundary conditions. This result is reasonable given the orientation of the assembly. Shifting the tube to the right side with mirrored boundary conditions moves a complete fuel pin row of two assemblies closer together, hence pushing the largest amount of fuel together and minimizing the flux trap gap between tubes. In

general, these results show that moving the tubes towards each other with the fuel assembly centered in the tube is the most reactive component configuration.

Thus, the following most reactive mechanical configuration is imposed on the NAC-MPC basket model: assemblies centered in the tubes, fuel tubes moved toward the center of the basket, maximum fuel tube opening, minimum disk opening, maximum disk thickness and close packed disk opening locations. The reactivity penalty associated with this configuration versus the nominal configuration is discussed in the transfer cask and storage cask criticality evaluations below.

[redacted]

Position	[redacted]	[redacted]	[redacted]
Nominal	[redacted]	[redacted]	[redacted]
Tubes Moved Toward the Basket Center	[redacted]	[redacted]	[redacted]
Tubes Moved Toward the Basket Shell	[redacted]	[redacted]	[redacted]
Assemblies Moved Toward the Basket Center	[redacted]	[redacted]	[redacted]
Assemblies Moved Toward the Basket Shell	[redacted]	[redacted]	[redacted]

Based on the cask analysis, the assembly moved towards [redacted] Δk_s of 0.004 to the reactivity of the nominal configur [redacted] case mechanical configuration, in the fuel/basket [redacted] moderator studies, is not adjusted from its assembly centered [redacted] with the assembly movement is accounted for by adding the [redacted] neutron multiplication factor (k_s) during k_s calculations.

To verify that the criticality axial models represent th [redacted] cask accident condition, with full density water, is [redacted] transfer disk is replaced with water. The results show a decrease [redacted]

[redacted]

6.4.3.3 Transfer Cask Criticality Evaluation

Under normal conditions, such as loading in a spent fuel pool, moderator (water) is present in the canister while it is in the transfer cask. Also, during draining and drying operations, moderator is present and its density will vary. Thus, the criticality evaluation of the transfer cask includes an evaluation of the reactivity effects of moderator density variation inside the cask. Off-normal and accident conditions are bounded by assuming the most reactive mechanical basket configuration as well as moderator intrusion into the fuel cladding (100% fuel failure).

Using the transfer cask criticality model, an evaluation of the assumption of 75% of ^{10}B in BORAL, the cumulative effect of worst case mechanical perturbations and the effect of moderator density variation is made. Table 6.4-5 shows transfer cask multiplication factors at various conditions. Table 6.4-5 shows that the assumption of 75% of the BORAL ^{10}B loading results in a 1.5% reactivity penalty and the cumulative effect of the worst case mechanical configuration results in an additional 1% reactivity penalty from the nominal configuration. Table 6.4-5 also shows that reactivity decreases monotonically with decreasing moderator density, and the optimum moderator density is at 1 g/cc. Under normal conditions involving loading, draining and drying, the maximum k_{eff} including bias and uncertainties is 0.8929. The CSAS output file (including an input listing) is shown in Figure 6.7-1. In the off-normal or accident situation involving fuel failure and moderator intrusion, the maximum k_{eff}, including biases and uncertainties, is 0.9021. The CSAS output file (including an output listing) is shown in Figure 6.7-2 Thus, the NAC-MPC transfer cask containing 36 Yankee Class fuel assemblies of the most reactive type in the most reactive configuration is well below the 0.95 NRC criticality safety limit, including all biases and uncertainties under normal, off-normal and accident conditions.

6.4.3.4 Storage Cask Criticality Evaluation

Under normal conditions, moderator is not present in the storage cask. However, access to the environment is possible via air inlets and the convective heat transfer annulus between the canister and the storage cask steel liner. This access provides paths for moderator intrusion during a flood Off-normal conditions evaluate moderator intrusion into the convective heat

transfer annulus. Under accident conditions, it is assumed that the canister confinement fails and moderator intrudes into the canister and fuel cladding (100% [redacted] failure) is evaluated along with moderator density variation. The accident condition analyses also [redacted] the effect of interior/exterior moderator density variations.

Using the storage cask criticality model, an evaluation is performed of moderator intrusion into the heat transfer annulus under off-normal conditions and into the canister under accident conditions. Table 6.4-6 shows the storage cask multiplication factors at various conditions. Under normal dry conditions, maximum k_{eff}, including biases and uncertainty, is [redacted], which is well subcritical. The CSAS output file (including an input listing) is shown in Figure 6.7-3. Under off-normal conditions involving flooding of the heat transfer annulus, the k_{eff} of the cask is even less. Under accident conditions involving full moderator intrusion into the canister and fuel clad gap, the maximum k_{eff} of the cask is 0.9018. The CSAS output file (including an input listing) is shown in Figure 6.7-4. Similar to the transfer cask analysis, the storage cask accident condition moderator density study evaluates a monotonic decrease in reactivity with moderator density inside and outside the cask. Thus, the NAC-MPC storage cask containing 36 fuel assemblies of the most reactive Yankee Class type in the most reactive configuration is well below the 0.95 NRC criticality safety limit, including all biases and uncertainties under normal, off-normal and accident conditions.

Based on the results of the criticality analysis for the [redacted] in Tables 6.4-5 and 6.4-6, respectively, there is no significant [redacted] changing reflector dimensions and material composition.

Table 6.4-1 Assembly Type Reactivity Evaluations

Assembly	Initial Enrichment (wt % ^{235}U)	In Basket k_{eff}	σ
Westinghouse Type A	4.94	0.8642	0.00105
Westinghouse Type B	4.94	0.8664	0.00102
United Nuclear Type A	4.00	0.8974	0.00087
United Nuclear Type B	4.00	0.8974	0.00106
Exxon Type A	4.00	0.8870	0.00111
Exxon Type B	4.00	0.8877	0.00111
Combustion Engineering Type A	3.90	0.8943	0.00060
Combustion Engineering Type B	3.90	0.8939	0.00163
Reconfigured Fuel Assembly	4.00	0.6280	0.0007

Table 6.4-2 Basket Tolerance Reactivity Evaluations

Analysis	k_{eff}	σ	Δk_{eff}
Nominal	0.8981	0.0007	-
Fuel Tube Maximum Opening	0.9018	0.0007	0.0037
Fuel Tube Minimum Opening	0.8916	0.0007	-0.0065
Disk Maximum Opening	0.8972	0.0007	-0.0009
Disk Minimum Opening	0.8991	0.0008	0.0010
Disk Maximum Thickness	0.8987	0.0008	0.0006
Disk Minimum Thickness	0.8972	0.0008	-0.0009
Loose Packed Disk Opening	0.8974	0.0008	-0.0007
Close Packed Disk Opening	0.8993	0.0007	0.0012

Table 6.4-3 Fuel Movement Reactivity Evaluations

Assembly Movement	Boundary Conditions	k_{eff}	σ	Δk_{eff}
Nominal	Reflective	0.8981	0.0007	-
Upper Right Corner	Mirrored	0.8954	0.0007	-0.0027
Upper Right Corner	Periodic	0.8943	0.0007	-0.0038
Lower Left Corner	Mirrored	0.8977	0.0007	-0.0004
Lower Left Corner	Periodic	0.8978	0.0008	-0.0003
Upper Left Corner	Mirrored	0.8963	0.0007	-0.0018
Upper Left Corner	Periodic	0.8961	0.0008	-0.0020
Right Side	Mirrored	0.8949	0.0007	-0.0032
Right Side	Periodic	0.8951	0.0007	-0.0030
Left Side	Mirrored	0.8978	0.0007	-0.0003
Left Side	Periodic	0.8972	0.0007	-0.0009

Table 6.4-4 Tube Movement Reactivity Evaluations

Tube Movement	Boundary Conditions	k_{eff}	σ	Δk_{eff}
Nominal	Reflective	0.8981	0.0007	-
Upper Right Corner	Mirrored	0.8999	0.0007	0.0018
Upper Right Corner	Periodic	0.8979	0.0007	-0.0002
Lower Left Corner	Mirrored	0.8984	0.0008	0.0003
Lower Left Corner	Periodic	0.8962	0.0007	-0.0019
Upper Left Corner	Mirrored	0.8991	0.0008	0.0010
Upper Left Corner	Periodic	0.8959	0.0007	-0.0022
Right Side	Mirrored	0.9005	0.0008	0.0024
Right Side	Periodic	0.8966	0.0007	-0.0015
Left Side	Mirrored	0.8968	0.0007	-0.0013
Left Side	Periodic	0.8976	0.0007	-0.0005

Table 6.4-5 Criticality Results for Transfer Cask

Cask Pitch (cm)	Basket Configuration	H_2O Inside (density g/cc)	H_2O Outside (density g/cc)	^{10}B in BORAL	k_{eff}	σ	k_s [1]
319.71	Nominal	1.0	1.0	100%	0.85035	0.00076	0.8684
319.71	Nominal	1.0	1.0	75%	0.86504	0.00070	0.8831
319.71	Worst Case	1.0	1.0	75%	0.87422	0.00076	0.8923
319.71	Worst Case	1.0	0.0001	75%	0.87488	0.00076	0.8929
319.71	Worst Case	0.8	0.0001	75%	0.82355	0.00074	0.8416
319.71	Worst Case	0.6	0.0001	75%	0.76550	0.00069	0.7835
319.71	Worst Case	0.4	0.0001	75%	0.69378	0.00064	0.7118
319.71	Worst Case	0.2	0.0001	75%	0.60267	0.00051	0.6206
319.71	Worst Case	0.1	0.0001	75%	0.55065	0.00042	0.5686
319.71	Worst Case	0.05	0.0001	75%	0.51859	0.00034	0.5365
319.71	Worst Case	0.01	0.0001	75%	0.46634	0.00032	0.4843
319.71	Worst Case + Water in Gap	1.0	1.0	75%	0.88403	0.00074	0.9021

1. Includes a Δk of 0.004 due to radial movement of fuel assembly toward basket center.

Table 6.4-6 Criticality Results for Storage Cask

Cask Pitch (cm)	Basket Configuration	H_2O Inside (density g/cc)	H_2O[1] Outside (density g/cc)	^{10}B	k	σ	k_s[2]
457.2	Nominal	0.0001	0.0001	75%	0.43088	0.00029	[illegible]
457.2	Worst Case	0.0001	1.0	75%	0.39800	0.00030	0.4159
457.2	"	0.0001	0.8	75%	0.39906	0.00031	[illegible]
457.2	"	0.0001	0.6	75%	0.39869	0.00032	0.4165
457.2	"	0.0001	0.4	75%	0.40071	0.00031	[illegible]
457.2	"	0.0001	0.2	75%	0.40963	0.00031	[illegible]
457.2	"	0.0001	0.1	75%	0.42134	0.00031	[illegible]
457.2	"	0.0001	0.05	75%	0.42924	0.00031	[illegible]
457.2	"	0.0001	0.01	75%	0.43241	0.00030	[illegible]
457.2	Worst Case + water in gap	1.0	1.0	75%	0.88376	0.00072	[illegible]
457.2	"	0.8	0.8	75%	0.83228	0.00072	[illegible]
457.2	"	0.6	0.6	75%	0.77378	0.00068	0.7918
457.2	"	0.4	0.4	75%	0.69781	0.00062	0.7158
457.2	"	0.2	0.2	75%	0.59996	0.00053	0.6179
457.2	"	0.1	0.1	75%	0.54264	0.00042	0.5606
457.2	"	0.05	0.05	75%	0.51048	0.00036	0.5284
457.2	"	0.01	0.01	75%	0.46246	0.00031	0.4804

1. Includes heat transfer annulus region.
2. Includes a Δk of 0.004 due to radial movement of fuel assembly toward basket center.

6.5 Critical Benchmark Experiments

This section provides the validation of the CSAS25 criticality analysis sequence contained in Version 4.3 of the SCALE package. This validation is required by the criticality safety standards ANSI/ANS-8.1 The section describes the method, computer program and cross section libraries used, the experimental data, the areas of applicability and the bias and margins of safety.

ANSI/ANS-8.17 prescribes the criteria to establish sub-criticality safety margins. This criteria is as follows:

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad (1)$$

where,

k_s = the calculated allowable maximum multiplication factor, k_{eff}, of the system being evaluated for all normal or credible abnormal conditions or events.

k_c = the mean k_{eff} that results from the calculation of the benchmark criticality experiments using a particular calculational method. If the calculated k_{eff} for the criticality experiments exhibit a trend with a parameter, then k_c shall be determined by extrapolation on the basis of a best fit to the calculated values. The criticality experiments used as benchmarks in computing k_c should have physical compositions, configurations, and nuclear characteristics (including reflectors) similar to those of the system being evaluated.

Δk_s = an allowance for:

(a) statistical or convergence uncertainties, or both, in the computation of k_s,
(b) material and fabrication tolerances, and
(c) geometric or material representations used in the computational method.

Δk_c = a margin for uncertainty in k_c, which includes allowance for:

(a) uncertainties in the critical experiments,
(b) statistical or convergence uncertainties, or both, in the computation of k_c,
(c) uncertainties due to extrapolation of k_c outside the range of experimental data, and

(d) uncertainties due to limitations in the geometrical or material representations used in the computational method.

Δk_m = an arbitrary margin to ensure the subcriticality of k_s.

The various uncertainties are combined statistically, if they are independent. Correlated uncertainties are combined additively.

The above equation can be rewritten as:

$$k_s \leq 1 - \Delta k_m - \Delta k_s - (1-k_c) - \Delta k_c \qquad (2)$$

Noting that the NRC requires a 5% subcriticality margin ($\Delta k_m = 0.05$) and the definition of the bias ($\beta = 1-k_c$), the above equation can then be written as:

$$k_s \leq 0.95 - \Delta k_s - \beta - \Delta\beta \qquad (3)$$

where $\Delta\beta = \Delta k_c$. Thus, k_s (the maximum allowable value for k_{eff}) must be below 0.95 minus the bias, uncertainties in the bias and uncertainties in the system being analyzed (i.e. Monte Carlo, mechanical and modeling). This is an upper safety limit criterion often used in the DOE criticality safety community.

Alternatively, this equation can be rewritten applying the bias and uncertainties to the k_{eff} of the system being analyzed as:

$$k_s = k_{eff} + \Delta k_s + \beta + \Delta\beta \leq 0.95 \qquad (4)$$

In equation 4, k_{eff} replaces k_s, and k_s has been redefined as the effective multiplication factor of the system being analyzed, including the method bias and all uncertainties. This is a maximum calculated k_{eff} criteria often used in LWR spent fuel storage and transport analyses.

Both β and $\Delta\beta$ are evaluated below for KENO-Va with the 27 group ENDF/B-IV library for use in criticality evaluations of LWR fuel in storage and transport casks.

6.5.1 Benchmark Experiments and Applicability

The criticality safety method is CSAS25 embedded in SCALE version 4.3 for the PC. CSAS25 includes: the SCALE Material Information Processor, BONAMI, NITAWL-II, and KENO-Va. The Material Information Processor generates number densities for standard compositions, prepares geometry data for resonance self-shielding, and creates data input files for the cross section processing codes. The BONAMI and NITAWL-II codes are used to prepare a resonance-corrected cross section library in AMPX working format. The KENO-Va code calculates the model k_{eff} using Monte Carlo techniques. The 27 group ENDF/B-IV neutron cross section library is used in this validation.

6.5.1.1 Description of Experiments

Sixty-three critical experiments were selected; nine Babcox and Wilcox (B&W) 2.46 wt % ^{235}U fuel storage (Baldwin), 10 Pacific Northwest Laboratory (PNL) 4.31 wt % ^{235}U lattice (Bierman, 1980), 21 PNL 2.35 and 4.31 wt % ^{235}U with metal reflectors (Bierman, 1979 & 1981), twelve PNL flux trap (Bierman 1980 and 1988) and 11 Valduc Critical Mass Laboratory (VCML) 4.74 wt % ^{235}U experiments, some involving moderator density variations (Manaranche). These experiments span a range of fuel enrichments, fuel rod pitches, neutron absorber sheet characteristics, shielding materials and geometries that are typical of LWR fuel in a cask.

The experiments are evaluated using three-dimensional models, as close to the actual experiment as possible, to achieve accurate results. Stochastic Monte Carlo error is kept within ±0.1 percent by executing at least 1,000 neutrons/generation for more than 400 generations.

6.5.1.2 Applicability of Experiments

All of the experiments chosen in this validation are applicable to either PWR, including Yankee Class, or BWR fuel. Fuel enrichments have covered a range from 2.35 up to 4.74 wt % ^{235}U typical of LWR fuel presently used. The experiment fuel rod and pitch characteristics are within the range of standard PWR or BWR fuel rods (i.e. pellet diameters from 0.78 to 1.2 cm, rod outside diameters from 0.95 to 1.88 cm and pitches from 1.26 to 1.87 cm). This is particularly true of the VCML (PWR rod type) and B&W experiments (BWR rod type). The H/U volume ratios of the experimental fuel arrays are within the range of PWR fuel assemblies (1.6 to 2.32) and BWR fuel assemblies (1.6 to 1.9).

[redacted]

Experiments covered the geometry and neutron absorber sheet arrangements typical of NAC basket designs. This included flux trap gap spacings of 3.81 cm such as in the NAC-STC basket, and gap spacing as low as 1.91 cm as in the NAC-MPC . The ^{10}B neutron absorber loadings are also typical of NAC basket designs (0.005 to 0.025) . The experiments covered the influence of water and metal reflector regions, including steel and lead, which are present in storage and transport cask shielding.

Confidence in predicting criticality, including bias and uncertainty, has been demonstrated for LWR fuel with enrichments up to 4.74 wt % ^{235}U and, based on the [illegible] trend with enrichment, confidence in extrapolating up to 4.94 wt % ^{235}U is high. Confidence in predicting criticality has been demonstrated for storage and transport arrays using flux trap or single neutron absorber sheet or simple spacing criticality control. Confidence in predicting criticality has been demonstrated for LWR fuel storage and transport arrays next to water and metal reflector regions.

6.5.2 Results of Benchmark Calculations

The k-effective results for the experiments are shown in Table 6.5-1, and a frequency distribution plot is provided in Figure 6.5-1. Five sets of cases are presented: Set 1 - B&W, Set 2 - PNL lattice, Set 3 - PNL reflector, Set 4 - PNL flux trap and Set 5 - VCML critical experiments.

The overall average and standard deviation of the sixty three cases is 0.9948 ± 0.0044. The average Monte Carlo error (statistical convergence) is ±0.0012 for the sixty-three cases. This uncertainty component is statistically subtracted from the uncertainties because it is previously included in the above standard deviation. The KENO-Va models are three-dimensional, fully explicit representations (no homogenizations) of the experimental geometry. Therefore, the uncertainty due

to limitations of geometrical modeling is taken to be 0.0. The experiments modeled cover the range of fuel types, enrichments, neutron absorber configurations, neutron absorber ^{10}B loading and metal reflector effects, so there are no extrapolations necessary outside of the range of data, and the uncertainty due to this is also taken to be 0.0. Based on the reported experimental error for the B&W cases, the reported error of the critical size number of rods for the PNL cases and the reported error for the critical height in the VCML cases, the experimental error is conservatively taken to be ±0.001. Criticality can then be represented as 1.000±0.001. This uncertainty component is statistically added to the sum of the other uncertainties because the bias is the difference between two random variates (i.e. criticality and code prediction and the uncertainty in the difference between two random variates is the statistical sum (rms) of their individual uncertainties).

Thus, the bias or average difference between code calculated and critical is $\beta=1-0.9948 = 0.0052$. The uncertainty in the bias, accounting for the statistical convergence (Monte Carlo error) and the uncertainty in criticality is $(0.0044^2 - 0.0012^2 + 0.0010^2)^{1/2} = 0.0043$. For 63 samples of criticality, the 95/95 one side tolerance factor is 2.012 (Owen). This results in a 95/95 one-sided uncertainty in the bias of $\Delta\beta=2.012\text{x}0.0043=0.0087$. Equation 4 now becomes:

$$k_{eff} + \Delta k_s + 0.0052 + 0.0087 \leq 0.95 \qquad (5)$$

where Δk_s becomes the uncertainty in k_s due to Monte Carlo error, mechanical and material tolerances and geometric or material representations. If the nominal representation of the system is evaluated for k_s, then the mechanical and material perturbation can be evaluated independently and can be combined statistically as the root sum of squares. If the worst case mechanical and material tolerances are used in the calculations of k_s (e. g., the most reactive positioning of fuel or basket components and 75% of the specified minimum boron loading), then Δk_s becomes 0.0 and the Monte Carlo error, σ_{mc}, can be combined statistically, since it is independent, with the uncertainty in the bias as:

$$k_{eff} + 0.0052 + \sqrt{0.0087^2 + (2\sigma)^2} \leq 0.95 \qquad (6)$$

6.5.2.1 Trends

Scatter plots of k_{eff} versus wt % ^{235}U, rod pitch, H/U volume ratio, average neutron group causing fission, ^{10}B loading for flux trap cases, and flux trap gap thickness are shown in Figures 6.5-2 through 6.5-7. Included in these scatter plots are linear regression lines with a corresponding correlation coefficient. This statistically indicates any trend or lack thereof. In particular, the correlation coefficient is a measure of the linear relationship between k_{eff} and a critical experiment parameter. If r is +1, a perfect linear relationship with a positive slope is indicated, and if r is -1, a perfect linear relationship with a negative slope is indicated. When r is 0, no linear relationship is indicated. The largest correlation coefficient indicated in the plots is 0.1302 (k_{eff} versus enrichment) and the lowest is 0.0048 (k_{eff} versus ^{10}B loading in flux trap experiments). Based on the correlation coefficients, no statistically significant trends exist over the range of variables studied. Most importantly, no trend is shown with flux trap gap spacing and/or ^{10}B loading. This is the major criticality control feature of the NAC-STC and the NAC-MPC basket.

6.5.3 [redacted]

NUREG/CR-6361, [redacted] and Storage Facilities" (NUREG) provides [redacted] the determination of bias and subcritical limits in [redacted] NUREG, a series of LWR critical experiments is [redacted] modeling. In Section 3, the critical experiments are modeled, and the [redacted] presented. The method utilized in the NUREG is [redacted] section library embedded in SCALE 4.3. Inputs are provided in [redacted] Section 4, a guide for the determination of bias and [redacted] ANSI/ANS-8.17 and statistical analysis of the trending [redacted] experiment selection and applicability are presented in Section 5. [redacted] 4 of the NUREG is described in detail below and is compared to the NAC approach presented in Sections 6.5, 6.5.1 and 6.5.2.

NAC has performed an extensive LWR critical benchmarking as documented in Sections [redacted] 6.5.2. The method used in NAC benchmarking/validation included the [redacted] criticality analysis sequence, with the 27 group ENDF/B-IV library [redacted] Trending in k_{eff} was evaluated for the following independent [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Δk_s = allowance for:

a) statistical or convergence uncertainties, or both, in computation of k_s,
b) material and fabrication tolerances, and
c) geometric or material representations used in computational method.

Δk_c = margin for uncertainty in k_s which includes allowance for:

a) uncertainties in critical experiments,
b) statistical or convergence uncertainties, or both, in computation of k_c,
c) uncertainties resulting from extrapolation of k_c outside range of experimental data, and
d) uncertainties resulting from limitations in geometrical or material representations used in the computational method.

Δk_m = arbitrary administrative margin to ensure subcriticality of k_s

The various uncertainties are combined statistically if they are independent. Correlated uncertainties are combined by addition.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

$$k_s + \Delta k_s \le \text{Upper Safety Limit}$$ [redacted]

where,

$$USL = 1 - \Delta k_m - \beta - \Delta\beta$$ [redacted]

This is the Upper Safety Limit criterion as described [redacted] methods are prescribed for the statistical determination [redacted] Administrative Margin (USL-1) and Single Sided Uniform [redacted] first method, $\Delta k_m = 0.05$ and a lower confidence band [redacted] regression of k_{eff} as a function of some system parameter [redacted] administrative margin is set to zero and a uniform lower [redacted] linear regression. The second method provides a criticality safety margin [redacted] 0.05. In cases where there are a limited number of data points, this method may indicate the need for a larger administrative margin. In both cases, all of the significant system parameters need to be studied to determine the strongest correlation.

In the analyses presented in Section 6.5.1 and 6.5.2, the bias and uncertainty are applied directly to the estimate of the system k_{eff}. Noting that the NRC requires a 5% subcritical margin ($\Delta k_m = 0.05$), Equation 5 can be rewritten applying the bias and uncertainty in the bias to the k_s of the system being analyzed as:

$$k_s + \Delta k_s + \beta + \Delta\beta \le 0.95 \qquad (6)$$

In Equation 6, the method bias and all uncertainties are added to k_s. This is the maximum k_s criterion defined in Section 6.5.2.

[redacted]

Both NUREG/CR-6361 and the NAC evaluation [redacted] for all of the major system [redacted] correlation. This is based on the observation that the [redacted] ± 1. Thus a constant bias with a 95/95 confidence [redacted] statistical analysis of the k_{eff} results produced a bias of 0.006[redacted]. Adding the two together and subtracting from 0.95 yield[redacted]]

To assure compliance with NUREG/CR-6361, an [redacted] USLSTATS computer code from NUREG/CR-6361, which [redacted] and bias uncertainty used in Section 6.5.2.

To evaluate the relative importance of the trend analysis to the upper safety limits, correlation coefficients are required for all independent variables. Table 6.5-8 [illegible] the correlation coefficient, R, for each linear fit of k_{eff} versus experimental parameter (R is extracted from Figure 6.5-2 through Figure 6.5-7 by taking the square root of the R^2 value). Based on the highest correlation coefficient and the method presented in NUREG/CR-6361, a USL is established based on the variation of k_{eff} with enrichment. Note that even the enrichment function shows a low statistical correlation coefficient (an |R| equal or near 1 would indicate a good fit). The output generated by USLSTATS is shown in Figure 6.5-8.

The NAC applied USL of 0.9361 bounds the calculated upper safety limits for all enrichment values above 3.0 wt % ^{235}U. Since the maximum reactivities in the NAC-MPC system are calculated at enrichments well above this level, the existing bias bounds the NUREG calculated USL. The parameters of the most reactive fuel element analysis are presented in Table 6.5-5.

Figure 6.5-1 KENO-Va Validation - 27 Group Library Results Frequency Distribution of K_{eff} Values

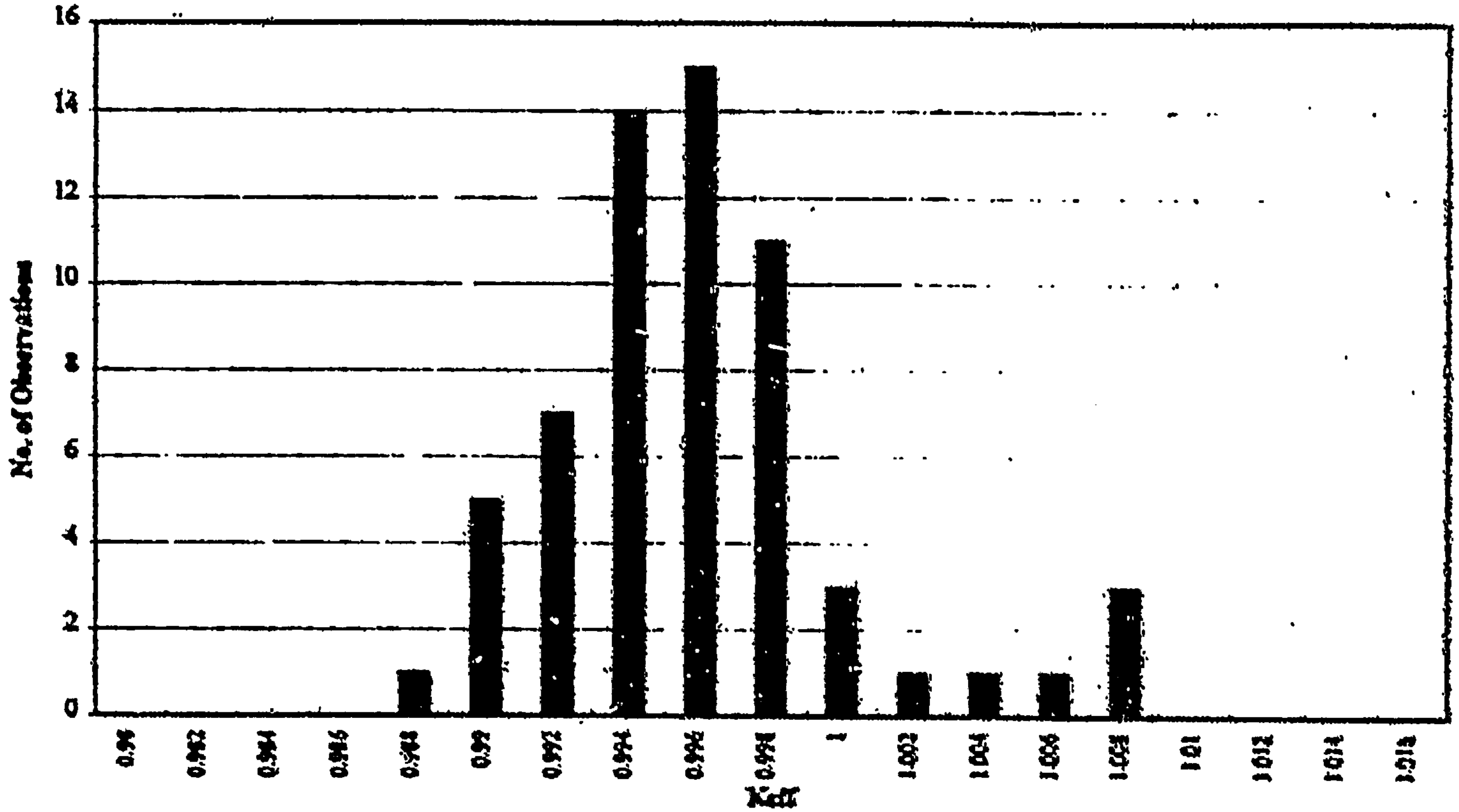


Figure 6.5-2 KENO-Va Validation -27 Group Library K_{eff} versus Enrichment

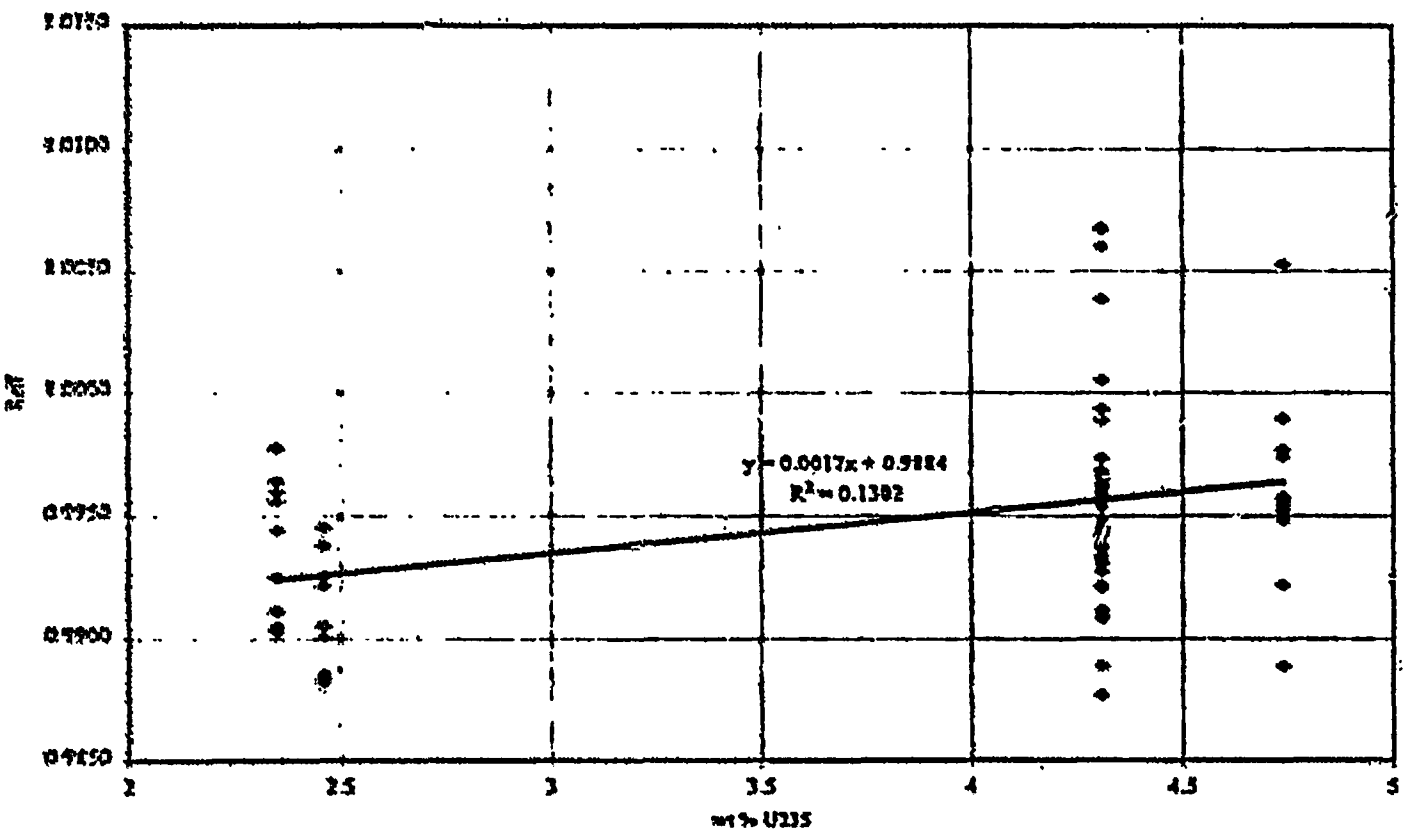


Figure 6.5-3 KENO-Va Validation - 27 Group Library K_{eff} versus Rod Pitch

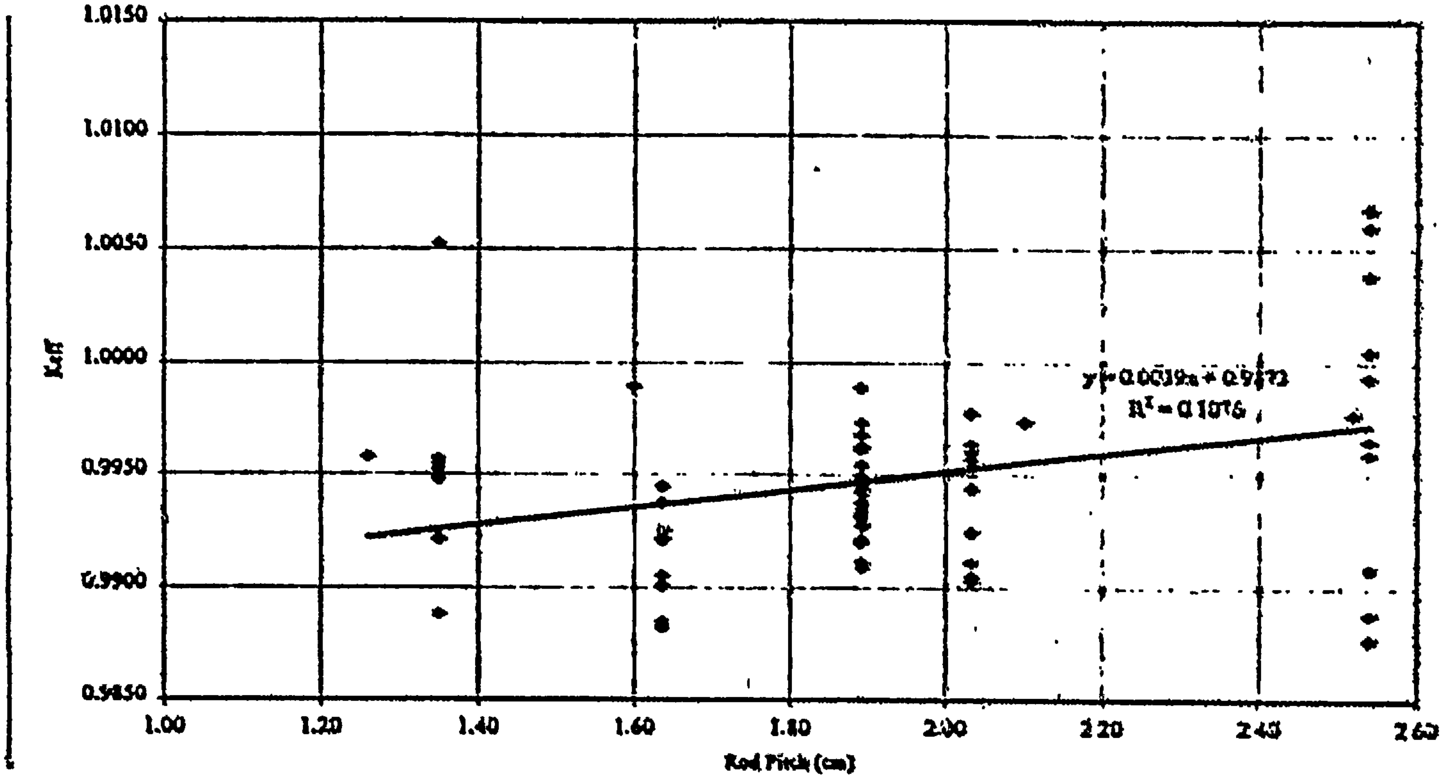


Figure 6.5-4 KENO-Va Validation -27 Group Library K_{eff} versus H/U Volume Ratio

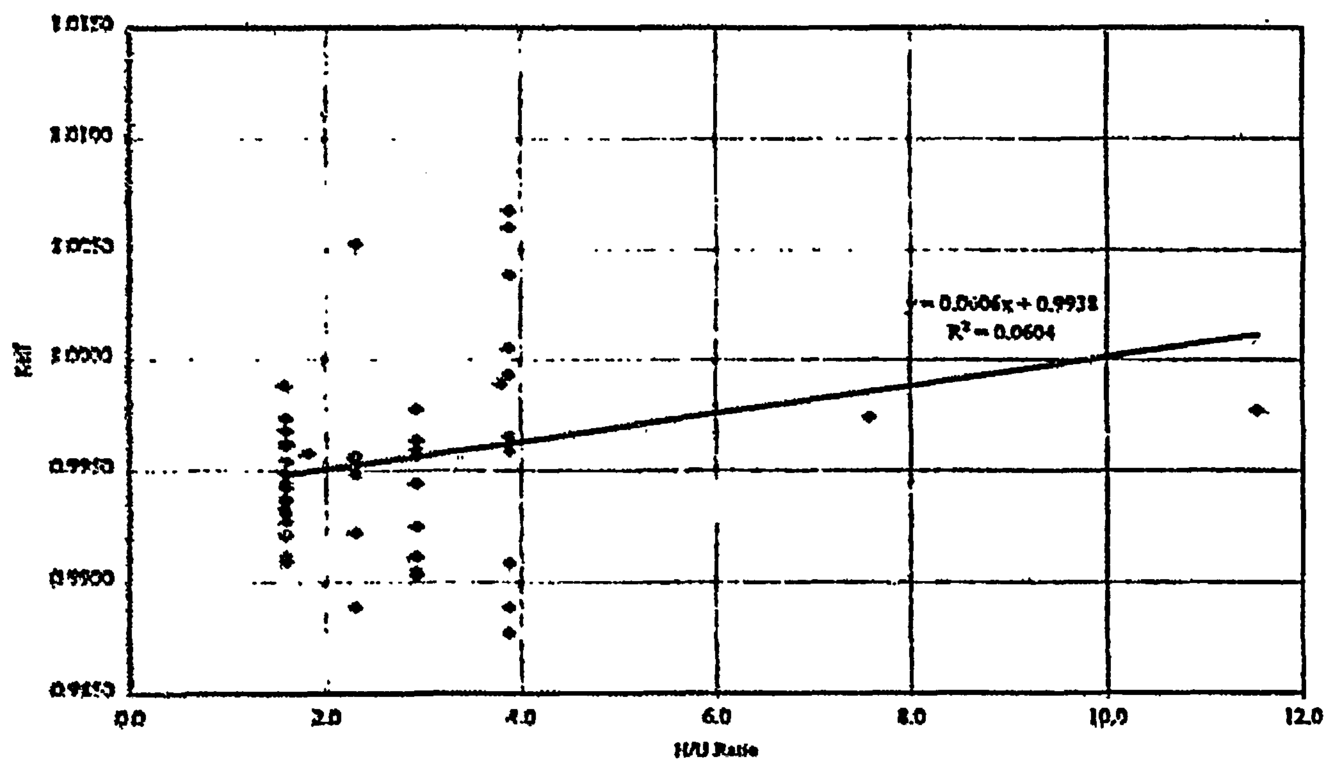


Figure 6.5-5 KENO-Va Validation -27 Group Library K_{eff} versus Average Group of Fission

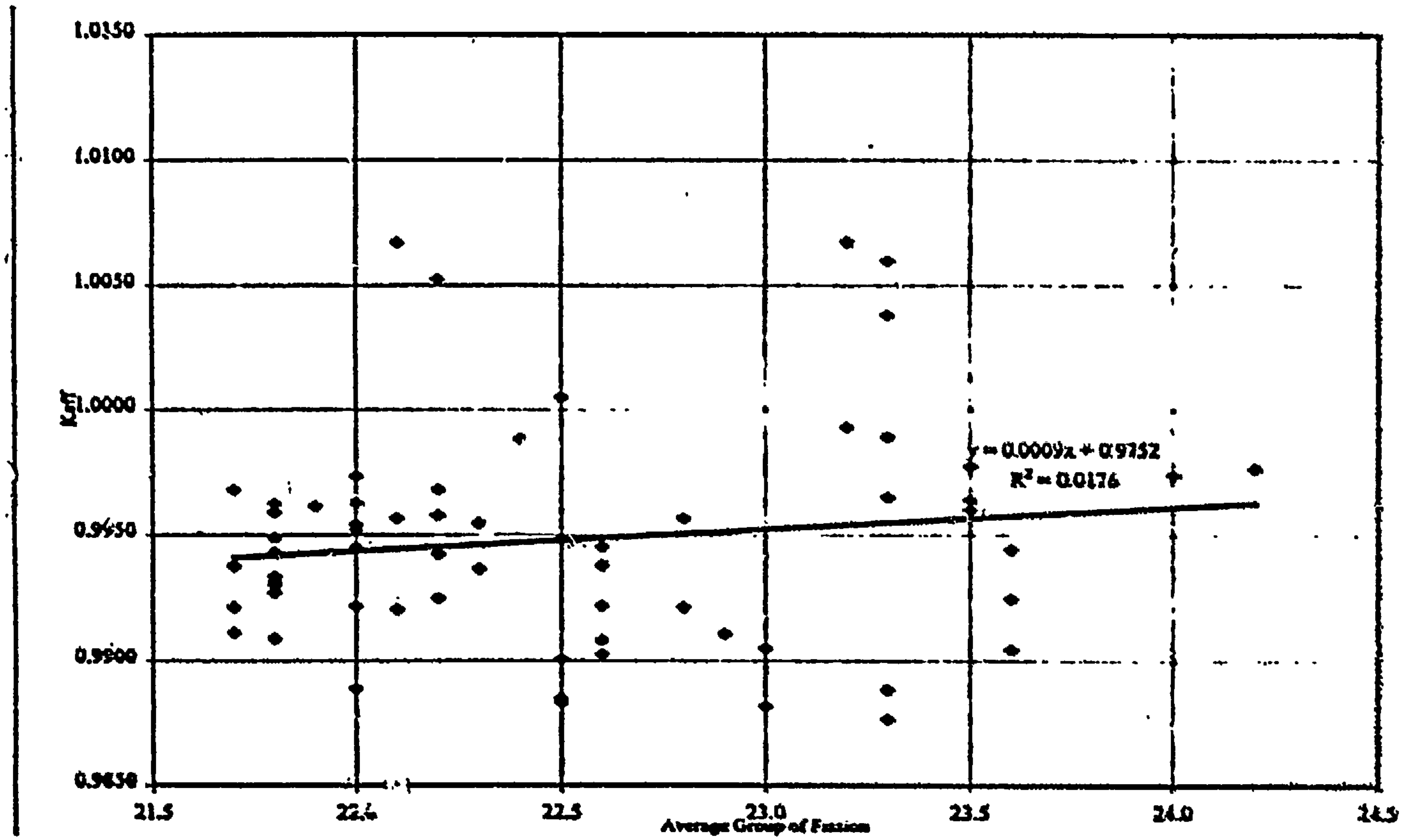


Figure 6.5-6 KENO-Va Validation - 27 Group Library K_{eff} versus ^{10}B Loading For Flux Trap Criticals

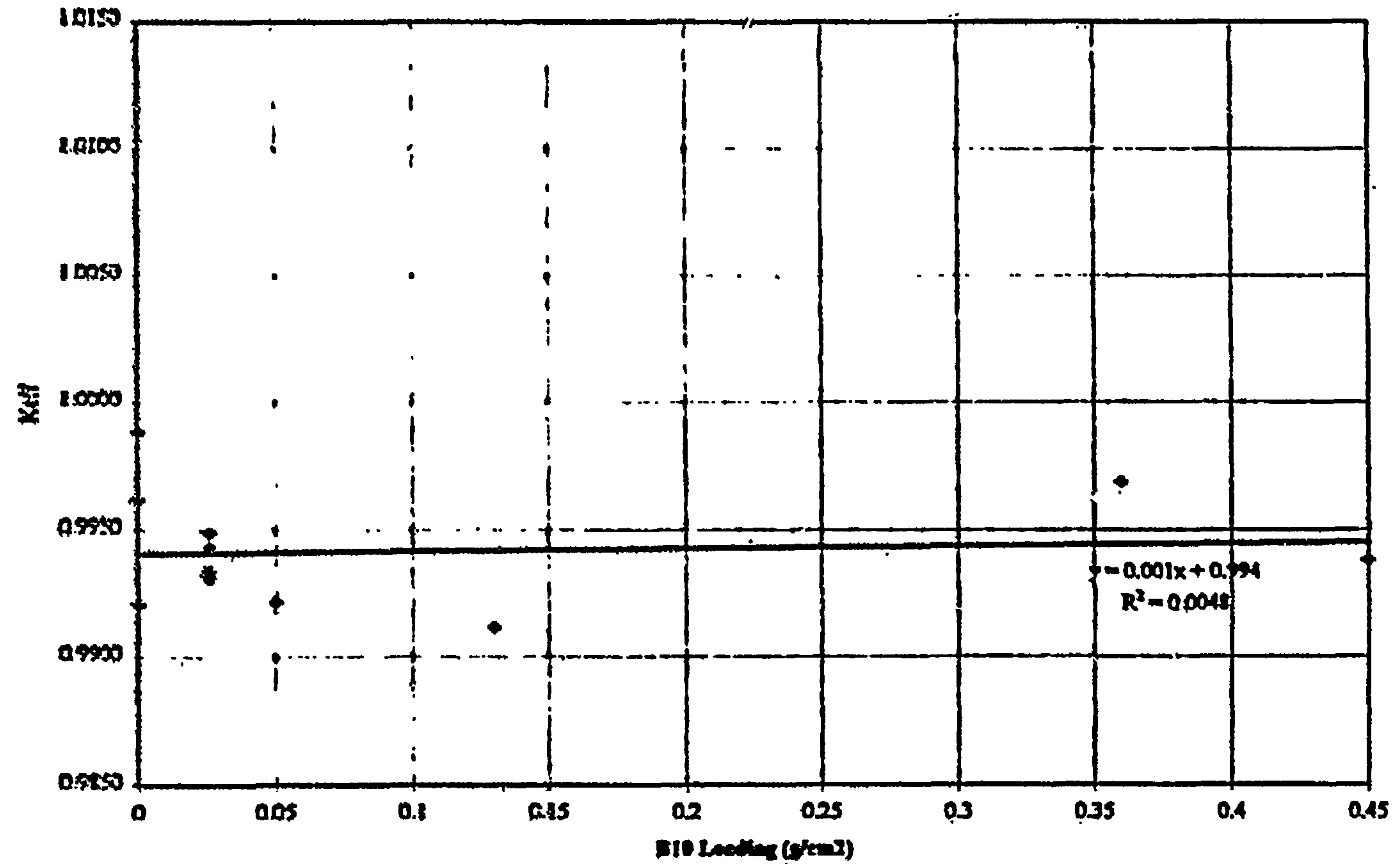
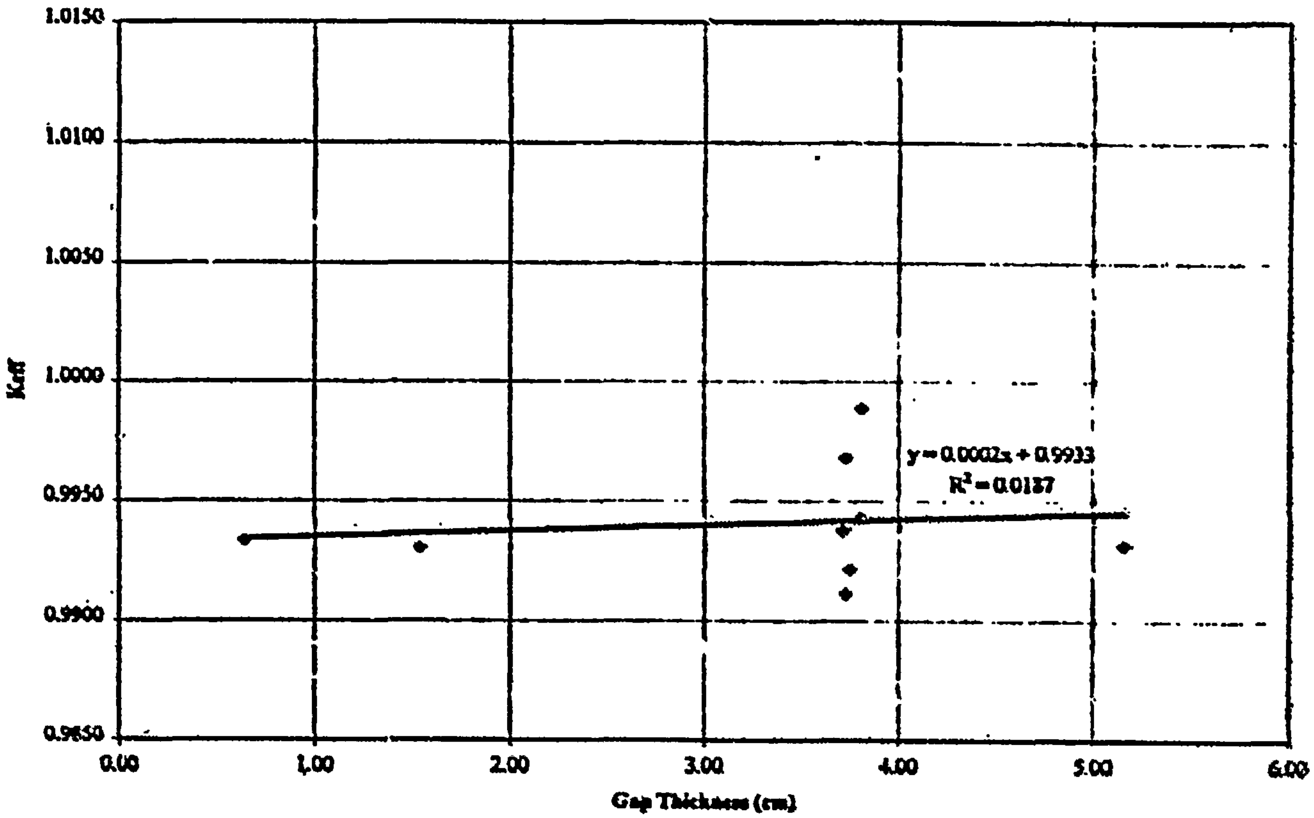


Figure 6.5-7 KENO-Va Validation -27 Group Library Results K_{eff} versus Flux Trap Critical Gap Thickness



[redacted]

```
uslstats: a utility to calculate upper subcritical
limits for criticality safety applications

*****************************************************************************
Version 1.3.4, February 12, 1998
Oak Ridge National Laboratory
*****************************************************************************

Input to statistical treatment from file:EN_KEFF.TXT

Title: 63 LWR CRITICAL EXPERIMENT KEFF VS ENRICHMENT

Proportion of the population = .995
Confidence of fit            = .950
Confidence on proportion     = .950
Number of observations       =   63
Minimum value of closed band =   0.00
Maximum value of closed band =   0.00
Administrative margin        =   0.05
```

independent variable - x	dependent variable - y	deviation in y	independent variable - x	dependent variable - y	deviation in y
2.35000E+00	9.96400E-01	1.00000E-03	4.31000E+00	9.96500E-01	1.10000E-03
2.35000E+00	9.94400E-01	1.00000E-03	4.31000E+00	1.00680E+00	2.12000E-03
2.35000E+00	9.90500E-01	1.00000E-03	4.31000E+00	1.00380E+00	1.20000E-03
2.35000E+00	9.96000E-01	1.10000E-03	4.31000E+00	9.48900E-01	1.10000E-03
2.35000E+00	9.97830E-01	1.00050E-03	4.31000E+00	9.95900E-01	1.10000E-03
2.35000E+00	9.92500E-01	1.00000E-03	4.31000E+00	1.00670E+00	1.00000E-03
2.35000E+00	9.90300E-01	9.00000E-04	4.31000E+00	1.00050E+00	1.10000E-03
2.35000E+00	9.95700E-01	1.00050E-03	4.31000E+00	9.90800E-01	1.10000E-03
2.35000E+00	9.91100E-01	1.00000E-03	4.31000E+00	9.98300E-01	1.20000E-03
2.46000E+00	9.92100E-01	1.10000E-03	4.31000E+00	9.92100E-01	1.20000E-03
2.46000E+00	9.92500E-01	9.00000E-04	4.31000E+00	9.91100E-01	1.20000E-03
2.46000E+00	9.93800E-01	9.00000E-04	4.31000E+00	9.96800E-01	1.30000E-03
2.46000E+00	9.90500E-01	1.00000E-03	4.31000E+00	9.93830E-01	1.20000E-05
2.46000E+00	9.86200E-01	1.00000E-03	4.31000E+00	9.93405E-01	1.00000E-03
2.46000E+00	9.94500E-01	1.00000E-03	4.31000E+00	9.93100E-01	1.00000E-03
2.46000E+00	9.92200E-01	1.00000E-03	4.31000E+00	9.84300E-01	1.00000E-03
2.46000E+00	9.83500E-01	1.00000E-03	4.31000E+00	9.93200E-01	1.00000E-03
2.46000E+00	9.81600E-01	1.00000E-03	4.31000E+00	9.91900E-01	1.00000E-03
2.46000E+00	9.93100E-01	9.00000E-04	4.31000E+00	9.92000E-01	1.00000E-03
4.31000E+00	9.95400E-01	1.40000E-03	4.31000E+00	9.96200E-01	1.00000E-03
4.31000E+00	9.94500E-01	1.36000E-03	4.74000E+00	9.92200E-01	1.30000E-03
4.31000E+00	9.97400E-01	1.30000E-03	4.74000E+00	9.88300E-01	1.30000E-03
4.31000E+00	9.96300E-01	1.10000E-03	4.74000E+00	9.95700E-01	1.30000E-03
4.31000E+00	9.92700E-01	1.20000E-03	4.74000E+00	1.03530E+00	1.19000E-03
4.31000E+00	9.93[illegible]E-01	1.20000E-03	4.74000E+00	9.95500E-01	1.20000E-03
4.31000E+00	9.96200E-01	1.20000E-03	4.74000E+00	9.94800E-01	1.30000E-03
4.31000E+00	9.97000E-01	[illegible]	4.74000E+00	9.95800E-01	1.20000E-03
4.31000E+00	9.96200E-01	1.20000E-03	4.74000E+00	9.95200E-01	1.20000E-03
4.31000E+00	9.96800E-01	1.20000E-03	4.74000E+00	9.98300E-01	1.30000E-03
4.31000E+00	9.97700E-01	[illegible]	4.74000E+00	9.97400E-01	1.20000E-03
4.31000E+00	9.92[illegible]E-01	[illegible]	4.74000E+00	9.97700E-01	1.10000E-03
4.31000E+00	1.00600E+00	[illegible]			

```
chi = [illegible] (upper bound = 9.49). The data tests normal.


Output from statistical treatment

63 LWR CRITICAL EXPERIMENT KEFF VS ENRICHMENT

Number of data points (n)                              [illegible]
Linear regression, k(X)                                [illegible]
Confidence on fit (1-gamma) [input]                    [illegible]
Confidence on proportion (alpha) [input]               [illegible]
Proportion of population falling above
lower tolerance interval (rho) [input]                 [illegible]
Minimum value of X                                     [illegible]
Maximum value of X                                     [illegible]
Average value of X                                     [illegible]
Average value of k                                     [illegible]
Minimum value of k                                     [illegible]
Variance of fit, s(k,X)^2                              [illegible]
Within variance, s(w)^2                                [illegible]
Pooled variance, s(p)^2                                [illegible]
Pooled std. deviation, s(p)                            [illegible]
C(alpha,rho)*s(p)                                      [illegible]
```

[redacted]

```
student-t ? (n-2,1-gamma)                              1.67078E+00
Confidence band width, W                               7.3606E-03
Minimum margin of subcriticality, C*s(p)-W             8.1273E-03

Upper subcritical limits: ( 2.35000 <= X <=  4.74000)
***** *********** *******

USL Method 1 (Confidence Band with
Administrative Margin)            USL1 = 0.9311 + ( 1.6748E-03)*X

USL Method 2 (Single-Sided Uniform
Width Closed Interval Approach)   USL2 = 0.9729 + ( 1.6748E-03)*X


USLs Evaluated Over Range of Parameter X:
**** ********* **** ***** ** *********

X:   2.35   2.69   3.03   3.37   3.72   4.06   4.40   4.74
-----------------------------------------------------------
USL-1:  0.9350 0.9356 0.9362 0.9367 0.9373 0.9379 0.9384 0.9390
USL-2:  0.9769 0.9775 0.9780 0.9786 0.9792 0.9797 0.9803 0.9809
-----------------------------------------------------------

*****************************************************************************
Thus spake USLSTATS
Finis.
```

Table 6.5-1 KENO-Va and 27 Group Library Validation Statistics

Criticals	Configuration	wt % ^{235}U	Pitch (cm)	Clad OD (cm)	Pellet OD (cm)	H/U	Sol. B (ppm)	Poison	g $^{10}B/cm^2$	Gap(cm)	Gap Den.	Ave. Grp	K_{eff}	σ
Set 1														
B&W-I	Cylindrical	2.46	1.636	1.206	1.03	1.6	0	NA	NA	0		22.8	0.9921	0.0011
B&W-II	3X3-14X14	2.46	1.636	1.206	1.03	1.6	1,037	NA	NA	0		22.2	0.9925	0.0009
B&W-III	3X3-14X14	2.46	1.636	1.206	1.03	1.6	764	NA	NA	1.636		22.6	0.9938	0.0009
B&W-IX	3X3-14X14	2.46	1.636	1.206	1.03	1.6	0	NA	NA	6.543		23	0.9905	0.0010
B&W-X	3X3-14X14	2.46	1.636	1.206	1.03	1.6	143	NA	NA	4.907		23	0.9882	0.0010
B&W-XI	3X3-14X14	2.46	1.636	1.206	1.03	1.6	514	Steel	0	1.636		22.6	0.9945	0.0010
B&W-XIII	3X3-14X14	2.46	1.636	1.206	1.03	1.6	15	B-Al	0.0052	1.636		22.6	0.9922	0.0010
B&W-XIV	3X3-14X14	2.46	1.636	1.206	1.03	1.6	92	B-Al	0.0040	1.636		22.5	0.9885	0.0010
B&W-XVII	3X3-14X14	2.46	1.636	1.206	1.03	1.6	487	B-Al	0.0008	1.636		22.5	0.9884	0.0010
B&W-XIX	3X3-14X14	2.46	1.636	1.206	1.03	1.6	634	B-Al	0.0003	1.636		22.5	0.9901	0.0009
												Average	0.9911	0.0023
Set 2														
PNL-031	17X13 Lattice	4.31	1.892	1.415	1.265	1.6	0	NA	NA	NA	NA	22.0	0.9954	0.0014
PNL-044	16X14 Lattice	4.31	1.892	1.415	1.265	1.6	0	NA	NA	NA	NA	22.0	0.9945	0.0013
PNL-045	14X16 Lattice	4.31	1.892	1.415	1.265	1.6	0	NA	NA	NA	NA	22.0	0.9974	0.0013
PNL-046	12x19 Lattice	4.31	1.892	1.415	1.265	1.6	0	NA	NA	NA	NA	22.0	0.9963	0.0013
PNL-047	4 11X14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.066	2.83		21.8	0.9927	0.0012
PNL-059	4 11X14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.030	2.83		21.8	0.9909	0.0012
PNL-095	4 11X14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.026	2.83		21.8	0.9962	0.0012
PNL-115	4 5X12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Aluminum	0	2.83		22.3	0.9937	0.0013
PNL-064	4 5X12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Steel (302)	0	2.83		22.2	0.9942	0.0012
PNL-077	4 5X12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Steel (485)	0	2.83		22.2	0.9968	0.0012
												Average	0.9948	0.0020

Table 6.5-1 KENO-Va and 27 Group Library Validation Statistics (continued)

Criticals	Configuration	wt % ^{235}U	Pitch (cm)	Clad OD (cm)	Pellet OD (cm)	H/U	Sol. B (ppm)	Poison	g $^{10}B/cm^2$	Gap(cm)	Gap Den.	Ave. GEs	K_{eff}	σ
Set 3										Cluster	Wall/Cluster			
PNL-STA	3X1 St Refl.	2.35	2.032	1.27	1.1176	2.9	0	NA	NA	10.65	0.00	23.5	0.9964	0.0012
PNL-STB	3X1 St Refl.	2.35	2.032	1.27	1.1176	2.9	0	NA	NA	11.20	1.32	23.6	0.9944	0.0010
PNL-STC	3X1 St Refl.	2.35	2.032	1.27	1.1176	2.9	0	NA	NA	10.36	2.62	23.6	0.9905	0.0010
PNL-PBA	3X1 Pb Refl.	2.35	2.032	1.27	1.1176	2.9	0	NA	NA	13.84	0.00	23.5	0.9960	0.0011
PNL-PBB	3X1 Pb Refl.	2.35	2.032	1.27	1.1176	2.9	0	NA	NA	13.72	0.66	23.5	0.9978	0.0010
PNL_PBC	3X1 Pb Refl.	2.35	2.032	1.27	1.1176	2.9	0	NA	NA	11.25	2.62	23.6	0.9925	0.0010
PNL-DUA	3X1 DU Refl.	2.35	2.032	1.27	1.1176	2.9	0	NA	NA	11.83	0.00	22.6	0.9903	0.0009
PNL-DUB	3X1 DU Refl.	2.35	2.032	1.27	1.1176	2.9	0	NA	NA	14.11	1.96	22.8	0.9957	0.0010
PNL-DUC	3X1 DU Refl.	2.35	2.032	1.27	1.1176	2.9	0	NA	NA	13.70	2.62	22.9	0.9918	0.0010
PNL-H20	3X1 H2O Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	8.24	inf	23.3	0.9877	0.0021
PNL-ST0	3X1 St Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	12.89	0	23.2	0.9993	0.0012
PNL-ST1	3X1 St Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	14.12	1.32	23.3	1.0000	0.0022
PNL-ST26	3X1 St Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	12.44	2.62	23.3	0.9963	0.0011
PNL-PB0	3X1 Pb Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	20.62	0	23.2	1.0063	0.0031
PNL-PB13	3X1 Pb Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	19.04	1.32	23.3	1.0018	0.0012
PNL-PB5	3X1 Pb Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	10.3	5.41	23.3	0.9969	0.0011
PNL-DU0	3X1 DU Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	15.38	0	21.8	0.9959	0.0011
PNL-DU13	3X1 DU Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	19.04	1.32	22.1	1.0067	0.0010
PNL-DU39	3X1 DU Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	18.05	3.91	22.5	1.0005	0.0011
PNL-DU54	3X1 DU Refl.	4.31	2.54	1.415	1.265	3.9	0	NA	NA	13.49	5.41	22.6	0.9908	0.0011
												Average	0.9964	0.0060

Table 6.5-1 KENO-Va and 27 Group Library Validation Statistics (continued)

Criticals	Configuration	wt % ^{235}U	Pitch (cm)	Clad OD (cm)	Pellet OD (cm)	H/U	Sol. B (ppm)	Poison	g $^{10}B/cm^2$	Gap(cm)	Gap Den.	Ave. Gfis	K_{eff}	σ
Set 4														
PNL-229	2x2 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	Aluminum	0	3.81	0.9982	22.4	0.9989	0.0012
PNL-230	2x2 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	BORAL	0.05	3.75	0.9982	21.7	0.9921	0.0012
PNL-228	2x2 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	BORAL	0.13	3.73	0.9982	21.7	0.9911	0.0012
PNL-218	2x2 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	BORAL	0.36	3.73	0.9982	21.7	0.9968	0.0013
PNL-231	2x2 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	BORAL	0.45	3.71	0.9982	21.7	0.9938	0.0012
PNL-127	2x1 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	BORAL	0.026	0.64	0.9982	21.8	0.9934	0.0010
PNL-126	2x1 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	BORAL	0.026	1.54	0.9982	21.8	0.9931	0.0010
PNL-128	2x1 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	BORAL	0.026	3.80	0.9982	21.8	0.9943	0.0010
PNL-125	2x1 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	BORAL	0.026	5.16	0.9982	21.8	0.9932	0.0010
PNL-124	2x1 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	BORAL	0.026	INF	0.9982	21.8	0.9949	0.0010
PNL-123-S	2x1 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	Steel	0	3.80	0.9982	22.1	0.9920	0.0010
PNL-124-S	2x1 Hex Trap	4.31	1.89	1.415	1.265	1.6	0	Steel	0	INF	0.9982	21.9	0.9962	0.0010
												Average	0.9941	0.0022
Set 5														
VCML	2x2 Water Gap	4.74	1.35	0.94	0.79	2.3	0	NA	NA	1.90	0	22.0	0.9922	0.0013
VCML	2x2 Water Gap	4.74	1.35	0.94	0.79	2.3	0	NA	NA	1.90	0.0323	22.0	0.9889	0.0013
VCML	2x2 Water Gap	4.74	1.35	0.94	0.79	2.3	0	NA	NA	1.90	0.2879	22.1	0.9957	0.0013
VCML	2x2 Water Gap	4.74	1.35	0.94	0.79	2.3	0	NA	NA	1.90	0.5540	22.2	1.0053	0.0011
VCML	2x2 Water Gap	4.74	1.35	0.94	0.79	2.3	0	NA	NA	2.50	0.9982	22.3	0.9955	0.0012
VCML	2x2 Water Gap	4.74	1.35	0.94	0.79	2.3	0	NA	NA	5.00	0.9982	22.5	0.9948	0.0013
VCML	Square Lattice	4.74	1.26	0.94	0.79	1.8	0	NA	NA	NA	NA	22.2	0.9938	0.0012
VCML	Square Lattice	4.74	1.35	0.94	0.79	2.3	0	NA	NA	NA	NA	22.0	0.9952	0.0012
VCML	Square Lattice	4.74	1.60	0.94	0.79	3.8	0	NA	NA	NA	NA	23.3	0.9989	0.0013
VCML	Square Lattice	4.74	2.10	0.94	0.79	7.6	0	NA	NA	NA	NA	24.0	0.9974	0.0012
VCML	Square Lattice	4.74	2.52	0.94	0.79	11.5	0	NA	NA	NA	NA	24.2	0.9977	0.0011
												Average	0.9961	0.0041

[redacted] [redacted]

[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

Table 6.5-3 Most Reactive Configuration System [redacted]

Parameter	[redacted]
Enrichment (wt % ^{235}U)	[redacted]
Rod pitch (cm)	[redacted]
H/U volume ratio	[redacted]
^{10}B loading (g/cm^2)	[redacted]
Average group causing fission	[redacted]
Flux gap thickness (cm)	[redacted]

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[redacted]

7.0 CONFINEMENT

The NAC-MPC transportable storage canister (canister) provides confinement for its radioactive contents in long-term storage. The confinement boundary is closed by welding that presents a leaktight barrier to the release of contents in all of the evaluated normal, off-normal and accident conditions.

The NAC-MPC canister contains an inert gas (helium). The confinement boundary retains the helium and also prevents the entry of outside air into the NAC-MPC. The exclusion of air precludes degradation of the fuel rod cladding over time, due to cladding oxidation failures.

The NAC-MPC canister confinement system meets the requirements of 10 CFR 72.24 for protection of the public from release of radioactive material, and 10 CFR 72.122 for protection of the spent fuel contents in long-term storage such that future handling of the contents would not pose an operational safety concern.

[redacted]

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7.1 Confinement Boundary

Confinement of the contents in long-term storage is provided by the transportable storage canister [redacted]. The welded canister [redacted] forms the confinement vessel.

The primary confinement boundary of the canister consists of the canister shell, bottom closure plate, shield lid, the two (2) port covers, and the welds that join these components. There are no bolted closures or mechanical seals in the primary confinement boundary. The confinement boundary welds are described in Table 7.1-1.

[redacted]

7.1.1 Confinement Vessel

[redacted]

[redacted] [redacted]

The canister consists of three (3) principal components. the canister shell, the shield lid, and the structural lid. The canister shell is a right circular cylinder constructed of 5/8-inch thick rolled Type 304L stainless steel plate. The edges of the rolled plate are joined using full penetration welds. It is closed at the bottom end by a 1-inch thick circular plate joined to the shell by a full penetration weld. The inside and outside diameter of the canister are 69.39 inches and 70.64 inches, respectively. The inside length is 121.5 inches. The overall external length of the canister is 122.5 inches. The canister is fabricated in accordance with the ASME Boiler and Pressure Vessel Code [redacted], Section III, Subsection NB, except for the top end weld [redacted] and their [redacted] examinations. [redacted]

After loading, the canister is closed at the top by a shield lid and a structural lid. The shield lid is a 5-inch-thick Type 304 stainless steel plate. It is joined to the canister shell using a field installed bevel weld. The shield lid contains the drain and fill penetrations and provides gamma radiation protection to the operators during the draining, drying and inerting operations. After the shield lid is welded in place, the canister is pressure tested and leak tested to ensure

leaktightness. Following draining, drying and inerting operations, the penetrations are closed with Type 304 stainless steel port covers that are welded in place with bevel welds. The operating procedures describing the handling steps to close the canister are presented in Chapter 8. The pressure and leak test procedures are described in Chapter 9.

A secondary, or redundant, confinement boundary [redacted] is [redacted] at the top of the canister by a structural lid, which is placed over the shield lid. The structural lid is a 3-inch thick Type 304L stainless steel plate. The structural lid provides the attachment points for lifting the loaded canister. The structural lid is welded to the shell using a field installed bevel weld. The weld specifications and weld inspection and acceptance criteria are presented in Sections 7.1.3.[redacted] and 7.1.3.[redacted], respectively.

The confinement boundaries are shown in Figures 7.1-1 and 7.1-2. As illustrated in Figure 7.1-2, the secondary [redacted] confinement boundary includes: the structural lid, the upper 3.5 inches of the canister shell and the joining weld. This boundary provides additional assurance of the leaktightness of the canister during its service life.

[redacted]

7.1.1.[redacted] Design Documents, Codes, and Standards

The canister [redacted] constructed in accordance with the license drawings presented in Section 1.5. The principal Codes and Standards that apply to the [redacted] design, fabrication and assembly are described in Sections 7.1.1 and 7.1.3 and are shown on the licensing drawings. Other Codes and Standards are applied as appropriate in the design or specification of the canister.

7.1.1.[redacted] Technical Requirements for [redacted] Canister [redacted]

The canister confines up to 36 intact or reconfigured Yankee Class fuel assemblies[redacted]. Over its 50-year design life, the canister precludes the release of radioactive contents and precludes the entry of air that could potentially damage the cladding of the stored spent fuel. The design of the canister to the requirements of ASME Section III, Subsection NB, ensures that the canister maintains confinement in all of the evaluated normal, off-normal, and accident conditions.

The design of the canister allows the recovery of stored spent fuel, [redacted] should that become necessary.

The canister has no exposed penetrations, no mechanical closures, and does not employ seals to maintain confinement. There is no requirement for continuous monitoring.

The design basis parameters for the Yankee Class fuel [redacted] are presented in Section 1.2.3. The design criteria that apply to the canister, as an element of the NAC-MPC dry storage system, are presented in Table 1.2-1.

[redacted]

7.1.1.[redacted] Release Rate

The primary confinement boundary is formed by a stainless steel plate joined by welding. The welds are visually inspected, nondestructively examined, and pressure tested to confirm integrity. Consequently, the confinement boundary is [redacted] leaktight. There is no maximum allowable leak rate specified for the NAC-MPC canister, as leakage to any degree up to the level of sensitivity of the leak test, is not acceptable. However, to demonstrate leaktightness of the shield lid, a leak [redacted] based on [redacted]

[redacted]

[redacted]

[redacted]

7.1.2 Confinement Penetrations

Two penetrations (with quick disconnect fittings) are provided in the [redacted] shield lid for operator use One penetration is used for draining residual water from the canister. It connects to a drain tube that extends to the bottom of the canister. The other penetration extends only to the underside of the shield lid. It is used to introduce air, or inert gas, into the top of the canister. Once draining is completed, either penetration may be used for vacuum drying and backfilling with helium. Following backfilling, both penetrations are closed with port covers that are [redacted]

welded to the shield lid. When the port covers are in place, the penetrations are not accessible. These port covers are subsequently enclosed and covered by the structural lid, which is also welded in place. The structural lid and the remainder of the canister have no penetrations.

[redacted]

7.1.3 Seals and Welds

This section describes the process used to properly [redacted] the confinement vessel [redacted] Weld processes [redacted] and acceptance criteria are described in Sections 7.1.3.2 and 7.1.3.3.

There are no elastomer or metallic seals used in the confinement boundary of the canister [redacted].

7.1.3.1 Fabrication

All cutting, machining, welding, and forming is in accordance with Section III, Article NB-4000 of the ASME Code, unless otherwise specified in the approved fabrication drawings and specifications consistent with the exception to the code described in Section [redacted]. License drawings are provided in Section 1.5. ASME code stamping of the canister is not required.

7.1.3.2 Welding Specifications

The canister [redacted] assembled using longitudinal [redacted] welded joints in the shell and circumferential welded joints at the bottom plate/shell juncture.

These welds are in accordance with ASME Code Section IX. The full penetration [redacted] weld joining the canister shell is radiographed in accordance with ASME Code Section V, Article 2. The weld joining the bottom plate to the canister shell is ultrasonically inspected in accordance with ASME Code Section V, Article 5. The acceptance criteria for these welds is as specified in ASME Code Section III, NB-5320 and NB-5330, respectively. [redacted]

After loading, the canister is closed by a shield lid and a structural lid using field installed bevel welds.

After the shield lid is welded in place, the canister is [redacted] pressure tested. Following draining, drying and inerting operations, the [redacted] are closed with port covers that are welded in place with bevel welds. The shield lid and port cover welds are liquid penetrant [redacted] in accordance with ASME Section V, Article 6. [redacted] Acceptance is in accordance with ASME [redacted] Section III, [redacted] NB-5350. The shield lid to canister shell weld is liquid penetrant [redacted] root and final passes in accordance with ASME [redacted] Section V, Article 6, and is pressure and leak tested to ensure leaktightness. The operating procedures describing the handling steps to seal the canister are presented in Chapter 8. The pressure and leak test procedures are described in Chapter 9.

A secondary, or redundant, confinement boundary is provided at the top end of the canister by a structural lid, which is placed over the shield lid. The structural lid is welded to the shell using a field-installed bevel weld. [redacted] NB-5350 (PT).

All welding procedures are written and qualified in accordance with Section IX of the ASME Code. Each welder and welding operator must be qualified in accordance with Section IX of the ASME Code.

The results of all weld examinations are recorded.

7.1.3.3 Testing, Inspection, and Examination

The following tests are performed to ensure satisfactory performance of the confinement vessel:

1. All components are visually examined for conformance with the fabrication drawings.
2. All welds that are directly visible are visually examined in accordance with the requirements of ASME Code Section V, Article 9.

3. The acceptance standards for visual examination of the canister [redacted] welded joints are as specified in ASME Code, Section III, [redacted] NB-4424 and NB-4427. Unacceptable weld defects are repaired in accordance with ASME Code Section III, Subarticle NB-4450, and visually re-examined.

4. Canister [redacted] welds designated to be examined by radiographic examination are examined in accordance with the requirements of Section V, Article 2 of the ASME Code. The minimum acceptance standards for radiographic examination are as specified in ASME Code Section III, [redacted] NB-5320. Welds designated for ultrasonic examination are examined in accordance with the requirements of Section V, Article 5, of the ASME Code. The [redacted] acceptance standards for ultrasonic examination are as specified in ASME Code Section III, [redacted] NB-5330. Unacceptable defects in the welds are repaired in accordance with ASME Code Section III, [redacted] NB-4450, and re-examined.

5. A written report of each weld [redacted] is prepared. At a minimum, the written report includes: identification of part, material, name and level of examiner, NDE procedure used and the findings or dispositions, if any.

6 All personnel performing nondestructive testing are qualified in accordance with American Society of Nondestructive Testing Recommended Practice No. SNT-TC-1A.

7. Individuals qualified for NDT Level I, NDT Level II, or NDT level III may perform nondestructive testing. Only Level II or Level III personnel may interpret the results of examination or make determination of the acceptability of examined parts.

[redacted] The vendor [redacted] completely assembles the canister [redacted] prior to shipping. The purpose of assembling the canister is to ensure that all items specified have been supplied and to test the fit of the shield lid assembly including drain tube and the structural lid[redacted]

9. A helium leak test is used to verify that the [redacted] wall [redacted] is leaktight. The containment vessel is pressurized to 22 psia through the either the drain port [redacted] to leak test the shield lid to canister shell weld [redacted]. The sensitivity of the helium leak test shall be at least [redacted], so as to demonstrate a leakage rate not greater than [redacted]. Any indication of a leak is an unacceptable condition and must be repaired.

7.1.4 Closure

The primary closure of the transportable storage canister consists of the welded shield lid and the two (2) welded port covers. There are no bolted closures or mechanical seals in the primary closure. A secondary, [redacted] closure is provided at the top end of the canister by the structural lid. The structural lid, when welded to the canister shell, fully encloses the shield lid and the [redacted] port covers. [redacted]

Figure 7.1-1 Transportable Storage Canister Primary and Secondary Confinement Boundaries

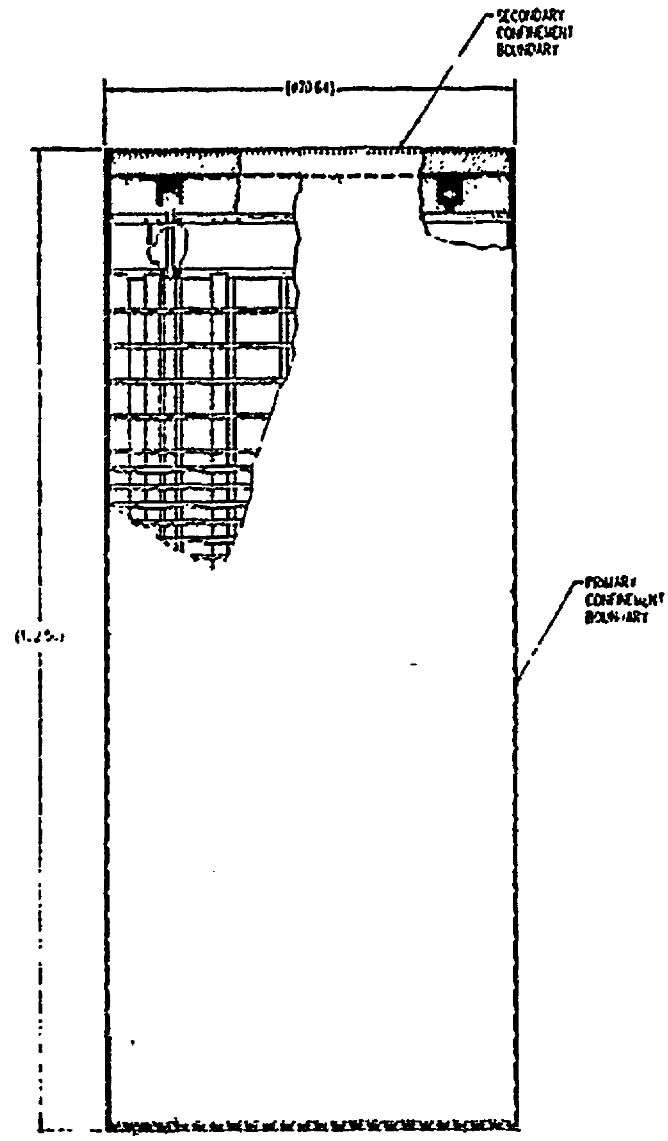

Figure 7.1-2 Confinement Boundary Detail at Shield Lid Penetration

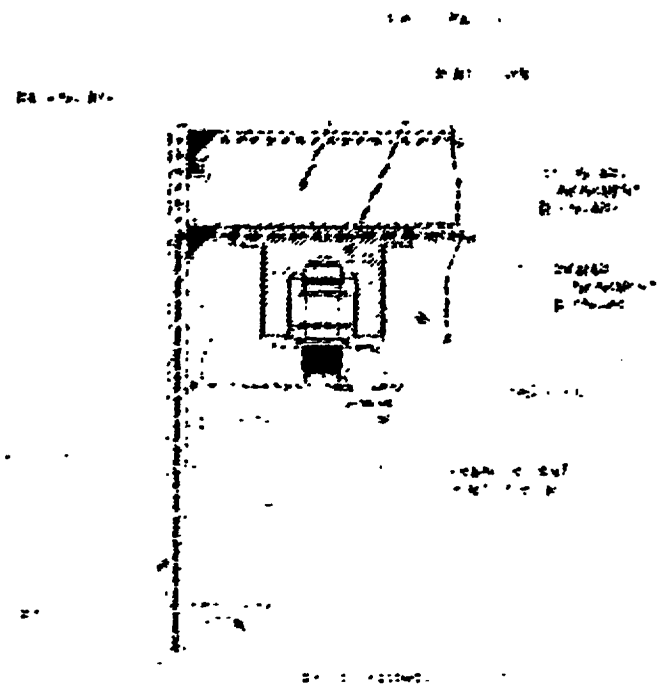

Table 7.1-1 Canister Confinement Boundary Welds

Confinement Boundary Welds		
MPC Weld Location	Weld Type	ASME Code Category (Section III, Subsection NB)
Shell longitudinal	Full penetration groove (shop weld)	A
Shell circumferential (if used)	Full penetration groove (shop weld)	B
[redacted] to shell	Full penetration groove (shop weld)	C
Shield lid to shell	Bevel (field weld)	C
Structural lid to shell	Bevel (field weld)	C
Vent and drain port covers to shield lid	Bevel (field weld)	C

7.2 Requirements for Normal Conditions of Storage

The canister is transferred to a vertical concrete storage cask using a transfer cask. During this transfer, the canister is subject to handling loads. The evaluation of the canister for normal handling loads is provided in Chapter 3.0. The principal design criteria for the NAC-MPC system are provided in Chapter 2.0.

Once the canister [redacted] is placed inside of the vertical concrete storage cask, it is effectively protected from direct loading due to natural phenomena, such as wind, snow and ice loading. The principal direct loading for normal operating conditions arises from increased internal pressure caused by decay heat, solar insolation, and ambient temperature. The normal operating internal pressure is evaluated in Chapter 3.0, as described in Section 7.2.2.

7.2.1 Release of Radioactive Material

The structural analysis of the canister for normal conditions of storage that is presented in Section 3.4.4 shows that the canister is not breached in any of the normal operating events. Consequently, there is no release of radioactive material during normal conditions of storage.

7.2.2 Pressurization of the Confinement Vessel

The canister is vacuum dried and backfilled with helium at one (1) atmosphere absolute prior to installing and welding the penetration port covers. In service, the internal pressure increases due to an increase in temperature of the helium and due to the postulated failure of fuel rod cladding [redacted]

The temperature of the helium increases due to the fuel decay heat, high ambient temperature, and the effect of full solar insolation on the concrete cask surface. Fuel cladding failure is assumed to release a fraction of the available fission gas and [redacted] radionuclides that are assumed to be [redacted] and all of the charge gas installed in the fuel rods at the time of manufacture of the rod. The evaluation conservatively adds the releasable inventory of the [redacted] fuel rods classified as failed to the inventory of the intact fuel that is assumed to fail in normal conditions [

The canister, shield lid, fittings, and the canister basket [redacted] are fabricated from materials that do not react with ordinary or borated spent fuel pool water to generate gases. The aluminum heat transfer disks, fuel tubes, and BORAL plates used for criticality control are protected by an oxide film that forms shortly after fabrication. This oxide layer effectively precludes further oxidation of the aluminum components or other reaction with water in the canister at temperatures less than 200°F, which is higher than the typical spent fuel pool water temperature. No steels requiring protective coatings or paints are used in the canister, shield lid, fittings, or basket. Therefore, there are no protective coatings or paints present that could interact with water to release gases.

The calculation of the canister pressure based on normal storage conditions is presented in Section 3.4.4.1.3 and is 11.5 psig. This pressure is well within the design basis internal pressure value of 50 psig. There are no adverse consequences, due to the internal pressure resulting from normal storage conditions.

Since the canister is vacuum dried and backfilled with helium prior to sealing, there are no significant moisture or gases, such as air, that remain in the canister. Consequently, there is no potential that radiolytic decomposition could cause an increase in canister internal pressure or result in a build-up of explosive gases in the canister.

7.3 Confinement Requirements for Hypothetical Accident Conditions

The evaluation of the canister for off-normal and accident condition loading is provided in Sections 3.5, 11.1 and 11.2, respectively.

Once the canister is placed inside the vertical concrete storage cask, it is effectively protected from direct loading due to natural phenomena, such as seismic events, flooding and tornado (wind driven) missiles. Accident conditions assume the cladding failure of all the fuel rods stored in the canister. Consequently, there is an increase in canister internal pressure due to the release of a fraction of the fission product and charge gases. The accident conditions internal pressure is 33 psig as calculated in Section 11.2.1.

For evaluation purposes, a class of accidents identified as off-normal is also considered in Section 11.1. This class of accidents is not considered here, since off-normal conditions are bounded by the hypothetical accident conditions

~~The structural analysis of the canister for off-normal and~~ [redacted] ~~presented in Chapter 11, shows that the canister is not~~ [redacted] ~~Consequently, based on a leaktight configuration, there~~ [redacted] ~~during off-normal or accident conditions of storage.~~

The resulting site boundary dose due to a hypothetical accident is, therefore, less than the 5 rem whole body ~~or organ (including skin)~~ dose at a ~~100 meter~~ minimum boundary required by 10CFR72.106 (b) for accident exposures

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8.0 OPERATING PROCEDURES

This chapter provides general guidance for using the NAC-MPC for storage operations. Three operating conditions are addressed. The first is loading the transportable storage canister [redacted], installing it in the vertical concrete cask [redacted], and transferring it to the storage (ISFSI) pad. The second is the removal of the loaded canister from the concrete cask. The third is opening the canister to remove spent fuel, in the unlikely event that this should be necessary.

The operating procedure for transferring a loaded canister from a concrete cask to the NAC-STC transport cask is described in Section 7.2.2 of the NAC-STC SAR.

Users are expected to develop site-specific procedures that incorporate the requirements presented here, consistent with the Operating Controls and Limits presented in Chapter 12. In addition, supplemental shielding may be employed to reduce radiation exposure for certain of the tasks specified by these procedures. Use of supplemental shielding is at the discretion of the user.

Operation of the NAC-MPC system requires the use of ancillary equipment items. The ancillary equipment supplied with the system is shown in Table 8.1-1. The system does not rely on the use of bolted closures, but bolts are used to secure retaining rings and lids. The hoist rings used for lifting the shield lid and canister have threaded fittings. Table 8.1-2 provides the torque values for installed bolts and hoist rings.

The design of the NAC-MPC is such that the potential for spread of contamination during handling and future transport of the canister is minimized. The concrete cask is constructed of new materials. The canister is loaded in the spent fuel pool, but is protected from gross contact with pool water by a jacket of clean water while it is in the transfer cask. Only the top of the open canister is exposed to contaminated pool water. The top of the canister is closed by the structural lid, which is not contaminated when it is installed. Consequently, the canister external surface is expected to be essentially clean.

When used in accordance with these procedures, the user dose is ALARA.

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8.1 Loading the NAC-MPC Storage System

The NAC-MPC storage system consists of three principal components: the transportable storage canister (canister), the transfer cask, and the vertical concrete cask ([redacted] cask). The transfer cask is used to hold the canister during loading and while the canister is being closed and sealed. The transfer cask is also used to transfer the canister to the [redacted] cask and to load the canister into the transport cask. The principal handling operations involve closing and sealing the canister by welding and [redacted] the [redacted] cask. [redacted]

This procedure assumes that the canister with an empty basket is installed in the transfer cask, that the transfer cask is positioned in the decontamination area or other suitable work station, and that the concrete cask is positioned on a heavy-haul transporter in the cask receiving area or other suitable staging area. The staging area should be within the handling "footprint" of the cask handling crane.

The operator must ensure that the fuel assemblies [redacted] to the requirements of Table 2.1-1 and the Certificate [redacted]

8.1.1 Loading and Closing the Transportable Storage Canister

1 Visually inspect the basket fuel tubes to ensure they are unobstructed and free of debris. Ensure that the welding zones on the canister, shield and structural lids, and the port covers are prepared for welding. Ensure transfer cask door lock bolts are installed and secure.

2 Flood the canister with clean water until the water is about 4 inches from the top of the canister.

Note: Do not fill the canister completely in order to avoid spilling water during the transfer to the spent fuel pool.

3. Attach a clean water line to the transfer cask.

4 If it is not already attached, attach the transfer cask lifting yoke to the cask handling crane, and engage the transfer cask lifting trunnions.

Note: The minimum temperature of the transfer cask (i.e., external ambient temperature) must be verified to be higher than 0°F prior to lifting. See Appendix 12A, Section 3.1.9

5 Raise the transfer cask and move it over the pool, following the prescribed travel path

6 Lower the transfer cask to the pool surface and turn on the clean water line to flood the annulus between the transfer cask and canister.

7. Lower the transfer cask as the annulus fills with clean water until the trunnions are at the surface and hold that position until clean water fills the remainder of the canister and overflows the sides of the transfer cask. Then lower the transfer cask to the bottom of the pool cask loading area.

   Note: If an intermediate shelf is used to avoid wetting the cask handling crane hook, follow the plant procedure for use of the extension piece.

8. Disengage the transfer cask lifting yoke to provide clear access to the canister.
9. Load the previously designated fuel assemblies into the canister.
10. Attach a three-legged sling to the shield lid using the swivel hoist rings.
11. Using the cask handling crane, or auxiliary hook, lower the shield lid until it rests in the top of the canister. Note the time that the shield lid is installed.

    Note: Ensure that the shield lid key slot aligns with the key welded to the canister shell.

    Caution: The operations from this point [redacted]

12. Raise the transfer cask until its top just clears the pool surface. Hold at that position, and using a suction pump, drain the pool water from above the shield lid. After the water is removed, continue to raise the cask.
13. As the cask is raised, spray the transfer cask outer surface with clean water to wash off any gross contamination.
14. When the cask is clear of the pool surface, but still over the pool, turn off the clean water flow to the annulus and allow the annulus water to drain to the pool. Move the cask to the decontamination area or other suitable work station.

    Note: Access to the top of the transfer cask is required. A suitable work platform may need to be erected.

15. Verify that the shield lid is level. Decontaminate the top of the transfer cask and shield lid as required to allow welding and inspection activities.

    Note: Supplemental shielding may be used for activities around the shield lid.

16. Insert the drain tube through the drain port of the shield lid into the basket drain tube sleeve. Torque the drain tube to 125 ± 5 ft-lbs. Install a mating quick-disconnect fitting in the vent line to open the vent. Remove the hoist rings.
17. Connect the suction pump to the drain port. Verify that the vent port is open. Remove approximately 50 gallons of water from the canister. Disconnect and remove the pump.
18. Install the semiautomated welding equipment.

19. Attach the [redacted]

20. Operate the welding equipment to complete the root weld joining the shield lid to the canister shell following approved procedures [redacted].

    [redacted]

[redacted] Prepare the weld and perform a liquid penetrant weld examination of the root pass. [redacted]

22 Complete welding of the shield lid to the canister wall and remove the weld equipment.

23. Prepare the weld and perform a liquid penetrant weld examination of the final pass. [redacted] the results of the weld examination.

24 Remove any lines attached to the drain port. Attach an air pressure line to the vent port. Pressurize the canister to 15 psig and hold the pressure. There must be no loss of pressure for 10 minutes. Ensure that the requirements of Technic[redacted]

25 Release the pressure and visually inspect the shield lid to canister shell weld for indications of leaks and defects. Record the results of the inspection.

26 Attach the suction pump to the drain line. Ensure that the vent line is open. Using the pump, remove the remaining free water from the canister cavity.

    Note: Steps 26 through 35 must be completed within 10 hours.

27. Remove any free water in the drain port cavity. Install the drain port cover.

    Note: If previously removed, reinstall the hydrogen gas detector to the vent port. Operate the detector to verify that the concentration of hydrogen gas is below 2.4%. If not, use the vacuum system to clear hydrogen gas from the cavity and the drain line.

28 Weld the drain port cover to the shield lid

29 Prepare the weld and perform a liquid penetrant examination of the drain port cover weld root and final passes. Record the results of the weld examination.

30 Attach the vacuum equipment to the vent port line

31 Operate the vacuum equipment, until a vacuum of 3 mm of mercury exists in the canister in accordance with the requirements of Technical Specification LCO 3.1.2.

32 Verify that no water remains in the canister by holding the vacuum for 30 minutes If water is present in the cavity, the pressure will rise as the water vaporizes Continue the vacuum hold cycle until there is no indicated rise in pressure after 30 minutes.

33. Backfill the canister cavity with helium [redacted]
34. Restart the vacuum equipment and evacuate the canister to 3[redacted] mm of mercury [redacted]
35. Backfill the canister cavity with helium, [redacted] pressurizing it to 22 psia [redacted]
36. Using a helium leak detector, verify [redacted] at the shield lid [redacted]

    Note: [redacted] within 36 hours.
37. Vent the canister helium pressure to one (1) atmosphere [redacted]
38. Remove any attachments to the vent port fitting. Dry any residual water that may be present in the [redacted] port cavity
39. Install the vent port cover and weld the vent port cover to the shield lid.
40. Prepare the weld and perform a liquid penetrant examination of the [redacted] port cover weld [redacted] and final passes. Record the results of the weld [redacted]
41. Remove any supplemental shielding used during shield lid closure activities.
42. Attach a three-legged sling to the structural lid using the swivel hoist rings.

    Note: Verify that the structural lid is stamped, or otherwise marked, to provide traceability of the canister contents. Verify that the structural lid weld backing ring is in place on the structural lid.
43. Using the cask handling or the auxiliary crane, install the structural lid in the top of the canister. Verify that the structural lid does not protrude above the canister shell and is approximately centered in the canister shell. Verify that the gap in the backing ring is not aligned with the shield lid alignment key. Remove the lifting slings and the hoist rings.
44. Install the automated welding equipment on the structural lid.
45. Complete the root weld pass joining the structural lid to the canister shell.
46. Prepare the weld and perform a liquid penetrant examination of the weld root pass and record the results of the weld examination.
47. Complete the remainder of the weld, performing NDE (progressive, liquid penetrant or ultrasonic testing) examination. Record the results of each weld examination.
48. Remove the welding equipment

[redacted] Prepare the weld and [redacted] perform a liquid penetrant examination of the final [redacted] pass. [redacted]

[redacted]

51. Install the transfer cask retaining ring.

[redacted]

8.1.2 Loading the Vertical Concrete Cask

This section of the loading procedure assumes that the [redacted] concrete [redacted] cask [redacted] is located on the bed of a heavy-haul trailer under the cask handling crane and that the [redacted] cask shield plug and lid are not in place.

1. Using a suitable crane, place the transfer adapter on the top of the [redacted] cask.
2. Using the transfer adapter bolt hole pattern, align the adapter to the [redacted] cask. Bolt the adapter to the cask using four (4) socket-head cap screws.
3. Verify that the bottom door connectors on the adapter plate are in the fully extended position.
4. If not already done, attach the transfer cask lifting yoke to the cask handling crane. Verify that the transfer cask retaining ring is installed.
5. Install six (6) swivel hoist rings in the structural lid of the canister. Verify that the hoist ring threads are fully engaged, and attach two (2) three-legged slings. Stack the slings on the top of the canister so they are available for use in lowering the canister into the concrete cask.
6. Engage the transfer cask trunnions with the transfer cask lifting yoke. Ensure that all lines are disconnected from the transfer cask.
7. Raise the transfer cask and move it over the concrete cask. Lower the transfer cask, ensuring that the transfer cask bottom door rails and connector tees align with the adapter plate rails and door connectors. Prior to final set down, remove transfer cask door lock bolts.

   Note: The minimum temperature of the transfer cask must be verified to be higher than 0°F (i.e., external ambient temperature) prior to lifting in accordance with Technical Specification LCO 3.1.9.
8. Ensure that the bottom door connector tees are engaged with the adapter plate door connectors.

9. Disengage the transfer cask yoke from the transfer cask and from the cask handling crane hook.
10. Return the cask handling crane hook to the top of the transfer cask and engage the two (2) three-legged slings attached to the canister by attaching the master [redacted] to the crane hook. Lift the canister slightly (about ½ inch) to take the canister weight off of the transfer cask bottom doors.

    Note: A load cell may be used to determine when the canister is supported by the crane. Avoid raising the canister to the point that the structural lid engages the transfer cask retaining ring, as this could result in lifting the transfer cask.

    [redacted]
11. Using the hydraulic system, open the bottom doors to access the concrete cask cavity.
12. Lower the canister into the concrete cask, using a slow crane speed as the canister nears the bottom of the [redacted] cask.
13. Disconnect the slings from the canister and close the transfer cask bottom doors.
14. Retrieve the transfer cask lifting yoke and attach the yoke to the transfer cask.
15. Lift the transfer cask off the concrete cask and return it to the decontamination area or designated work station.
16. Using the auxiliary crane, remove the adapter plate from the top of the concrete cask.
17. Remove the swivel hoist rings from the structural lid and replace them with bolts.
18. Using the auxiliary crane, retrieve the shield plug and install the shield plug in the top of the concrete cask.
19. Using the auxiliary crane, retrieve the concrete cask lid and install the lid in the top of the concrete cask using six stainless steel bolts.
20. Ensure that there is no foreign material left at the top of the concrete cask. Install the tamper-indicating seal.

8.1.3 Transporting the Vertical Concrete Cask

This section of the procedure assumes that the loaded concrete cask is positioned on a heavy-haul trailer.

1. Using a suitable towing vehicle, tow the heavy haul trailer to the dry storage pad (ISFSI). Verify that the bed of the trailer is approximately at the same height as the pad surface.
2. Install four (4) hydraulic jacks at the four (4) designated jacking points at the bottom cooling air vents.

3. Raise the concrete cask approximately 3 inches.

   [redacted]

4. Move the air-bearing rig set under the cask.

   Note: A hydraulic skid may also be used to move the [redacted] cask. The height the [redacted] cask is raised depends upon the height of the skid or air pad set used, [redacted]

5. Inflate the air-bearing rig set. Remove the four (4) hydraulic jacks.
6. Using a suitable towing vehicle, move the [redacted] cask from the bed of the transporter to the designated location on the storage pad.
7. Turn off the air-bearing rig set, allowing it to deflate.
8. Reinstall the four (4) hydraulic jacks and raise the concrete cask approximately 3 inches.

   [redacted]

9. Remove the air-bearing rig set pads. Ensure that the surface [redacted] under the cask is free of foreign objects.
10. Lower the concrete cask to the surface.

    Note: Ensure that the spacing between concrete c[illegible]

11. Remove the four (4) hydraulic jacks.
12. Install screens in the inlets and outlets [illegible].
13. Install/connect the temperature monitoring equipment.
14. Scribe/stamp the concrete cask name plate to indicate loading.
15. Verify that the concrete cask surface dose [illegible] requirements of Technical Specification LCO 3.2.5 [illegible] exceed 50 mrem per hour on the sides and 35 mrem per hour [illegible] measured at the inlets and outlets shall be less than 100 mrem per hour [illegible] that is the extension of the external surface.).

Figure 8.1-1 Vent and Drain Port Locations

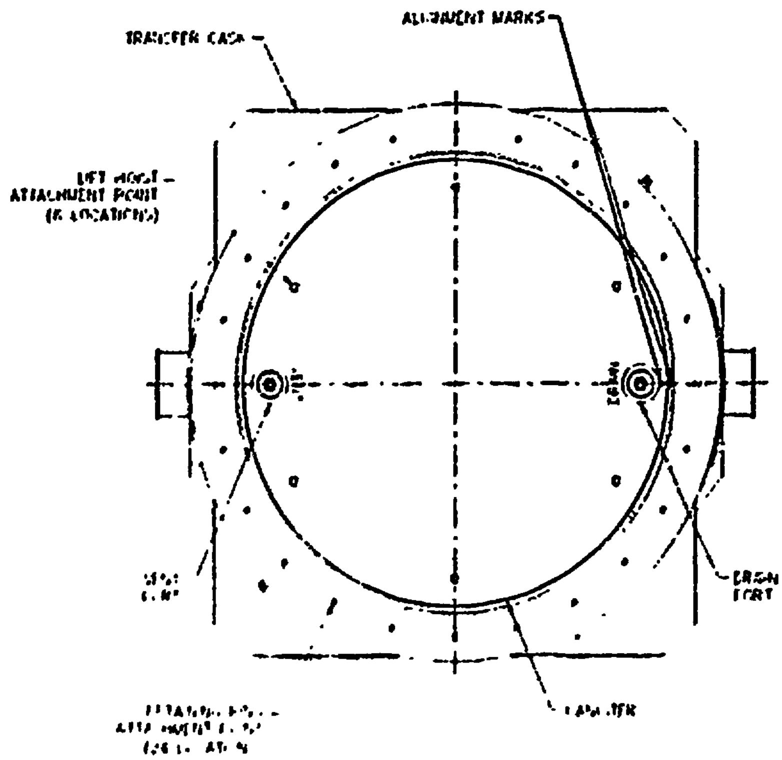

Table 8.1-1 List of Ancillary Equipment

Item	Description
Transfer Cask Lifting Yoke	Required for lifting and moving the transfer cask.
Transport Trailer (Optional)	Heavy-haul (double drop frame) trailer required for moving the loaded and empty concrete [redacted] cask to and from the ISFSI pad.
Helium Supply System	Supplies helium to the canister for helium backfill and purging operations.
Vacuum Drying System	Used for evacuating the canister. Used to remove residual water, air and initial helium backfill.
Automated Welding System	Used for welding the shield lid and structural lid to the canister shell.
Self-Priming Pump	Used to remove water from the canister.
Shield Lid Sling	A three-legged sling used for lifting the shield lid. It is also used to lift the [redacted] cask shield plug and lid.
Canister Sling	A set of 2 three-legged slings joined by a master link, used for lifting the structural lid by itself, or for lifting the canister when the structural lid is welded to it. The master link allows the slings to be loaded simultaneously during the lift.
Transfer Adapter	Used to align the transfer cask to the [redacted] cask or transport cask. Provides the platform for the operation of the transfer cask bottom doors
Hydraulic Unit	Operates the bottom doors of the transfer cask.
Lift Pump Unit	Jacking system for raising and lowering the concrete cask.
Air Pad Rig Set	Air cushion system used for moving the concrete cask.

Table 8.1-2 Torque Values

Fastener	Torque Value (ft-lbs)	Torque Pattern
Transfer Adapter Bolts	40 ± 5	None
Transfer Cask Retaining Ring Bolts	100 ± 10	None
[redacted] Cask Lid Bolts	40 ± 5	None
Lifting Hoist Rings[redacted]		
[redacted] Shield Lid	800 + 80, - 0	None
[redacted] Structural Lid	800 + 80, - 0	None
[redacted] Cask Lid	800 + 80, - 0	None
[redacted]ly engaged		
Canister and Lid Plug Bolts	Hand Tight	None
Transfer Cask Door Lock Bolts	Hand Tight	None
[redacted]	[redacted]	[redacted]

8.2 Removal of the [redacted] Canister from the [redacted] Concrete [redacted] Cask

Removal of the loaded canister from the concrete [redacted] cask is expected to occur at the time of shipment of the canistered fuel [redacted] off site. Alternately, removal could be required in the unlikely event of an accident condition that rendered the [redacted] cask or canister unsuitable for continued long-term storage or for transport. This procedure identifies the general steps to return the loaded canister to the transfer cask and return the transfer cask to the decontamination station, or other designated work area. Since these steps are the reverse of those undertaken to place the canister in the concrete cask, as described in Section 8.1.2, they are summarized here.

At the option of the user, the canister may be removed from the [redacted] cask and transferred to another concrete cask or to the NAC-STC transport cask at the ISFSI site. This transfer is done using the transfer cask, which provides shielding for the canister contents during the transfer.

1. Using the hydraulic jacking system and the air pad set, move the concrete cask from the ISFSI pad to the heavy-haul trailer. The bed of the trailer must be approximately level with the surface of the pad.

   Caution: Do not exceed a maximum lift height of 6 inches when lifting the concrete cask to install the air pad set in accordance with the requirements of Technical Specification LCO 3.1.8.

2. Tow the transporter to the cask receiving area or other designated work station.
3. Remove the concrete cask shield plug and lid. Install the hoist rings in the canister structural lid. Verify that the hoist ring threads are fully engaged and attach the lift slings. Install the transfer adapter.
4. Retrieve the transfer cask and position it on the transfer adapter on the top of the concrete cask.

   Note: The minimum temperature of the transfer cask must be verified to be higher than 0°F (i.e., external ambient temperature) prior to lifting in accordance with Technical Specification LCO 3.1.9.

5. Open the shield doors. Attach the canister lift slings to the cask handling crane hook.

   Caution: The three-legged sling master links must be at least 67 inches above the canister lid. (Refer to Technical Specifications, Appendix 12A, Section 4.5.4)

6. Raise the canister into the transfer cask. Use caution to avoid contacting the transfer cask retaining ring with the canister.

7. Close the shield doors. Lower the canister to rest on the bottom doors. Disconnect the canister slings from the crane hook.
8. Retrieve the transfer cask lifting yoke. Engage the transfer cask trunnions and move the transfer cask to the decontamination area or designated work station.
   Note: Prior to moving transfer cask, install and secure door lock bolts.

After the transfer cask containing the canister is in the decontamination area or other suitable work station, additional operations may be performed on the canister. It may be opened, transferred to another [redacted] cask, or placed in the NAC-STC transport cask.

8.3 Unloading the Transportable Storage Canister

Circumstances could arise that dictate the opening of a previously loaded canister and the removal of the stored spent fuel [illegible]. This section describes the basic operations needed to open the sealed canister. It is assumed that the canister is positioned in the transfer cask and that the transfer cask is in the decontamination station or other suitable work station. The principal mechanical operations are the cutting of the closure welds, filling with water, and the removal of the spent fuel [illegible]. Supplemental shielding is used as required.

1. Remove the transfer cask retaining ring.
2. Survey the top of the canister to establish the radiation level and contamination level at the structural lid.
3. Set up the weld cutting equipment to cut the structural lid weld (Abrasive grinding, hydrolaser, or similar cutting equipment.).
4. Tent the top of the transfer cask as required.
5. Operate the cutting equipment to cut the structural lid weld.
   Note: Monitor for any out-gassing. ~~Wear respiratory protection as required.~~
6. Remove the cutting equipment and attach a three-legged sling to the structural lid.
7. Using the auxiliary crane, lift the structural lid off of the canister and out of the transfer cask.
8. Survey the top of the shield lid to determine radiation and contamination levels. Use supplemental shielding as necessary. Decontaminate the top of the shield lid, if necessary.
9. Tent the top of the transfer cask, if required. Using an abrasive grinder ~~and wearing suitable respiratory protection,~~ cut the welds joining the vent and drain port covers to the shield lid.
10. Remove the port covers. Monitor for any out-gassing and survey the radiation level at the quick-disconnect fittings. Attach a manually valved line with a vacuum bottle to the vent port quick-disconnect. Open the valve to the vacuum bottle to obtain a gas sample from the vent line. Analyze the gas sample to determine the make up of the canister atmosphere.
    Caution: The canister could be pressurized.
11. Attach a nitrogen gas line to the drain port quick-disconnect and a discharge line from the vent port quick-disconnect to an off-gas handling system. Monitor the vent line, so that the radiation level of the discharge gas and the temperature of the discharge gas are indicated.
    Note: Any significant radiation level in the discharge gas indicates the presence of fission gas products. The temperature of the gas indicates the thermal conditions in the canister.
    Caution: Discharge gas temperature could initially be above 400°F.

12. Continue to flow nitrogen through the line, until there is no evidence of fission gas activity in the discharge line. Continue to monitor the gas discharge temperature. When there is no additional evidence of fission gas, stop the nitrogen flow and disconnect the drain and vent port line connections.

    (Caution: the discharge line and fittings may be very hot.)

13. Attach a s ce of clean water [redacted] to the drain port quick-disconnect. Attach a discharge line to the vent port quick-disconnect. Start the flow of clean water [redacted]. The discharge line will initially discharge hot gas, but after the canister fills, it will discharge hot water. [redacted] (Caution: Relatively cool water may flash to steam as it encounters hot surfaces within the canister. If there are grossly failed or ruptured fuel rods within the canister, very high levels of radiation could rapidly appear at the discharge line. The radiation level of the discharge gas or water should be continuously monitored.)

    Note: The fuel cooldown procedure must conform [redacted] Specification LCO 3.1.6.

14. Continue to flow water through the canister until the exit water temperature stabilizes, [redacted] stop the flow of water.
15. Connect a suction pump to the drain port and [redacted] Disconnect and remove the pump.
16. Attach a hydrogen gas detector to the vent port. [redacted] gas is less than 2.4%.
17. Set up the weld cutting equipment to cut the shield lid weld (Abrasive grinding, hydrolaser, or similar cutting equipment.). Route the vent line to avoid interference with the weld cutting operation.
18. Operate the cutting equipment to cut the shield lid weld.

    Note: Stop the cutting operation if the hydrogen gas detector indicates a concentration of hydrogen gas above 2.4%. Clear the gas before proceeding with the cutting operation.

19. Remove the cutting equipment. Install the shield lid lifting hoist rings and attach a three-legged sling. Attach a line to the sling master link to aid in attaching the sling to the crane hook.
20. Attach the clean water line to the transfer cask.

[redacted] Retrieve the transfer cask lifting yoke and engage the transfer cask lifting trunnions.

[redacted]

[redacted] Move the transfer cask over the pool and lower the bottom of the transfer cask to the surface. Start the flow of clean water to the transfer cask annulus. Continue to lower the transfer cask, as the annulus fills with clean water, until the top of the transfer cask is about 4 inches above the pool surface. Hold this position until clean water fills the top of the transfer cask.

[redacted] Lower the transfer cask to the bottom of the cask loading area and remove the lifting yoke.

[redacted] Attach the shield lid lifting sling to the crane hook.

[redacted] Slowly lift the shield lid. Move the shield lid to one side after it is raised clear of the transfer cask (Caution: The drain line tube is suspended from the under side of the shield lid. The lid should be raised as straight as possible until the tube clears the canister basket. [redacted] the shield lid is removed from the pool. The under side of the shield lid [redacted] drain line could be highly contaminated.).

26. Visually inspect the fuel for damage.

At this point, the spent fuel could be transferred from the canister to the fuel racks. If the fuel is damaged, special rigging could be required to remove the fuel. In addition, the bottom of the canister could be highly contaminated. Care must be exercised in the handling of the transfer cask when it is removed from the pool. Highly radioactive particles could rest on flat surfaces of the transfer cask resulting in high dose rates.

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9.2 Maintenance Program

The NAC-MPC storage system is a passive system. There are no active components or systems incorporated in the design. Consequently, there is a minimal amount of maintenance that is required over its lifetime.

The system has no valves, gaskets, rupture discs or seals, and there are no accessible penetrations. Consequently, there is no maintenance associated with these types of features.

9.2.1 Continuing Maintenance Requirements

Recommended maintenance in normal conditions

1 Daily surveillance of the storage casks

Visual inspection of air vents for detection of blockage.

Verify that "outlet screens" [illegible] place, whole and secure.

Measure and record the ambient temperature and air outlet temperature for each vertical concrete cask upon placement in service. Thereafter, the temperatures shall be recorded on a daily basis to verify the continuing thermal performance of the system. [redacted].

Visual inspection of the ISFSI site for security and safeguards

2 Annual inspection of the storage cask exterior

Visual inspection of surfaces for chipping, spalling or other surface defects. If found, a defect should be corrected by regrouting the affected area. Refer to [redacted] for surface defect limits and required actions.

Re-application of corrosion inhibiting [illegible] coatings on accessible surfaces

It is not necessary to inspect the canister during the storage period as long as normal conditions exist.

9.2.2 Required Maintenance of First Storage System Placed in Service

[redacted]

[redacted] to the NRC for Canisters that are subsequently [redacted] case.

9.1.3 Leak Tests

The canister [redacted] leak tested at the time of use. After the pressure test described in Section 9.1.2, the canister is drained of residual water, vacuum dried and backfilled with helium. The canister [redacted] is pressurized with helium to 22 psia. The shield lid [redacted] to canister shell weld is helium leak tested. The leak test is performed at a sensitivity of at least 4.0 x 10^{4} cm^3/sec (helium). Any indication of a leak is unacceptable and repair of the leak is required.

9.1.4 Component Tests

The components of the NAC-MPC do not require any special tests in addition to the material receipt, dimensional, and form and fit tests described above, or as described below.

9.1.4.1 Valves, Rupture Disks and Fluid Transport Devices

The NAC-MPC canister and storage cask do not contain rupture disks or fluid transport devices. There are no valves that are part of the confinement boundary for transport or storage. Quick-disconnect valves are installed in the canister vent and drain ports of the shield lid. [redacted] These valves are intended to be convenience items for the operator, as they provide a means of quickly connecting (or disconnecting) ancillary drain and vent lines to the canister [redacted]. The quick-disconnect fittings consist of male and female halves. The male fitting is installed in the canister and the female fitting is used as the connecting piece. The male fitting is automatically closes when the mating fitting is removed, however, no credit is taken for this sealing feature. During storage and transport, these fittings are not accessible, as port covers that are welded in place cover them when the canister is closed. As presented for storage, the canister has no accessible valves or fittings.

9.1.4.2 Gaskets

The NAC-MPC canister and concrete cask have no mechanical seals or gaskets that form an integral part of the package, and there are no mechanical seals or gaskets in the confinement boundary.

9.1.5 Shielding Tests

Based on the conservative design of the NAC-MPC storage cask for shielding criteria and the detailed construction requirements, no shielding tests of the concrete storage cask are required.

9.1.6 Neutron-Absorber Tests

After manufacturing, a statistical sample of each lot of BORAL panels is tested using wet chemistry and/or neutron attenuation techniques to verify a minimum ^{10}B content at the ends of the panels. Any panel in which B^{10} loading is less than the specified minimum is rejected.

9.1.7 Thermal Tests

No thermal acceptance testing of the NAC-MPC system is required during construction. Temperature measurements are taken at the air outlets of the storage cask during operation in accordance with Chapter 12.0 as verification of the thermal performance of the storage system.

9.1.8 Cask Identification

A stamped stainless steel nameplate, as shown on Drawing No. 455-856, is permanently attached on the outer surface of the storage cask. The nameplate includes the following information:

Vertical Concrete Cask

Owner	(Utility Name)
Designer	NAC International Inc.
Fabricator:	(Vendor Name)
Date of Manufacture	(mm/dd/yy)
Model Number	(MPC-TR)
Cask No	(XXX)
Date of Loading	(mm/dd/yy)
Empty Weight	(Pounds [kilograms])

[redacted]

The basket assembly welds are [redacted] examined [redacted] ASME Code Section V, Article 6. The acceptance criteria are in accordance with ASME Code Section III, [redacted] NG-5350.

All welding of [redacted] components is performed using procedures and welders qualified in accordance with the ASME Code Section IX. [redacted]

9.1.1.2 Fabrication Inspections

Materials used in the fabrication of the NAC-MPC storage cask [redacted] canister [redacted] are procured with certifications and supporting documentation, as necessary, to assure compliance with procurement specifications. All materials are receipt inspected for appropriate acceptance requirements and for traceability to required material certification.

The canister [redacted] fabricated to the requirements of ASME Code Section III, Subsection NB. Specific exceptions to the ASME Code are described in Chapter 2 and 7. The basket assembly is fabricated to ASME Code Section III, Subsection NG. Shop fabricated components of the storage cask are fabricated in accordance with ANSI AWS D1.1-96 [redacted]

A complete dimensional inspection of all critical components and a components fit-up test is performed on the canister [redacted] assembly to ensure proper assembly in the field. Acceptance criteria for dimensions shall conform to the fabrication drawings.

Concrete strength and density shall be field verified to American Concrete Institute (ACI) and American Society for Testing and Materials (ASTM) standards to ensure adequacy. Reinforcing steel is installed per specification requirements based on ACI-318.

On completion of fabrication, the canister, [redacted] basket, and other shop fabricated components shall be inspected for cleanliness. All components shall be free of any foreign material, oil, grease and solvents. Carbon steel components assembled for the storage cask shall be coated with a corrosion-resistant paint.

9.1.2 Structural and Pressure Test

The canister is pressure tested at the time of use. After loading of the canister basket with spent fuel [redacted], the shield lid is welded in place after approximately 50 gallons of water are removed from the canister. Prior to removing the remaining spent fuel pool water from the canister, the canister is pressure tested at [redacted] This pressure is held for 10 minutes. Any loss of pressure during the test period is unacceptable and the leak must be located and repaired. The pressure test is described in Section 8.1.

[redacted]

9.1 Acceptance Tests

The acceptance tests ensure that the storage cask [redacted] canister, [redacted] are fabricated, assembled, inspected and tested in accordance with the requirements of this SAR and the license drawings.

9.1.1 Visual and Nondestructive Examination Inspections

The acceptance test program establishes a set of visual inspections and nondestructive examination or test requirements for the fabrication and assembly of the storage cask [redacted] canister [redacted]. Satisfactory results for these inspections, examinations and tests demonstrate that the components comply with the requirements of the SAR and the license drawings, and [redacted] initial operation of the storage system complies with regulatory requirements.

A fit-up test of the canister and its components is performed during the acceptance inspection. The fit-up test demonstrates that the canister, basket, shield lid and structural lid can be properly assembled during fuel loading and canister closure operations.

[redacted]

A visual inspection is performed on all materials and welds used for storage cask, canister [redacted] and basket fabrication. The visual inspection applies to finished surfaces of the components. All welds (shop and field installed) are visually inspected for defects prior to the nondestructive examinations that are specified. The welding of the canister [redacted] is performed in accordance with ASME Code, Section III, Subsection NB-4000, except as [redacted].

The visual inspections of the canister [redacted] welds are performed in accordance with the ASME Code Section V, Article 9. Acceptance criteria for the visual examinations of the canister [redacted] welds are in accordance with ASME Code [redacted]

Welding of the storage cask's steel components, including field installed welds, is performed in accordance with [redacted] ANSI/AWS D1.1-96 and is inspected in accordance with ANSI/AWS D1.1, Section 8.15 [redacted]

[redacted]. Weld procedures and welder qualifications shall be in accordance with ANSI/AWS D1.1, Section 5, or ASME Code Section IX.

Welding of the basket assembly for spent fuel [redacted] is performed in accordance with ASME Code Section III [redacted] NG-4000. Visual examination of the welds is performed per the requirements of ASME Code Section V, Article 9. Acceptance criteria for the visual examination of the basket assembly welds are that of ASME Code Section [redacted] Any required weld repairs are performed in accordance with ASME Code Section III, [redacted] NG-4450 and are reexamined in accordance with the original acceptance criteria.

[redacted]

9.1.1.1 Nondestructive Weld Examination

All of the welds of the canister assembly are nondestructively examined in addition to the visual examination previously discussed. In [redacted] ASME Code Section III, Subsection NB requirements for confinement vessels, the canister [redacted] shell welds are volumetrically examined by radiography (RT) [redacted] ASME Code Section V, Article 2, with acceptance criteria in accordance with ASME Code Section III, [redacted] NB-5320. The weld that joins the bottom plate to the canister shell is ultrasonically (UT) examined per ASME Code Section V, Article 5, with acceptance criteria in accordance with ASME Code Section III, NB-5330. The finished surface of the [redacted] liquid penetrant examined in accordance with ASME [redacted] in accordance with ASME Code Section III, NB-5350. The shield lid to canister shell weld and the structural lid to shell weld, as well as the vent and drain port [redacted] to shield lid welds, are field welds that are performed after the canister is loaded. [redacted] The root and final passes of the shield lid to canister shell weld are liquid penetrant (PT) examined per ASME Code, Section V, Article 6. The acceptance criteria are in accordance with ASME Code, Section III, [redacted] NB-5350. The canister vent port cover and drain port [redacted] penetrant examined, i.e., root and final surfaces, in [redacted] Article 6. Acceptance criteria are specified in ASME [redacted] structural lid to canister shell weld is either: 1) [redacted] with ASME Code Section V, Article 5, with the final [redacted] accordance with ASME Code Section V, Article 6 or 2) [redacted]

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

This chapter specifies the acceptance criteria and the maintenance program for the NAC-MPC storage system primary components: vertical concrete cask (storage cask), transportable storage canister (canister). The design of the NAC-MPC system requires shop fabrication of the canister shell with the bottom plate, the shield and structural lids for the canister, and the basket that holds the spent fuel. The storage cask consists of reinforced concrete placed around steel components that are integral to the performance of the storage cask. These steel components include: a liner that forms the central cavity of the storage cask, a set of air outlet passage-ways that allow cooling to the stored canister, a shield plug, a steel closure lid, and a steel base. The base includes: the air inlets and associated pathways, provides a pedestal upon which the canister rests, and provides a structural support for raising the storage cask. The steel components are shop fabricated. The reinforcing steel will be bent in the shop and delivered to the storage cask construction site. The storage cask construction will include the erection of the cask liner onto the steel base. The concrete is placed around the liner after the reinforcing steel has been properly erected.

As described in Chapter 8, the storage cask is intended to be lifted by hydraulic jacks and moved using air pads under the base. It does not have lifting trunnions.

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10.0 RADIATION PROTECTION

10.1 Ensuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable (ALARA)

The NAC-MPC provides radiation protection for all areas and systems that may expose personnel to radiation or radioactive materials. The components of the NAC-MPC system that require operation, maintenance and inspection are designed, fabricated, located, and shielded to minimize radiation exposure to personnel.

10.1.1 Policy Considerations

It is the policy of NAC to ensure that the NAC-MPC system is designed, so operation, inspection, repair and maintenance can be carried out while maintaining occupational exposure as low as reasonably achievable (ALARA).

10.1.2 Design Considerations

The design of the NAC-MPC system complies with the requirement of 10 CFR 72.3 concerning ALARA and meets the requirements of 10 CFR 72.126(a) and 10 CFR 20.1101 with regard to maintaining occupational radiation exposures ALARA. Specific design features that demonstrate the ALARA philosophy are:

- Material selection and surface preparation that facilitate decontamination.
- A basket configuration that allows spent fuel loading using accepted standard practice and current experience.
- Positive clean water flow in the transfer cask/canister annulus to minimize the potential for contamination of the canister surface during in-pool loading.
- Passive confinement, thermal, criticality, and shielding systems that require no maintenance.
- Thick steel and concrete walls to reduce the side surface dose rate to 40 mrem/hr (average).

- Nonplanar cooling air pathways to minimize radiation streaming at the inlets and outlets of the concrete cask.

- Use of remote, automated outlet air temperature measurement to reduce surveillance time.

10.1.3 Operational Considerations

The ALARA philosophy has been incorporated into the procedural steps necessary to operate the NAC-MPC in accordance with its design. The following features or actions, which comprise a baseline radiological controls approach, have been incorporated in the design or procedures to minimize occupational radiation exposure:

- Use of prefabricated, shaped temporary shielding during automated welding equipment set up and removal, manual welding, and weld inspection of the shielding and structural lids and for use during all of the canister closing and sealing operations.

- Use of automatic equipment for welding the shield lid and structural lid to the canister shell.

- Decontamination of the exterior surface of the transfer cask, welding of the shield lid, and pressure testing of the canister while the canister remains filled with water.

- Use of quick disconnect fittings at penetrations to facilitate required service connections.

- Use of remote handling equipment, where practical, to reduce radiation exposure.

The operational procedures at a particular facility will be determined by the user's operational conditions and facilities.

10.2 Radiation Protection Design Features

The description of the radiation shielding design is provided in Chapter 5.0. The design basis radiation exposure rates are summarized in this section and in Chapter 2.0. The principal radiation protection design features are the shielding necessary to meet the design objectives, the placement of penetrations near the edge of the canister shield lid to reduce operator exposure and handling time, and the use of shaped supplemental shielding for work on and around the shield and structural lids. This supplemental shielding reduces operator dose rates during the welding, inspection, draining, drying and backfilling operations that seal the canister.

Radiation exposure rates at various work locations were determined for the principal NAC-MPC operational steps. These exposure rates were determined using a combination of the SAS1 and SKYSHINE III computer codes. The use of SAS1 is described in Chapter 5.0. The SKYSHINE-III code is discussed in Section 10.4. The calculated dose rates decrease with time.

10.2.1 Design Basis for Normal Storage Conditions

The radiation protection design basis for the NAC-MPC storage cask is derived from 10 CFR 72 and the applicable ALARA guidelines. The design basis surface dose rates and the calculated 1 meter dose rates are shown below. The calculated dose rates at these, and at other dose points, are also reported in Chapter 5.0, "Shielding Evaluation."

Concrete Storage Cask	Design Basis Maximum Surface Dose Rate (mrem/hr)	1 Meter Maximum Dose Rate (mrem/hr)
Side wall	50.0	20.0
Air inlet/air outlet	100.0	5.0
Top lid	55.0	15.0

Activities associated with closing the canister, including welding of the shield and structural lids, draining, drying, backfilling and testing, will employ temporary shielding to minimize personnel dose in the performance of those tasks.

10.2.2 Design Basis for Accident Conditions

Damage to the NAC-MPC cask after a design basis accident will not result in a radiation exposure at the controlled area boundary in excess of 5 rem to the whole body or any organ, [redacted]. The high energy missile impact is estimated to reduce the concrete shielding thickness, locally at the point of impact, by 6 inches. This reduction in shielding results in a calculated dose rate of 120 mrem/hr at one meter. There are no other design basis accident conditions that result in a greater estimated loss of shielding.

Two hypothetical accident events that evaluate storage cask tip over and the rupture of 100% of the fuel rods | are considered in Chapter 11. There are no design basis events that result in the tip over of the NAC-MPC storage cask or the release of any radioactive material from the canister.

10.3 Estimated On-Site Collective Dose Assessment

Occupational radiation exposures (person-mrem) resulting from the use of the NAC-MPC storage system are calculated using estimated exposure rates presented in Chapter 5.0 and Section 10.2.1. Exposure was evaluated by identifying the tasks and estimating the duration and number of personnel performing those tasks based on industry experience. The tasks identified were based on the design basis operating procedures, as presented in Chapter 8.0.

Dose rates were initially estimated based on the design basis fuel assembly for shielding, the Combustion Engineering Type A [redacted] Since use of these dose rates over predict the total dose for the use of the NAC-MPC system for a 16 cask storage array, the dose rates are adjusted to account for fuel cooling time representative of an ISFSI. The effect of the adjustment is to reduce the maximum estimated total dose for loading each canister by about 20%. This adjustment is described in Section 10.3.2. It is also applied in the calculation of the ISFSI boundary dose rates.

10.3.1 Estimated Collective Dose for Loading a Single NAC-MPC

This section estimates the collective dose due to the loading, sealing, transfer and placement of a single NAC-MPC containing design basis fuel. This analysis assumes that the exposure incurred by the operators is independent of background radiation, as background will vary with site specificity The number of persons allocated to task completion is generally the minimum number required for the task.

Working area exposure rates are assigned based on the orientation of the worker with respect to the source and take into account the use of temporary shielding.

Table 10 3-1 summarizes the estimated total exposure, by task, attributable to the loading, transfer, sealing and placement of a design basis NAC-MPC.

Due to the additional cooling of casks in a typical array, the actual loading, transfer and storage dose associated with the 16 storage cask array is significantly lower than that of a cask array consisting of design basis casks. Based on the source term weighting factors presented in Section 10.3.2, the occupational dose is approximately 20% lower for a typical cask array.

10.3.1 Estimated Annual Dose Due to Routine Operations

Once in place, the ISFSI will require limited ongoing maintenance and surveillance throughout its design life. The annual dose evaluation considers the combination of the requirements specified in Chapter 12.0 and tasks that are anticipated to be representative of an operational facility. Typically, no maintenance of the storage system is expected to be required annually. Collective dose due to certain events, such as clearing the blockage of air vents, is accounted for in Chapter 11.0.

Routine operations are expected to include:

- A daily visual inspection of the cask array. This inspection consists of the electronic measurement of air outlet temperatures and inspection for blockage of the inlet and outlet vents. Outlet temperature indicators are located away from the cask array. Temperature surveillance is assumed to be performed by one operator and require 1 minute per cask. Inspection of the vents is assumed to take one operator 2 minutes per cask.

- A daily security inspection of the security fence and equipment surrounding the storage area. This surveillance is assumed to require 5 minutes and 1 security officer.

- Grounds maintenance performed every other week by 1 maintenance technician. Grounds maintenance is assumed to require 0.5 hour.

- Quarterly radiological surveillance. The surveillance consists of a radiological survey comprised of a surface radiation measurement on each cask, the determination and/or verification of general area exposure rates and radiological postings. This surveillance is assumed to require 1 hour and 1 person.

- Annual inspection of the general condition of the storage casks. This inspection is estimated to require 15 minutes per cask and require 2 technicians.

[redacted]

weighting factors are based on the decayed spectra of the fuel neutron and fuel gamma components of the source term. Table 10.3-2 presents the weighting factors applied to the individual casks in the array. The arrangement of the array is shown in Figure 10.3-1.

For this evaluation, it is assumed that the storage cask array consists of four casks containing the design basis Combustion Engineering Type A 36,000 MWD/MTU burnup, 8 year cooled fuel, and twelve casks loaded with CE Type A 36,000 MWD/MTU burned fuel with cool times representative of the staggered core discharge. The assumed cool times of the fuel in the cask array are shown in Table 10.3-3. The design basis casks represent not only the design basis CE fuel assembly but also any of the other Yankee fuel categories at the minimum cool time and maximum burnup limit. As stated in Section 5.4.3 maximum design basis cask surface dose rates were employed to establish the minimum cool time for each of the fuel categories.

Based on these assumptions, the annual operation and surveillance requirements result in an estimated annual collective exposure of 626 person-mrem for the 16 cask ISFSI. The estimated annual exposure, by task, is shown in Table 10.3-4.

10.3.3 Estimated Collective Dose for Unloading a Single NAC-MPC

This section estimates the collective dose due to the transfer, opening and unloading a single NAC-MPC containing design basis fuel. This analysis assumes that the exposure incurred by the operators is independent of background radiation, as background radiation varies with each site. The number of persons allocated to task completion is generally the minimum number required for the task. Working area exposure rates are assigned based on the orientation of the worker with respect to the source and take into account the use of temporary shielding.

Table 10.3-5 summarizes the estimated total exposure, by task, attributable to the transfer, opening and unloading of a design basis NAC-MPC.

Figure 10.3-1 Typical ISFSI 16 Cask Array Layout

FIGURE WITHHELD UNDER 10 CFR 2.390

Table 10.3-1 Estimated Person-Mrem Exposure for Operation of the NAC-MPC

Activity	Personnel	Duration (hr)	Average Dose Rate (mrem/hr)	Exposure (pers-mrem)
Load Canister	2	[redacted]	[redacted]	[redacted]
Move to Decon Area	2	[redacted]	[redacted]	[redacted]
Setup, Weld Shield Lid, and Inspect	2	[redacted]	[redacted]	[redacted]
Weld	[redacted]	[redacted]	[redacted]	[redacted]
Drain/Dry/Backfill and Leak Test	2	[redacted]	[redacted]	[redacted]
Vacuum Drying	[redacted]	[redacted]	[redacted]	[redacted]
Weld and Inspect Port Covers	2	[redacted]	[redacted]	[redacted]
Setup, Weld Structural Lid and	2	[redacted]	[redacted]	[redacted]
Inspect Weld	[redacted]	[redacted]	[redacted]	[redacted]
Transfer to Storage Cask	4	[redacted]	[redacted]	[redacted]
Position on ISFSI Pad	2	[redacted]	[redacted]	[redacted]
Total		[redacted]		[redacted]

Table 10.3-2 Storage Cask Radiation Spectra Weighting Factors

Cask	Neutron Weighting Factor	Gamma Weighting Factor	Cask	Neutron Weighting Factor	Gamma Weighting Factor
A-1	1	1	B-1	.83	.74
A-2	1	1	B-2	.80	.71
A-3	1	1	B-3	.78	.69
A-4	1	1	B-4	.75	.67
A-5	.96	.93	B-5	.72	.65
A-6	.93	.86	B-6	.70	.63
A-7	.90	.82	B-7	.67	.61
A-8	.86	.77	B-8	.65	.59

Table 10.3-3 Assumed Cooling Time for the Storage Casks in the ISFSI Array

Cask	Cooling Time	Cask	Cooling Time
A-1	8 yr.	B-1	13 yr.
A-2	8 yr.	B-2	14 yr.
A-3	8 yr.	B-3	15 yr.
A-4	8 yr.	B-4	16 yr.
A-5	9 yr.	B-5	17 yr.
A-6	10 yr.	B-6	18 yr.
A-7	11 yr.	B-7	19 yr.
A-8	12 yr.	B-8	20 yr.

Table 10.3-4 Estimate of Annual Exposures for a 16 Cask Array

Activity	Dose Rate Distance (meters)	Frequency	Time (hr)	Dose Rate (mrem/hr)	Personnel Required	Total Exposure (person-mrem)
Visual inspection and temperature readings	10	365	0.8	1.3	1	380
Security surveillance	10	365	0.08	1.3	1	38
Radiological surveillance	5	4	1.0	2.7	1	11
Annual inspection	0.3	1	4.0	20.2	2	162
Grounds maintenance	5	26	0.5	2.7	1	35
Total [redacted]						626

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Setup and Cut Shield Lid	2	4.0	21.5	172.0
Move to Pool	2	1.1	11.3	23.3
Unload Canister	2	2.5	5.0	25.0
Total		[illegible]		[illegible]

10.4 Exposures to the Public

[redacted] SKYSHINE-III code is used to evaluate the placement of the controlled area boundary for the 16 storage cask array [redacted] integrate over the source direction and energy variables.

The cask array is explicitly modeled in the code, with the source term from each cask represented as top and side surface sources. Surface source emission fluxes are provided from 1D SAS1 shielding evaluations. The top and side source energy distributions for both neutron and gamma radiation are taken from the design basis cask in Tables 5.4-4 and 5.4-5. As stated in Section 10.2, the associated reference cask source strengths are multiplied by weighting factors to correct for the differences in cooling times.

Version 5.0.0 of SKYSHINE-III, explicitly calculates cask self shielding based on the cask geometry and arrangement of the cask array. A ray tracing technique is utilized. Given the source position on the cask surface and the direction cosines for the source emission, geometric tests are made to see if any adjacent casks are in the path of the emission. If so, the emission history does not contribute to the air scatter dose. Also, given the source position on the cask surface and the direction cosines for the source to detector location, geometric tests are made to see if any adjacent casks are in the source path. If so, the emission position does not contribute to the uncollided dose at the detector location.

The performance of the SKYSHINE-III Code is benchmarked by modeling a set of Kansas State University ^{60}Co skyshine experiments and by modeling two Kansas State University neutron computational benchmarks. The code compared well with these benchmarks for both neutron and gamma doses versus distance.

Annual exposures based on a 2,080 hour work year were determined at distances ranging from 100 to 300 meters surrounding a 2 x 8 cask array (Figure 10.3-1) as shown in Table 10.4-1. The storage

array is assumed to have a fuel population reflective of that of an operating reactor facility and that fuel cool time will range from 8 to 20 years.

[redacted]

While these analyses are performed for typical arrays of casks containing design-basis spent fuel assemblies, full compliance with the requirements of 10 CFR 72.104(a) can only be demonstrated on a site-specific basis. Consequently, each ISFSI licensee performs a site-specific dose analysis for the facility to show that the requirements of 10 CFR 72 are met. Site-specific boundary distances may vary significantly based on fuel type, fuel cooling time, exposure duration and number of casks in service.

Figure 10.4-1 Controlled Area Boundary Determination for a 16 Cask Array

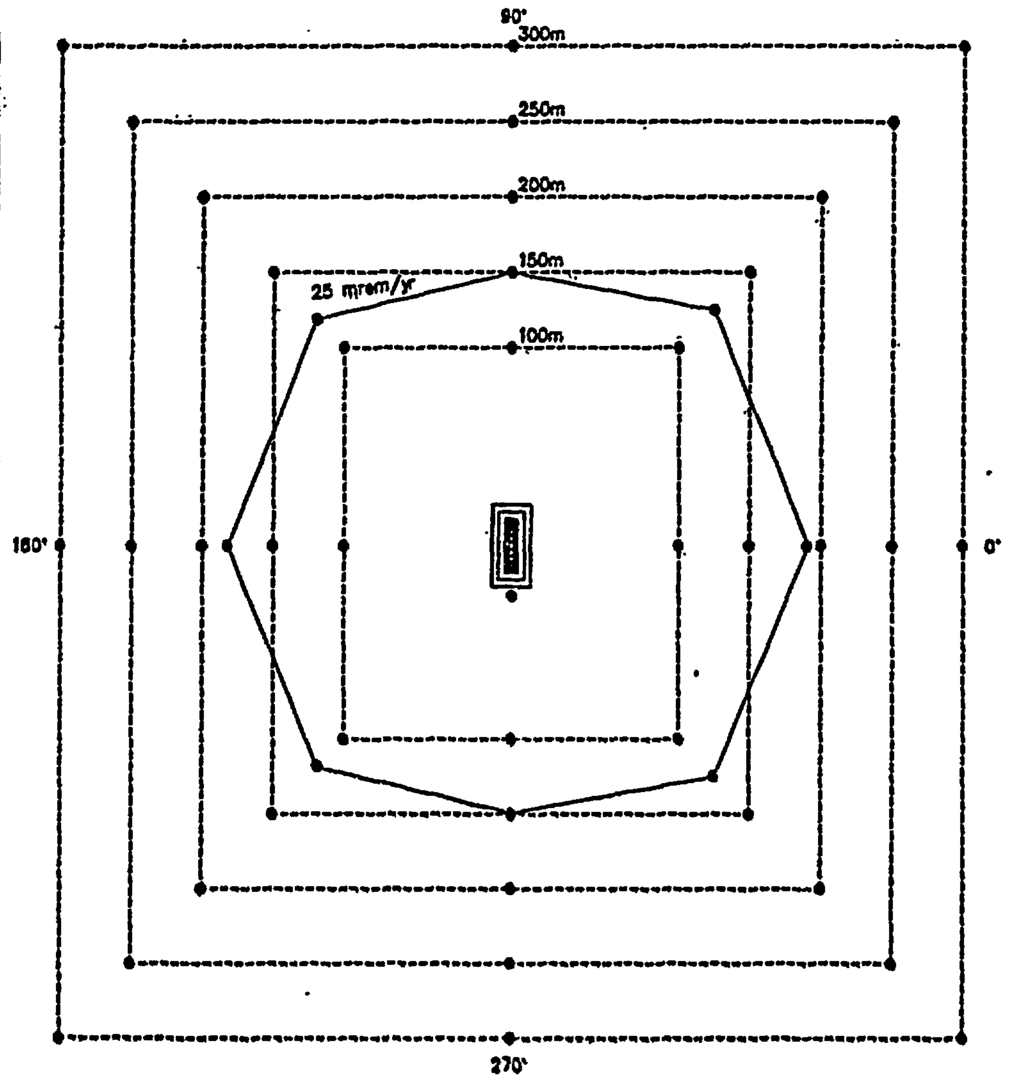

Table 10.4-1 [redacted]

[redacted]

Table 10.4-2 Controlled Area Boundary for 2 x 8 Cask Array [redacted]

Orientation (degrees)	[redacted]
0	190.49
45	176.95
90	149.95
135	167.95
180	182.48
225	165.74
270	146.68
315	177.62

Notes:

1. Denotes counter clockwise orientation from X axis shown in Figure 10.3-1
2. Boundary distance is from security fence. In case of 45, 135, 225, and 315 orientations, boundary distance is diagonally from corner of security fence.
3. Based on 2,080 hours of occupancy.

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[redacted] .... 11.2-[redacted]

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11.0 ACCIDENT ANALYSIS

The analyses of the off-normal and accident design events, including those identified by ANSI/ANS 57.9-1992, are presented in this section. Section 11.1 describes the off-normal events that could occur during the use of the NAC-MPC storage system, possibly as often as once per calendar year. Section 11.2 addresses very low probability events that might occur once during the lifetime of the ISFSI or hypothetical events that are postulated because their consequences may result in the maximum potential impact on the surrounding environment. Section 11.3 describes the design basis load conditions for the transportable storage canister. As described in Section 11.3, the canister is analyzed for loads imposed during transportation. These transport condition loads envelope the loads for the storage condition analyzed herein.

This chapter demonstrates that the NAC-MPC satisfies the requirements of 10 CFR 72.24 and 10 CFR 72.122 for off-normal and accident conditions. These analyses are based on conservative assumptions to ensure that the consequences of off-normal conditions and accident events are bounded by the reported results. The actual response of the NAC-MPC system to the postulated events will be much better than that reported, i.e., stresses, temperatures, and radiation doses will be lower than predicted. If required for a site specific application, a more detailed evaluation could be used to extend the limits defined by the events evaluated in this section.

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11.1 Off-Normal Events

This section evaluates postulated events that might occur once during any calendar year of operations. The actual occurrence of any of these events is unlikely.

11.1.1 Blockage of Half of the Air Inlets

This section evaluates the NAC-MPC storage cask for the steady state effects of a blockage of one-half of the air inlets at the normal ambient temperature (75°F).

11.1.1.1 Cause of Event

The likely cause of air inlet blockage is debris deposited in the inlets by wind or by intrusion of a burrowing animal. It is expected that screens over the inlets would preclude such animals and would exclude debris from the inlet channels.

This event would be detected visually by the persons inspecting the air inlets and gathering outlet air temperature data on a daily basis. It could also be detected by security forces, or other operations personnel engaged in other routine activities, such as fence inspection or grounds maintenance.

11.1.1.2 Analysis of the Blockage Event

Off-normal temperature conditions are evaluated using the thermal models described in Section 4.4.1. Air mass flow and air carried heat are calculated using the airflow model described in Section 4.4.1.1. The maximum component temperatures due to one-half of the air inlets being blocked are compared to the allowable component temperatures for the off-normal event (see Table 4.1-4).

Component	1/2 Inlets Blocked Max Temp. (°F)	Allowable Temp. (°F)
Fuel Cladding	565	[redacted]
Support Disks	531	800
Heat Transfer Disks	529	[redacted]
Canister Shell	318	800
Concrete	168	350

This evaluation shows that the component temperatures are within the allowable temperature range for the condition of one-half of the inlets blocked.

11.1.1.3 Radiological Consequences

There are no significant radiological consequences for this event. Personnel will be subject to an estimated maximum contact dose rate of 240 mrem/hr when clearing the inlets. If it is assumed that a worker kneeling with his hands on the inlets would require 15 minutes to clear the inlets, the estimated maximum extremity dose is 60 mrem. The whole body dose would be significantly less.

11.1.1.4 NAC-MPC Performance

There are no adverse consequences for this off-normal condition. The maximum component temperatures are less then the allowable temperatures. The NAC-MPC storage cask continues to perform its function with one-half of the air inlets blocked.

11.1.1.5 Recovery and/or Corrective Actions

The debris blocking the inlets must be manually removed. The nature of the debris may indicate that other actions are required to prevent recurrence of the blockage.

11.1.2 Canister Off-Normal Handling Load

This section evaluates the consequence of loads on the transportable storage canister during the installation of the canister in the storage cask, or removal of the canister from the storage or transfer casks.

11.1.2.1 Cause of Event

Unintended loads could be applied to the canister, due to misalignment or faulty crane operation or inattention of the operators.

Detection of the event is expected to occur by observation of the event or banging or scraping noise associated with movement of the canister. The event is expected to be obvious to the operators at the time of occurrence.

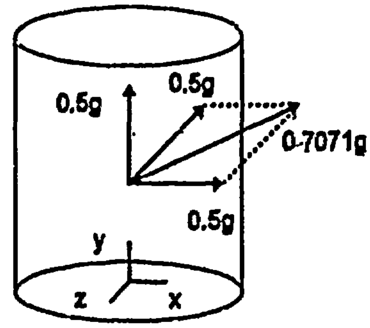
11.1.2.2 Analysis of the Canister Off-Normal Handling Load Event

The canister structural analysis, including lifting loads, was evaluated using an ANSYS finite element model. The model is described in Section 3.4.4-1.

The off-normal handling load condition is assumed to consist of loads of 0.5 g applied in all directions (i.e., in the global x, y, and z directions) in addition to the 1.1 g lifting load (an additional 10 percent load is included as a dynamic load factor during lifts) applied in the finite element model. The stresses resulting from off-normal handling are estimated by combining the normal handling stresses at the design off-normal internal pressure of 11 psig with the stress results from a 20 g side and a 20 g bottom end impact of the canister ratioed to the off-normal 0.5 g-loading.

A design off normal internal pressure is calculated based on the condition of 10 percent fuel rod failure with 30 percent fission gas release at a conservative gas temperature of 630°F. The calculated off normal internal pressure is 14.6 psig. A design internal off normal pressure of 11 psig is conservatively used in the canister finite element model as shown in Figure 3.4.4.1-1.

The 0.5 g acceleration in the vertical direction is additive to the 1.1 g acceleration for normal lifting. The two 0.5 g accelerations in the horizontal directions result in a single horizontal acceleration of 0.7071 g as shown below:



$$\sqrt{(0.5g)^2+(0.5g)^2}=0.7071g$$

The stresses obtained from the 20 g side and 20 g bottom end impacts of the canister were then scaled to obtain the additive off-normal handling stresses for 0.7071g side load ($\sigma_{0.7071g}$) and 0.5g vertical load ($\sigma_{0.5g}$) as follows:

$$\sigma_{0.7071g}=\sigma_{1-ft,side}\left(\frac{0.7071g}{20g}\right)$$

and,

$$\sigma_{0.5g}=\sigma_{1-ft,bottom\,end}\left(\frac{0.5g}{20g}\right)$$

Where 20 g is the deceleration applied in the canister model analysis.

The off-normal handling stresses for the side and the vertical g-loading were then added to the normal-handling stresses to obtain the total off-normal handling stresses.

The pertinent stress results from the 20 g side impact and 20 g bottom end of the canister are summarized below. These stress results are presented in Section 11.3.1.

Side: S_{pm} = 13,884 psi
S_{pm+pb} = 24,044 psi

Bottom End: S_{pm} = 1,961 psi
S_{pm+pb} = 4,273 psi

The side and bottom end impact stresses scaled to off-normal handling g-loads were calculated as:

Side:

$$\left(S_{pm}\right)_{0.7071g}=\left(S_{pm}\right)_{20g}\left(\frac{0.7071g}{20g}\right)=13{,}884\,\text{psi}\left(\frac{0.7071g}{20g}\right)=491\text{ psi }(0.491\text{ ksi})$$

$$\left(S_{pm+pb}\right)_{0.7071g}=\left(P_{pm+pb}\right)_{20g}\left(\frac{0.7071g}{20g}\right)=24{,}044\,\text{psi}\left(\frac{0.7071g}{20g}\right)=850\text{ psi }(0.85\text{ ksi}$$

Vertical (bottom):

$$\left(S_{pm}\right)_{0.5g}=\left(S_{pm}\right)_{20g}\left(\frac{0.5g}{20g}\right)=1{,}961\,\text{psi}\left(\frac{0.5g}{20g}\right)=49\text{ psi }(0.049\text{ ksi})$$

$$\left(S_{pm+pb}\right)_{0.5g}=\left(S_{pm+pb}\right)_{20g}\left(\frac{0.5g}{20g}\right)=4{,}273\,\text{psi}\left(\frac{0.5g}{20g}\right)=107\text{ psi }(0.107\text{ ksi})$$

The total stresses for a load condition that includes off-normal handling is obtained by adding the off-normal handling stresses to the normal handling stresses [redacted]

$$\begin{aligned}\left(S_{pm}\right)_{Case\,w/Off-normal}&=\left(S_{pm}\right)_{Normal}+\left(S_{pm}\right)_{0.7071g}+\left(S_{pm}\right)_{0.5g}\\&=\left(S_{pm}\right)_{Normal}+0.491\text{ ksi}+0.049\text{ ksi}\\&=\left(S_{pm}\right)_{Normal}+0.54\text{ ksi}\end{aligned}$$

$$\begin{aligned}\left(S_{pm+pb}\right)_{Case\,w/Off-normal}&=\left(S_{pm+pb}\right)_{Normal}+\left(S_{pm+pb}\right)_{0.7071g}+\left(S_{pm+pb}\right)_{0.5g}\\&=\left(S_{pm+pb}\right)_{Normal}+0.85\text{ ksi}+0.107\text{ ksi}\\&=\left(S_{pm+pb}\right)_{Normal}+0.96\text{ ksi}\end{aligned}$$

The maximum primary membrane stress (S_{pm}) for normal handling occurs at location 13. The maximum primary membrane plus bending stress (S_{pm+pb}) for normal handling occurs at location 2 These stress locations are shown in Figure 3.4.4.1-4. Allowable stresses are determined at 250°F in accordance with ASME Code Section NB, Service Level C (See Table 2-4).

For location 13:

Normal Handling P_m Stress (ksi)	12.07
Additional P_m Stress Due to Off-Normal Handling (ksi)	0.54
Off-Normal Handling P_m Stress (ksi)	12.61
Allowable P_m Stress (ksi)	20.30
Margin of Safety	+0.61

For location 2:	
Normal [redacted] Stress (ksi)	26.15
Additional [redacted] Stress Due to Off-Normal Handling (ksi)	0.96
Off-Normal Handling [redacted] Stress (ksi)	27.11
Allowable [redacted] Stress (ksi)	30.06
Margin of Safety	+0.11

These results show that the canister maintains a positive margin of safety for the off-normal handling condition.

11.1.2.3 Radiological Consequences

There are no radiological consequences for this off-normal event.

11.1.2.4 NAC-MPC Performance

This evaluation shows that the stress induced in the canister as a consequence of the assumed off-normal handling loading is within the allowable stress for Service Level C loading. There is no deterioration of canister performance.

11.1.2.5 Recovery and/or Corrective Actions

Operations should be halted until the cause of the misalignment, interference or faulty operation is identified and corrected. Since the radiation level of the canister sides and bottom is high, extreme caution should be exercised if inspection of these surfaces is required.

11.1.3 Failure of Instrumentation

The NAC-MPC system uses an electronic temperature sensing system to read and record the outlet air temperature at each of the four air out'-ts on each storage cask. The temperatures are read and recorded during a daily inspection of the ISFSI.

11.1.3.1 Cause of Accident

Failure of the temperature measuring instrumentation could occur as a result of component failure, or as a result of another accident condition that interrupted power or damaged the sensing or reader terminals.

The failure is expected to be identified by the lack of a reading at the temperature reader terminal. Alternately, a malfunction could result in a disparity between outlet temperatures or between similar storage casks.

11.1.3.2 Analysis of Instrumentation Failure

Since the temperatures of each outlet of each storage cask are recorded daily, there is early opportunity to identify and correct a defect. Because the canister and concrete cask are a large heat sink, and because there are few conditions that could result in a cooling air temperature increase, there is no concern about the temporary loss of remote sensing and monitoring of the outlet air temperature.

The principal condition that could cause an increase in temperature is the blockage of the cooling air inlets or outlets. The purpose of the daily inspection is to ensure that the inlets and outlets are not obstructed, such that the cooling efficiency of the system is reduced. As shown in Section 11.2.8, even if all of the inlets and outlets for a single cask are blocked immediately after the temperature is read, it would take more than 24 hours before any component approached its allowable temperature limit. There would be no consequence, if the affected storage cask continued to operate in normal storage conditions.

11.1.3.3 Radiological Consequences For This Accident

There are no radiological consequences for this event.

11.1.3.4 NAC-MPC Performance

The NAC-MPC canister and storage cask are a large thermal sink. During the period of loss of instrumentation, no significant change in canister temperature will occur under normal conditions.

11.1.3.5 Recovery and Corrective Actions

This event requires that the temperature reporting equipment be either replaced or repaired and calibrated. Prior to repair or replacement, the temperature shall be recorded manually.

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

This section evaluates the NAC-MPC for the steady state effects of high and low ambient temperature conditions.

11.1.4.1 Cause of Event

Large geographical areas of the United States are subjected to sustained summer temperatures in the 90 to 100°F range and winter temperatures that are significantly below zero. To bound the expected steady state temperatures of the canister and storage cask during these severe ambient conditions, analyses were performed to calculate the steady state storage cask, canister, and fuel cladding temperatures for a 100°F ambient temperature and 24-hour average solar loads. Similarly, winter weather analyses were performed for a -40°F ambient temperature with no solar load. The maximum thermal load of 12.5 kW was applied for these analyses. Neither ambient temperature condition is expected to last more than several days.

Detection of off-normal ambient temperatures would occur during the daily measurement of ambient temperature and storage cask outlet air temperature.

11.1.4.2 Analysis of the Off-Normal Ambient Temperature Event

Off-normal temperature conditions are evaluated using the thermal models described in Section 4.4.1. The temperature profile for the concrete cask and for the air flow for the steady state conditions associated with a 100°F ambient condition are shown in Figures 11.1.4-1 and 11.1.4-2, respectively. Similar profiles for the -40°F ambient temperature condition are shown in Figures 11.1.4-3 and 11.1.4-4. The principal component temperatures for each of these ambient temperature conditions are summarized below.

Component	100°F Ambient Max Temp. (°F)	-40°F Ambient Max Temp. (°F)	Allowable Temp. (°F)
Fuel Cladding	587	453	[redacted]
Support Disks	554	412	800
Heat Transfer Disks	552	411	[redacted]
Canister Shell	347	187	800
Concrete	196	5	350

This evaluation shows that the component temperatures are within the allowable values for the off-normal ambient conditions.

The thermal stress evaluation for these off-normal conditions are bounded by that for the accident condition with 125°F ambient temperature (Section 11.2.10), since the accident condition has the maximum temperature gradient through the storage cask concrete wall.

Stress intensities corresponding to thermal loads in the canister were evaluated using an ANSYS finite element model as described in Section 3.4.4. The thermal stresses occur in the canister as a result of the maximum temperature gradients in the canister. The finite element analysis assumes that the canister contains the maximum heat load of 12.5 kW, as this ensures the largest gradient for either load condition (i.e., if the canister was at a uniform -40°F, no temperature gradient would exist and no thermal stress would be induced in the canister). No other loads are applied, and the thermal stress is classified as secondary. The smallest margin of safety is +2.75, which occurs at location 13, at the center of the bottom plate (Figure 11.2.1-1). The thermal stresses for the support disks and weldments due to these off-normal conditions are bounded in the thermal stress analysis presented in Section 3.4.4.1.

11.1.4.3 Radiological Consequences

There are no radiological consequences for this off-normal event.

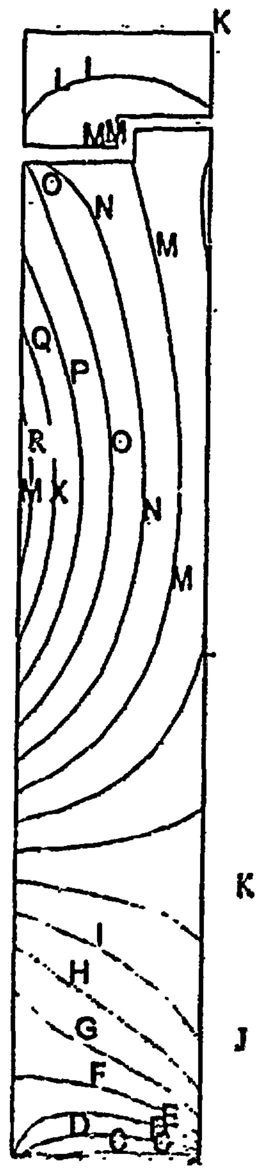
11.1.4.4 NAC-MPC Performance

There are no adverse consequences for this off-normal condition. The maximum component temperatures are within the allowable temperature values. The materials used are not subject to low temperature brittle fracture.

11.1.4.5 Corrective Actions

No corrective actions are required for this off-normal condition.

Figure 11.1.4-1 Temperature Profile of the Concrete Cask in 100°F Ambient Steady State Conditions



	°F
MX=	196
*A=	103
*B=	108
C=	113
D=	119
E=	124
F=	129
G=	134
H=	140
I=	145
J=	150
K=	156
L=	161
M=	166
N=	172
O=	177
P=	182
Q=	188
R=	193

MX = Maximum Temperature

*These temperatures occur over short distances around the inlet vent, but are not shown due to the scale of the figure.

Figure 11.1.4-2 Temperature Profile of the Air Flow Stream in 100°F Ambient Steady State Conditions

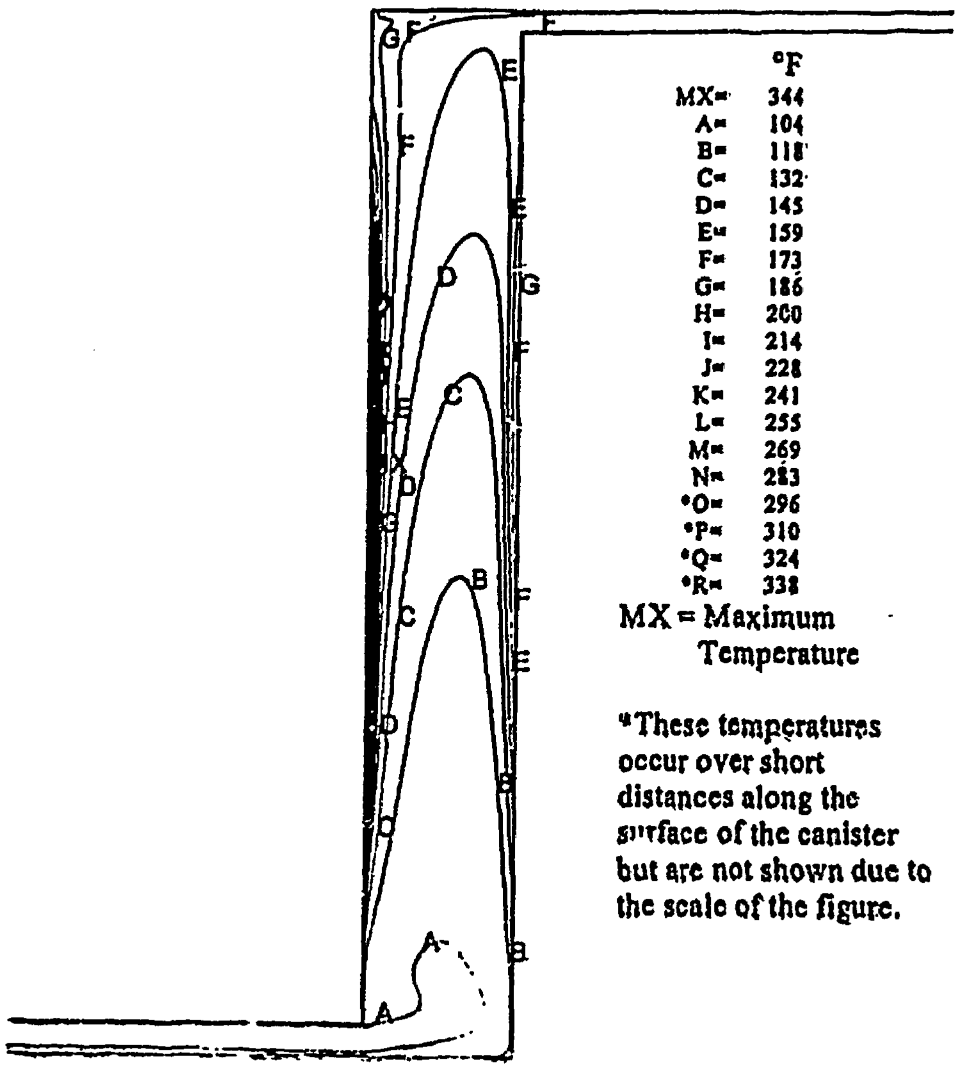


Figure 11.1.4-3 Temperature Profile of the Concrete in -40°F Ambient Steady State Conditions

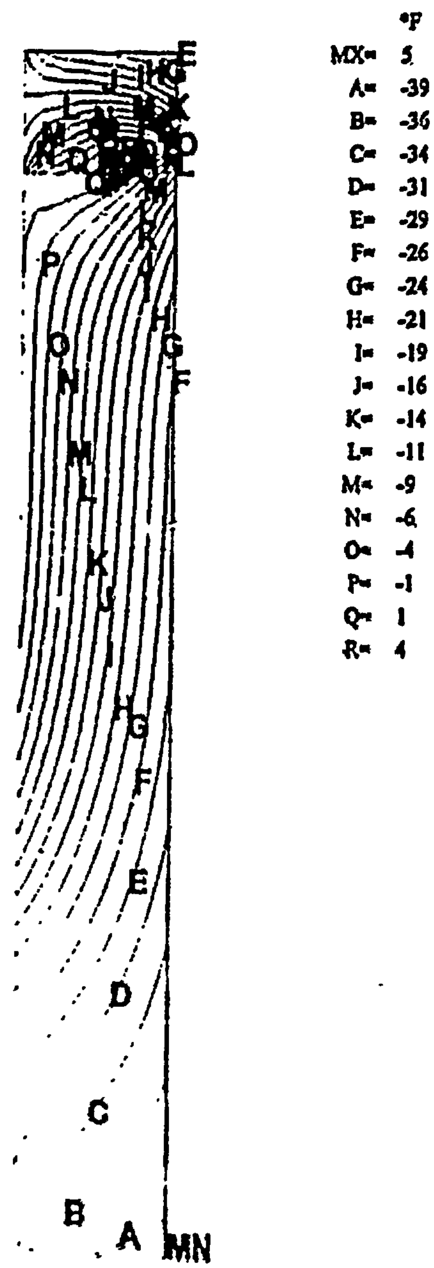
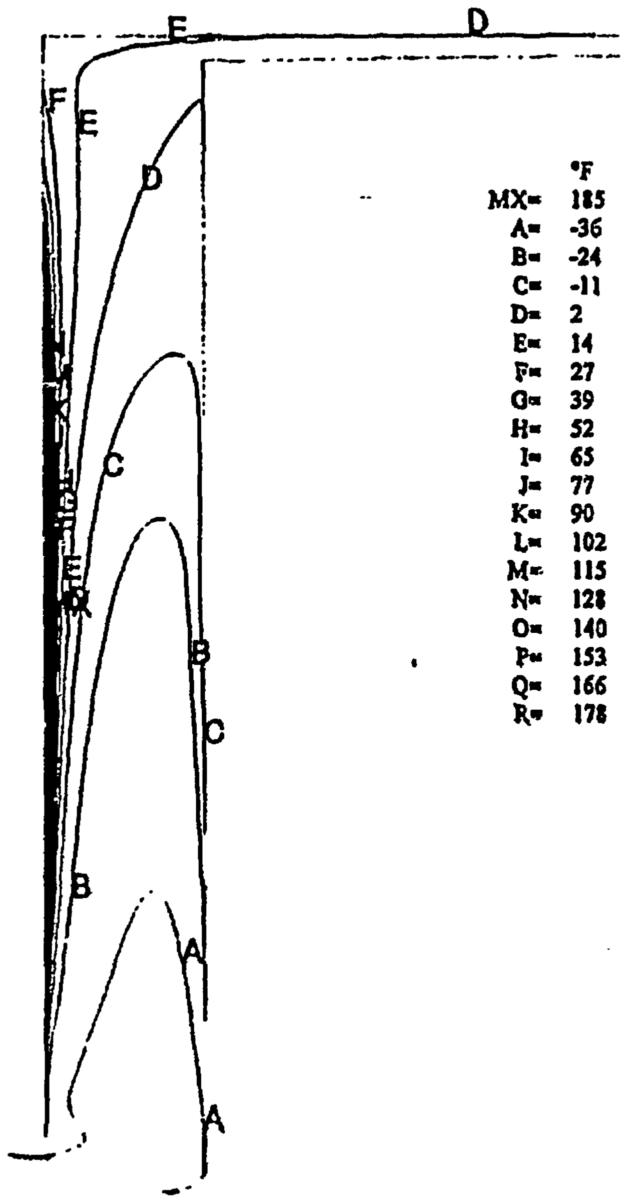


Figure 11.1.4-4 Temperature Profile of the Air Flow Stream in -40°F Ambient Steady State Condition



11.1.5 Small Release of Radioactive Particulate From the Canister Exterior

The procedures for loading the canister provide for steps to ensure that the canister exterior surface does not come into contact with contaminated spent fuel pool water, and the exterior surface of the canister is surveyed by smear at the top end to verify canister surface conditions. No particulate release from the canister exterior surface is expected to occur in normal use.

11.1.5.1 Cause of Event

In spite of precautions taken to preclude contamination to the external surface of the canister, it is possible that a portion of the canister surface may become slightly contaminated and that the contamination will go undetected. Surface contamination could become airborne and be released as a result of the air flow over the canister surface.

Detection of the release of small amounts of radioactive particles over time would be difficult to ascertain. The release would likely not be at a level that would result in detection by any of the long-term radiation dose monitoring methods (such as TLDs) normally employed. It is possible that a suspected release could be verified by a smear survey of the air outlets.

11.1.5.2 Analysis

A calculation was made to determine the level of surface contamination that results in a dose of one (1) mrem annually at a point 100 meters from the ISFSI site. The calculation shows that at the minimum distance of 100 meters permitted by 10 CFR 72, a residual contamination limit of approximately 20,000 dpm/100 cm^2 β-γ and 200 dpm/100 cm^2 α activity, on the surface of each of 16 casks yields a dose of one (1) mrem annually.

The method for determining the residual contamination limit is based on the plume dispersion calculations presented in U. S. NRC Regulatory Guides 1.109 and 1.145.

11.1.5.3 Radiological Consequences

The projected dose at a boundary located 100 meters from the ISFISI is estimated to be less than one (1) mrem annually due to the postulated surface contamination. This dose is based on a postulated surface contamination of approximately 20,000 dpm/100 cm^2 β-γ and 200 dpm/100 cm^2 α, for each of the 16 storage casks in the design basis ISFSI. This analysis is highly conservative and demonstrates that the potential off-site radiological consequences from the release of surface contamination on the canisters is negligible.

11.1.5.4 NAC-MPC Performance

Procedural steps are employed to ensure that the canister surface is generally free of surface contamination prior to its installation in the storage cask. The surface of the canister is free of traps that could hold contamination. The presence of external surface contamination on the canister is unlikely.

11.1.5.5 Corrective Actions

No corrective action is required, since the radiological consequence is negligible.

11.2.13.2.3 Tornado Effects on the Canister

The postulated tornado wind loading and missile impacts are not capable of overturning the cask, or penetrating the boundary established by the concrete cask. Consequently, there is no effect on the canister.

11.2.13.3 Radiological Consequences

This evaluation shows that there is little potential for significant damage to the concrete cask, which provides radiation shielding.

Under the worst tornado missile impact, a penetration of 5.68 inches into the concrete shield is possible. This would result in a local surface radiation dose rate at the point of penetration of 125 mrem/hr. Since the area of reduced shielding is very small, there would not be a noticeable increase in the dose rate at the site boundary. The estimated dose rate is well below the 1000 mrem/hr limit for dose rates one meter from the cask wall after an accident.

Repair of the damage would likely require two persons to form the damaged area and apply grout. This is estimated to require 10 minutes. The estimated extremity dose is 125 mrem. The whole body dose would be less. The dose rate of clearing inlets and outlets has been estimated in Section 11.1.1.3 at approximately 25 mrem per opening.

11.2.13.4 NAC-MPC Performance

The concrete cask is demonstrated to be stable in tornado wind loading in conjunction with impact from a high energy tornado missile. The performance of the NAC-MPC is not significantly affected by the tornado event.

The storage cask has a safety factor of [illegible] against overturning under the sustained maximum wind loading. The maximum concrete shell stresses under maximum tornado wind loading are [illegible] psi in shear and 11.51 psi in bending, which are well below their respective ACI [illegible] code allowables of [illegible] psi and [illegible] psi, respectively.

The concrete shell of the storage cask is more than three times thicker than the calculated missile penetration depth, which is adequate to prevent significant scabbing of the concrete.

11.2.13.5 Recovery and/or Corrective Actions

A tornado event is not expected to result in the need to take any corrective action other than an inspection of the ISFSI. This inspection would be directed at ensuring that inlets and outlets had not become blocked by wind-blown debris and at checking for obvious (concrete) surface damage.

As shown in the evaluation, in the worse case, a missile could dislodge concrete to a depth of approximately 6 inches. To repair the cask surface this area would need to be filled with grout. As noted, penetration to 6 inches is unlikely, since the reinforcing bar cage is encountered at a depth of 2 to 3 inches, depending on the location on the cask surface.

11.2 Accidents

This section provides the results of analyses of the design basis and hypothetical accident conditions evaluated for the NAC-MPC system. The analyses presented show that the NAC-MPC system has substantial design margin of safety and provides protection to the public and to occupational personnel. In addition to these design basis accidents, this section addresses very low probability events that might occur over the lifetime of the ISFSI or hypothetical events that are postulated because their consequences may result in the maximum potential impact on the immediate environment.

11.2.1 Accident Pressurization

Accident pressurization is a hypothetical event that assumes the failure of all of the fuel rods contained within the canister [redacted] There are no storage conditions that are expected to lead to the rupture of all of the fuel rods [redacted]

11.2.1.1 Cause of Pressurization

The hypothetical breach of all of the fuel rods in a canister would release the fission and fill gases to the interior of the canister.

11.2.1.2 Analysis of Accident Pressurization

11.2.1.2.1 Maximum Canister Internal Pressure

The analysis requires the calculation of the free volume of the canister, calculation of the quantity of fill and fission gas in the 36 fuel assemblies, and the subsequent calculation of the pressure in the canister if these gases are added to the helium pressure (initially at 1 atm) already present in the canister (Section 4.4.5). The quantity of fission gases was conservatively estimated assuming that 30% of the total gases present are released from the fuel. The bulk temperature of the helium is conservatively taken to be [redacted]°F. [redacted]

The internal pressure is a function of rod-fill, fission and canister backfill gases. All of the gases, except the fission gases, are assumed to be helium. The total pressure for each volume are found by calculating the molar quantity of each gas and summing those directly. The design basis fuel assembly for the internal pressure calculation is the Combustion Engineering Type A assembly. This assembly has the highest fuel rod back-fill pressure (315 psig) and received the highest burnup (36,000 MWD/MTU) (Section 2.1.1).

The number of moles of the backfill gases are calculated using the Ideal Gas Law, PV = NRT. Backfill gases for the canister and cavity are assumed to be initially at 1 atmosphere. The quantity of fission gas is derived [redacted]

The number of moles of gas in the canister is.

$$N = N_{Fill\ Gas\ Fd} + N_{Rod\ Bkf\ Fd} + 0.3\left(N_{Fission\ Gas}\right)$$

The number of moles of helium contained in the canister as backfill and the number of moles of gas in the fuel rods (as helium backfill and fission products) were calculated in Section 4.4.5

The number of moles of gas due to the hypothetical failure of 100% of the fuel rods is

$$N = \text{[redacted]}\ \frac{\text{Moles}}{\text{Cask}} + 77.05\ \frac{\text{Moles}}{\text{Cask}} + 0.3\left(\text{[redacted]}\ \frac{\text{Moles}}{\text{Cask}}\right)$$

(canister backfill) (rod backfill) (fission gas)

$$N = \text{[redacted]}\ \frac{\text{Moles}}{\text{Cask}}$$

Based on an assumed [redacted] temperature of [redacted]°F, the maximum pressure in the canister is

$$P = \frac{\left(\text{[redacted]}\ \frac{\text{Moles}}{\text{Cask}}\right) \times \left(0.0821\ \frac{\text{atm}\cdot\ell}{\text{mole}\cdot\text{K}}\right) \times 616.45\ \text{K}}{\left(4,377.03\ \frac{\ell}{\text{Cask}}\right)} = \text{[redacted]}\ \text{atm} = \text{[redacted]}\ \text{psia} = \text{[redacted]}\ \text{psig}$$

11.2.1.2.2 Maximum Canister Stress Due to Internal Pressure

The stresses that result in the canister due to the internal pressure were evaluated using the ANSYS finite element model described in Section 3.4.4. The pressure used for the [redacted] psig [redacted]. The results of the analysis are shown in Tables 11.2.1-1 (primary membrane stress) and 11.2.1-2 (primary membrane plus bending stress). The locations of the canister analysis sections are shown in Figure 11.2.1-1.

[redacted]

These results show that the minimum margin of safety for primary membrane stress is +[redacted] at location 1. The minimum margin of safety for primary membrane plus bending stress is [redacted] at location 2.

11.2.1.3 Radiological Consequences

There are no radiological consequences for this accident.

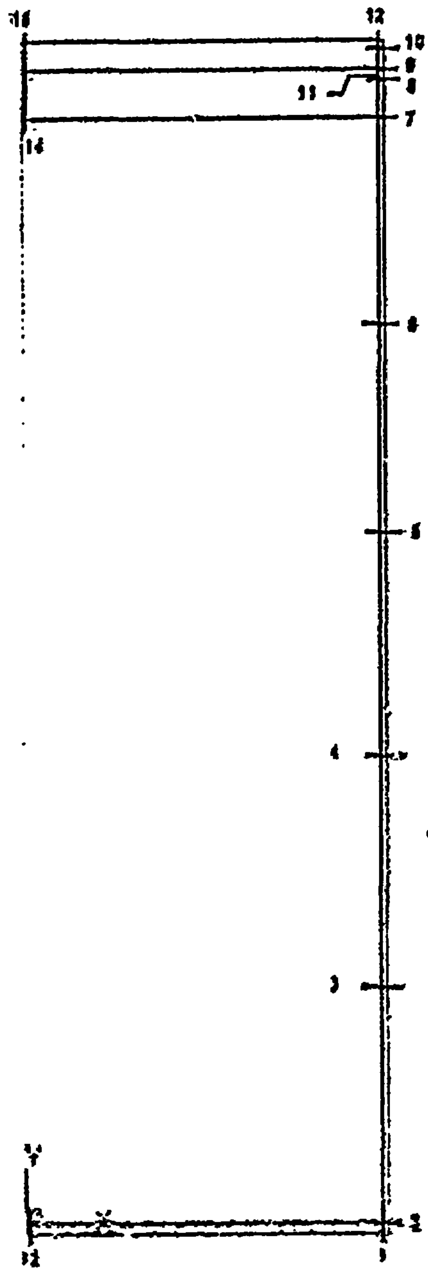
11.2.1.4 NAC-MPC Performance

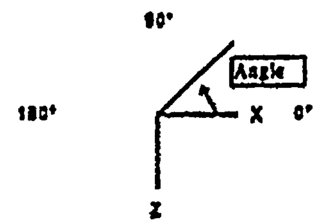
This analysis demonstrates that the canister performance is not significantly affected by the increase in internal pressure that results from the hypothetical rupture of all of the fuel rods contained in the canister. There is a positive margin of safety throughout the canister.

11.2.1.5 Recovery and/or Corrective Actions

There are no recovery or corrective actions required for this hypothetical accident event. The rupture of fuel rods within the canister is unlikely to be detected by any measurements or inspections that could be undertaken from the exterior of the canister or storage cask.

Figure 11.2.1-1 Section Location for Canister Stress Evaluation





Top View of Axis

Location of sections at Z=0 (Note: these sections are repeated at each angular division in the 180° model).

Loc	Node 1 X1	Node 1 Y1	Node 2 X2	Node 2 Y2
1	34.835	0.0	34.835	1.0
2	34.835	1.0	35.32	1.0
3	34.835	23.7	35.32	23.7
4	34.835	46.4	35.32	46.4
5	34.835	69.1	35.32	69.1
6	34.835	91.8	35.32	91.8
7	34.835	114.6	35.32	114.6
8	34.835	118.8	35.32	118.8
9	34.835	119.8	35.32	119.8
10	34.835	121.92	35.32	121.92
11	34.835	118.6	34.835	118.6
12	34.835	[illegible]	34.835	122.6
13	0.01	0.0	0.01	1.0
14	0.01	114.6	0.01	119.2
15	0.01	119.8	0.01	121.6

Table 11.2.1-1 Canister Primary Membrane Stress (ksi) Due to Internal Pressure ([redacted] psig) for the Accident Pressurization Condition

Location	Angle	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity	Allowable Stress	Margin of Safety
1	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
3	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
4	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
5	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
6	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
7	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
8	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
9	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
10	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
11	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
12	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
13	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
14	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
15	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

A stress of "0" indicates that the stress at the location is positive, but less than 0.1 ksi. Components x, y, and z correspond to the radial, circumferential, and axial directions, respectively

Table 11.2.1-2 Canister Primary Membrane Plus Bending Stress (ksi) Due to Internal Pressure ([redacted] psig) for the Accident Pressurization Condition

Location	Angle	SX	SY	SZ	SXY	SYZ	SXZ	Stress Intensity	Allowable Stress	Margin of Safety
1	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
2	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
3	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
4	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
5	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
6	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
7	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
8	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
9	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
10	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
11	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
12	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
13	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
14	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
15	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

A stress of "0" indicates that the stress at the location is positive, but less than 0.1 ksi.

11.2.2 Earthquake Event

11.2.2.1 Cause of Earthquake

Earthquakes are natural phenomena, which the cask might be subjected at any U.S. site. The design basis seismic event is described and discussed in Section 2.2.3.2. This analysis shows that the concrete storage cask containing the loaded canister does not tip over [redacted] in the design basis earthquake. The design basis earthquake is one imparting [redacted] a horizontal [redacted] of 0.25g [redacted]

11.2.2.2 Earthquake Analysis

The concrete storage cask is a very stiff structure. Although free-standing, it has been analyzed as a cantilever fixed at the base (Roark). For the purpose of calculating seismic loads, the cask is treated as a rigid body attached to the ground and equivalent static analysis methods were used to calculated loads, stresses, and overturning moments.

The natural frequencies of the empty concrete storage cask and the storage cask and canister are calculated to be 89 cycles per second and 36.1 cycles per second, respectively. The natural frequency of the empty storage cask was calculated using the frequency equation.

$$fn = \left(\frac{Kn}{2\pi}\right)\sqrt{\frac{EIg}{WL^4}}$$ (Roark, Table 36, Case 3)

The natural frequency of the loaded canister is calculated using:

$$\frac{1}{\omega^2} = \frac{1}{\omega^2_f} + \frac{1}{\omega^2_s}$$ (Blevins, Equation 8-30)

where

$$\omega_f = \frac{\lambda^2}{2\pi L^2}\sqrt{\frac{EI}{M}}$$ and

$$\omega_3 = \frac{\lambda}{2\pi L}\sqrt{\frac{KG}{M}}$$

Since the natural frequency is above 33 cycles per second, there is no dynamic amplification and the storage cask and canister are considered to be rigid bodies (NUREG-0800). Static analysis is applied to the evaluation of the earthquake event response.

The following paragraphs present the calculations for acceleration, overturning/restoring forces, and moments for the fully loaded concrete cask.

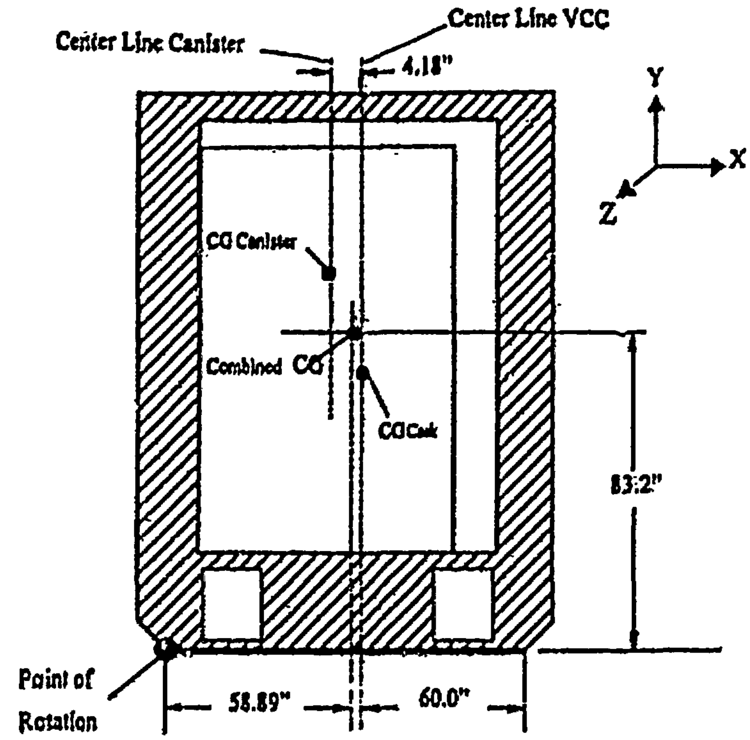
Because the canister is not attached to the concrete cask, the combined center of gravity (CG) for the concrete cask, with the canister in its maximum off-center position, must be calculated. For convenience, a point of rotation (P.O.R.) is established at the outside lower edge of the concrete cask.

The inside diameter of the concrete cask is 79.0" and the outside diameter of the canister is 70.64"; therefore, the maximum eccentricity between the two is $\left(\frac{79.0"-70.64"}{2}\right) = 4.18"$

The horizontal displacement, x, of the combined CG due to eccentric placement of the canister is:

$x = \left(\frac{54730 \times 4.18}{206100}\right) = 1.11"$, therefore, the distance from the postulated point of rotation to the horizontal center of gravity is: $(60.0 - 1.11) = \underline{58.89 \text{ in.}}$

The vertical location of the CG remains at 83.2" The various moment arms resulting from this calculation are shown in the sketch below.



To maintain the concrete cask in equilibrium, the restoring moment, M_R, must be greater than, or equal to, the overturning moment (i.e., $M_R \geq M_O$). [redacted] derivation shows that the 0.25g acceleration of the [redacted] acceleration required to flip the concrete cask over.

The combination of horizontal and vertical acceleration components [redacted] approach of ASCE 4-86, which considers that when the maximum [redacted] occurs, the responses from the other two components are 40% of [redacted] component of acceleration is obtained by scaling the corresponding ordinates of the horizontal components by two-thirds.

Let $a_x = a_z = a$ = horizontal acceleration components

$a_y = (2/3)a$ = vertical acceleration component

Ω_h = resultant of two horizontal acceleration components

Ω_v = vertical acceleration component

There are two combinations that have to be analyzed:

[redacted]

[redacted]

$G_v = 0.4 \times a_y = 0.267 \times a$

[redacted]

For the cask to resist overturning, the restoring moment, [redacted] greater than or equal to the overturning moment, M_o:

$$M_R \geq M_O \text{, or}$$

$$F_r \times b \geq F_o \times d \Rightarrow (W \times 1 - W \times G_v) \times b \geq \text{[redacted]} \times d$$

where,

- d = center of gravity measured from the base [redacted]
- b = horizontal distance from the point of rotation to the CG [redacted]
- W = the weight of the concrete cask (206,100 lbs)
- F_o = overturning force
- F_r = restoring force

Substituting for G_h and G_v gives:

1) $(1-0.667a)\frac{b}{d} \geq 0.566 \times a$

$$a \leq \frac{b/d}{0.566 + 0.667(b/d)}$$

1) $a \leq \frac{58.89/83.2}{0.566 + 0.667 \times 58.89/83.2}$

$$a \leq 0.682g$$

2) $(1-0.267a)\frac{b}{d} \geq 1.077a$

$$a \leq \frac{b/d}{1.077 + 0.267(b/d)}$$

2) $a \leq \frac{58.89/83.2}{1.077 + 0.267 \times 58.89/83.2}$

$$a \leq 0.559g$$

Therefore, the minimum acceleration that may cause a tip over of a fully loaded concrete cask is [redacted]. Since the 0.25g design basis earthquake acceleration is less than [redacted], the [redacted] cask will not tip over.

[redacted]

[redacted]

[redacted]

$$\mu(1-G_v)W \geq G_h W$$

where
μ = coefficient of friction
N = normal force
W = weight of the concrete cask
G_v = vertical acceleration component
G_h = resultant of horizontal acceleration component

Substituting for G_h and G_v for the two combination conditions:

1) $\mu(1-0.667a) \geq 0.566a$ $\qquad$ 2) $\mu(1-0.267a) \geq 1.077a$

$$\mu \geq \frac{0.566a}{1-0.667a} \qquad \mu \geq \frac{1.077a}{1-0.267a}$$

For a = 0.25g

1) $\mu \geq 0.17$ $\qquad$ 2) $\mu \geq 0.29$

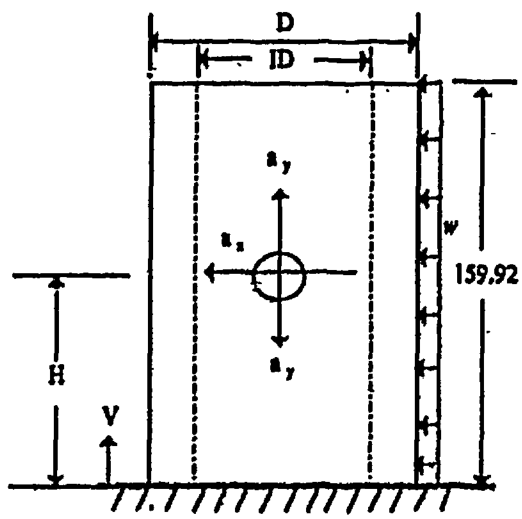
The analysis shows that the minimum coefficient of friction, μ, required to prevent sliding of the concrete cask is 0.29. The coefficient of friction between the steel bottom plate of the concrete cask and the concrete surface of the storage pad, 0.35 (Funk), is greater than the coefficient of friction required to prevent sliding of the concrete cask. Therefore, the concrete cask will not slide under design-basis earthquake conditions. The factor of safety is 0.35 / 0.29 = 1.21 which is greater than the factor of safety of 1.1 required by ANSI/ANS-57.9.

The stresses in the concrete due to the design basis G-loads are conservatively calculated below. The fully loaded concrete cask is considered to be fixed at its base and subjected to seismic loads equal to 0.25 g in the two orthogonal horizontal directions and the vertical direction.

The accelerations are:

$a_x = (0.25^2 + 0.25^2)^{0.5} = 0.354$ g (horizontal direction)

$a_y = \pm 0.25$ g (vertical direction)



The following parameters are used in the calculation:

H	= 83.2 in. (Location of Center of Gravity)	
W_{vcc}	= 206,100 lb. (VCC 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}
I	$= \pi (D^4 - ID^4)/64$	$= 10.492 \times 10^6$ in.^{4}
S_{outer}	$= 2I/D$	$= 163{,}937.5$ in.^{3}
S_{inner}	$= 2I/(ID)$	$= 244{,}000.0$ in.^{3}
w	$= a_x W_{vcc} / 159.92$	$= 456.22$ lb / in.

The maximum bending moment (M) at support is:

$$M = w(159.92)^2/2 = 5.834 \times 10^6 \text{ in.-lb}$$

With $a_y = +0.25$g, the maximum tensile stress at the outer and inner surfaces of the concrete shell are:

$$\sigma_{v\,outer} = (M / S_{outer}) + (a_y (W_{vcc})/A) = +35.59 + 7.30 = 42.9 \text{ psi}$$
$$\sigma_{v\,inner} = (M / S_{inner}) + (a_y (W_{vcc})/A) = +23.91 + 7.30 = 31.2 \text{ psi}$$

With $a_y = \pm 0.25g$, the maximum compressive stress at the outer and inner surfaces of the concrete shell are:

$$\sigma_{v\,outer} = (M / S_{outer}) + (a_y (W_{vcc}) / A) = -35.59 - 7.30 = -42.9 \text{ psi}$$
$$\sigma_{v\,inner} = (M / S_{inner}) + (a_y (W_{vcc}) / A) = -23.91 - 7.30 = -31.2 \text{ psi}$$

The compressive stresses are included in the load combination No. 5 in Table 3.4.4.2-1, since they are governing stresses for the load combination. As shown in Tables 3.4.4.2-1 and 3.4.4.2-2, the maximum combined stresses for the load combination of dead, live, thermal and earthquake are below the allowable stress.

11.2.2.3 Radiological Consequences

There are no radiological consequences for this accident.

11.2.2.4 NAC-MPC Performance

This analysis shows that the Yankee NAC-MPC vertical concrete cask performance is not affected by the design basis earthquake. The vertical concrete cask does not tip over for the design-basis earthquake having ground accelerations of 0.25 g.

11.2.2.5 Recovery and/or Corrective Actions

Inspection of the storage casks is required following an earthquake accident. While the cask does not tip over, there is a potential for movement of a cask relative to other casks and for superficial damage at the bottom edge due to that movement. The temperature monitoring system should be checked for operation as movement of a cask could have disconnected the monitoring system.

11.2.3 Explosion

The flood analysis presented in Section 11.2.6 shows that the NAC-MPC system would not experience adverse effects due to a pressure of 22 psig applied to the canister. The vertical concrete cask will also be unaffected. This pressure is considered to bound any explosions occurring in the vicinity of the ISFSI.

11.2.3.1 Cause of Accident

An explosion is an unlikely event because administrative controls will exclude explosive substances in the vicinity of the ISFSI. No flammable or explosive substances are stored or used at the storage facility; therefore, an explosion affecting the site is extremely unlikely. This evaluation is provided in order to provide a bounding pressure that could be used in the event that the potential of an explosion must be considered at a given site.

11.2.3.2 Evaluation of the Explosion Event

The NAC-MPC canister shell was evaluated in Section 11.2.6 for the effects of a flood having a depth of 50 feet. The water exerts an external hydrostatic pressure of 22 psig on the canister, which results in stress in the canister shell.

The maximum primary membrane stress calculated in the canister is 8.82 ksi. The allowable stress for accident conditions is 40.08 ksi. The margin of safety for primary membrane stress is + 3.54.

The maximum primary membrane plus bending stress calculated in the canister is 19.18 ksi. The allowable primary membrane plus bending stress for accident conditions is 60.12 ksi. The margin of safety for primary membrane plus bending stress is + 2.13.

Consequently, there is no adverse consequence to the canister as a result of the 22 psig external pressure. This pressure conservatively bounds an explosion event.

The concrete cask is a monolithic structure that is not affected by the explosion overpressure.

11.2.3.3 Radiological Consequences

There are no radiological consequences for this accident.

11.2.3.4 NAC-MPC Performance

This analysis shows that the NAC-MPC system performance is not affected by explosion over pressure.

11.2.3.5 Recovery and/or Corrective Actions

In the unlikely event of a nearby explosion, inspection of the storage casks is required to ensure that the air inlets and outlets are free of debris and to ensure that the monitoring system is intact. There are no recovery or corrective actions required for this accident event.

11.2.4 Failure of All Fuel Rods With a Subsequent Ground Level Breach of the Canister

This section addresses the potential mechanistic [illegible] radioactive gas, volatile and particulate material from the canister.

As described in Chapters 3, 4, and 11, the NAC-MPC is evaluated for normal conditions and for a series of off-normal and accident events that include cask tip over, [illegible] explosion, lightning, earthquake, loss of shielding, adiabatic heat up, and tornado generated missiles. The evaluations show that for these design basis events, there is no mechanistic failure of the confinement boundary of the canister, i.e., the canister maintains its structural integrity. As described in Chapter 7 and in Section 8.1.1, the canister is tested to demonstrate that it is leaktight as defined by ANSI N14.5-1997.

Therefore, no further evaluation of this potential accident condition is required.

11.2.5 Fire Accident

This section evaluates the effects of a hypothetical fire accident as a bounding condition. A fire accident is a very unlikely occurrence in the storage cask lifetime.

11.2.5.1 Cause of Accident

There is no probable cause for this accident event. There are no flammable materials in the area of the ISFSI. While it is possible that a transport vehicle could start a fire while transferring a loaded storage cask at the ISFSI, this fire would be confined to the vehicle and would be rapidly extinguished by the persons performing the transfer operations.

Detection of the event would be by observation of fire or smoke.

11.2.5.2 Accident Analysis

The bounding hypothetical fire accident condition is [redacted] following conditions are applied:

1) The air inlets are not blocked; air flows into the inlets, [redacted] surface, and out the outlets.

2) A temperature of 1475°F is applied to the boundary [redacted] up to an elevation of 10 inches above the base, to force [redacted] flows through the inlets. Both the top and bottom [redacted] temperature specification of 1475°F applied to them.

3) The fire condition is applied for 8 minutes.

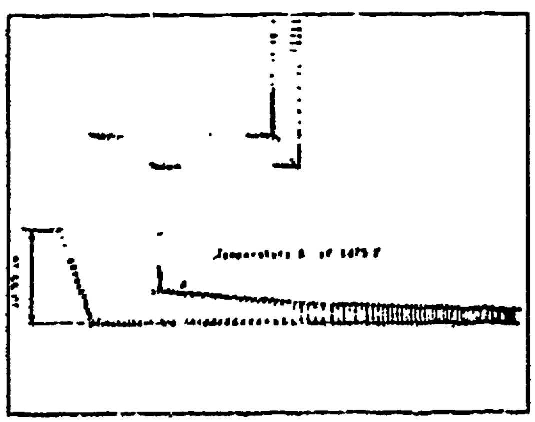
4) After the fire condition, the air inlet temperature is returned to 75°F.

5) Solar insolation is applied during the fire, since the fire condition is applied only to the base of the concrete cask.

[redacted]

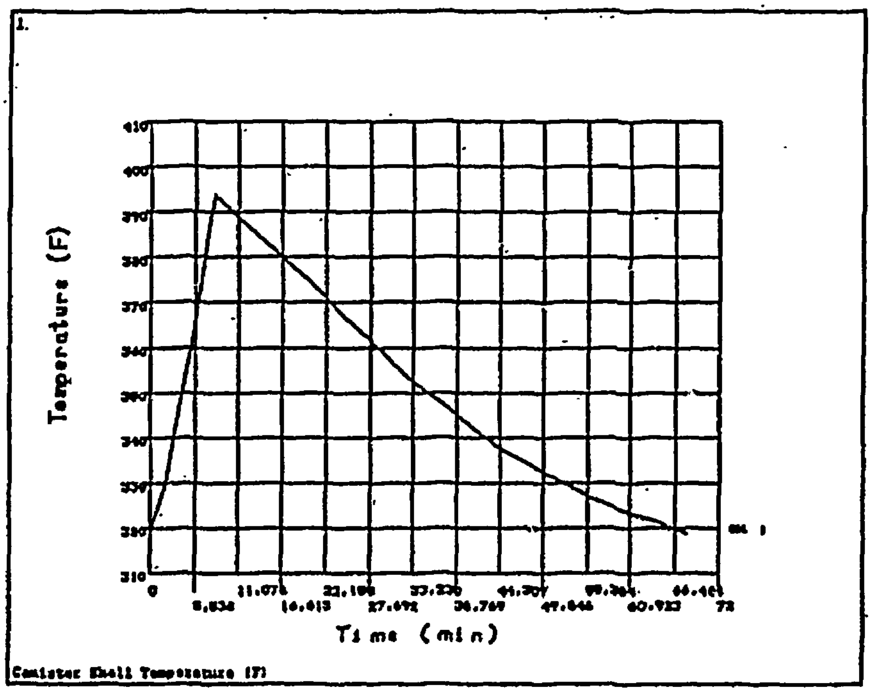
[redacted]

[redacted]



This condition is applied for 8 minutes. After the 8-minute period, the 1475°F boundary condition is removed and the external conditions at the time of normal operating steady state conditions are applied. This initiates the cool down phase, which is analyzed for an additional 60 minutes.

[redacted]



[redacted]

Component	Steady State Temperature (°F)	[redacted]	[redacted]
Canister shell	319	[redacted]	[redacted]
Heat transfer disk	327	[redacted]	[redacted]
Support disk	339	[redacted]	[redacted]
Fuel cladding	363	[redacted]	[redacted]
Bulk concrete	133	[redacted]	[redacted]

[redacted]

11.2.5.3 Radiological Consequences

There are no significant radiological consequences for this accident. There may be local spalling of concrete during the fire event, which could lead to some minor reduction in shielding effectiveness. The principal effect would be local increases in radiation dose rate on the [redacted] cask surface.

11.2.5.4 NAC-MPC Performance

The peak temperature of the [redacted] is [redacted]°F, which is reached [redacted] after the fire starts. [redacted] The concrete temperature increases only slightly during the hypothetical event. [redacted] is much less than the short-term allowable fuel temperature of [redacted]°F. The fuel, canister, and support disk are not significantly affected by the accident.

Even for the severe 10 CFR 71 hypothetical thermal accident (fire), the NAC-MPC meets its storage performance requirements.

11.2.5.5 Recovery and/or Corrective Actions

In the unlikely event of such a severe fire at an ISFSI site, the operator will take appropriate immediate response to suppress and extinguish the fire. Following the fire, the concrete cask should be inspected for general deterioration of the concrete, loss of shielding (spalling of concrete), exposed reinforcing bar, and surface discoloration that could affect heat rejection. This inspection would determine the repair activities necessary to return the concrete storage cask to its design basis configuration.

11.2.6 Flood

This evaluation shows that the NAC-MPC system is not adversely affected by a design basis flood having a depth of water of 50 feet and a flow velocity of 15 feet per second.

11.2.6.1 Causes of Flood

The probability of a flood event is highly dependent on land contour and environmental factors that are specific to each ISFSI site. Most reactor sites are not susceptible to flooding as a result of site selection characteristics. This evaluation considers design basis flood conditions of a 50-foot depth of water having a velocity of 15 feet per second. This flood is fully immersing for the NAC-MPC system.

11.2.6.2 Flood Analysis

The concrete cask is considered to be resting on a flat level concrete pad when subjected to a flood velocity pressure distributed uniformly over the projected area of the concrete cask. Because of the concrete cask geometry, rigidity, and large mass, it is analyzed as a rigid body. Assuming full immersion of the concrete cask and steady-state flow conditions, the drag force, F_D, is calculated using classical fluid mechanics for turbulent flow conditions. A safety factor of 1.1 for stability against overturning and sliding is applied. The coefficient of friction between carbon steel and concrete used in this analysis is 0.35 (Funk).

The buoyancy force, F_b, is calculated from the weight of water (62.4 lb/ft^3) displaced by the fully immersed concrete cask. The displacement volume of the concrete cask containing the loaded canister is 1030 ft^3. The displacement volume is the occupied volume less the free space in the central annular cavity of the concrete cask.

$$\begin{aligned} F_b &= \text{Vol} \times 62.4 \text{ lb/ft}^3 \\ &= 64.27 \text{ kip} \end{aligned}$$

Assuming the steady-state flow conditions for a rigid cylinder, the total drag force of the water on the concrete cask is given by the formula:

$$F_D = (C_D)(\rho)(V^2)\left(\frac{A}{2}\right) \qquad \text{(Roberson)}$$
$$= 21.72 \text{ kip}$$

where,

C_D = Drag coefficient, which is dependent upon the Reynolds Number (Re): For flow velocities greater than 6 ft/sec, the value of C_D approaches 0.7. (Roberson)
ρ = mass density of water = 1.94 slugs/ft^3
D = VCC outside diameter (128 in. / 12 = 10.67 ft)
V = velocity of water flow (15 ft/sec)
A = projected area of the VCC normal to water flow (10.67 ft × 13.33 ft = 142.2 ft^2)

The drag force required to overturn the concrete cask is determined by summing the moments of the drag force and the submerged weight (weight of the cask less the buoyant force) about a point on the bottom edge of the cask. This method assumes a pinned connection, i.e., the cask will rotate about the point on the edge rather than slide. When these moments are in equilibrium, the cask is at the point of overturning.

$$F_D \times \left(\frac{h}{2}\right) = (W_{VCC} - F_b) \times r$$

and

$$F_D = 113.42 \text{ kip}$$

where:

h = concrete cask overall height (159.92 in., use 160 in. / 12 in./ft = 13.33 ft)
W_{VCC} = concrete weight = 206.1 kips
F_b = buoyant force = 64.27 kips
r = concrete cask radius (64 in. / 12 in./ft = 5.33 ft)

Solving the drag force equation for the velocity, V, that is required to overturn the concrete cask.

$$V = \sqrt{\frac{2F_D}{C_D \rho A}}$$

= 32.68 ft/sec. (including safety factor of 1.1)

To prevent sliding, the minimum coefficient of friction (with a safety factor of 1.1) between the carbon steel bottom plate of the concrete cask and the concrete surface upon which it rests is,

$$\mu_{min} = \frac{(1.1)F_{D15}}{F_y}$$

where, F_y = the immersed weight of the concrete cask

$$\mu_{min} = \frac{(1.1)21.72\text{kip}}{(206.10 - 64.27)\text{kip}} = 0.17$$

The analysis shows that the minimum coefficient of friction, μ, required to prevent sliding of the concrete cask is 0.17. The coefficient of friction between the steel bottom plate of the concrete cask and the concrete surface of the storage pad (0.35) is greater than the coefficient of friction required to prevent sliding of the concrete cask. Therefore, the concrete cask will not slide under design-basis flood conditions.

The water velocity required to overturn the concrete cask is greater than the design-basis velocity of 15 ft/sec. Therefore, the concrete cask will not be overturned under design basis flood conditions.

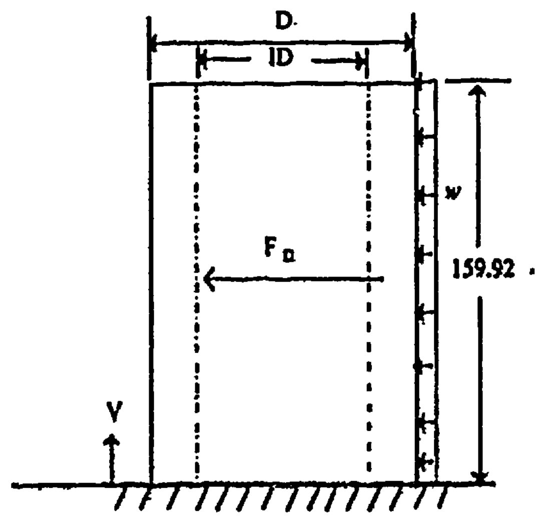
The flood depth of 50 feet exerts a hydrostatic pressure on the canister and the concrete cask. The water exerts a pressure of 22 psig (50 x 62.4/144) on the canister, which results in stresses in the canister shell.

The maximum primary membrane stress in the canister is 8.82 ksi. The allowable stress for accident conditions (2.4S m) is 40.08 ksi The margin of safety for primary membrane stress is +3.54.

The maximum primary membrane plus bending stress is 19.18 ksi. The allowable primary membrane plus bending stress for accident conditions (3.65m) is 60.12 ksi. The margin of safety is +2.13.

Consequently, there is no adverse consequence to the canister as a result of the hydrostatic pressure due to the flood condition.

The concrete cask is a thick monolithic structure and is not affected by the design basis flood hydrostatic pressure. However, the stresses in the concrete due to the drag force (F_D) are conservatively calculated as shown below. The concrete cask is considered to be fixed at its base.



F_D	$= 21{,}720$ lb		
D	= 128 in. (concrete exterior diameter)		
ID	= 86 in. (concrete interior diameter)		
A	$= \pi(D^2 - ID^2)/4$	$= 7{,}059.2$ in.^{2}	
I	$= \pi(D^4 - ID^4)/64$	$= 10.492 \times 10^6$ in.^{4}	(Moment of Inertia)
S_{outer}	$= 2I/D$	$= 163{,}937.5$ in.^{3}	(Section Modulus for inner surface)
S_{inner}	$= 2I/(ID)$	$= 244{,}000.0$ in.^{3}	(Section Modulus for outer surface)
w	$= F_D/159.92$	$= 135.82$ lbf/in.	
M	$= w(159.92)^2/2$	$= 1.737 \times 10^6$ in.-lb	(Bending Moment at the base)

Maximum stresses at the base surface:

$\sigma_{V_{outer}} = M / S_{outer} = 10.6$ psi (tension or compression)
$\sigma_{V_{inner}} = M / S_{inner} = 7.1$ psi (tension or compression)

The compressive stresses are included in load combination No. 7 in Table 3.4.4.2-1, since they are the governing stresses for the load combination. As shown in Tables 3.4.4.2-1 and 3.4.4.2-2, the maximum combined stresses for the load combination due to dead, live, thermal and flood loading, are below the allowable stress.

11.2.6.3 Radiological Consequences

There are no radiological consequences for this accident.

11.2.6.4 NAC-MPC Performance

This analysis shows that the Yankee NAC-MPC vertical concrete cask system performance is not affected by the design basis flood. The analysis demonstrates that the concrete cask will not slide and will not overturn in the design-basis flood. The hydrostatic pressure exerted by the 50 foot depth of water does not produce significant stress in the canister.

11.2.6.5 Recovery and/or Corrective Actions

Inspection of the storage casks is required following this accident event. While the casks do not tip over or slide, there is a potential for the collection of debris or the accumulation of silt at the base of the cask, which could clog or obstruct the air inlets. Operation of the temperature monitoring system must be verified, as flood conditions may have impaired its operation.

11.2.7 Fresh Fuel Loading in the Canister

This section evaluates the effects of an inadvertent loading of up to 36 fresh, unburned Yankee class fuel assemblies in the canister. This event is not a credible event.

11.2.7.1 Cause of Accident

The cause of this event would be operator and/or procedural error. The design basis criticality condition demonstrates that the canister is designed to accommodate fresh fuel without a resulting criticality event.

This accident is expected to be identified immediately by observation of the condition of the fuel installed in the canister or by a review of the fuel handling records.

11.2.7.2 Analysis of Fresh Fuel Loading in the Canister

The criticality analysis presented in Chapter 6 assumes the loading of up to 36 Yankee class fuel assemblies having no burn up. This analysis shows that the maximum K_{eff} for the canister in the dry normal condition is 0.4463. The maximum K_{eff} in the accident conditions is 0.8978. The accident condition assumes the most reactive configuration of the fuel and full moderator intrusion.

The design of the NAC-MPC is adequate to preclude any effects due to this accident condition.

11.2.7.3 Radiological Consequences For This Accident

There are no radiological consequences for this event.

11.2.7.4 NAC-MPC Performance

The criticality control features of the NAC-MPC canister and basket ensure that the K_{eff} of the fuel is less than 0.95 for all loading conditions of fresh fuel. There is no adverse impact on the NAC-MPC due to this event.

11.2.7.5 Recovery and Corrective Actions

This event requires that the canister be unloaded when the incorrect loading is identified. The controls placed on the movement of fuel assemblies is such that this accident event will not occur.

11.2.8 Full Blockage of Air Inlets and Outlets

This section evaluates the NAC-MPC for the steady state effects of full blockage of the air inlets and outlets at the normal ambient temperature (75°F). It estimates the duration of the event that would result in the concrete reaching its design basis limiting temperature of 350°F.

11.2.8.1 Cause of Event

The likely cause of complete air inlet and outlet blockage is the covering of the cask with earth in a catastrophic event such as greater than design basis earthquake or a land slide. This event is a bounding condition accident that is not credible.

This event would be detected by inspection of general conditions at the site following such an event. It would be detected visually by the persons inspecting the ISFSI site.

11.2.8.2 Analysis of the Blockage Event

The accident temperature conditions are evaluated using the thermal models described in Section 4.4.1. The analysis assumes initial normal storage conditions, with the sudden loss of convective cooling of the canister. Heat is then rejected from the canister to the storage cask liner by radiation and conduction. The loss of convective cooling results in the fairly rapid and sustained heat-up of the canister and the concrete cask. To account for the loss of convective cooling in the ANSYS air flow model (Section 4.4.1.1), the elements in the model were replaced with thermal elements, employing the axis-symmetric option. This option assures that there is no axial temperature difference across the element. This model is used to evaluate the thermal transient resulting from the postulated boundary conditions.

The thermal transient analysis evaluated the concrete temperature conditions for 100 hours. At the end of 24 hours, the concrete temperature is 287°F, an increase of 122°F, compared with the initial (normal conditions) concrete temperature, 165°F. The maximum temperature in the concrete will exceed the thermal design criteria temperature of 350°F at 45.7 hours after the start of the event.

Other conservative assumptions include no internal convection and no heat transfer to the surrounding environment (i.e., the dirt covering the cask in the case of a landslide).

[redacted]

Considering the maximum temperature difference of 250°F between the canister shell and fuel region (as shown in Table 4.1-4, the ΔT between canister shell and fuel clad is 563 - 319 = 244°F for normal condition of storage), the maximum fuel temperature will reach [redacted]°F ([redacted]) in the same 45.7-hour period, which is less than the short-term allowable temperature of [redacted]°F.

11.2.8.3 Radiological Consequences

There are no significant radiological consequences for this event, as the NAC-MPC retains its shielding performance. Dose is incurred as a consequence of uncovering the storage cask and vent system. Since the dose rates at the air inlets and outlets are higher than the nominal rate (35 mrem/hr) at the cask wall, personnel will be subject to an estimated maximum dose rate of 100 mrem/hr when clearing the inlets and outlets. If it is assumed that a worker kneeling with his hands on the inlets or outlets would require 15 minutes to clear each inlet or outlet, the estimated extremity dose is 200 mrem for the 8 [illegible]. The whole body dose would be slightly less. In addition, some dose is incurred cleaning debris away from the cask body. This dose is estimated at 50 mrem, assuming 2 hours is spent near the cask exterior surface.

11.2.8.4 NAC-MPC Performance

There are no adverse consequences for this accident condition, assuming that debris is cleared within a reasonable time (< 45 hours). The maximum component temperatures are less than the allowable temperatures. The NAC-MPC continues to satisfactorily perform the cooling function with all the inlets and outlets blocked.

11.2.8.5 Recovery and Corrective Actions

The debris blocking the vents shall be manually removed. In addition, a considerable effort may be involved in clearing the area around storage casks. No actions are required with regard to the casks proper, provided that the vents are cleared within 45 hours.

11.2.9 Lightning

This analysis demonstrates that the NAC-MPC storage cask does not experience adverse effects due to a lightning strike.

11.2.9.1 Cause of Accident

A lightning strike is a random weather related event. Since the NAC-MPC storage cask is located on an unsheltered pad, the storage cask may be subject to a lightning strike. The probability of a lightning strike is primarily dependent on the geographical location of the ISFSI site, as some geographical regions experience a higher frequency of storms containing lightning than others.

The analysis of the consequence of a lightning strike assumes that the lightning strikes the upper most metal surface and proceeds through the storage cask liner to the ground. The electrical current flow path results in current induced Joulean heating along that path.

A lightning strike on a storage cask may be visually detected at the time of the strike, or by visible surface discoloration at the point of entry or exit of the current flow. Most reactor sites in locations experiencing a frequency of lightning bearing storms have lightning strike detection systems as an aid to ensuring stability of site electric power.

11.2.9.2 Analysis of the Lightning Strike Event

The current path analyzed is from a strike point on the outer radius of the [illegible] flange [illegible] storage cask, down through the carbon steel liner and the bottom plate to the ground.

The integrated maximum current for a lightning strike is a peak current of 250 kiloamps over a period of 260 microseconds, and a continuing current of up to 2 kiloamps for 2 seconds in the case of severe lightning discharges ([illegible]).

From Joule's Law, the amount of thermal energy developed by the combined currents is given by ([illegible])

$$Q = 0.0009478\,R\left[I_1^2\left(dt_1\right) + I_2^2\left(dt_2\right)\right]$$
$$= (22.98 \times 10^3)\ R\ \text{Btu}$$

where,

Q = thermal energy (BTU)
I_1 = peak current (amps)
I_2 = continuing current (amps)
dt_1 = duration of peak current (seconds)
dt_2 = duration of continuing current (seconds)
R = resistance (ohms)

The maximum lightning discharge is assumed to attach to the smallest current-carrying component, that is, the top flange connected to the outer lid.

The propagation of the lightning through the carbon steel cask liner, which is both permeable and conductive, is considered to be a transient. For static conditions, the current would be distributed throughout the shell. In a transient condition the current will be near the surface of the conductor. It is assumed that the current is contained in a 90 degree sector of the circular cross section of the steel liner as opposed to the entire cross section. The depth of the current penetration (δ) is estimated as (Fink):

$$\delta = \frac{1}{\sqrt{\pi\mu f\sigma}}\ \text{(m)}$$

where,

μ = permeability of the conductor = $100\mu_0$ (Summer)
and μ_0 = 4πE-7 Henries/m
σ = electrical conductivity (siemens/meter) = $1/\rho$
ρ = resistivity = 1.978E-08 (ohm-m) (Fink)
f = frequency of the field (Hz)

The pulse is represented conservatively as a half sine form, so that the equivalent $f = 2 \times \tau$, where τ is the referenced pulse duration. There are two skin depths to be computed, corresponding to different pulse duration. The larger effective frequency will result in a smaller effective area to conduct the current. The effective resistance is computed as (Fink):

$$R = \frac{\rho l}{a}$$

where,

$R =$ resistance (ohms)
$\rho =$ resistivity = 9.78E-08 (ohm-m)
$l =$ length of conductor path
$a =$ area of conductor (m^2)
$\quad = (R_{shell})(\delta)(\pi/4)$
$R_{shell} =$ 35.75 inches = 0.9081 m

Using the current level of the pulse and the duration in conjunction with the carbon steel liner, the resulting energy into the shell is:

	Peak	Continuous	Expression
Time (s) τ	2.60 E-05	2.0	----------
Peak Current (KA)	250	2	----------
Frequency (Hz)	2E4 (conservative)	2.5E-01	$1/(2\tau)$
Skin depth (m) δ	3.52E-04	3.15E-02	$1/\sqrt{\pi\mu f\sigma}$
Conductor length (m) L	4.1	4.1	----------
Conductor area (m^2) A_s	2.51E-4	2.24E-2	$R_{shell}\delta(\pi/4)$
Resistance (ohm) Ω	1.60E-03	1.79E-5	$(L/A)\rho$
Q (Btu)	24.6	14	$R = \frac{\rho l}{a}$

It is conservatively assumed that this thermal energy dissipation occurs in the localized volume of the carbon steel involved in the current flow path through the flange to the inner liner. Assuming no heat loss or thermal diffusion beyond the current flow boundary, the maximum temperature increase in the flange due to this thermal energy dissipation is calculated as (Black):

$$\Delta T = \frac{Q}{mc}$$

where,

$\Delta T =$ temperature change (°F)
$Q =$ thermal energy (BTU)
$c =$ 0.113 Btu/lb °F
$m =$ mass (lbm)

The ΔT_1 for the peak current (250KA, 260 μsec) for the region of the current is computed based on the mass associated with this region.

$$m = (.284\ \mathrm{lb/in^3}) \times A_c \times L = (.284)(2.51E\text{-}4)(159.92)/C_f = 17.7\ \mathrm{lbm}$$

$$C_f = (.0254^3)\ \text{(converts m}^3\text{ to in}^3\text{)}$$

$$\Delta T_1 = 24.6/(17.7 \times .113) = 12.3°F$$

The ΔT_2 for the continuous current (2KA, 2 sec) for the region of the current is computed based on the mass associated with this region

$$m = (.284\ \mathrm{lb/in^3}) \times A_c \times L = (.284)(2.24E\text{-}2)(159.92)/(.0254^3) = 1577\ \mathrm{lbm}$$

$$\Delta T_2 = 14/(1577 \times .113) = 0.001°F$$

The ΔT_1 would correspond to the increase in the maximum temperature of the steel. For the concrete to experience an increase in temperature, the heat must disperse from the steel liner surface throughout the steel. Using the total thickness of the steel, over the 90 degree sector, the increase in temperature is:

$$m = (.284\ lb/in^3) \times (39.25^2 - 35.75^2)\pi/4 \times L = 9{,}364\ lbm$$

$$\Delta T_1 = (24.6+.14)/(9{,}364 \times .113) = 0.02\ F$$

Therefore the increase in temperature of the concrete is not significant.

11.2.9.3 Radiological Consequences

There are no radiological consequences for this accident.

11.2.9.4 NAC-MPC Performance

This analysis shows that the NAC-MPC vertical concrete cask's performance is not affected by a lightning strike. When lightning strikes the cask and travels through the 3.5-inch thick carbon steel storage cask liner, the maximum increases in the steel and concrete temperatures is 12.3°F and 0.0°F, respectively.

11.2.9.5 Recovery and/or Corrective Actions

There are no recovery or corrective actions required for this accident event.

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

This analysis evaluates the NAC-MPC response to storage operation with an ambient temperature of 125°F. The NAC-MPC is evaluated at steady state conditions.

11.2.10.1 Cause of Accident

The cause of this condition is a weather event that causes the NAC-MPC to be subject to a 125°F ambient temperature with full-solar insolance. A contents heat load of 12.5 kW is assumed.

The condition is analyzed in accordance with the requirements of ANSI/ANS 57.9 to evaluate a credible worse case thermal loading. This condition is considered in the load combinations described in Chapter 2.

Detection of the ambient high temperature condition would occur during the daily measurement of ambient temperature and storage cask outlet air temperature.

11.2.10.2 Analysis of the 125°F Ambient Temperature Event

The severe high temperature condition is evaluated using the thermal models described in Section 4.4.1. The principal component temperatures for this ambient condition are:

Component	125°F Ambient Max Temp. (°F)	Allowable Max Temp. (°F)
Fuel Cladding	607	1058
Support Disks	575	800
Heat Transfer Disks	574	700
Canister Shell	372	800
Concrete	228	350

This evaluation shows that the component temperatures are within the allowable temperature for the severe high ambient temperature conditions.

This event is considered an "accident condition" for the purpose of this evaluation. Consequently, the thermal stress (secondary) occurring in the canister due to this thermal loading is not evaluated.

However, the thermal stresses for the concrete cask are calculated using the same methodology as shown in Section 3.4.4.2 for the stress calculation for normal conditions of storage. The same finite element model is used with the boundary temperatures based oncrete cask thermal analysis results (Section 4.4.4.1). The maximum vertical and c rential (compressive) stresses at the concrete inside surface are calculated to be 660.1 psi and 127.5 psi, respectively. These stresses are included in the load combination (No. 4) as presented in Table 3.4.4.2-1. The maximum tensile forces in the vertical and hoop reinforcing bars are 19,380 lbs. and 23,196 lbs., respectively. Comparison of these forces and the allowable forces is shown in Table 3.4.4.2-2. As shown in Tables 3.4.4.2-1 and 3.4.4.2-2, the maximum combined concrete stress and reinforcing bar force for the load combination of dead (D), live (L) and accident temperature (T_a) are below the allowable stress and force, respectively.

11.2.10.3 Radiological Consequences

There are no radiological consequences for this event.

11.2.10.4 NAC-MPC Performance

There are no adverse consequences for this accident condition. The maximum component temperatures are less than the allowable temperatures for accident conditions, and are also less than the temperature limits for normal conditions of storage.

11.2.10.5 Corrective Actions

No corrective actions are required for this accident condition.

11.2.11 Storage Cask 6-Inch Drop

This analysis evaluates a loaded concrete storage cask for a 6-inch drop onto a concrete storage pad. This evaluation shows that neither the concrete storage cask nor the ...nister experience significant adverse effects due to the 6-inch drop accident

11.2.11.1 Cause of Accident

The concrete storage cask, containing the loaded canister, must be raised approximately 3-inches in order to install the inflatable air-pads beneath it. The air pads use pressurized air to allow the cask to be moved across the surfaces of the transporter and the ISFSI pad to the designated positions. The cask is raised using hydraulic jacks installed at jack-points in the air inlets.

This accident assumes the failure of one or more of the jacks or of the air pad system, resulting in a drop of the cask. A lift of about 3-inches is required to install and remove the air pads. This analysis conservatively evaluates the consequences of a 6-inch drop.

This accident would be detected by persons jacking or moving the storage cask.

11.2.11.2 Analysis of the 6-Inch Drop Event

A bottom end impact is assumed to occur normal to the concrete pad surface, transmitting the maximum g-load to the concrete storage cask and the canister. The energy absorption is computed as the product of the compressive force acting on the storage cask and its displacement. Conservatively assuming the storage pad is an infinitely rigid surface, exhibiting a uniform isotropic crush strength, the concrete body of the storage cask will crush until the impact energy is absorbed.

A compressive strength of 4,000 psi is used for the storage cask concrete. This is conservative, since it is the minimum specified value and it ignores any energy absorption by the internal friction of the aggregate as crushing proceeds.

The canister rests upon a stand/inlet system, shown in Figure 11.2.11-1, designed to allow cooling of the canister. Following the initial impact, the inlet system will partially collapse, providing an energy absorption mechanism that somewhat reduces the deceleration force on the canister. A detailed evaluation is provided in Section 11.2.11.2.2. The weight of the empty concrete storage cask is 151,364 pounds. The loaded canister weight is 54,730 pounds.

11.2.11.2.1 Evaluation of the Concrete Storage Cask

In the 6-inch bottom drop, the cylindrical portion of the concrete is in contact with the steel bottom baseplate that also supports the stand/inlet system. The baseplate is assumed to be part of an infinitely rigid storage pad. No credit is taken for the crush properties of the storage pad or the underlying soil layer. Therefore, energy absorbed by the crushing of the cylindrical concrete region of the storage cask equals the product of the compressive strength of the concrete, the crush depth of the concrete, and the projected area of the concrete cylinder. Crushing of the concrete continues until the energy absorbed equals the potential energy of the storage cask at the initial drop height. The canister is not rigidly attached to the concrete cask, so it is not considered to contribute to the concrete crushing. The energy balance equation is

$$w(h+\delta) = P_e A \delta,$$

where,

h = 6 in., the drop height

δ = the crush depth of the concrete cask

P_e = 4000 psi, the compressive strength of the concrete

A = $\pi(R_1^2 - R_2^2)$
= 7,059 in^2, the projected area of the concrete shield wall

w = 151,364 lb × 1.1, the weight of the concrete cask, conservatively multiplied by 1.1 to account for variations in the concrete

It is assumed that the maximum force that can be exerted on the concrete cask is the compressive strength of the concrete multiplied by the area of the concrete being crushed.. Therefore, the deceleration of the storage cask is dependent upon the compressive strength of the concrete, the cross-sectional area of the storage cask cylindrical body, and the weight of the empty storage cask with lid. The deceleration due to crush is:

$$a_{ysc} = \frac{P_o A}{W} = 169.6 \quad g's.$$

The crush distance computed from the energy balance equation is:

$$\delta = \frac{hw}{P_o A - w} = \frac{(6)(151{,}364 \times 1.1)}{(4000)(7{,}059) - (151{,}364 \times 1.1)} = 0.036\text{inch}$$

11.2.11.2.2 Evaluation of the Canister for a 6-inch Bottom-End Drop of the Storage Cask

Upon a bottom-end impact of the concrete storage cask on the storage pad, the canister produces a force on the stand/inlet system located near the bottom of the storage cask. It is expected that the steel plate above the air inlet channels will yield. Calculating the flow stress, σ_f, for the material:

$$\sigma_f = \frac{\sigma_y + \sigma_u}{2}$$

$$= 45{,}400 \text{ psi},$$

where,

σ_u = 58,000 psi, A36 carbon steel ultimate strength at 200 °F.

σ_y = 32,800 psi, A36 carbon steel yield strength at 200 °F.

The flow stress results in the deformation of the steel plate above the air inlet vents, and produces a force through the section. The total force is

$$F = 8(\sigma_f bt),$$

$$= 1.092 \times 10^6 \text{ pounds}$$

where,

b = 1.5 in., the width of the load bearing area

t = 2.0 in., the thickness of the stand

This force results in an acceleration to the loaded canister (weighing w_c = 54,730 pounds) of:

$$g = \frac{F}{w_c} = 19.95$$

The area of contact between the canister bottom and stand is twice as large as the cross-sectional area of the region that yields; therefore, the stand's platform will not be perforated. The expected crush distance, δ, of the crush area is computed by an energy balance equation. The crush distance is:

$$\delta = \frac{hw_c}{P_c A - w_c} = \frac{(6)(54{,}730 \times 1.1)}{(45{,}400)(8)(2)(1.5) - (54{,}730 \times 1.1)} = 0.35 \text{in.}$$

which will reduce the height of the air inlet region by only 0.35 inches.

11.2.11.3 Radiological Consequences

There are no radiological consequences for this accident.

11.2.11.4 NAC-MPC Performance

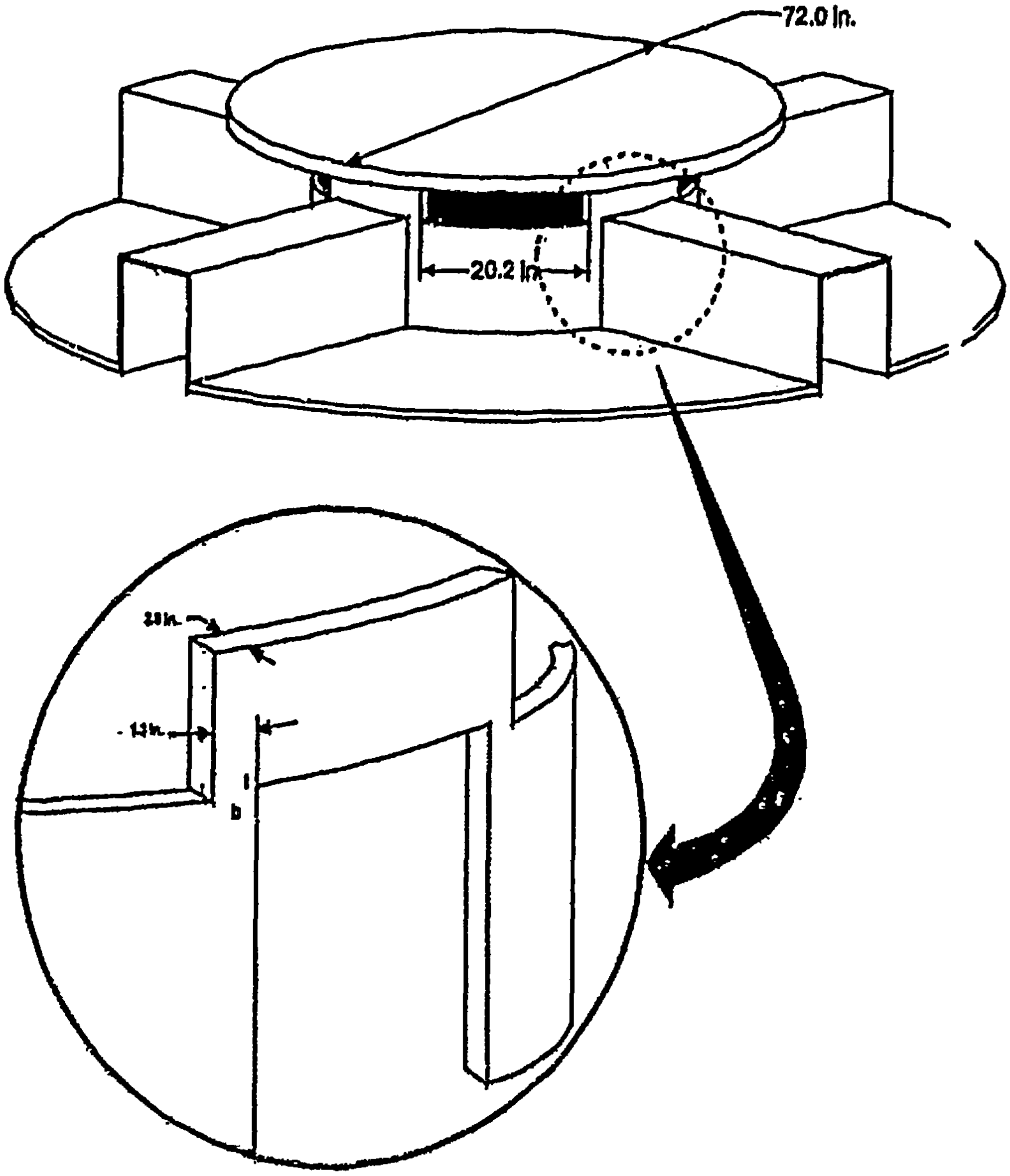
Evaluations of the NAC-MPC concrete storage cask for a 6-inch bottom end drop accident results in a maximum deceleration of 169.6 g for the storage cask, which does not reduce the shielding effectiveness of the cask. The base support, which contains the air inlets, is crushed approximately 0.35 inch. The effect of the reduction of the inlet area by the 6-inch drop is to reduce cooling air flow. The consequences of the loss of one-half of the air inlets is evaluated in Section 11.1.1, which bounds this condition.

The maximum deceleration of 19.95 g for the canister and its internals as a result of the 6-inch storage cask drop, is bounded by the 56.1 g loading evaluated in Section 11.3, where the adequacy of the canister is demonstrated. The NAC-MPC storage cask system is structurally adequate to withstand a 6-inch drop accident.

11.2.11.5 Recovery and/or Corrective Actions

Even though the storage cask system remains functional and no immediate recovery steps are required, the canister should be moved to a new concrete storage cask as soon as one is available. The damaged storage cask should be inspected for stability, and repaired as required prior to continued use.

Figure 11.2.11-1 Storage Cask Base Plate



11.2.12 Tip Over of the [redacted] Cask

This is a hypothetical accident condition that presents a bounding case for evaluation. There are no design basis accidents that result in the tip over of the [redacted] concrete cask [redacted].

11.2.12.1 Cause of Tip Over

A tip over event is possible in an earthquake that exceeds the design basis described in Section 11.2.2. There are no other events that are expected to result in a tip over of the concrete cask.

A tip over of the concrete storage cask would be observed during a survey of the site following the earthquake or other catastrophic event. The tipped over orientation of the concrete cask would be obvious by inspection.

11.2.12.2 Analysis of the Tip Over Event

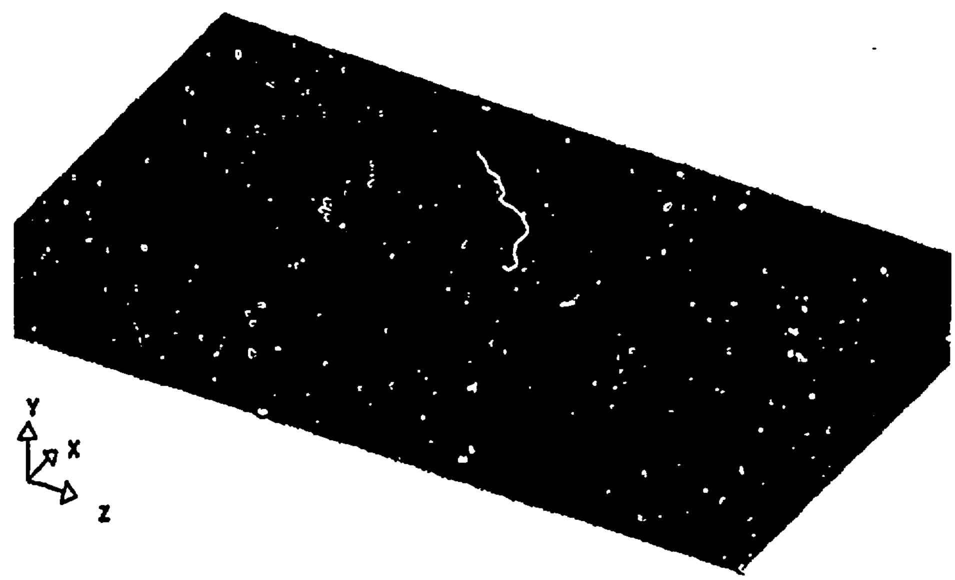
For a tip over event to occur, the center of gravity of the concrete cask and loaded canister must be displaced beyond the outer radius of the concrete cask, i.e., the point of rotation. When the center of gravity passes beyond the point of rotation, the potential energy of the concrete cask and canister will be converted to kinetic energy as the concrete cask and canister rotate toward a horizontal orientation on the ISFSI pad. The subsequent motion of the cask is governed by structural characteristics of the cask, ISFSI pad and underlying soil.

The objective of the evaluation of the response of the concrete cask during the tip over event is to determine the maximum acceleration to be used in the structural evaluation of the loaded canister and basket (Section 11.2.12.3). The methodology to determine the concrete cask response follows the methodology contained in NUREG/CR-6608 ("Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet Onto Concrete Pads"). The LS-DYNA program is used in this evaluation.

11.2.12.2.1 Model Description

As shown below, the finite element model includes a half section of the concrete cask, the concrete ISFSI pad and soil subgrade. The concrete pad in the model corresponds to a pad 30-feet by 30-feet square and 3-feet thick, supporting one concrete cask in the center of the pad. The

[redacted]



11.2.12.2.2 [redacted]

[redacted]

[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

[redacted]

[redacted]

where:

A_{pad} = the area of the concrete pad in square inches

k = 250 psi/in. is the soil stiffness for the soil under the pad

α = 1.08 for a square area

For the 30 x 30 square foot pad, the modulus of elasticity of the soil is calculated as [illegible] psi. The concrete cask steel liner has the properties:

Density	= 0.284 lb/in^3
Poisson's ratio	= 0.31
Modulus of elasticity	= 2.9E7 psi

To account for the weight of the shield plug, the loaded canister, and the concrete cask pedestal, effective densities are used for the elements in the first row of the steel liner elements adjacent to the impact plane of symmetry. These densities represent the regions (6° in the circumferential direction) of the steel liner subjected to the weight of the shield plug, the loaded canister and the pedestal, during the side impact (tip over) condition. The contact angle (6°) is determined based on the canister/basket analysis for the tip over condition (Section 11.2.12.3). Based on the status of the gap elements at the outer surface of the canister shell (representing the interface between the canister shell and the concrete cask steel liner) in the finite element model used in Section 11.2.12.3, the contact angle between the canister shell and the concrete cask liner is between 4.5° and 9° for the half-symmetry model.

11.2.12.2.3 Boundary Conditions and Initial Conditions

A friction coefficient of 0.25 is used at the interface between the steel liner and the concrete shell, between the concrete cask and the pad, and between the pad and the soil. For all the embedded faces (three side surfaces and the bottom surface) of the soil in the model, the displacements in the direction normal to the surface are restrained. The symmetry boundary conditions are applied for all nodes at the plane of symmetry.

The initial condition corresponds to the concrete cask in a horizontal position with an initial vertical velocity into the concrete pad. The pad and soil are initially at rest.

The initial velocity is simulated by applying an angular velocity (ω) of 1.516 radians/sec to the entire cask. The angular velocity value is computed by considering energy conservation at the cask "center of gravity over corner" tip condition versus the side impact condition.

From energy conservation:

$$mgh = \frac{I\omega^2}{2}$$

where,

mg = total weight of the loaded concrete cask = 206,094 lbs

h = height change of the concrete cask center of gravity (L_{CG}) = $\sqrt{R^2 + \left(\frac{L_{CG}}{2}\right)^2} - R$

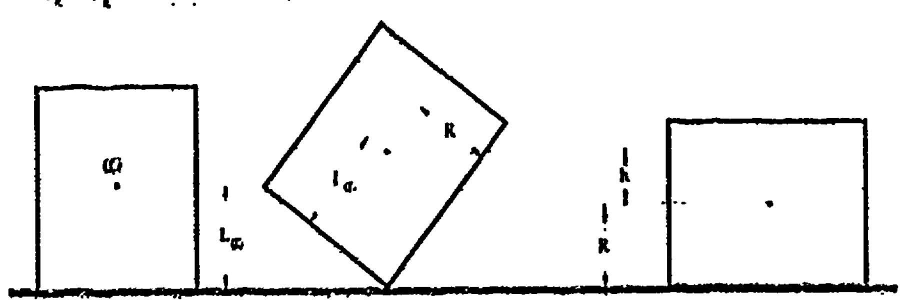
= 40.97 inches

where,

L_{CG} = location of the center of gravity above the pad for the concrete cask = 83.2 inches

R = radius of the concrete cask = 64 inches

I = total mass moment of inertia of the concrete cask about the pivot point = 6,899,360 lb-sec²-inch



Concrete Cask Tip-Over Height Change

[redacted]

[redacted] cylinder is used:

$$I = \frac{m}{12}(3R^2 + 4L^2) + md^2$$

Where,

m = mass of the cylinder
R = radius of the cylinder
L = height of the cylinder
d = distance between the two pivot axes

The angular velocity is given by $\omega = \sqrt{\frac{2mgh}{I}} = 1.516$ radians/sec

11.2.12.2.4 Filter Frequency

The accelerations are evaluated at the inner surface of the [redacted], which physically corresponds to the interface of the liner and the loaded canister [redacted] the plane of impact. Following the methodology contained in NUREG/CR-6608, the [redacted] filter is applied to the nodal accelerations. The filter frequency is based on the [redacted] mode of the cask.

The fundamental natural frequency of a beam in transverse [redacted] is given by Blevins as:

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

$f_s = 31.1\ \lambda^2 = 479.5$ Hz

Area of steel liner $= \pi((43)^2 - (39.5)^2) =$ [redacted]

Moment of inertia of steel liner $= \frac{\pi}{4}((43)^4 - (39.5)^4)$ [redacted]

$f_s = 36.1\ \lambda^2 = 556.6$ Hz

Since the concrete cask is short compared to its diameter, the contribution of the flexibility due to shear is also incorporated. This is accomplished by using Dunkerley's formula (Blevins). The system frequency is:

$$\frac{1}{f^2} = \frac{1}{f_s^2} + \frac{1}{f_b^2}$$

Thus, the system frequency is $f = 363.3$ Hz. A cut-off frequency of 375 Hz is conservatively applied to filter the analysis results and measure the peak accelerations.

[redacted] [redacted]

[redacted]

[redacted]

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

11.2.12.2.6 Validation of the Analysis Methodology

Tip over tests of a steel billet onto a concrete pad were conducted and reported in NUREG/CR-6608. The purpose of the test was to provide data, against which analysis methodology could be validated. Using the geometry described in the benchmark along with the modeling methodology, these analyses were performed using the soil input data in Section 11.2.12.2.2.

Using the filter frequency reported in the benchmark, the following results are obtained.

Comparison of Accelerations of NUREG/CR-6608 Benchmark

Nodes / Gauge Location	Maximum Experiment (g)	NAC Analysis (g)
16115 / A1	237.5	237.1
17265 / A5	231.5	229.4

11.2.12.3 Analysis of the Canister and Basket

Structural evaluations are performed for the transportable storage canister and fuel basket support disks for tip-over accident conditions. An ANSYS finite element model is used to evaluate this side impact loading condition. Based on the evaluation presented in Section 11.2.12.3.3, the bounding basket drop angle of 45° is used for the evaluation of the tip-over accident. Thermal conditions are bounded by evaluating the canister and basket using material properties at [illegible] and applying the stress allowables for the maximum temperature at the extreme heat thermal accident condition.

Comparison of maximum stress results to the allowable stress intensities shows that the canister and support disks are structurally adequate for the concrete cask tip-over condition and satisfies the stress criteria in accordance with the ASME Code, Section III, Division I, Subsection NB and NG, respectively.

11.2.12.3.1 Canister and Basket Model

A finite element analysis, using the ANSYS program, is performed to evaluate the canister and support disks for the tip over accident condition. Output is in the form of linearized stresses taken across component sections. Additionally, a buckling evaluation is performed on the support disk in accordance with NUREG/CR-6322.

A symmetric one-half (180°) model is used to represent the upper portion of the canister and basket assembly, which is subject to the highest g-load during the tip-over accident. As shown in Figure 11.2.12.3-1, the model consists of the upper portion of the canister, the top weldment and the five support disks. The canister is constructed using ANSYS solid (SOLID45) elements. ANSYS SHELL63 elements are used to model the top weldment and the support disks. The model for the canister assembly includes the shield and structural lids. Figure 11.2.12.3-2 shows the basket top weldment and five support disks. A detailed view of the basket support disk showing applied loads is shown in Figure 11.2.12.3-3. For the evaluation of the basket support disk, a basket angle of 45° is considered. This orientation is the most critical during side impact conditions. The weight of the fuel assembly, aluminum heat transfer disks, rods and spacers, and fuel tube (factored by appropriate g-load) are conservatively represented as concentrated forces at mid-span of the ligaments of the support disks.

[redacted] modeled using a ring of COMBIN40 spring gap elements between the shield lid and the canister in the axial direction at the lid lower outside radius. In addition, CONTAC52 elements are used to model interaction between the structural lid and canister shell, and the shield lid and canister shell, just below the respective lid weld joints. The size of the CONTAC52 gaps is determined from the nominal dimensions of contacting components. The COMBIN40 elements between the structural and shield lids are assigned a gap size of 0.08 inch, based on the flatness tolerances of the structural and shield lids. A gap of 1E-8 in. is used between the backing ring and shield lid. Note that the gap sizes in the axial direction do not have a significant effect on the side impact analysis results. All gap-spring elements are assigned a stiffness of 1E6 lb/in.

Symmetry boundary conditions are applied at the plane of symmetry of the model, as well as at the cut boundary of the model (canister shell near the 5th support disk). All nodes on the concrete cask side gap elements at the outer surface of the canister shell are fixed in all degrees of freedom (UX, UY, and UZ). In addition, the axial (UY) and in-plane rotational degrees of freedom (ROTX and ROTZ) of the basket nodes are fixed.

Loading of the model includes an internal pressure of 11.5 psig (normal condition of storage) applied to the canister, concentrated loads are applied to all slots (except the central slot where there is no fuel assembly) and the inertial loads.

For the inertial loads, a maximum acceleration of 45 g is conservatively applied to the entire model. The maximum acceleration of the steel liner at the locations of the top support disk and the top of the canister lid during the tip over event is determined to be 27.5 g and 31.4 g, respectively. The main pulse of deceleration lasts for 30 milliseconds. To determine the effect of the rapid application of the inertia loading for the top support disk, a dynamic load factor (DLF) is computed using the mode shapes of a loaded support disk. The mode shapes are extracted using ANSYS for the first four modes. Using the acceleration time history developed

[redacted]

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]

A uniform temperature of 75°F is applied to the mod[redacted] solution. During post processing, temperatures of [redacted] the support disks and the canister, respectively, to de[redacted] temperatures are the maximum calculated temperatures for [redacted] condition (ambient temperature = 125°F).

11.2.12.3.2 Analysis Results for the Canister

The stress results of the canister for the tip over accident condition are summarized in Tables 11.2.12.3-1 and 11.2.12.3-2 for primary membrane and primary membrane plus bending stresses, respectively. The sectional stresses at 14 axial locations are obtained for each angular division of the model (a total of 41 angular locations for each axial location). The locations for the stress sections are shown in Figure 11.2.12.3-4.

The stress evaluation for the canister is performed in accordance with the ASME Code, Section III, Subsection NB, by comparing the linearized stresses of critical sections of the structure against the allowable stresses. Allowable stresses are computed at a temperature of 372°F (maximum canister temperature is 319°F for normal condition of storage). The allowable stresses for accident conditions are taken from Subsection NB as shown below. S_m and S_u are 16.05 ksi and 59.17 ksi, respectively, for Type 304L stainless steel (canister shell and structural lid). S_m and S_u are 19.06 ksi and 64.85 ksi, respectively, for Type 630 stainless steel (shield lid).

	Accident (Level D) Allowable Stress
P_m	Lesser of 0.7 S_u or 2.4 S_m
P_m+P_b	Lesser of 1.0 S_u or 3.6 S_m

During the tip over accident, the canister shell at the structural and shield lids is subjected to the inertial load of the lids, which results in highly localized bearing stresses (Sections 8 through 12 at angular locations of 0 and 4.5 degrees). This stress is predominant because the weights of the structural and shielding lids are transferred to the canister shell through the thickness of the weld (7/8 inch weld for stuctural lid and 1 inch weld for shield lid). According to ASME Section III, Appendix F, bearing stresses need not be evaluated for Level D service (accident) conditions. Therefore, the P_m stresses are not presented for the region local to the impact (angular locations of 0 and 4.5 degree) for Sections 8, 9, and 10 in Table 11.2.12.3-1. Stresses are conservatively presented for all locations (including bearing region) for Sections 11 and 12 (conservative).

The inertial load of the shield lid also results in localized bending stresses at the canister shell (at the shield lid weld, i.e. Section 8). This stress state is identified as the case of a cylindrical shell at the junction with a head, as shown in Table NB-3217-1 of the ASME Code, Section III, Subsection NB. The localized bending stress at the structural discontinuity is classified as secondary (Q) stress. In accordance with ASME Section III, Appendix F, secondary stresses are not considered for the stress evaluation for Level D (accident) conditions. Therefore, the P_m+P_b stresses are not presented for the region local to the impact for Section 8 (angular locations of 0 and 4.5 degree) in Table 11.2.12.3-2.

The stress evaluation results for tip-over accident conditions show that the minimum margin of safety in the canister is +0.25 for P_m (Sections 12) and +0.10 for P_m+P_b (Section 10).

11.2.12.3.3 Analysis Results for the Support Disk

To evaluate the most critical regions of the support disk, a series of cross sections are considered. To aid in the identification of these sections, Figure 11.2.12.3-5 shows the locations on a support disk. Table 11.2.12.3-3 lists the cross sections versus Point 1 and Point 2, which spans the cross section of the ligament in the plane of the support disk.

The stress evaluation for the support disk is performed according to ASME, Section III, Subsection NG. According to this subsection, linearized stresses of cross sections of the structure are to be compared against the allowable stresses. The allowable stresses for tip over accident conditions are taken from Subsection NG as shown below at the maximum support disk temperature of 575°F (accident-extreme heat). The S_m and S_u are 42.3 ksi and 127.1 ksi, respectively, for 17-4 PH stainless steel.

	Accident (Level D) Allowable Stresses
P_m	Lesser of 0.7 S_u or 2.4 S_m
P_m+P_b	Lesser of 1.0 S_u or 3.6 S_m

The stress evaluation results for the support disks for tip-over impact condition are presented in Tables 11.2.12.3-4 through 11.2.12.3-13. The tables list the 40 highest P_m and P_m+P_b stress intensities for Disks 1 through 5. The minimum margin of safety is +0.21, which occurs in Disk 5 (See Figure 11.2.12.3-4 for identification of support disks). The highest P_m and P_m+P_b stresses occur in Disk 5, with the minimum margin of safety of 1.51 and 0.21, respectively. Locations of the 10 highest P_m and P_m+P_b linearized stresses for Disk 5 are given in Figure 11.2.12.3-6 and 11.2.12.3-7.

11.2.12.3.4 Support Disk Buckling Evaluation

The fuel basket support disks are subjected to compressive and/or inertial loads during impact conditions. For the tip-over accident, the support disks experience in-plane loads. The in-plane loads apply compressive forces and in-plane bending moments on the support disk. Buckling of the support disk is evaluated in accordance with the methods and acceptance criteria of NUREG/CR-6322. Because the ASME Code identifies 17-4PH disk material as ferritic steel, the formulas for non-austenitic steel are used.

[redacted]

[redacted]

- P = applied axial compressive loads, kips
- M = applied bending moment, kips-inch
- P_a = allowable axial compressive load, kips
- P_{cr} = critical axial compression load, kips
- P_e = Euler buckling loads, kips
- P_y = average yield load, equal to profile area times specified minimum yield stress, kips (for normal operating condition)
- C_c = column slenderness ratio separating elastic and inelastic buckling
- C_m = coefficient applied to bending term in interaction equation
- M_m = critical moment that can be resisted by a plastically designed member in the absence of axial load, kip-in.
- M_p = plastic moment, kip-in.
- F_a = axial compressive stress permitted in the absence of bending moment, ksi
- F_e = Euler stress for a prismatic member divided by factor of safety, ksi
- k = ratio of effective column length to actual unsupported length
- l = unsupported length of member, in.
- r = radius of gyration, in.
- S_y = yield stress, ksi
- A = cross sectional area of member, in^2
- Z_x = plastic section modulus, in^3
- λ = allowable reduction factor, dimensionless

From NUREG/CR-6322, the following equations are used to evaluate the support disk:

[redacted]	[redacted]

[redacted]	[redacted]

[redacted]

[redacted]

[redacted]

and $\lambda_c = \frac{1}{\pi}\left(\frac{kl}{r}\right)\sqrt{\frac{S_y}{E}}$	(accident conditions)

$P_e = 1.92 \times A \times F_e$

$$F_e = \frac{\pi^2 E}{1.3\left(\frac{kl}{r}\right)^2}$$	(Level D - Accident)

$P_y = S_y \times A$

$C_m = 0.85$ for members with joint translation (sideways)

$M_p = S_y \times Z$

$$M_m = M_p\left(1.07 - \frac{\left(\frac{l}{r}\right)\sqrt{S_y}}{3160}\right) \le M_p$$

[redacted]

To determine the margin of safety:

$$P_1 = \frac{P}{P_a}; \quad M_1 = \frac{C_m M}{\left(1 - P/P_e\right) M_a} \quad (P_1 + M_1 \le 1)$$

and

$$P_2 = \frac{P}{P_y}; \quad M_2 = \frac{M}{1.18 M_p} \quad (P_2 + M_2 \le 1)$$

The margins of safety are

$$MS1 = \frac{1}{P_1 + M_1} - 1$$

and

$$MS2 = \frac{1}{P_2 + M_2} - 1$$

Buckling evaluation results are provided in Tables [illegible] through [illegible]. As the tables illustrate, [illegible] meet the requirements of [illegible].

[redacted]

[redacted]

[redacted] evaluation.

As shown in Figure 11.2.12.3-9, a finite element model for the support disk is generated using the ANSYS program to perform analyses for six side impact cases:

Case No.	Side Impact Acceleration (g)	Basket Orientation (Degree)
1	20	0
2	20	22.5
3	20	45
4	55	0
5	55	22.5
6	55	45

The model consists of a support disk and a section of the canister shell, as shown in Figure 11.2.12.3-9. ANSYS SHELL63 elements are used to model the disk and canister. As shown in Figure 11.2.12.3-10, CONTAC52 elements are used to simulate the interface between the support disk and the canister shell. A value of 1.0 × 10[illegible] lb/inch is used for the stiffness of CONTAC52 elements. BEAM3 elements with very small properties (Area = 5 × 10[illegible] inch², I_{zz} = 5 × 10[illegible] inch⁴, Modulus of Elasticity = 1 psi) are applied at the same locations as the CONTAC52 elements to prevent rigid body motion of the model. The canister shell outer diameter is constrained to prevent translation of the canister. Applying a concentrated load (with the acceleration factor) simulates the fuel assembly and tube weight at the midpoint of each ligament. The value of the force varies according to the impact orientation. There is no fuel

[redacted]

[redacted]

[redacted]	[redacted]	
	[redacted]	[redacted]
0°	[redacted]	[redacted]
22.5°	[redacted]	[redacted]
45°	[redacted]	[redacted]

The maximum stress intensity occurs in the 45° [redacted] the 45° orientation is the bounding case for the [redacted] over event.

11.2.12.4 Radiological Consequences

There is an adverse radiological consequence in the hypothetical tipover event since the bottom end of the storage cask and the canister have significantly less shielding than the sides and tops of these same components. The dose rate at 1 meter is calculated to be approximately 156 rem/hour, and the dose at 5 meters is estimated to be approximately 1 rem/hour. Consequently, following a tip-over event, supplemental shielding should be used until the concrete cask can be righted. Stringent access controls must be applied to ensure that personnel do not enter the area of radiation shine from the exposed bottom of the tipped over concrete cask.

11.2.12.5 NAC-MPC Performance

Functionally the NAC-MPC does not have significant adverse consequences from the tip-over event. The storage cask and canister maintain design basis shielding, geometry control of contents, and contents confinement performance requirements.

Damage to the edge and surface of the concrete cask may occur in the tipover, which could result in marginally higher dose rates at the bottom edge or at surface cracks in the concrete. This

increased dose rate is not expected to be significant, but would be dependent on the specific damage incurred.

11.2.12.6 Recovery and/or Corrective Actions

The most important recovery step required following a cask tipover accident is the uprighting of the concrete cask to eliminate the high dose rate from the exposed bottom end. The uprighting operation will require a heavy lift capability and rigging expertise. The concrete cask must be returned to the vertical by rotation around a convenient bottom edge. The concrete cask should be returned to the vertical using a method and rigging that controls the rotation to vertical.

Surface and top and bottom edges of the concrete cask would be expected to exhibit cracking and possibly loss of concrete down to the layer of reinforcing bar. If only minor damage occurs, the concrete may be repairable using grout. Otherwise, it may be necessary to remove the canister and install it in a new storage cask.

The storage pad must be repaired to preclude the intrusion of water that could cause further deterioration of the pad in freeze-thaw cycles.

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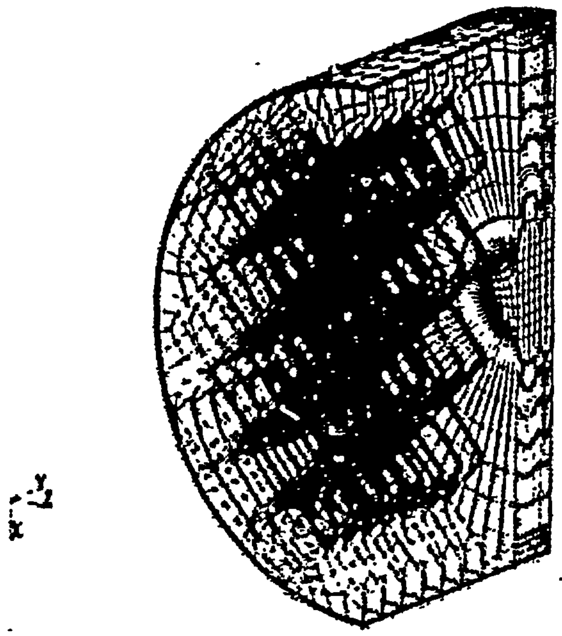
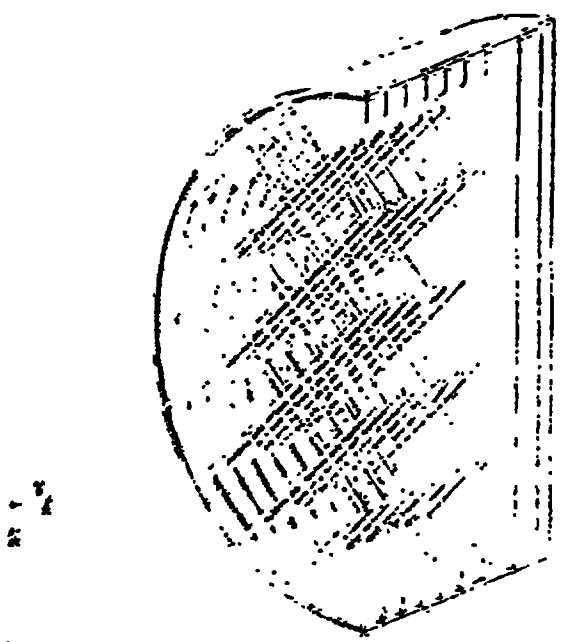


Model With Element Edges Displayed



Model Without Element Edges Displayed

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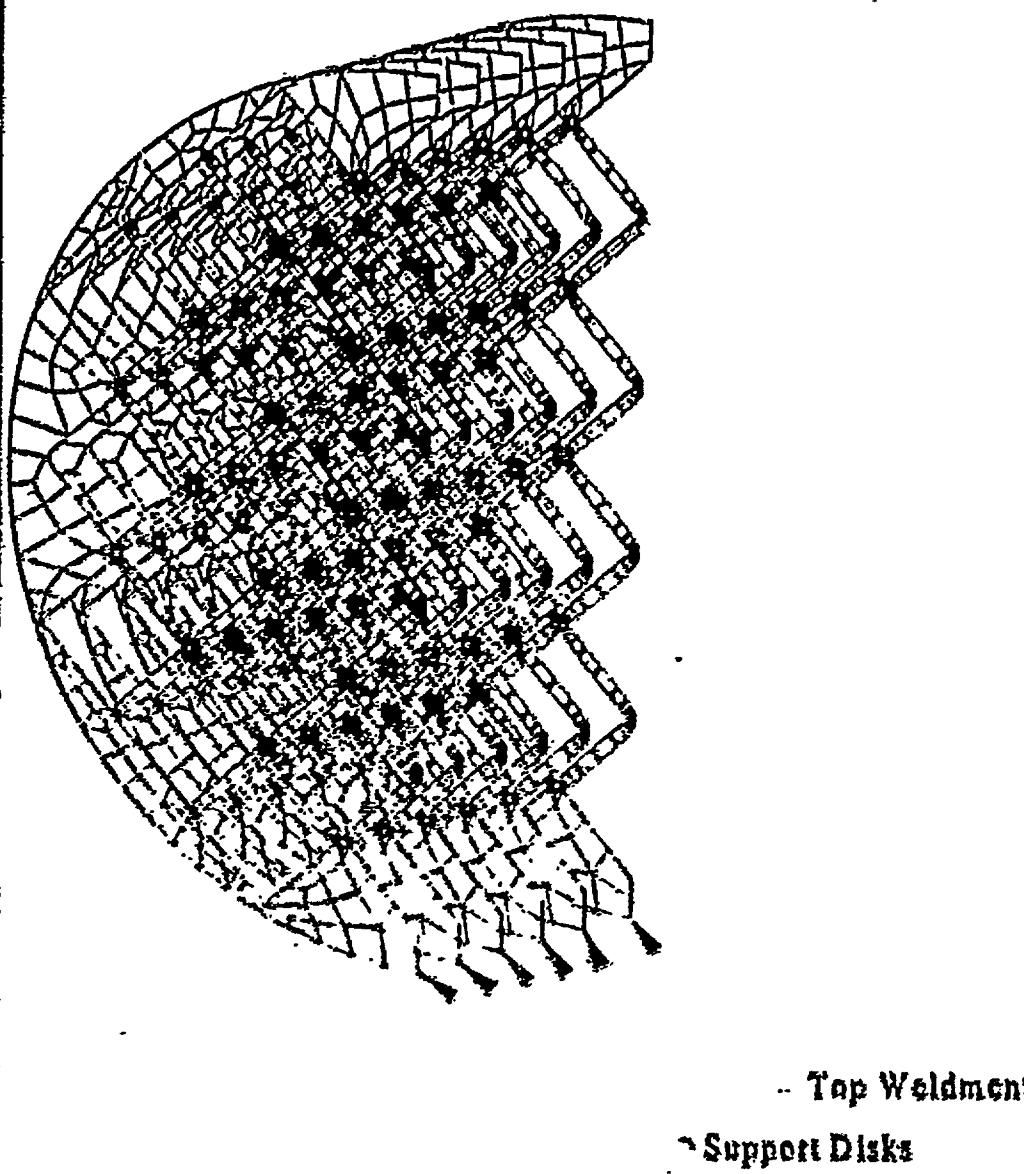

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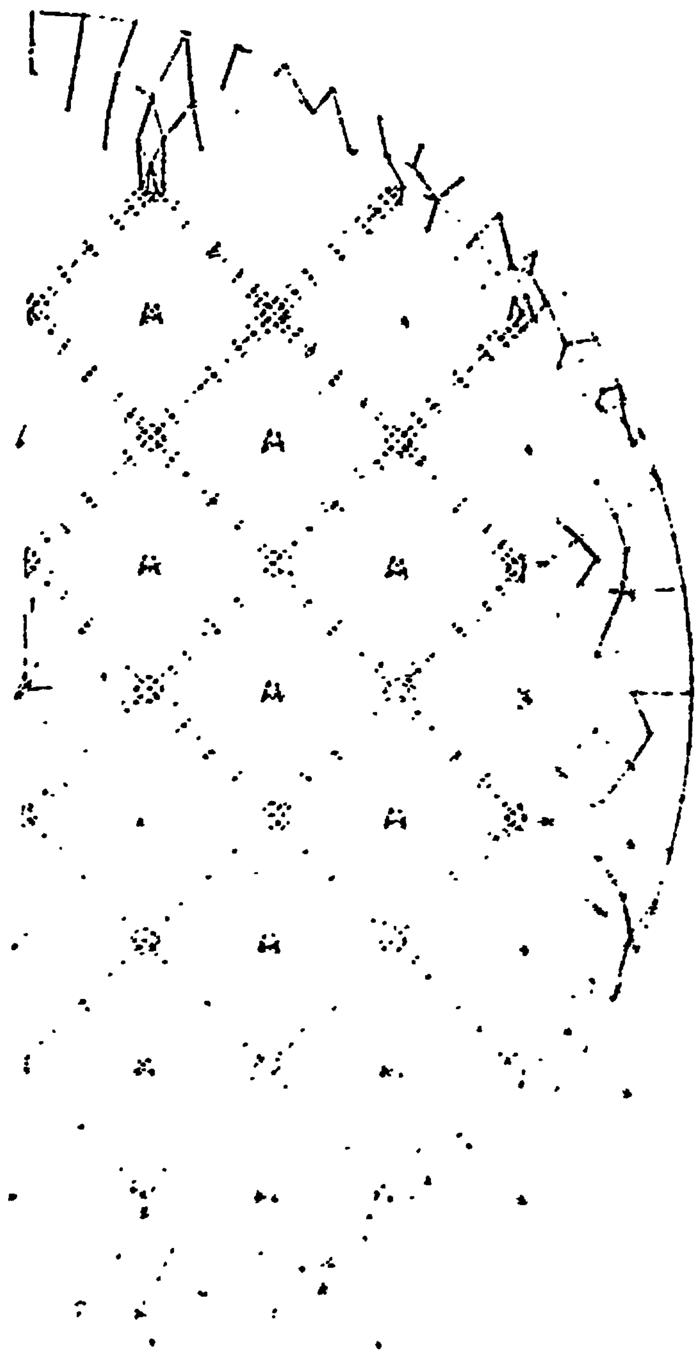

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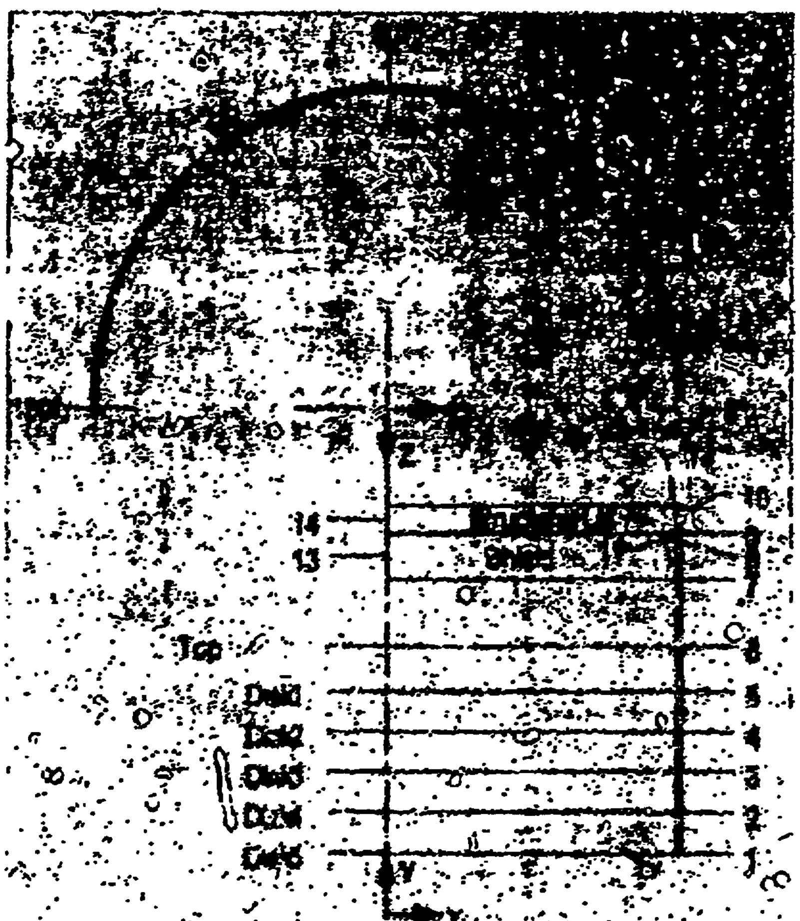

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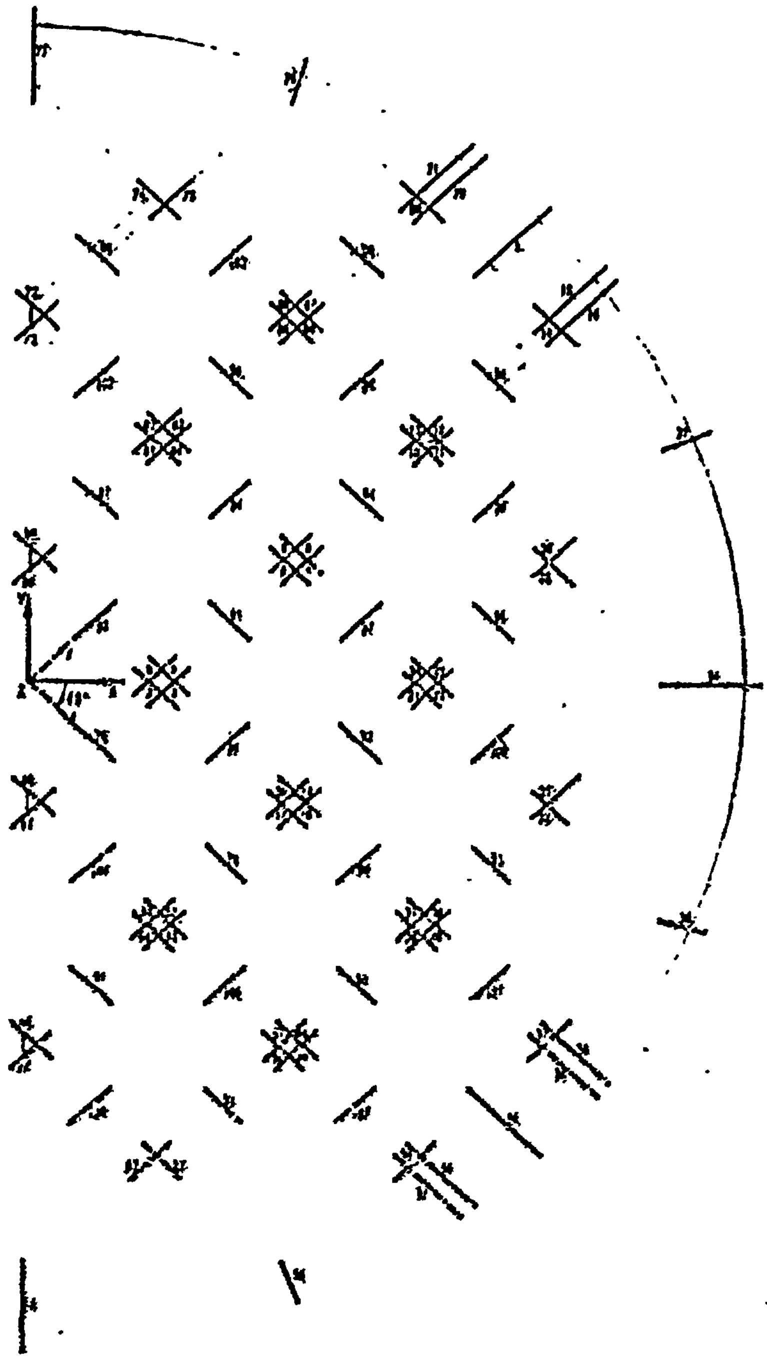

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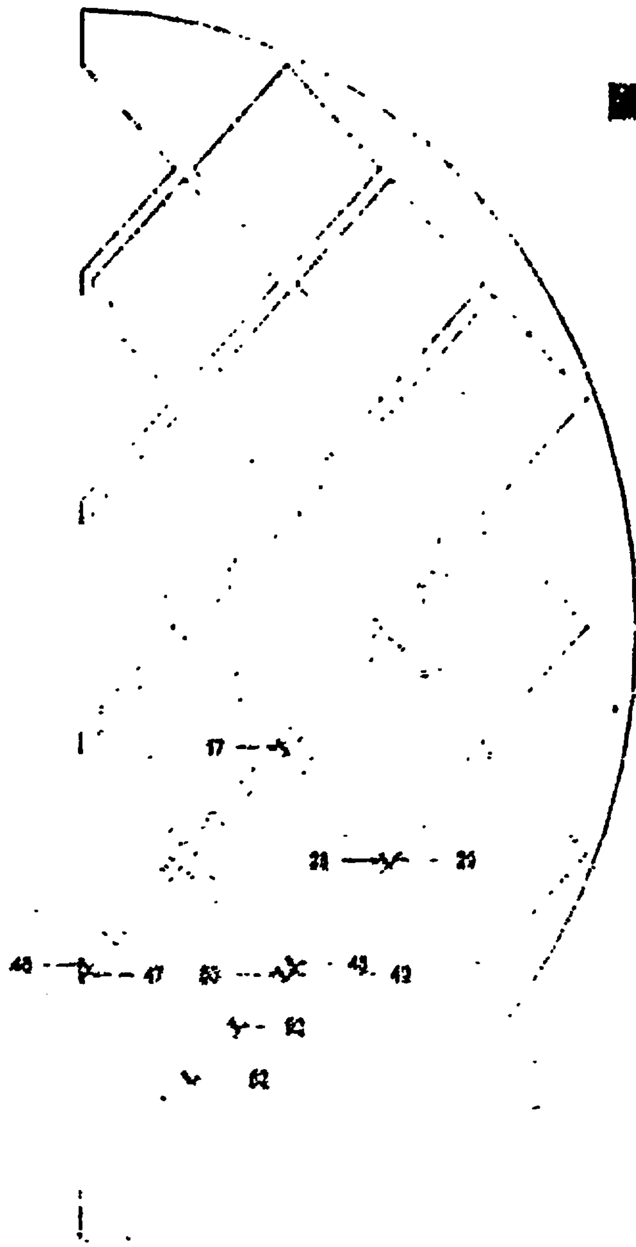

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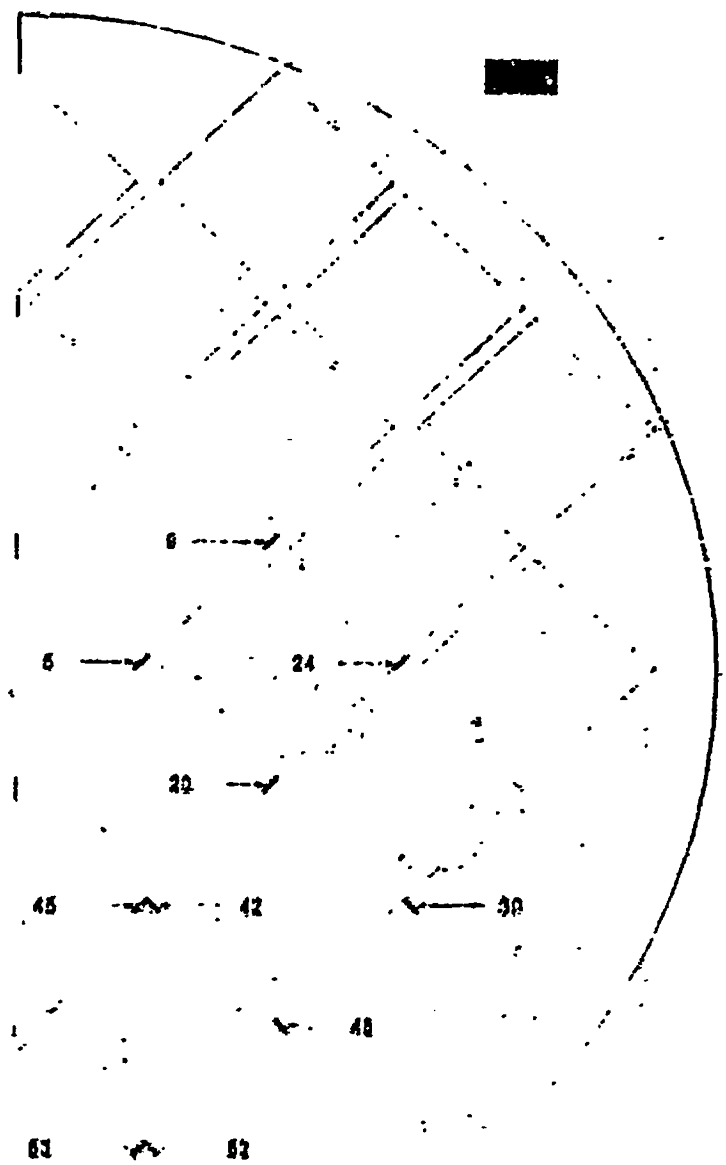

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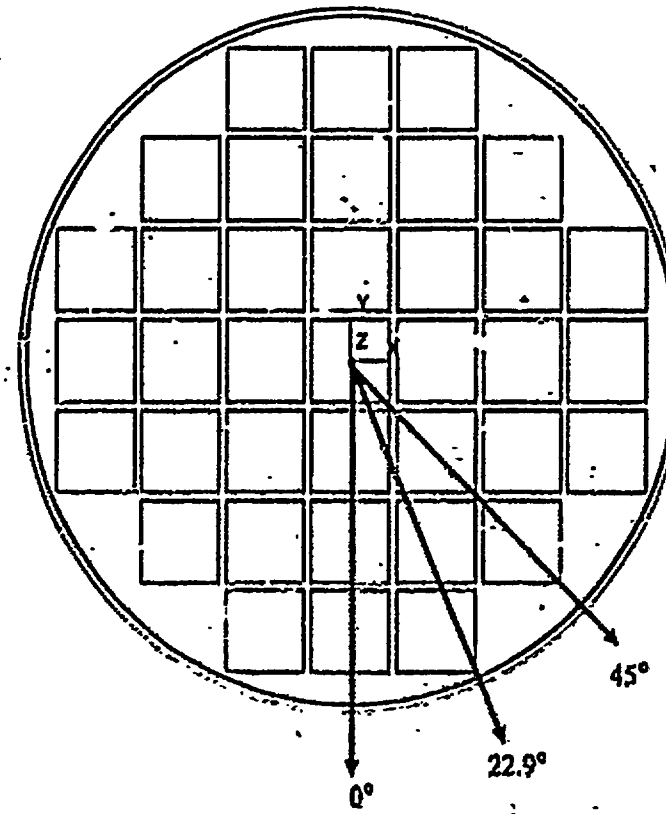

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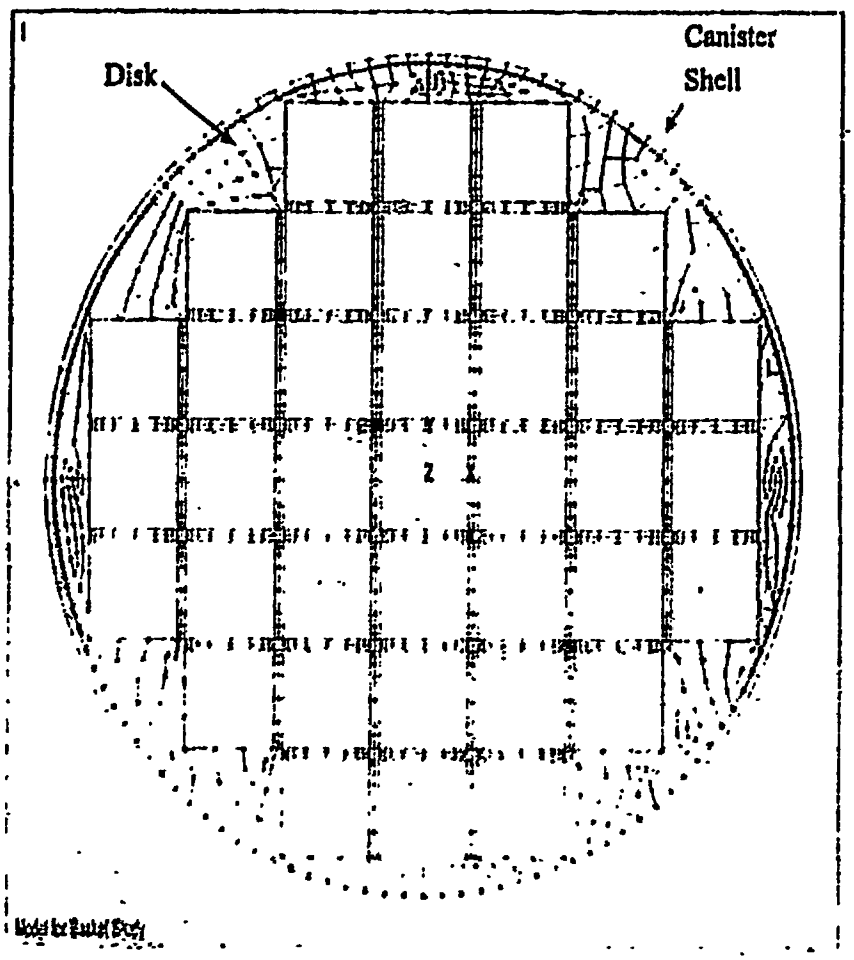

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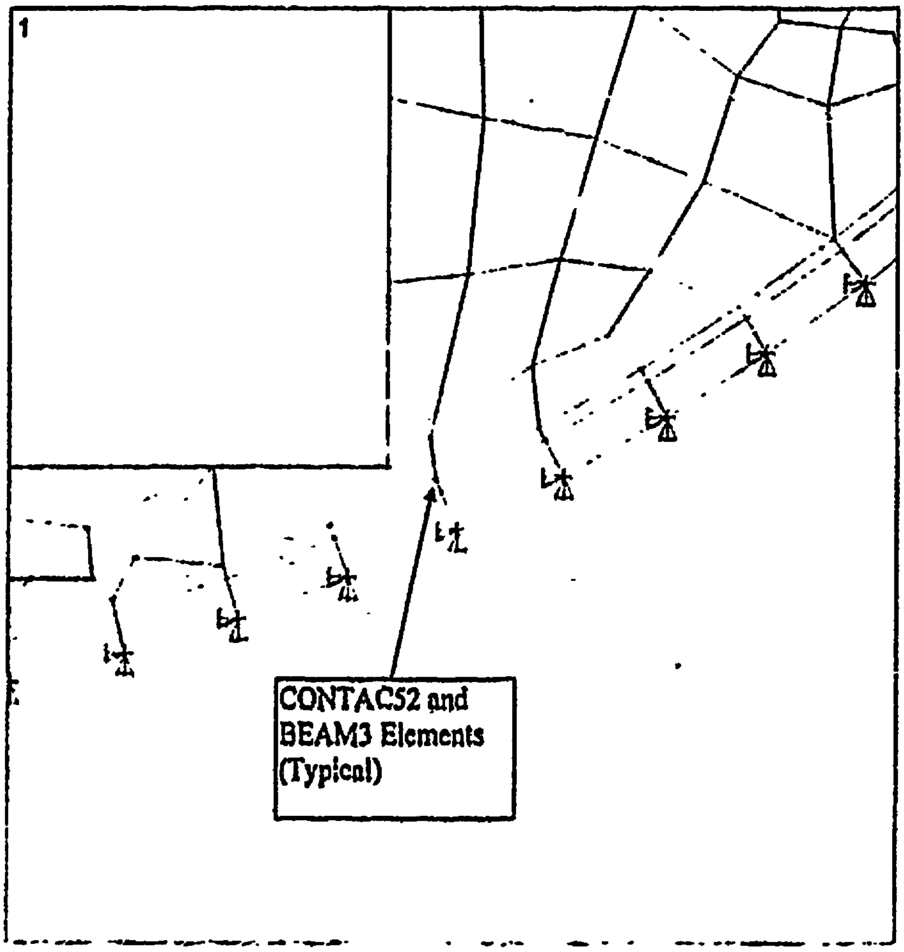

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11	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
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14	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

Notes:

1 Stresses are not presented for the region with localized bearing [illegible] need not be evaluated for Level D Service (Accident) Conditions.

Table 11.2.12-4[illegible] [illegible] Primary Membrane [illegible]

Section	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
1	0.00	14.30	10.50	22.20	0.90	0.30	11.20	[illegible]	[illegible]	[illegible]
2	0.00	11.90	1.70	29.30	15.60	0.50	[illegible]	[illegible]	[illegible]	[illegible]
3	0.00	22.70	1.60	25.90	0.40	0.20	[illegible]	[illegible]	[illegible]	[illegible]
4	0.04	23.10	22.20	24.30	10.10	0.04	0.60	[illegible]	[illegible]	[illegible]
5	0.00	22.90	2.00	27.60	0.00	10.10	10.40	[illegible]	[illegible]	[illegible]
6	20.00	1.80	5.00	0.00	114.30	0.00	10.10	[illegible]	[illegible]	[illegible]
7	0.00	11.40	12.20	54.70	11.10	1.50	13.20	[illegible]	[illegible]	[illegible]
8	0.00	41.00	2.00	3.10	0.00	0.00	1.70	[illegible]	[illegible]	[illegible]
9	0.00	116.30	33.90	11.60	10.20	[illegible]	1.30	[illegible]	[illegible]	[illegible]
10	0.00	130.30	21.20	112.90	1.00	1.30	2.10	[illegible]	[illegible]	[illegible]
11	0.00	21.30	13.10	113.00	113.00	0.30	10.50	[illegible]	[illegible]	[illegible]
12	0.00	140.30	30.00	-15.90	16.10	0.60	17.40	[illegible]	[illegible]	[illegible]
13	0.00	31.00	0.00	0.70	10.10	0.00	0.00	[illegible]	[illegible]	[illegible]
14	0.00	10.30	0.00	1.50	0.00	0.00	6.60	[illegible]	[illegible]	[illegible]

Notes:

1 Stresses are not presented for the region with bearing stress (membrane) and secondary (Q) stress [illegible] Section III, Appendix F, both bearing stresses and secondary stresses are not required [illegible] Level D Service (Accident) Conditions.

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[illegible]	[illegible]	[illegible]	[redacted]	[redacted]	[redacted]	[redacted]
[illegible]	[illegible]	34	12.29	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	36	12.86	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	38	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	40	12.91	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	42	11.75	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	44	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	46	19.89	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	48	19.32	[illegible]	[illegible]	[illegible]
[illegible]	49	50	25.16	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	52	25.16	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	54	31.52	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	56	18.7	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	58	19.23	[illegible]	[illegible]	[illegible]
30	59	60	19.85	[illegible]	[illegible]	[illegible]
31	61	62	19.32	[illegible]	[illegible]	[illegible]
[illegible]	63	64	25.16	[illegible]	[illegible]	[illegible]
[illegible]	65	66	25.16	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	68	31.57	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	70	22.15	[illegible]	[illegible]	[illegible]
[illegible]	71	72	25.07	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	74	25.07	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	76	25.69	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	78	31.52	[illegible]	[illegible]	[illegible]
40	79	80	0	0	[illegible]	0
41	81	82	0.62	[illegible]	0	[illegible]
42	83	84	5.46	[illegible]	[illegible]	[illegible]
43	85	86	7.03	[illegible]	[illegible]	[illegible]
44	87	88	6.71	[illegible]	[illegible]	[illegible]

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[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
11	2.51	43.95	37.41	2.27	[illegible]	[illegible]	[illegible]	[illegible]
4	1.55	44.00	37.34	1.43	[illegible]	[illegible]	[illegible]	[illegible]
30	10.19	35.87	31.97	1.34	[illegible]	[illegible]	[illegible]	[illegible]
31	4.05	43.95	37.31	3.45	[illegible]	[illegible]	[illegible]	[illegible]
13	0.97	35.92	32.00	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2	3.02	44.00	37.34	3.72	[illegible]	[illegible]	[illegible]	[illegible]
60	11.39	[illegible]	31.97	6.48	[illegible]	[illegible]	[illegible]	[illegible]
62	[illegible]	[illegible]	34.56	6.41	[illegible]	[illegible]	[illegible]	[illegible]
83	[illegible]	[illegible]	34.56	2.46	[illegible]	[illegible]	[illegible]	[illegible]
44	6.94	39.79	34.56	1.80	[illegible]	[illegible]	[illegible]	[illegible]
46	8.37	39.79	34.56	1.20	7.00	[illegible]	[illegible]	[illegible]
2	3.62	44.00	37.34	2.79	8.17	[illegible]	[illegible]	[illegible]
41	0.83	44.00	37.34	2.63	[illegible]	[illegible]	[illegible]	[illegible]
91	5.94	39.75	34.53	1.65	6.99	[illegible]	[illegible]	[illegible]
28	10.19	35.92	32.00	0.36	6.00	[illegible]	[illegible]	[illegible]
106	7.77	39.79	34.56	1.18	7.00	[illegible]	[illegible]	[illegible]
83	8.54	35.92	32.00	0.63	6.00	5.75	[illegible]	[illegible]
60	7.94	44.00	37.34	2.65	8.17	[illegible]	[illegible]	[illegible]
8	0.01	44.00	37.34	3.10	8.17	[illegible]	[illegible]	[illegible]
17	8.94	39.75	34.53	1.79	6.99	[illegible]	[illegible]	[illegible]
79	8.06	[illegible]	34.56	1.10	7.00	[illegible]	[illegible]	[illegible]
63	0.22	44.00	37.34	2.79	8.17	[illegible]	[illegible]	[illegible]
47	7.75	39.75	34.53	0.51	6.99	[illegible]	[illegible]	[illegible]
105	6.38	43.95	37.31	1.27	8.16	7.95	[illegible]	2.67
92	8.53	35.92	32.00	0.04	6.00	5.75	5.09	2.67

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

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[redacted]

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[redacted]

11.2.13 Tornado and Tornado Driven Missiles

This analysis evaluates the strength and stability of the concrete storage cask for a maximum tornado wind loading and for the impacts of tornado generated missiles. The design basis tornado characteristics have been selected in accordance with Regulatory Guide 1.76.

Classical techniques are used to evaluate the loading conditions. Cask stability analysis for the maximum tornado wind loading is based on NUREG-0800, Section 3.3.1, "Wind Loadings," and Section 3.3.2, "Tornado Loadings." Loads, due to tornado generated missiles, are based on NUREG-0800, Section 3.5.3, "Barrier Design Procedures."

11.2.13.1 Cause of Tornado Event

A tornado is a random weather event having a higher probability of occurrence at certain times of the year and in certain geographical areas.

Wind loading and tornado driven missiles have the potential for causing damage from pressure differential loading and from impact loadings.

A tornado event is expected to be visually observed. Warning of a tornado probability and of atornado sighting may be received f n the National Weather Service, local radio and television, local law enforcement officials, and site personnel.

11.2.13.2 Analysis of the Tornado Event

Classical analysis is applied to the evaluation of the consequences of tornado wind and missile events.

The concrete cask stability in a maximum tornado wind is evaluated based on the design wind pressure calculated in accordance with ANSI/ASCE 7-93 and using classical free body stability analysis methods.

Local damage to the concrete shell is assessed using a formula developed by the National Defense Research Committee. This formula has been selected as the basis for predicting depth

of missile penetration and minimum concrete thickness requirements to prevent scabbing of the concrete. Penetration depths calculated using this formula have been shown to provide reasonable correlation with test results (EPRI Report NP-440).

The local shear strength of the concrete shell is evaluated based on ACI 349-85, Section 11.11.2.1, discounting the reinforcing and steel shell.

The concrete shell shear capacity and shear-friction reinforcing steel area requirements are also evaluated for missile loading using ACI 349-85, Section 11.7.

The cask has the following properties that are considered in this evaluation:

H =	Height = 160 in
D_o =	Outside Diameter = 128 in
D_i =	Inside Diameter of concrete shell = 86 in
W_{VCC} =	Weight of the storage cask with canister, basket and full fuel load
=	206,100 lbs
A_c =	Cross section area of concrete shell = 7,059 in^2
I_c =	Moment of inertia of concrete shell = 10.49×10^6 in^4
f_c' =	Compressive strength of concrete shell = 4,000 psi

11.2.13.2.1 Tornado Wind Loading (Storage Cask)

The tornado wind velocity is transformed into an effective pressure applied to the cask using procedures delineated in ANSI/ASCE 7-93 Building Code Requirements for Minimum Design Loads in Buildings and Other Structures. The maximum pressure, q, is determined from the maximum tornado wind velocity as follows:

$$q = (0.00256)\,V^2 \text{ psf}$$

Where:

V = Maximum tornado wind speed = 360 mph

The velocity pressure exposure coefficient for local terrain effects, K, Importance Factor, I, and the Gust Factor, G, may be taken as unity (1) for evaluating the effects of tornado wind velocity pressure.

Then:

$$q = (0.00256)(360)^2 = 331.8 \text{ psf}$$

Considering the cask is small with respect to the tornado radius, the velocity pressure is assumed uniform over the projected area of the cask.

Then the total wind loading on the projected area of the cask, F_w, is then computed as:

$$F_w = q \times G \times C_f \times A_p$$

where:

q = Effective velocity pressure (psf)

C_f = Force Coefficient = 0.50 (ASCE 7-93, Table 12 with $D\,q^{1/2}$ = 194 for a moderately smooth surface, h/D = 13.33 ft /10.66 ft = 1.25)

A_f = Projected area of cask = 10.67 ft x 13.33 ft = 142.2 ft^2

F_w = 331.8 x 0.50 x 142.2 = 23,590 lbs

The wind overturning moment, M_w, is computed as:

$M_w = F_w \times H/2$, where, H, is the cask height

M_w = 23,590 lbs x 160. in/12 x 1/2 = 157,267. ft-lbs

The stability moment, M_s, of the cask with the canister, basket and full fuel load about an edge of the base, is:

$M_s = W_{VCC} \times D_o/2$

where:

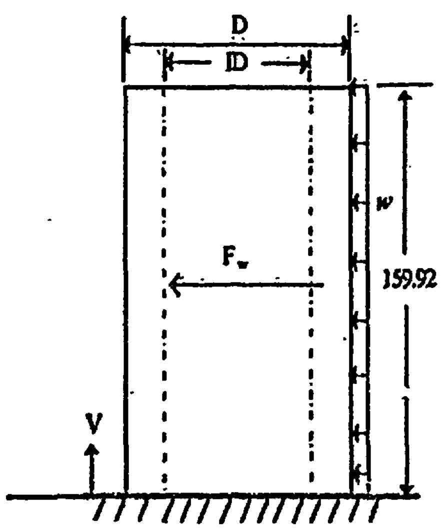
D_o = Diameter of the cask = 128 in

W_{VCC} = Weight of the cask with canister
= 206,100 lbs

M_s = 206,100 lbs x 128. In /12 x 1/2 = 1.099 x 10^6 ft-lbs

Thus, the cask has a Safety Factor of approximately +7 (1.099 x 10^6 /1.573 x 10^5) against overturning with respect to the maximum tornado wind loading and requires only a coefficient of friction of about 0.12 (23.6 x 10^3 / 206. x 10^3) to be developed between the concrete cask base and ISFSI support deck to inhibit sliding via friction.

The stresses in the concrete due to the tornado wind load are conservatively calculated as below. The concrete cask is considered to be fixed at its base.



F_W = 21,720 lb
D = 128 in. (concrete exterior diameter)
ID = 86 in. (concrete interior diameter)

A	$= \pi (D^2 - ID^2) / 4$	$= 7,059.2$ in.^{2}	
I	$= \pi (D^4 - ID^4) / 64$	$= 10.492 \times 10^6$ in.^{4}	(Moment of Inertia)
S_{outer}	$= 2I / D$	$= 163,937.5$ in.^{3}	(Section Modulus for inner surface)
S_{inner}	$= 2I / (ID)$	$= 244,000.0$ in.^{3}	(Section Modulus for outer surface)
w	$= F_w / 159.92$	$= 147.5$ lbf/in	
M	$= w (159.92)^2 / 2$	$= 1.886 \times 10^6$ in.-lb	(Bending Moment at the base)

Maximum stresses at the base surface:

$\sigma_{v\,outer} = M / S_{outer} = 11.5$ psi (tension or compression)
$\sigma_{v\,inner} = M / S_{inner} = 7.7$ psi (tension or compression)

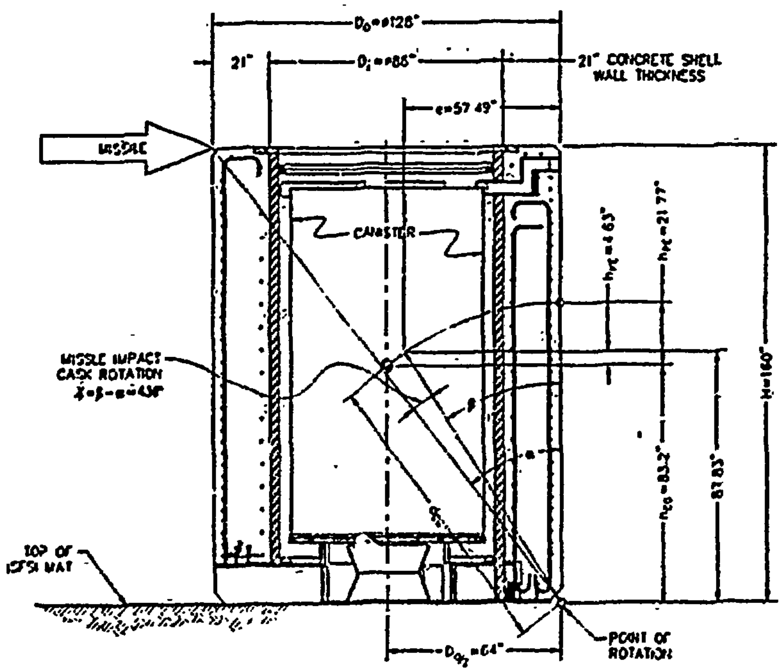
The compressive stresses are included in the load combination No. 8 in Table 3.4.4.2-1, since they are governing stresses for the load combination. As shown in Tables 3.4.4.2-1 and 3.4.4.2-2, the maximum combined stresses for the load combination of dead, live, thermal and tornado wind are below the allowable stress.

11.2.13.2.2 Tornado Missile Loading (Storage Cask)

The NAC-MPC concrete cask is designed to withstand the effects of impacts associated with postulated tornado generated missiles identified in NUREG-0800, Section 3.5.1.4.III.4, Spectrum I missiles. Consisting of: 1) a high kinetic energy missile (3,960 lb automobile, with a frontal area of 20 square feet that deforms on impact); 2) a 275 lb, 8 in diameter armor piercing artillery shell; and 3) a small 1-inch diameter solid steel sphere. All of these missiles are assumed to

impact in a manner that produces the maximum damage at a velocity of 126 mph (35 percent of the maximum tornado wind speed of 360 mph). The cask has been evaluated for impact effects associated with each of the above missiles.

The principal dimensions and moment arms used in this evaluation are shown below:



The concrete cask has no openings except for the four outlets at the top and four inlets at the bottom of the cask. These openings are configured such that a 1-inch diameter solid steel missile cannot directly enter the concrete cask interior. In addition, the canister is protected from the inlet (bottom) openings by a steel pedestal (bottom plate), and from the outlet (top) openings by the canister structural and shield lids. Therefore, a detailed analysis of the impact of a 1-inch diameter steel missile is not required.

Concrete Shell Local Damage Prediction (Penetration Missile)

Local damage to the cask body has been assessed using the National Defense Research Committee (NDRC) formula. This formula has been selected as the basis for predicting depth of penetration and minimum concrete thickness requirements to prevent scabbing. Penetration depths calculated using this formula have been shown to provide reasonable correlation with test results (EPRI Report NP-440, Section 4 "Local Response Evaluation").

Concrete shell penetration depths are calculated as follows:

For $x/2d \leq 2.0$:

where:

x = Missile penetration depth
d = Missile diameter = 8 in
$x = [4KNWd^{-0.8}(V/1000)^{1.8}]^{0.5}$

where:

K = Coefficient depending on concrete strength
$= 180/(f_c')^{1/2} = 180/(4000)^{1/2} = 2.846$
N = 1.14 Shape factor for sharp nosed missiles
W = Missile weight = 275 lb
V = Missile velocity = 126 mph = 185 ft/sec

$x = [(4)(2.846)(1.14)(275)(8^{-0.8})(185/1000)^{1.8}]^{0.5}$
$= 5.68$ inches
$x/2d = 5.68/(2)(8) = 0.355 < 2.0$

The minimum concrete shell thickness required to prevent scabbing is three times the predicted penetration depth of 5.68 inches based on the NDRC formula, or 17.04 inches. The concrete cask wall thickness includes 21 inches of concrete, which is more than the thickness required to

prevent damage due to the penetration missile. This analysis conservatively neglects the 3.5 inch steel shell at the inside face of the concrete shell.

Closure Plate Local Damage Prediction (Penetration Missile)

The concrete cask is closed with a 1.5-inch thick steel plate bolted in place. The following missile penetration analysis shows that the 1.5-inch steel closure plate is adequate to withstand the impact of the 275 lb armor piercing missile, impacting at 126 mph.

The perforation thickness of the closure steel plate is calculated by the Ballistic Research Laboratories Formula with K = 1, formula number 2-7, in Section 2.2 of Topical Report BC-TOP-9A, Revision 2.

$$T = [0.5M_m V^2]^{2/3}/672d$$
$$= 0.516 \text{ inch}$$

where:

T = Perforation thickness
M_m = Missile mass = W/g = 275 lbs/32.2 ft/sec^2 = 8.54 slugs
g = Acceleration of gravity = 32.2 ft/sec^2
W, V and d are as defined above

BC-TOP-9A, recommends that the plate thickness be 25 percent greater than the calculated perforation thickness, T, to prevent perforation.

Thus, the recommended plate thickness is: 1.25 x 0.516 in. = 0.645 in.

The closure plate is 1.5 inches thick; therefore, the plate is adequate to withstand the local impingement damage due to the specified armor piercing missile.

Overall Damage Prediction for a Tornado Missile Impact (High Energy Missile)

The concrete cask is a freestanding structure. Therefore, the principal consideration in overall damage response is the potential of upsetting or overturning the cask as a result of the impact of a

high energy missile. Based on the following analysis, it is concluded that the cask can sustain an impact from the defined high energy missile and does not overturn.

From the principle of conservation of momentum, the impulse of the force from the missile impact on the cask must equal the change in angular momentum of the cask. Also, the impulse force due to the impact of the missile must equal the change in linear momentum of the missile. These relationships may be expressed as follows:

Change in momentum of the missile, during the deformation phase:

$$\int_{t_1}^{t_2} (F)(dt) = M_M (v_2 - v_1)$$

where:

F = Impact Impulse force on missile
M_M = Mass of missile = 3960 lbs/g = 123 slugs/12 =
= 10.25 lbm (lb sec^2 /in)
t_1 = Time at missile impact
t_2 = Time at conclusion of deformation phase
v_1 = Velocity of missile at impact = 126 mph = 185 ft/sec
v_2 = Velocity of missile at time t_2

The change in angular momentum of the cask, about the bottom outside edge/rim, opposite the side of impact is:

$$\int_{t_1}^{t_2} (M_c)(dt) = \int_{t_1}^{t_2} (H)(F)\,dt = I_m(\varpi_1 - \varpi_2)$$

Thus, $\int (F)(dt) = M_M (v_2 - v_1) = I_m (\varpi_1 - \varpi_2)/H$

Where:

M_c = Moment of the impact force on the cask

I_m = Storage cask mass moment of inertia, about point of rotation on the bottom rim

ϖ_1 = Angular velocity at time t_1

ϖ_2 = Angular velocity at time t_2

M_{VCC} = Mass of the VCC cask = W_{VCC}/g

= 6,400 slugs/12 = 533.4 lbm (lb sec^2 /in)

I_{mx} = Mass moment of inertia of VCC cask about x axis through center of gravity

$$\cong 1/12(M_{VCC})(3r^2 + H^2)$$

$$\cong (1/12)(533.4)\,[(3)(64)^2 + (160)^2] = 1.684 \times 10^6 \text{ in}^2\text{lbm}$$

$I_m = I_{mx} + (M_{VCC})(d_{CG})^2$, where: d_{CG} is the distance (105 inches) between the cask CG and a rotation point on base rim

Thus, $I_m \cong 1.684 \times 10^6 + (533.4)(105)^2$

$= 7.565 \times 10^6$ in^{2}lbm

Based on conservation of momentum, the impulse of the impact force on the missile is equated to the impulse of the force on the cask.

$$M_M(v_2 - v_1) = I_m(\varpi_1 - \varpi_2)/H,$$

at time t_1, v_1 = 185 ft/sec and ϖ_1 = 0 rad/sec

at time t_2, = v_2 = 0 ft/sec based on the following:

During the restitution phase, the final velocity of the missile will depend upon the coefficient of restitution of the missile, the geometry of the missile and target, the angle of incidence, and on the amount of energy dissipated in deforming the missile and target. It is assumed, based on tests conducted by EPRI (Ref. EPRI Report NP-440), that the final velocity of the missile, v_f, following the impact is zero. If it is conservatively assumed that all of the missile energy is transferred to the cask:

Then: $(10.25)(v_2 - 185 \text{ ft/sec} \times 12 \text{ in/ft}) = 7.565 \times 10^6 \text{ in}^2\text{lbm}\,(0 - \varpi_2)/160$

Setting v_2 = 0 and solving for ϖ_2,

ϖ_2 = 0.481 rad/sec

and $v_2 = 205\ \varpi_2$

Then, $v_2 = (205)(0.481) = 98.6$ in/sec

Equating the impulse of the force on the missile during restitution to the impulse of the force on the cask yields:

$$-[M(v_f - v_2] = I_m(\varpi_f - \varpi_2)/H$$

With: $v_f = 0$, $v_2 = 98.6$ in/sec and $\varpi_2 = 0.481$ rad/sec

Then: $-[10.25(0-98.6)] = 7.565 \times 10^6\ \text{in}^2\text{lbm}\ (\varpi_f - 0.481)/160$

$\varpi_f = 0.502$ rad/sec

Thus, the final energy of the cask following the impact, E_k, is:

$$E_k = (I_m)(\varpi_f)^2/(2)$$
$$= (7.565 \times 10^6)(0.502)^2/(2)$$
$$E_k = 9.53 \times 10^5\ \text{in-lb}_f$$

The energy required to overturn cask must be equal to or greater than its potential energy, E_p:

$$E_p = (W_{VCC})(h_{PE})$$
$$E_p = 206{,}100\ \text{lbs} \times 21.77\ \text{in}$$
$$E_p = 4.487 \times 10^6\ \text{in-lb}_f$$

The high energy tornado generated missile imparts insufficient kinetic energy to produce a cask overturning due to the missile impact.

Combined Tornado Wind and Missile Loading (High Energy Missile)

The cask rotation due to the heavy missile impact is calculated as:

$$h_{EK} = E_k / W_{VCC} = 9.53 \times 10^5\ \text{in-lb}_f / 206{,}100.\ \text{lbs} = 4.63\ \text{in}$$

Then: $\cos\beta = (h_{CG} + h_{KE}) / d_{CG}$

$\cos\beta = (83.2 + 4.63) / 104.97 = 0.8367$

$\beta = 33.21$ deg

$\cos\alpha = 83.2 / 104.97 = 0.7926$

$\alpha = 37.57$ deg

$e = d_{CG} \sin\beta$

$e = 104.97 \sin 33.21 = 57.49$ in

Thus, cask rotation after impact = $\alpha - \beta = 37.57 - 33.21 = 4.36$ deg

Available gravity restoration moment after missile impact = $(W_{VCC})(e)$

= 206,100 lb x 57.49 in / 12 =

987,391 ft-lb >> Tornado Wind Moment = 157,267. ft-lb

Therefore, the combined effects of tornado wind loading and the high energy missile impact loading will not overturn the cask.

Local Shear Strength Capacity of Concrete Shell (High Energy Missile)

This section evaluates the shear strength of the concrete at the top edge of the concrete shell due to a high energy missile impact based on ACI 349-85, Chapter 11, Section 11.11.2.1, on concrete punching shear strength.

The force developed by the missile using the methodology presented in Topical Report, BC-TOP-9A, is:

$F = 0.625(v)(W_M)$

$F = 0.625(185 \text{ ft/sec})(3960 \text{ lbs}) = 457.8$ kips

$F_u = LF \times F = 1.1 \times 457.8 = 503.6$ kips

Based on a rectangular missile contact area, having proportions of 2 horizontal to 1 vertical and the top of the area flush with the top of the concrete cask, the required missile contact area based on the concrete punching shear strength, neglecting reinforcing is:

$$V_c = (2+4/\beta_c)(f_c')^{1/2} b_o\, d, \text{ where } \beta_c = 2/1 = 2$$
$$V_c = 4(f_c')^{1/2} b_o\, d$$
$$d = 21\text{in} - 3\text{ in} = 18\text{ in}$$
$$(f_c')^{1/2} = 63.24\text{ psi, where } f_c' = 4{,}000\text{ psi}$$
b_o = perimeter of punching shear area at d/2 from missile contact area
$$b_o = (2b + 18) + 2(b + 9) = 4b + 36$$
$$V_u = \Phi(V_c + V_s)\text{, where } V_s = 0\text{, assuming no steel shear}$$
$$V_u = \Phi V_c = \Phi 4 (f_c')^{1/2} b_o\, d = (.85)(4)(63.24)(4b + 36)(18)$$
$$= 15{,}481\, b + 139{,}330$$

Setting, V_u equal to F_u and solving for b:

$$503.6 \times 10^3 = 15{,}481\, b + 139{,}330$$
$$b = 23.53\text{ inches (say 2.0 ft)}$$

The implied missile impact area required = 2b x b = 2 x 2 x 2 = 8.0 sq ft < 20.0 sq ft.

Thus, the concrete shell alone, based on the concrete conical punching strength and discounting the steel reinforcement and shell, has sufficient capacity to react to the high energy missile impact force.

The effects of tornado winds and missiles have been considered both separately and combined in accordance with NUREG-800, Section 3.3.2 II.3.d. For the case of tornado wind plus missile loading, the stability of the cask has been assessed and found to be acceptable. Equating the kinetic energy of the cask following missile impact to the potential energy yields a maximum postulated rotation of the cask, as a result of the impact, of 4.36 degrees. Applying the total tornado wind load to the cask in this configuration results in an available restoring moment considerably greater than the tornado wind overturning moment. Therefore, overturning of the cask under the combined effects of tornado winds, plus tornado-generated missiles, does not occur.

11.3 Design Basis Loading of the Transportable Storage Canister

The Transportable Storage Canister (canister) is designed to be stored in the NAC-MPC storage cask and transported in the NAC-STC Storable Transport Cask. The NAC-STC is licensed to transport spent fuel in accordance with 10CFR71 and has been issued Certificate of Compliance Number 71-9235. The NAC-STC has a design weight when loaded of 250,000 pounds and is intended to be transported by rail.

An amendment to the Safety Analysis Report (SAR) for the NAC-STC is being requested, in conjunction with this Safety Analysis Report for storage of Yankee class fuel, to include the loaded canister as authorized contents.

The load condition imposed on the canister and its basket by the transport conditions—including the 30 foot end and side impacts and the fire accident—are more rigorous than those imposed by design basis storage normal, off-normal and accident conditions.

Sections 11.1.2 and 11.2.11 use the results of [redacted] impact conditions to show adequate performance of [redacted] handling condition (Section 11.1.2) and of the [redacted] drop accident (Section 11.2.11). This section summarizes [redacted] the results of sections 11.1.2 and 11.2.11 are based. The [redacted] in accordance with ASME Section III, Subsection NB, and Subsection [redacted]

The complete evaluation of the transport normal and accident conditions loading on the canister and basket is presented in the NAC-STC SAR, Docket Number 71-9235.

11.3.1 Canister Impact Analysis

The canister is a right-circular shell fabricated from rolled 5/8-inch thick, Type 304L stainless steel plate and closed by a 1-inch thick, Type 304L stainless steel plate that is welded to one end of the shell. The canister is closed at the top end by the installation and welding of the 5-inch thick, Type 304 stainless steel shield lid and the 3-inch thick, Type 304L stainless steel structural lid.

11.3.1.1 Finite Element Model Description - Canister

A finite element model of the canister was constructed using ANSYS solid (SOLID45) elements. The model represents a one-half (180°) section of the canister and basket. The basket support discs were modeled with three-dimensional shell (SHELL63) elements. The model uses gap-spring elements to simulate contact between adjacent components. Interaction between the basket and canister were accomplished using three-dimensional gap elements (CONTACT52) along the periphery of the support disks. Contact between the canister and the cask inner shell is also modeled using CONTACT52 gap elements. Contact between the canister structural lid and shield lid is modeled using COMBIN40 combination elements in the axial degree of freedom. Simulation of the backing ring is accomplished using a ring of COMBIN40 spring gap elements connecting the shield lid and the canister in the axial direction at the lid lower outside radius. In addition, CONTACT52 elements are used to model interaction between the structural lid and canister shell and the shield lid and canister shell just below the respective lid weld joints. The size of the CONTACT52 gaps were determined from the 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 are assigned a stiffness of 1E8 lb/in.

Boundary conditions were applied to enforce symmetry at the plane of symmetry of the model. All nodes on the cask shell side of the canister to cask spring gap elements were fixed in all degrees of freedom. In addition, the axial and inplane rotational degrees of freedom of the basket nodes were fixed.

Figure 11.3.1.1-1 is a plot of the entire canister finite element model. An isolated view of the canister shield and structural lids portion of the model is presented in Figure 11.3.1.1-2 and an enlarged view of the model in the structural lid and shield lid weld regions is shown in Figure 11.3.1.1-3. The canister bottom plate portion of the model is shown in Figure 11.3.1.1-4. Identification of the sections for evaluating the linearized stresses in the canister is shown in Figure 11.3.1.1-5.

In the bottom end impact orientation, the fuel weight as well as the basket weight are transferred to the canister bottom plate. In the finite element model, the canister content weight is represented by applying a pressure load to the surface of the canister bottom plate. The support

disks inside the canister as shown in Figure 11.3.1.1-1 are set to be in-active for the bottom end impact analysis. The canister bottom plate is considered to be fully supported.

For the side impact condition, the loads from the canister contents weight is transferred through the support disks into the canister wall, which is considered to be backed by a rigid shell support of 71.0 inside diameter. The basket, canister and the assumed support shell have different radii, which implies that the contact angle between the components is dependent on the loading. Gap elements between the basket and the canister allow the interface to be dependent on the loading. The interface between the canister and the support shell is also represented by gap elements. The load due to the contents is applied to the basket via pressure acting in the plane of the disks. The weight is assumed to act over the effective width of 8.254 inches in which the disk is 0.5 inches thick. The content weight includes the 900 pounds per fuel assembly (for 36 fuel assemblies) plus the fuel tube weight (58.6 pounds/tube). This weight is distributed over the 22 support disks.

Use of the minimum size gap between the transportable storage canister structural and shield lids, 1×10^{-4} inch, results in the maximum stress and minimum margin of safety for the transportable storage canister impact analysis. Use of the maximum gap size (0.08 inches) results in a reduced margin of safety in some section locations, but does not result in the maximum stress / lowest margin of safety condition.

Figure 11.3.1.1-1 Canister Assembly Finite Element Model

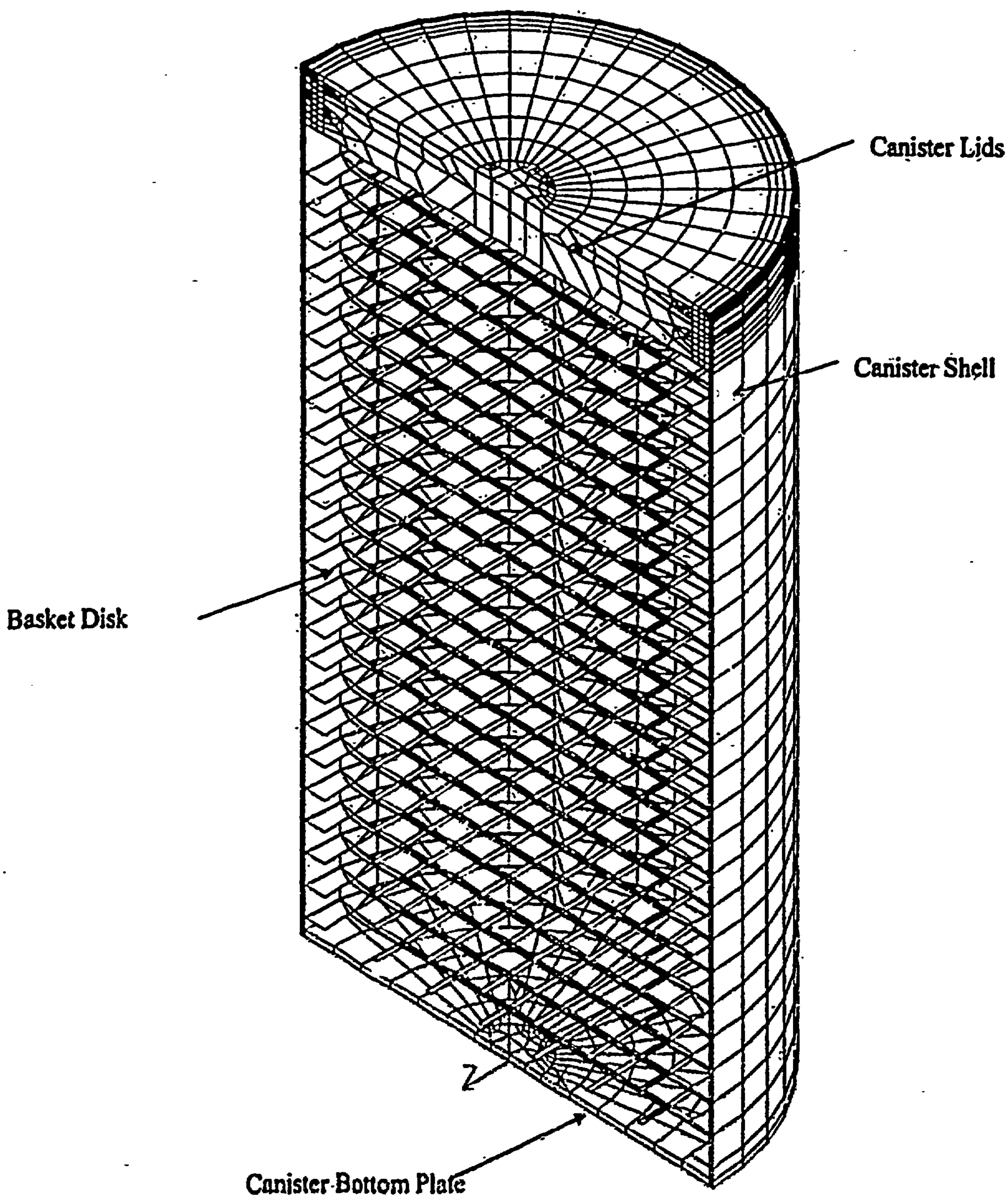


Figure 11.3.1.1-2 Canister Structural and Shield Lid Finite Element Mesh

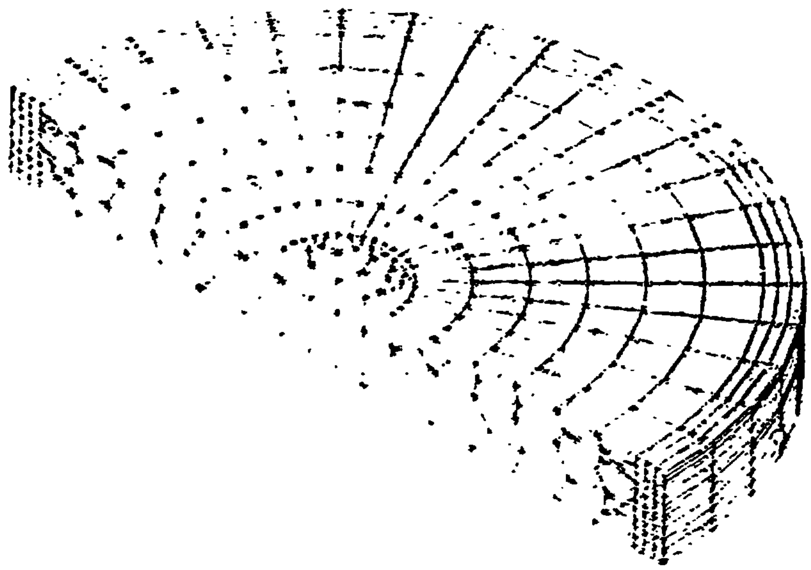


Figure 11.3.1.1-3 Structural and Shield Lid Weld Regions Finite Element Mesh

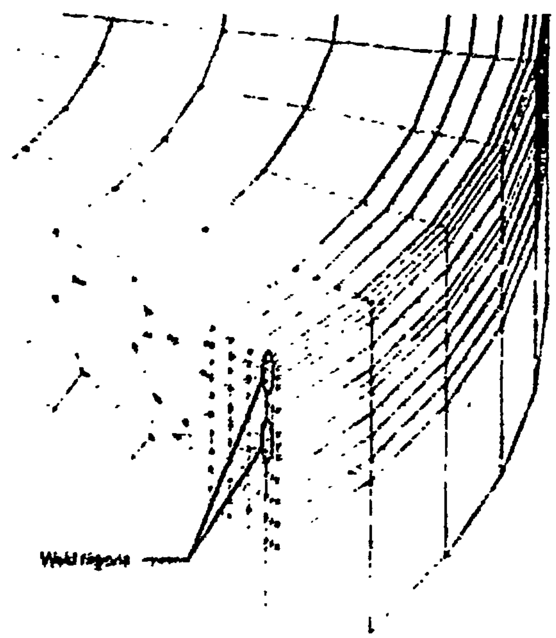
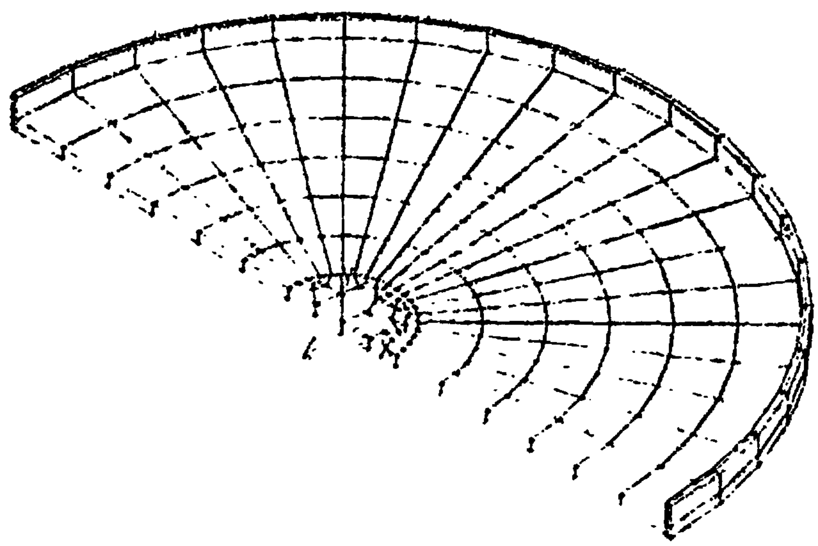


Figure 11.3.1.1-4 Canister Bottom Plate Finite Element Mesh



11.3.1.2 Canister Bottom and Side Impact Analysis

This section documents the evaluation of the canister for the 56.1 g bottom end impact and 55 g side impact load conditions. In addition to the impact loads, a 20 psig pressure load internal to the canister is considered. Note that the use of 20 psig is conservative, since the maximum internal pressure for the canister is 11.32 psi for normal conditions of storage. To determine the effect of the 20 psig pressure load, analyses with and without the pressure load were performed. The maximum nodal stresses are summarized as.

Impact Orientation	Internal Pressure (psi)	Maximum Stress Intensity (ksi)
Bottom End	0	4.6
Bottom End	20	3.0
Side	0	24.5
Side	20	27.1

It is concluded that the impact loading without pressure is the limiting condition for the bottom end impact case. For the side impact case, the addition of pressure to the impact loading is limiting. Therefore, a 20 psig pressure is conservatively applied to the inner surface of the canister model for the side impact conditions, while no internal pressure is used for the bottom end impact analyses.

The analysis results of the 56.1 g bottom impact conditions are presented in Tables 11.3.1.2-1 and 11.3.2-2. Results for the 55 g side impact condition are shown in Tables 11.3.1.2-3 and 11.3.1.2-4. The section stresses presented in the tables are identified by a section number. A cross-section of the canister showing the section numbers is presented in Figure 11.3.1.1-5. A summary of the canister minimum Margins of Safety for the evaluated impact conditions are shown in Table 11.3.1.2-5. The margins of safety are calculated as: M.S. = (allowable stress/S.I.) –1, where S.I. is the calculated stress intensity.

For the bottom end impacts, the stresses are essentially uniform around the circumference. For the side impact, the stresses vary around the circumference. Therefore, the circumferential angle at which the maximum stress occurs is noted in the table, in parentheses beside the section number. The allowable stresses presented in the tables are for Type 304L stainless steel, except

for section 10, which is for Type 304 stainless steel. These allowables are evaluated at 350°F (maximum calculated temperature in the canister is 319°F for normal conditions of storage).

Additionally, stress results for a 20 g bottom end impact and a 20 g side impact are listed in Table 11.3.1.2-6 through 11.3.1.2-9. Since the results are bounded by those of the 56.1 g bottom end impact and the 55 g side impact conditions, the stresses are listed without showing the Margin of Safety.

Table 11.3.1.2-1 Canister Analysis Results for the 56.1 g Bottom End Impact (Primary Membrane Stress)

Section[1] No.	Component Stresses (psi)						Principal Stresses (psi)			S.I.	Allow. Stress	Margin of Safety
	SX	SY	SZ	SXY	SYZ	SXZ	S1	S2	S3			
1	-70.7	-1971.0	-361.1	176.2	70.3	22.4	-51.6	-360.9	-1990.0	1938.0	39000	19.12
2	289.7	-5185.0	-1182.0	91.4	64.3	94.4	297.4	-1187.0	-5187.0	5485.0	39000	6.11
3	2.1	-4867.0	0.9	0.0	0.2	0.0	2.1	0.9	-4867.0	4869.0	39000	7.01
4	-2131.0	-2251.0	-1084.0	0.0	734.3	0.0	-729.3	-2131.0	-2605.0	1876.0	39000	19.79
5	2510.0	-2095.0	-1535.0	-310.9	-0.9	276.1	2549.0	-1553.0	-2117.0	4667.0	39000	7.36
6	651.9	1897.0	-676.0	15.4	64.7	-106.6	1900.0	661.0	-686.0	2583.0	39000	14.07
7	-3028.0	979.9	-1533.0	-368.1	39.6	5[illegible].9	1014.0	-1531.0	-3063.0	4077.0	39000	8.57
8	568.3	-3496.0	-2019.0	466.6	119.6	210.4	659.5	-2031.0	-3554.0	4214.0	39000	8.25
9	82.5	-682.0	94.2	6.5	58.3	-0.6	96.5	82.5	-686.4	785.0	39000	48.68
10	183.8	-98.9	166.1	43.7	-77.1	6.7	195.2	183.6	-125.7	320.9	[illegible]	[illegible]
11	-469.6	-9.4	-467.5	48.0	-75.2	-1.2	7.4	-470.1	-483.8	491.2	39000	78.40

1. Sections are identified in Figure 11.3.1.1-5. Stresses are reported in a cylindrical coordinate system (X,Y,Z) corresponding to radial, circumferential and axial directions respectively.

Table 11.3.1.2-2 Canister Analysis Results for the 56.1 g Bottom End Impact (Primary Membrane Plus Primary Bending Stress)

Section[1] No.	Component Stresses (psi)						Principal Stresses (psi)			S.I.	Allow. Stress	Margin of Safety
	SX	SY	SZ	SXY	SYZ	SXZ	S1	S2	S3			
1	398.0	-2678.0	-380.7	125.9	67.7	37.0	405.2	-380.8	-2685.0	3090.0	58500	17.93
2	-1988.0	-7553.0	85.1	0.0	-32.4	0.0	85.2	-1938.0	-7553.0	7638.0	58500	6.66
3	2.1	-4867.0	2.9	1.1	0.4	-0.1	3.0	2.1	-4867.0	4870.0	58500	11.01
4	-2259.0	-3096.0	-704.5	0.0	805.1	0.0	-458.8	-2259.0	-3342.0	2883.0	58500	19.29
5	1450.0	-10380.0	-4379.0	301.5	3.7	410.7	1486.0	-4407.0	-10390.0	11880.0	58500	3.92
6	3324.0	5269.0	932.2	617.6	53.3	-170.6	5567.0	3041.0	917.1	4650.0	58500	11.58
7	-2361.0	8360.0	938.8	-472.4	41.6	196.1	8401.0	950.4	-2393.0	10790.0	58500	4.42
8	4403.0	-1535.0	-491.6	605.7	160.4	345.7	4490.0	-503.9	-1659.0	6149.0	58500	8.51
9	105.1	-666.4	100.3	8.1	60.4	-1.7	108.1	104.0	-691.1	797.2	58500	72.38
10	6941.0	230.8	6895.0	28.2	-72.6	35.9	6960.0	6876.0	229.9	6730.0	56200	[illegible]
11	-4093.0	-195.3	-4093.0	47.4	-79.5	-18.6	-193.1	-4079.0	-4115.0	3922.0	58500	13.92

1. Sections are identified in Figure 11.3.1.1-5. Stresses are reported in a cylindrical coordinate system (X,Y,Z) corresponding to radial, circumferential and axial directions respectively.

Table 11.3.1.2-3 Canister Analysis Results for the 55 g Side Impact + Internal Pressure (20 psi) (Primary Membrane Stress)

Section[1]	Component Stresses (psi)						Principal Stresses (psi)				Allow.	Margin of
No.	SX	SY	SZ	SXY	SYZ	SXZ	S1	S2	S3	S.I.	Stress	Safety
1(0°)	-14880.4	1060.1	-9311.6	-235.5	-21.1	-904.6	1063.2	-9163.7	-14827.2	15917.7	39000	1.45
2(0°)	-3395.4	62.4	-7415.7	-314.3	-438.9	-478.6	111.6	-3358.7	-7501.7	7612.8	39000	4.12
3(180°)	-4.6	-1213.2	586.7	0.1	-4.0	-45.8	590.3	-8.1	-1213.2	1802.5	39000	20.64
4(9°)	-16945.4	3043.0	-4750.2	-333.0	2851.1	2020.7	3978.4	-5329.0	-17301.9	21276.1	39000	0.83
5(0°)	-10863.5	1232.1	-7804.7	-1756.4	1333.8	92.4	1662.0	-7961.0	-11146.6	12803.4	39000	2.05
6(0°)	-23467.7	-3813.8	-11125.6	-2768.3	1168.1	38.0	-3262.2	-11293.4	23855.7	20594.5	39000	0.89
7(9)	-11503.1	654.7	-4158.7	-31.6	1865.5	961.7	1299.2	-4673.6	-11639.5	12939.7	39000	2.01
8(0°)	-19367.6	-4979.8	-8614.2	-982.6	865.8	-756.5	-4597.7	-8785.2	-19472.5	14773.8	39000	1.64
9	-2146.5	-16.3	1105.2	-2.9	-14.7	-78.0	1107.3	-15.5	-2148.6	3255.9	39000	10.98
10	-1032.3	-6.4	331.8	-59.4	-2.8	-27.1	332.3	-4.9	-1036.3	1368.4	[illegible]	[illegible]
11	-1131.4	-1.3	373.3	-25.5	-5.1	-32.8	374.1	-0.7	-1132.5	1506.8	39000	24.88

1. Sections are identified in Figure 11.3.1.1-5. Stresses are reported in a cylindrical coordinate system (X,Y,Z) corresponding to radial, circumferential and axial directions respectively.

Table 11.3.1.2-4 Canister Analysis Results for the 55 g Side Impact + Internal Pressure (20 psi) (Primary Membrane Plus Primary Bending Stress)

Section[1]	Component Stresses (psi)						Principal Stresses (psi)				Allow.	Margin of
No.	SX	SY	SZ	SXY	SYZ	SXZ	S1	S2	S3	S.I.	Stress	Safety
1(0°)	-24841.3	457.1	-12761.5	-195.8	106.1	-824.5	459.5	-12709.0	-24904.3	23365.6	68500	1.31
2(0°)	-2630.9	-1326.5	-8993.8	-293.8	-641.0	-126.3	-1218.5	-2682.3	-9050.5	7832.0	68500	6.47
3(0°)	89.9	2039.5	3198.2	2.7	38.7	156.9	3207.7	2038.5	81.9	3125.9	68500	17.71
4(9°)	-13894.0	9446.8	-2019.6	93.3	2241.9	2829.1	9886.2	-1813.0	-14533.6	24421.9	68500	1.40
5(0°)	-14251.0	2077.3	-6923.9	-7052.1	1046.5	-16.1	2448.4	-7037.2	-14512.6	16966.3	58500	2.45
6(0°)	-32968.0	-7832.0	-15215.2	-4140.9	1477.5	185.3	-6920.8	-16456.4	-33639.1	26718.3	68500	1.19
7(9°)	-9079.8	4969.3	-1993.4	53.6	1402.0	1674.6	5250.3	-1894.8	-9460.5	14711.9	58500	2.98
8(0°)	-28144.4	-7245.8	-12646.1	-2107.7	1351.6	-307.9	-6716.3	-12971.2	-28354.1	21643.1	68500	1.70
9	-2172.7	-33.2	1088.4	-2.8	-14.9	-75.6	1089.5	-33.5	-2174.8	3264.3	68500	16.92
10	-1360.0	-10.9	297.9	-63.0	-2.9	-40.8	292.0	-8.0	-1364.2	1663.1	[illegible]	[illegible]
11	-1344.3	-1.2	354.1	-25.6	-5.0	-39.5	355.1	-0.7	-1348.4	1700.8	68500	33.40

1. Sections are identified in Figure 11.3.1.1-5. Stresses are reported in a cylindrical coordinate system (X,Y,Z) corresponding to radial, circumferential and axial directions respectively

Table 11.3.1.2-5 Summary of Minimum Margin of Safety for Canister Impact Analysis

Drop Orientation	Loading Condition	Stress Evaluated	Minimum Margin of Safety	Section No.*
bottom end	56.1 g impact	P_m	6.11	2
bottom end	56.1 g impact	$P_m + P_b$	3.92	5
side	55 g impact + internal pressure (20 psi)	P_m	0.83	4
side	55 g impact + internal pressure (20 psi)	$P_m + P_b$	1.19	6

* See Figure 11.3.1.1-5 for section locations.

Table 11.3.1.2-6 Bottom End Impact (20 g) - Primary Membrane Stresses[1]

Section	P_m Stresses (psi)						Principal Stresses (psi)			
No.	SX	SY	SZ	SXY	SYZ	SXZ	S1	S2	S3	S.I.
1	-25.5	-704.5	-129.2	63.1	25.1	8.0	-18.5	-129.1	-711.3	692.8
2	103.7	-1854.0	-422.7	32.7	19.4	33.8	106.4	-424.7	-1855.0	1961.0
3	0.8	-1740.0	0.3	0.0	0.1	0.0	0.8	0.3	-1740.0	1741.0
4	-760.4	-800.7	-389.0	0.0	263.2	0.0	-260.7	-760.4	-929.1	668.4
5	905.9	-756.4	-546.9	-107.1	0.2	99.9	919.5	-553.5	763.5	1683.0
6	196.5	704.2	-233.2	5.1	30.5	-37.5	705.2	199.7	-237.4	942.6
7	-1072.0	339.2	-546.8	-127.7	13.9	19.5	350.9	-546.2	-1084.0	1435.0
8	207.7	-1240.0	-718.6	162.8	42.2	74.7	232.4	-723.0	-1260.0	1493.0
9	29.5	-243.2	33.7	2.3	20.8	-0.2	35.3	29.5	-244.7	280.0
10	70.7	23.7	69.5	16.0	-20.2	0.7	80.9	71.0	12.1	68.8
11	-161.1	50.4	-157.0	16.1	-39.4	-1.4	58.9	-161.5	-165.1	223.9

1. Sections are identified in Figure 11.3.1-5. Stresses are reported in a cylindrical coordinate system (X,Y,Z) corresponding to radial, circumferential and axial directions respectively.

Table 11.3.1.2-7 Bottom End Impact (20 g) Primary Membrane Plus Primary Bending Stresses[1]

Section	$P_m + P_b$ Stresses (psi)						Principal Stresses (psi)			
No.	SX	SY	SZ	SXY	SYZ	SXZ	S1	S2	S3	S.I.
1	142.5	-957.2	-136.2	45.1	24.2	13.3	145.1	-136.2	-959.7	1105.0
2	-711.1	-2701.0	30.4	0.0	-11.5	0.0	30.5	-711.1	-2701.0	2731.0
3	0.8	-1741.0	1.0	0.4	0.1	0.0	1.053	0.8	-1741.0	1742.0
4	-808.8	-1111.0	-254.6	0.0	290.6	0.0	-165.3	-808.8	-1200.0	1035.0
5	522.1	-3735.0	1569.0	111.2	1.6	147.5	535.4	-1580.0	-3738.0	4273.0
6	1194.0	1910.0	344.1	288.9	19.6	-60.7	2012.0	1097.0	338.7	1674.0
7	-838.6	2970.0	331.1	-166.6	14.8	69.5	2977.0	335.2	-850.0	3827.0
8	1555.0	-560.8	-178.8	213.3	56.8	122.6	1586.0	-183.1	-587.0	2173.0
9	37.6	-244.7	35.9	2.9	21.5	-0.6	38.0	37.2	-246.4	284.4
10	2529.0	140.9	2515.0	13.6	-17.2	12.0	2536.0	2508.0	140.7	2395.0
11	-1383.0	-19.4	-1389.0	14.6	-41.9	-6.1	-17.9	-1380.0	-1393.0	1375.0

1. Sections are identified in Figure 11.3.1-5. Stresses are reported in a cylindrical coordinate system (X,Y,Z) corresponding to radial, circumferential and axial directions respectively.

Table 11.3.1.2-8 Side Impact (20 g) + Internal Pressure (20 psi) - Primary Membrane Stresses[1]

Section	P_m Stresses (psi)						Principal Stresses (psi)			
No.	SX	SY	SZ	SXY	SYZ	SXZ	S1	S2	S3	S.I.
1	-10227.0	856.0	-4685.1	375.9	199.7	-908.3	873.9	-4543.6	-10386.4	11262.0
2	5431.7	1884.3	-732.0	-276.8	-292.7	-442.5	5481.0	1902.2	-800.0	6281.1
3	-692.7	765.9	1360.0	4.7	-5.3	114.9	1367.4	765.9	-699.1	2065.7
4	-292.6	2090.9	659.1	159.2	1019.3	427.2	2660.3	279.9	-482.3	3142.7
5	-9019.0	-32.9	-3701.6	-1049.6	954.2	-599.5	345.1	-3914.4	-9184.7	9529.7
6	-15561.2	-2614.2	-4803.6	-1996.5	811.3	-748.8	-2010.2	-5071.0	-15896.8	13883.5
7	1925.2	1097.9	814.8	69.9	485.8	-45.5	1931.5	1460.7	446.1	1484.8
8	-14051.2	-2852.2	-3555.8	-677.4	842.2	-1202.7	-2161.2	-4080.1	-14219.0	12058.9
9	-478.9	-198.6	806.7	71.4	22.0	-22.2	807.5	-181.7	-496.6	1304.5
10	-425.3	-1.7	78.9	-25.7	6.7	-10.9	79.7	-0.7	-427.1	506.8
11	-382.5	-2.7	174.5	-16.4	2.8	-13.3	174.9	-2.1	-383.6	558.5

1. Sections are identified in Figure 11.3.1.1-5. Stresses are reported in a cylindrical coordinate system (X,Y,Z) corresponding to radial, circumferential and axial directions respectively.

Table 11.3.1.2-9 Side Impact (20 g) + Internal Pressure (20 psi) - Primary Membrane Plus Primary Bending Stresses[1]

Section	$P_m + P_b$ Stresses (psi)						Principal Stresses (psi)			
No.	SX	SY	SZ	SXY	SYZ	SXZ	S1	S2	S3	S.I.
1	-26047.2	-2107.7	-10104.3	413.8	217.7	-1116.8	-2095.1	-10030.9	-26131.1	24044.4
2	-4429.3	-11241.0	1092.6	364.4	886.8	-75.4	1156.6	-4409.4	-11324.9	12478.3
3	-759.6	1193.5	2829.1	7.8	-5.9	227.3	2842.8	1198.5	-774.0	3617.7
4	599.0	4274.1	2236.7	208.0	755.2	1088.4	4603.4	2456.9	50.1	4553.0
5	-11146.6	26.7	-3044.1	-1644.2	728.4	-827.9	466.5	-3185.6	-11450.7	11912.1
6	-19916.8	-4548.8	-6797.0	-2643.5	1026.7	-594.2	-3707.8	-7185.0	-20447.7	16735.7
7	2385.6	321.5	419.3	86.5	664.6	-648.9	1677.8	651.1	-962.1	2639.3
8	-18046.4	-4398.9	-5812.7	-1420.9	1203.8	-864.6	-3522.2	-6506.6	-18235.2	14711.9
9	-956.6	-172.7	869.4	75.4	9.5	10.3	869.6	-165.6	-963.9	1833.0
10	-1095.8	-29.5	-493.6	-25.3	7.2	-18.4	-28.7	-493.2	-1096.8	1068.5
11	-814.4	-19.7	-190.1	-16.0	3.0	-15.9	-19.4	-189.8	-815.2	795.8

1. Sections are identified in Figure 11.3.1.1-5. Stresses are reported in a cylindrical coordinate system (X,Y,Z) corresponding to radial, circumferential and axial directions respectively.

11.3.1.3 Canister Buckling Evaluation for the Bottom End Impact

The canister shell is axially loaded by the weights of the structural lid, the shield lid, and the inertial weight of the shell during a bottom end impact. The impact load amplification factor is 56.1g. The shell is evaluated as an unsupported, right circular cylinder using a critical buckling load per Blake, 2nd Edition, "Practical Stress Analysis in Engineering Design."

$$S_{cr} = \frac{E\left(0.605 - 10^{-7} M^2\right)}{M\left(1 + 0.004\phi\right)}$$

$$= 40.3 \text{ ksi}$$

The canister material is Type 304L stainless steel. Conservatively assume the material temperature to be at 400°F for this impact condition.

$E = 26.5E{+}03$ ksi $R = (69.39 + 0.625)/2$

$= 35.01$ inches (mid-radius of the canister shell)

$S_y = 17.5$ ksi $t = 0.625$ inches (thickness of the canister)

$\phi = E/S_y$ and $m = R/t$

$= 1514.3$ $= 56.0$

The axial compression load in the canister shell is:

$$P_a = [\,(\pi/4)(69.03^2)(8)(0.291) + (\pi/4)(70.64^2 - 69.39^2)(121.5)(0.291)\,]\,(56.1)$$

$$P_a = 761{,}457 \text{ pounds}$$

and the axial compression stress is:

$$S_a = \frac{P_A}{\left(\pi/4\right)\left(70.64^2 - 69.39^2\right)}$$

$$S_a = 5{,}540 \text{ psi}$$

The margin of safety is:

$$(S_{cr}/S_a) - 1 = +\ 6.3$$

11.3.2 Canister Fuel Basket [redacted] Analysis

[redacted]

The fuel basket in the transportable storage canister is designed to contain up to 36 Yankee class fuel assemblies. The basket structure has a right circular cylinder configuration and consists of 36 square tubes supported by 22 circular support disks, and a circular top and bottom plate, which are retained by eight axial tie rods. The support disks and top and bottom plates are seperated and supported by split spacers at the tie rods. The configuration of the basket is shown in Figure 11.3.2-1.

Each fuel tube has an 7.8-inch square inside dimension, a 0.048-inch thick wall, and can hold one intact fuel assembly. The fuel assemblies together with the tubes are laterally supported in square holes in the stainless steel support disks. Each circular support disk is 0.5 inches thick and 69.15 inches in diameter. There are three different web widths in the support disks. One web width is 0.750 inches between the holes, one web width is 0.810 inches between the holes, and one web width is 0.875 inches between holes. The top and bottom plates are both 0.5 inch thick and have the same diameter as the support disks. The disks are spaced and retained by tie rods and split spacers (spacers) at eight locations near the periphery of each disk to form an integral basket assembly. The fuel basket contains the fuel and is enclosed by the canister. The canister has a 70.64 inch outer diameter and 5/8 inch thick walls. The overall length of the canister cavity is 122.5 inches, which encompasses the entire fuel assembly length and the thickness of the shield lid and structural lid. The canister shell is fabricated from Type 304L stainless steel.

The material of the support disks is 17-4 PH stainless steel. The top plate and the bottom plate are fabricated from Type 304 stainless steel. The fuel tubes are made from Type 304 stainless steel, which also encases the BORAL neutron absorbing material. The tie rods and spacers are fabricated from Type 304 stainless steel. The fuel tubes are not structural components, and are not considered in the basket evaluation. The tie rods and spacers locate and structurally assemble the circular support disks, heat transfer disks, and the top and bottom plates to form an integral assembly. The spacers carry the weight of the support disks, heat transfer disks, endplate and

their own weight in the end impact loading condition. The end impact analysis uses classical closed form methods that are evaluated independent of the finite element basket model. The support disk structural evaluation is performed using a finite element model of a single disk.

The structural analysis of the basket components is in accordance with ASME Code, Section III, Division 1, Subsection NG, "Core Support Structures." In addition, the stainless steel/BORAL composite fuel tube has been evaluated for a postulated impact load.

11.3.2.1 Stress Evaluation of Support Disk [redacted]

To determine the structural adequacy of the support disks [redacted] a load equal to the weight of the fuel and tubes multiplied by an amplification factor is applied to the support disk structure to simulate an end impact. For accident conditions, the amplification [redacted] for the end impact is 56.1 g.

A finite element analysis is performed, utilizing the ANSYS computer code, to calculate the stresses in a support disk in accordance with ASME Code Section III, Subsection NG. In this subsection, linearized stresses of cross sections of the structure are compared to the allowable stresses. The maximum primary membrane stress intensity calculated in the support disk is compared to the allowable stress limits for accident conditions is, 0.7 S_u or 2.4 S_m, whichever is less.

11.3.2.1.1 Finite Element Model Description

A finite element model is used to evaluate the basket support disk for the end impact accident condition in which the loads are perpendicular to the plane of the disk.

The model for the end drop condition is constructed using ANSYS SHELL63 elements. It consists of a single support disk with a thickness of 0.5 inches. The shell elements accommodate the out-of-plane bending, which is present in the end-drop condition. In the end drop, the support disk is restrained by the split spacers on the eight tie rods. The nodes corresponding to the location of the tie rods are restrained in the out of plane direction (the cask axial direction). Four additional in-plane transitional restraints are specified at the outer edge of the model (located 90° apart from each other) in the tangential direction to prohibit rigid body

[redacted]

[redacted]

The temperature distribution of the support disk [redacted] the thermal analysis. The bounding thermal cond[redacted] determine the material properties. Allowable [redacted] temperatures of 539° F for Thermal Condition 2 and [redacted]

To determine the most critical regions, a series of cross sections [redacted] locations are identified in Figure 11.3.2-3. Table 11.3.2-1 [redacted] cross section end points.

11.3.2.1.2 Support Disk End Impact Analysis Results

A structural analysis is performed using ANSYS to evaluate the effect of a 56.1 g end impact which corresponds to the most severe out-of-plane loading. Linearized stresses at the cross sections identified in Figure 11.3.2-3 are compared to stress allowables in accordance with the ASME Code, Section III, Subsection NG.

The stress evaluation results for the 56.1 g end impact condition are:

Thermal Condition	P_m Stress Intensity (ksi)	M.S.	P_m+P_b Stress Intensity (ksi)	M.S.
2	0	N/A	52.9	1.42
3	0	N/A	53.9	1.51

[redacted]

[redacted]

[redacted]

[redacted]

11.3.2.2 Evaluation of Tie Rods and Spacers for an End Impact Condition

The design end impact loading for the basket is 56.1 g. The structural capacity of the spacers supporting the basket is evaluated using classical analysis. Accident loading due to the 56.1 g impact of the fuel basket was compared to the stress limit of 0.7 S_u in accordance with Subarticle NF 1440 of the ASME Code.

No detailed evaluation of the tie rods is required. The tie rods serve basket assembly purposes and are not part of the load path for the condition evaluated. The tie rods are loaded during fabrication by a 190 ft-lbs preload. Under impact conditions, the preload will be reduced. The tie rod design is, therefore, acceptable by inspection.

During the end impact, the spacers are loaded with the weight of 22 support disks, the aluminum heat transfer disks, one end plate, and the weight of the spacers. The load is resisted by the effective area of 8 spacers. The compressive stresses are calculated on the effective area of the spacer.

The material allowable stress is conservatively selected at a temperature of 500°F. The analysis input is :

stress limits	=	0.7 S_u (accident condition) (more limiting than 2.4 S_m)
loading criteria (g)	=	56.1g (accident condition)
evaluation temperature	=	500°F

Canister Basket Parameters

fuel basket weight	=	9,530 lbs
bottom weldment weight	=	438 lbs
fuel tube weight (36 tubes)	=	2,164 lbs
rod diameter	=	1.13 in
spacer outer diameter	=	2.50 in

Materials

tie rod	=	SA 479 Type 304 Stainless Steel
spacer	=	A511 Type 304 Stainless Steel

Material Allowable

Type 304 SS	=	S_m = 17,500 psi (500°F)
	=	S_u = 63,500 psi (500°F)

The spacer load is calculated as follows:

Total weight of basket	=	9,530 lbs
Less weight of bottom weldment	=	-438 lbs
Less weight of fuel tubes	=	-2,164 lbs
Therefore,		
1 g load on spacers	=	6,928 lbs
Applied g level	=	56.1g
End impact load on spacers	=	6,928 x 56.1
	=	388,661 lbs

The effective area of one spacer at each of eight locations supporting the weight of the support disks is equal to the net area of the spacer and is calculated as:

$$A = \frac{3.14 \times \left(2.5^2 - 1.25^2\right)}{4}$$
$$= 3.68 \text{ in}^2$$

The average compressive stress, S_c, in the spacer is:

$$S_c = \frac{388{,}661}{8 \times 3.68}$$

$$= 13{,}202 \text{ psi}$$

The allowable stress for Type 304 SS under accident conditions is 0.7 S_u.

$$S_u = 63{,}500 \text{ psi}$$
$$0.7\,S_u = 0.7 \times 63{,}500$$
$$= 44{,}500 \text{ psi}$$

The margin of safety (MS), which is defined as $\frac{0.7S_u}{S_c} - 1$, is calculated as:

$$\frac{44{,}450}{13{,}202} - 1 = 2.37$$

Therefore, the spacers are structurally adequate for a 56.1 g end impact under accident conditions.

Figure 11.3.2-1 Canistered Yankee Class Fuel Basket Assembly

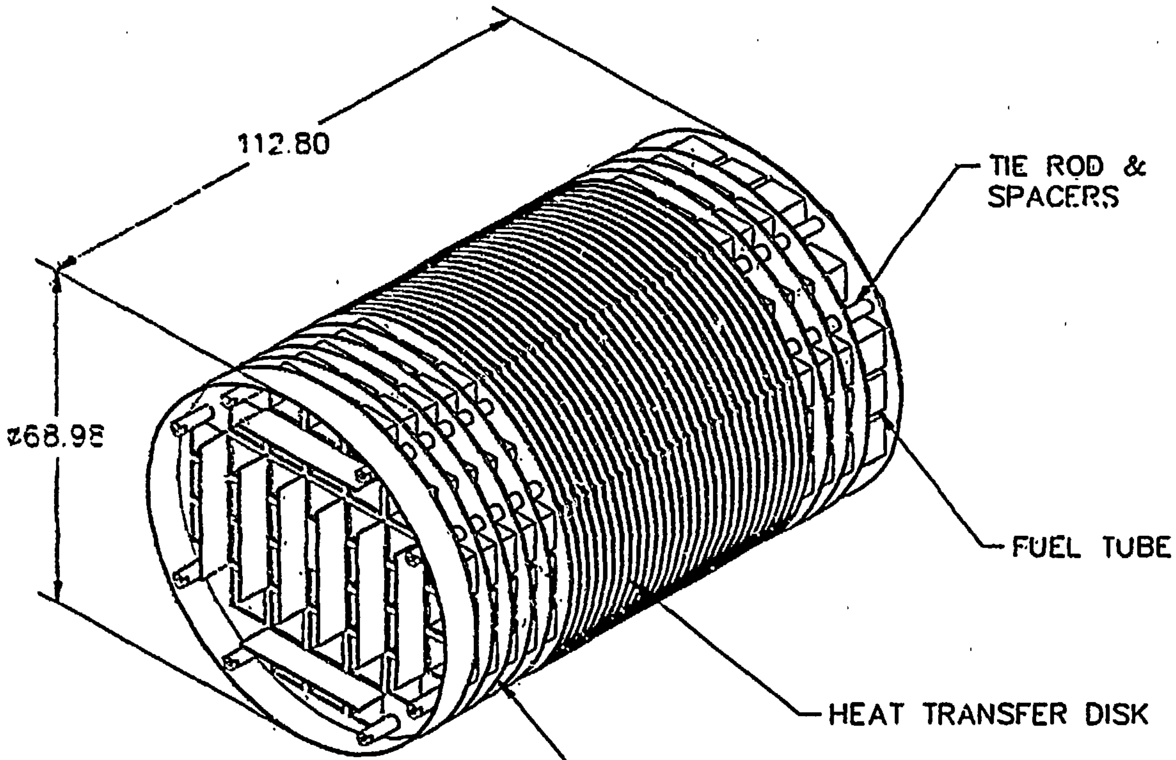


Figure 11.3.2-2 Fuel Basket Support Disk [redacted]

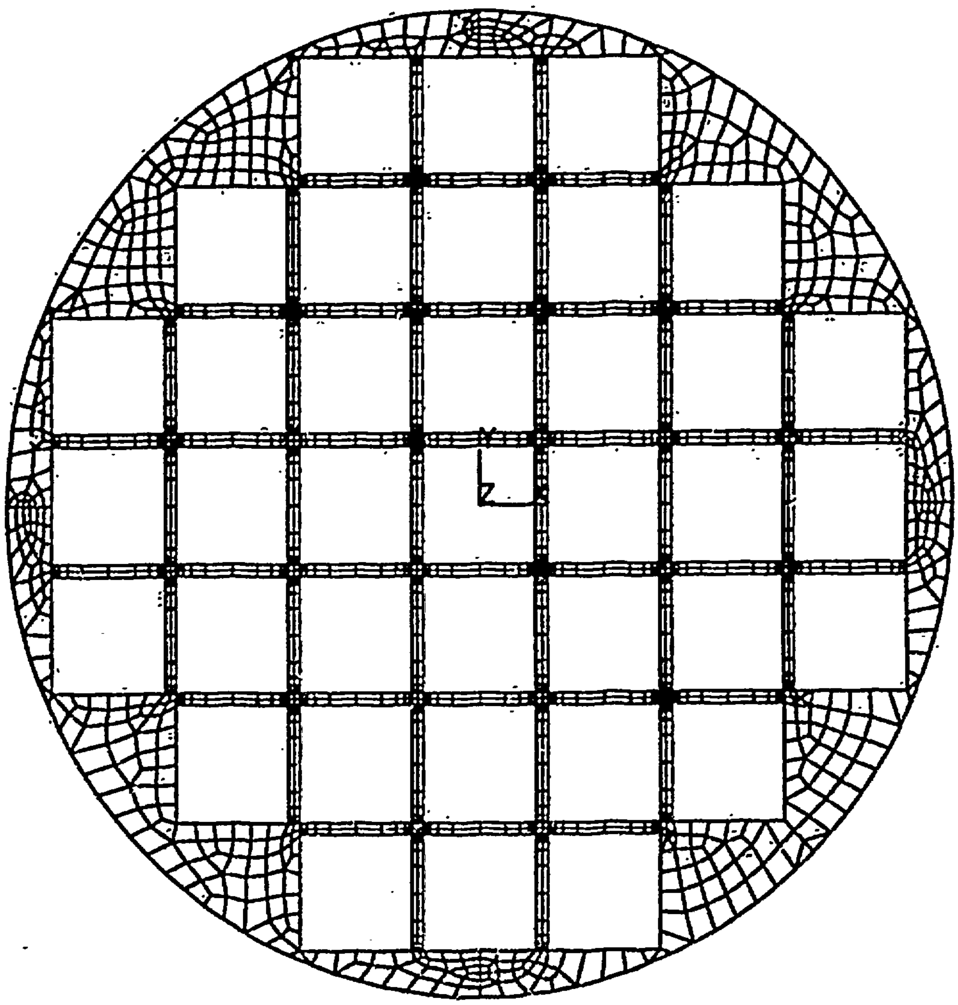


Figure 11.3.2-3 [redacted]

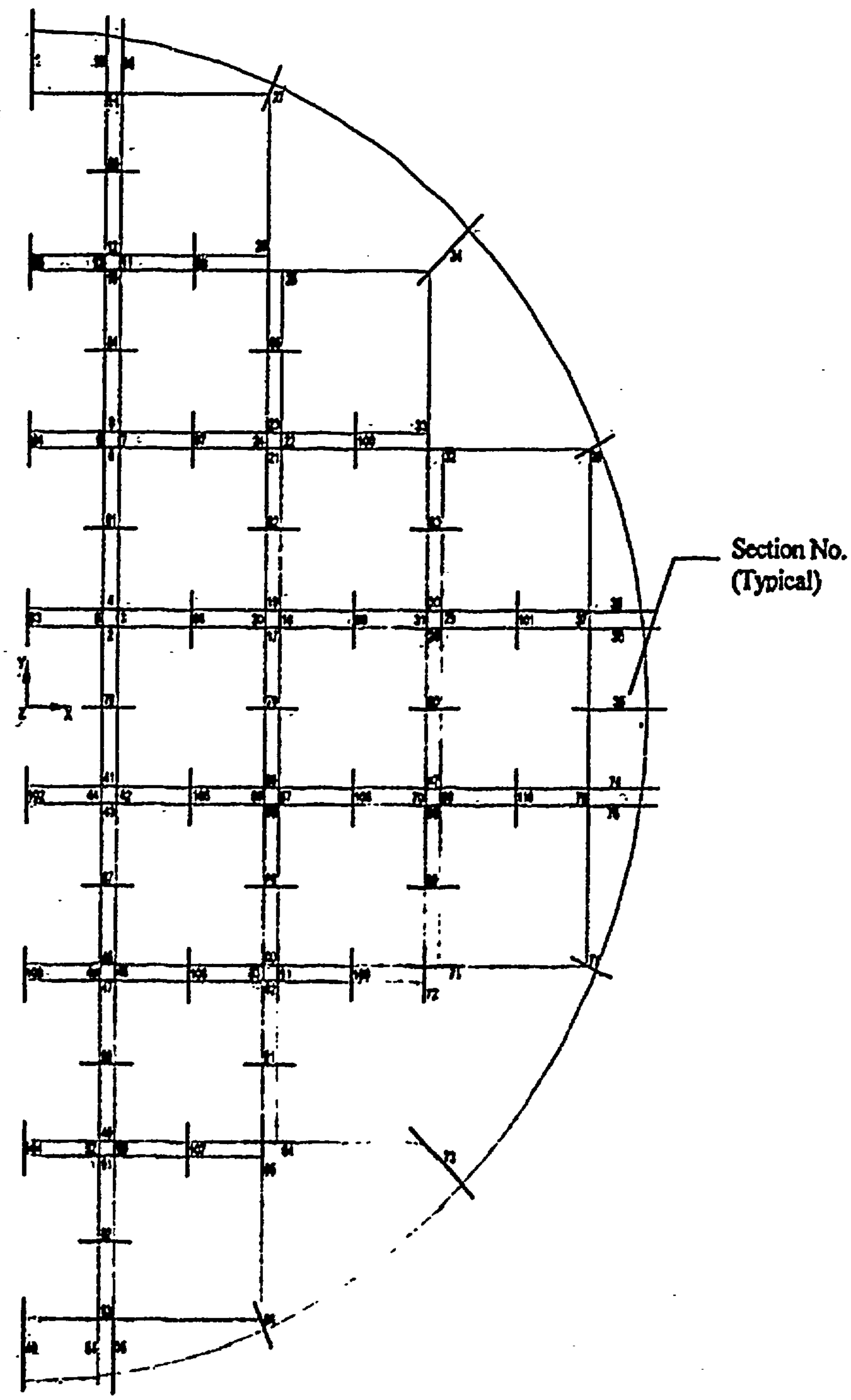


Table 11.3.2-1 [redacted] Support Disk [redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]		
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	10	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	11	12	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	13	14	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	15	16	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	17	18	[redacted]	[redacted]	[redacted]	[redacted]
10	19	20	[illegible]	[redacted]	[redacted]	[redacted]
11	21	22	5.00	[redacted]	[redacted]	[redacted]
12	23	24	5.08	[redacted]	[redacted]	[redacted]
13	25	26	4.13	[redacted]	[redacted]	[redacted]
14	27	28	4.13	[redacted]	[redacted]	[redacted]
15	29	30	4.13	[redacted]	[redacted]	[redacted]
16	31	32	5.00	[redacted]	[redacted]	[redacted]
17	33	34	13.26	[redacted]	[redacted]	[redacted]
18	35	36	14.07	[redacted]	[redacted]	[redacted]
19	37	38	14.07	[redacted]	[redacted]	[redacted]
20	39	40	13.26	5.00	[redacted]	[redacted]
21	41	42	13.26	13.26	[illegible]	[illegible]
22	43	44	14.07	13.26	[illegible]	[illegible]
23	45	46	14.07	14.07	[illegible]	[illegible]
24	47	48	13.26	14.07	[illegible]	[illegible]
25	49	50	13.26	22.32	[illegible]	[illegible]
26	51	52	13.26	22.32	[illegible]	[illegible]
27	53	54	13.26	31.32	[illegible]	[illegible]
28	55	56	22.32	[illegible]	[illegible]	[illegible]
29	57	58	23.07	4.13	23.07	5.00
30	59	60	23.07	5.00	22.32	5.00
31	61	62	22.32	5.00	22.32	4.13
32	63	64	22.32	13.26	[illegible]	13.26
33	65	66	22.32	13.26	22.32	14.07
34	67	68	22.32	22.32	24.39	24.59
35	69	70	31.32	0.00	[illegible]	0.00

Table 11.3.2-1 [redacted] Support Disk [redacted]

			[redacted]			
[redacted]	[redacted]	[redacted]	[redacted]		[redacted]	
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
45	89	90	4.13	[redacted]	[redacted]	[redacted]
46	91	92	5.00	[redacted]	[redacted]	[redacted]
47	93	94	5.00	[redacted]	[redacted]	[redacted]
48	95	96	4.13	[redacted]	[redacted]	[redacted]
49	97	98	4.13	[redacted]	[redacted]	[redacted]
50	99	100	5.00	[redacted]	[redacted]	[redacted]
51	101	102	5.00	[redacted]	[redacted]	[redacted]
52	103	104	4.13	[redacted]	[redacted]	[redacted]
53	105	106	4.13	[redacted]	[redacted]	[redacted]
54	107	108	4.13	[redacted]	[redacted]	[redacted]
55	109	110	5.00	[redacted]	[redacted]	[redacted]
56	111	112	13.26	4.13	[redacted]	[redacted]
57	113	114	14.07	4.13	[redacted]	[redacted]
58	115	116	14.07	5.00	[redacted]	[redacted]
59	117	118	13.26	5.00	[redacted]	[redacted]
60	119	120	13.26	13.26	[redacted]	[redacted]
61	121	122	14.07	13.26	[redacted]	[redacted]
62	123	124	14.07	14.07	[redacted]	[redacted]
63	125	126	13.26	14.07	[redacted]	[redacted]
64	127	128	13.26	22.32	[redacted]	[redacted]
65	129	130	13.26	22.32	[redacted]	[redacted]
66	131	132	13.26	31.32	[redacted]	[redacted]
67	133	134	22.32	4.13	[redacted]	[redacted]

Table 11.3.2-1 [redacted] Support Disk
[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	150	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	151	152	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	153	154	[redacted]	[redacted]	[redacted]	[redacted]
78	155	156	4.13	[redacted]	[redacted]	[redacted]
79	157	158	13.26	[redacted]	[redacted]	[redacted]
80	159	160	22.32	[redacted]	[redacted]	[redacted]
81	161	162	4.13	[redacted]	[redacted]	[redacted]
82	163	164	13.26	[redacted]	[redacted]	[redacted]
83	165	166	22.32	[redacted]	[redacted]	[redacted]
84	167	168	4.13	[redacted]	[redacted]	[redacted]
85	169	170	13.26	[redacted]	[redacted]	[redacted]
86	171	172	4.13	27.20	[redacted]	27.20
87	173	174	4.13	9.13	[redacted]	[redacted]
88	175	176	13.26	9.13	[redacted]	[redacted]
89	177	178	22.32	9.13	[redacted]	[redacted]
90	179	180	4.13	18.19	[redacted]	[redacted]
91	181	182	13.26	18.19	[redacted]	[redacted]
92	183	184	4.13	27.20	[redacted]	27.20
93	185	186	0.00	4.13	0.00	5.00
94	187	188	0.00	13.26	0.00	14.07
95	189	190	0.00	22.32	0.00	23.07
96	191	192	9.13	4.13	9.13	5.00
97	193	194	9.13	13.26	[redacted]	14.07
98	195	196	9.13	22.32	[redacted]	[redacted]
99	197	198	18.19	4.13	[redacted]	[redacted]
100	199	200	18.19	13.26	[redacted]	[redacted]

Table 11.3.2-1 [redacted] Support Disk
[redacted]

			[redacted]			
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[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
105	209	210	-10.02	[redacted]	[redacted]	[redacted]
106	211	212	-20.29	[redacted]	[redacted]	[redacted]
107	213	214	-10.02	[redacted]	[redacted]	[redacted]
108	215	216	-31.18	[redacted]	[redacted]	[redacted]
109	217	218	5.39	[redacted]	[redacted]	[redacted]
110	219	220	5.39	[redacted]	[redacted]	[redacted]

Table 11.3.2-[redacted] [redacted]

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[redacted]	[redacted]	15.9	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	-27.8	[redacted]	[redacted]	[redacted]	[redacted]
87	[redacted]	-46.7	[redacted]	[redacted]	[redacted]	[redacted]
92	-0.1	-46.5	0.3	[redacted]	[redacted]	[redacted]
[redacted]	-45.4	0.0	0.3	[redacted]	[redacted]	[redacted]
13	-45.7	6.6	1.3	[redacted]	[redacted]	[redacted]
94	-45.6	-0.1	-0.1	[redacted]	[redacted]	[redacted]
[redacted]	0.1	-45.5	-0.2	[redacted]	[redacted]	[redacted]
104	-45.5	-0.1	-0.2	[redacted]	[redacted]	[redacted]
95	-44.5	-0.1	-0.2	[redacted]	[redacted]	[redacted]
78	0.1	-44.0	0.3	[redacted]	[redacted]	[redacted]
102	-44.0	-0.1	0.3	[redacted]	[redacted]	[redacted]
93	-43.8	-0.1	0.1	[illegible]	127.8	1.92
57	6.3	35.1	-10.2	38.3	127.8	2.34
66	-35.1	6.3	-10.2	38.3	127.8	2.34
89	-0.6	37.7	3.3	38.0	127.8	2.36
88	-0.5	-37.5	1.4	37.5	127.8	2.41
20	37.2	-28.2	0.0	37.2	127.8	2.43
45	28.1	-36.9	1.0	37.0	127.8	2.46
87	-0.3	-36.0	0.8	36.0	127.8	2.55
82	0.3	-35.7	-0.5	36.0	127.8	2.55
37	3.8	-25.9	3.9	35.5	127.8	2.60
81	0.2	-35.2	-0.3	35.4	127.8	2.61
63	26.3	8.1	-4.3	35.4	127.8	2.61
106	-34.8	0.4	1.4	35.3	127.8	2.62
107	-34.0	0.5	-3.3	35.1	127.8	2.64
83	0.5	33.9	2.9	34.9	127.8	2.66
97	34.4	0.3	-0.7	34.7	127.8	2.68
105	-34.4	0.2	0.8	34.6	127.8	2.69
96	-34.4	0.2	-0.2	34.6	127.8	2.69

Note: See Figure 11.3.2-3 for section locations and definition of coordinate system.

Table 11.3.2-[illegible] [illegible]

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
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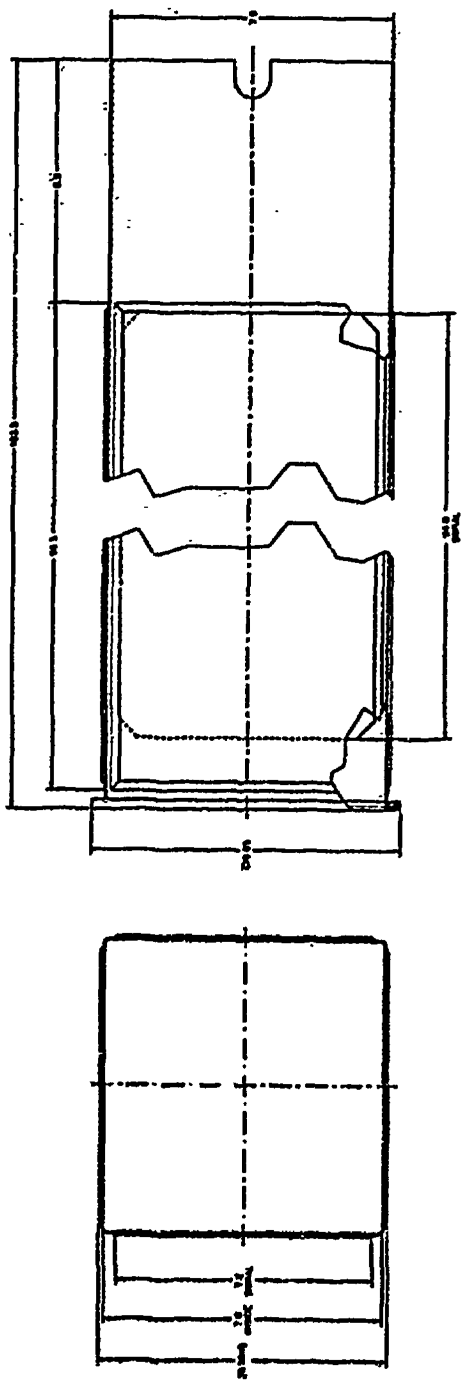
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	-51.7	-19.2	-0.9	[illegible]	[illegible]	[illegible]
[illegible]	-6.5	-51.4	2.7	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	-24.5	-0.4	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	-6.1	2.5	[illegible]	[illegible]	[illegible]
[illegible]	-0.1	-46.7	0.3	[illegible]	[illegible]	[illegible]
103	-46.7	0.0	0.3	[illegible]	[illegible]	[illegible]
67	-6.6	-46.5	-1.7	[illegible]	[illegible]	[illegible]
94	-45.8	-0.1	-0.1	[illegible]	[illegible]	[illegible]
13	-45.6	-6.8	1.2	[illegible]	[illegible]	[illegible]
80	-0.1	-45.3	-0.2	[illegible]	[illegible]	[illegible]
78	-0.1	-45.3	0.3	[illegible]	[illegible]	[illegible]
104	-45.3	-0.1	-0.2	[illegible]	[illegible]	[illegible]
102	-45.3	-0.1	0.3	[illegible]	[illegible]	[illegible]
93	-45.0	-0.1	0.1	45.0	[illegible]	[illegible]
95	-44.3	-0.1	-0.2	44.3	[illegible]	[illegible]
20	-38.2	-28.4	-0.2	38.2	135.0	[illegible]
45	-28.3	-37.9	1.2	38.0	135.0	[illegible]
88	-0.5	-37.7	1.7	37.8	135.0	2.57
89	-0.6	-37.5	-3.1	37.8	135.0	2.57
77	-6.1	-34.3	-9.9	37.4	135.0	2.61
66	-34.3	-6.1	-9.9	37.4	135.0	2.61
87	-0.3	-36.9	1.0	36.9	135.0	2.66
81	0.2	-36.1	-0.5	36.3	135.0	2.72
82	0.3	-36.0	-0.7	36.3	135.0	2.72
106	-35.1	0.3	1.7	35.6	135.0	2.79
105	-35.3	0.2	1.0	35.5	135.0	2.80
96	-35.3	0.2	-0.3	35.5	135.0	2.80
57	8.7	-25.6	3.8	35.1	135.0	2.85
8	-27.5	-35.0	-0.4	35.0	135.0	2.85
97	-34.7	0.3	-0.9	35.0	135.0	2.86
53	-26.0	7.9	-4.2	35.0	135.0	2.86
107	-33.8	0.5	-3.1	34.9	135.0	2.87

Note: See Figure 11.3.2-3 for section locations and definition of coordinate system.

11.3.2.[redacted] Fuel Tube Analysis

The BORAL neutron poison plates are supported by the fuel tubes within the canister basket. The fuel tube must preserve the geometry of the BORAL in the 6 inch drop and tip over accident events of the storage cask. [redacted] The fuel tube has been evaluated for an end drop of 56.1 g and a side drop of 55 g for the hypothetical accident events for transport. That analysis is presented in the Safety Analysis Report for the NAC-STC, docket 71-9235. That analysis shows that the BORAL neutron poison remains in place in the end and side drop conditions. The fuel tube configuration is shown in Figure 11.3.2.[redacted]-1.

Figure 11.3.2.[redacted]-1 Yankee Class Fuel Basket Tube Configuration



11.3.2[redacted] Fuel Basket Weldment Analysis for End Impact Conditions

The response of the top and the bottom weldment plates of the fuel basket assembly to a 56.1g accident impact load are examined. The top and bottom weldment are 0.5-inch thick plates constructed of SA240, Type 304 stainless steel. The weldments support their own weight plus the weight of 36 fuel assembly tubes.

A finite element analysis is performed for both plates, since the support for each weldment is different due to the location of the welded ribs for each. Eight structural ribs, eight tie-rod ends, and a circumferential ring support the top weldment and its loads during a top end drop. These structural components are modeled as zero-translation restraints in the direction of the end drop. The load from the fuel tube (2,198 pounds) are represented as point forces applied to the nodes at the periphery of the fuel assembly slots. An average point force is applied. The application of the nodal loads at the slot periphery is accurate since the tube weight is transmitted to the edge of the slot, which provides support to the fuel tubes in the end drop condition. Both models use the SHELL63 element which permits out of plane loading. The finite element models represent one-quarter sections of the weldments. Figures 11.3.2[redacted]-1 through 11.3.2[redacted]-4 show the finite element models for the weldments.

Figure 11.3.2.[redacted]-1 Top Weldment Finite Element Model with Structural Boundary Conditions

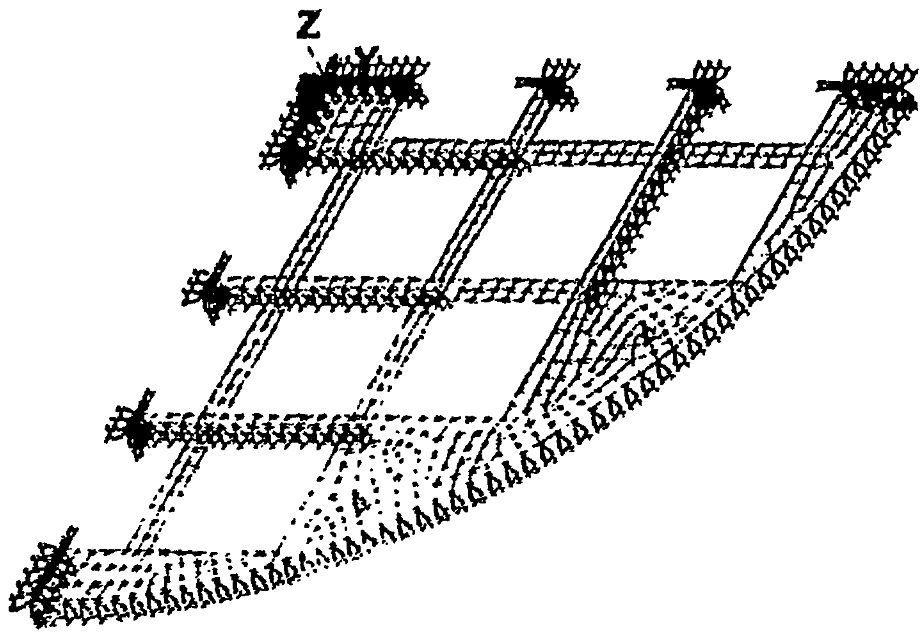
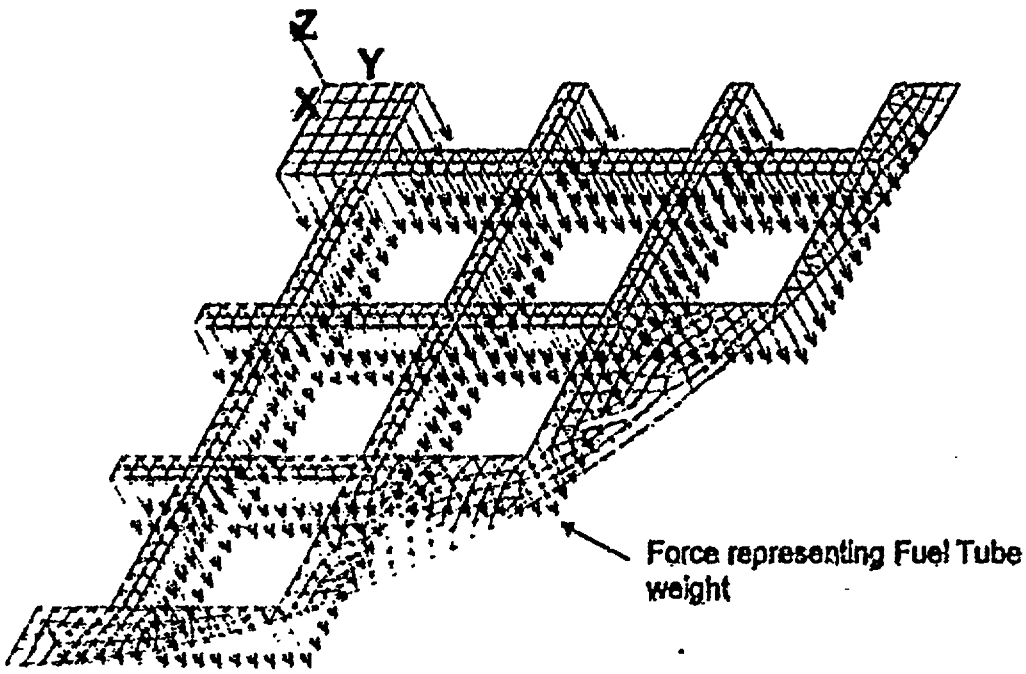


Figure 11.3.2.[redacted]-2 Top Weldment Finite Element Model with Structural Applied Loads[1]



1. Displacement conditions not shown.

Figure 11.3.2.█-3 Bottom Weldment Finite Element Model with Structural Boundary Conditions

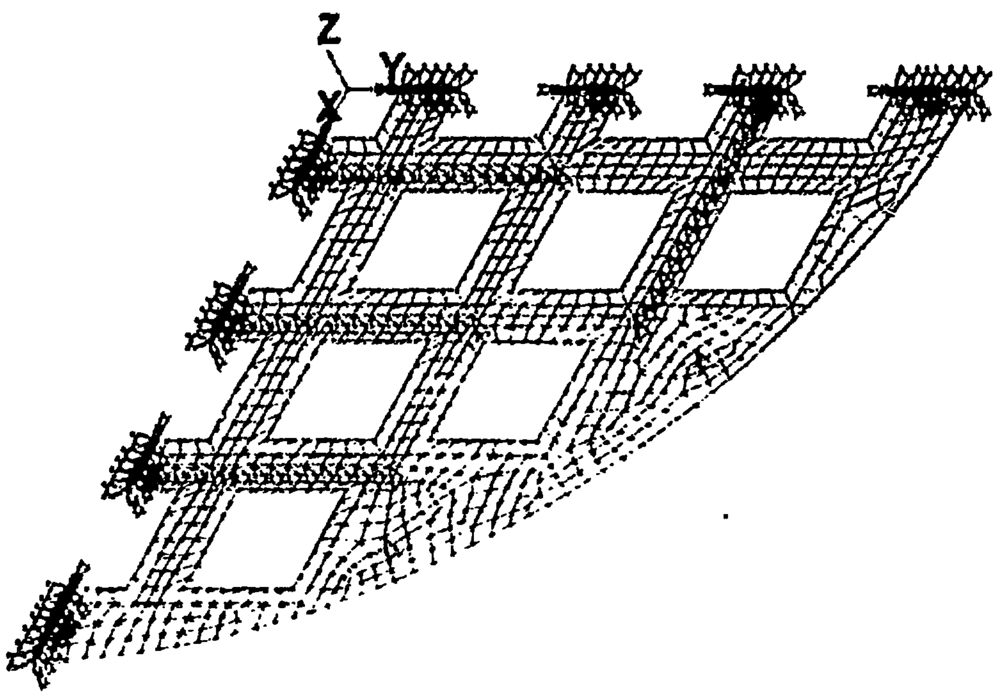
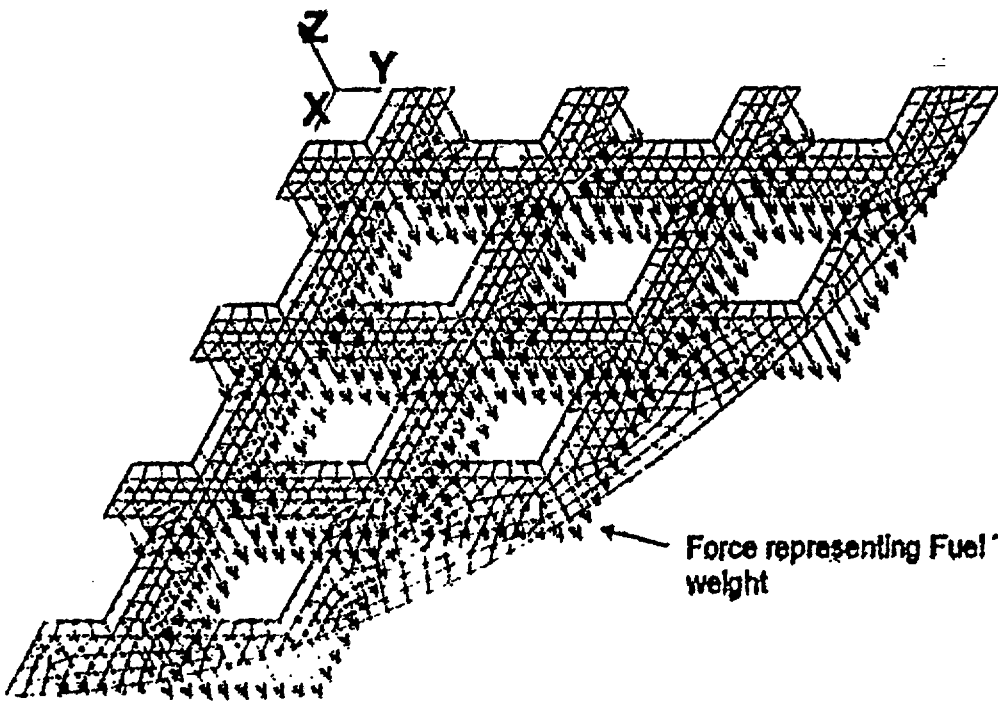


Figure 11.3.2.█-4 Bottom Weldment Finite Element Model with Structural Applied Conditions[1]



1. Displacement conditions not shown.

11.3.2.[illegible].1 Results of Fuel Basket Weldment Analyses

The analysis using the applied nodal forces demonstrates that the weldment design satisfies the primary membrane (P_m) and the primary membrane plus bending ($P_m + P_b$) stress criteria in ASME Code, Section III Division I, Subsection NG. Conservatively, nodal stresses, in lieu of sectional stresses, are used to compare with the stress allowable. For the end impact conditions, the P_m stresses are essentially zero since the weldments are subjected to bending load only. Hence, the following criteria for the $P_m + P_b$ stresses was used in the evaluation:

$P_m + P_b < 3.6S_m$ or S_u, whichever is less.

(Note: For Type 304 stainless steel in these temperature ranges, S_u is smaller than $3.6 S_m$.)

The margin of (MS) was calculated as:

$$M.S. = \frac{\text{Allowable Stress}}{\text{Nodal Stress Intensity}} - 1$$

The minimum margins of safety for each weldment for the end drop condition are shown in [illegible] table below. The allowable stress is determined based on a temperature of 530°F. This temperature is established by using the maximum temperature of the support disk for normal conditions of storage. The minimum Margins of Safety for the top/bottom weldments for the 56.1 g end impact are:

Component	$P_m + P_b$ (ksi)	Allowable (ksi)	M.S.
Top Weldment	58.0	62.2	+0.1
Bottom Weldment	48.1	62.2	+0.3

11.3.2.[redacted].2 Top Weldment Structural Rib Buckling Evaluation

The structural ribs on the top weldment are subjected to axial loads during a top end drop. End constraints on the ribs during a top end drop consist of fixed at the end welded to the top weldment and free at the other end. Because there are no closed solutions readily available for evaluating a plate for buckling loads with end constraints matching those of the top weldment ribs, a closed-form solution for the buckling of a column was used to analyze a 1-inch section of one of the ribs.

For a column under axial loading with one end fixed and the other end free, the critical load (P_{cr}) is determined by:

$$P_{cr} = \frac{\pi^2 EI}{(KL)^2}$$

where,

- I = moment of inertia,
- E = modulus of elasticity,
- L = length of the column, and
- K = effective length factor (K = 2 for a column with one end fixed and the other free).

Evaluating a 1-inch section of one of the ribs at the temperature of 540°F yields:

$$P_{cr} = \frac{\pi^2 \left(25.576 \times 10^6 \,\mathrm{lbf/in^2}\right) \frac{1}{12}(1.0\mathrm{in})(0.38\mathrm{in})^3}{(2 \times 6.80\mathrm{in})^2} = 6{,}241 \mathrm{lb}$$

For the 30-foot top end drop, the sum of the forces on the nodes representing the ribs was a maximum of 3,681 lb. Thus, the maximum load (P) on a 1-inch section of one of the structural ribs is:

$$P = \frac{3{,}681\text{ lb}}{27.5\text{ in}/2}(1\text{ in}) = 268\text{ lb}$$

Thus, the margin of safety (MS) for buckling of one of the structural ribs of the top weldment during a 30-foot top end drop is:

$$\text{M.S.} = \frac{6{,}241\text{ lb}}{268\text{ lb}} - 1 = 22.3$$

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[redacted] the reconfigured fuel [redacted] impact events.

11.4.1 Shell Casing Side Impact

The shell weldment is analyzed for bending [redacted] longitudinal direction as a simple span with [redacted] temperature of 750°F. Because the assembly is [redacted] maximum deflection at the weldment center, δ, is limited to 0.0[redacted] remaining energy will be transferred into the basket fuel tube assembly.

In accordance with ASME Section III, Subsection NF [redacted] subjected to axial compression or compression due to bending [redacted] effective if the width-thickness ratio, b/t, meets the following criterion:

$$\frac{b}{t} \le \frac{238}{\sqrt{S_y}}$$

Since, for the shell casing:

$$\frac{b}{t} = \frac{7.125-(2)(0.128)}{0.128} = 53.7 < \frac{238}{\sqrt{S_y}} = \frac{238}{\sqrt{17.3}} = 57.2$$

The shell casing meets the criteria and no reduction in allowable stress is applied.

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where:

$$M = \frac{0.555 \times (20+1)(0.53)}{4} = \ldots$$

For the Service [illegible] D condition, the bending [redacted]

$$f_b = \frac{M}{S} = 8.57 \text{ ksi}$$

where:

$$M = \frac{Pe}{4} = \frac{0.555 \times (57+1)(0.53)}{4} = 4.27 \text{ in-kip}$$

$P = \text{total weight} \times (\ \ + 1)$

$$e = \ldots \left(\frac{2.0+1.75}{2}\right) = 0.53 \text{ in.}$$

$$S = \frac{bd^2}{6} = \frac{\frac{3}{8} \times \left(2.12 - \frac{3}{16}\right)^2}{6} = 0.23 \text{ in}^3$$

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11.5 [redacted]

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[redacted] Margin of Safeties are:

Stress Type	Analysis Stress (ksi)	[redacted]	[redacted]
P_m	3.19	[redacted]	[redacted]
$P_m + P_b$	1.66	[redacted]	[redacted]
$P + Q$	6.99	[redacted]	[redacted]

This confirms that the canister closure weld is acceptable for normal operation conditions.

11.5.2 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.

[redacted]

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Table of Contents

12.0 OPERATING CONTROLS AND LIMITS .... [redacted]

12.1 Proposed Operating Controls and Limits .... [redacted]

12.2 [redacted] .... [redacted]

[redacted] [redacted] .... [redacted]

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List of Tables

Table [redacted] NAC-MPC System Controls and Limits .... [redacted]

12.0 OPERATING CONTROLS AND LIMITS

This chapter identifies operating controls and limits, technical parameters and surveillance requirements imposed to ensure the safe operation of the NAC-MPC System. [redacted]

[redacted] the user in meeting the requirements of [redacted] System. Section 4.8 of Appendix 12A presents [redacted] service.

12.1 Proposed Operating Controls and Limits

The NAC-MPC System is designed to provide passive dry storage of containerized Yankee Class spent fuel. The system has few operating controls. The principal controls and limits for the NAC-MPC System are satisfied by the selection of fuel for storage that meets the technical specifications presented in Section 2.1 and in Tables 12A2-1 and 12A2-2 of Appendix A. The general areas where controls and limits are necessary for safe operation of the NAC-MPC System are shown in Table 12-1. The conditions for use of the system that are defined in the table, are based on the specifications and functionality of the system and on the safety assessments for normal and accident conditions. Appendix 12B presents the bases for the Technical Specifications, which describe the development of the operating controls and limits.

12.2 [redacted]

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Training is considered in two venues. The first is [redacted] is dry run training. Classroom training considers [redacted] controls and limits and their bases. Dry run training is performed at the licensee's site and considers equipment fitup, interfacing and operations, including documentation of tests, inspections and test conditions.

Specific information included in any training topic may be site specific, but must consider and include, the approved site procedure(s) to be used in NAC-MPC System handling, loading, closing, and storage. Each training program should be developed in accordance with the licensee's general site training program requirements.

12.3 Special Requirements for the First System Placed in Service

The thermal performance of the first NAC-MPC System placed in service at a site shall be documented as described in Section 4.8 of Appendix 12A to this Chapter.

12.4 Surveillance After an Accident

The NAC-MPC storage cask shall be inspected within 24 hours after the occurrence of a tornado, earthquake (larger than 5.0 on the Richter scale), or other natural disaster in the area of the ISFSI. This inspection must specifically verify that the storage cask inlets and outlets are not blocked. At least one-half of the inlets and outlets must be cleared to restore air circulation within 24 hours. Provision must be made for removing the storage cask from service if the inlets and

[redacted]

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Table [redacted] NAC-MPC System Controls and Limits

Control or Limit	Applicable Technical Specification	Condition or Item Controlled
1. Fuel Characteristics	[redacted]	Type and Condition
	[redacted]	Dimensions and Weight
	[redacted]	Burnup and [redacted] Initial Enrichment
	[redacted]	Cool Time
[redacted]	[redacted]	[redacted]
[redacted] Canister	[redacted]	[redacted]
Fuel Loading	[redacted]	Weight and Number of Assemblies
Pressure Testing	[redacted]	Internal Pressure
Draining	[redacted]	Time to Drain
Drying	[redacted]	Vacuum Pressure
	3.1.2	[redacted]
Backfilling	[redacted]	Helium Pressure
Sealing	[redacted]	[redacted]
External Surface	[redacted]	Level of Contamination
3. Concrete Cask	[redacted]	Surface Dose Rates
	Note 1	Cask Spacing
	3.1.8	Cask Hand[redacted]
4. Surveillance	3.1.7	Air Inlets and Outlets
	3.1.7	Air Outlet Temperature
	Note 2	Annual Vertical Concrete Cask Concrete Inspection
5. Transfer Cask	3.1.9	Minimum [redacted]
6. ISFSI Concrete Pad	Note 3	Pad Concr[redacted]
	Note 3	Pad Subs[redacted]
	Note 3	Pad Concrete [redacted]

1. Limits are presented in Section 4.5.2 of Appendix 12A.
2. Limits are applied annually.
3. Limits are verified at the time of construction of the [redacted]

APPENDIX 12A

NAC-MPC SYSTEM
TECHNICAL SPECIFICATIONS

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1.0 USE AND APPLICATION

1.1 Definitions

---NOTE---

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

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CANISTER	See TRANSPORTABLE STORAGE CANISTER
CONCRETE CASK	See VERTICAL CONCRETE CASK
FUEL DEBRIS	FUEL DEBRIS is fuel with known or suspected defects, such as ruptured rods, severed rods, or loose fuel pellets.
INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)	The facility within the perimeter fence licensed for storage of spent fuel within NAC-MPC SYSTEMs (see also 10 CFR 72.3).
INTACT FUEL ASSEMBLY	INTACT FUEL ASSEMBLY is a fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks 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).

1.1 Definitions (continued)

LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on an NAC-MPC SYSTEM while it is being loaded with fuel assemblies. LOADING OPERATIONS begin when the first fuel assembly is placed in the CANISTER and end when the NAC-MPC SYSTEM is secured on the transporter.
RECONFIGURED FUEL ASSEMBLY (RFA)	A stainless steel canister 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 fuel rods or FUEL DEBRIS from any type of Yankee Class spent fuel assembly.
NAC-MPC SYSTEM	NAC-MPC SYSTEM is the container approved for storage of spent fuel assemblies at the ISFSI. The NAC-MPC SYSTEM consists of a CONCRETE CASK and a CANISTER.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI, while an NAC-MPC SYSTEM containing spent fuel is sitting on the storage pad within the ISFSI perimeter.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities involved in moving a loaded NAC-MPC SYSTEM to the ISFSI. TRANSPORT OPERATIONS begin when the NAC-MPC SYSTEM is first secured on the transporter and end when the NAC-MPC SYSTEM is at its destination and no longer secured on the transporter.

1.1 Definitions (continued)

TRANSPORTABLE STORAGE CANISTER (CANISTER)	TRANSPORTABLE STORAGE CANISTER is the sealed container that consists of a tube and disk fuel basket in a cylindrical canister shell that is welded to a baseplate, shield lid with welded port covers, and structural lid. The CANISTER provides the confinement boundary for the confined spent fuel.
TRANSFER CASK	TRANSFER CASK is a shielded lifting device that holds the CANISTER during LOADING and UNLOADING OPERATIONS and during closure welding, vacuum drying, leak testing, and non-destructive examination of the CANISTER closure welds. The TRANSFER CASK is also used to transfer the CANISTER into and from the CONCRETE CASK.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on an NAC-MPC SYSTEM to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the NAC-MPC SYSTEM is no longer secured on the transporter and end when the last fuel assembly is removed from the NAC-MPC SYSTEM.
VERTICAL CONCRETE CASK (CONCRETE CASK)	CONCRETE CASK is the cask that receives and holds the sealed CANISTER. It provides the gamma and neutron shielding and convective cooling of the spent fuel confined in the CANISTER.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

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

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

BACKGROUND

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

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

1.2 Logical Connectors (continued)

EXAMPLES The following examples illustrate the use of logical connectors.

EXAMPLES <u>EXAMPLE 1.2-1</u>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met	A.1 Verify. . . <u>AND</u> A.2 Restore. . .	

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

1.2 Logical Connectors (continued)

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. LCO not met	A.1	Stop...	
	OR		
	A.2.1	Verify...	
	AND		
	A.2.2		
	A.2.2.1	Reduce...	
		OR	
	A.2.2.2	Perform...	
	OR		
	A.3	Remove...	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector "OR" and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector "AND." Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector "OR" indicated that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operations (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the NAC-MPC SYSTEM. The ACTIONS associated with an LCO state conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition, unless otherwise specified, provided that the NAC-MPC SYSTEM is in a specified condition stated in the Applicability of the LCO. Prior to the expiration of the specified Completion Time, Required Actions must be completed. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the NAC-MPC SYSTEM is not within the LCO Applicability. Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1	12 hours
	AND	
	B.2 Perform Action B.2	36 hours

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

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within six hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One System not within limit.	A.1 Restore System to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1 AND B.2 Complete action B.2	12 hours 36 hours

When a System is determined not to meet the LCO, Condition A is entered. If the System is not restored within seven days, Condition B is also entered, and the Completion Time clocks for Required Actions B.1 and B.2 start. If the System is restored after Condition B is entered, Conditions A and B are exited; therefore, the Required Actions of condition B may be terminated.

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. LCO not met	A.1	Restore compliance with LCO	4 hours
B. Required Action and associated Completion Time not met.	B.1	Complete action B.1	6 hours
	AND		
	B.2	Complete action B.2	12 hours

The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times to be tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times are tracked for each component.

1.3 Completion Times (continued)

EXAMPLES	EXAMPLE 1.3-3 (continued)
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of requirements of the Frequency column of each SR.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

1.4 Frequency

EXAMPLES The following examples illustrate the various ways that Frequencies are specified.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, SR 3.0.2 allows an extension of the time interval to 1.25 times the interval specified in the Frequency for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment or variables are outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

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

1.4 Frequency (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits	Once within 12 hours prior to starting activity AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

2.0 FUNCTIONAL AND OPERATING LIMITS

2.1 Functional and Operating Limits

2.1.1 Fuel to be Stored in the NAC-MPC SYSTEM

INTACT FUEL ASSEMBLIES and FUEL DEBRIS meeting the limits specified in Table 12A2-1 may be stored in the NAC-MPC SYSTEM.

2.1.2 Preferential Fuel Loading

Preferential fuel loading shall be used whenever fuel assemblies with significantly different post-irradiation cooling times (zone year) are to be loaded in the same CANISTER. That is, fuel assemblies with the longest post-irradiation cooling times shall be loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post-irradiation cooling times shall be placed toward the center of the basket.

Preferential fuel loading may also be used whenever fuel assemblies having different cladding are to be loaded in the same CANISTER. Zircaloy clad fuel assemblies should be loaded in periphery positions of the basket, with stainless steel clad fuel assemblies loaded in the interior positions of the basket.

2.2 Functional and Operating Limit Violations

If any Functional and Operating Limits of Table 12A2-1 are violated, the following actions shall be completed:

2.2.1 The affected fuel assemblies shall be placed in a safe condition.

2.2.2 Within 24 hours, notify the NRC Operations Center.

2.2.3 Within 30 days, submit a special report that describes the cause of the violation and actions taken to restore compliance and prevent recurrence.

Table 12A2-1
Fuel Assembly Limits

I. NAC-MPC CANISTER

A. Allowable Contents

1. Uranium oxide Yankee Class INTACT FUEL ASSEMBLIES listed in Table 12A2-2 and meet the following specifications:

a. Cladding Type:	Zircaloy or Stainless Steel as specified in Table 12A2-2 for the applicable fuel assembly array/class
b. Enrichment:	As specified in Table 12A2-2 for the applicable fuel assembly array/class
c. Decay Heat Per Assembly:	
i. Zircaloy-Clad Fuel:	≤347 Watts
ii. Stainless Steel-Clad Fuel:	≤264 Watts
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Zircaloy-Clad Fuel:	As specified in Table 12A2-2 for the applicable fuel assembly array/class.
ii. Stainless Steel-Clad Fuel:	As specified in Table 12A2-2 for the applicable fuel assembly array/class.

Table 12A2-1
Fuel Assembly Limits (Continued)

f. Nominal Fuel Assembly Length:	≤ 111.8 inches
g. Nominal Fuel Assembly Width:	≤ 7.64 inches
h. Fuel Assembly Weight:	
i. Zircaloy-Clad Fuel:	≤ 850 lbs
ii. Stainless Steel-Clad Fuel:	≤ 900 lbs

2. Uranium oxide fuel rods or FUEL DEBRIS placed in RECONFIGURED FUEL ASSEMBLIES (RFA). The original fuel assemblies for the FUEL DEBRIS shall meet the criteria specified in Table 12A2-2 for the fuel assembly array/class, and meet the following additional specifications:

a. Cladding Type:	Zircaloy or Stainless Steel as specified in Table 12A2-2 for the applicable fuel assembly array/class
b. Enrichment:	As specified in Table 12A2-2 for the applic le fuel assembly array/class
c. Decay Heat Per RFA:	≤ 102 Watts
d. Post-Irradiation Cooling Time and Average Burnup Per Original Assembly:	
i Zircaloy-Clad Fuel:	As specified in Table 12A2-2 for the applicable fuel assembly array/class.

Table 12A2-1
Fuel Assembly Limits (Continued)

ii. Stainless Steel-Clad Fuel:	As specified in Table 12A2-2 for the applicable fuel assembly array/class.
e. Nominal Original Fuel Assembly Length:	≤ 111.8 inches
f. Nominal Original Fuel Assembly Width:	≤ 7.64 inches
g. Fuel Debris Weight:	≤ 850 lbs, including RFA
h. Maximum kg U per RFA:	≤ 146.23 lbs

B. Quantity per CANISTER:
Up to thirty-six (36) RFAs containing FUEL DEBRIS. Any remaining CANISTER fuel tube locations not filled with RFAs may be filled with the following:

a. Up to thirty-six (36) Zircaloy-clad INTACT FUEL assemblies;
b. Up to thirty-four (34) Stainless Steel-clad INTACT FUEL assemblies; as limited by the maximum fuel content weight limit of 30,600 pounds;
c. Mixed load of up to thirty-five (35) INTACT FUEL assemblies; as limited by the maximum fuel content weight limit of 30,600 pounds.

C. Fuel assemblies shall not contain control components.

D. Fuel assemblies shall not contain empty fuel rod positions. Any fuel rod that has been removed shall be replaced by a solid Zircaloy or stainless steel rod.

Table 12A2-2 INTACT FUEL ASSEMBLY Characteristics

	Combustion Engineering	Combustion Engineering	Exxon	Exxon	Exxon	Exxon	Westinghouse	Westinghouse	United Nuclear	United Nuclear
Assembly Configuration	Type A	Type B	Type A	Type B	Type A	Type B	Type A	Type B	Type A	Type B
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	3.50	3.50	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	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
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 load at a maximum burnup of 32,000 MWD/MTU, a minimum enrichment of 3.5 wt% and cool time of 8.0 years. The maximum decay heat for this assembly is 0.304 kW

Table 12A2-2 INTACT FUEL ASSEMBLY Characteristics (Continued)

	Combustion Engineering	Combustion Engineering	Exxon	Exxon	Exxon	Exxon	Westinghouse	Westinghouse	United Nuclear	United Nuclear
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)	33	33	33	33	33	33	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)	Variable	[illegible]	622.1	622.1	634.3	646.4	N/A	N/A	8	8
Number of Grid Spacers	8	8	8	8	6	8	N/A	N/A	6	6

2 United Nuclear assemblies fabricated from Zircaloy incorporate displacement rods 95.32 inches long and 0.361-inch in outside diameter
3 Five grid spacers are Zircaloy-4. The bottom spacer is Inconel 625
4 Estimated weight

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3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.
LCO 3.0.3	Not applicable to an NAC-MPC SYSTEM.
LCO 3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of an NAC-MPC SYSTEM.
LCO 3.0.5	Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the System to return to service under administrative control to perform the testing.
LCO 3.0.6	Not applicable to an NAC-MPC SYSTEM.
LCO 3.0.7	Not applicable to an NAC-MPC SYSTEM.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO, except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed from the time of discovery up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.3 (continued)	When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made, unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an NAC-MPC SYSTEM.

3.1 NAC-MPC SYSTEM Integrity

3.1.1 CANISTER Pressure Testing

LCO 3.1.1 The CANISTER shall be pneumatically pressure tested with an acceptance criteria of no observable weld defects or loss of pressure as specified in Table 12A3-1.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

---NOTE---

Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER pressure test requirements not met.	A.1 Establish CANISTER pressure test within acceptance criteria.	20 hours
B. Required Action and Associated Completion Time not met.	B.1 Remove all fuel assemblies from the NAC-MPC SYSTEM.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify CANISTER pressure test acceptance criteria are met.	Within 20 hours after removal of CANISTER from spent fuel pool.

3.1 NAC-MPC SYSTEM Integrity

3.1.2 CANISTER Vacuum Drying Pressure

LCO 3.1.2 The CANISTER vacuum drying pressure shall meet the limit specified in Table 12A3-1.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

---NOTE---

Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER vacuum drying pressure limit not met.	A.1 Establish CANISTER cavity vacuum drying pressure within limit.	10 hours
B. Required Action and Associated Completion Time not met.	B.1 Remove all fuel assemblies from the NAC-MPC SYSTEM.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify CANISTER cavity vacuum drying pressure is within limit	Within 10 hours after completion of CANISTER vacuum drying.

3.1 NAC-MPC SYSTEM Integrity

3.1.3 CANISTER Helium Backfill Pressure

LCO 3.1.3 The CANISTER helium backfill pressure shall meet the limit specified in Table 12A3-1.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

NOTE

Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER helium backfill pressure limit not met.	A.1 Establish CANISTER helium backfill pressure within limit.	10 hours
B. Required Action and Associated Completion Time not met.	B.1 Remove all fuel assemblies from the NAC-MPC SYSTEM.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify CANISTER helium backfill pressure is within limit	Within 10 hours after completion of CANISTER draining.

3.1 NAC-MPC SYSTEM Integrity

3.1.4 <u>CANISTER Helium Leak Rate</u>

LCO 3.1.4 There shall be no indication of a helium leak at a test sensitivity of 4 x 10^{-8} cm^3/sec (helium) through the CANISTER shield lid to CANISTER shell confinement weld to demonstrate a helium leak rate less than 8 x 10^{-8} cm^3/sec (helium) as specified in Table 12A3-1.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

---NOTE---

Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER helium leak rate limit not met.	A.1 Establish CANISTER helium leak rate within limit.	36 hours
B. Required Action and Associated Completion Time not met.	B.1 Remove all fuel assemblies from the NAC-MPC SYSTEM.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify CANISTER helium leak rate is within limit	Within 36 hours after completion of CANISTER vacuum drying.

3.1 NAC-MPC SYSTEM Integrity

3.1.5 <u>CANISTER Maximum Time in TRANSFER CASK</u>

LCO 3.1.5 The time duration from completion of backfilling the CANISTER with helium through completion of the CANISTER transfer operation from the TRANSFER CASK to the CONCRETE CASK shall not exceed 36 hours.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

----------------------------------NOTE----------------------------------
Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO time limit exceeded	A.1 Place TRANSFER CASK with helium filled loaded CANISTER in spent fuel pool for a minimum of 24 hours prior to restart of LOADING OPERATIONS sequence.	48 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Monitor elapsed time from start of helium backfill until completion of transfer of loaded CANISTER into CONCRETE CASK.	At completion of vacuum dryness verification test <u>AND</u> 3 hours thereafter.

3.1 NAC-MPC SYSTEM Integrity

3.1.6 Fuel Cooldown Requirements

LCO 3.1.6 A loaded CANISTER and its fuel contents shall be cooled down in accordance with the following specifications:

a. Nitrogen gas flush for a minimum of 10 minutes

b. Minimum cooling water temperature of 70 °F

c. Cooling water flow rate of 5 (+3, -0) gallons per minute at inlet pressure of 10 (+10, -0) psig

d. Maintain cooling water flow through CANISTER until outlet water temperature ≤ 200 °F

e. Maximum canister pressure ≤ 50 psig

APPLICABILITY: During UNLOADING OPERATIONS

-----NOTE-----

The LCO is only applicable to wet UNLOADING OPERATIONS.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER cooldown requirements not met.	A.1 Initiate actions to meet CANISTER cooldown requirements.	Immediately

3.1 NAC-MPC SYSTEM Integrity

3.1.6 Fuel Cooldown Requirements (Continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Initiate CANISTER cooldown flow to loaded CANISTER.	Within 30 hours after removal of CANISTER from CONCRETE CASK and placement in Transfer Cask.
SR 3.1.6.2	Verify that the cooldown water temperature and flow rate are within limits.	Once within 1 hour prior to initiating cooldown AND 1 hour thereafter.

3.1 NAC-MPC SYSTEM Integrity

3.1.7 CONCRETE CASK Thermal Performance

LCO 3.1.7 The outlet vent air temperature shall meet the limit specified in Table 12A3-1.

APPLICABILITY: During STORAGE OPERATIONS

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CONCRETE CASK Outlet Air Temperature Limit not met	A.1 Inspect CONCRETE CASK air inlet and outlet vents for blockage	12 hours
	AND	
	A.2 Verify calibration of temperature instrumentation.	24 hours
	AND	
	A.3 Administratively verify correct fuel loading.	12 hours
B. Required Actions and Associated Completion Times not met.	B.1 Remove all fuel assemblies from NAC-MPC SYSTEM.	30 days

3.1 NAC-MPC SYSTEM INTEGRITY

3.1.7 CONCRETE CASK Thermal Performance (Continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify outlet vent air temperature is within limit.	Once within 24 hours of start of STORAGE OPERATIONS AND 24 hours thereafter.

3.1 NAC-MPC SYSTEM INTEGRITY

3.1.8 <u>CONCRETE CASK Maximum Lifting Height</u>

LCO 3.1.8 A CONCRETE CASK containing a CANISTER loaded with INTACT FUEL ASSEMBLYs or RECONFIGURED FUEL ASSEMBLYs shall be lifted in accordance with the following requirements

a. A lift height $\leq$ 6 inches when oriented vertically.

APPLICABILITY: During TRANSPORT OPERATIONS

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NAC-MPC SYSTEM lifting requirements not met.	A.1 Initiate actions to meet CONCRETE CASK maximum lifting height.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify CONCRETE CASK lifting requirements are met.	After the CONCRETE CASK is raised to install or remove air pad and prior to the transporter beginning to move the NAC-MPC SYSTEM to/from the ISFSI.

3.1 NAC-MPC SYSTEM Integrity

3.1.9 <u>TRANSFER CASK Minimum Operating Temperature</u>

LCO 3.1.9	The TRANSFER CASK shall not be used for loaded CANISTER transfer operations outside of the fuel handling facility when the external ambient temperature is ≤ 0°F.
APPLICABILITY:	During LOADING or UNLOADING OPERATIONS

ACTIONS

------------------------------NOTE------------------------------
Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. External ambient temperature below LCO limit	A.1 Do not perform TRANSFER CASK operations external to the facility.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.9.1	Measure external ambient temperature.	Prior to start of LOADING or UNLOADING OPERATIONS <u>AND</u> 1 hour thereafter.

Table 12A3-1
CANISTER Limits

CANISTER	LIMITS
NAC-MPC CANISTER	
a. CANISTER Vacuum Drying Pressure	≤ 3 mm of Mercury for ≥ 30 min
b. CANISTER Helium Leak Rate	≤ 8×10^{-3} std cc/sec (helium)
c. CANISTER Helium Backfill Pressure	0 (+1, -0) psig
d. CANISTER Pressure Test	15.0 (+2,-0) psig for ≥ 10 min
e. Outlet Vent Air Temperature	≤ 92 °F above ambient temperature (°F)

3.2 NAC-MPC SYSTEM Radiation Protection

3.2.1 NAC-MPC SYSTEM Average Surface Dose Rates

LCO 3.2.1 CONCRETE CASK dose rates shall be measured at the locations shown in Figure 12A3-1. The average surface dose rates of each CONCRETE CASK shall not exceed:

a. 50 mrem/hour (neutron + gamma) on the side (on the concrete surfaces)

b. 35 mrem/hour (neutron + gamma) on the top;

c. 100 mrem/hour (neutron + gamma) at air inlet and outlet vents.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

NOTE

Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CONCRETE CASK average surface dose rate limits not met.	A.1 Administratively verify correct fuel loading. AND	24 hours

3.2 NAC-MPC SYSTEM Radiation Protection

3.2.1 CONCRETE CASK Average Surface Dose Rates (Continued)

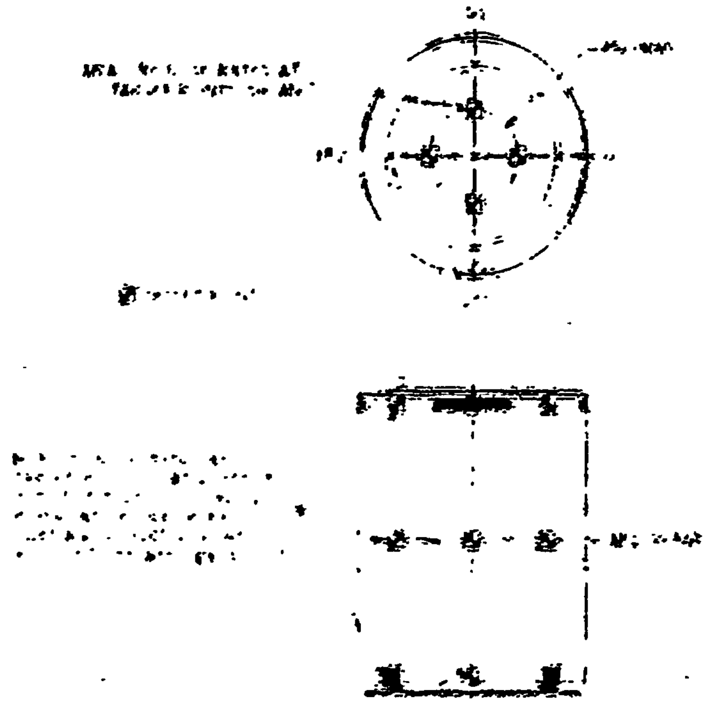
	A.2	Perform analysis to verify compliance with the ISFSI offsite radiation protection requirements of 10CFR20 and 10CFR72.	Prior to TRANSPORT OPERATIONS
B. Required Action and Associated Completion Time not met.	B.1	Remove all fuel assemblies from the NAC-MPC SYSTEM.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify average surface dose rates of CONCRETE CASK containing fuel assemblies are within limits.	Prior to TRANSPORT OPERATIONS.

NAC-MPC SYSTEM Average Surface Dose Rate
3.2.1

Figure 12A3-1
CONCRETE CASK Surface Dose Rate Measurement



3.2 NAC-MPC SYSTEM Radiation Protection

3.2.2 CANISTER Surface Contamination

LCO 3.2.2 Removable contamination on the accessible exterior surfaces of the CANISTER shall each not exceed:

a. 1000 dpm/100 cm^2 from beta and gamma sources and

b. 20 dpm/100 cm^2 from alpha sources.

APPLICABILITY. During LOADING OPERATIONS

ACTIONS

---NOTE---

Separate Condition entry is allowed for each NAC-MPC SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CANISTER removable surface contamination limits not met	A.1 Restore CANISTER removable surface contamination to within limits.	Prior to TRANSPORT OPERATIONS

3.2 NAC-MPC SYSTEM Radiation Protection

3.2.2 CANISTER Surface Contamination (Continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify that the removable contamination on the accessible exterior surfaces of the CANISTER containing fuel is within limits.	Prior to TRANSPORT OPERATIONS
SR 3.2.2.2	Verify that the removable contamination on the accessible interior surfaces of the TRANSFER CASK do not exceed limits.	Prior to loading empty CANISTER for performance of next LOADING OPERATIONS sequence

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site Location

Not applicable

4.2 Storage Features

4.2.1 Storage Cask

The NAC-MPC SYSTEM consists of the VERTICAL CONCRETE CASK (CONCRETE CASK) and its integral TRANSPORTABLE STORAGE CANISTER (CANISTER).

4.2.2 Storage Capacity

The total storage capacity of the ISFSI is limited by plant-specific license conditions.

4.2.3 Storage Pad(s)

Not applicable

4.3 Codes and Standards

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Add[illegible] through 1997, is the governing Code for the NAC-MPC Storage System.

4.3.1 Exceptions to Codes, Standards, and Criteria

Table 12A4-1 lists all approved exceptions.

Table 12A4-1

List of ASME Code Exceptions for the NAC-MPC SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
CANISTER	NB-1100	Statement of requirements for Code stamping of components.	CANISTER is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
CANISTER Shield Lid and Structural Lid Welds	NB-4243	Full penetration welds required for Category C joints (flat head to main shell per NB-3352.3)	Shield lid and structural lid to canister shell welds are not full penetration welds. These field welds are performed independently to provide a redundant closure. Leaktightness of the canister is verified by testing.
CANISTER Structural Lid Weld	NB-4421	Requires removal of backing ring.	Structural lid to canister shell weld uses a backing ring that is not removed. The backing ring permits completion of the groove weld. It is not considered in any analyses. It has no detrimental effect on the canister's function.
CANISTER Vent Port Cover and Drain Port Cover to Shield Lid Welds; Shield Lid to Canister Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root and final surface liquid penetrant examination to be performed per ASME Code Section V, Article 6, with acceptance in accordance with NB-5350.

Table 12A4-1

List of ASME Code Exceptions for the NAC-MPC SYSTEM (Continued)

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
CANISTER Structural Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	The CANISTER structural lid to canister shell closure weld is performed in the field following fuel assembly loading. The structural lid-to-shell weld will be verified by either ultrasonic (UT) or progressive liquid penetrant (PT) examination. If progressive PT examination is used, at a minimum, it will include the root and final surfaces and sufficient intermediate layers to detect critical flaws. If UT examination is used, it will be followed by a final surface PT examination. For either UT or PT examination, the maximum, undetectable flaw size is demonstrated to be smaller than the critical flaw size. The critical flaw size is determined in accordance with ASME Section XI methods. The examination of the weld will be performed by qualified personnel per ASME Code Section V, Articles 5 (UT) and 6 (PT) with acceptance per ASME Code Section III, NB-5330 (UT) and NB-5350 for (PT).

Table 12A4-1

List of ASME Code Exceptions for the NAC-MPC SYSTEM (Continued)

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
CANISTER Vessel and Shield Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The CANISTER shield lid to shell weld is performed in the field following fuel assembly loading. The CANISTER is then pneumatically (air-over-water) pressure tested as defined in Chapter 9 and described in Chapter 8. Accessibility for leakage inspections precludes a Code compliant hydrostatic test. The shield lid-to-shell weld is re-examined by liquid penetrant (PT) examination following the pneumatic test. The vent port and drain port cover welds are examined by root and final PT examination. The structural lid secondary enclosure weld is examined by progressive PT or UT and final surface PT.
CANISTER Vessel	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. The function of the CANISTER is to confine radioactive contents under normal, off-normal, and accident conditions of storage. The CANISTER vessel is designed to withstand a maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.

Table 12A4-1

List of ASME Code Exceptions for the NAC-MPC SYSTEM (Continued)

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
CANISTER Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The NAC-MPC Storage System is marked and identified in accordance with 10 CFR 72 requirements. Code stamping is not required. The QA data package will be in accordance with NAC's approved QA program.
CANISTER Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The NAC-MPC Storage System will be marked and identified in accordance with 10 CFR 72 requirements. No Code stamping is required. The CANISTER basket data package will be in conformance with NAC's approved QA program.
CANISTER Vessel and Basket Assembly Material	NB-2130/NG-2130	States requirements for certification of material to NCA-3861 and NCA-3862	The NAC-MPC CANISTER Vessel and Basket Assembly component materials are procured in accordance with the specifications for materials in ASME Code Section II. The component materials will be obtained from NAC approved Suppliers in accordance with NAC's approved QA program.

4.4 Site Specific Parameters and Analyses

Site-specific parameters and analyses that will need verification by the NAC-MPC SYSTEM user, are as a minimum, as follows:

1. The temperature of 75°F is the maximum average yearly temperature. The average daily ambient temperature shall be 100°F or less.

2. The temperature extremes of 125°F with incident solar radiation and -40°F for storage of the CANISTER inside the CONCRETE CASK.

3. The horizontal and vertical seismic acceleration levels are bounded by the values shown below.

Design-Basis Earthquake Input on the Top Surface of an ISFSI Pad

Horizontal g-level in each of Two Orthogonal Directions	Corresponding Vertical g-level (upward)
0.25g	0.25 x 0.667 = 0.167g

4. The analyzed flood condition of 15 fps water velocity and a height of 50 feet of water (full submergence of the loaded cask) are not exceeded.

5. The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the fuel tank of the cask handling equipment used to move the loaded CONCRETE CASK onto the ISFSI site contains no more than 50 gallons of fuel.

4.4 Site Specific Parameters and Analyses (continued)

6. In addition to the requirement of 10 CFR 72.212(b)(2)(ii), the ISFSI pad and foundation shall include the following characteristics as applicable to the drop and tip over analyses:

a.	Concrete thickness	36 inches maximum
b.	Pad Subsoil thickness (backfill)	72 inch minimum
c.	Concrete compressive strength	≤3,000 psi at 28 days (maximum)
d.	Concrete density (ρ)	$125 \le \rho \le 140$ lbs/ft^3
e.	Soil density (ρ)	$85 \le \rho \le 115$ lbs/ft^3
f.	Soil Stiffness	≤ 250 psi/in.

The concrete pad maximum thickness excludes the ISFSI pad footer. Steel reinforcement is used in the pad. The placement of the reinforcement, including its area and spacing, are determined by analysis and installed per ACI 318.

7. In cases where engineered features (i.e., berms, shield walls) are used to ensure that requirements of 10 CFR 72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assessment Category on a site specific basis.

4.5 Design Specifications

4.5.1 Specifications Important for Criticality Control

4.5.1.1 CANISTER-INTACT AND RECONFIGURED FUEL ASSEMBLIES

1. Minimum flux trap size: 0.75 inch
2. Minimum ^{10}B loading in the Boral neutron absorbers: 0.01 g/cm^2

4.5.2 Specification Important for Thermal Performance

1. The spacing of the NAC-MPC SYSTEM shall be 15 (+1, -0) feet (center-to-center).
2. Helium shall have a minimum purity of 99.9%.
3. Time limits for CANISTER LOADING OPERATIONS are established for the following activities:

a.	Draining of water from the CANISTER	20 hours
b.	Vacuum drying of the CANISTER	10 hours
c.	Transfer of CONCRETE CASK after helium backfill	36 hours

4.5.3 Specification Important to Shielding Performance

The CONCRETE CASK exterior surface shall be visually inspected annually for any damage (chipping, spalling, etc.). Any defects larger than one inch in diameter (or width) and deeper than one inch shall be regrouted, according to the grout manufacturer's recommendations.

4.5.4 Specification Important to CANISTER Lifting

The minimum distance from the master link of the CANISTER lifting slings to the top of the CANISTER shall be 67 inches.

4.6 NAC-MPC SYSTEM Training

Training modules shall be developed under the general licensee's training program as required by 10 CFR 72.212(b)(6). Training modules shall require a comprehensive, program for the operation and maintenance of the NAC-MPC SYSTEM and the Independent Spent Fuel Storage Installation (ISFSI). The training modules shall include the following elements, at a minimum:

- Regulatory Requirements Overview
- NAC-MPC SYSTEM Design and Operational Features
- ISFSI Facility Design (overview)
- Certificate of Compliance Conditions
- Technical Specifications, Controls, Limits and Conditions of Use
- Identification of Components and Equipment Important to Safety
- Surveillance Requirements
- NAC-MPC SYSTEM and ISFSI procedures, including:
  - Documentation, Inspection and Compliance Requirements
  - Handling the CONCRETE CASK and Empty CANISTER
  - Handling the Transfer Cask
  - Loading and Closing the CANISTER
  - Loading the CONCRETE CASK
  - Moving the CONCRETE CASK and CANISTER and Placement on the ISFSI
- Special Processes and Equipment, including Leak Testing, Welding and Weld Examination
- Auxiliary Equipment, including Lifting Yokes and Slings
- Off-Normal and Accident Conditions, Response and Corrective Actions
- Radiological Safety and ALARA
- Operating Experience

Training session participation should be documented as required to establish qualification to performed the designated tasks

4.7 Dry Run Training

A dry run training exercise of the loading, closure, handling, unloading, and transfer of the NAC-MPC Storage System shall be conducted by the licensee before the system is initially loaded. This demonstrates equipment fitup and interfacing, provides the opportunity to illustrate key features, operations, inspections and test conditions. It also allows comparison of procedural steps to component handling requirements. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to, the following:

- Moving the Concrete Cask into its Designated Loading Area
- Moving the Transfer Cask Holding the Empty Canister into the Spent Fuel Pool
- Loading One or More Dummy Fuel Assemblies into the Canister, Including Independent Verification
- Installing the Shield Lid
- Removal of the Transfer Cask from the Spent Fuel Pool
- Closing and Sealing of the Canister to Demonstrate Pressure Testing, Vacuum Drying, Helium Backfilling, Welding, Weld Inspection and Documentation, and Leak Testing
- Transfer Cask Movement Through the Designated Load Path
- Transfer Cask Installation on the Concrete Cask
- Placement of the Canister in the Concrete Cask
- Transport of the Concrete Cask to the ISFSI
- Canister Unloading, Including Reflooding and Weld Removal or Cutting

Demonstration of closing and sealing the canister may be performed using a mockup of the canister. The mockup should closely approximate the actual canister to allow qualification of personnel in the welding and testing tasks as required. The closed mockup is also used to demonstrate the activities necessary to open and unload the canister.

Participation in dry run training should be documented as required to establish qualification to perform designated tasks.

4.8 Special Requirements for First NAC-MPC SYSTEM Placed in Service

The heat transfer characteristics of the NAC-MPC SYSTEM will be recorded by temperature measurements of the first NAC-MPC SYSTEM placed in service with a heat load equal to or greater than 7.5 kW.

A letter report summarizing the results of the measurements shall be submitted to the NRC for each cask subsequently loaded with a higher heat load, up to the 12.5 kW maximum heat load for the NAC-MPC SYSTEM. The calculation and the measured temperature data shall be reported to the NRC in accordance with 10 CFR 72.4. The calculation and comparison need not be reported to the NRC for CANISTERs that are subsequently loaded with lesser loads than the latest reported case.

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APPENDIX 12B

NAC-MPC SYSTEM
TECHNICAL SPECIFICATIONS BASES

Appendix 12B
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B 1.0 Introduction

This Appendix presents the design or operational condition, or regulatory requirement, which establishes the bases for the Technical Specifications provided in Appendix 12A.

The section and paragraph numbering used in this Appendix corresponds to the numbering used in Appendix 12A for the Functional and Operating Limits (Section 2.0) and the Limiting Condition for Operations or Surveillance (Section 3.0). This allows direct comparison of the limit or condition described in Appendix 12A, with its corresponding bases in this Appendix.

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B 2.0 Functional And Operating Limits

B 2.1 Fuel to be Stored in the NAC-MPC SYSTEM

BASES

BACKGROUND

The NAC-MPC SYSTEM design requires specifications for the spent fuel to be stored, such as the type of spent fuel, minimum and maximum allowable enrichment prior to irradiation, maximum burnup, minimum acceptable post-irradiation cooling time prior to storage, maximum decay heat, and conditions of the spent fuel (i.e., intact fuel or failed fuel). Other important limitations are the dimensions and weight of the fuel assemblies.

Requirements for fuel to be loaded into the NAC-MPC SYSTEM are specified in Sections 2.1.1 and 2.1.2 of Appendix 12A.

Specific limitations for the NAC-MPC SYSTEM are specified in Table 12A2-1 as referred to in the Functional and Operating Limits, Section 2.1.1 of Appendix 12A. These limitations support the assumptions and inputs used in the thermal, structural, shielding, and criticality evaluations performed for the NAC-MPC SYSTEM.

Actions required to respond to violations of any Functional and Operating Limits are provided in Section 2.2.

APPLICABLE SAFETY ANALYSES

To ensure that the shield lid is not placed on a canister containing an unauthorized fuel assembly, facility procedures require verification of the loaded fuel assemblies to ensure that the correct fuel assemblies have been loaded in the canister.

FUNCTIONAL AND OPERATING LIMITS

2.1.1

Functional and Operating Limit 2.1.1 refers to Table 12A2-1 for the specific fuel assembly characteristic limits for Yankee Class fuel assemblies authorized for loading into the NAC-MPC SYSTEM. These fuel assembly characteristics include parameters such as cladding material, enrichment, decay heat generation, post-irradiation cooling time, burnup, and fuel assembly length, width, and weight. Table 12A2-2 is referenced from Table 12A2-1 and provides additional specific fuel characteristic limits for the fuel assemblies based on the fuel assembly array/class type.

The fuel assembly characteristic limits of Tables 12A2-1 and 12A2-2 must be met to ensure that the thermal, structural, shielding, and criticality analyses supporting the NAC-MPC SYSTEM Safety Analysis Report are bounding.

2.1.2

Functional and Operating Limit 2.1.2 requires preferential loading of fuel assemblies with significantly different post-irradiation cooling times. This is required to prevent a cooler assembly from heating up due to being surrounded by hotter fuel assemblies. For the purposes of complying with this Functional and Operating Limit, only fuel assemblies with post-irradiation cooling times differing by two years or greater need to be loaded preferentially. This is based on the fact that the heat-up phenomenon can only occur with significant differences in decay heat emission characteristics between adjacent fuel assemblies having different post-irradiation cooling times.

FUNCTIONAL AND OPERATING LIMITS VIOLATIONS

2.2.1

If any Functional and Operating Limits of 2.1.1 or 2.1.2 are violated, the limitations on fuel assemblies to be loaded are not met. Action must be taken to place the affected fuel assembly(s) in a safe condition. This safe condition may be established by returning the affected fuel assembly(s) to the spent fuel pool. However, it is acceptable for the affected fuel assemblies to temporarily remain in the NAC-MPC SYSTEM, in a wet or dry condition, if that is determined to be a safe condition.

2.2.2 and 2.2.3

Notification of the Functional and Operating Limit violation to the NRC is required within 24 hours. Written reporting of the violation must be accomplished within 30 days. This notification and written report are independent of any reports and notification that may be required by 10 CFR 72.216.

REFERENCES

1. SAR, Sections 2.1, 4.4; Chapters 5 and 6.

B 3.0 Limiting Condition for Operation (LCO) Applicability

BASES

LCOs LCO 3.0.1, 3.0.2, 3.0.4, and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1 LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the NAC-MPC SYSTEM is in the specified conditions of the Applicability statement of each Specification).

LCO 3.0.2 LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within the specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and,

b. Completion of the Required Actions is not required when an LCO is meet within the specified Completion Time, unless otherwise specified.

There are two basic Required Action types. The first Required Action type specifies a time limit, the Completion Time to restore a system or component or to restore variables to within specified limits, in which the LCO must be met. Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS. The second Required Action type specifies the remedial measures that permit continued activities that are not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

LCO 3.0.2 (continued)

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillance, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

LCO 3.0.3

This specification is not applicable to the NAC-MPC SYSTEM because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the facility in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. NAC-MPC SYSTEM conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in NAC-MPC SYSTEM activities being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the NAC-MPC SYSTEM. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

LCO 3.0.4 (continued)	The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability. The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of the NAC-MPC SYSTEM. Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.
LCO 3.0.5	LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with the ACTIONS. The sole purpose of the Specification is to provide an exception to LCO 3.0.2 (e.g. to not comply with the applicable Required Action[s]) to allow the performance of testing to demonstrate: a. The equipment being returned to service meets the LCO; or b. Other equipment meets the applicable LCOs. The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed testing. This Specification does not provide time to perform any other preventive or corrective maintenance.
LCO 3.0.6	Not Applicable
LCO 3.0.7	Not Applicable

B 3.0 Surveillance Requirement (SR) Applicability

BASES

Surveillance Requirements (SRs)	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillance is performed to verify that systems and components meet the LCO and variables are within specified limits. Failure to meet Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or,

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the NAC-MPC SYSTEM is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including those invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post maintenance testing is required. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance

SR 3.0.1 (continued)

testing may not be possible in the current specified conditions in the Applicability, due to the necessary NAC-MPC SYSTEM parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Suveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the affected equipment in an alternative manner.

SR 3.0.2 (continued)	The provisions of SR 3.0.2 are not intended to be used repeatedly, merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes: consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency, based not on time intervals, but upon specified NAC-MPC SYSTEM conditions, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by the Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility, which is not intended to be used as an operational convenience to extent Surveillance intervals.

SR 3.0.3 (continued)

If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe operation of NAC-MPC SYSTEM activities.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on equipment that has been determined to not meet the LCO.

SR 3.0.4 (continued)

When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s), since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in a SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not in this situation, LCO 3.0.4 will govern any restrictions that may be (or may not) apply to specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of the NAC-MPC SYSTEM.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances, when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO, prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met.

Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 3.1 NAC-MPC SYSTEM Integrity

B 3.1.1 CANISTER Pressure Testing

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies that meet the requirements of the Functional and Operating Limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the welds are inspected and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is backfilled with helium and leak tested. The CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination and dose measurements are completed prior to moving the TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred to the CONCRETE CASK, average CONCRETE CASK surface dose rate measurements are taken. The CONCRETE CASK is moved to the ISFSI

Pneumatic pressure testing of the shield lid-to-CANISTER shell weld provides assurance that the weld is structurally sound and will perform its primary function of the confinement of radioactive material in the CANISTER and maintenance of the inert helium atmosphere. Prior to draining, vacuum drying and backfilling the CANISTER with helium, a pressure test at 1.25 times the normal conditions design pressure is performed to meet the requirements of the ASME Code, Section III, Subsection NB (Ref. 2). Following completion of the pressure test, a final dye penetrant (PT) examination of the shield lid-to-shell weld is performed

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied upon are the fuel pellet matrix, the metallic

fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on maintaining the cladding temperatures below established long-term limits. The confinement of the radioactive materials in the CANISTER and the maintenance of an inert helium atmosphere in the cavity is provided by the CANISTER's confinement boundary, including the field welded shield lid. Performance of the pressure test confirms the structural integrity of the shield lid to shell weld. The seal integrity of the shield lid to shell weld is further confirmed by subsequent helium leakage testing.

LCO

Verifying by pressure test that the shield lid to shell weld is structurally adequate at a pressure exceeding MNOP ensures the integrity of the confinement boundary.

APPLICABILITY OPERATIONS

The pneumatic pressure test is performed during LOADING OPERATIONS before the CANISTER is transferred to the CONCRETE CASK and transported to the ISFSI. TRANSPORT OPERATIONS would not commence if the pressure test did not meet the acceptance criteria. Therefore, CANISTER pressure testing is not required during TRANSPORT OPERATIONS or STORAGE OPERATIONS.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions

<u>A.1</u>

If the pressure test criteria are not met, actions must be taken to meet the LCO. The Completion Time is sufficient to determine and correct most failures, which could cause the pressure test to fail

B.1

If the CANISTER pressure test cannot be performed in compliance with the acceptance criteria, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated a maximum of three times prior to performing B.1. The time frame for completing B.1 can not be extended by re-performing A.1. The Completion Time is reasonably based on the time required to reflood the CANISTER, remove the shield lid to shell weld, move the TRANSFER CASK and CANISTER into the spent fuel pool, remove the shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The primary design consideration of the CANISTER is that it is structurally adequate to ensure the confinement of radioactive material and maintenance of the inert helium cavity atmosphere during long-term storage of the spent nuclear fuel.

Prior to placing the CANISTER in storage, a pressure test must be successfully performed. The Surveillance must be performed within 20 hours after removal of the TRANSFER CASK and CANISTER from the spent fuel pool. This allows sufficient time to perform the Surveillance following shield lid welding operations. If the pressure test criteria can not be met, the cavity can be quickly refilled with water.

REFERENCES

1. SAR Section 7.1 and 8.1
2. ASME Code, Division 1, Section III, Subsection ND

B 3.1 NAC-MPC SYSTEM Integrity

B 3.1.2 CANISTER Vacuum Drying Pressure

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operating Limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid welds are inspected and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is backfilled with helium and leak tested. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, average CONCRETE CASK surface dose rate measurements are taken. The CONCRETE CASK is then moved to the ISFSI.

CANISTER cavity vacuum drying is utilized to remove residual moisture from the CANISTER cavity after the water is drained from the CANISTER. Any water not drained from the CANISTER cavity evaporates, due to the vacuum. This is aided by the temperature increase, due to the heat generation of the fuel.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of design basis spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on storage in an inert atmosphere. This is accomplished by removing water from the CANISTER and backfilling the cavity with helium. The thermal analysis assumes that the CANISTER cavity is dried and filled with helium.

LCO

A vacuum pressure, meeting the limit specified in Table 12A3-1, indicates that liquid water has evaporated and been removed from the CANISTER cavity. Removing water from the CANISTER cavity helps to ensure the long-term maintenance of fuel cladding integrity.

APPLICABILITY OPERATIONS

Cavity vacuum drying is performed during LOADING OPERATIONS before the TRANSFER CASK holding the CANISTER is moved to transfer the CANISTER to the CONCRETE CASK. Therefore, the vacuum requirements do not apply after the CANISTER is backfilled with helium and leak tested prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

<u>A.1</u>

If the CANISTER cavity vacuum drying pressure limit cannot be met, actions must be taken to meet the LCO. Failure to successfully complete cavity vacuum drying could have many causes, such as failure of the vacuum drying system, inadequate draining, ice clogging of the drain lines, or leaking CANISTER welds. The Completion Time is sufficient to determine and correct most failure mechanisms.

<u>B.1</u>

If the CANISTER fuel cavity cannot be successfully vacuum dried, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated a maximum of three times prior to performing B.1. The time frame for completing B.1 can not be extended by re-performing A.1. The Completion Time is reasonably based on the time

required to reflood the CANISTER, perform fuel cooldown operations, cut the shield lid weld, move the TRANSFER CASK into the spent fuel pool, and remove the CANISTER shield lid in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. Cavity dryness is demonstrated by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time. A low vacuum pressure is an indication that the cavity is dry. The surveillance must be performed within 10 hours after completion of CANISTER draining. This allows sufficient time to backfill the CANISTER cavity with helium, while minimizing the time the fuel is in the CANISTER without water or the assumed inert atmosphere in the cavity.

REFERENCES

1. SAR Sections 7.1 and 8.1.

B 3.1 NAC-MPC SYSTEM Integrity

B 3.1.3 CANISTER Helium Backfill Pressure

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operating Limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid welds are inspected and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is backfilled with helium and leak tested. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, average CONCRETE CASK surface dose rate measurements are taken. The CONCRETE CASK is then moved to the ISFSI.

Backfilling of the CANISTER cavity with helium promotes heat transfer from the spent fuel to the CANISTER structure and the inert atmosphere protects the fuel cladding. Providing a helium pressure equal to atmospheric pressure ensures that there will be no in-leakage of air over the life of the CANISTER, which might be harmful to the heat transfer features of the NAC-MPC SYSTEM and harmful to the fuel.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on the ability of the NAC-MPC SYSTEM to remove heat from the CANISTER and reject it to the

environment. This is accomplished by removing water from the CANISTER cavity and backfilling the cavity with an inert gas.

The thermal analyses of the CANISTER assume that the CANISTER cavity is dried and filled with dry helium.

LCO

Backfilling the CANISTER cavity with helium at a pressure equal to atmospheric pressure ensures that there is no air in-leakage into the CANISTER, which could decrease the heat transfer properties and result in increased cladding temperatures and damage to the fuel cladding over the storage period. The helium backfill pressure specified in Table 12A3-1 was selected based on a minimum helium purity of 99.9% to ensure that the CANISTER internal pressure and heat transfer from the CANISTER to the environment is maintained consistent with the design and analysis basis of the CANISTER.

APPLICABILITY

Helium backfill is performed during LOADING OPERATIONS, before the TRANSFER CASK and CANISTER are moved to the CONCRETE CASK for transfer of the CANISTER. Therefore, the backfill pressure requirements do not apply after the CANISTER is backfilled with helium and leak tested prior to TRANSPORT OPERATIONS and STORAGE OPERATIONS.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs, that do not meet the LCO are governed by subsequent condition entry and application of associated Required Actions.

A.1

If the backfill pressure cannot be obtained, actions must be taken to meet the LCO. The Completion Time is sufficient to determine and correct most failures, which would prevent backfilling of the CANISTER cavity with helium.

B.1

If the CANISTER cavity cannot be backfilled with helium to the specified pressure, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated a maximum of three times prior to performing B.1. The time frame for completing B.1 can not be extended by re-performing A.1. The Completion Time is reasonably based on the time required to re-flood the CANISTER, perform cooldown operations, cut the CANISTER shield lid weld, move the TRANSFER CASK and CANISTER into the spent fuel pool, remove the CANISTER shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The long-term integrity of the stored fuel is dependent on the storage in a dry, inert atmosphere and maintenance of adequate heat transfer mechanisms. Filling the CANISTER cavity with helium at a pressure within the range specified in Table 12A3-1 will ensure that there will be no air in-leakage, which could potentially damage the fuel. This pressure of helium gas is sufficient to maintain fuel cladding temperatures within acceptable levels.

Backfilling of the CANISTER cavity must be performed successfully on each CANISTER before placing it in storage. The Surveillance must be performed within 10 hours after draining the CANISTER. This allows sufficient time to backfill the annulus with helium, while minimizing the time the loaded CANISTER is in the TRANSFER CASK without the assumed inert atmosphere in the cavity.

REFERENCES

1. SAR Sections 4.4, 7.1 and 8.1.

B 3.1 NAC-MPC SYSTEM Integrity

B 3.1.4 CANISTER Helium Leak Rate

BASES

BACKGROUND

A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operating Limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid welds are inspected and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is backfilled with helium and leak tested. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, average CONCRETE CASK surface dose rate measurements are taken. The CONCRETE CASK is then moved to the ISFSI.

Backfilling the CANISTER cavity with helium promotes heat transfer from the fuel to the CANISTER shell. The inert atmosphere protects the fuel cladding. Prior to transferring the CANISTER to the CONCRETE CASK, the CANISTER helium leak rate is verified to meet leak tight requirements to ensure that the fuel is confined.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity (including fission product gases, fuel fines, volatiles, and crud) during the storage of spent fuel in the CANISTER is ensured by the multiple confinement boundaries and systems. The barriers relied on are: the fuel pellet matrix, the metallic fuel cladding tubes where the fuel pellets are contained, and the CANISTER where the fuel assemblies are stored. Long-term integrity of the fuel and cladding depends on maintaining an inert atmosphere, and maintaining the cladding temperatures below established long-term limits. This is accomplished by removing water from the CANISTER and backfilling the cavity with helium.

LCO

Verifying that the CANISTER cavity helium leak rate is below the leak tight limit specified in Table 12A3-1 ensures the CANISTER shield lid is sealed. Verifying the helium-leakage rate is below leak tight levels will also ensure that the assumptions in the accident analyses and radiological evaluations are maintained.

APPLICABILITY

The leak tight helium leak rate verification is performed during LOADING OPERATIONS before the TRANSFER CASK and integral CANISTER are moved for transfer operations to the CONCRETE CASK. TRANSPORT OPERATIONS would not commence if the CANISTER helium leak rate was not below the test sensitivity. Therefore, CANISTER leak rate testing is not required during TRANSPORT OPERATIONS or STORAGE OPERATIONS.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the helium leak rate limit is not met, actions must be taken to meet the LCO. The Completion Time is sufficient to determine and correct most failures, which could cause a helium leak rate in excess of the limit.

B.1

If the CANISTER leak rate cannot be brought within the limit, the fuel must be placed in a safe condition. Corrective actions may be taken after the fuel is placed in a safe condition to perform the A.1 action provided that the initial conditions for performing A.1 are met. A.1 may be repeated a maximum of three times prior to performing B.1. The time frame for completing B.1 can not be extended by re-performing A.1 The Completion Time is reasonably based on the time required to re-flood the CANISTER, perform fuel cooldown operations,

cut the CANISTER shield lid weld, move the TRANSFER CASK into the spent fuel pool, remove the CANISTER shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

The primary design consideration of the CANISTER is that it is leak tight to ensure that off-site dose limits are not exceeded and to ensure that the helium remains in the CANISTER during long-term storage. Long-term integrity of the stored fuel is dependent on storage in a dry, inert environment.

Verifying that the helium leak rate meets leak tight requirements must be performed successfully on each CANISTER prior to placing it in storage. The surveillance must be performed within 36 hours after completion of CANISTER vacuum drying. This allows sufficient time to backfill the CANISTER cavity with helium and perform the leak test, while minimizing the time the fuel is in the CANISTER without water or the assumed helium atmosphere in the cavity.

REFERENCES

1. SAR Sections 7.1 and 8.1.

B 3.1 NAC-MPC SYSTEM Integrity

B 3.1.5 CANISTER Maximum Time in the TRANSFER CASK

BASES

BACKGROUND	A TRANSFER CASK with an empty CANISTER is placed into the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Functional and Operating Limits. A shield lid is then placed on the CANISTER. The TRANSFER CASK and CANISTER are raised out of the spent fuel pool. The TRANSFER CASK and CANISTER are then moved into the cask decontamination area, where dose rates are measured and the CANISTER shield lid is welded to the CANISTER shell and the lid welds are inspected and pressure tested. The water is drained from the CANISTER, and CANISTER cavity vacuum drying is performed. The CANISTER cavity is backfilled with helium and leak tested. Additional dose rates are measured, and the CANISTER vent port and drain port covers and structural lid are installed and welded. Non-destructive examinations are performed on the welds. Contamination measurements are completed prior to moving TRANSFER CASK and CANISTER in position to transfer the CANISTER to the CONCRETE CASK. After the CANISTER is transferred, average CONCRETE CASK surface dose rate measurements are taken. The CONCRETE CASK is then moved to the ISFSI. Backfilling the CANISTER cavity with helium promotes heat transfer from the fuel and the inert atmosphere protects the fuel cladding. Limiting the total time the loaded CANISTER is in the TRANSFER CASK, prior to its placement in the CONCRETE CASK, ensures that the short-term temperature limits established in the Safety Analysis Report for the spent fuel cladding and CANISTER materials are not exceeded.
APPLICABLE SAFETY ANALYSIS	Limiting the total time a loaded CANISTER backfilled with helium is authorized in the TRANSFER CASK, prior to placement in the CONCRETE CASK, ensures that the short-term temperature limits for the spent fuel cladding and CANISTER materials are not exceeded. Upon placement of the loaded CANISTER in the CONCRETE CASK, the temperatures of the CANISTER and stored spent fuel will return to below the established long-term temperature limits due to the more

efficient passive heat transfer characteristics of the CONCRETE CASK. Ensuring temperatures are maintained below short-term limits for a limited time period and returning them to below long-term limits will prevent damage to the spent fuel cladding and the as-analyzed performance of the CANISTER materials.

LCO

Limiting the length of time that the loaded CANISTER backfilled with helium is allowed to remain in the TRANSFER CASK ensures that the spent fuel cladding and CANISTER material temperatures remain below the short-term temperature limits established in the SAR for the NAC-MPC SYSTEM. The time duration is a function of the design of the TRANSFER CASK and the NAC-MPC SYSTEM.

APPLICABILITY

The elapsed time restrictions on the loaded CANISTER apply during LOADING OPERATIONS from the completion point of the CANISTER vacuum dryness verification through completion of the transfer from the TRANSFER CASK to the CONCRETE CASK.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate condition entry is allowed for each NAC-MPC system. This is acceptable, since the Required Actions for each condition provide appropriate compensatory measures for each NAC-MPC SYSTEM not meeting the LCO. Subsequent NAC-MPC systems that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the LCO time limit is exceeded, the TRANSFER CASK containing the loaded CANISTER will be returned to the spent fuel pool to allow the cooler spent fuel pool water to reduce the TRANSFER CASK, CANISTER and stored spent fuel temperatures. After a minimum period of 24 hours and when problems, which caused a delay in transferring the loaded CANISTER to the CONCRETE CASK are resolved, if any, the LOADING OPERATIONS can be re-commenced.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

The elapsed time shall be monitored from completion of CANISTER dryness verification until transfer operations into the CONCRETE CASK are completed.

REFERENCES

1. SAR Sections 4.1 and 8.1.

B 3.1 NAC-MPC SYSTEM Integrity

B 3.1.6 Fuel Cooldown Requirement

BASES

BACKGROUND

In the event that a CANISTER must be unloaded, the CONCRETE CASK with its enclosed CANISTER is returned to the fuel building, the CANISTER is removed from the CONCRETE CASK using the TRANSFER CASK, and the TRANSFER CASK and CANISTER are placed in the cask preparation area to begin the process of fuel unloading. The structural lid and vent and drain port cover welds are removed. The CANISTER cavity gas is sampled to determine the level of radioactive gases in the cavity. A flow of nitrogen gas is established to flush radioactive gases from the cavity. A cooldown system is attached to the drain connection (inlet) and vent connection (outlet). A controlled water flow rate with a specified minimum water temperature is established to the drain connection with the steam and water being discharged from the vent to the spent fuel pool or radioactive water treatment system. Cooling water flow is maintained until the CANISTER is filled and the contents sufficiently cooled down to allow placement of the TRANSFER CASK and CANISTER in the spent fuel pool.

Following cooldown, the shield lid weld is removed and the TRANSFER CASK and CANISTER are placed in the fuel pool. The shield lid is removed and the fuel assemblies are removed and placed in storage rack locations. The TRANSFER CASK and CANISTER are removed from the spent fuel pool and decontaminated.

APPLICABLE SAFETY ANALYSIS

The use of a controlled cooldown process allows the reflooding of the CANISTER and cooling of the stored fuel assemblies in a manner which precludes the creation of excessive thermal stresses in the fuel cladding, which could result in cladding rupture and steam pressures in the cavity that could exceed the CANISTER's design pressure.

By controlling the water flow rate and minimum water temperature, the rate of fuel cooldown is controlled, thereby preventing fuel cladding failure and maintenance of a steam pressure within the CANISTER design pressure, thus ensuring CANISTER structural integrity.

LCO

Controlling the inlet water flow rate and temperature ensures that there is no excessive thermally induced stress in the fuel cladding leading to failure and the steam pressure will be maintained below analyzed design values. The exit water temperature is monitored to ensure that the CANISTER contents are sufficiently cooled down to allow return of the CANISTER to the spent fuel pool for fuel assembly unloading.

APPLICABILITY

The inlet water flow rate and temperature and water/steam outlet temperatures are controlled and measured during UNLOADING OPERATIONS after the CANISTER has been transferred to the TRANSFER CASK from the CONCRETE CASK. Therefore, the CANISTER fuel cooldown LCO does not apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS. A note has been added to the Applicability for LCO 3.1.6, which states that the APPLICABILITY is only applicable to wet UNLOADING OPERATIONS. This is acceptable, since the intent of the LCO is to avoid uncontrolled CANISTER pressurization due to steam creation during CANISTER reflooding. This is not a concern for dry UNLOADING OPERATIONS.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each NAC-MPC SYSTEM. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

<u>A.1</u>

If the inlet water flow rate and minimum temperature requirements are not met, actions must be taken to restore the parameters to within the limits. The Completion Time is defined as Immediately to ensure actions are taken to correct the LCO before fuel cladding damage or overpressurization of the CANISTER has occurred. No additional actions are appropriate, since this LCO applies during UNLOADING OPERATIONS, which cannot proceed until the LCO is met.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This SR ensures that the temperatures of the CANISTER, basket and fuel contents do not exceed short-term limits prior to initiation of cooldown operations.

SR 3.1.6.2

The long-term integrity of the fuel assembly is dependant on the material condition of the fuel assembly cladding. Minimizing cladding damage, due to excessive thermally induced stresses in the cooldown process, ensures continuing cladding integrity for future wet or dry fuel storage. By controlling the flow rate and entry water temperature, the creation of steam pressures exceeding design values is prevented.

REFERENCES

1. SAR, Sections 4.4 and 8.3, and Chapter 3.

B 3.1 NAC-MPC SYSTEM Integrity

B 3.1.7 CONCRETE CASK Thermal Performance

BASES

BACKGROUND

The NAC-MPC SYSTEM consists of a CONCRETE CASK containing a CANISTER loaded with spent fuel assemblies stored in a dry inert atmosphere. The CANISTER is designed to reject the decay heat from the stored spent fuel and to maintain cladding temperatures within long-term limits. The CONCRETE CASK is designed to provide cooling air flow to the external surfaces by a passive natural convection mode of heat transfer.

Verification of the temperature of the outlet air, when corrected for ambient temperature, provides confirmation that the function of the CONCRETE CASK is within design parameters and the CANISTER contains fuel assemblies authorized for storage in accordance with the Functional and Operating Limits in Section 2.0.

APPLICABLE SAFETY ANALYSIS

The limit on the outlet air temperature is based on the thermal analyses in the SAR (Ref. 1). The limit was selected to ensure that the fuel cladding and CANISTER structural components will not exceed normal storage condition long-term temperature limitations. Cladding and structural component temperatures exceeding long-term limits could result in fuel assembly cladding damage and a reduction in the structural integrity of the fuel basket and CANISTER.

LCO

The limit on outlet air temperature was selected to ensure that long-term storage temperature limits will not be exceeded.

APPLICABILITY

The outlet air temperature limit applies during STORAGE OPERATIONS. Due to the large thermal inertia of the CONCRETE CASK, thermal performance can only be determined once the CONCRETE CASK is placed on the ISFSI pad. Therefore, the LCO is not applicable to LOADING OPERATIONS and TRANSPORT OPERATIONS.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each NAC-MPC SYSTEM. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each NAC-MPC SYSTEM not meeting the LCO. Subsequent NAC-MPC SYSTEMs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the air outlet temperature is not within limit, it could be a result of the blockage or air flow restriction of one or more of the air inlet and outlets, thereby reducing cooling air flow to the CANISTER. Such blockage could result in an excessive heat-up of the remaining air flow, a reduction in the heat rejection capability of the CANISTER and basket, and potentially, increases in fuel cladding and structural component temperatures exceeding long-term allowables. The visual inspection and removal of any blockage in the air inlets and outlets will allow the air flow to be restored. The completion time is based on the time required to perform such inspections under appropriate radiological control procedures.

A.2

If the air outlet temperature is not within limit, it could be a result of faulty, damaged or out of calibration temperature measuring equipment. The verification of proper functioning, calibration and accuracy of the temperature instrumentation could result in air outlet temperatures meeting the limit. The Completion Time is based on the time required to verify instrument performance.

A.3

If the air outlet temperature exceeds the limit, it could be an indication that one or more fuel assemblies with higher decay heat than specified in the Functional and Operational Limits in Section 2.1 was inadvertently loaded into the CANISTER. An administrative verification of the CANISTER fuel loading, by means such as review of video recordings of fuel loadings and records of the loaded fuel

assembly serial numbers, can establish whether a misloaded fuel assembly is the cause of the out-of-limit condition. The Completion Time is based on the time to perform such verification.

B.1

If the air outlet temperature cannot be brought into compliance with the limit, the thermal performance of the NAC-MPC SYSTEM is not within the thermal analysis in the SAR. The fuel must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonably based on the time required to return the NAC-MPC SYSTEM to the fuel building, transfer the CANISTER into the TRANSFER CASK, remove the structural lid and port cover welds, cooldown the CANISTER and fuel assemblies, remove the shield lid weld and unload the fuel assemblies from the CANISTER in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

The NAC-MPC SYSTEM thermal performance is determined after the NAC-MPC SYSTEM is placed in STORAGE OPERATIONS at the ISFSI pad and achieves thermal equilibrium. The SURVEILLANCE will then be performed every 24 hours for the period the NAC-MPC SYSTEM is in STORAGE OPERATIONS.

REFERENCES

1. SAR Sections 4.4, 8.2.

B 3.1 NAC-MPC SYSTEM Integrity

B 3.1.8 CONCRETE CASK Maximum Lifting Height

BASES

BACKGROUND

A loaded CONCRETE CASK is transported between the loading facility and the ISFSI using a heavy haul trailer. The CONCRETE CASK is handled in the vertical orientation. The height to which the CONCRETE CASK is lifted is limited to ensure that its structural integrity, and that of the installed CANISTER, are not compromised should it be dropped.

APPLICABLE SAFETY ANALYSIS

The structural analyses of the CONCRETE CASK and CANISTER demonstrate that the drop of a CONCRETE CASK from the Technical Specification height limits to a surface having the structural characteristics described in Design Features Section 4.4.6, will not compromise the NAC-MPC SYSTEM integrity or result in physical damage to the contained fuel assemblies. The structural analyses evaluated a CONCRETE CASK tip-over event onto an ISFSI surface also having structural characteristics, as described in Design Features, Section 4.4.6.

LCO

Limiting the CONCRETE CASK lifting height during TRANSPORT OPERATIONS maintains the NAC-MPC SYSTEM within the design and analysis basis. The maximum lifting height is a function of the NAC-MPC SYSTEM design. The lift height requirement is specified in LCO 3.1.8 for the vertical orientation.

Site Specific Parameters and Analysis, Section 4.4, provides the characteristics of the drop surface assumed in the analyses. As required by 10 CFR 72.212(b)(3), each licensee must "...determine whether or not the reactor site parameters...are enveloped by the cask design bases..."; therefore, licensees must evaluate the site transport route to assure that it is bounded by Section 4.4 or that potential drop accidents are bounded by the CANISTER design basis 55 g deceleration (horizontal orientation) or 56.1 g deceleration (vertical orientation).

APPLICABILITY

CONCRETE CASK lifting height restrictions apply during TRANSPORT OPERATIONS, which include movement of the CONCRETE CASK while secured on the heavy haul trailer. CONCRETE CASK and TRANSFER CASK handling and drop events postulated to occur in the fuel loading facilities are addressed in the user's FSAR or PSDAR.

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate condition entry is allowed for each NAC-MPC SYSTEM. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each NAC-MPC SYSTEM not meeting the LCO. Subsequent NAC-MPC SYSTEMs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

<u>A.1</u>

If none of the CONCRETE CASK lifting height requirements are met, immediate action must be initiated and completed expeditiously to comply with the lifting height requirements, in order to preserve the NAC-MPC SYSTEM design and analysis basis.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The CONCRETE CASK lift height requirements of LCO 3.1.8 or the site-specific lift height requirements developed under Design Features Section 4.4.6 must be verified to be met after the CONCRETE CASK is secured to the transporter and prior to the transporter beginning to move the CONCRETE CASK to the ISFSI. This ensures potential drop accidents during TRANSPORT OPERATIONS are bounded by the drop analyses.

REFERENCES

1. SAR, Sections 8.1, 8.3, and 11.2.4.

B 3.1 NAC-MPC SYSTEM Integrity

B 3.1.9 TRANSFER CASK Minimum Operating Temperature

BASES

BACKGROUND

The TRANSFER CASK is a shielded handling device designed to lift and protect the CANISTER during fuel LOADING and UNLOADING OPERATIONS. It is used to perform the vertical transfer of the CANISTER to and from the CONCRETE CASK. This transfer operation can occur within the confines of the fuel loading facility or in the open environment adjacent to the facility.

The structural integrity of the TRANSFER CASK and its capability to handle and shield a loaded CANISTER is ensured by maintaining the TRANSFER CASK ferrous material temperatures significantly above the materials' nil ductility transition temperatures (NDTT), thereby precluding brittle fracture.

APPLICABLE SAFETY ANALYSIS

The structural analysis of the TRANSFER CASK is based on the ductile performance of the structural material. The TRANSFER CASK structural materials were selected for their low temperature fracture toughness. In accordance with NRC Reg Guide 7.11 (Ref. 1), the lowest service temperature of a ferrous material component should be established at a minimum of 40°F above the NDTT for the material. For the NAC-MPC transfer cask, the NDTT established in the SAR is -50°F. Therefore the minimum ambient temperature limit of 0°F is established. Conservatively, the decay heat from the contained spent fuel is not assumed to maintain the TRANSFER CASK material temperatures above ambient.

LCO

Limiting the TRANSFER CASK operations outside of covered or heated facilities when the external ambient temperature is below the minimum temperature limit maintains the NAC-MPC SYSTEM within the design and analysis basis of the SAR (Ref. 2). The minimum operating temperature selected is based on the properties of the materials of construction of the TRANSFER CASK.

APPLICABILITY

The minimum operating temperature limit applies for TRANSFER CASK operations external to the fuel facility during LOADING or UNLOADING OPERATIONS

ACTIONS

A note has been added to the ACTIONS, which states that, for this LCO, separate condition entry is allowed for each NAC-MPC SYSTEM. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each NAC-MPC SYSTEM not meeting the LCO. Subsequent NAC-MPC SYSTEMs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

<u>A.1</u>

For external TRANSFER CASK operations, if the external ambient temperature is at, or below, the minimum operating temperature limit, immediate action must be initiated to stop the LOADING or UNLOADING OPERATIONS sequence to ensure that the TRANSFER CASK is not operated outside of the fuel facility.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

The external ambient temperature shall be measured, prior to and during the LOADING or UNLOADING OPERATIONS, to ensure that the ambient temperature does not fall below the established TRANSFER CASK minimum operating temperature for operations external to the fuel building.

REFERENCES

1. NRC RG 7.11.

2. SAR, Sections 2.2, 3.4, 4.1, 8.1 and 8.3.

B 3.2 NAC-MPC SYSTEM Radiation Protection

B 3.2.1 CONCRETE CASK Average Surface Dose Rates

BASES

BACKGROUND	The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public (Ref. 1). Occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and within the limits of 10 CFR Part 20. Radiation doses to the public are limited for both normal and accident conditions in accordance with 10 CFR 72.
APPLICABLE SAFETY ANALYSIS	The CONCRETE CASK average surface dose rates are not an assumption in any accident analysis, but are used to ensure compliance with regulatory limits on occupational dose and dose to the public.
LCO	The limits on CONCRETE CASK average surface dose rates are based on the shielding analysis of the NAC-MPC SYSTEM (Ref. 2). The limits are selected to minimize radiation exposure to the public and maintain occupational dose ALARA to personnel working in the vicinity of the NAC-MPC SYSTEMs. The LCO specifies sufficient locations for taking dose rate measurements to ensure the dose rates measured are indicative of the effectiveness of the shielding material.
APPLICABILITY	The CONCRETE CASK average surface dose rates apply during LOADING OPERATIONS. These limits ensure that the CONCRETE CASK average surface dose rates during TRANSPORT OPERATIONS, STORAGE OPERATIONS, and UNLOADING OPERATIONS are bounded by the shielding safety analyses. Radiation doses during STORAGE OPERATIONS are monitored by the NAC-MPC SYSTEM user in accordance with the plant-specific radiation protection program required by 10 CFR 72.212(b)(6).
ACTIONS	A note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each loaded CONCRETE CASK. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each

CONCRETE CASK not meeting the LCO. Subsequent NAC-MPC SYSTEMs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the CONCRETE CASK average surface dose rates are not within limits, it could be an indication that a fuel assembly was inadvertently loaded into the CANISTER that did not meet the Functional and Operating Limits in Section 2.0. Administrative verification of the CANISTER fuel loading, by means such as review of video recordings and records of the loaded fuel assembly serial numbers, can establish whether a misloaded fuel assembly is the cause of the out-of-limit condition. The Completion time is based on the time required to perform such a verification.

A.2

If the CONCRETE CASK average surface dose rates are not within limits and it is determined that the CONCRETE CASK was loaded with the correct fuel assemblies, an analysis may be performed. This analysis will determine if the CONCRETE CASK, once located at the ISFSI, would result in the ISFSI offsite or occupational calculated doses exceeding regulatory limits in 10 CFR Part 20 or 10 CFR Part 72. If it is determined that the out of limit average surface dose rates do not result in an the regulatory limits being exceeded, TRANSPORT OPERATIONS may proceed.

B.1

If it is verified that the fuel was misloaded or that the ISFSI offsite radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 will not be met with the CONCRETE CASK average surface dose rates above the LCO limit, the fuel assemblies must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonably based on the time required to transfer the CANISTER to the TRANSFER CASK, remove the structural lid and vent and drain port cover welds, perform fuel cooldown operations, cut the shield lid weld, move the TRANSFER CASK and CANISTER into the spent fuel pool, remove the shield lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLALNCE REQUIREMENTSS

SR 3.2.1.1

This SR ensures that the CONCRETE CASK average surface dose rates are within the LCO limits prior to transporting the NAC-MPC SYSTEM to the ISFSI. The surface dose rates are measured approximately at the locations indicated on Figure 12A3-1, following standard industry practices for determining average surface dose rates for large containers.

REFERENCES

1. 10 CFR Parts 20 and 72.
2. SAR Sections 5.1 and 8.1

B 3.2 NAC-MPC SYSTEM Radiation Protection

B 3.2.2 CANISTER Surface Contamination

BASES

BACKGROUND

A TRANSFER CASK containing an empty CANISTER is immersed in the spent fuel pool in order to load the spent fuel assemblies. The external surfaces of the CANISTER are maintained clean by the application of clean water to the annulus of the TRANSFER CASK. However, there is potential for the surface of the CANISTER to become contaminated with the radioactive material in the spent fuel pool water. This contamination is removed prior to moving the CONCRETE CASK containing the CANISTER to the ISFSI in order to minimize the radioactive contamination to personnel or the environment. This allows the ISFSI to be entered without additional radiological controls to prevent the spread of contamination and reduces personnel dose, due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices.

APPLICABLE SAFETY ANALYSIS

The radiation protection measures implemented at the ISFSI are based on the assumption that the exterior surfaces of the CANISTER have been decontaminated. Failure to decontaminate the surfaces of the CANISTER could lead to higher-than-projected occupational dose and potential site contamination.

LCO

Removable surface contamination on the CANISTER exterior surfaces is limited to 1000 dpm/100 cm² from beta and gamma sources and 20 dpm/100 cm² from alpha sources. These limits are taken from the guidance in IE Circular 81-07 (Ref. 2) and are based on the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. Only loose contamination is controlled, as fixed contamination will not result from the CANISTER loading process. Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels, which would cause significant personnel skin dose.

LCO 3.2.2 requires removable contamination to be within the specified limits for the accessible exterior surfaces of the CANISTER. The location and number of CANISTER surface swipes used to determine compliance with this LCO are determined based on standard industry practice and the user's plant-specific contamination measurement program for objects of this size. Accessible portions of the CANISTER are the upper portion of the CANISTER external shell wall accessible after draining of the TRANSFER CASK annulus and the structural lid. The user shall determine a reasonable number and location of swipes for the accessible portion of the CANISTER. The objective is to determine a removable contamination value representative of the entire upper circumference of the CANISTER and the structural lid, while implementing sound ALARA practices.

Verification swipes and measurements of removable surface contamination levels on the inside surfaces of the TRANSFER CASK shall be performed following transfer of the CANISTER to the CONCRETE CASK. These measurements will provide indirect evidence that the inaccessible surfaces of the CANISTER do not have removable contamination levels exceeding the limit.

APPLICABILITY

Verification that the CANISTER accessible surface contamination is less than the LCO limit is performed during LOADING OPERATIONS. This occurs before TRANSPORT OPERATIONS and STORAGE OPERATIONS. Measurement of the CANISTER surface contamination is unnecessary during UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the subject CANISTER to the ISFSI.

ACTIONS

A Note has been added to the ACTIONS, which states that, for this LCO, separate Condition entry is allowed for each CANISTER. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory measures for each CANISTER not meeting the LCO. Subsequent CANISTERs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the removable surface contamination of the CANISTER that has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the CANISTER and bring the removable surface contamination within limits. The Completion Time of "Prior to TRANSPORT OPERATIONS" is appropriate, given that the time needed to complete the decontamination is indeterminate and surface contamination does not affect the safe storage of the spent fuel assemblies.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

This SR verifies that the removable surface contamination on the accessible surface of the CANISTER is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification prior to initiating TRANSPORT OPERATIONS in order to confirm that the CANISTER can be moved to the ISFSI without spreading loose contamination.

SR 3.2.2.2

This SR verifies that the removable surface contamination on the interior surfaces of the TRANSFER CASK is less than the limits, thereby providing indirect confirmation that the removable surface contamination on the inaccessible surfaces of the CANISTER are within the limits. It also confirms that the proper functioning of the annulus clean water fill system. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification prior to initiating the next LOADING OPERATIONS sequence.

REFERENCES

1. SAR Section 8.1
2. NRC IE Circular 81-07.

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