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NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION REPORT

Docket No. 72-1026
FuelSolutions^{TM} Storage System
Certificate of Compliance No. 1026

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PRELIMINARY SAFETY EVALUATION REPORT

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INTRODUCTION

By letter dated February 3, 1998, as supplemented, BNFL Fuel Solutions (BFS), formerly Westinghouse Electric Company, requested approval of the storage components of the FuelSolutions™ Spent Fuel Management System (SFMS), formerly Wesflex, under the provisions of 10 CFR Part 72, Subpart K. In support of this request, BFS submitted a Safety Analysis Report (SAR) for the FuelSolutions™ Storage System. By letter dated March 23, 2000, BFS submitted Revision 4 to the SAR[1], which supersedes all previous revisions to the SAR. The SAR, submitted by BFS, follows the format of Regulatory Guide 3.61. This SER uses the Section-level format of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems[2], with some differences implemented for clarity and consistency.

The Nuclear Regulatory Commission's (NRC) staff review of the SAR addresses the handling and dry storage of spent fuel in a single dry storage cask design, the FuelSolutions™ Storage System. The cask would be used at an Independent Spent Fuel Storage Installation (ISFSI) that would be licensed under 10 CFR Part 72[3] at a reactor site operating with a 10 CFR Part 50 license. The staff's assessment is based on whether the applicant meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel and of 10 CFR Part 20 for radiation protection. Decommissioning, to the extent that it is treated in the SAR, presumes that, as a bounding case, the FuelSolutions™ Storage System is unloaded and subsequently decontaminated before disposition or disposal.

The staff has reviewed Revision 4 to the SAR for the FuelSolutions™ Storage System. Based on the statements and representations contained in the SAR, and the conditions given in the Certificate of Compliance (CoC), the staff concludes that the FuelSolutions™ Storage System meets the requirements of 10 CFR 72.

While components of the FuelSolutions™ Storage System are designed to be used in conjunction with a transportation cask for a dual-purpose function, the use or certification of the FuelSolutions™ Storage System under 10 CFR Part 71 for the off-site transport of the spent fuel contents is not a subject of this SER. Certification for transportation of the spent fuel contents occurs upon the completion of a separate staff review for a 10 CFR Part 71 Certificate of Compliance for transportation.

References

1. FuelSolutions™ Storage System Safety Analysis Report, Rev. 4, BNFL Fuel Solutions Corporation, March 2000.

2. U.S. Nuclear Regulatory Commission, “Standard Review Plan for Dry Cask Storage Systems,” NUREG-1536, January 1997.

3. U.S. Code of Federal Regulations. “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,” Title 10, Part 72.

LIST OF ACRONYMS USED

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ALARA	As Low As Is Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AW/OS	Automated Welding/Opening System
AWS	American Welding Society
BFS	BNFL Fuel Solutions
BNFL	British Nuclear Fuels PLC
BRP	Big Rock Point plant
B&W	Babcock and Wilcox
BWR	Boiling Water Reactor
CE	Combustion Engineering
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
DBE	Design Basis Earthquake
DBT	Design Basis Tornado
DBW	Design Basis Wind
DLF	Dynamic Load Factor

DU	Depleted Uranium
EN	Electroless Nickel
GE	General Electric
INEEL	Idaho National Engineering and Environmental Laboratory
ISFSI	Independent Spent Fuel Storage Installation
ISG	Interim Staff Guidance
kW	Kilowatts
lbm	Pounds (Mass)
LCO	Limiting Condition for Operation
LEL	Lower Explosive Limit
LHGR	Linear Heat Generation Rate
LLNL	Lawrence Livermore National Laboratory
LST	Lowest Service Temperature
LTP	Long-Term Performance
MCNP	Monte Carlo Neutron Photon Code
MPC	Multi Purpose Canister
MTIHM	Metric Ton of Initial Heavy Metal
MWD/MTU	MegaWatt Days Per Metric Ton of Uranium
NDE	Nondestructive Examination
NDT	Nil Ductility Temperature
NFPA	National Fire Protection Association
NQA-1	Nuclear Quality Assurance - 1
NRC	Nuclear Regulatory Commission
PGA	Peak Ground Acceleration

PNNL	Pacific Northwest National Laboratory
ppm	Parts Per Million
psia	Pounds Per Square Inch Absolute
psig	Pounds Per Square Inch Gauge
PT	Liquid Penetrant Examination
PWR	Pressurized Water Reactor
QA	Quality Assurance
RC	Reinforced Concrete
RT	Radiographic Examination
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SFMS	Spent Fuel Management System
SNF	Spent Nuclear Fuel
SR	Surveillance Requirement
SRP	Standard Review Plan
SSCs	Structures, Systems and Components
TEDE	Total Effective Dose Equivalent
TPA	Thimble Plug Assembly
TS	Technical Specifications
USL	Upper Subcritical Limit
VDS	Vacuum Drying System
W	Westinghouse

1.0 GENERAL DESCRIPTION

The objective of the review of the general description of the FuelSolutions™ Storage System is to ensure that the applicant has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the cask system.

1.1 System Description and Operational Features

The FuelSolutions™ Storage System is a dry storage system that uses a stainless steel storage canister stored within the central cavity of a cylindrical concrete storage cask. The concrete storage cask provides radiation shielding and contains internal air flow paths that allow decay heat from the canister spent fuel contents to be removed by natural air circulation around the canister wall.

The principal components of the FuelSolutions™ Storage System are the W21 and W74 canisters, the W150 concrete storage cask, and the W100 transfer cask. The transfer cask is used to move the loaded canisters to and from the storage cask. Canisters are placed in the storage cask either by positioning the transfer cask on top of the storage cask and lowering the canister in, or by positioning the transfer and storage casks on their sides and transferring the canister horizontally. Each FuelSolutions™ Storage System component has been classified as important to safety, safety related, or not important to safety in Table 2.1-1 of the storage system SAR (WSNF-200).

The FuelSolutions™ Storage System is designed to store up to 21 Pressurized Water Reactor (PWR) assemblies in the W21 canister, or up to 64 Big Rock Point Boiling Water Reactor (BWR) assemblies in the W74 canister. The W21 canister is designed to accommodate nearly all domestic commercial spent nuclear fuel. PWR fuel assemblies are grouped into 14 classes based on the characteristics in Tables 2.1-3 and 2.1-4 of the Technical Specifications (TS) for the W21 canister. The majority of PWR assemblies can be stored with control components. The W74 canister is designed to accommodate the three assembly types used at Big Rock Point (BRP). The limiting characteristics of these three assembly types are given in Table 2.1-2 of the TS for the W74 canister.

1.1.1 W21 and W74 Canisters

The W21 and W74 storage canisters, evaluated in canister SARs WSNF-201 and WSNF-203, respectively, each have several designs consisting of different materials of construction and different dimensions, depending on the storage conditions and the fuel being stored. All structural components of the canister are constructed of high-strength carbon or stainless steel. Any carbon steel used in the canister is coated with electroless nickel for corrosion protection. Each canister has an outside diameter of about 66 inches and a length of either about 182 inches for short canisters or 192 inches for long canisters. A typical canister consists of a shell assembly, top and bottom inner closure plates, vent and drain port covers, internal basket assembly, top and bottom shield plugs, and top and bottom outer closure plates. The canister shell, top and bottom inner closure plates, and vent and drain port covers form the confinement boundary.

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

W21 Basket Assembly

The W21 PWR fuel basket assembly is a right circular cylinder configuration with 21 stainless steel guide tubes for PWR contents. The guide tubes are laterally supported by a series of 2.0-inch and 0.75-inch thick spacer plates, held in position by support rods that run through support rod sleeves placed between the spacer plates. The square guide tubes include neutron poison sheets (Boral) on all four sides for criticality control. The W21 canister has two classes of canister, W21M and W21T, differing in materials of construction used for the canister shell and basket assembly. Each class of canister has four different types. The W21T canister class consists of a long lead (LL), long steel (LS), short lead (SL), and short steel (SS) canister. The W21M canister has long, depleted uranium (LD); long steel (LS); short, depleted uranium (SD), and short steel (SS) designs. See Figure 1-1, W21 Canister.

W74 Basket Assembly

The W74 Big Rock Point BWR fuel basket assembly consists of two right circular cylindrical baskets, with a total of 56 guide tubes (the blocked five center holes and the eight support tube holes do not have guide tubes) and a capacity of up to 64 assemblies. The ten unfueled guide tube positions, five in the center of each half-basket assembly, are mechanically blocked to prevent fuel assemblies from being loaded in these positions. The guide tubes are supported by a series of 0.75-inch thick spacer plates, held in position by support rods that run through support rod sleeves placed between the spacer plates. The square guide tubes include neutron poison sheets (borated stainless steel) on either one side, or two sides opposite from each other, in an arrangement within the basket that assures that there is a poison sheet between all of the assemblies. The W74 canister has two classes of canister, W74M and W74T, differing in materials of construction used for the canister shell and basket assembly. Each canister class has only a long steel (LS) design. See Figure 1-2, W74 Canister.

1.1.2 W150 Storage Cask

The W150 is the storage overpack for both the W21 and W74 canisters. The long version of the cask is 230 inches high and the short version is 220 inches high. Both versions have an outside diameter of 138 inches. The inside diameter is nominally 73 inches, and the wall thickness is 32.5 inches, which includes reinforced concrete and a 2.0-in thick steel liner. The maximum W150 storage cask weight is 253,204 lbs. empty, and 334,092 lbs. loaded with the heaviest canister. The storage cask provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during long-term storage. The storage cask has an annular air passage to allow the natural circulation of air around the canister. The air outlet vent duct flow path is off-set to prevent direct radiation streaming through the vent when the canister is in place. The spent fuel decay heat is transferred from the fuel assemblies to the guide tubes, and then via conduction through the spacer plates and radiation to the canister wall. Heat flows by radiation and convection from the canister wall to the circulating air and is exhausted through the air outlets. The passive cooling system is designed to maintain acceptable reinforced concrete and peak cladding temperatures for the authorized fuel types during storage. See Figure 1-3, W150 Storage Cask.

1.1.3 W100 Transfer Cask

The transfer cask provides shielding during canister movements between the spent fuel pool and the storage cask. The cask is a multi-wall (steel/lead/steel/water/steel) design, with about an 85-inch outside diameter. The cask has an overall length of 209 inches, and a maximum empty weight, with the top cover on and neutron shield filled, of 112,107 lbs. Covers are bolted on each end of the cask to allow access to the cask cavity from either side. Both the top and bottom covers consist of a thick steel plate and solid neutron shielding encased in a steel shell. The bottom cover has O-rings to prevent spent fuel pool water from entering the cask during loading operations. The top cover includes a secondary central cover for ram access during horizontal loading and unloading operations, and does not have O-rings, since the top cover is used only outside of the spent fuel pool.

The transfer cask neutron shield cavity is filled with clean demineralized water either prior to placing the transfer cask in the spent fuel pool or upon its removal, depending on the site crane capacity. Two quick-connect fittings are used to drain and fill the neutron shield cavity, and to prevent intrusion of contaminated spent fuel pool water. To prevent spent fuel pool water from contaminating the annular region between the transfer cask and the fuel canister, an inflatable annulus seal is used during spent fuel loading. The seal is inserted at the top end of the annulus and inflated, after filling the annulus with clean demineralized water.

Heat removal from the transfer cask is primarily by conduction through the cask wall. A high emissivity, low absorptivity coating is applied to the exterior of the liquid neutron shield jacket to facilitate radiative heat transfer to the environment. A thermocouple probe is also included, at the exterior of the cask structural shell, to ensure that the transfer cask system temperatures are within limits during horizontal transfer. See Figure 1-4, Transfer Cask.

1.1.4 Auxiliary Equipment

Section 1.2.1.4 of the storage system SAR describes the following principal auxiliary equipment necessary to operate the FuelSolutions™ Storage System in accordance with its design:

- Annulus Seal - keeps spent fuel pool water out of the canister-transfer cask annulus.
- Shield Plug Retainers - keeps the canister and shield plug in place during removal from the spent fuel pool.
- Vacuum Drying System (VDS) - dries sealed spent fuel canisters.
- Inner Closure Plate Strongback - provides supplementary support for inner closure plate during draining and leak testing operations.
- Automated Welding/Opening System - welds/opens canister closure welds.
- Helium Leak Detector - detects and locates leaks in canister closure welds.
- Horizontal Transfer Trailer/Skid - moves transfer cask with loaded fuel canister from spent fuel pool building to the ISFSI.

- Hydraulic Ram System - pushes or pulls canister during horizontal transfer to or from a storage cask.
- Upender/Downender - rotates storage cask from vertical to horizontal position.
- Storage Cask Impact Limiter - energy absorbing pad assembly in event of storage cask drop accident.
- Horizontal Lid Handling Fixture - lifting device for handling storage and transfer cask lids during horizontal transfer.
- Vertical Transporter/Transport Trailer - vehicles capable of moving empty or loaded storage cask in vertical position.
- Air Pallet System - pallets inserted under cask to provide reduced friction surface to facilitate vertical transfer and placement on storage pad.
- Vertical Canister Lift Fixture - lifts transfer cask for loading and vertical transfer operations.
- Cask Lifting Yoke - used for vertical lifting, handling, and up-ending/down-ending inside fuel building.
- Cask Cavity Axial Spacer - used in transfer cask when short fuel canister is being transferred.
- Docking Collar - provides shielding during vertical and horizontal transfer.
- Cask Restraints - prevent cask separation during transfer.
- Empty Canister Lift Fixture - used for handling of empty canister.
- Standard Lifting Slings - support system operations for handling various items associated with the FuelSolutions™ Storage System.

1.1.5 FuelSolutions™ Storage System Cask Arrays

Section 1.4 of the storage system SAR depicts a typical storage pad layout for an ISFSI. Spacing limitations on cask arrays (15 feet minimum center to center) are discussed in Section 4.0 of the Technical Specifications (TS). Measurements of center to center spacing are required in Sections 8.1.10.7 and 8.1.11.4 of the storage system Operating Procedures. TS 5.3.5 controls the maximum allowable surface dose rates for any individual cask.

1.2 Drawings

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

1.3 Cask Contents

The approved contents for the FuelSolutions™ Storage System are: (1) up to 21 intact zircaloy-clad PWR spent fuel assemblies having the characteristics of one of the fuel assembly classes described in Tables 2.1-3 or 2.1-4 of the W21 canister TS, or (2) up to 64 intact Big Rock Point BWR assemblies described in Table 2.1-2 of the W74 canister TS. Intact fuel assemblies are defined as having no defects greater than hairline cracks or pinhole leaks. The enrichment and physical, thermal, and radiological characteristics of the approved contents are given in the CoC Appendix A, the TS fuel specifications.

PWR and BWR fuel assemblies are qualified for shipment based on the fuel cooling tables in Section 5 of each canister SAR. The tables provide a required minimum cooling time for fuel assemblies with various enrichments and burnup levels. PWR fuel assemblies with burnup levels greater than 60,000 MWD/MTU and BWR assemblies with burnup levels greater than 40,000 MWD/MTU are not allowed in the FuelSolutions™ Storage System.

Intact PWR assemblies can be stored with or without control components. Stainless steel spacers may be used in the W21 canister to radially or axially position intact PWR assemblies that are significantly smaller than the available cavity dimensions. Missing or damaged fuel rods must be replaced with dummy rods which displace an equal amount of water as the original rods to allow storage in the W21 canister.

Intact Big Rock Point BWR assemblies are stored in the W74 canister without flow channels and are accommodated without the need for radial or axial spacers. Missing or damaged fuel rods must be replaced with dummy rods which displace an equal amount of water as the original rods to allow storage in the W74 canister.

1.4 Qualification of Applicant

BFS is the prime contractor for design, fabrication, construction, assembly, testing, and operation of the FuelSolutions™ Storage System. Fabrication, construction, assembly, and operation may be performed by Westinghouse, a BNFL company, or another licensee as the prime contractor. The BFS Quality Assurance (QA) program, designed and administered to meet the criteria of 10 CFR Part 72, Subpart G, is applicable to all design, fabrication, construction, testing, operation, modification, and decommissioning activities that are important to safety. Section 1.3 of the storage system SAR adequately details The applicant's technical qualifications and previous experience in the area of dry cask storage licensing.

1.5 Quality Assurance

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

1.6 Evaluation Findings

F1.1 A general description and discussion of the FuelSolutions™ Storage System is presented in Section 1 of the storage system SAR and each canister SAR (Rev. 4), with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

F1.2 Drawings for SSCs important to safety are presented in Section 1 of the storage system SAR and each canister SAR. Specific SSCs are evaluated in Sections 3 through 14 of this SER.

F1.3 Specifications for the spent fuel to be stored in the FuelSolutions™ Storage System are provided in Section 2 of the storage system and canister SARs.

F1.4 The technical qualifications of the applicant to engage in the proposed activities are identified in Section 1.3 of the storage system SAR.

F1.5 The quality assurance program is described in Section 13 of the storage system SAR and addressed in Section 13 of this SER.

F1.6 The FuelSolutions™ Storage System was not reviewed in this SER for use as a transportation cask.

F1.7 The staff concludes that the information presented in this section of the storage system and canister SARs satisfies the requirements for the general description under 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation itself, Regulatory Guide 3.61, and accepted dry storage practices detailed in NUREG-1536.[2]

1.7 References

1. FuelSolutions™ Storage System Safety Analysis Report, Rev. 4, BNFL Fuel Solutions Corporation, March 2000.

2. U.S. Nuclear Regulatory Commission, “Standard Review Plan for Dry Cask Storage Systems,” NUREG-1536, January 1997.

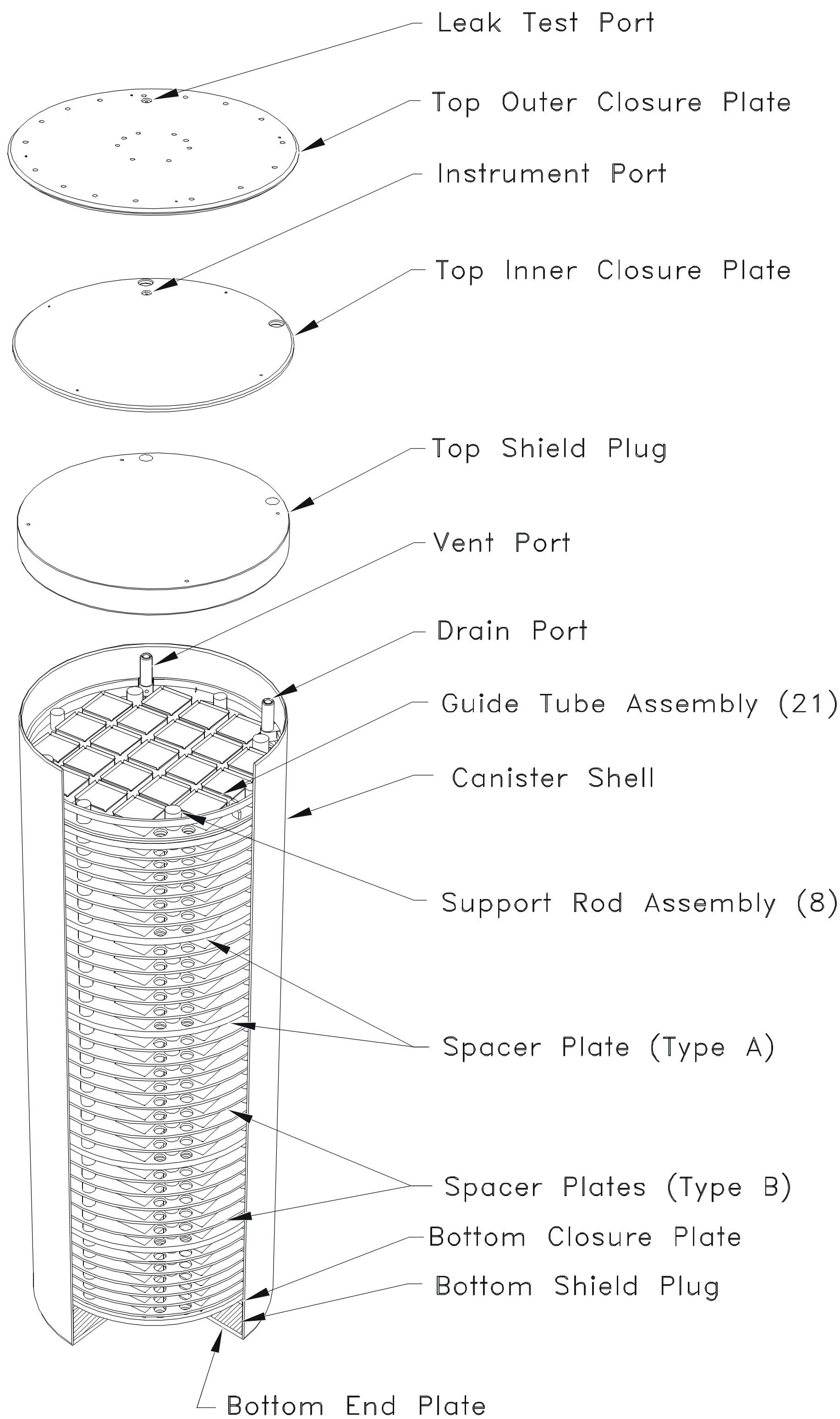

Figure 1-1 W21 Canister

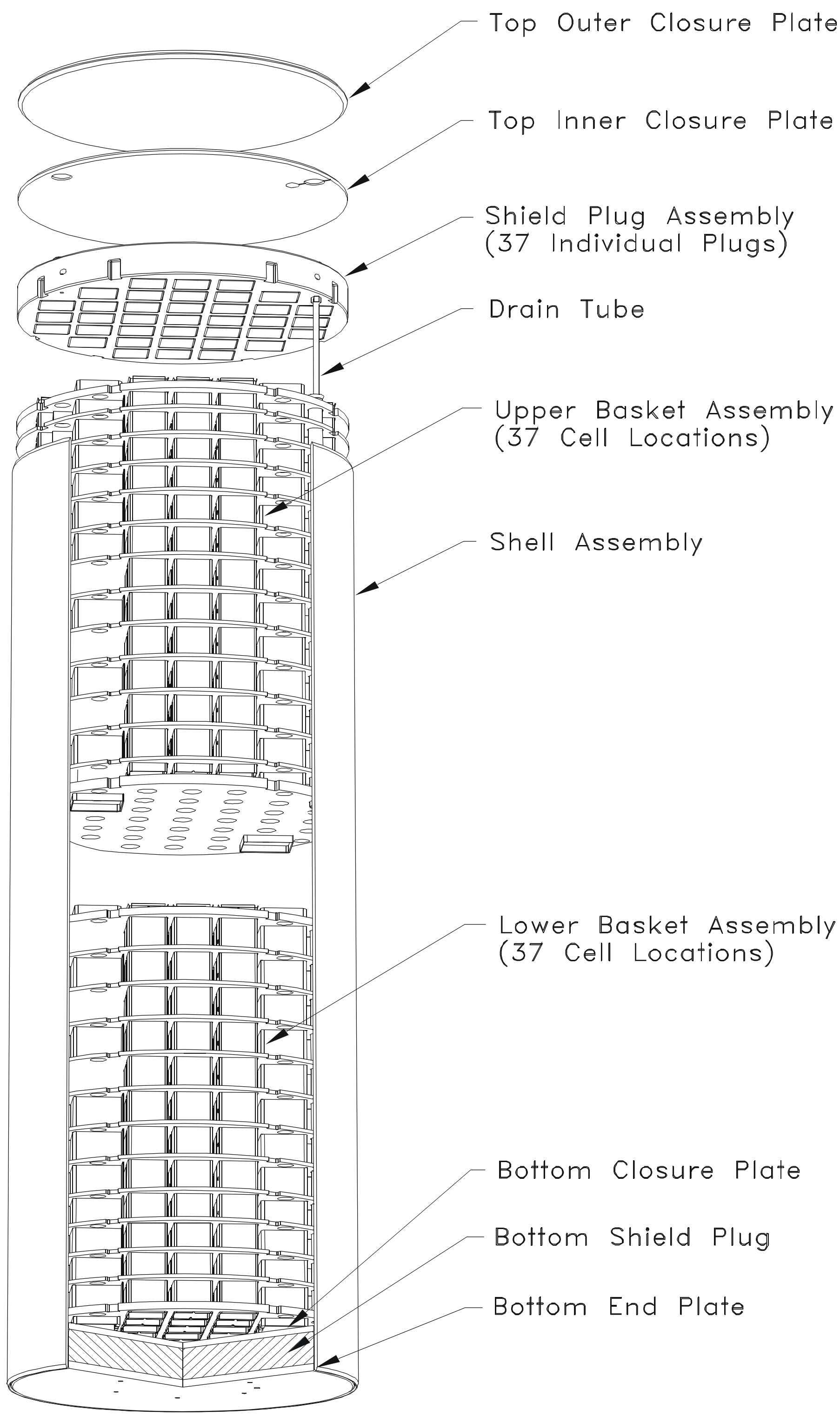

Figure 1-2 W74 Canister

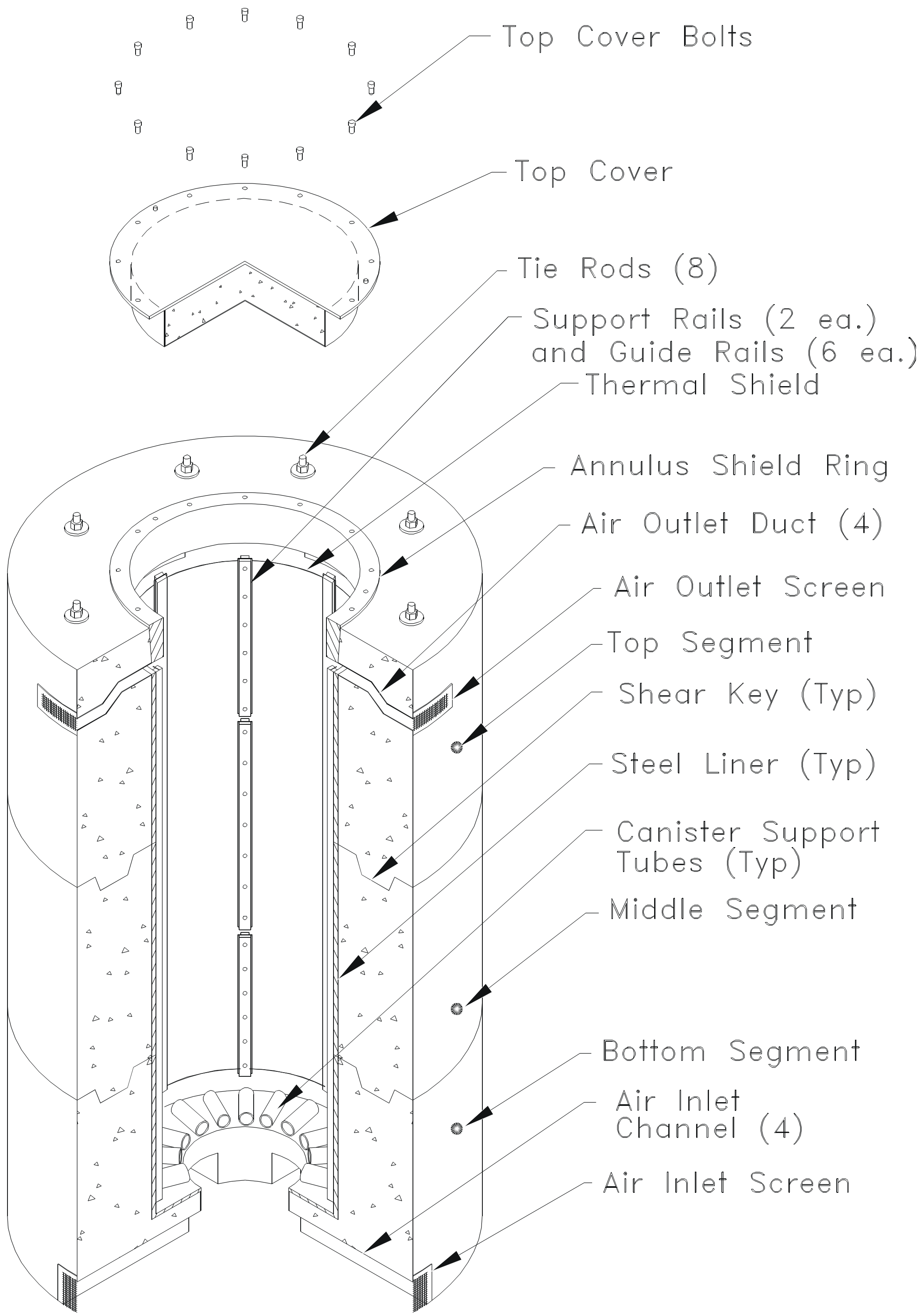

Figure 1-3 W150 Storage Cask

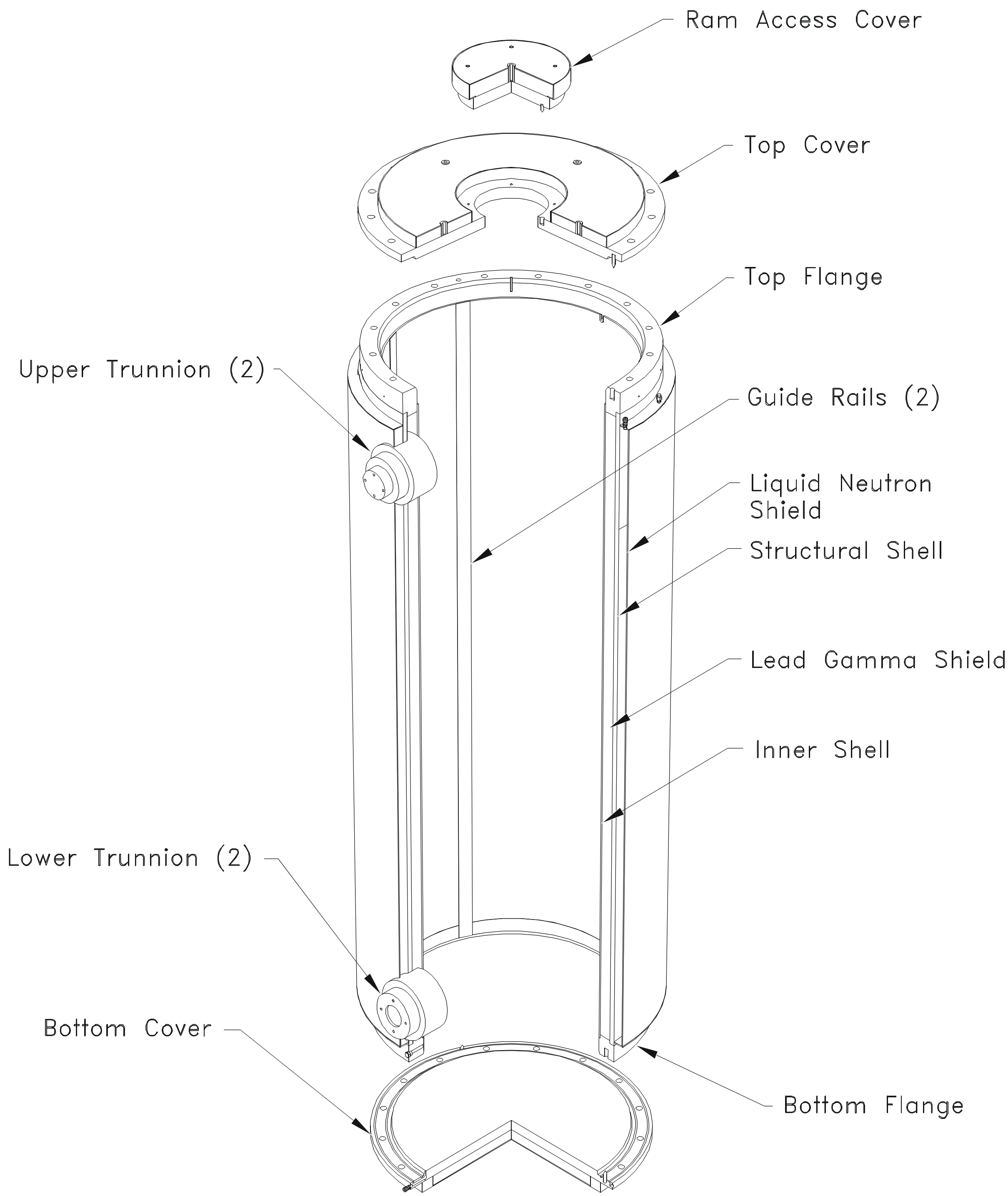

Figure 1-4 W100 Transfer Cask

2.0 PRINCIPAL DESIGN CRITERIA

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

2.1 Structures, Systems and Components Important to Safety

The applicant presented a summary of the principal FuelSolutions™ Storage System design criteria in Tables 2.0-1 and 2.0-2 of the SAR, for the transfer cask and the storage cask, respectively. Tables 2.0-1 of the W74 and W21 SARs provide design criteria for each canister. As shown in Table 2.1-1 of the WSNF-200, each FuelSolutions™ Storage System component is assigned a safety classification that is based on the component's function and on an assessment of the consequences of component failure.

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

The applicant's design bases summary for the FuelSolutions™ Storage System identified the range of spent fuel configurations and characteristics, the enveloping conditions of use, and the bounding site characteristics.

2.2.1 Spent Fuel Specifications

The FuelSolutions™ Storage System is designed to store up to 21 PWR or 64 BWR spent fuel assemblies. Detailed fuel assembly characteristics for the authorized contents are given in Table 2.1-2 of the W74 canister TS and in Tables 2.1-3 and 2.1-4 of the W21 canister TS. These characteristics include: manufacturer, assembly array, physical assembly dimensions, maximum and minimum enrichments, maximum burnup, minimum cool time, maximum decay heat, total weights per assembly, initial uranium weight per assembly and applicable cooling table. Detailed parameters regarding the configuration of individual fuel rods, fill gas pressures, and fuel assembly hardware are also provided.

Fuel assemblies shall not contain empty fuel rod positions. Any fuel rod that has been removed shall be replaced by a solid dummy rod, as specified in the TS. Control components are authorized for storage, as specified in the TS.

Section 2.2 of the canister specific SARs specify the bounding fuel types for the criticality and shielding evaluations and provides the design bases maximum decay heat load. The enrichments, burnups, decay heat rates, and cooling times vary, as specified in the cooling tables, and are specified in the TS.

2.2.2 External Conditions

Section 2.3 of the SAR identifies the site environmental conditions and natural phenomena for which the storage system is analyzed during the period of storage. The SAR presents analyses which demonstrate that the FuelSolutions™ Storage System meets the design criteria in subsequent SAR sections, and which is further evaluated in Sections 3 through 14 of this SER.

SAR Sections 2 and 11 identify the normal, off-normal, and accident conditions for which the FuelSolutions™ design has been evaluated. The staff's evaluation of the system's response to off-normal and accident conditions is located in Section 11 of this SER. The TS, in Section 4.3, identifies the site-specific parameters and analyses that are required to be verified by the FuelSolutions™ system users.

2.3 Design Criteria for Safety Protection Systems

A summary of the Safety Protection Systems is provided in Section 2.4 of the SAR. These systems are further evaluated in Sections 3 through 14 of this SER.

2.3.1 General

The design life of the FuelSolutions™ Storage System is 100 years. The codes and standards of design and construction of the system are specified in Section 3.1.2 of the SAR. The system is approved for a 20-year period.

2.3.2 Structural

The applicant performed deterministic and probabilistic evaluations to show that a tip-over of the storage cask onto the ISFSI pad during storage conditions is not a credible event. The evaluation results showed: (1) a minimum factor of 1.1 against tip-over based on a peak ground acceleration (PGA = 0.5g) which bounds all reactor sites east of the Rocky Mountain Front, and (2) the probability of occurrence of a beyond-design basis external event which could lead to cask tip-over is acceptably low ($\leq 1 \times 10^{-4}$). The staff reviewed the applicant's evaluation results and concurred that storage cask tip-over onto the ISFSI pad is non-credible. Pursuant to 10CFR72.236(l), only credible accident conditions need be addressed in Section 11 of the SAR. The staff notes that the applicant performed a consequence analysis for the tip-over event. The staff did not perform an acceptance review of the radionuclide release fractions used by the applicant, as this was not required for the staff to render its finding that a tip-over event is a non-credible event. The staff believes that additional data and analyses would be required to render a judgement on the release fractions and activities used by The applicant in this analysis. However, the storage cask is evaluated for a tip-over accident during up- or downending operation of the storage cask while it is secured to the j-skid in Section 3 of the SAR.

Section 3 of the SER evaluates the structural integrity of the FuelSolutions™ Storage System under the combined normal, off-normal, and design basis accident condition loadings. For the W150 Storage Cask, loads are combined together in accordance with the load combinations identified in NUREG-1536 for both the reinforced concrete components and the structural steel components, which are consistent with those load combinations specified in ANSI/ANS57.9. For the W100 Transfer Cask, and the W21 and W74 Storage Canisters, the loading combinations are categorized based on the Section III service level criteria of the ASME Boiler and Pressure Vessel Code for evaluation against the corresponding allowable values. The FuelSolutions™ Storage System structural components are designed to protect the stored spent nuclear fuel assemblies from significant structural degradation, preserve retrievability, and maintain subcriticality and confinement.

2.3.3 Thermal

The thermal analysis is presented in Section 4 of the SAR. The FuelSolutions™ Storage System is designed to passively reject decay heat under normal, off-normal and accident conditions. Heat removal by conduction, radiation, and natural convection (both internal and external to the canister) was addressed in the thermal analyses. The thermal design criteria include maintaining fuel cladding integrity, for both for low and high burnup fuel (to a maximum burnup of 60,000 MWD/MTU) and ensuring that the temperatures of materials and components important to safety are within the design limit.

2.3.4 Shielding/Confinement/Radiation Protection

Sections 5, 7, and 10 of this SER evaluate the FuelSolutions™ W150 storage cask design criteria which protects occupational workers and members of the public against direct radiation and radioactive material releases, and which minimizes doses after any postulated off-normal or accident condition, sufficient to meet the requirements of 10 CFR Part 72. Section 11 of this SER evaluates the effect of radiological consequences for hypothetical accidents. The FuelSolutions™ canisters use a welded closure system to provide confinement. Radiation exposure is minimized by the steel and concrete shields and by operational procedures.

2.3.5 Criticality

The FuelSolutions™ Storage System has been designed to assure that the effective neutron multiplication factor is less than or equal to 0.95 under all credible conditions. Section 6 of this SER evaluates the control methods which maintain the subcriticality of the system. The control methods used include a neutron absorbing material in the basket and the basket geometry. The continued efficacy of the neutron absorber plates over a 20-year storage period is assured by testing at the time of manufacture and by the design of the W21 and W74 fuel canisters. The neutron flux in the dry cask over the storage period is also very low such that depletion of the Boron-10 in the neutron absorber is negligible.

2.3.6 Operating Procedures

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

2.3.7 Acceptance Tests and Maintenance

The acceptance tests and maintenance of the FuelSolutions™ Storage System are described in Section 9 of the SAR, including the commitments, industry standard, and regulatory requirements used to establish the acceptance, maintenance, and periodic surveillance tests.

2.3.8 Decommissioning

Decommissioning of the FuelSolutions™ Storage System is described in Section 2.5 of the SAR and evaluated in Section 14 of this SER.

2.4 Evaluation Findings

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

3.0 STRUCTURAL EVALUATION

Scope

This section reviews and evaluates the structural design of the components of the FuelSolutions™ Storage System. Structural design features and design criteria are reviewed, and analyses related to structural performance under normal, off-normal, accident, and extreme natural phenomena events are evaluated.

Loads and load combinations are reviewed to meet those specified in the Standard Review Plan (SRP) (NUREG-1536[1]). Material specifications are reviewed and compared with acceptable codes and standards. The constructions of the components (e.g., fabrication, examination and testing) are reviewed to ensure they meet the requirements of the design code. The analytical procedures and design assumptions are reviewed for appropriateness and acceptability. The analytical results are reviewed to ensure the acceptance criteria of the governing design codes have been met.

3.1 Structural Design

3.1.1 Structural Design Features

The FuelSolutions™ Storage System includes the W150 Storage Cask, W100 Transfer Cask, and two types of Dry Storage Canisters. The W150 Storage Cask is a modular precast, reinforced concrete cylindrical structure used to house a loaded canister for storage. It provides biological shielding, structural protection, and passive convective heat removal of a loaded canister during dry storage at an ISFSI. The W100 Transfer Cask is a lead-shielded stainless steel cask used for on-site transfer of a loaded canister. The transfer cask provides biological shielding and structural protection for the canister during transfer. The two dry storage canisters, W21 and W74, are of similar designs with variations in the internal fuel basket assembly to receive different types of fuel assemblies. The W21 canister will receive up to 21 PWR fuel assemblies and the W74 canister will receive up to 64 BWR fuel assemblies. The canisters are all welded stainless steel cylindrical vessels with flat ends and an internal fuel basket assembly. The dry storage canister provides confinement, criticality control and passive heat removal for the spent nuclear fuel (SNF) during storage and transfer. A complete description of the FuelSolutions™ Storage System is provided in Section 1 of the storage system SAR and the respective canister SARs.

3.1.2 Structural Design criteria

This subsection discusses the codes and standards, individual loads, load combinations, and allowable stresses used in the design evaluation of the storage system components.

3.1.2.1 Codes and Standards

Codes and standards used for the design evaluation of the various storage system components are reviewed to ensure they are acceptable in accordance with NUREG-1536.

W150 Storage Cask

The W150 Storage Cask includes both reinforced concrete and structural steel components. The reinforced concrete component is designed in accordance with ACI 349[2], and constructed with the applicable requirements of ACI 318[3]. The steel component is designed and constructed in accordance with the requirements of AISC[4] Manual of Steel Construction.

W100 Transfer Cask

The structural steel components of the transfer cask, with the exception of lifting trunnions, are designed and fabricated in accordance with the applicable requirements of ASME Code Section III, Subsection NF[5]. The lifting trunnions are designed for non-redundant lifting devices in accordance with NUREG-0612[6] and ANSI N14.6[7].

W21 and W74 Storage canisters

Both canisters are designed and constructed based on the ASME Code. The structural components include the shell assembly and an internal basket assembly. The internal basket assembly is designed and constructed as a core support structure in accordance with the applicable requirements of Section III, Subsection NG[8] of the ASME Code. The shell assembly is designed and constructed as a Class 1 pressure vessel component in accordance with Section III, Subsection NB[9] of the ASME Code. The only exception to the Code is the top end inner and outer closure plate welds to the canister shell. However, the top end closure design meets the requirements recommended in the ISG-4, Rev. 1[10] and thus is acceptable.

3.1.2.2 Design Loadings

Design loadings are described in Section 2.3 of the storage system SAR and the respective canister SARs. Design basis conditions considered for the storage system components are: Normal Conditions, Off-Normal Conditions, Accident Conditions, and Natural Phenomena Events. Normal conditions include loads (dead, live, thermal, and pressure) that occur regularly during normal operation. Off-normal conditions are defined in accordance with NUREG-1536 and ANSI/ANS-57.9[11] including events which, although not occurring regularly, can be expected to occur with moderate frequency on the order of no more than once a year. The design-basis postulated accident conditions and natural phenomena events are in accordance with NUREG-1536 and ANSI/ANS-57.9. Individual loads induced by the postulated accident and natural phenomena events are defined by the applicant as described below.

Tip-Over

The applicant performed comprehensive evaluations to show that FuelSolutions™ storage cask tip-over onto the ISFSI pad during storage conditions is not a credible event. The evaluation included deterministic evaluations, probabilistic evaluations, and consequence evaluations. The results of the evaluations showed: (1) a minimum factor of safety of 1.1 against tip-over based on a peak ground acceleration (PGA) which bounds all reactor sites east of the Rocky Mountain Front, and (2) the probability of occurrence of a beyond-design basis external event which could lead to cask tip-over is acceptably low ($\leq 1 \times 10^{-4}$). As the result, the only tip-over accident postulated to occur is that during up- or down-ending operation of the storage cask while secured to the J-skid. Since the cask up- or down-ending operations can occur only over the impact limiter (recessed into the ISFSI pad) in the storage cask handling area, the maximum dynamic acceleration of the cask is determined to be 21.9g. The dynamic acceleration is then increased by the dynamic load factor (DLF) of 1.22 to an equivalent static g-loading of 26.7g. A bounding static acceleration load of 28g is used for the tip-over structural analysis of the storage cask. Similarly a bounding equivalent static acceleration load of 30g is conservatively used for the tip-over structural evaluation of the canisters.

Handling Drop Accident

In addition to tip-over, there are two postulated handling drop accidents for the FuelSolutions™ Storage System components. The first scenario is a bottom end 36-in vertical drop of the storage cask. This drop is postulated to occur when the storage cask is lifted to be placed onto or removed from the J-skid. The second scenario is a 72-in side drop of the transfer cask. The transfer cask side drop is postulated to occur at the canister transfer pad outside of the fuel handling building.

The methodology used to evaluate the drop accident accelerations is based on the approach presented in NUREG/CR-6608[12] by the Lawrence Livermore National Laboratory (LLNL). The maximum peak rigid-body accelerations for the end drop in the storage cask, canister, and basket assembly are 57g, 28g, and 26g, respectively. For the storage cask body, with the lowest axial vibration frequency of 162 Hz, the elastic DLF is determined to be 1.5. Thus, the equivalent static loading for the cask body is 57g x 1.5 = 86g. An acceleration of 89g is conservatively applied to the cask body in the stress analysis. The maximum DLF for an undamped system subjected to a half-sine wave pulse is 1.75. Therefore, the maximum equivalent static acceleration for the canister and basket assembly is 1.75 x 28g = 49g. A bounding equivalent static acceleration of 50g is conservatively used in the structural evaluation of the canister and the basket assembly for the storage cask end drop condition. The maximum transfer cask peak rigid-body acceleration resulting from the 72-in side drop is 46g. Based on the lowest cask natural frequency of 82 Hz and the drop duration of 22.5 msec, the maximum DLF for a half-sine wave pulse is 1.3. Thus, the equivalent static loading is 1.3 x 46g = 60g. A static loading equivalent to an acceleration of 60g is applied to the transfer cask and the canisters in the stress analysis for the postulated 72-in side drop of the transfer cask.

Explosive Over-pressure

The explosive over-pressure effects are bounded by the design basis tornado wind loads because there are no potentially explosive materials stored in close proximity to the ISFSI. However, the general licensee should compare site-specific hazards to the design basis tornado wind to assure that the resulting explosive over-pressures are indeed bounded. The canister is completely enclosed in the storage cask or transfer cask. Therefore, the canisters are not subject to direct explosive over-pressures. For the purpose of providing a design basis external pressure criterion for comparison to site-specific hazards, the explosive over-pressures acting on the canisters are taken to be the same as the external pressures due to flooding.

Flood

The design basis flood for the storage cask is defined to be a 50-foot flood height [this is conservatively based on the water immersion depth of 10CFR71.73(c)(5)], and a water velocity of 21 ft/sec. Because of the infrequent use of the transfer cask, the transfer cask is not evaluated for flood conditions. However, the licensee's operating procedures should prohibit transfer cask usage during flood conditions**.** The canisters are evaluated for a 50-foot hydraulic head of water which corresponds to an enveloping design basis flood of 50-foot flood height. The canisters are designed for fresh water optimum moderation with neutron absorbers and as such criticality safety during flooding is assured.

Tornado and Tornado Missile

The storage cask and transfer cask are designed to withstand the loadings associated with tornados, as prescribed by Regulatory Guide 1.76[13]. The loading characteristics specified for Region I, which is the most severe for the United States, is used as the design basis tornado loading. The design basis tornado missiles are based on NUREG-0800[14], Section 3.5.1.4, Paragraph III.4. Accordingly, three types of tornado missiles are postulated including: (1) a 4000-pound automobile (with a frontal crashing area of 20 square feet), (2) a 275-pound, 8-in diameter armor piercing artillery shell (simulated by an 8-in diameter steel cylinder), and (3) a 1-in diameter solid steel sphere. The storage cask is evaluated for the effects of tornado missile impact, including overturning and sliding of the cask. The stability of the transfer cask is evaluated to show that sliding and/or overturning of the transfer cask when secured to the transfer skid and trailer does not occur. Since the canister is completely enclosed in the storage cask or transfer cask, the canister is not subject to tornado winds and tornado generated missiles.

Earthquakes

The design basis earthquake (DBE) is defined as having free field peak ground acceleration of 0.25g's acting simultaneously in two orthogonal horizontal directions, and the vertical direction. The storage cask and transfer cask are analyzed for earthquake loads using equivalent static moment equilibrium methods. The storage cask is shown to be stable, with no tip-over under the influence of the inertia forces caused by the combined (horizontal and vertical) DBE seismic accelerations. The transfer cask is analyzed for stability under the influence of the inertia forces caused by the combined DBE seismic accelerations during loading operations. Similarly, the canisters are evaluated for the effects of seismic accelerations with the canister in the storage cask or the transfer cask.

Wind

The design basis wind (DBW) speeds for the evaluation of the storage cask and transfer cask is taken to be 150 mph in accordance with ASCE 7[15]. The effects of DBW are included as an off- normal condition loading. The resulting pressure due to DBW is 62 psf for the storage cask and 74 psf for the transfer cask. Canisters are completely enclosed in the storage cask or transfer cask. Therefore, canisters are not subjected to wind-related loading.

Lightning

The storage cask steel liner and reinforcement include grounding provisions meeting the applicable requirements of the National Fire Protection Association (NFPA) Codes. Provisions for the attachment of an external ground cable from the storage cask to an ISFSI grounding grid are provided. The need for such grounding is to be evaluated by the licensee in accordance with site-specific conditions. The licensee's operating procedures should require that canister transfer operations should not be conducted outside the fuel handling facility during the weather conditions in which lightning could occur.

Snow and Ice

Loads due to snow and ice are considered live loads. Based on ASCE 7, Section 7.0 and a conservative bounding value of 100 psf ground snow loads, the snow and ice loading on the storage cask is 101 psf. The transfer cask is not evaluated for snow and ice loading because of the infrequent, short term use. Canisters are protected by the storage cask or the transfer cask. Therefore, the canisters are not subjected to any snow and ice related loads.

Fire

No combustible materials are used in the construction of the storage cask or transfer cask. The ISFSI is typically sited in areas free of combustible materials and combustible materials such as vehicle fuel during transfer operations are carefully controlled (i.e., not to exceed 70 gallons) in order to limit the duration of the fire to 5 minutes. Nevertheless, the storage cask and transfer cask are evaluated for the effects of a postulated fire accident.

The postulated fire accident is defined using the transportation cask fire criteria described in 10CFR71.73, with the exception of the fire duration. The casks are assumed to be engulfed in a hydrocarbon fuel/air fire of sufficient extent and ambient conditions to provide an average emissivity coefficient of 0.9, with an average flame temperature of at least 1475^{o} F for a period of 5-minutes. The resulting temperature effects due to the fire accident are included in the structural analysis of the storage cask, transfer cask, and the canisters.

3.1.2.3 Loading Combinations

The FuelSolutions™ Storage System is subjected to normal, off-normal, postulated accident and natural phenomena condition loadings. The loadings are combined together in accordance with the load combinations specified in NUREG-1536.

Storage Cask

The storage cask is subject to normal, off-normal, postulated accidents, and natural phenomena condition loadings as follows:

- Normal Loads - Normal ambient conditions, dead, live, handling, snow and ice.

- Off-normal Loads - Extreme ambient conditions, wind, cask misalignment during cask horizontal transfer.

- Postulated Accident and Natural Phenomena Loads - Complete blockage of storage cask air inlet/outlet vents, cask end drops, cask tip-over while secured on J-skid, flood, fire, tornados, earthquake.

The above loadings are combined together in accordance with the load combinations identified in NUREG-1536, Table 3-1, for the structural evaluations of reinforced concrete and structural steel storage cask components. Loading combinations are summarized in Table 2.3-6 and Table 2.3-7 of the WSNF-200.

Transfer Cask

The transfer cask is subjected to the normal, off-normal, postulated accident, and natural phenomena event loadings:

- Normal Loads - Normal ambient conditions, internal pressure (neutron shields only), dead, lifting and handling.

- Off-Normal Loads - Off-normal ambient conditions, internal pressures (neutron shield only), wind, misalignment during canister horizontal transfer.

- Postulated Accident and Natural Phenomena Loads - Loss of neutron shield, cask side drops, fire, tornados, earthquake.

The above loadings are combined together and used in the structural analysis of the transfer cask. The load combinations of NUREG-1536, Table 3.1 are used for steel structures for allowable stress design (without the strength factors). Load combinations are summarized in Table 2.3-8 of WSNF-200. Load combination stresses are categorized based on the ASME Code, Subsection NF, Service Level Criteria for evaluation against the associated allowable stress intensity values.

Canisters

The canisters are subject to normal, off-normal, postulated accidents, and natural phenomena condition loadings as the following:

- Normal Loads - Normal ambient conditions, internal pressure, dead weight, handling.

- Off-Normal Loads - Extreme ambient condition, off-normal internal pressure, re-flood, misalignment during canister horizontal transfer operation.

- Postulated Accident and Natural Phenomena Loads - Complete blockage of storage cask air inlet/outlet vents, transfer cask loss of neutron shield, cask drops, cask tip-over, fire, accident internal pressure, flood, earthquake.

The canister internal basket and shell assembly components are designed for the loading combinations shown in Table 2.3-1 of WSNF - 201 and - 203. Load combination stresses are categorized based on the ASME service level criteria for evaluation against the associated allowable stress intensity values.

3.1.2.4 Allowable Stresses

Allowable stresses are developed for the various FuelSolutions™ Storage System components based on the appropriate design codes and loading conditions. The storage cask concrete stress and section allowable are provided in Table 3.1-2, the storage cask structural steel allowable stresses are provided in Table 3.1-4, and the transfer cask allowable stresses are provided in Table 3.1-5 of WSNF-200. The allowable stresses for the canister shell and the basket assembly are provided in Tables 3.1-5 and 3.1-6 of WSNF-201 and in Tables 3.1-4 and 3.1-5 of WSNF-203.

3.1.3 Weights and Center of Gravity

The bounding weights and center of gravity locations for the storage cask are provided in Table 3.2-1; the weights of a transfer cask for lifting from the spent fuel pool, with or without the neutron shield, is provided in Table 3.2-2; the dry loaded transfer cask weights and center of gravity locations are provided in Table 3.2-3 of WSNF-200.

Summary of canister weights and centers of gravity are provided in Tables 3.2-1 through 3.2-2 of WSNF-201 and in Table 3.2-1 of WSNF-203.

3.1.4 Materials

The staff reviewed the materials information presented for the FuelSolutions™ W150 Storage Cask, W100 Transfer Cask, W74 and W21 Storage Canisters to determine whether the materials of construction of the FuelSolutions™ Storage System meet the requirements of 10 CFR Part 72. In particular, the following aspects were reviewed: materials' selection; applicable codes and standards; weld specifications, consumables and qualifications; bolt specification; chemical and galvanic reactions; coatings; and long term performance issues, such as, delayed cracking, brittle fracture, corrosion, lead slumping and changes in toughness (brittle fracture), and thermal aging.

The principal design, fabrication and material criteria and applicable codes and specifications for the FuelSolutions™ Storage System components are listed tables in Chapter 2 and 3 of their respective SARs. The corresponding safety classifications are also defined. Essential additional materials information and applicable specification information are also identified on the general arrangement drawings in Chapter 1 of the respective SARs and the Engineering drawings.

W 150 STORAGE CASK

The W150 storage cask is a modular concrete structure with a concrete and steel cover at the top end. The specifications for the concrete are addressed in Section 3.1.2.1 of this document. There is a prohibition on the use of aluminum for embeds and a prohibition of its use in mixing, transportation and placement of the concrete.

Within the cavity of the storage cask is a carbon steel liner, an aluminum thermal shield, austenitic steel guide rails with hardened stainless steel facing to provide bearing surfaces during canister transfer operations, and stainless support rails. The inside diameter surface of the aluminum thermal shield is coated with a high emissivity, temperature resistant coating. High strength tie rods are used to secure the concrete precast segments of the storage cask together. The canister is supported vertically on the bottom of the storage cask cavity by radially arranged, austenitic stainless steel seamless tube pipe sections. The exact ASTM designations for each of the materials and applicable welding specification are given in Table 3.3.1 of the WSNF-200 SAR. Further details of the materials of construction are stated in the materials schedule and General Notes found in Section 1.5.1 Drawings of the WSNF-200 SAR.

The W150 storage cask is designed for temperatures as low as -40°F. The cask materials of construction are selected to provide protection against brittle fracture failure down to that temperature. The storage cask liner (shell, bottom plate, and shield ring) and top cover plate are considered Category III (as defined by NUREG/CR-1815) components and must have sufficient fracture toughness to prevent fracture initiation at minor defects typical of good fabrication practices. Section 5.3.1(4) of NUREG/CR-1815 states that this can be achieved by specifying a material with a minimum energy absorption (Cv) of 15 ft-lb at 10°F. Therefore, these components are fabricated from carbon steel with a supplementary specification requirement of having the above fracture toughness. The storage cask cover is attached by steel cap screw bolts. As stated in Section 5 of NUREG/CR-1815, bolts are generally not considered as fracture-critical components because multiple load paths exist and bolting systems are redundant.

W100 TRANSFER CASK

The W100 transfer cask consists of the cask body and the top and bottom cover assemblies which are all austenitic stainless steel. The primary structural components of the transfer cask body are stainless. Hardened-steel rails are welded to the inner shell cavity to facilitate horizontal canister transfer. The annulus formed by the transfer cask inner liner, structural shell, and top and bottom end flanges is filled with ASTM-B-29 chemical lead for gamma shielding. Since the lead is in-effect encapsulated, slumping is not a problem.

The transfer cask body includes a neutron shield jacket of stainless steel. The transfer cask upper and lower trunnions are nitrogen strengthened stainless. The transfer cask top end closure and ram access cover are stainless. The ram access cover is bolted to the stainless steel top cover which, in turn, is bolted to the transfer cask body top stainless flange. The bottom end closure is formed by the bottom cover bolted to the transfer cask body bottom flange. All of the transfer cask covers include thick structural steel plates and an encased solid neutron shielding of RX-277 or NS-3 (boron loaded resin polymers).

The W100 transfer cask has a stainless steel inner liner and an outer structural steel shell, with lead gamma shielding in the annular space between them. The ends of the transfer cask body include stainless forged flanges that are welded to the structural shell and inner liner. The upper and lower lifting trunnions are welded to the structural shell and inner liner. The covers at each end of the cask are bolted to the flanges.

Structural components of the transfer cask that are important to safety are designed in accordance with the criteria of ASME Code, Section III, Subsection NF, for Class 1 component supports. These criteria are applicable to the transfer cask inner liner, structural shell, top flange, bottom flange, lower trunnions, neutron shield jacket, top cover, bottom cover, ram access cover, closure bolts, and all structural welds, including the trunnion-to-shell welds. A summary of the transfer cask component functions, safety classes, and applicable codes and standards is given in Table 3.1-1 of the WSNF-200 SAR.

The transfer cask lower trunnions are used only for rotating the transfer cask between vertical and horizontal orientation and supporting the transfer cask horizontally on the transfer skid. The lower trunnions are not interfacing lift points for any critical lift conditions. As such, the lower trunnions do not have to meet the requirements of NUREG-0612 and are designed in accordance with Subsection NF of the ASME Code, as discussed above.

W21 CANISTER

There are eight different W21 canister assembly configurations, all of which are similar in design. The W21M and W21T canisters, which are designed for on-site storage in the W150 storage cask, consist of a shell assembly and a basket assembly. The variations in W21M and W21T class canisters (e.g., materials and dimensions) are identified in Table 1.2-1 of the W21 SAR, WSNF-201. From a materials standpoint, the main difference between the M and T versions is the use of 316 or 304 stainless steel with the option of using a reduced carbon version of 304 for increased resistance to stress corrosion cracking in the T version. General arrangement drawings containing the bill of materials for the W21M and W21T class canisters are provided in Section 1.5.1 of the W21 SAR. The basket assembly, which is sealed inside the canister shell assembly cavity, maintains the positions of the spent fuel assemblies and neutron absorbing materials. The function of each of the W21 canister assembly components is summarized in Table 3.1-1 of the W21 SAR.

Depleted Uranium (DU) and carbon steel are used for the W21M shield plugs. Lead and carbon steel are used for the W21T shield plugs. The canister shell and closure plates for both the W21M and W21T class canisters are fabricated from austenitic stainless steel to provide corrosion protection and fracture toughness. The specific material designations for the FuelSolutions™ W21T and W21M class canister shell assemblies are in the Bill of Materials on the drawings in Section 1.5.1 of the W21 SAR.

The canister top end closure plate welds are partial penetration welds that are structurally qualified by analysis. The inner closure plate welds are inspected by performing a liquid penetrant examination of the root pass and final weld surface. The integrity of the top end inner closure plate welds is verified by performing a pneumatic pressure test, and a helium leak test in accordance with the TS requirements contained in Section 12.3 of the SAR. The outer closure plate weld is inspected by performing a liquid penetrant examination of the root pass, intermediate pass, and final weld surface. This weld non-destructive examination (NDE) is in compliance with NRC Interim Staff Guidance #4 (ISG-4). The associated critical flaw size evaluation to support the NDE acceptance basis is provided in Section 3.9.5 of the W21 SAR. This critical flaw size evaluation also supports the optional use of UT inspection of the outer closure plate to shell weld.

The structural analysis of the W21 canister, in conjunction with the redundant closures and nondestructive examination, pneumatic pressure testing, and helium leak testing performed during canister fabrication and canister closure, provides assurance of canister closure integrity in lieu of the specific weld joint requirements of Section III, Subsection NB.

The W21 canister basket assembly contains borated aluminum neutron absorber sheets (Boral) and a thin outer wrapper. The Boral panels are sealed between the inner structural tube and the outer wrapper. No borated materials are formed, bent, welded, or used as structural members. The carbon steel spacer plates and support sleeves are coated with electroless nickel for corrosion protection. This is discussed further in Section 3.4.1 of the W21 SAR. The specific material specification details for the W21T and W21M class basket assemblies are shown in the Bill of Materials on the drawings in Section 1.5.1 of the SAR.

The field closure welds are designed to provide structural integrity while minimizing heat input and distortion of the canister shell. Shop fit-up of the inner closure plate with the canister shell is carefully controlled to minimize weld shrinkage during field welding. The outer closure plate is designed to allow for weld shrinkage. All canister top end closure welds are liquid dye penetrant examined; the inner closure weld at the root and final weld passes, and the outer closure weld at the root, intermediate, and final weld passes.

The canister shell assembly has redundant welded closure plates of Type 304 or 316 stainless. The shell assembly stainless steel material provides corrosion protection and has minimum susceptibility to weld sensitization. Each shell assembly contains a top and bottom shield plug. DU or lead are used for the long cavity canister shield plugs, and carbon steel or lead for the short cavity canister shield plugs. A comparison of the shield plug assembly materials and dimensions used in each canister configuration is provided in Table 1.2-2 of the W21 SAR. The DU or lead shield plugs are encased in stainless steel, so there is no concern for lead slumping or oxidation. The top carbon steel shield plugs are coated with a suitable coating to provide corrosion protection.

The FuelSolutions™ W21M and W21T canister basket assemblies are all similar in construction. A comparison of the various basket assembly materials and dimensions is provided in Table 1.2-2 of the SAR. The W21M and W21T canister basket assembly designs are shown on the general arrangement drawings in Section 1.5.1 of the W21 SAR. Each basket assembly contains two types of spacer plates. The 2-in thick Type A spacer plate is fabricated from a single XM-19 stainless steel plate. The 3/4-in thick Type A and B spacer plates are fabricated from electroless nickel coated carbon steel.

Each guide tube assembly consists of an inner guide tube, four neutron absorber panels, and an outer wrapper. The guide tube materials and cross section dimensions are identical for all W21 basket assembly designs. Only the guide tube lengths are varied to fit within the cavities of each W21 canister shell assembly. The inner guide tube and outer wrapper are fabricated from Type 316 stainless steel. All of the W21 basket assemblies use a borated aluminum material as the fixed neutron absorber panels.

The W21 canister shell subassembly provides primary confinement for storage conditions. As such, the W21 canister shell assembly confinement components are designed in accordance with the applicable requirements of Subsection NB of the ASME Code as discussed in Section 2.1.2 of the SAR. The W21 canister cylindrical shell seam welds are full penetration groove welds designed and RT inspected per Subsection NB. The W21 shell assembly includes redundant confinement welds. The weld details between the top inner and top outer closure plates, and the cylindrical shell are partial penetration groove welds. All canister top closure welds are liquid penetrant examined: the inner closure weld at the root and final weld passes, and the outer closure weld at the root, intermediate, and final weld passes. The full penetration weld between the bottom closure plate and the canister shell is examined by both liquid dye penetrant and radiographic examination.

Not all of the non-pressure retaining materials specified in WSNF-200 are listed as ASME Section III materials. According to the applicant, canisters are purchased, controlled and manufactured using a graded quality approach in accordance with the NRC approved BFS Quality Assurance Program based on NQA-1, NRC Regulatory Guide 7.10 and NUREG/CR-6407 criteria.

The acceptance criteria of Subsection NB of the ASME Code are applied to the W21 closure welds with additional conservative measures to assure that the design margins inherent in the NB rules are maintained. The canister shell assembly components which do not provide confinement, are designed and tested in accordance with the design criteria of Section III, Subsection NF of the ASME Code.

W 74 CANISTER

The structural materials used for the W74 canisters are listed in Section 3.1.1 of WSNF-203. Tables 3.3-1 to 3.3-9 of the SAR list the primary function of each cask component along with information on material specification, welding process and weld filler metal (if applicable), and material mechanical properties. In addition, Section 3.3 of the SAR discusses pertinent additional information on the usage and mechanical properties of the materials.

The design and materials are so similar to that in the W21 canister that they will not be addressed here in the detail allotted to the W21 materials. Sufficient detail is in the W74 SAR as indicated above. The different materials not covered above in the discussion of the W21 canister are: the bottom shield plug material is ASTM A36: the top shield plug material is ASTM A516, Grade 55 or 60; there is more extensive use of XM-19 stainless steel; and the nonstructural neutron absorber sheet in the canister is boron alloyed stainless steel meeting ASTM A887, Type 304 B5.

All plate materials used in the fabrication of the canister shell and inner and outer top and bottom closure plates are austenitic stainless steels meeting the requirements of the ASME Boiler and Pressure Vessel Code (the Code). Paragraph NB-2530 of the Code ensures that there are no significant sub-surface defects that could cause delimitations or other propagating defects during welding and fabrication by requiring 100% straight beam UT from at least one major surface of the plates.

Material properties are taken from Section II, Part D of the Code or from NUREG reports. These materials meet the applicable requirements of Section III, Subsection NB (Class 1) of the Code or an acceptable alternative that provides an acceptable level of quality and safety.

The W74 canister cylindrical shell seam welds are full penetration groove welds, designed and radiographically inspected (RT) in accordance with Subsection NB of the Code. The W74 shell assembly includes redundant confinement welds at the joints between the shell and the inner and outer closure plates. The welds are partial penetration welds. All canister top and closure welds are liquid penetrant (PT) examined with the inner seal closure weld examined after the root pass and final weld passes. The outer closure weld receives PT examination at the root, intermediate and final weld passes. The full penetration weld between the bottom closure plate and the canister shell is examined by both PT and RT. The canister shell extension and bottom end plate attachment welds are non-pressure boundary welds and are PT examined. Further minimization of weld shrinkage stresses is realized by specifying that all filler materials have a minimum ferrite content of 10FN (ferrite number), which has been shown to result in less weld shrinkage.

The closure weld for the outer cover plate is performed according to the requirements of ASME Code to the quality level of Section III, Class 1 materials. The weld closure design for the W74 (and W21) canister has been designed to minimize weld shrinkage. For example, the shield plug is separate from the inner closure plate, minimizing weld shrinkage constraint by incorporating an inner closure plate-to-shell weld gap that is optimized to provide the design partial penetration root depth and minimizing the potential weld shrinkage volume. The gap is larger than that required by welding considerations alone, thereby aiding in the placement of the shield plug in the spent fuel pool after spent fuel loading. In addition, to prevent weld shrinkage during tack and root pass welding of the outer closure plate to canister shell, high ferrite content filler metal is used which has been shown to minimize shrinkage.

3.1.4.1 Welds

The applicant has stated that canister welds, except the redundant canister closure welds, are performed, tested and inspected in accordance with ASME Code, Section III, Subsection NB. The redundant canister closure welds are partial penetration welds. The applicant has taken exception to the ASME code by using an acceptable alternative that employs multi-pass redundant welds subject to multi-level liquid penetrant examination and a combined pneumatic pressure and helium leak rate test at a hydrostatic test pressure to assure structural integrity and leak tightness, along with an appropriate reduction factor in allowable applied stress. The staff has found this to be acceptable and consistent with the guidance in ISG-4, Rev. 1. Such a redundant confinement boundary weld complies with 10CFR72.236(e) and is consistent with NUREG-1536. Welding of the storage cask will be performed by a permitted process and filler materials according to American Welding Society (AWS), D1.1 or American Institute of Steel Construction (AISC). Structural welding of the transfer cask will be performed in accordance with ASME Code, Section III, Subsection NF. The FuelSolutions™ system materials of construction (e.g., stainless, carbon, low alloy steels etc.) are readily welded using commonly available welding techniques, including remote welding. The welds are well characterized on the drawings and standard welding symbols and notations in accordance with AWS standards are used.

3.1.4.2 Coatings

The electroless nickel (EN) coating on carbon steel spacer plates is for corrosion protection following fabrication and during the brief immersion period during fuel loading in the spent fuel pool. The electroless nickel coating is acceptable to the staff based upon its successful and wide use in the electronics, petrochemical, automotive, and food industries.

3.1.4.3 Brittle Fracture

Brittle fracture considerations for the FuelSolutions™ Storage Cask and the Transfer Cask are discussed under their respective Materials sections.

Brittle Fracture Considerations for the W21and W74 Canisters

Both the W21 and W74 canister shell and basket assemblies are designed using materials which provide assurance of safety against failure due to brittle fracture. The fracture toughness requirements used for the W21 and W74 canisters are based on a Lowest Service Temperature (LST) for all on-site storage and transfer conditions which produce adequate fracture toughness. TSs have been established in Section 12.3 of the respective SARs which limit the minimum temperature of the transfer cask structural shell to 40°F during normal transfer operations, when the ambient air temperature is below 32°F. However, a conservative LST of 0°F is used to establish the fracture toughness requirements for the W21 and W74 canister assemblies.

The W21 and W74 canister shell assembly confinement components are designed in accordance with the fracture toughness requirements of ASME NB-2300. The W21 and W74 canister shell assembly confinement components are fabricated entirely from austenitic stainless steels. These materials do not undergo a ductile-to-brittle transition in the temperature range of interest (i.e., down to 0°F), and thus are not susceptible to brittle fracture. Accordingly, impact testing is not required for austenitic stainless steels in accordance with ASME NB-2311(a)(6).

The W21 and W74 carbon steel shield plugs are designed in accordance with the fracture toughness requirements of ASME NF-2300. As per NF-2311(b)(7) impact testing is not required for materials for which the maximum stress does not exceed 6,000 psi tension or is compressive since brittle fracture failure under these conditions is not credible. As shown in the W21 canister shell structural evaluation, the maximum stress in the top and bottom carbon steel shield plugs is less than 6,000 psi for the storage cask bottom end drop, which is the controlling on-site storage and transfer load condition. Therefore, brittle fracture failure of both top and bottom W21 canister carbon steel shield plugs is not a credible failure mode and impact testing is not required. Impact testing will be performed on the W74 top shield plugs in accordance with NF-2300.

The W21 DU and lead shield plugs are encased and supported by components fabricated entirely from austenitic stainless steels. These materials do not undergo a ductile-to-brittle transition. Accordingly, impact testing of the DU and lead casing plate materials is not required in accordance with NF-2311(b)(5).

The W21 and W74 canister basket assembly are designed in accordance with the fracture toughness requirements of NG-2300, except that the impact testing requirements of NUREG/CR-1815 are used instead of NG-2330. Since the basket assembly components do not provide confinement, the fracture toughness testing requirements from NUREG/CR-1815 for Category II steel are used. These requirements assure that the fracture toughness of the material is sufficient to prevent fracture initiation of pre-existing cracks under dynamic loading.

The W21 and W74 basket assembly structural components are fabricated from SA-240, Type 316 and SA-479, Type XM19 austenitic stainless steels; SA-564, Grade 630 precipitation hardened steel; SA-517, Grade P or F, or A514, Grade P or F, and SA-106, Grade C carbon steels; and SA-240, Type 304, SA-312, Type 304L, and SA-240, Type XM-19 stainless steels. The austenitic stainless steel materials do not undergo a ductile-to-brittle transition. Accordingly, impact testing of austenitic stainless steels is not required as per NG-2311(a)(5). The W21M and W21T and W74 carbon steel spacer plate material and W21T support rod segment material, all experience ductile-to-brittle transitions at a temperature lower than the NUREG/CR-1815 prescribed maximum NDT temperatures. Drop weight testing of these materials in accordance with ASTM E-208 will be performed to demonstrate that the NDT temperature is at or below the T_{NDT} test temperature.

The effects of irradiation on W21 and W74 material toughness properties were considered in accordance with the requirements of ASME NG-2332(d). The licensee's evaluation, based on an exposure of 100 years, found that the exposure will not change the fracture toughness properties of SA-517 or A514 carbon steels, which receive the greatest flux.

3.2 General Standards for Cask Storage System

The cask storage system is evaluated to show positive closure, adequate safety factors for lifting devices, no significant chemical or galvanic reactions, and acceptable service life. In addition, the structural analyses for the cask storage system components must show sufficient structural capabilities to withstand the postulated worst-case loads under normal, off-normal, accident and nature phenomena events with adequate margins of safety to preclude the following:

- Unacceptable risk of criticality,
- Unacceptable release of radioactive materials,
- Unacceptable level of radiations,
- Impairment of ready retrievability.

3.2.1 Positive Closure

Both the storage cask and the transfer cask have bolted closures to secure and retain the loaded canister within the cavity of these casks. Therefore, inadvertent opening of these casks is not possible. The canisters are welded shut and have no penetrations. In addition, a lock wire is installed in one or more of the top closure bolts of the storage cask as tamper proof device.

3.2.2 Lifting Devices

Storage Cask Lifting Devices

The storage cask is lifted vertically either from the top end using four of the tie-rods or from the bottom end using four jacks and air pallets. The design load for the storage cask vertical lift is equal to the dead weight of the heaviest loaded storage cask plus an additional 5% for as-built uncertainties and 15% for dynamic effects. In the case of the top end lift, spreader beams are used to ensure that the load is evenly distributed among all four tie-rods. The applicant provided analysis to show that the resulting maximum tensile force in each tie rod due to the vertical lift is less than the minimum preload applied to the tie-rods. The cask may also be lifted from the bottom using four jacks positioned inside the inlet vent openings. It is conservatively assumed that the total lift load is supported by only two diametrically opposed jacks in the event of uneven jacking. The analysis result showed that each jack must provide a bearing area equal to or greater than 34.0-in^2.

Transfer Cask Lifting Devices

The transfer cask has two integral lifting trunnions which are located near the top end and are used to lift the cask vertically. The requirements of NUREG-0612 and ANSI-N14.6 for critical lifts are applied to these trunnions and their attachment welds. The critical lift load is defined as the maximum weight of the transfer cask and contents plus an additional 15% to account for dynamic effects due to crane hoist motion.

Canister Lifting Devices

The canister lifting devices are designed in accordance with the requirements of ANSI N14.6 for critical lift conditions, including vertical canister transfer, vertical lifts of the empty canister prior to fuel loading, and vertical lifts of the canister top shield plug and top closure plates.

During vertical canister transfer operations, the canister is lifted and lowered using the canister vertical lift fixture, which engages a lift adapter bolted onto the top closure plate by sixteen (16) 1-1/8 inch diameter bolts. The closure plate and its attachment weld to the canister shell are designed in accordance with ANSI N14.6 for non-redundant lifting devices. Thus, stress design factors of six (6) for yield strength and ten (10) for ultimate tensile strength are used for the vertical lifting condition.

The empty canister, including the shell assembly and basket assembly, is lifted vertically from the top shield plug support ring and placed into the transfer cask prior to loading spent fuel. The empty canister is lifted with a lifting fixture that engages the bottom of the shield plug support ring in four (4) locations having a total contact length of 20-in. The canister top shield plug, inner closure plate, and outer closure plate are each lifted from four attachment points on the top surface of the plates for placement on the canister inside the spent fuel building. The design load for the attachments includes a 15% increase for crane hoist motion. In addition, for design purposes, the full design weight of the components is conservatively assumed to be supported by only two lifting attachments.

3.2.3 Chemical and Galvanic Reactions

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

Chemical and Galvanic Reactions - W100 Transfer Cask

No significant chemical, galvanic or other reactions are expected for the W100 transfer cask. The W100 transfer cask is constructed with austenitic stainless steel, lead and solid neutron shielding material. Austenitic stainless has a long history of non-galvanic behavior due to is passive oxide surface film. As to its direct contact with lead, this combination has been used in similar casks for more than 30 years and no degradation has been found. The solid neutron shielding material is encapsulated.

Chemical and Galvanic Reactions - W150 Storage Cask

No significant chemical, galvanic or other reactions are expected for the W150 storage cask. The storage cask is constructed of reinforced concrete coated carbon steel, and austenitic stainless steel. During on-site storage the environment inside the cask is dry and warm. The cask exterior is exposed to the environment but carbon steel components are all protected with acceptable coatings.

Dissimilar materials of the storage cask in contact with one another are steel and concrete. Portland cement concrete provides an environment which protects the carbon steel from corrosion because of its alkaline nature which passivates the steel.

Contact between the canister assembly and the storage cask occurs at the external surface of the canister, the Nitronic-60 faced support rails, and the stainless steel support pipes. No reaction or galvanic coupling is expected. All exposed surfaces of the canister shell are austenitic stainless steel. Prolonged use of stainless steel in contact with stainless steel or the coated carbon steel produces no significant chemical, galvanic or other reactions.

Chemical and Galvanic Reactions - W21 Canister

The service conditions for W21 canisters include immersion in PWR fuel pools, vacuum drying (hot) conditions, on-site storage conditions (helium backfill), off-site transportation, and potentially, canister opening (water reflood) conditions. PWR spent fuel pools have relatively high concentrations of boric acid, giving the pool water a mildly acidic pH (4.0 to 4.5).

The austenitic stainless steel confinement boundary for the FuelSolutions™ W21 canister was evaluated for the effects of corrosion during dry storage and found to provide sufficient protection against failures due to corrosion. The licensee may elect to use low carbon austenitic stainless steel materials, which are not susceptible to weld sensitization for the canister's confinement boundary components as discussed in the W21 SAR Sections 3.1.1.1 as a means of providing additional protection against stress corrosion cracking.

The W21 canisters are constructed from stainless steel, electroless nickel coated carbon steel, aluminum/boron carbide (used for neutron absorber panels), and lead or DU. The aluminum/boron carbide, lead, and DU materials are all encased and seal welded in stainless steel to alleviate any effects of water immersion of the neutron absorber panel material and possible aluminum corrosion and hydrogen generation. The corrosion of stainless steels is generally extremely low, as these materials quickly form a protective passive film in the spent fuel pool environments. The electroless nickel coating on carbon steel spacer plates is for corrosion protection following canister fabrication and during the brief immersion period during fuel loading in the spent fuel pool. Electroless nickel coatings are widely used for corrosion protection and wear resistance in the electronics, petrochemical, automotive, and food industries, most often on steel and alloy steel substrates. During immersion and subsequent canister sealing, hydrogen production is relatively low. When compared to carbo-zinc or aluminum flame spray coating systems in boric acid, the hydrogen generation rate of electroless nickel is much lower than that of carbo-zinc and in the same range as aluminum flame spray. Once dried and sealed, the corrosion mechanism is removed and the nickel coating becomes inert.

The W21 canisters were evaluated to determine the potential for chemical, galvanic, or other reactions in the intended service conditions which may lead to the production of hydrogen gas, as required by NRC Bulletin 96-04. The associated hydrogen generation analysis of the W21 canister considers the effects of radiolytic generation of spent fuel pool water and corrosion of the canister materials under the most limiting service conditions. The results of the hydrogen generation analysis show that the estimated time to reach a concentration of 10% of the Lower Explosive Limit (LEL) (0.4% hydrogen by volume) in the canister cavity is approximately 4.3 hours, based on average hydrogen generation rates. For less conservative hydrogen generation rate assumptions, the time to reach a concentration of 10% of the LEL is as long as 14 hours. Therefore, monitoring the gas in the W21 canister cavity and purging (when necessary) prior to welding the top inner closure plate to the canister shell is performed to eliminate the potential for a hydrogen gas burn event and assure the safety of the public and plant personnel.

Chemical and Galvanic Reactions - W74 Canister

The service conditions for the W74 canisters include immersion in BWR fuel pools, vacuum drying (hot) conditions, on-site storage conditions (helium backfill), off-site transportation, and potentially canister opening (water reflood) conditions. BWR spent fuel pools are generally filled with air saturated, demineralized water having a neutral pH (5.6 to 7.1) and low impurities.

W74 canisters are constructed from austenitic stainless steel, electroless nickel coated carbon steel, and borated stainless steel (used for neutron absorber panels). The corrosion of austenitic stainless steels is generally extremely low, as these materials quickly form a protective passive film in the spent fuel pool environments. The EN coating on carbon steel spacer plates is for corrosion protection following canister fabrication and during the brief immersion period during fuel loading and canister sealing. Electroless nickel coatings are widely used for corrosion protection and wear resistance in the electronics, petrochemical, automotive, and food industries, most often on steel and alloy steel substrates. During immersion and subsequent canister sealing, hydrogen production is relatively low. When compared to carbo-zinc or aluminum flame spray coating systems in boric acid, the EN hydrogen generation rate of hydrogen is much lower than that of carbo-zinc and in the same range as aluminum flame spray. Once the canister is drained, dried, sealed, and backfilled with helium, the corrosion mechanism is removed and the coating is inert during storage.

The W74 canisters were evaluated to determine the potential for chemical, galvanic, or other reactions in the intended service conditions, as required by NRC Bulletin 96-04. The hydrogen generation analysis of the W74 canisters considered the effects of radiolytic generation of spent fuel pool water and corrosion of the canister materials under the most limiting service conditions.

The results of the hydrogen generation analysis show that the estimated time to reach a concentration limit of 10% of the Lower Explosive Limit (LEL) (0.4% hydrogen by volume) in the canister cavity is approximately 42 hours. Therefore, continuous monitoring for combustible gases will be performed prior to and for the duration of all welding, cutting, grinding or other spark producing activities until the canister has been drained down (loading), or until the inner closure plate has been removed from the canister (unloading). An inert gas purge of the canister cavity may also be instituted after the small volume of water is removed during loading or unloading activities as an additional measure against combustible gas buildup and to provide a means to reduce combustible gas concentration levels below 10% of the LEL, if necessary.

3.2.4 Design service life

The term of the 10CFR72 Certificate of Compliance is 20 years. According to the applicant, the W100 Transfer Cask is designed for 40 years and the W150 Storage Cask, the W21 Canister and the W74 Canister are designed for 100 years. To ensure the minimum materials design life of 20 years, the applicant has followed the applicable requirements of the ACI and ASME codes, including the regulatory requirements of 10CFR72. The applicant has used materials of known characteristics, designed, fabricated, inspected and tested under the Quality Assurance Program described in Chapter 13 of the respective SARs, For example, materials which are susceptible to corrosion are selected to be inherently resistant to corrosion (such as stainless steel) or are provided with adequate corrosion protection such as coatings. In addition, the provisions in the TS will ensure that the integrity of the materials is assured throughout the component's service life. On this basis, the staff finds that there is reasonable assurance that the structural components will maintain their integrity throughout the term of the Certificate of Compliance.

3.3 Normal Conditions

3.3.1 Storage Cask

The storage cask is evaluated using a combination of finite element analyses and hand calculations. The hand calculations are performed in accordance with the ACI 349 Code and are used for determination of the minimum reinforcement, section capacities, and tie rod anchor requirements. The storage cask concrete forces and moments, as well as the steel component stresses, due to the applied loads and load combinations are calculated using the ANSYS 5.4[16] finite element model. The finite element model is a half-symmetry model (180^{o}) of the storage cask. The model includes the concrete segments, the steel liner segments with bottom plate and top ring, the tie rods, and the top cover plate. Key features of the model also include air inlet and outlet openings and the joints between the three cask segments, including shear keys at the segment interfaces. The storage cask is evaluated for dead weight loads in both vertical orientation and horizontal orientation. The cask is evaluated for all live loads encountered during normal operation. Normal live loads include those associated with ice and snow, handling loads during up-ending/down-ending of the cask, transfer cask load during vertical transfer of the canister, and horizontal canister transfer loads. The storage cask internal forces and moments due to thermal gradients are based on temperatures obtained from the thermal analysis at 162 predetermined key points in Chapter 4 of WSNF-200. The nodal temperature distribution in the storage cask finite element model is obtained by interpolation between the key point values. The load combinations, maximum forces and stresses for the storage cask reinforced concrete and steel components are summarized respectively in Table 3.5-2 and Table 3.5-3, of WSNF-200.

The results of the storage cask structural evaluations demonstrated that the applicable design criteria are satisfied for all normal operating conditions. As shown in Table 3.5-2, all design margins for reinforced concrete components are above +0.4. The design margins for storage cask structural steel components are generally very high except for the tie-rods. The tie-rod design margin for the load combination that included thermal stress is +0.08 as shown in Table 3.5-3. The canister support pipes at the bottom of the cavity are evaluated using the bounding canister weight of 82,000 lb. and increased by 15%. The support pipe maximum stress of 14.6 ksi is lower than the corresponding normal allowable of 18.8 ksi for type 304 stainless steel pipes. Thus, the supporting pipes are adequate to support the heaviest loaded canister.

The storage cask has been shown to have sufficient cavity clearance to allow the canister shell to expand freely for normal and off-normal conditions. The nominal axial and radial clearances are 0.75-in and 0.4-in, respectively. The calculated differential thermal expansion of the canister and the storage cask using the worst temperature differential between the two components is 0.67-in axially and 0.12-in radially. Thus, the storage cask provides adequate cavity clearances to allow free thermal expansion of the canister under normal and off-normal temperature conditions.

3.3.2 Transfer Cask

The transfer cask shell stresses due to dead weight while oriented vertically are calculated using the ANSYS axisymmetric finite element model. The axisymmetric model includes the transfer cask ram access cover, top flange, top cover, inner liner, structural shell, gamma shield, bottom flange, and bottom cover. The cask neutron shields are not modeled but their weights are included in the model by adjusting the element density. Gap elements which can transfer only compressive forces are used to model the nonlinear interface between the gamma shield (lead) and the cask shells.

The transfer cask stresses due to dead weight, while oriented horizontally and secured on the transfer skid and trailer, are determined by using the two half symmetry finite element models. The two finite element models represent the top and bottom halves of the transfer cask. The top end half model includes the transfer cask ram access cover, top cover, inner liner, structural shell, upper trunnion, and associated welds. The bottom half model includes the transfer cask bottom cover, bottom flange, inner liner, structural shell, and lower trunnion. In addition to dead weight stresses, the stresses in the transfer cask shell due to lifting and handling operations are also evaluated by using the two half-symmetry finite element models.

3.3.2.1 Dead weight

Two dead weight loading conditions are evaluated: (1) Vertical dead weight when the cask is resting vertically on the bottom cover, and (2) Horizontal dead weight when cask is secured horizontally on transfer skid/trailer by the upper and lower trunnions. The results of the dead weight static analysis showed that the transfer cask shell stresses are controlled by the horizontal dead weight loading condition. Therefore, only the stresses calculated from horizontal dead weight are used in the cask load combinations. The stress results from the static analyses are summarized in Table 3.5-4 of WSNF-200. The maximum primary membrane plus bending stress intensities in the cask liner and structural shell are 4.2 ksi and 7.1 ksi, respectively. The maximum peak stress intensity in the transfer cask due to horizontal dead weight is 10.7 ksi, occurring in the welds at the junction of the upper trunnion and structural shell. These maximum stresses are included in the transfer cask load combination evaluation in Section 3.3.2.5.

3.3.2.2 Lifting and Handling Loads

The transfer cask is evaluated for all loads associated with normal lifting and handling conditions. The controlling lifting and handling conditions considered in the structural evaluation include the following: (1) Vertical lifting, (2) Down-ending/up-ending operations, (3) On-site transportation, (4) Horizontal canister transfer, and (5) Vertical canister transfer. The transfer cask shell stresses under normal lifting and handling loading conditions are calculated by linear elastic static analyses using the transfer cask upper and lower half-symmetry finite element models. The resulting stresses are summarized in Table 3.5-6 of WSNF-200.

3.3.2.3 Normal Thermal Loads

The transfer cask is evaluated for the most limiting thermal gradients occurring during vertical handling inside the fuel building and during the normal ambient conditions. The transfer cask thermal analysis is performed by using the axisymmetric finite element model described in Section 3.3.2. Temperature dependent material properties are used. The normal condition thermal stress intensities are summarized in Table 3.5-7 of WSNF-200 and are included in the transfer cask load combinations. The differential thermal expansion of the canister and the transfer cask is calculated by using the largest temperature differential between the two components. The minimum axial and radial clearance between the transfer cask cavity and canister are 0.08-in and 0.42-in, respectively. Thus, the transfer cask allows free radial and axial thermal expansions of the canister shell under normal thermal conditions.

3.3.2.4 Pressure and Fatigue Evaluation

The transfer cask is not a pressure vessel. The only component that experiences a pressure load is the neutron shield jacket designed for a bounding pressure of 50 psig. However, the stress in the neutron shield jacket due to this pressure load (e.g., 8.5 ksi) is well below the allowable values (e.g., 20 ksi) of the stainless steel neutron shield jacket material.

The applicant provided an evaluation to show that detailed fatigue analysis is not required because the transfer cask met the rules of ASME, Section III, NC-3219.2, Condition B.

3.3.2.5 Transfer Cask Normal Condition Load Combinations

The transfer cask load combination stresses are conservatively calculated by combining the maximum transfer cask component stress intensities due to each individual load condition irrespective of sign and location. The resulting stress intensities for the normal condition load combination are reported in Table 3.5-8 of WSNF-200. Since positive design margins are shown for all transfer cask components in Table 3.5-8, the transfer cask has met the design normal load conditions.

3.3.3 Canisters

The FuelSolutions™ Storage System has two different types of canisters, designated as W21 and W74 canisters, based on how many spent fuel assemblies each canister can hold. Within each type of canister there are two different classes, “M” and “T”, differing in materials of constructions used for the canister shell and basket assembly. The technical basis for the design, fabrication, and operation of the canisters are provided in two Safety Analysis Reports: (1) FuelSolutions™ W21 Canister Storage SAR (WSNF-201) and (2) FuelSolutions™ W74 Canister Storage SAR (WSNF-203). The structural evaluations for each canister are provided in the respective (Chapter 3) SARs.

The structural evaluation of the canister shell and basket assemblies is performed using a combination of finite element analyses and classical closed form solutions. The evaluations performed for all normal, off-normal, and accident conditions and loading combinations, along with the analytical methods employed, are summarized in the following tables of the respective SAR documents:

- Table 3.1-3, WSNF-201, for W21 canister shell assemblies.
- Table 3.1-4, WSNF-201, for W21 canister basket assemblies.
- Table 3.1-2, WSNF-203, for W74 canister shell assemblies.
- Table 3.1-3, WSNF-203, for W74 canister basket assemblies.

Under normal conditions, canisters are evaluated for all loads occurring during normal or routine operation. Loads considered in the structural evaluation include normal temperature, normal internal pressure, dead weight, and normal handling. The results of the evaluation demonstrate that the canisters and their internal basket structures can withstand the effects of normal condition loads without affecting structural safety functions and remain in compliance with the applicable acceptance criteria.

3.3.3.1 Normal Temperature Loads

Normal Hot Conditions

The canisters are evaluated both in the transfer cask and in the storage cask for the steady-state thermal gradient resulting from normal, off-normal, and accident conditions and the design-basis heat load. Differential thermal expansions between the canister basket assembly components and the canister shell assembly are evaluated. The results of this evaluation demonstrate that the canister basket assemblies expand freely within the canister shell under normal, off-normal, and accident conditions. For the normal hot condition, the canister shell assembly is evaluated for a bounding internal pressure of 10.0 psig which is derived by assuming 1% of the fuel rods fail under normal conditions of dry storage. The canister basket assembly is not affected by canister internal pressure. Thermal stresses in the canister shell assembly and basket assembly are calculated using linear elastic finite element analysis and are shown to be lower than the Service Level A allowable primary plus secondary stress intensity. Fatigue evaluations are performed to show that fatigue is not a concern for the shell assembly or the basket assembly structural components.

Extreme Cold Temperature

The extreme cold temperature condition is defined as steady-state ambient temperature of 0^o F with no fuel decay heat and no insolation. This assumed cold condition will result in a uniform temperature of 0^o F throughout the canister. For this condition, the effects of differential thermal expansion between dissimilar materials and the potential for brittle fracture and freezing of liquids are considered. The evaluation showed no significant stresses resulting from differential thermal expansion.

Brittle fracture of the canister components for a lowest service temperature of 0^o F is addressed in Section 3.1.2.3 of the respective canister SARs. Since the canister does not contain any liquids, there is no potential damage due to freezing of liquids.

3.3.3.2 Internal Pressure

The canister shell assembly is evaluated for the normal transfer and storage internal pressure load of 10 psig and the bounding internal pressure load of 30 psig associated with the canister draining operation. After installation of the inner closure plate, a compressed gas pressure of 30 psig is applied to the canister cavity to speed the water draining process during canister closure operation. A uniform pressure load of 30 psig is applied to the inner surface of the inner closure plate and the cylindrical shell. In addition, a 1g vertical acceleration is applied to the canister shell finite element model to account for the self weight of the canister shell assembly. The analysis results showed that maximum stress intensities in the canister shell assembly due to the water drainage pressure loading are below the Service Level A allowable stress intensities.

3.3.3.3 Dead Weight Load

The canisters are transferred either in the vertical or horizontal orientations but stored only in the vertical orientation. Thus, canister shell and basket assembly components must be evaluated for both vertical and horizontal dead weight loading conditions. For vertical dead load condition, the bottom end of the canister rests on either the transfer cask bottom cover plate or the storage cask canister support tubes. The support conditions are nearly uniform over the bottom surface of the canister shell assembly. For horizontal dead load condition, the canister is supported by two rails in the transfer cask or storage cask. The centerline of the rails in both the transfer cask and storage cask are located at 22.5^o on either side of the centerline of the canister. The dead weight structural evaluations of the canister are performed by a combination of finite element analyses and hand calculations. The results showed that the maximum stress intensities in the canister shell and basket assembly components are less than the corresponding Service Level A allowable stress intensities with large margins.

3.3.3.4 Normal Handling

The FuelSolutions™ Storage System is designed to transfer the canister both vertically and horizontally between the transfer cask, storage cask, and transportation cask. Three normal handling load conditions are evaluated: (1) vertical canister transfer handling, (2) horizontal canister transfer handling, and (3) on-site transport. Vertical canister transfer handling loads are equal to canister dead weight plus an additional 15% increase to account for dynamic effects due to crane hoist motion. Horizontal canister transfer loads are equal to a hydraulic ram force of 45 kips applied to the top or bottom end of the canister. The hydraulic ram force is derived by assuming a coefficient of friction between the canister shell and the support rails to be 0.5 and a bounding canister weight of 90,000 pounds. On-site transport shock and vibration loads are small. Design-basis on-site transport shock and vibration loads of 0.6g vertical, 0.3g longitudinal, and 0.2g lateral are conservatively used to provide a bounding structural evaluation. The structural evaluation results showed that the maximum stresses in the canister shell assembly and the basket assembly components due to these handling loads are less than the Service Level A allowable stresses.

3.3.3.5 Load Combinations

As stated in Section 3.1.2.3, the canister shell and basket assembly components are designed for the loading combinations shown in Table 2.3-1 of the respective canister SARs. Stresses due to these normal load combinations are conservatively calculated by adding the maximum stresses due to the individual load conditions absolutely and irrespective of location. An exception is for the canister shell load combination evaluation in which all loads, except the horizontal dead weight, are superimposed on the canister finite element model. Horizontal dead weight stresses are then added to the stresses from the finite element model analysis absolutely and irrespective of location. The results show that the maximum stresses due to all normal load combinations are less than the corresponding Service Level A allowable stresses. The governing load combination stress results for normal load combinations are summarized in the following Tables in the respective SARs:

- W21 canister shell assembly - Table 3.5-5, WSNF-201
- W21 basket assembly - Table 3.5-6, WSNF-201
- W74 canister shell assembly - Table 3.5-6, WSNF-203
- W74 basket assembly - Table 3.5-7, WSNF-203

3.4 Off-Normal Conditions

3.4.1 Storage and Transfer Casks

3.4.1.1 Off-Normal Temperature

For the storage cask, the normal thermal loads used in the storage cask evaluations are conservatively selected to bound both normal and off-normal conditions. Therefore, the off-normal thermal condition is bounded by the normal thermal load conditions.

For the transfer cask, off-normal thermal stresses are determined for the following two loading conditions:

1. Off-normal Cold: Ambient temperature of -40^{o} F, no insolation.

2. Off-Normal Hot: Ambient temperature of 125^{o} F, full insolation.

The resulting transfer cask general thermal stress intensities for the bounding off-normal thermal load condition are shown in Table 3.6-1 of WSNF-200. The thermal stresses are combined with other stresses in the load combinations.

3.4.1.2 Wind Load

The design basis wind load for the storage cask is included in the off-normal load combination. The wind load is conservatively based on the design tornado wind load (e.g., 356 psf). The wind load and load combinations are presented in Table 3.6-2 of WSNF-200.

The design basis wind load for the transfer cask is 74 psf (0.514 psi). The transfer cask stresses due to the design basis wind load are calculated by scaling the stresses calculated for tornado wind loads by the ratio of the applied loads. The wind pressure is assumed to act either normal to the top or bottom ends of the cask or normal to the side of the cask. The maximum stresses in the transfer cask due to the design basis wind load are summarized in Table 3.6-1 of WSNF-200.

3.4.1.3 Cask Misalignment or Interference

The effects on the storage cask due to the canister misalignment or interference during horizontal transfer from the transfer cask to the storage cask are evaluated. The misalignment load, however, is limited to 70,000 pounds which is the maximum force can be exerted by the hydraulic ram. The load is less than the live load used in the evaluations under normal conditions (i.e., transfer cask bounding weight of 200,000 lbs. is applied as a vertical load around the top end of the storage cask). Thus, for the storage cask, the misalignment condition is bounded by the normal load condition. No additional analysis is required. The effects on the transfer cask due to canister misalignment during horizontal transfer from the storage cask to the transfer cask are evaluated. Similar to the storage cask, the load is limited by the maximum hydraulic ram force of 70,000 pounds. The stresses in the transfer cask due to the off-normal load are evaluated by hand calculations.

The transfer cask lower trunnions are evaluated for the bounding off-normal load of 90,000 pounds applied by the cask restraints. The lower trunnion stresses are calculated by taking a ratio of the stresses calculated for the transfer cask down-ending operation during normal handling conditions. The ratio is based on the down-ending design load of 200 kips (100 kips per trunnion) versus the bounding misalignment load of 90 kips (45 kips per trunnion). The resulting stresses are shown in Table 3.6-1 of WSNF-200.

3.4.1.4 Off-Normal Load Combinations

Applicable off-normal load combinations for the storage cask and transfer cask are evaluated as discussed in Section 3.1.2.3. The resulting stresses are presented in Table 3.6-2 through Table 3.6-4 of the SAR. All cask components are shown to be adequate to withstand the off-normal design load combinations and have significant design margins.

3.4.2 Canisters (W21 and W74)

The W21 and W74 canisters are evaluated for all credible and significant design basis events resulting from off-normal operation conditions. The results of the evaluations performed demonstrate that the canisters can withstand the effects of off-normal events without affecting structural safety function and remain in compliance with the applicable acceptance criteria.

3.4.2.1 Off-Normal Ambient Conditions

The storage cask system may be subjected to extreme ambient conditions for short periods of time. In order to bound the expected temperatures, the steady-state temperatures of the canister and fuel cladding are calculated for a maximum 125° F ambient temperature with maximum insolation and for a minimum -40° F ambient without insolation. The design basis heat loads for the canisters are used for this analysis. The off-normal thermal and handling conditions are evaluated for Service Level B stress limits. However, the canister reflood case is a condition where the canister is in the process of being removed from service and the Service Level C stress limits are applicable. Consequently, the short-term thermal stresses resulting from the reflood operation are not combined with other off-normal or accident load conditions. The canister shell assembly is evaluated for a bounding off-normal internal pressure load of 16 psig. Differential thermal expansion of the canister components is evaluated under all normal, off-normal, and accident conditions to assure that the basket assembly expands freely within the shell assembly and the shell assembly expands freely within the storage cask and/or transfer cask under all thermal conditions.

3.4.2.2 Off-Normal Internal Pressure

In addition to the backfill helium, the applicant assumes a concurrent non-mechanistic failure of 10% of the fuel rods with complete release of their fill gas and 30% of their fission gasses into the canister cavity. At the extreme off-normal condition canister gas temperature, the bounding internal pressure is 12.3 psig for W74 canisters and 15.9 psig for W21 canisters. The bounding 16.0 psig internal pressure is used in the canister shell assembly stress analysis. The resulting stresses for this loading condition are provided in Table 3.6.1 of the respective SARs.

3.4.2.3 Cask Misalignment

The FuelSolutions™ Storage System is evaluated for a maximum hydraulic ram pushing load of 70 kips and a maximum hydraulic ram pulling load of 50 kips resulting from cask misalignment or interference during horizontal canister transfer. The canister shell maximum primary stress intensities resulting from a cask misalignment ram pull force applied to either the top or bottom ends of the canister are summarized in Table 3.6.1 of the respective SARs.

3.4.2.4 Canister Opening/Reflood

The canisters are evaluated for the effects of reflooding the canister after the canister cavity is drained and dried. This could occur prior to or after dry storage. The maximum internal pressure for this condition is estimated to be 100 psig. The stresses in the canister shell due to reflood internal pressure plus vertical dead weight loading is evaluated on an elastic basis using an axisymmetric finite element model. The dead weight of the canister basket assembly and spent nuclear fuels is modeled as a uniform pressure load over the inner surface of the canister shell bottom closure plate. A bounding weight of 60 kips is conservatively used for the basket and the fuel. A bounding reflood internal pressure load of 125 psi is applied to the inner surfaces of the canister shell cavity. The maximum canister shell stresses resulting from reflood internal pressure are summarized in Table 3.6.1 of the respective SARs. The results show that the maximum stresses in the canister shell assembly due to reflood internal pressure of 125 psi are less than the allowable stress intensities.

3.4.2.5 Off-Normal Load Combinations

The load combinations for the canisters include normal loads, and off-normal loads as described in Section 3.1.2.3.. The evaluation of load combinations is simplified by identifying the bounding load combinations. The canister shell load combination results for off-normal conditions are reported in Table 3.6-2 of the SARs. The results show that all stresses are within code allowable values. The basket assembly components are not affected by internal pressure loading or cask misalignment loading. Therefore, the basket assembly off-normal load combinations are identical to basket normal load combinations. No new evaluation is required for the basket assembly off-normal load combinations.

3.5 Accident Conditions and Natural Phenomena Events

3.5.1 Storage and Transfer Casks

The transfer cask and the storage cask are evaluated for a range of postulated accident and natural phenomena events. Postulated accidents include handling drops, fire, explosions, blocked inlet and outlet vents, and loss of transfer cask neutron shielding. Extreme Natural phenomena events include flood, tornado and tornado missiles, and the design basis earthquake. As discussed in Section 3.1.2.3, the accident and natural phenomena event loads are combined together with the normal loads in accordance with the load combinations specified in NUREG-1536.

3.5.1.1 Handling Drop Accidents

Three handling drop accident events are postulated: (1) a storage cask end drop, (2) storage cask tip-over during up- or down-ending of the cask while secured to the J-skid, and (3) a transfer cask side drop.

Storage Cask End Drop

The methodology used to evaluate the storage cask end drop loads is based on the approach presented in NUREG/CR-6608. The drop analysis is performed using the nonlinear transient dynamic finite element code LS-DYNA. The finite element model includes the storage cask and canister, concrete pad, and the underlying soils. The applicant performed benchmark analysis to validate the finite element analytical model and the LS-DYNA code by comparing computer analysis results with the LLNL billet test measurements presented in NUREG/CR-6608[11].

There are five cases evaluated for the end drop of a storage cask. All cases are free drops from a height of 36-in and impact flat on the cask’s bottom end. The first three cases are used to evaluate the maximum g-loads on the cask and canisters based on three different pad thickness and soil stiffness conditions. The three pad and soil combinations are summarized in Table 3.7-2 of WSNF-200. Any of these combinations are acceptable. However, a stiffer pad and soil combinations cannot be used unless a site-specific analysis is performed to demonstrate that the drop load is bounded by the design drop load. The fourth case is performed to show that the pad modeled in the analysis is of adequate size to eliminate edge effects. The pad and soil radii of the model are both increased by 50%. Drop analysis is performed for the most critical case of the 3'-0" storage pad supported by the 10 ksi soil. The analysis showed that the difference in g- loads between the larger pad and the base model pad are insignificant. Therefore, it can be concluded that the pad size used in the base model is of sufficient size to eliminate edge effects of the concrete pad. The canister is supported by sixteen stainless steel pipes that are oriented radially and welded to the storage cask bottom liner plate. The functions of the canister support pipes include providing spacing between the bottom of the canister and storage cask cavity (to allow air flow for passive convective cooling) and limiting the canister g-loads during a postulated end drop. For the fifth case, the finite element model is used to evaluate the potential for a canister to "bottom out" due to the complete collapse of the support pipes. In this analysis, the heaviest canister assembly and the lowest support pipe stiffness are assumed. The result of this analysis showed that the support pipe do not bottom out. The maximum displacement of the support pipes for this drop condition is 2.64-in versus an available pipe crush distance of 3.36-in.

The peak rigid-body accelerations in the storage cask, canister shell, and canister basket assembly for each drop case are summarized in Table 3.7-3 of WSNF-200. As shown in the table, the maximum peak rigid-body accelerations in the storage cask, canister shell, and basket assembly are 57g, 28g, and 26g, respectively. In general, the maximum peak rigid-body acceleration occurs for the case with the 3-ft thick pad on the 10-ksi soil. But the canister shell and basket assembly accelerations are controlled by crushing of the support pipes and thus do not vary significantly with variations in pad thickness and soil stiffness.

Based on the cask body lowest axial vibration frequency of 162 Hz, the elastic DLF is determined to be 1.5 for a triangular pulse. Thus, the equivalent static loading for the cask body is 57g x 1.5 = 85.5g. An acceleration of 89g is conservatively applied to the storage cask body in the ANSYS model for stress analysis. The storage cask top cover plate basic frequency is only 21.8 Hz as determined based on simply supported plate with uniformly distributed loads. Thus, the cover plate is soft compared to the cask body and, as the result, the DLF for the cover plate is lower than 1.0. An equivalent static loading of 59g (assuming DLF=1.0) is conservatively used for the cover plate design. Since the cover plate is included in the ANSYS model and since an equivalent static load of 89g is applied to the ANSYS model, the equivalent density of the cover plate is adjusted for the end drop condition by the factor of 59/89 = 0.66.

The results of the storage cask stress analysis for the postulated end drop accident are presented in Table 3.7-5 for reinforced concrete components and Table 3.7-6 for steel components in WSNF-200.

Storage Cask Tip-Over

The only tip-over accident postulated to occur is that during up- or down-ending operations of the storage cask while it is secured to the J-skid and over the impact limiter (which is recessed into the pad) in the storage cask handling area of the ISFSI. The cask tip-over evaluation is performed in two steps: (1) the rigid body dynamic response is obtained by the energy balancing method to determine the peak g-loading, and (2) the peak rigid body g-loading amplified by the DLF is used to evaluate the storage cask using the ANSYS model and finite element analysis.

The energy balancing method is based on the assumption that the kinetic energy of the falling cask is balanced with the strain energy of the impact limiter crush. Two storage cask tip-over cases are evaluated to assure that all possible conditions are bounded:

(1) Impact based on cold temperature (0°F) properties of polyurethane foam that produces the maximum cask impact g-load to be used in the cask structural analysis.

(2) Impact based on hot temperature (100°F) properties of polyurethane foam that results in impact limiter maximum crush depth to assure that the foam is not subjected to excessive strains (i.e., bottom out).

The maximum dynamic acceleration is determined to be 21.9g. The elastic DLF for the cask tip-over event is determined to be 1.22. Thus, the equivalent static load for structural analysis is 1.22 x 21.9g=26.7g. For conservatism, a static acceleration of 28g is used in the cask structural evaluation and applied along the entire length of the cask. The storage cask evaluation is performed using the finite element model and the ANSYS program. The results of the storage cask tip-over stress analysis are presented in Table 3.7-5 for reinforce concrete (RC) components and Table 3.7-6 for steel components in WSNF-200.

A centrifugal force acting on the canister during the postulated tip-over event will cause the canister to slide out of the cask and must be restrained by the top cover plate and the bolts. The centrifugal force of the canister at the maximum angular velocity is calculated to be 56.5 kips and the centrifugal force of the cask cover's own mass is 6.5 kips. Modeling the top cover plate as having a fixed edge at the bolt circle, the maximum bending stress in the cover plate is determined to be 4.2 ksi, which is well below the accident allowable stress of 121.7 ksi. The bolt tension, including the prying force due to edge moments, is 13.7 kips. The bolt shear due to the 28g top cover load is calculated to be 14 kips per bolt. Using the AISC interaction equation for bolts, the tensile allowable for bolts in the presence of this shear load is reduced from 82.8 kips to 77.4 kips per bolt. There are twelve (12) bolts for the cover plate and the total centrifugal force (i.e., 56.5 + 6.5=63 kips) is even less than the allowable tensile force of one bolt. Thus, the centrifugal force of the canister acting on the storage cask top cover plate during the postulated cask tip-over is not consequential for the bolts.

Transfer Cask Side Drop

The transfer cask 72-in side drop is postulated to occur at the canister horizontal transfer pad outside the fuel building. The transfer pad and the elastic modulus of the underlying soil are defined in the TS. The site-specific pad design should be evaluated by the general licensee to confirm that the design basis deceleration loads for the transfer cask and canister are not exceeded.

Determination of the drop g-loads for the W100 transfer cask side drop is performed using LS-DYNA finite element model that includes the transfer cask, loaded canister assembly, reinforced concrete pad, and underlying soil. The transfer cask model includes the main structural components (i.e., inner shell, outer shell, top and bottom flanges and cover plates) and the lead gamma shield. Linear elastic properties are used for all structural materials. The lead is modeled with the nonlinear elastic-plastic material properties presented in Table 3.3-8 of WSNF-200. The transfer cask liquid neutron shields and jackets are included in the model for its weight only. The neutron shield is assumed to fail in the side drop event and is thus conservatively neglected in the structural model. The canister is modeled as a solid cylinder with a mass density and modulus of elasticity adjusted to provide the weight and a transverse natural vibration frequency representative of the canister assembly.

The maximum transfer cask peak rigid-body acceleration resulting from the 72-in side drop is 46g. Based on the cask lowest natural frequency of 82 Hz and the drop duration of 22.5 msec, the DLF for a half-sine wave impulse loading is 1.3. Therefore, the amplified acceleration is equal to 1.3 x 46g=60g. The stress analysis of the transfer cask is based on an equivalent static loading of 60g. The maximum primary stress intensities for a 72-in side drop are presented in Table 3.7-8 of the WSNF-200.

3.5.1.2 Explosive Over-Pressure

Since no potentially explosive materials are stored in close proximity to the ISFSI, the applicant assumes that the design basis tornado wind load (356 psf) envelopes the explosive over-pressure loads on the storage and transfer casks. Therefore, no analysis is performed for explosive over-pressure loading. However, the licensee should compare site-specific hazards to the design basis tornado wind loading to assure that the explosive over-pressure loads are bounded by the tornado wind load.

3.5.1.3 Flood

The lateral drag force resulting from the design basis flood is estimated to be 47.1 kips. The net buoyant weight of a loaded storage cask is 182.5 kips (i.e., the weight of the dry loaded storage cask less the weight of water displaced by the storage cask and canister). The friction force resisting sliding is equal to the product of the net buoyant weight and the static friction coefficient between the bottom surface of the cask and the ISFSI pad. The applicant assumes a lower bound static coefficient of friction of 0.3. The resisting force is thus calculated to be 0.3 x 182.5=54.75 kips. The factor of safety against sliding due to the design basis flood is calculated as 54.75/47.1=1.16.

Overturning stability of the storage cask during the design basis flood is evaluated by assuming that the cask is pinned at its outer bottom edge on the side opposite to the direction of water flow. The maximum overturning moment resulting from the drag force acting at the mid-height of the cask is 47.1 x 9.59=451.7 kip-ft. The restoring moment based on the buoyant weight of the storage cask is 182.5 x 5.75=1049 kip-ft. Thus, the resulting factor of safety against a cask overturning due to the design basis flood is 1049/451.7=2.3. Based on the analysis results it can be concluded that the storage cask remains upright and that it will not slide or be overturned by the lateral drag force of the design basis flood water.

3.5.1.4 Fire

The applicant performed thermal transient analysis of the storage cask for the postulated accident fire event. It concludes that the concrete exterior surface exceeds the ACI short term local allowable temperature for a depth of less than two inches. The concrete wall thickness is approximately 30.5-in. Therefore, it is postulated that there will be some local spalling of exterior surface concrete due to the excessive heat of the postulated fire accident but the remaining concrete wall will still permit the storage cask to perform its shielding function and allow unloading of the cask for detailed inspection.

For the transfer cask, the only structural consequence of the fire event must be evaluated is its effect on the cask material properties. The thermal stresses are secondary stresses and do not require evaluation under the postulated accident events. The applicant performed thermal analysis to show that all transfer cask components, except for the neutron shield jacket, remain well within allowable temperatures for their materials of construction. The loss of neutron shield will not affect the cask structural performance. The effects of a postulated fire on the lifting trunnions are also considered. The trunnion outer surfaces are directly exposed to the fire and consequently their temperatures are assumed to be close to that of the neutron shield jacket, i.e., 1400^{o} F. However, the allowable stresses of the stainless steel will not change significantly after exposure to this temperature for only a short period of time (fire duration is limited by the TS to be only five minutes). Thus, the transfer cask would continue to perform its functions to allow unloading and inspection following the fire.

3.5.1.5 Tornado and Tornado Missile

The storage cask and transfer cask are designed to withstand loads associated with tornados at an ISFSI located anywhere in the contiguous United States. The structural evaluation of the storage cask and transfer cask includes local damage, stability, and stresses due to tornado winds and tornado generated missiles.

Storage Cask

The storage cask is evaluated for the effects of wind-driven tornado missiles to assure that there is no significant local damage that allows the missile to perforate or pass through the storage cask and potentially damage the canister. The evaluation includes two wind-driven tornado missiles: (1) a 275-pound, eight inches diameter armor piercing artillery shell (simulated by an 8-in diameter steel cylinder), and (2) a one inch diameter solid steel sphere.

Local damage of the storage cask side wall due to horizontal impact of the armor piercing missile is based on the modified National Defense Research Committee formula as presented by ASCE 7. The depth of penetration due to the 275-pound 8-in diameter, armor piercing artillery shell with a horizontal impact velocity of 126 mph (185 feet per second) is calculated to be 5.4-in. This value is much less than the storage cask side wall thickness of 30.75-in. Thus, the cask wall is not penetrated. Local damage of the storage cask top steel cover due to vertical impact of the armor piercing missile is evaluated using the Ballistic Research Laboratory (BRL) formula. The thickness of the steel plate required to prevent perforation is 0.65-in. Since the thickness of the storage cask steel cover plate is 1.0-in, the armor piercing missile will not perforate the storage cask steel cover plate. A direct impact of the one inch solid sphere in the storage cask outlet vent is evaluated. The outlet vents are designed with a step so that the canister is protected from direct impact by the one inch diameter small missile. However, even if the small missile strikes the canister directly, the canister shell thickness is sufficient to prevent missile perforation.

The storage cask overturning stability due to a 4,000 pound automobile tornado missile is evaluated using the principals of conservation of energy. The force from the missile impact is assumed to act at the top of the storage cask and the cask is assumed to rotate about its bottom edge. The angular velocity of the storage cask after impact by the 4,000 pound tornado missile at a horizontal impact velocity of 126 mph is calculated to be 0.264 rad/sec. The kinetic energy of the cask due to this rotational velocity is 7.0×10^{5} in-lbs. By equating rotational kinetic energy of the cask to the potential energy of the cask for a given rotation about the bottom edge, the storage cask is found to rotate only 2° from the vertical. A rotation of 29.9° from the vertical is required to bring the cask C.G. outside the edge for overturning. Therefore, a tornado-generated large missile will not cause the cask to overturn.

The combined effects of the tornado wind and missile loads on the overturning stability of the storage cask are evaluated in accordance with NUREG-0800. The maximum lateral pressure load due to the tornado wind load (i.e., 356 psf) is applied to the storage cask with an imposed rotation of 2.0° due to the tornado missile load. The overturning moment due to tornado wind is $9.35x10^6$ in-lbs. The restoring moment for the storage cask is $19.15x10^6$ in-lbs. Therefore, the factor of safety against cask overturnings due to combined effects of the tornado wind and missile load is greater than 2.0.

The storage cask is also evaluated for sliding using the conservation of momentum. Equating the missile momentum to the cask linear momentum after impact, the cask velocity is 29.6 in/sec. Assuming the lower bound friction coefficient of 0.3 and lower bound storage cask weight of 300,000 pounds, the friction force is 90,000 pounds. The wind force is 78,500 pounds which is calculated by the wind pressure of 356 psf times the cask projected area of 220.5 ft^2. The net force resisting sliding is thus equal to 90,000-78,500= 11,500 pounds. With this net resisting force and an initial velocity of 29.6 in/sec, the cask slides a maximum of 32.9-in before stopping. The evaluation is based on conservative assumptions and the maximum sliding distance is still less than the cask spacing of approximately 4 feet.

Storage cask stresses due to tornado wind loads are obtained by the finite element analysis. The wind and tornado missile loads are applied simultaneously with the wind loads applied as a uniform pressure and the automobile missile impact load spread over a 3-ft x 3-ft area at the top of the cask. The stresses are presented in Table 3.7-5 and Table 3.7-6 of WSNF-200.

Transfer Cask

The evaluation for the tornado missile impact for the transfer cask is similar to the evaluations performed for the storage cask. The transfer cask is evaluated for local damage due to impact of the armor piercing missile using the same BRL formula. The plate thickness required to prevent missile perforation is shown to be 0.84-in. The thickness of the transfer cask shell and the end plate are 1.5-in and 3.0-in, respectively. Thus, the transfer cask will adequately protect the canister and its contents from the small and intermediate tornado missiles.

The stability of the transfer cask with the 4,000 pounds tornado missile striking in the most vulnerable position is evaluated to show that the sliding and /or overturning of the transfer cask does not occur. The force from the missile impact is assumed to act at the middle of the transfer cask. The transfer cask, which is attached to the transfer skid and trailer, is assumed to rotate about the outside edge of the transfer trailer tires. Based on the principal of energy conservation, the cask is found to rotate only 1.7° from vertical. This angle is much less than the 31.3° rotation required to overturn the transfer cask. The combined effects of the tornado wind and missile loads on the overturning stability of the transfer cask are also evaluated in accordance with NUREG-0800. It was shown that the restoring moment for the transfer cask is larger than the overturning moment due to the tornado wind load and that the factor of safety against the overturning of the transfer cask is approximately equal to 1.7.

The transfer cask stress analysis due to missile impact and tornado wind loads are evaluated using a combination of closed form hand calculations and finite element analysis. The missile impact load is applied as a uniform pressure in combination with the tornado wind pressure load. The analysis considers impact conditions that the missile impacts on transfer cask top end, bottom end, and the side. The results of the transfer cask stress analysis showed that the stresses in the transfer cask are within the ASME Code allowable stresses.

3.5.1.6 Earthquake

The storage cask and transfer cask effects due to DBE and all associated load combinations are evaluated for all possible modes of failure. The DBE peak accelerations are defined as 0.25g in two orthogonal horizontal directions and 0.25g in the vertical direction. The failure modes evaluated include overturning stability, sliding stability, and structural failure of the casks due to internal forces and moments induced by a DBE earthquake. The overturning analysis is conservatively performed using static balance of the overturning and restoring moments. Peak accelerations in all three directions are assumed to act simultaneously. The overturning stability and sliding stability are controlled by the long storage cask in the free standing vertical storage condition. The factor of safety against cask overturning due to DBE is calculated to be 1.22. The maximum sliding distance due to DBE is 0.3-in based on a net sliding friction coefficient of 0.3 between the cask bottom and the storage pad. The sliding distance is insignificant and would not cause an impact between adjacent casks.

The stresses in the storage cask due to the design basis earthquake are calculated by the finite element analysis. Bounding DBE loads of 0.82g horizontally and 0.51g vertically are assumed for the stress analysis (the actual peak design values are 0.25g vertically and 0.35g horizontally). The storage cask is assumed to be supported at the bottom. The storage cask stress analysis results are presented in Table 3.7-5 and Table 3.7-6 of WSNF-200.

The transfer cask is evaluated for the DBE while supported horizontally at the upper and lower trunnions by the transfer skid and trailer. Vertical loads on the transfer cask are reacted at all four trunnions, longitudinal loads are reacted by the two lower trunnions, and transverse loads are reacted by the upper and lower trunnions on one side of the cask. The transfer cask is conservatively evaluated for the combined effects of a 1.0g horizontal acceleration in the transverse direction, a 1.0g horizontal acceleration in the longitudinal direction, and a 0.6g in the vertical direction. The analysis results showed that the stresses in the transfer cask are low as compared to the ASME allowable stresses. Thus, the ability of the transfer cask to maintain criticality control and provide radiation shielding is not affected by the DBE event.

3.5.1.7 Fully Blocked Inlet and Outlet Vents

The blockage of the vents results in increased temperatures inside the storage cask due to the loss of air flow. This results in higher inside surface temperatures and larger thermal gradients in the storage cask wall section. Short term allowable concrete temperature for the maximum heat load would be reached in 41 hours after termination of the air flow. Thus, temperature distribution after 41 hours of the transient heat-up is used as the bounding temperatures for the thermal accident analysis of the storage cask. This is conservative because the TSs called for daily (once every 24 hours) inspection of the vents and thus protecting the cask from exceeding the assumed accident condition temperature distribution.

The thermal accident loading is only combined with the dead and live loads of the cask and the load combination results are shown in Table 3.7-5 and Table 3.7-6 of WSNF-200.

3.5.1.8 Loss of Transfer Cask Neutron Shielding

The neutron shield jacket thickness is not sufficient to prevent perforation as the consequence of tornado missile impact by the armor piercing shell. However, the lost of liquid neutron shielding material do not have any significant effects on cask stresses. The loss of neutron shielding, however, must be included in the thermal and shielding evaluations of the cask.

3.5.1.9 Load Combinations

The accident condition load combinations for the storage cask and the transfer cask are discussed in Section 3.1.2.3 of the SER. The load combinations included both normal loads and accident loads. The load combinations for the storage cask reinforced concrete sections and steel components are provided in Table 3.7-5 and Table 3.7-6, respectively in WSNF-200. The transfer cask maximum stresses due to the individual accidents and load combinations are presented in Table 3.7-8 and Table 3.7-9, respectively in WSNF-200. The load combination tables showed that the stresses in both the W150 storage cask and the W100 transfer cask are within the applicable code allowable.

3.5.2 Canister Shell and Basket Assembly

The W21 and W74 are evaluated for a range of postulated accident and natural phenomena events. Postulated accidents included cask handling drops, loss of transfer cask neutron shielding, fire, accident internal pressure, and fully blocked inlet and outlet vents. Natural phenomena events included design basis earthquake and flood. The accident and natural phenomena event loads are combined with the concurrent normal loads in accordance with loading combinations specified in NUREG-1536.

3.5.2.1 Handling Drop Accidents

Handling drop accidents are postulated for the canister systems while the canister systems are placed inside either a storage cask or the transfer cask. The canister systems are evaluated for: (1) a storage cask end drop, (2) a storage cask tip-over during up- or down-ending operations, and (3) a transfer cask side drop.

Storage Cask End Drop

The accident storage cask end drop scenario is a postulated vertical drop onto the bottom end of the storage cask from a height of 36-in. The canister systems are evaluated for the end drop scenario using equivalent static loads. The equivalent static end drop load is equal to the peak rigid-body canister assembly acceleration multiplied by a DLF to account for possible dynamic amplifications. The storage cask bottom end drop onto the ISFSI pad results in a canister assembly peak rigid-body response of 28g. The rigid-body response is conservatively characterized as a half-sine wave pulse and the maximum DLF for an undamped system for a half-sine wave pulse is 1.75. Thus, the maximum equivalent static acceleration for the canister assembly for the end drop is 1.75 x 28g = 49g. A bounding static acceleration of 50g is used for the evaluation of the W21 and W74.

Canister Shell Assembly

The stresses in the canister shell components are evaluated for a bounding 50g bottom end drop using a combination of finite element analysis and closed form hand calculations. The bounding canister shell stress intensities resulting from 50g bottom end drop loads are reported in WSNF-201, Table 3.7-4 for the W21 and WSNF-203, Table 3.7-1 for the W74. The results show that all stresses within the canister shell are lower than the ASME Code, Section III, Subsection NB, Service Level D allowable stresses.

The canister shell is also evaluated for buckling for the 50g bottom end drop in accordance with ASME Code Case N-284. The bottom end drop condition results in the highest axial compressive stress in the shell. Internal pressure loads that result in tensile stress in the shell, thereby offsetting the impact loads, are conservatively neglected for buckling evaluation. The results show that the canister shell buckling interaction ratio is only 0.14 for the bounding 50g bottom end drop load. The shell will not buckle when the ratio is less than 1.0. Thus, it can be concluded that the canister shell does not buckle due to the 50g bottom end drop load.

The canister top closure plate and its attachment welds to the canister shell are controlled by the canister lifting operation. The top closure plate and its attachment welds must sustain the dead weight of the loaded canister and designed in accordance with ANSI N14.6 stress factors of six (6) against yield strength and ten (10) against ultimate tensile strength.

The W21 canister top inner closure plate and the top carbon steel shield plug are evaluated by hand calculations assuming the plate behaves as a simply supported circular plate subject to a uniform pressure load due to its own weight. The results show that the stresses in these components due to the 50g bottom end drop loads are lower than the Service Level D allowable stresses. The W74 canister top shield plug assembly is evaluated for the bounding 50g bottom end drop load using a quarter symmetry finite element model. The top shield plug model includes only the shield plate. However, the weight of the shield caps is accounted for in the model by adjusting the density of the elements in the region of the shield caps. The W74 top end shield plug is designed in accordance with Subsection NF of the ASME Code. The maximum primary membrane and primary membrane plus bending stress intensities for the shield plug assembly due to the 50g bottom end drop are 5.0 ksi and 27.1 ksi, respectively. The allowable primary membrane and membrane plus bending stress intensities are 31.9 ksi and 47.9 ksi, respectively. Thus, the W74 top shield plug assembly stresses meet the stress criteria of the design code.

Canister Basket Assembly

(a) W21Basket Assembly

The W21 spacer plates support only its own weight during the storage cask bottom end drop except for the bottom end spacer plate which supports the guide tube assembly via the attachment brackets. The fuel assemblies within the basket assembly are supported vertically by the canister bottom end closure. The load capacity of the guide tube attachment brackets has been determined to be 17.9g. An upper bound failure load of 20g is used for the purpose of estimating the guide tube assembly loads on the bottom end spacer plate. The W21 spacer plates are designed for a bounding 50g bottom end drop acceleration load. The stresses in the spacer plates are equal to 50 times the maximum vertical dead weight stresses. The maximum stresses in the bottom end spacer plate are taken as the larger of 20 times the bottom end plate (with the guide tubes attached) vertical stresses or 50 times the bottom end spacer plate (self-weight only without the guide tubes) vertical dead weight stresses. The maximum stresses in the W21 spacer plates due to the storage cask bottom end drop are presented in WSNF-201, Table 3.7-5. The results show that the W21spacer plate stresses are less than the ASME Section III, Subsection NG, Service level D allowable stresses.

For the postulated storage cask end drop condition, the W21 support rod assembly (e.g., support rod segments and support sleeves) provides longitudinal support for the spacer plates. The support rod assembly components are evaluated as a beam-column system in accordance with NUREG/CR-6322 for stainless steel linear type component supports subjected to combined axial compression and bending. The results show that both the support rod segments and the support sleeves meet the allowable stress design criteria for the bounding 50g storage cask end drop loading condition. The maximum support rod interaction ratio is 0.84 and the maximum support sleeve interaction ratio is 0.77. Since the interaction ratio is less than 1.0, the W21 support rods and support sleeves meet the buckling design criteria of Section III, Appendix F, Article F-1334.5, of the ASME Code.

The W21 guide tubes are loaded only by their own weight during the storage cask bottom end drop condition. A bounding equivalent static acceleration load of 50g is conservatively used in the stress calculations. The guide tube area at the bottom end is reduced due to the 6.5-in wide cutout on all four faces. The uniform axial compressive stress in the guide tube is calculated to be 14.0 ksi. The corresponding service Level D allowable primary membrane stress intensity, conservatively based on Type 316 stainless steel material properties at 700^{o} F, is 39.1 ksi. Therefore, the W21 guide tubes meet the stress acceptance criteria for the postulated storage cask bottom end drop condition. The guide tube is checked for elastic buckling by hand calculations. The design allowable buckling stress is 137.0 ksi which is much larger than the calculated compressive stress of 14.0 ksi. Therefore, the guide tube does not buckle under the 50g bottom end drop condition.

(b) W74 Basket Assembly

For a storage cask bottom end drop, the W74 spacer plates, with the exception of the bottom end long-term Performance (LTP) spacer plate in the W74M upper and lower basket assemblies, support only their own weight. The bottom end LTP spacer plate in the W74M upper and lower basket assemblies support the weight of the guide tube assemblies in addition to its own weight. The guide tube attachment brackets which secure the guide tubes to the bottom end LTP spacer plate are not designed to withstand the storage cask end drop load, but will fail at an acceleration load of 20g. The W74 spacer plate bottom end drop stresses are equal to 50 times the maximum vertical dead weight stresses of the spacer plate. The maximum stresses in the bottom end LTP spacer plate for the bottom end drop are taken as the larger of 20 times the maximum vertical dead weight stresses in the bottom end LTP spacer plate (i.e., with guide tubes attached) or 50 times the maximum vertical dead weight stresses in the top end LTP spacer plate (i.e., spacer plate self-weight only). The maximum stresses in the W74 spacer plates due to the storage cask bottom end drop are presented in WSNF-203, Table 3.7-2. The results show that the W74 spacer plate stresses due to the storage cask end drop are less than the corresponding Service Level D allowable stresses.

The stresses in the LTP spacer plate attachment welds (e.g., to the support tubes) are evaluated using hand calculations for the 50g storage cask bottom end drop loads. The maximum shear stress due to combined axial and bending loads in the most heavily loaded LTP spacer plate attachment weld is 3.2 ksi. The allowable attachment welds shear stresses, based on the SA-240, Type XM-19 stainless steel at a bounding temperature of 700^{o} F and a 40% weld efficiency reduction factor, is 13.8 ksi. Thus, the LTP spacer plate attachment weld is adequate for the storage cask bottom end drop load condition.

The W74 engagement spacer plate is evaluated for a bounding bottom end drop acceleration load of 50g using finite element analysis. The engagement spacer plate is loaded by its self weight, and the total weight of 32 fuel assemblies in 28 guide tube assemblies and four damaged fuel cans (no fuel in the five guide tube locations at the center of the basket). Loads are modeled as uniform pressure loads applied to the regions of the engagement spacer plate over which the various components are supported. The self weight of the engagement spacer plate is included by applying a 50g acceleration load to the model. The maximum stresses in the engagement spacer plate due to the bounding 50g bottom end drop are reported in WSNF-203, Table 3.7-2. The results show that the engagement spacer plate stresses resulting from the 50g bottom end drop are lower than the Service Level D allowable stresses for SA-240, Type XM-19 stainless steel at a bounding design temperature of 600^{o}F.

When subject to the storage cask bottom end drop loading, the upper basket assembly support tubes provide longitudinal support only for the upper basket assembly spacer plates and support sleeves but the lower basket assembly support tubes provide longitudinal support for the entire upper basket assembly and its contents plus the lower assembly spacer plates, guide tube assemblies, and support sleeves. Therefore, the lower basket assembly support tubes bound that of the upper basket assembly. The stresses in the W74 lower basket assembly support tubes due to the cask bottom end drop are determined by hand calculations. The total axial load acting at the bottom end of the four support tubes in the lower basket assembly is equal to the inertial load of the upper basket assembly containing 32 fuel assemblies and four damaged fuel cans, plus the weight of lower basket assembly, without the fuel, or 1,750 kips for the 50g end drop accelerations. The resulting maximum primary membrane plus bending stress intensity in the W74 lower basket support tubes is 19.7 ksi. The corresponding Service Level D membrane and membrane plus bending stress intensities for SA-240, Type XM-19 stainless steel at an upper bound temperature of 700^{o}F are 60.6 ksi and 90.8 ksi, respectively. The W74 support tube is also evaluated for buckling for the storage cask end drop using the criteria of NUREG/CR-6322 for linear-type support subjected to combined axial compression and bending. The results show that the highest buckling interaction ratio is 0.75. Since the buckling interaction ratio is less than 1.0, buckling will not occur due to the 50g end drop load.

The applicant evaluated the support sleeve and the guide tube for the 50g storage cask bottom end drop loads. The stress evaluation results showed large margins for the guide tube and support sleeve against the code allowable stresses and that the guide tube and the support sleeve met the buckling criteria of NUREG/CR-6322 for the 50g storage cask bottom end drop loading condition.

(c) Fuel Rods

Structural failure of the fuel rod cladding in the event of a storage cask handling drop accidents is evaluated. The canisters are designed to withstand a bounding storage cask end drop of 50g and bounding storage cask tip-over and transfer cask side drop loads of 30g and 60g respectively. For transverse impact loads resulting from the tip-over or side drop, the fuel assemblies are supported by the guide tubes. The guide tubes are evaluated for two loading conditions: (1) linear elastic analysis of a uniform fuel load assumption, and (2) elastic-plastic analysis based on concentrated fuel loads at the grid spacers. The bounding 60g transfer cask side drop acceleration loads are used. The results showed that the guide tube stresses and maximum displacements are within acceptable limits. Also, studies (e.g., UCID 21246, LLNL) have shown that the lowest buckling load is 63g for the most limiting fuel assemblies. Thus, it can be concluded that the structural integrity of the fuel rod cladding will be maintained in the event of a postulated storage cask tip-over or transfer cask side drop.

For the postulated storage cask end drop, the fuel rod cladding is evaluated to assure that integrity of the fuel assembly will be maintained. The dominant failure mode of the fuel rod for the end drop loading is buckling. Classical Euler buckling solutions are used to determine the impact load at which the onset of buckling occurs in each fuel type. For the BWR fuel type (to be stored in the W74 canister), the Euler buckling load is 86g. The Westinghouse 17x17 OFA fuel type has the lowest buckling capacity (e.g., 10.1g) for a bottom end drop. Recognizing that the onset of buckling did not imply rupture of the fuel cladding, a bounding fuel rod stress evaluation is performed based on the maximum possible lateral deflection of the fuel rod. For the Westinghouse 17x17 fuel type, the critical g-load at which the fuel rod cladding stress reaches its yield stress is determined to be 52g with no moment restraint at the fuel assembly grid spacers and 86g with moment restraint. Since the lower bound load capacity of the fuel rod (52g) exceeds the bounding end drop load of 50g, the integrity of the fuel rod will be maintained for the bounding design end drop loads.

Test data for fuel assemblies with high burnup generally shows an increase in the strength of zircaloy fuel cladding. For buckling of high burnup fuel up to 60,000 MWD/MTU, the cladding wall thickness is adjusted for a maximum of 70 μm oxide layer and an oxide-to-metal ratio of 1.56. The applicant stated that test data has shown that high burnup fuel retains sufficient strength and ductility during relatively short-term fast strain loading conditions. Therefore, the same elastic modulus (10.4×10^6 psi) and dynamic yield strength (80.5 ksi) of zircaloy at 750°F may be used for the buckling evaluation of fuel rods with high burnup. The critical buckling load for the high burnup fuel is determined to be 49g with no moment restraint and 80g with moment restraint from adjacent fuel rod spans. In light of the calculated maximum g-load of the basket assembly is 49g based on a conservative dynamic load factor of 1.75, it is concluded that the fuel rod integrity is maintained for the high burnup fuels in the event of an end drop.

Storage Cask Tip-Over

The storage cask tip-over event results in a peak rigid body acceleration of 21.9g at the top end of the storage cask. The structural evaluation of the W21 and W74 canisters is performed using equivalent static loads. The equivalent static acceleration load is equal to the peak rigid-body acceleration multiplied by a DLF to account for possible dynamic amplification. The transverse natural frequencies of the canisters are greater than 100Hz. Based on the impact load rise time and the canister transverse natural period, the DLF is approximately 1.2. Therefore, the maximum equivalent static acceleration at the top end of the canister for the storage tip-over is 1.2 x 21.9g = 26.3g. Conservatively, a bounding equivalent static acceleration load of 30g is used for the canister tip-over structural evaluation.

W21 and W74 Canister Shell Assembly

The structural evaluation of the W21 and W74 canister shell assembly for the postulated storage cask tip-over condition is performed using the three-dimensional, ½-symmetry finite element model representing the top end region of the W21M-LS canister shell and basket assembly. The W21M-LS canister shell and basket assembly are bounding because the total weight of the basket assembly and the fuel assemblies of the W21 canisters are higher than those of the W74 canisters (e.g., 56.1 kips versus 51.6 kips). Since the tip-over loads at the bottom end of the canister shell assembly is much lower than those at the top end, the stresses in the bottom end are bounded by those in the top end region and, consequently, need not be evaluated.

For the tip-over condition, the canister shell is loaded by its own weight plus the weight of the basket assembly and fuel. The stresses in the canister shell due to the tip-over load condition are calculated using a linear-elastic analysis. The maximum stresses in the canister shell components are summarized in WSNF-201, Table 3.7-4 for the W21 canister and WSNF-203, Table 3.7-1, for the W74 canister. The maximum stresses in the canister shell due to the bounding 30g tip-over load are shown to be lower than the corresponding ASME Section III, Subsection NB, Service Level D allowable stresses.

W21 Basket Assembly

(a) Spacer Plates

The W21 basket assembly spacer plate is designed for a bounding 30g tip-over load. Two stress analyses are performed for the spacer plate. A linear-elastic stress analysis is performed based on a uniform fuel load assumption. A plastic stress analysis is performed assuming the fuel load as concentrated loads applied to the spacer plates through the fuel grids. The primary purpose of the plastic stress analysis is to determine the postulated maximum spacer plate permanent deformation for consideration in the criticality evaluation. Only the most highly loaded W21 carbon steel and stainless steel spacer plates are evaluated. The spacer plate is assumed to be supported only at the locations of the storage cask rails, conservatively neglecting the support provided by the canister shell. Finite element analysis is performed for the linear-elastic stress analysis. The results show that the maximum primary membrane and primary membrane plus bending stress intensities due to the bounding 30g storage cask tip-over loading are less than the corresponding ASME Section III, Subsection NG, Service Level D allowable stress intensities. The Plastic stress analysis is performed using a ½-symmetry multi-span plane-stress finite element model described in WSNF-201, Section 3.9.3.3.3. The results of the spacer plate plastic tip-over analysis show that plastic strain in the most highly loaded W21 spacer plates occur only in a few very local regions. The maximum permanent deformations in the most highly loaded W21 carbon steel and stainless steel spacer plates resulting from the bounding 30g tip-over load are 0.005-in and 0.002-in, respectively.

Buckling evaluations of the spacer plates are performed for the storage cask tip-over, considering both beam-column buckling in accordance with NUREG/CR-6322 and general instability in accordance with Appendix F of the ASME Code. The maximum axial compressive stress and bending stress in the W21 carbon steel and stainless steel spacer plate ligaments are summarized in WSNF-201, Table 3.7-7 along with the resulting interaction ratios. It is seen that the highest interaction ratios resulting from the bounding 30g tip-over load are 0.31 for carbon steel plate and 0.70 for the stainless steel spacer plate. In addition to the elastic beam-column buckling analysis, general plastic instability of the W21 spacer plate is evaluated for the 30g storage cask tip-over load. The results show that the most highly loaded W21, 3/4-in thick, carbon steel spacer plate has a load factor of 2.2 against plastic instability.

(b) Support Rod Assemblies and Guide Tubes

The support rod assembly and guide tube loads due to the storage cask tip-over are bounded by those due to the transfer cask side drop. Therefore, the support rod assembly and guide tube stresses are not evaluated for the storage cask tip-over condition.

<u>W74 Basket Assembly</u>

(a) General and LTP Spacer Plates

The W74 general and LTP spacer plates are relied upon to support and maintain the positions of the guide tube assemblies and fuel assemblies for criticality control during the storage cask tip-over. For the tip-over evaluation, the W74 spacer plates are assumed to be supported only at the locations of the storage cask rails, conservatively neglecting the support provided by the canister shell. Only the most highly loaded W74 general and LTP spacer plates are evaluated for the 30g storage cask tip-over load. A linear-elastic stress analysis is performed for the spacer plates based on a uniform fuel load assumption. In addition, a plastic stress analysis is performed assuming the fuel load is applied to the spacer plates as concentrated loads at the fuel grid locations. The primary purpose of the plastic stress analysis is to determine the maximum spacer plate permanent deformation resulting from the cask tip-over for consideration in the criticality evaluation.

The spacer plate loading from the weights of support tubes, support sleeves, guide tube assemblies, damaged fuel canisters, and fuel assemblies are determined for the bounding 30g tip-over loading condition. Linear-elastic static analyses are performed for the most highly loaded W74 general and LTP spacer plates. The results show that the maximum primary membrane and primary membrane plus bending stress intensities in the most highly loaded W74 general and LTP spacer plates are lower than the ASME Section III, Subsection NG, Service Level D allowable stress intensities. For the plastic stress analysis, the fuel loads are conservatively applied to the supporting basket assembly structure as concentrated loads at the fuel grid spacer locations. The worst case loading for each W74 spacer plate results when the fuel assembly grid spacer is located directly over that spacer plate. The results of the W74 general and LTP spacer plate tip-over plastic analyses show that the spacer plate strain and permanent deformation due to the bounding tip-over loads are small. The maximum plastic strain and deformation are 0.7% and 0.005-in, respectively which occurs in the general spacer plate.

The buckling evaluation of the W74 spacer plates for the postulated storage cask tip-over considered both the beam-column buckling in accordance with NUREG/CR-6322 and the general plastic instability in accordance with Appendix F of the ASME Code. For the beam-column buckling, evaluations are performed for the most highly stressed ligaments in the most highly loaded W74 general and LTP spacer plate for the bounding 30g tip-over loading both with or without the bounding normal thermal loads superimposed. The maximum axial compressive and bending stress in the most highly loaded W74 general and LTP spacer plate ligaments are summarized in WSNF-203, Table 3.7-5. The buckling evaluation show that the highest interaction ratio in the W74 general spacer plate is 0.44 and the highest interaction ratio in the W74 LTP spacer plate is 0.60. Since the ligament, which subjected to combined compression and bending, has an interaction ratio less than 1.0, it will not buckle.

General plastic instability of the W74 spacer plate is evaluated for the bounding 30g storage cask tip-over load. Since the general stability is proportional to the bending stiffness, the 3/4-in thick W74 general spacer plate controls the plate stability. General instability occurs when the W74 general spacer plate experiences a global lateral plate buckling mode. The large displacement, plastic analysis shows that the most highly loaded W74 general spacer plate remains stable up to 150% of the bounding tip-over load, both with or without thermal loading superimposed. The deformed shape of the most highly loaded W74 general spacer plate at 150% of the bounding 30g tip-over load (i.e., 45g) plus the thermal load is shown in WSNF-203, Figure 3.7-3.

(b) Engagement Spacer Plate

The engagement spacer plate loads due to the postulated storage tip-over condition are approximately equal to half of those due to the transfer cask side drop. Therefore, the engagement plate stresses are controlled by the transfer cask side drop condition and are not evaluated for the tip-over drop condition.

(c) Support Tubes, Support Sleeves, and Guide Tubes

Loads due to the postulated storage cask tip-over condition are bounded by those due to the transfer cask side drop condition. Therefore, these components are not evaluated for the storage tip-over condition.

Transfer Cask Side Drop

The transfer cask is evaluated for a postulated side drop from a height of 72-in. The transfer cask side drop event results in a peak rigid body response of 46g. The W21 and W74 canisters are evaluated for the postulated transfer cask side drop event using equivalent static loads. The equivalent static load is equal to the peak acceleration load multiplied by a DLF to account for possible dynamic amplification. The maximum DLF for the canister assembly is 1.15 based on the natural frequencies of the canister and the rise time of the acceleration load. Thus, the maximum equivalent static load for the transfer cask side drop is 1.15 x 46 = 52.9g. A bounding 60g side drop acceleration load is used for the canister side drop structural evaluation.

W21 and W74 Canister Shell Assembly

For the transfer cask side drop condition, the canister shell assembly is supported by the transfer cask inner shell and loaded by its own weight in addition to the weight of the basket assembly and fuels contained in the canister shell cavity. The side drop load condition is evaluated using the three dimensional half-symmetry finite element model represents the top end region of the W21M-LS canister shell and basket assembly. The W21M-LS canister is selected as the bounding design over the W74 canister shells since it has the largest overall combined weight for the basket assembly and fuel and the heaviest top shield plug. A bounding equivalent static deceleration load of 60g is applied to the finite element model. The inertial loads of the fuel assembly and guide tubes are applied as uniform pressure loads over the width of the supporting spacer plate.

The stresses in the canister shell due to the 60g side drop load condition are calculated using a linear-elastic analysis. The maximum primary membrane and primary membrane plus bending stresses intensities in the canister shell are reported in WSNF-201, Table 3.7-4 for the W21 canister, and WSNF-203, Table 3.7.1 for the W74 canister. The maximum stress intensities are shown to be lower than the corresponding ASME Section III, Subsection NB, Service Level D allowable stresses.

W21 Basket Assembly

(a) Spacer Plates

The W21 basket assembly spacer plate is designed for a bounding 60g transfer cask side drop impact load. Two stress analyses are performed for the spacer plate. A linear-elastic stress analysis is performed based on a uniform fuel load assumption. A plastic stress analysis is performed assuming the fuel load as concentrated loads applied to the spacer plates through the fuel grids. The primary purpose of the plastic stress analysis is to determine the postulated maximum spacer plate permanent deformation for consideration in the criticality evaluation. Only the most highly loaded W21 carbon steel and stainless steel spacer plates are evaluated. Since the side drop acceleration load does not vary in magnitude over the length of the basket assembly, the most heavily loaded W21 spacer plates are those which support the largest tributary weight. The spacer plate in-plane tributary weights are defined as the portion of SNF assembly (and fuel spacer, if required), guide tube, support rod, and support sleeve weights that are supported by each spacer plate in the transverse direction, combined with the spacer plate self-weight. A total of four spacer plate side drop impact azimuth orientations are analyzed, including 0^o, 15^o, 30^o, and 45^o. The spacer plate loading from the support rod, guide tube, and fuel assemblies are determined for each impact orientation and applied to the spacer plate finite element model. Linear-elastic stress analysis is performed. The maximum primary membrane and primary membrane plus bending stress intensities in the most highly loaded W21 carbon steel and stainless steel spacer plates are summarized in WSNF-201, Table 3.7-8 for each side drop orientation. The results show that the maximum primary membrane and primary membrane plus bending stress intensities due to the bounding 60g transfer cask side drop are less than the corresponding ASME Section III, Subsection NG, Service Level D allowable stress intensities. The Plastic stress analysis is performed using the full multi-span plane-stress finite element model described in WSNF-201, Section 3.9.3.3.4. The finite element model include three spacer plates; the center spacer plate over which the fuel grid spacer are assumed to be positioned, and the adjacent spacer plates on either side. The results of the W21 carbon steel and stainless steel spacer plate 60g side drop plastic analyses are summarized in WSNF-201, Table 3.7-9. The analysis shows that the plastic strain in the most highly loaded W21 spacer plates for the bounding 60g side drop load is small and occurs only in localized regions. The maximum permanent deformations in the most highly loaded W21 carbon steel and stainless steel spacer plates resulting from the bounding 60g loading are 0.03-in and 0.049-in, respectively.

Buckling evaluations of the spacer plates are performed for the postulated transfer cask side drop condition, considering both beam-column buckling in accordance with NUREG/CR-6322 and general instability in accordance with Appendix F of the ASME Code. The maximum axial compressive stress and bending stress in the W21 carbon steel and stainless steel spacer plate ligaments are summarized in WSNF-201, Table 3.7-10 along with the resulting interaction ratios. It is seen that the highest interaction ratios resulting from the bounding 60g side drop impact load are 0.49 for carbon steel plate and 0.94 for the stainless steel spacer plate. Since both ratios are less than 1.0, the spacer plates satisfied the elastic beam-column stress criteria for local buckling. In addition to the elastic beam-column local buckling analysis, general plastic instability of the W21 spacer plate is evaluated for the 60g transfer cask side drop load. General instability is controlled by the thin carbon steel spacer plates. The results show that the most highly loaded W21, 3/4-in thick, carbon steel spacer plate has a load factor of 1.5 against general plastic instability. Thus, the spacer plates will not buckle under the 60g transfer cask side drop condition.

(b) Support Rod and Support Sleeve Assemblies

The support rod assembly is evaluated using the simple beam theory. Two sections of the support rod are evaluated: the cantilever section at the top and bottom end and the longest span section between any two adjacent 2-in thick stainless steel spacer plates. Since the spacer rod segments are tightened to 150 ft-lbs, the ends of the support rod span are assumed as fixed ends. The bounding side drop acceleration of 60g is considered for the rod evaluation. The beam analyses show that the maximum stresses in the W21 support rod due to the 60g side drop load are less than the Service Level D allowable stresses and that the support rod threads are structurally adequate for the shearing force induced by the 60g side drop impact. The support sleeve is supported full length by the support rod. It is conservatively assumed that the support sleeve supports no transverse loads but the weight of the support sleeve is included in the support rod beam analysis.

(c) Guide Tubes

When subjected to the transfer cask side drop loading, each guide tube is supported by the basket spacer plates and loaded by its own weight in addition to the weight of a spent fuel assembly. The W21 guide tubes are evaluated for a bounding 60g transfer cask side drop acceleration load. The side drop stress evaluation of the W21 guide tubes considered two bounding conditions for the fuel loading on the guide tube: (1) Linear elastic analysis based on a uniform fuel load assumption, and (2) A plastic analysis assuming the fuel loads are concentrated at the spent fuel assembly grid spacers. The side drop evaluations of the guide tubes are described in the sections below and the results are summarized in Table 3.7-5, WSNF-201.

Uniform fuel loading-linear elastic stress analysis:

Fuel weight calculations are performed for the largest guide tube free spans of each W21 canister. The results of the hand calculations show that the maximum bending stress occurs in the W21M-SD/W21T-SL guide tubes for the W15x15 fuel with control components. A detailed finite element model analysis of the guide tube assembly is performed for the 60g side drop. The W21 guide tube neutron absorber sheets and stainless steel wrapper are not included in the finite element model, conservatively neglecting the support they provide to the guide tube. However, their mass is included in the model. The uniformly distributed fuel load is applied as a pressure load on the supporting guide tube panel. The maximum membrane and membrane plus bending stress intensities in the guide tube are 13.6 ksi and 54.0 ksi, respectively. The Service Level D allowable primary membrane and membrane plus bending stress intensities are 39.1 ksi and 58.7 ksi, respectively. Thus, the guide tube stresses are within the Code allowable limits.

Concentrated fuel loading at the grid spacers:

The 60g side drop evaluation of the W21 guide tube with a concentrated load at the fuel grid spacer is performed using the ½-symmetry multi-span finite element model described in WSNF-201, Section 3.9.3. The purpose of the analysis is to determine the maximum guide tube permanent displacements resulting from the postulated transfer cask side drop. The fuel load is applied to the model at the mid-span of the largest guide tube span (i.e., midway between two spacer plate supports) since this loading will result in the largest guide tube displacements for a given load. For the plastic analysis, the following loads are applied: (1) a 60g side drop acceleration applied to account for the guide tube self-weight, (2) an imposed displacement on the guide tube bottom panel nodes at the region of the fuel grid spacer, and (3) a uniform pressure load applied to the adjacent spans which do not have the imposed displacement. The magnitude of the imposed displacement is chosen based on the maximum distance from the edge of grid spacer to the outmost fuel rod. The plastic material properties of Type 316 stainless steel at a bounding temperature of 700^{o}F (Table 3.3-4, WSNF-201) are used for the analysis. The analysis results show that the maximum stress intensity and maximum equivalent plastic strain at the extreme fibers (i.e., top and bottom) of the guide tube panel are 39.0 ksi and 13.9%, respectively. The minimum elongation of Type 316 stainless steel is about 40%. The plastic strain in the guide tube occurs only in the span which supports the fuel grid spacer and is limited to a small region near the edge of the fuel grid spacer footprint. The applicant also performed a plastic analysis based on uniformly distributed fuel assembly loads and the maximum permanent displacement at the center of the guide tube panel is 0.079-in. The displacement is considered in the criticality evaluation.

<u>W74 Basket Assembly</u>

(a) General and LTP Spacer Plates

The W74 general and LTP spacer plates are evaluated for the bounding 60g transfer cask side drop loads. Two loading conditions are considered: (1) Uniform fuel loading (i.e., fuel weight is distributed uniformly to the basket assembly spacer plates), and (2) Concentrated fuel loading at the fuel assembly grid spacers. The uniform fuel loading assumption is used for elastic system stress analysis and the concentrated fuel loading assumption is used for plastic stress analysis to determine the maximum spacer plate permanent deformation in the most highly loaded spacer plates.

The linear-elastic stress analysis showed that the highest primary membrane and primary membrane plus bending stress intensities in the most heavily loaded W74 general spacer plate are 35.3 ksi and 78.9 ksi, respectively. The Service Level D allowable primary membrane and primary membrane plus bending stress intensities are 75.4 ksi and 113.1 ksi, respectively. Similarly, the maximum primary membrane and primary membrane plus bending stress intensities in the most highly loaded W74 LTP spacer plate are 17.4 ksi and 51.3 ksi, respectively. The Service Level D allowable primary membrane and primary membrane plus bending stress intensities for SA-240, Type XM-19 stainless steel at 650^{o}F are 61.1 ksi and 91.5 ksi, respectively. Thus, based on uniform fuel loading, the calculated stresses in the spacer plates are less than the ASME Code allowable stresses.

The results of the W74 general and LTP spacer plate 60g side drop plastic analyses shows that the plastic strains in the most highly loaded W74 general and LTP spacer plates are small and occur only in localized regions. The maximum equivalent plastic strain in the most highly loaded W74 general spacer plate is 0.5% and occurs for the 0^{o} azimuth impact orientation. The plastic strain is less than the minimum elongation of 16% of the spacer plate material. The maximum permanent deformation in the most highly loaded W74 general spacer plate results from the 45^{o} azimuth impact orientation is calculated to be 0.04-in only.

Spacer plate buckling for the 60g side drop condition is considered for both beam-column local buckling and general plastic instability in accordance with Appendix F of the ASME Code. The maximum axial compressive stress and bending stress in the spacer plate ligaments are summarized in Tables 3.7-7 through Table 3.7-10 of WSNF-203. The results show that the interaction ratios are less than 1.0. Therefore, beam-column local buckling of the spacer plate ligaments will not occur. In addition to the elastic beam-column buckling analysis, general plastic instability of the W74 spacer plate is evaluated using plastic large deflection analysis. General instability is controlled by the W74 general spacer plates. The results show that the most highly loaded W74 general spacer plate remains stable up to 90g side drop loads. Thus, the minimum design margin against plastic instability is 1.5 for the W74 general spacer plates.

(b) Engagement Spacer Plate

The W74 engagement spacer plate is evaluated for the bounding 60g side drop loads. The analysis shows that the maximum primary plus secondary plus peak stress intensity of 16.4 ksi results from the 36^{o} azimuth impact orientation. This stress intensity is conservatively compared with the allowable Service Level D primary membrane stress intensity. The allowable stress intensity for SA-240, Type XM-19 stainless steel material at the design temperature of 600^{o}F is 61.4 ksi. The elastic stability of the W74 engagement spacer plate is evaluated for the effects of a bounding free drop impact load. The assumed impact load consists of 60g in-plane and 45g out-of-plane. The evaluation demonstrates that the engagement spacer plate does not fail due to elastic buckling. Therefore, an adequate design margin and factor of safety exist for the engagement spacer plate for the transfer cask side drop loading condition.

(c) Support Tubes and Support Sleeves

The support tube stress resulting from the 60g side drop load is evaluated using simple beam theory. For the side drop condition, the horizontally oriented support tubes are loaded in the vertical direction by their own weight plus the weight of support sleeves and the fuel assembly inside the support tube. The support tube stresses are calculated for the 60g side drop load and shown to be much less than the ASME Code allowable. In the horizontal orientation, the support sleeve supports only its own weight. As such, the stresses in the support sleeves due the bounding 60g side drops are very small and do not control the design.

(d) Guide Tube

For the transfer cask side drop condition, the W74 guide tube assemblies are loaded by the initial load due to their own weight plus the weight of fuel assembly within each guide tube. The guide tube structural evaluation is performed for a bounding 60g side drop load.

Stress evaluation:

The guide tube stresses are determined using a linear-elastic static analysis and uniform fuel loading. The maximum primary membrane and membrane plus bending stress intensities in the guide tube are 15.4 ksi, and 54.6 ksi, respectively. The Service Level D allowable primary membrane and primary membrane plus bending stress intensities for SA-240,Type 316 stainless steel at the design temperature of 715^{o}F are 38.8 ksi, and 58.3 ksi, respectively. Therefore, the maximum guide tube stresses are within the Code allowable.

Permanent deformation evaluation:

The W74 guide tubes are evaluated for the bounding 60g side drop load on a plastic basis to determine the maximum guide tube permanent deformation. Two fuel loading conditions are considered for the guide tube: (1) a uniform fuel loading, and (2) concentrated fuel loading at fuel grid spacers. The maximum guide tube permanent deformation at the center of the panel that supports the uniformly distributed fuel load is 0.066-in. The maximum permanent deformation of the guide tube, occurring at the location of the concentrated grid spacer loading is 0.128-in. These guide tube deformations are considered in the criticality evaluation.

3.5.2.2 Flood

The W21 and W74 canisters in the storage cask are evaluated for the effects of a design-basis-flood resulting from extreme natural phenomena event. The canisters are conservatively evaluated for a 50-foot flood height. The canisters are protected from the lateral forces due to the flood current by the storage cask. Consequently, the only flood load affecting the canister is the 21.7 psi hydrostatic pressure load due to the 50-foot flood water head. Since the hydrostatic pressure on the canisters are less than the design basis 69psig accident internal pressure load, the canister shell stresses due to flood are bounded by the shell stresses due to accident internal pressure in Section 3.5.2.7.

3.5.2.3 Fire

The W21 and W74 canisters in the storage cask are evaluated for a postulated accidental fire event defined in Section 2.3.3.4 of WSNF-200. The maximum canister pressure as a result of the postulated fire event is bounded by the design basis accident pressure of 69psig. Further, the external heating acts to reduce the thermal gradients on the canister, resulting in a lower thermal stress than the normal ambient conditions. Thus, this loading condition is bounded by the accident pressure condition evaluated in Section 3.5.2. 7.

3.5.2.4 Earthquake

The FuelSolutions™ DBE accelerations are 0.35g horizontal and 0.25g vertical. Since the lowest fundamental frequency of the canisters is greater than 33 Hz, no amplification of the seismic loading is needed for the canisters. However, bounding accelerations of 1.1g along the canister longitudinal axis (e.g., vertical) and 1.3g transverse (e.g., horizontal) to the canister longitudinal axis is conservatively used for the structural evaluation of the W21 and W74 canister basket assembly.

The canister shell assembly stresses due to the design basis earthquake loads are bounded by those due to the postulated cask drop conditions. Due to the relative magnitudes of the seismic loads and drop accident loads (i.e., 0.25g versus a 50g end drop and 0.35g versus a 60g side drop), the canister shell stresses need not be calculated for the earthquake load condition.

A bounding seismic evaluation of the basket assembly components is performed by using a 1.3g transverse acceleration combined with a 1.1g longitudinal acceleration. The spacer plate and guide tube stresses due to the transverse seismic acceleration are calculated by scaling the maximum stresses due to the horizontal dead weight load by a factor of 1.3. Similarly, the stresses due to the longitudinal earthquake acceleration are determined by scaling the maximum stresses due to the vertical dead weight load by a factor of 1.1. The resulting stresses due to transverse and longitudinal earthquake loads are added together to arrive at the total seismic stresses. The resulting stresses are shown in Table 3.7-5, WSNF-201 for the W21 canisters and Table 3.7-2, WSNF-203 for the W74 canisters.

3.5.2.5 Fully Blocked Inlet and Outlet Vents

The canisters, loaded with design basis fuel assemblies and dry stored vertically in the storage cask, are evaluated for a postulated accident thermal event in which complete blockage of all storage cask inlet and outlet vents occurs. A steady-state ambient temperature of 100^{o} F with insolation is assumed. The thermal analysis showed (Chapter 4 of the corresponding W21 and W74 SARs) that the maximum temperatures in the canister assembly due to storage cask vent blockages are bounded by those resulting from the transfer cask loss of neutron shield accident condition.

3.5.2.6 Loss of Transfer Cask Neutron Shield

The canisters, while loaded with SNF and inside the transfer cask, are evaluated for a postulated accident event in which the transfer cask lost all its liquid neutron shield. A steady-state ambient temperature of 100^{o} F with insolation is assumed to concur concurrently. The canister internal pressure resulting from the loss of neutron shield accident is less than the design basis accident pressure of 69.0 psig in Section 3.5.2.7 below. Thus, the canister shell stresses will be bounded by the design basis accident pressure analysis.

The temperature difference between the spacer plate and the canister shell is reduced during the loss of a neutron shield accident. Because of the reduced thermal gradient, the thermal stresses in the basket assembly are reduced during the postulated loss of a neutron shield accident. However, the support sleeves, support rods, and support tubes are subjected to compressive forces resulting from differential thermal expansion of dissimilar materials at elevated temperatures. The applicant performed analysis to show that these components will not buckle under the most severe thermal condition.

3.5.2.7 Accident Internal Pressure

The maximum accident internal pressure is calculated based on the required backfill quantity of helium, and a concurrent non-mechanistic failure of 100% of the fuel rods with complete release of their fill gas and 30% of their fission gasses into the canister cavity. The maximum accident internal pressure is 68.7 psig for the W21 canister and 30.0 psig for the W74 canister. For all canisters with the exception of the W21T-LL, a bounding accident internal pressure of 69 psig is conservatively used for the structural evaluation. The maximum accident pressure for the W21T-LL canister is 45.0 psig. The W21T-LL canister shell stresses are calculated based on a bounding 45.1 psig accident internal pressures.

The structural evaluation of the canisters for the accident internal pressure load is performed by using the axisymmetric finite element model. The resulting canister shell stresses are provided in Table 3.7-4 of WSNF-201 for the W21 canisters and Table 3.7-1 of WSNF-203 for the W74 canisters. All canister shell stresses due to accident internal pressure are within the code allowable values.

3.5.2.8 Load Combinations

Accident condition and natural phenomena event load combinations for the W21 and W74 canisters are discussed in Section 3.1.2.3 of the SER. The load combinations include normal loads, off-normal loads, and accident loads. The load combinations for the canisters are summarized in Table 2.3-1 of the respective SAR (WSNF-201or WSNF-203). The maximum stresses in the canister shell due to load combination are provided in Table 3.7-11 through Table 3.7-15, and the maximum stresses in the canister basket assembly due to load combination are provided in Table 3.7-16 of the respective SAR for the canisters (i.e., WSNF-201 for W21canisters and WSNF-203 for W74 canisters). These load combination stress tables showed that the maximum stresses in the canister shell and basket assembly are within the applicable code allowable stresses.

3.6 Evaluation Findings

The staff concludes that the structural design of the FuelSolutions™ Storage System is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that the FuelSolutions™ Storage System will enable safe storage of spent nuclear fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted practices, and limited confirmatory analysis.

F3.1 SSCs important to safety are described in sufficient detail to enable an evaluation of their structural effectiveness. The FuelSolutions™ Storage System is designed to accommodate the combined loads of normal, off-normal, accident, and the natural phenomena events, in accordance with 10 CFR 72.24. The staff has reviewed the codes and standards used in structural design and find that they are acceptable.

F3.2 The staff has reviewed the design of FuelSolutions™ Storage System and finds that the cask storage system meets the requirements of positive closure, adequate safety factors for lifting devices, no significant chemical or galvanic reactions, and acceptable service life, in accordance with 10 CFR Part 72.120, Part 122, and Part 236.

F3.3 The FuelSolutions™ Storage System is designed and fabricated so that the stored spent nuclear fuel is maintained in a subcritical condition, in accordance with 10 CFR Part 72.124 and Part 72.236. The configuration of the stored spent nuclear fuel is essentially unchanged under the normal, off-normal, accident, and the natural phenomenon events. Additional criticality evaluations are discussed in Section 6 of this SER.

F3.4 The FuelSolutions™ Storage System will reasonably maintain confinement of the stored spent nuclear fuels, in accordance with 10 CFR Part 72.236. The cask and components important to safety have adequate structural integrity for the handling, packaging, transfer, and storage to preclude unacceptable leaks or radiation levels.

F3.5 The FuelSolutions™ Storage System is designed to allow ready retrieval of spent nuclear fuel for further processing or disposal without the release of radioactive materials to the environment or excessive radiation exposures to the workers. No accident or natural phenomena events analyzed will result in damage of the FuelSolutions™ Storage System that will prevent retrieval of the stored spent nuclear fuel.

F3.6 The SAR describes the material properties that are used for the construction of SSCs important to safety and the suitability of these materials for their intended functions in sufficient detail to facilitate evaluation of their effectiveness.

F3.7 The design and selection of materials of the FuelSolutions™ Storage System provides adequate protection of the stored spent fuels against degradation that might otherwise lead to fuel rod cladding gross rupture.

F3.8 The materials that comprise the FuelSolutions™ Storage System will maintain their mechanical properties during all conditions of operation so the spent fuel can be safely stored for a minimum of 20 years and maintenance can be conducted as required.

F3.9 The staff concludes that the structural design of the FuelSolutions™ Storage System is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied.

3.7 References

1. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, 1997

2. American Concrete Institute, "Code Requirements for Nuclear safety-related Concrete Structures," ACI 349, 1990

3. American Concrete Institute, "Building Code Requirements for Reinforced Concrete," ACI 318, 1989

4. AISC Manual of Steel Construction- Allowable Stress Design, Ninth Edition, American Institute of Steel Construction, 1989

5. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NF, Component Supports, 1998 Edition

6. U.S. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, 1980

7. ANSI N14.6, "American National Standards for Radioactive Material Lifting Devices for Shipping Containers weighing 10,000 lbs (4500 kg) or more.", 1993

8. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NG, Core Support Structures, 1998 Edition

9. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, Class 1 Components, 1998 Edition

10. U.S. Nuclear Regulatory Commission, "SFPO Director's Interim Staff Guidance," ISG-4, Rev.1, 1999

11. ANSI/ANS-57.9, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)," American National Standards Institute, 1984

12. U.S. Nuclear Regulatory Commission, "Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet Onto Concrete Pads," NUREG/CR-6608, 1998

13. U.S. Nuclear Regulatory Commission, "Design -Basis Tornado for Nuclear Power Plants," Regulatory Guide 1.76, April 1974

14. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Power Reactors," NUREG-0800, 1981

15. American Society of Civil Engineers, "Minimum Design Loads for Buildings and Other Structures," ASCE 7 - 93

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4.0 THERMAL EVALUATION

The thermal review ensures that the cask component and fuel material temperatures of the FuelSolutions Canister Storage System will remain within the allowable values or criteria for normal, off-normal, and accident conditions. These objectives include confirmation that the fuel cladding temperature will be maintained below specified limits throughout the storage period to protect the cladding against degradation that could lead to gross ruptures, as required by 10 CFR 72.122. This portion of the review also confirms that the cask has been evaluated using acceptable analytic techniques and/or testing methods.

4.1 Spent Fuel Cladding

A general outline of the thermal criteria and analytic methods are addressed in the WSNF-200 SAR. Design specific calculations are presented in the W21 and the W74 SARs. The applicant applies two thermal rating criteria to address the axial heat flux profile for a spent nuclear fuel assembly: a maximum heat load rating and a maximum linear heat generation rate. Peaking factors are used to determine the magnitude of heat generation per unit length (i.e., kW/in). Spent nuclear fuel assemblies with a lower burnup have a higher maximum axial peaking factor when compared to higher burnup fuels (i.e., 1.3 versus 1.1 for PWR fuels and 1.45 versus 1.2 for BWR fuels). Therefore, an adjustment factor is included for low burnup fuels to extend the required cooling time such that the maximum allowable temperature limit is not exceeded. For example, low burnup fuel assemblies with high peaking factors could exceed the maximum allowable clad temperature limit at a total canister heat load rating below the 22.0 kW limit for the W21 storage system. Extending the cooling time ensures that the maximum allowable kW/in linear heat generation rate is not exceeded.

BFS Methodology for Calculating Maximum Allowable Cladding Temperature Limits

NUREG-1536 provides guidance on acceptable methods for meeting the regulatory requirements specified in 10CFR72.122 (h) {The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures...} and (i) {Storage systems must be designed to allow ready retrieval of spent fuel...}. One acceptable method, employed by most applicants, is to calculate a maximum allowable cladding temperature limit using methodologies that are based on a calculation of the amount of accumulated strain (or creep) the cladding experiences over time. This method provides reasonable assurance that the requirements of 10 CFR 72.122 are met.

The thermal design criteria for preventing fuel cladding degradation are presented in Section 4.3 of the SAR. For the W21 canister, a creep methodology was used to calculate cladding temperature limits for PWR fuels having burnups up to 60,000 MWD/MTU for assumed storage times of up to 100 years. For the W74 canister, the same creep model and storage times were used to calculate temperature limits for BWR fuels having burnups up to 40,000 MWD/MTU.

The applicant used a creep correlation that was originally developed by H. Spilker and his colleagues[1] to calculate the allowable fuel cladding temperature limits for the FuelSolutions W21 and W74 canisters. The Spilker correlation was developed from a database containing creep measurements (i.e., creep strain versus time) for unirradiated Zircaloy-4 cladding materials tested at temperatures and stresses ranging from 250-400°C and 80-150 MPa, respectively, and a test duration up to 10,000 hours. As addressed in the Spilker paper, creep strains of unirradiated Zircaloy-4 material were, in general, greater than the creep strains for irradiated material. This led Spilker and his colleagues to conclude that a creep equation derived from unirradiated Zircaloy-4 cladding could conservatively describe the creep behavior of irradiated cladding.

In general, the Spilker creep correlation is a mathematical relationship that can be used to calculate creep strains of Zircaloy cladding as a function of time for known cladding hoop stress and cladding temperatures. Through detailed thermal calculations described below, the applicant calculated the bounding fuel rod average internal gas temperature from which a fuel rod pressure and corresponding cladding hoop stresses were calculated. Using the calculated hoop stresses and a correction factor, the applicant calculated the maximum allowable cladding temperature that corresponds to an accumulated cladding creep strain (or creep strain limit) of 1% over a 100-year dry storage period. As noted above, the creep strains of the Spilker unirradiated Zircaloy-4 data were greater than the creep strains for irradiated material. Thus, the calculated unirradiated strain values were corrected to coincide with measured values of irradiated creep strains by using a correction factor of 2 (i.e., the correction factor was derived from the ratio of the measured creep strains for unirradiated Zircaloy-4 to the measured creep strains for irradiated Zircaloy-4).

With technical assistance from Pacific Northwest National Laboratory (PNNL)[2], the staff performed an independent evaluation to determine whether the Spilker correlation methodology could be used to calculate cladding temperature limits that would preclude gross rupture of the cladding during normal conditions of storage. The methodology of PNL-6189[3] was used by the staff and PNNL to independently verify that the Spilker correlation with a creep reduction factor of 2 was supported by both data and the independent theoretical models of PNL-6189. Based on its independent evaluation, the staff concludes that the applicant's use of the Spilker correlation with a creep reduction factor of 2 is acceptable.

With respect to the 1% accumulated creep strain (or 1% strain limit), the staff reviewed the information and data contained in both the applicant's submittals and in publically available literature regarding creep tests and other tests that measure failure strains. The staff concluded that the mechanical properties of uniform elongation and total strain-to-failure are influenced by the amount of oxide and local concentration of hydrogen in the cladding. Because local concentration is not easily measured nor calculated, a conservative limit is placed on average hydrogen concentration and corresponding oxide thickness. This approach was adopted to preclude excessive local hydrogen concentrations. Based on the currently available information and data, the staff concluded that the use of a 1% strain limit is defendable for spent fuels having oxide thicknesses less than 70 micrometers, irrespective of the burnup.

It should be noted that the applicability of the Spilker correlation methodology to stainless steel clad spent fuels was not established in the SAR for this application. Therefore, stainless steel clad fuel will not be allowed to be stored in the FuelSolutions W21 or W74 canisters without modifications to the SAR and staff review and approval.

The applicability of a 1% creep strain limit for fuels with high burnup (i.e., burnups exceeding 45,000 MWD/MTU) was also assessed. PNNL concluded that zircaloy cladding can withstand uniform creep strains of about 1% or more before the cladding can become perforated if the average cladding hydrogen concentrations in the cladding are less than about 400 to 500 parts per million (ppm). This amount of hydrogen concentration corresponds to an oxide thickness of approximately 70 micrometers (µm) using the recommended hydrogen pickup fraction of 0.15 from Lanning and his colleagues[4]. PNNL also concluded that the strength and ductility of irradiated zircaloy do not appear to be significantly affected by corrosion-induced hydrides at hydrogen concentration levels up to about 400 ppm. Further, the PNL-6189 methodology of creep by grain boundary sliding provides a theoretical basis to expect cladding uniform creep strains greater than 1% for cladding with hydrogen levels in the 400-500 ppm range.

Reasonable agreement was observed between the staff calculated temperature limits using the PNL-6189 methodology and the applicant calculated temperature limits using the Spilker correlation methodology over a range of temperatures and stresses. Therefore, for spent fuel with oxide thicknesses less than 70 µm and burnups up to 60,000 MWD/MTU, the staff has reasonable assurance that (1) the BFS Spilker correlation creep methodology can be used to calculate maximum allowable cladding temperature limits and (2) these temperature limits will preclude the cladding from developing gross ruptures during storage.

PWR fuel assemblies with burnups between 45,000 and 60,000 MWD/MTU will be allowed to be stored in the W21 canister if the cladding oxide thickness is limited to 70 µm. The licensee is required by the TS 5.3.7 to establish a program to measure the cladding oxide thickness to assure it is less than or equal to 70 µm. Fuel assemblies that do not meet these criteria are considered damaged fuel and must be treated like damaged fuel. Oxide measurements are not required for burnups below 45,000 MWD/MTU.

Peak Rod Pressure for Calculating Maximum Allowable Temperature Limits

An important input to the BFS creep methodology is the derivation of the gas pressure within the fuel pin. The applicant developed a simplified method for establishing the pressure in a fuel pin. This method incorporates plant operating pressure and core inlet temperature at normal end-of-(fuel) life operating conditions. The volume averaged gas pressure within the limiting fuel rod at end of life (EOL) operating condition is assumed to be at the plant operating pressure (i.e., 2250 psia). From that thermal state, the average pressure for the limiting fuel pin was corrected to storage conditions within the cask. This corrected pressure is then used to calculate the cladding hoop stress. The average gas temperature for the peak fuel rod at storage conditions was established to be 288.7°C for PWR fuel and 257.4°C for the Big Rock Point BWR fuel. The gas temperature limits were derived from thermal sensitivity analyses that altered the canister heat load until the maximum allowable clad temperature limit was obtained. When calculating the cladding hoop stress, the fuel rod cladding thickness was reduced by 100 µm, the conservatively assumed oxidation thickness. By monitoring the temperature history of the cladding from time of removal from the spent fuel pool to a 100-year dry storage period, a temperature limit is established such that the calculated cladding strain does not exceed 1% for a given burnup level and fuel assembly type. The calculated cumulative cladding strain also included the strain associated with the heatup during vacuum drying. Additionally, The applicant performed confirmatory calculations using a Westinghouse fuel code to demonstrate the acceptability of their simplified approach.

To assess the acceptability of the simplified methodology for deriving the maximum fuel pin pressure, the staff, with the assistance of the PNNL, performed independent thermal analysis of the W21 canister storage system. The thermal temperature gradients calculated in the staff's independent analysis (using the COBRA-SFS computer code) were inputs to the boundary conditions of the FRAPCON-3 computer code. The FRAPCON-3 computer code was used by the staff to calculate detailed fuel pin pressure and temperature profiles. The FRAPCON-3 calculation validated the acceptability of the BFS simplified analytic process for assessing the pressure within the limiting fuel pin of non-IFBA (Integral Fuel Burnable Absorber) PWR fuel rod at dry storage conditions.

Maximum Allowable Temperature Limits

By implementing the BFS creep method and the peak rod pressure method, as outlined above, the applicant calculated a long-term temperature limit of 350°C for PWR fuel in the FuelSolutions W21 canister under normal conditions of storage. This temperature limit is conservatively lower than the staff calculated temperature limit which was derived by using the PNL-6189 methodology. The applicant also calculated a long-term temperature limit of 385.5°C for the BWR fuel in the W74 canister. The W74 temperature limit was also confirmed by staff independent calculations using the PNL-6189 methodology.

For off-normal conditions, the temperature limit is maintained below 400°C to inhibit major annealing of the cladding. This includes imposing administrative controls over the period of time that the fuel can be left in a vacuum (further addressed later in this SER).

The applicant established a short-term temperature limit of 570°C (1058°F) for hypothetical accident conditions for zircaloy clad fuel in accordance with the guidance of NUREG-1536 and PNL-4835[5]. The staff finds this short-term temperature limit acceptable.

Tables 4-1 and 4-2 summarize the fuel temperature limits for the W21 and W74 storage systems, respectively.

Table 4-1
Fuel Temperature Limits (°F) for W21 Storage System

W21 Canister	Normal Conditions	Off-Normal Conditions	Accident Condition
PWR Fuel Cladding (0-60,000 MWD/MTU)	662 (350°C)	752 (400°C)	1058 (570°C)

Table 4-2 Fuel Temperature Limits (°F) for W74 Storage System			
W74 Canister	**Normal Conditions**	**Off-Normal Condition**	**Accident Condition**
BWR Fuel Cladding (40,000 MWD/MTU)	726 (385.5^{o}C)	752 (400^{o}C)	1058 (570^{o}C)

Conclusions Related to the Spent Fuel Cladding Integrity

The staff has reviewed and finds acceptable the BFS method for calculating maximum allowable cladding temperature limits using the Spilker creep correlation methodology and the assumptions related to calculated peak rod pressures. The methodology is consistent with the goals and objectives of the NRC's Standard Review Plan, NUREG-1536.

4.2 Storage System Thermal Design

The thermal criteria for the FuelSolutions Canister Storage System are presented in Sections 2 and 4 of the SAR. Tables 4-3 (PWR) and 4-4 (BWR) identify the thermal rating for the respective Storage System.

Table 4-3 W21 Canister Thermal Rating for Storage			
Cooling Table	**Axial Heat Profile**	**Q_{max} (kW)**	**$LHGR_{max}$ (kW/in)**
0-60,000 MWD/MTU	Maximum Fuel Thermal Output	22.0	0.161
W21 Basket Structure Thermal Rating		25.1	0.184
Storage and Transfer Cask Thermal Rating		28.0	0.253

Table 4-4 W74 Canister Thermal Rating for Storage			
Cooling Table	**Axial Heat Profile**	**Q_{max} (kW)**	**$LHGR_{max}$ (kW/in)**
0 to 40,000 MWD/MTU	Maximum Fuel Thermal Output	24.8	0.216
W74 Basket Structure Thermal Rating		26.4	0.230
Storage and Transfer Cask Thermal Rating		28.0	0.253

4.2.1 Design Criteria

The applicant documented the FuelSolutions Canister Storage System design criteria used to demonstrate compliance with 10 CFR Part 72 requirements for 20-year storage of spent nuclear fuel. These design criteria encompass normal, off-normal, and postulated accident conditions. The design basis events are identified in Table 4.1-2 and Table 4.1-3 of the SARs.

The thermal criteria for the FuelSolutions overpack with the loaded canister are presented in Section 2 of the SAR. Tables 4-5 and 4-6 list the design temperature limits for the canister components.

Table 4-5 FuelSolutions W21
Component Normal, Off-normal, and Accident Temperature Limits (°F)

W21 Canister	Normal Conditions	Off-Normal Conditions	Accident Conditions
PWR Fuel Cladding (0-60,000 MWD/MTU)	662 (350°C)	752 (400°C)	1058 (570°C)
Load Bearing Carbon Steel	700	1000	1000
Load Bearing Stainless Steel	800	1000	1000
Lead Shielding	620	620	620
DU Shielding	2071	2071	2071
Boral	850	1000	1000

Table 4-6 FuelSolutions W74
Component Normal, Off-normal, and Accident Temperature Limits (°F)

W74 Canister	Normal Conditions	Off-Normal Conditions	Accident Conditions
BWR Fuel Cladding (40 MWD/MTU)	726 (385.5°C)	752 (400°C)	1058 (570°C)
Load Bearing Carbon Steel	700	1000	1000
Load Bearing Stainless Steel	800	1000	1000
Borated Stainless Steel	1000	1000	1000

4.2.2 Design Features

Fuel Canister

The thermal design features of the FuelSolutions fuel canister consist of the following components:

1. The FuelSolutions canister includes carbon steel spacer plates for increased thermal conductance;
2. Basket assembly layouts are configured to maximize convective flow areas for horizontal transport;
3. Helium backfill gas in the canister is used to enhance both conductive and convective heat transfer across void spaces in the basket. The quantity of inert gas in terms of moles needed for canister cavity backfill is determined in order to achieve a pressure of 10 psig (1.68 atm) in the canister cavity under normal hot storage conditions (100 ^{o}F ambient conditions) with 1% rod failure.

When compared to other inert gases, the helium backfill used in the canister provides superior heat conduction from the fuel to the canister wall. The helium backfill also provides an inert atmosphere to limit the potential for degradation of the internal components. The heat transfer effectiveness of the helium gas was demonstrated on full scale casks at the Idaho National Engineering and Environmental Laboratory (INEEL). The INEEL experiments also provided data with nitrogen and vacuum conditions inside the fuel canister. These experiments provided data for validating computer codes used by the applicant and the NRC in licensing spent nuclear fuel storage casks. The data also confirmed that spent nuclear fuel can be safely stored in dry casks.

Overpack Design

The thermal design features of the FuelSolutions overpack consist of the following:

1. Vertical storage overpack is made of modular precast, reinforced concrete.
2. Passive natural convection driven cooling is provided by vent openings to allow air to flow at the base of the cask, rise upward through the annulus between the canister and the thermal shield and between the thermal shield and concrete wall, and flow out the vent openings near the top of the cask.
3. The thermal shield and the convective air cooling maintain the temperatures of the concrete below allowable values.
4. A thermal shield, with a coating on one side only, maximizes the radiative heat transfer from the canister while minimizing the heat transfer to the concrete wall.
5. Thick carbon steel liner provides structural protection for the concrete and promotes axial heat conduction.
6. The size and location of the vents and flow annulus through the storage cask are designed to minimize expansion, contraction, and frictional flow pressure losses for the passive air that flows through the cask, thereby maximizing convective heat removal at lower differential air temperatures.
7. The inlet and outlet vent screens are sized to minimize resistance to air flow.

8. The canister is supported vertically above the bottom of the storage cask cavity by multiple metal tubes. These tubes provide paths for air flow to distribute under the canister and into the annulus with minimum resistance.
9. Centering guide rail system is used to maintain a minimum separation between the canister and the storage cask inner wall.
10. Two thermocouples are provided for storage cask temperature monitoring. One is located at mid-height and mid-thickness of the storage cask concrete wall, and the second thermocouple is located at mid-height at the liner/concrete interface. The thermocouples are monitored daily, as required by the technical specification to assure that the maximum short-term allowable surface concrete temperature (350°F) is not exceeded.

With respect to the design of the overpack, the internal air passage between the canister and the overpack inner surface provides the primary means for decay heat removal. Approximately 75% percent of the heat generated in the canister plus the solar heat on the overpack is removed through the air passage (see Figure 4-1). This heat removal is accomplished through natural convection cooling. The cooling is passive in that it uses differential air density buoyancy (also characterized as the chimney effect) to drive the air flow up the passage between the canister outer wall and the concrete overpack inner surfaces.

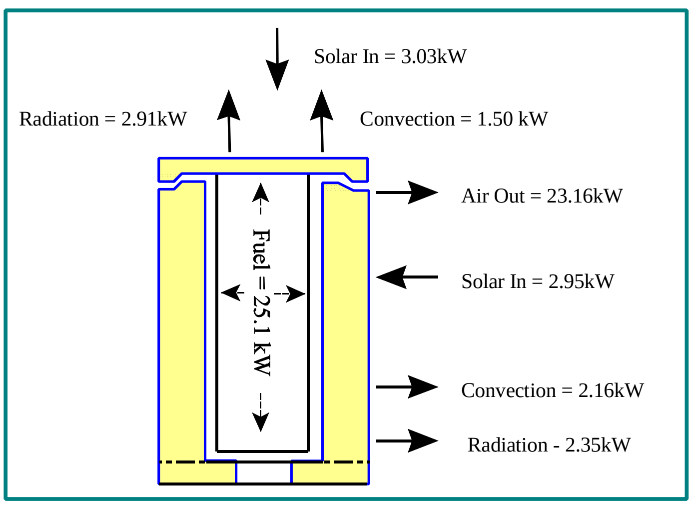

Figure 4-1: Design Basis Storage System Heat Balance

<u>Transfer Cask</u>

The thermal design features of the FuelSolutions transfer cask consist of the following:

1. Radiological shielding is provided by stainless steel inner and outer liners with lead between the liners, and a neutron shield consisting of an outer jacket forming an annular cavity that is filled with water.

2. Potential degradation in heat transfer due to the different thermal expansion characteristics of lead and steel is conservatively addressed by assuming a uniform gap at the interface between the outer radius of the lead surface and the steel.
3. The transfer cask is designed, analyzed, and fabricated in accordance with the applicable provisions of the American Society of Mechanical Engineers (ASME) Code Section III, Subsection NF.

Figure 4-2 illustrates the heat balance within the transfer cask at normal hot conditions for the W21 canister.

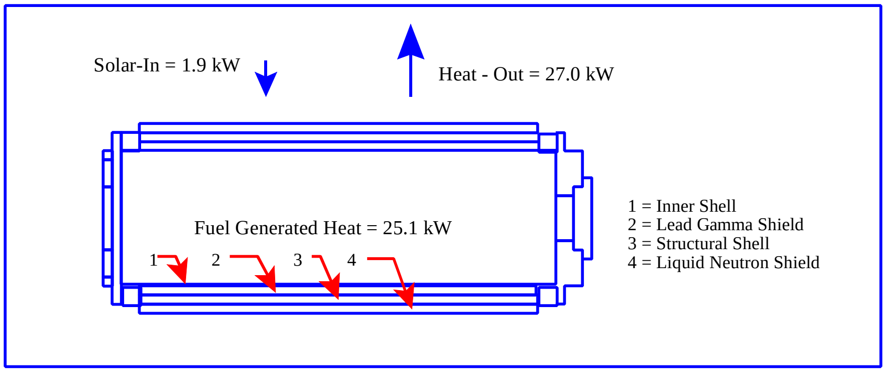

Figure 4-2: Transfer Cask Heat Balance Illustration

4.3 Thermal Load Specification/Ambient Temperature

The boundary conditions in the thermal analysis are specified in Section 4 of the WSNF-200, -201, and -203 SARs. The operating modes are identified in Tables 4.1-1 and 4.1-2 in the WSNF-200 and referenced in the WSNF-201 and -203 SARs. The off-normal and accident high ambient temperature cases include a solar insolation boundary condition on the top and side surfaces of the overpack.

__Canister Heat Load Limits__

The thermal load specifications for an overpack loaded with the canister are addressed in Sections 2 and 4 of the SAR. The maximum heat load, burnup and linear heat generation rate (LHGR) for the W-21 canister are 22.0 kW, 60,000 MWD/MTU, and 0.161 kW/in., respectively. The maximum heat load, burnup and LHGR for the W74 canister are 24.8 kW, 40,000 MWD/MTU, and 0.216 kW/in, respectively. For the storage and transfer casks, the maximum thermal rating is 28 kW and the maximum linear heat generation rate is 0.253 kW/in.

These limits on decay heat loads for the canisters are based on the calculated maximum cladding temperature limits for normal conditions. The thermal rating for the storage cask is based from the temperature limits for the concrete overpack. Solar thermal loads, as identified in 10 CFR 71.71, were incorporated into the analysis, as appropriate. The thermal loads apply to normal, off-normal, and accident conditions. During a postulated fire accident, the thermal load on the overpack includes the heat generated from the engulfing fire.

Storage System Array Spacing

The Storage Systems are designed to be placed in multiple cask arrays on an ISFSI pad (described in Section 1.4 of the SAR). A minimum cask spacing of 15 feet, center-to-center is established to provide for adequate ventilation flow and to provide access for handling, monitoring and maintenance.

Decay Heat Loads and Preferential Fuel Loading

The staff has reviewed and confirmed the design basis decay heat loads for the specific fuel designs. The staff has also verified that the bounding decay heats have been properly addressed in the analyses. With the assistance from PNNL, the staff confirmed that the applicant's decay heat for 60,000 MWD/MTU PWR fuel bounds the loading of lower burnup fuel with regards to establishing the maximum allowable temperature limits. Within the bounds of the analyses, the method employed by FuelSolutions does not impose thermal constraints on mixing low and high burnup fuel assemblies in one canister (known as preferential loading). Other non-thermal considerations may be used in optimization of fuel loading patterns (e.g., reduction of radiation dose from high-burnup fuel by placing low-burnup fuel on the periphery).

Ambient Temperatures

The ambient temperature boundary conditions assumed as design bounding values for the thermal evaluation of the FuelSolutions Canister Storage System are listed in Table 4-7.

Table 4-7
FuelSolutions Canister Storage System Design Ambient Temperatures

Condition	Temperature (°F)
Normal Annual Average	77
Normal Maximum Design Event Condition	100
Normal Canister Loading	120
Off-Normal Minimum Design Event Condition	-40
Off-Normal Maximum Design Event Condition	125

The heat transfer between the soil and the overpack was assumed to be negligible.

Postulated Fire Accidents

The applicant used limiting thermal boundary conditions for the postulated fire accident analyses in accordance with NUREG-1536. Seventy gallons of combustible fuel was assumed to feed the engulfing fire. Section 8.1.10.3, of the operating procedures limits the tow vehicle fuel supply to be less than or equal to 70 gallons. The duration of burn time for the 70 gallon fuel supply is 5 minutes. The applicant assumed a surface convection heat transfer coefficient of 3.8 Btu/hr-ft^2-°F for all exterior surfaces of the storage cask. Radiation heat exchange is based on a flame temperature of 1475°F, a flame emissivity of 0.9 (per 10CFR71.73(c)(4)), and a surface emissivity of 0.9 to yield an effective surface absorptivity coefficient of 0.883. For the postulated internal vent fire, fifteen gallons of the combustible fuel was conservatively assumed to accumulate to a depth of 1 inch beneath the cask within the air inlets. It is not credible that more than 15 gallons of fuel could accumulate in the air inlet without flowing out the bottom of the cask.

4.4 Model Specification

Configuration

The staff verified that the analytic model used to perform the thermal analyses was described in the SAR. With the assistance of the PNNL, an independent thermal computer model of the W21 storage system was developed using the information provided in the SAR. Applying various assumptions, including those used by The applicant, the PNNL calculations reproduced the calculated results in the SAR. The evaluations followed the guidance described in NUREG-1536.

Material Properties

The staff verified that the material compositions and thermal properties used in the safety analysis are addressed in the SAR and are appropriate. Of major interest in the staff's evaluation was The applicant's analytic modeling of natural circulation cooling by the helium gas inside the canister. The material properties and assumptions are discussed in Section 4.7 of the SAR. The analytic models were benchmarked with data obtained at INEEL on full scale spent fuel casks. Good agreement with the experimental data was obtained.

While not specifically calculating natural-circulation cooling within the fuel assemblies, the use of the Manteufel and Todreas[6] correlation incorporates data that includes some level of natural circulation cooling from helium gas. The staff's confirmatory analyses (see below) confirmed the acceptability of the Applicant's analytic modeling assumptions.

4.5 Thermal Analysis

4.5.1 Computer Programs

The thermal-fluid dynamic analysis was performed using the multidimensional SINDA/FLUINT computer code. This finite difference, lumped parameter code was developed under the sponsorship of the NASA Johnson Space Center. The SINDA/FLUINT computer code was used for the analysis and licensing of several transportation packages for nuclear material, including the RTG transportation package and the TRUPACT-II transportation packages.

A major feature of the SINDA/FLUINT code is the ability to use thermal sub-models to represent common geometry sections of the canister shell, guide tubes, spacer plates, etc. A thermal sub-model is defined as a thermal model which contains the necessary information to be independently solved for the temperatures of the components which it simulates, but which depends on one or more other thermal sub-models for some or all of its boundary conditions. Section 4 of the SAR provides details of the analytic model.

To demonstrate the applicability and use of the SINDA/FLUINT code, the applicant benchmarked the code with a full scale, 15kW fuel thermal output cask (VSC-17), instrumented with 98 thermocouples. The tests were conducted at the INEEL. The agreement between the SINDA/FLUINT predictions and the experimental test results was good.

The SINDA/FLUINT code was also benchmarked with the FLUENT computer code. The FLUENT computer code was approved for use by the staff on other spent fuel cask designs.

As addressed in Section 4.2.2 of this SER, helium backfill is used in the canister to optimize the heat transfer rate from the fuel to the canister wall and to provide an inert atmosphere to limit the potential for degradation of the internal components. The heat transfer effectiveness of the helium gas was demonstrated on full scale casks at the INEEL. However, the FuelSolutions Canister Storage System has one significant design difference when compared to the tested canisters. Spacers within the FuelSolutions canister limit the natural circulation of helium coolant to the distance between the spacers (e.g., natural convection does not flow from the bottom to the top of the cask outside the fuel assembly region, See Figure 4-3). This limits the buoyancy driving head that can influence the helium velocity. A lower buoyancy head can restrict the maximum velocity of the helium gas. A lower helium velocity has the potential for lowering the convective heat transfer rates. This issue led the staff to question the degree of natural circulation cooling provided in the FuelSolutions design and the ability of the vendor to adequately calculate the heat transfer and helium flow characteristics inside the FuelSolutions canister.

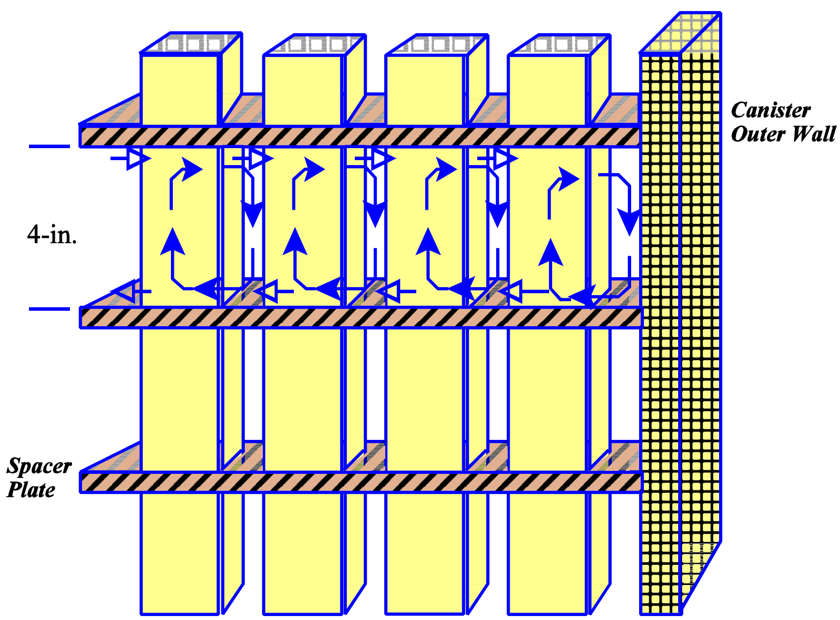

Figure 4-3 Flow Pattern for Vertical Canister

To resolve this technical issue, the staff, with the assistance of the PNNL, performed independent analyses using the COBRA-SFS computer code to augment the validation of the applicant's analytic conclusions. The COBRA-SFS computer code was validated with all of the spent fuel casks tested at the INEEL facility. The results of the COBRA-SFS and SINDA/FLUINT analyses are described in the following paragraphs.

Independent NRC Audit of the FuelSolutions Thermal Analysis

An independent confirmatory analysis of the FuelSolutions W21 Storage System was performed with the assistance of the PNNL using the COBRA-SFS computer software program. This confirmatory analysis was performed on a canister heat load of 25.1 kW (note that the SAR canister rating is 22 kW). The COBRA-SFS computer code solves the conservation equations of mass, momentum, and energy using finite difference equations derived from performing control volume balances on mass, momentum and energy. Empirical relationships are used when needed to close the set of equations. The code computes three-dimensional flow with two momentum equations; an axial momentum equation and a transverse momentum equation. The transverse momentum equation accounts for all momentum in the plane orthogonal to the axial momentum direction. The code also has the capability to model free-field three-dimensional Novier-Stokes flows.

The COBRA calculation modeled the helium gas flow through the fuel assemblies. The radial distribution of temperatures (hottest in the center assembly, coolest in the outer assemblies) results in a modest recirculation of the helium gas upward in the middle assemblies and downward in the outer assemblies. This feature was not explicitly included in the applicant's calculation. While The applicant did not specifically calculate natural-circulation cooling within the fuel assemblies, the use of the Manteufel and Todreas correlation incorporates data that includes some level of natural circulation cooling from helium gas.

PNNL investigated four cases:

1. First, the flow field between the horizontal spacer plates was analyzed as a solid material with enhanced heat transfer to account for natural circulation cooling. This case assumed a conservative heat transfer correlation of Nusselt number equal to 3.66 for the gaseous regions between the spacer plates. A peak clad temperature (PCT) of 799 °F (426 ^{o}C) was calculated.

2. Second, a Nusselt number similar to that employed by The applicant was modeled. In this case, the predictions were very similar, with a calculated PCT of 715 °F (379 ^{o}C).

3. Third, the gaseous region between a single set of spacers in the mid-axial region of the canister was modeled in the COBRA-SFS simulation. A Nusselt number of 3.66 correlation was used between all other spacer regions. The PCT for this case was 753 °F (401 ^{o}C).

4. Fourth, calculation number (3) above, was re-analyzed with assumed fission gases from the fuel rods mixed with the helium fill gas. This calculation (addressed later in this SER) is intended to assess the impact of reduced heat transfer coefficient and increased bulk gas density that results from breaching all the fuel rods within the canister.

A summary of the results is provided in Table 4-8.

Table 4-8
COBRA Parametric Results for the W21 Storage System Design

	BFS SAR	COBRA-SFS Calculated Results			
		Case 1	Case 2	Case 3	Case 4
Peak Clad Temp °F	715	799	715	753	780
°C	379	426	379	401	416

The COBRA-SFS model for the W21 canister was developed from the information presented in the SAR. Two base simulations (i.e., Cases 2 and 3) were performed. First a COBRA-SFS model was developed to treat the radial heat transfer in the regions between the horizontal spacer plates using a similar Nusselt number approach as applied by The applicant. This validated the staff's input model for COBRA-SFS code as providing similar results compared to the applicant's analyses. Next, although data exists for a range of spent fuel storage systems, there is no comparable full scale cask thermal performance data with multiple horizontal spacers, as designed into FuelSolutions. However, data for enhanced heat transfer due to natural convection between plates was identified by The applicant (e.g., Figures 4.4-4 and 4.7-5 in the WSNF-201 SAR). As an independent confirmatory calculation of the physics that exists in the FuelSolutions design, the staff requested PNNL to confirm, through detailed analyses, the thermal characteristics or response of the helium heat transfer mechanisms (e.g., flow and heat transfer rates) that occurs between the spacers.

The first base simulation (Case 2) COBRA-SFS model using the enhanced Nusselt number correlation in between the spacer regions duplicated the applicant's predictions extremely well. The results are presented in Figure 4-4. The radial temperatures are provided at the peak axial temperature level of the spent fuel cladding in the center assembly for all of the components, with the exception of the annulus temperatures, which represent the peak temperature of the flow field which occurs at the inner and outer annulus outlets. For simplicity, only a few points from the FuelSolutions SAR have been provided. Results from the second base simulation (Case 3) COBRA-SFS model are included and presented in Figure 4-5. The radial heat transfer coefficient of the gas between the spacers was treated with a Nusselt number of 3.66, with the exception of the single region that was nodalized in greater detail. It is projected that the peak temperature predicted by the COBRA-SFS model would approach the 715 °F predicted by the BFS model if the flow among all of the spacers was treated in a similarly detailed fashion. The staff, therefore, finds the thermal models used by the applicant acceptable.

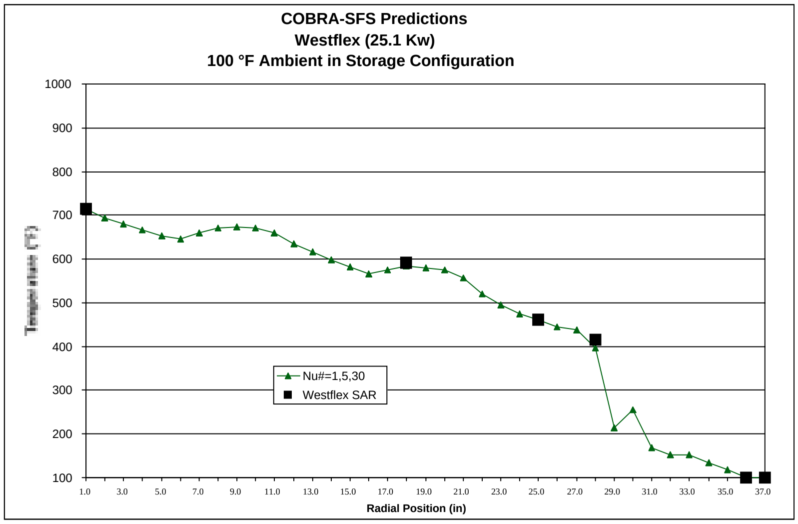

Figure 4-4: COBRA-SFS Predictions of FuelSolutions W21 Storage System Radial Temperatures using BFS Level of Enhanced Heat Transfer Between Spacers

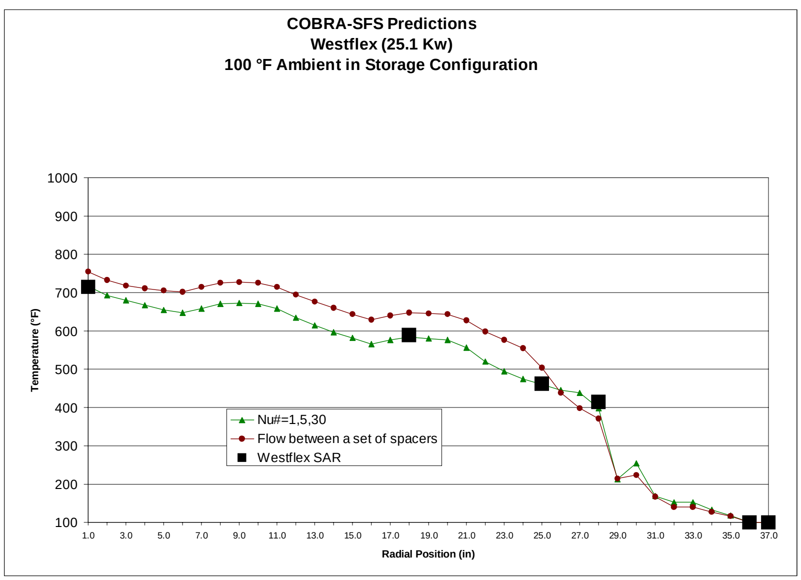

Figure 4-5: COBRA-SFS Predictions of FuelSolutions W21 Storage System Radial Temperatures using 1) BFS Level of Enhanced Heat Transfer Between Spacers and 2) Detailed Flow model Results.

4.5.2 Temperature Calculations

Normal Conditions

The results of temperature calculations for normal, off-normal, and accident conditions are presented in Section 4 of the SAR. The normal and off-normal temperature calculations were performed at the three normal and two off-normal ambient temperatures, as identified in Table 4-7 of this SER. The accident temperature calculations were performed for an extreme ambient temperature, as discussed in Section 4.3 of this SER, and for a hypothetical maximum fire enveloping a loaded FuelSolutions overpack. All cases assumed the maximum design basket-specific decay heat load. Key FuelSolutions system component temperatures under normal, off-normal, and accident conditions for both canister designs are summarized in Tables 4-9 and 4-10.

Table 4-9
Calculated Maximum FuelSolutions W21 System Component Temperatures at Normal Conditions

Component	Normal W21 °F	Normal Cold Storage W21 °F	Normal Hot Storage °F	Allowable Temperature °F
Fuel Cladding	643 (339.3°C)	632 (333.3°C)	662 (349.9°C)	662 (350.0°C)
Guide Tube	604	587	624	800
Spacer Plates: Stainless Steel Carbon Steel	 595 591	 576 571	 616 612	 800 700
Support Rod	417	368	440	650
Helium Bulk	437	397	458	N/A
Canister Shell	365	306	388	800
Maximum Concrete	152	52	177	200

Table 4-10 Calculated Maximum FuelSolutions W74 System Component Temperatures at Normal Conditions				
Component	**Normal W74 °F**	**Normal Cold Storage W74 °F**	**Normal Hot Storage W74 °F**	**Allowable Temperature W74 °F**
Fuel Cladding	705 (373.7°C)	668 (353.2°C)	724 (384.6°C)	726 (385.5°C)
Guide Tube	668	627	690	800
Spacer Plates: Stainless Steel Carbon Steel	 466 652	 408 608	 488 673	 800 700
Support Tube	540	485	563	800
Helium Bulk	473	415	495	N/A
Canister Shell	407	337	431	800
Maximum Concrete	150	49	175	200

Off-Normal Conditions

The thermal performance of the FuelSolutions canisters was analyzed at off-normal cold storage (-40°F) and hot storage (125°F) ambient conditions. For horizontal canister transfer conditions, the canisters were evaluated at -40°F ambient temperature, zero decay heat load, and no insolation. Tables 4-11 and 4-12 highlight the results for the off-normal events analyzed for the W21 and W74 storage systems, respectively.

The temperature profile for the canisters within the transfer cask was evaluated under vacuum drying, off-normal cold, and off-normal hot conditions. As addressed in Section 4.3.2.3, the use of the BFS creep model for deriving the maximum allowable peak cladding temperature under storage conditions imposes a short-term cladding temperature limit of 400°C.

Table 4-11 Storage Off-Normal Conditions for W21 Storage System			
Component	Off-Normal Cold Storage Peak Temp	Off-Normal Hot Storage	Allowable Temperature Limit
Peak Fuel Rod Cladding	598°F (314.5°C)	734°F (389.8°C)	752°F (400°C)
Guide Tube	550°F	696°F	1000°F
Spacer Plates: Stainless Steel Carbon Steel	 539°F 534°F	 687°F 683°F	 1000°F 1000°F
Support Tube	325°F	497°F	1000°F
Helium Bulk	357°F	515°F	N/A
Canister Shell	261°F	440°F	1000°F
Max. Concrete	6°F	211°F	350°F

Table 4-12 Storage Off-Normal Conditions for W74 Storage System			
Component	Off-Normal Cold Storage Peak Temp	Off-Normal Hot Storage	Allowable Temperature Limit
Peak Fuel Rod Cladding	664°F (334.2°C)	744°F (395.8°C)	752°F (400°C)
Guide Tube	589°F	738°F	1000°F
Spacer Plates: Stainless Steel* Carbon Steel	 367°F 571°F	 529°F 721°F	 1000°F 1000°F
Support Tube	444°F	608°F	1000°F
Helium Bulk	375°F	536°F	N/A
Canister Shell	291°F	471°F	1000°F
Max. Concrete	3°F	208°F	350°F

* Located at the top and bottom of the canister (W74 only)

To maintain the peak cladding temperature below 400°C during vacuum drying operations, helium may be periodically injected into the canister to cool the clad. This process is identified by the applicant as the “Cask Loading Cycle Method.” It is summarized in the following phases.

Phase 1-2: The spent fuel assemblies, immersed in water inside the canister/transfer cask, are removed from the spent fuel pool to a wash down pit or other work area where the canister closure is welded in place.

Phase 2-3: Water is drained from the canister and simultaneously replaced with helium. Water is maintained in the annulus to minimize fuel heat-up.

Phase 3-4: The remaining water in the canister is removed in a vacuum. If, left unattended at vacuum drying, a steady-state temperature of 473°C for the W21 canister, will be attained in excess of 30 hours. The vacuum drying procedure limits the initial vacuum drying period to 12 hours for the W21 canister and to 7 hours for the W74 canister. Subsequent vacuum drying (if needed) is to be performed using the 8-hour/4-hour vacuum/cooling cycle for the W21 canister and a 4-hour/4-hour vacuum/cooling cycle for the W74 canister.

Phase 4-5: 4-hour inspection of the canister prior to initiating the second vacuum drying cycle.

Phase 5-6: Perform a second vacuum drying cycle and a final helium backfill.

The applicant performed an assessment to quantify the heat loads that would result in a steady-state peak clad temperature below 400 °C under unattended vacuum conditions, thereby not requiring implementation of the Cask Loading Cycle Method. Tables 4-13-A and -B list those conditions for the W21 and W74 canisters, respectively.

Table 4-13-A
W-21 Canister Conditions Not Requiring Use of the Cask Loading Cycle Method

Canister Load	Heat Load (kW)	Steady-State Peak Cladding Temperature
Vacuum Drying w/annulus temperature ≤ 212°F (No Flow)	17.5 kW	397°C (747°F)
Vacuum Drying w/annulus temperature ≤ 120°F (5 gpm flow)	19.0 kW	399°C (750°F)
Vertical Handling	19.0 kW	386°C (727°F)

Table 4-13-B W-74 Canister Conditions Not Requiring Use of the Cask Loading Cycle Method		
Canister Load	Heat Load (kW)	Steady-State Peak Cladding Temperature
Vacuum Drying w/annulus temperature ≤ 212°F (No Flow)	12.2 kW	400°C (752°F)
Vacuum Drying w/annulus temperature ≤ 120°F (5 gpm flow)	13.2 kW	400°C (752°F)

Heat Transfer Rate of Helium With Gases from Failed Fuel Pins

Under accident conditions, 100% of the fuel pins are assumed to experience pinhole leaks as a bounding assumption for calculating cansiter pressure and heat transfer. The staff requested the applicant to address the reduction in the heat transfer rate inside the canister as the helium gas is diluted with gases escaping from the fuel pins to the canister. In response to the staff's inquiry, the applicant stated that an increase in the density of the gas mixture will increase the Rayleigh number by the square of the density ratio, while an increase in dynamic viscosity will have the predictable effect of decreasing the convection flow and, hence the Rayleigh number (the Rayleigh number is used to calculate the convective heat transfer coefficient). In addition, a lower thermal conductivity will increase the Rayleigh number and, hence the strength of the convection flow. A lower thermal conductivity will tend to prevent the gas stream from giving up its thermal energy to the surroundings by conduction. This is one reason why a nitrogen (versus helium) environment is seen as providing higher levels of convective flows, while still yielding higher component temperatures. Since the basket designs for W21 and W74 canisters permit convection heat transfer to occur nearly everywhere within the basket and under all orientations and ambient conditions, the net effect of a fission gas release will be an increase in the overall thermal performance of the canister, not a decrease. An analysis performed by the applicant illustrated that for hot normal storage conditions, the peak fuel rod cladding temperature for the no failed fuel case was 380°C, and the peak fuel rod cladding temperature for a canister with failed fuel was 352°C.

To confirm the applicant's responses to this issue, the staff requested PNNL to perform independent confirmatory calculations using the COBRA-SFS computer code. The purpose of the analysis was to assess the net contribution of fission product gases on the calculated peak clad temperature. The PNNL calculations were unable to confirm the applicant's evaluation that compensatory buoyancy forces enhance the thermal performance of the canister. The PNNL analysis validated the proposition that gas flow velocities increased for higher density gases. However, the analysis was unable to confirm the claim that the increased flow velocity overcompensates for the reduction in gas convective heat transfer coefficient such that the net effect is a decrease in peak clad temperature. The PNNL results for the FuelSolutions W21 canister calculation resulted in a peak cladding temperature increase of 20 °F. This temperature increase was negligible and the staff has reasonable assurance that any degradation in gas convective properties will not result in gross cladding failures, as stipulated in 10 CFR Part 72.

Thermal Evaluation for Accident Conditions of Storage

Section 4.6 of the SAR addresses the consequences of postulated accidents. The analyses included the blockage of all air vents and fire. Two postulated fire events were calculated. One calculation assumed that the fire engulfed the Storage System. The second calculation assumed the fire burned within the Storage System inlet vents. The duration of the fire was 5 minutes. Section 8.1.10 of the operating procedures limits the amount of combustible fuel near the Storage System to 70 gallons, "The tow vehicle fuel supply shall not exceed 70 gallons." The burning duration for 70 gallons of fuel is conservatively calculated as 5 minutes.

Vent Blockage Accident

For the vent blockage accident event, the analysis assumes that the inlet and outlet vents become fully blocked. An internal air flow distribution is developed by this blockage, wherein the air flow circulates between the canister, the thermal shield and the steel liner of the overpack. The strength of this internal flow is computed within the overpack thermal model as a function of the density differences between the inner and outer air columns. The analysis assumed blockage durations of 80 hours and 70 hours for the W21 and W74 Storage Systems, respectively. The temperature rise in the canister and overpack component is primarily a function of the thermal mass of the component since heat loss to ambient is greatly reduced for this condition. The maximum overpack concrete temperature is the controlling component for the blockage event. The analysis demonstrated that upon reaching steady-state temperature conditions, neither the fuel cladding nor the canister component allowable temperature limits would be exceeded.

The maximum cask liner temperature predicted to occur after 60 hours is 339 ^oF for the W21 overpack and 345 ^oF at 58 hours for the W74 overpack. This is below the 350^oF allowable temperature limit. For both the W21 and W74 casks, the maximum concrete temperature is 350^oF at 60 and 58 hours, respectively. Assuming a heat load of 28 kW (design limit for the overpack), the short-term allowable concrete surface temperature (350^oF) is reached after 41 hours for the rated Q_{max} (28kW) and 55 hours for the rated $LHGR_{max}$ (0.253 kW/in). The TS require daily monitoring and corrective action to mitigate any vent blockage conditions. This assures that the concrete will not exceed its temperature limit.

Fire Accident

Storage Cask Fire

The applicant presented results of two fire analyses in their SAR. One analysis assumed that the fire engulfed the Storage System. The second analysis assumed the fire was within the Storage System inlet vents. Both events assumed a 5-minute fire duration. The analysis simulates 70 gallons of combustible fuel.

For the fire analyses, less than two-inches of concrete exceeded the short-term allowable concrete temperature of 350 ^oF. This transient thermal response of a small fraction of the concrete is allowable for a fire condition. Both fire analyses assumed an initial steady state temperature distribution for normal hot storage conditions (100^oF ambient temperature). The engulfing fire assumed a natural convection flow upward through the annulus induced by the buoyancy driving forces. Since the air/gas temperature becomes hotter than the canister heat shield and overpack liner temperatures, the gas inside the storage overpack annulus reverses direction. Hot combustion gas flows into the annulus gap from the top vents, and exits the overpack through the bottom channels. The downward gas flow transfers heat to the canister shell and the overpack liner until the fuel is depleted and normal ambient air flow is established. In accordance with NUREG-1536 and 10CFR71.73(c)(4), an average emissivity coefficient of 0.9 and an average flame temperature of 1475°F were assumed for the duration of the 5-minute fire burn with the flame limited to within 10 feet of the external surface. A maximum flame velocity of 15 m/s was used. This results in a high convective heat transfer coefficient of 3.8 Btu/hr-ft^2-°F on the exterior surfaces of the overpack.

For the inlet vent fire, the combustible fuel is assumed to burn within the overpack inlet vents. The temperature of the air within the overpack inlet vents and the base section bottom penetrations was set to the fire temperature 1475°F. The 1475°F gases rose along the vertical annulus gaps and exhausted from the outlet vents. For the W21 and W74 canisters, the calculated peak temperatures are summarized in Table 4-14.

Table 4-14
Storage Cask Maximum Temperatures from Postulated Fire Accidents

Component/Location	Engulfing Fire	Inlet Vent Fire	Allowable Material Temperature (°F)
Canister Shell	475	478	1000
Inner Air Gap	1337	1391	n/a
Thermal Shield	530	558	600
Outer Air Gap	1382	1370	n/a
Liner	205	202	1000
Surface Concrete	744 (Cask Side Wall Between Inlet Vents) 916 (Corner of Inlet Vents)	200 (Cask Side Wall at Liner Interface) 269 (Concrete in the Inlet Duct)	350

Less than two inches of the concrete surface exceeded the allowable short-term surface material temperature limit. Therefore, the storage system will adequately withstand the effects of a postulated fire accident. Section 11.2.5.4 in WSNF-200 identifies the corrective actions following a postulated fire. Table 4-15 lists the corrective actions for a postulated fire event.

Table 4-15 Corrective Actions For The Concrete Overpack Following A Postulated Fire	
Concrete Damage Following a Postulated Fire	Action Required
Hairline cracks or no visible damage	No action required
Significant local cracking or minor spalling	Repair the storage cask concrete in place using approved procedures
Significant surface cracking and substantial spalling, little or no exposed reinforcing steel	Retrieve the canister to the transfer cask, repair the storage cask concrete using approved procedures, then return the storage cask to service
Substantial spalling and concrete damage below the depth of the reinforcing steel	Remove the storage cask from service

Due to the large thermal capacitance of the canister, the temperature rise within the canister during the 5 minute fire is much lower than the shell. At the end of the fire event, normal ventilation at ambient air temperature is resumed. The peak clad temperature remains below the short-term thermal limit of 570^{o}C.

4.5.3 Pressure Analysis

Section 4 of the SAR addresses the FuelSolutions canisters calculated pressures for normal, off-normal, and accident conditions. The maximum internal pressure was calculated using the free volume of the canister, and the ideal gas law, accounting for the backfill helium gas along with a fraction of the stored fuel helium fill gas and fission product gas. The normal, off-normal, and accident conditions were differentiated by the assumption of the fraction of stored spent fuel which contributed to fill gas and fission gas to the canister. These fractions were 1%, 10%, and 100%, respectively for the normal, off-normal, and accident cases, which are in agreement with NUREG-1536. The quantity in moles of inert gas needed for canister cavity backfill is determined in order to achieve 10 psig in the canister cavity under normal hot storage conditions (e.g., 100 ^{o}F ambient temperature) with 1% rod failure. The maximum canister internal pressure is determined assuming rupture of 1%, 10%, and 100% of the fuel rods under normal, off-normal and accident conditions, respectively. The pressure calculations assumed the release of 100% of the fill gas and 30% of the rod fission gas for each postulated failed fuel rod. The resulting W21 and W74 pressures are summarized in Table 4-16.

Table 4-16 Calculated Maximum Canister Pressures for Normal, Off-Normal, and Accident Conditions			
Condition	**W21 Calculated Canister Pressure, 60 GWd/MTU (psig)***		**W74 Calculated Canister Pressure 40 GWd/MTU BRP Fuel (psig)**
	w/o Control Components	**w/ Control Components**	
Normal	10.0	10.0	10.0
Off-Normal (Max)	15.4	15.9	12.3
Accident (Max.)	64.0	68.7	30.0

* Canister pressures are less for lower fuel burnup.

The calculated maximum pressure for both canister designs under all postulated conditions remains below its appropriate design pressure.

4.6.1 Transfer Cask

The FuelSolutions Transfer Cask is a right circular cylindrical vessel with covers on both ends. It is designed to support a variety of FuelSolutions canister transfer operations. The transfer cask is used to handle canisters during canister fuel loading, canister closure, and on-site transport operations. It also provides the capability for transferring loaded canisters to or from the storage cask or shipping cask in a horizontal or vertical orientation. The transfer cask is composed of a stainless steel inner liner and an outer structural shell, with lead gamma shielding in the annular space between them (see Figure 4-2). A neutron shield, consisting of an outer jacket forming an annular cavity that is filled with water, surrounds the structural shell. The structural shell of the transfer cask is sized to accommodate localized loads from the lifting trunnions and other design basis loadings. In order to maintain the weight of the loaded transfer cask at less than 100 tons, the neutron shield cavity may be empty for operations in the spent fuel pool and refilled after the loaded cask is placed in the decontamination area. All exposed surfaces of the transfer cask are constructed of polished stainless steel to ease decontamination.

Heat removal from the Transfer Cask is accomplished primarily by conduction through the walls of the cask. In addition, the liquid in the neutron shield facilitates heat conduction and convection to maintain fuel cladding and structural component temperatures within allowable values. The outer surface of the liquid neutron shield jacket is provided with a high emissivity, low absorptivity coating to enhance radiation heat transfer from the cask while minimizing the effects of insolation. The transfer cask is also equipped with a thermocouple probe that is attached to the outer surface of the transfer cask structural shell at mid-height of the cask. This thermocouple is used to monitor the cask temperatures following canister closure operations to provide a direct indication that system temperatures are within acceptable values.

During canister reflooding and opening operations, the canister shell can be cooled within the transfer cask by circulating clean water through the annulus between the transfer cask and the canister shell. Recirculation and cooling of the water are accomplished by plant make-up cooling located near the top and bottom ends of the cask. The cask port locations are offset circumferentially by 180 degrees to provide increased water circulation. In addition, an auxiliary drain connection through the cask bottom cover may also be used.

The thermal analyses of the transfer cask were performed using the SINDA/FLUINT computer program. As previously addressed in this SER, the staff performed independent confirmatory analyses of the FLUINT computer code and found it acceptable.

<u>Thermal Rating of the Transfer Cask</u>

To assess the thermal rating of the transfer cask, the applicant varied the heat loads with the maximum thermal profile until the maximum allowable temperature of 620°F was reached for the lead shielding. The lead shielding reached the allowable temperature limit prior to any other thermal criteria for the cask components. Table 4-17 summarizes the material temperature limits for the transfer cask. The transfer cask heat load rating which corresponds to this allowable lead shield temperature is 31 kW. However, since the storage cask is more thermal limiting, a transfer cask maximum allowable heat load of 28 kW is established.

Table 4-17
Transfer Cask Allowable Material Temperatures

Material	Maximum Average Temperature (°F)	Maximum Local Temperature (°F)	Maximum Local Short-Term Temperature (°F)	Minimum Average Temperature (°F)
Lead Shielding	620	620	620	n/a
Liquid Neutron Shielding	293	n/a	n/a	32
NS-3/RX-277 Solid Neutron Shielding	n/a	350	n/a	n/a
Rubber O-Rings	250	250	250	n/a
Structural Stainless Steel	800	800	1000	n/a
Neutron Shield Jacket	325	325	325	n/a

Similar analyses were performed for establishing the maximum linear heat generation rate. The transfer cask heat load rating which corresponds to this allowable lead shield temperature with the maximum thermal gradient profile is 24.0 kW. However, since the storage cask is more thermally limiting, the applicant set the transfer cask maximum allowable heat load to 21.0 kW, with the maximum thermal gradient profile. Table 4-18 summarizes the transfer cask thermal rating.

Table 4-18
Transfer Cask Thermal Rating

Design Conditions	Q_{max} (kW)	$LHGR_{max}$ (kW/in)
Maximum Thermal Profile	28.0	0.204
Maximum Thermal Gradient Profile	21.0	0.253

Normal and Off-Normal Transfer Cask Evaluations

The thermal model used in the evaluation of the transfer cask for off-normal conditions is the same as that used for normal conditions. The off-normal conditions evaluated include off-normal cold transfer (e.g., -40°F ambient temperature) and off-normal hot transfer (e.g., 125°F ambient temperature). The maximum material temperatures for the off-normal conditions are summarized in Table 4-19.

Table 4-19
Transfer Cask Maximum Material Temperature for Off-Normal Conditions

Cask Component	Maximum Thermal Profile °F		Allowable Material Temperature (°F)
	Cold Transfer (28kW)	Hot Transfer (28kW)	
Canister Shell	575	627	1000
O-Rings	41	195	250
Peak Lead	258	360	620
Average Lead	208	321	620
Peak Neutron Shield Fluid	147	272	293
Average Neutron Shield Fluid	120	252	293
RX-277 / NS-3	100	252	350
Stainless Steel	263	365	1000
Thermocouple	151	274	-

Transfer Cask Loss of Neutron Shield Event

The limiting accident events for the transfer cask are the loss of a liquid neutron shield and the postulated fire accident. The thermal parameters for these accident conditions are defined in Section 4.6 in the SAR. For the loss of neutron shield accident event, a steady-state analysis was performed at the cask thermal rating. The only analytic difference between the normal and accident transfer models is that the accident analysis assumes that the liquid neutron shield is empty (filled with air at one atmosphere, instead of water). Heat transfer within the empty liquid neutron shield is through radiation and convection. Table 4-20 summarizes the material temperatures for the loss of a neutron shield accident.

Table 4-20
Transfer cask Maximum Material Temperatures for Loss of Neutron Shield Accident

Cask Component	Temperatures ^oF		Allowable Material Temperature (^oF)
	Qmax = 28 kW	$LHGR_{max}$ = 0.253 kW/in	
Canister Shell	750	793	1000
O-Rings	225	165	250
Peak Lead	568	577	620
Average Lead	482	402	620
RX-277-NS-3	286	256	350
Stainless Steel	573	583	1000
Thermocouple	514	471	-

Transfer Cask Fire Event

Section 4.6.2 of the SAR evaluates the consequences of a postulated fire for the transfer cask. The applicant used limiting thermal boundary conditions for the postulated fire accident analysis in accordance with NUREG-1536 and 10CFR71.73. The analysis assumed seventy gallons of combustible fuel fed the engulfing fire. Section 8.1.10.3, of the operating procedures limits the tow vehicle fuel supply to be less than or equal to 70 gallons. The fire is postulated to burn for 5 minutes.

The initial conditions for the analysis assumed a steady-state operation at the cask thermal rating of 28kW, an empty liquid neturon shield filled with air at 1 atm., and an ambient temperature of 100 ^oF (hot transfer). Heat transfer within the empty liquid neutron shield is provided by radiation and convection. The coating on the transfer cask neutron shield outer shell is rated for 325^oF and was postulated not to survive. A coating emissivity of 0.85 is conservatively assumed during the fire event. The transfer cask neutron shield shell post-event outer surface emissivity is assumed to be 0.8 since the external surface of the cask will be darkened by soot during the fire. The transfer cask is assumed to be engulfed in a hydrocarbon fire/air with an average emissivity of 0.9 and an average flame temperature of 1475°F. Convective heat transfer from the flame to the cask occurs via a forced convection flame velocity of 15 m/sec.

For this event, the solid neutron shield material exceeds its allowable material temperature in localized regions. The liquid neutron shield outer shell exceeds the 1000°F short-term allowable temperatures, but remains well below the material melting temperature of 2600°F. All stainless steel material in the transfer cask remains below the 1000°F short-term stainless steel allowable temperature during the fire transient. The 325°F allowable temperature for the neutron shield jacket coating does not apply during the fire accident, since the coating is not assumed to be present following the event. Similarly, the O-ring allowable temperature of 250°F does not apply to the fire event since the O-rings are not required to maintain a seal. Table 4-21 summarizes the temperature history for the fire accident event.

Table 4-21
Transfer Cask Maximum Material Temperature for the Fire Accident Event

Component	Pre-Fire Steady-State Temperatures (°F)	Fire Transient (5-minutes) Maximum Temperature (°F)	Post-Fire Cool-Down/Steady-State Temperatures (°F)	Short-Term Allowable Temperature (°F)
Canister Shell	618	618	753	1000
Cask Inner Shell	347	347	574	1000
Cask Lead Shield	344	344	571	620
Cask Outer Shell	260	279	522	1000
Liquid Neutron Shield Skin	246	1136	260	1000
Top Flange Joint	233	568	270	1000
Bottom Flange Joint (Seal Location)	171	516	204	1000

Transfer Cask Analyses for the W21 and W74 Canisters

Previous discussions address the design limits for the transfer cask. These design limits were based on a decay heat load (e.g., 28 kW) that a fuel canister can withstand without exceeding the material property temperature limits. These design limits were based on the maximum decay heat loading (e.g., 28 kW) that the transfer cask can withstand without exceeding the material property temperature limits. As discussed in Sections 4.1 and 4.2 above, the maximum decay heat loads for the W21 and W74 canisters are 22.0 kW and 24.8 kW, respectively. The off-normal peak clad temperature limit is 400 ^{o}C (752 ^{o}F) for both canisters. The short-term peak clad temperature limit for accident conditions is 570^{o}C (1058 ^{o}F) for both canisters.

Analyses for the transfer cask with W21 and W74 canisters were performed in the WSNF-201 and WSNF-203 SARs. Tables 4-22, 4-23 and 4-24 summarize the peak cladding temperature for normal, off-normal and accident conditions, respectively, for the transfer cask with W21 and W74 canisters. The analyses were performed with a heat load in excess of the maximum canister rating, as identified above. Even with a heat load greater than permitted for the respective W21 and W74 canisters, the applicant demonstrated that the maximum allowable short-term temperature limit is not exceeded for either canister. The staff finds the analyses acceptable.

Table 4-22 Transfer Cask Calculated Cladding Temperatures for Normal Conditions

	Normal Transfer	Normal Cold Transfer	Normal Hot Transfer	Temperature Limit
W21 Peak Clad Temperature (25.1 kW Heat Load)	379 ^{o}C	363 ^{o}C	382 ^{o}C	400 ^{o}C
W74 Peak Clad Temperature (26.4 kW Heat Load)	394 ^{o}C	377 ^{o}C	398 ^{o}C	400 ^{o}C

Table 4-23 Transfer Cask Calculated Cladding Temperatures for Off-Normal Conditions

	Off-Normal Cold Transfer	Off-Normal Hot Transfer	Temperature Limit
W21 Peak Clad Temperature (25.1 kW Heat Load)	356 ^{o}C	387 ^{o}C	400 ^{o}C
W74 Peak Clad Temperature (24.8 kW Heat Load)	369 ^{o}C	389 ^{o}C	400 ^{o}C

Table 4-24 Transfer Cask Calculated Cladding Temperatures for Accident Conditions				
	Loss of Neutron Shield Event	Fire Event (28 kW Heat Load)		Short-Term Temperature Limit
		Pre-Fire Event Steady-State Temperature	Maximum Fire Event Temperature	
W21 Peak Clad Temperature	454 ^oC (25.1 kW Heat Load)	406 ^oC	454 ^oC	570 ^oC
W74 Peak Clad Temperature	465 ^oC (26.4 kW Heat Load)	406 ^oC	454 ^oC	570 ^oC

Summary

The staff has reviewed the thermal evaluations of the FuelSolutions Transfer Cask and finds them acceptable.

4.6.2 Reflood Analysis During Fuel Unloading Operation

In the event that the canister needs to be unloaded after an extended period in dry storage, the fuel and the canister will be cooled by injecting water into the canister prior to opening the canister and unloading the fuel. During canister injection or reflooding operations, quench water is introduced into the cavity through the canister drain line. The initial water flow is set at 5 gpm and at an inlet temperature of 100 ^oF. The maximum permissible flow rate is 10 gpm. The 10 gpm maximum quench flow rate assures that only one spacer plate is generating steam at a given time. The quench water is introduced into the cavity through the bottom closure plate and flashes to steam. The saturated steam rises through the canister cavity to the open vent port at the top of the canister. During this process, the saturated steam becomes superheated as it comes in contact with the hot fuel and canister internals. The superheated steam exits through the open canister vent and exhausts to a heat sink, typically the plant's spent fuel pool. As the water level steadily rises inside the canister, localized boiling at the top surface of the water is expected to occur because of the large inventory of heat that is stored in the canister internals.

Following reflooding, the canister may be further cooled by continuing the operation of the transfer cask annulus cooling. The post reflood cooldown is evaluated by a licensee on a site-specific basis. The plant-specific considerations that go into this evaluation include: canister's SNF decay heat, ambient conditions, water conditions, site-specific temperature limitations, and the lead time needed to initiate annulus cooling operations to prevent canister water boiling.

The applicant used the HEATING 7.2f computer code to determine the transient steam generation rate for the bottom closure plate. The steam venting rate is determined using the

RELAP5 computer code. The first phase of the reflood tranisent is the initial quench water impingement on the canister cavity bottom closure plate. For steam generation analysis, a stagnant pool of saturated water is assumed to be present above the bottom closure plate at the start of the transient. This is conservative since heat input to the water is neglected. Additionally, the subcooling effect of inlet water at 10 gpm is sufficent to overcome the heat flux from the bottom closure plate, assuming no interruption of the quench flow. Heat input from the canister internals directly above the bottom closure plate is considered. The entire bottom closure plate transient is conservatively evaluated at the lowest possible saturation temperature (212 ^{o}F), and the heat flux reduction for temperatures in the transition boiling regime is ignored.

As the water level rises inside the canister, the internals and the fuel are cooled. The limiting basket components considered in the analysis are the large spacer plates located throughout the axial length of the cansiter. The carbon steel spacer plates are the bounding components for steam generation during the remaining canister reflood transient. The carbon steel spacer plates have a higher conductivity and higher ratio of surface area to mass than the stainless steel plates and therefore release heat at a much faster rate.

The mass balance between the steam generation rate and the vent capacity during the bottom closure plate quench transient results in a maximum cansiter pressure of 12 psig, which is well below the 100 psig maximum reflood pressure. The reflooding process yields a series of transient pressure increases due to spacer plate quenching with rising cavity water level. The mass balance for the limiting spacer plate quench transient results in a peak canister pressure of 70 psig, which is also well below the 100 psig maximum reflood pressure.

Based on the applicant's analyses, the staff finds acceptable the reflooding of the canister prior to unloading of the fuel. In addition, the staff notes that similar procedures have been applied to other casks and demonstrated successfully.

4.7 Evaluation Findings

10 CFR Part 72 requires an analysis and evaluation of the dry cask storage system thermal design and performance to demonstrate that the cask will permit safe storage of the spent fuel for a minimum of 20 years. This section reviewed the performance of the long-term storage overpack (FuelSolutions W21 and W74 Canister Storage Systems) and the associated spent fuel transfer cask used to load and unload the dry canister storage system and for various plant operations, such as onsite transport of SNF, including loading and unloading operations of SNF in the canister. The staff concludes that the FuelSolutions Canister Storage System design fulfills the following acceptance criteria:

1. Fuel cladding temperature at the beginning of the dry cask storage is below the anticipated damage-threshold temperatures for normal conditions.

2. Fuel cladding temperatures (zircaloy) are maintained below 570 ^{o}C (1058 ^{o}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).

3. The maximum internal pressure of the cask remains within the design pressures for normal, off-normal, and accident conditions assuming rupture of 1%, 10%, and 100% of

the fuel rods, respectively. Assumptions for pressure calculations include release of 100% of the fill gas and 30% of the significant radioactive gases in the fuel rods.

4. Cask and fuel materials are 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.

5. For each fuel type proposed for storage, the dry cask storage system provides reasonable assurance that the degradation will not lead to gross ruptures, or that the fuel will be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage.

6. The cask system is passively cooled.

7. The thermal performance of the cask is within the allowable design criteria specified in Section 2 (e.g., materials, decay heat specifications) and Section 3 (e.g., thermal stress analysis) of the SAR for normal, off-normal, and accident conditions.

The following summarizes the staff's finding regarding the thermal evaluation of the FuelSolutions Canister Storage System:

F4.1 SSCs important to safety are described in sufficient detail in Sections 1.2 and 2.3 of the SAR to enable an evaluation of their thermal effectiveness. SSCs important to safety remain within their operating temperature ranges.

F4.2 The FuelSolutions W150 Storage Cask (overpack) with the loaded W21 or W74 cansiter is designed with a heat-removal capability that is verifiable and reliable, consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems.

F4.3 The staff finds, in accordance with 10 CFR 72.122(h), that the spent fuel cladding is protected against degradation leading to gross ruptures by maintaining the cladding temperature for zircaloy clad below the temperature limits. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F4.4 The staff concludes that the thermal responses of the W21 and W74 Canister Storage Systems, described in the SARs are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The thermal evaluations of the FuelSolutions Canister Storage Systems provide reasonable assurance that the FuelSolutions overpack with the loaded W21 or W74 canister will allow safe handling and storage of spent fuel for a certified life of 20 years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

4.8 References

1. H. Spilker, et al, “Spent LWR Fuel Dry Storage in large Transport and Storage Casks after Extended Burnup,” Journal of Nuclear Materials, Vol. 250, pp. 63-74, 1997.

2. Memorandum to: M. W. Hodges from K. Gruss, "PNNL Technical Evaluation Report on Cladding Behavior for High Burnup Fuels," with attachment by E.R. Gilbert, C.E. Beyer, and E.P. Simonen, Pacific Northwest National Laboratory, “Technical Evaluation Report of WCAP-15168 (Dry Storage of High Burnup Spent Fuel)”, February 2000.

3. I.S. Levy, et al., Pacific Northwest Laboratory, “Recommended Temperature limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas,” PNL-6189, May 1987.

4. D.D. Lanning et al., Pacific Northwest National Laboratory, “FRAPCON-3: Modifications to Fuel Rod Material Properties and Performance Models for high Burnup Applications,” NUREG/CR-6534, Vol. 1 (PNNL-11513, Vol. 1), 1997.

5. A.B. Johnson, Jr., and E.R. Gilbert, Pacific Northwest Laboratory, “Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases,” PNL-4835, September 1983.

6. Manteufel, R.D., and Todreas, N.E., Effective Thermal Conductivity and Edge Conductance Model for a Spent Fuel Assembly, Nuclear Technology, Vol. 105, pp. 421-440, March 1994.

5.0 SHIELDING EVALUATION

The shielding review evaluates the capability of the FuelSolutions™ Storage System shielding features to provide adequate protection against direct radiation from its contents. This review included the calculation of the dose rates from both photon and neutron radiation at locations near the cask and at specific distances away from the cask. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d)[1, 2]. An overall assessment of compliance with 10 CFR Part 72 dose limits for members of the public is discussed in Section 10 (Radiation Protection) of the SER and includes direct radiation, effluent releases, and radiation from other uranium fuel-cycle operations.

5.1 Shielding Design Features and Design Criteria

5.1.1 Shielding Design Features

The principle shielding components of the FuelSolutions™ Storage System are a concrete storage cask, a steel-lead-water composite transfer cask, and a shielded canister. The storage cask is designed to provide both photon and neutron shielding. There are two versions of the concrete cask, a long version and a short version. The difference between the two being the length of the center section of the concrete cask. The effective radial and axial shielding of both models are the same. The principal components of the storage cask radial shielding are the 2.0-in thick carbon steel liner and 30.5-in thick reinforced concrete body. The shielding at the top of the storage cask consists of 0.25-in thick steel, 12.75-in thick concrete, and 1.25-in thick steel. The shielding at the bottom of the storage cask consists of 1.0-in thick steel and 18.0-in thick concrete.

The canisters provide radial and axial shielding. Both canister shells are 0.63-in thick stainless steel. The axial top shielding consists of an inner and outer closure plate, an optional steel top plate, a shield plug, which varies among steel, lead, and depleted uranium depending on the canister design, and an optional steel bottom plate. When the shield plug is a material other than steel, the top and bottom steel plates are used to encase the shield plug. The shielding through the bottom of the canister consists of a steel end plate, shield plug, and a closure plate.

The W21 canister has two classes of canister, W21M and W21T, differing in materials of construction used for the canister shell and basket assembly. Each class of canister has four different types. The W21T canister class consists of a long lead (LL), long steel (LS), short lead (SL), and short steel (SS) canister. The W21M canister has long, depleted uranium (LD); long steel (LS); short, depleted uranium (SD), and short steel (SS) designs. Details of the W21 canister shielding are provided in Table 5-1 below.

The W74 canister has two classes of canister, W74M and W74T, differing in materials of construction used for the canister shell and basket assembly. Each canister class has only a long steel (LS) design. The shielding provided by this design includes a 5.8-in thick steel shield plug on the bottom encased in a 1.0-in thick closure plate and a 1.8-in thick steel end plate. The top of the canister has a 1.0-in thick inner closure plate and a 2.0-in thick outer closure plate. The top shield plug is 7.25-in thick steel.

Table 5-1 W21 Canister Details								
Class	W21M				W21T			
Type	-LD	-LS	-SD	-SS	-LD	-LS	-SL	-SS
Radial Shell	0.63-in Stainless steel (all Types)							
Top Closure Details								
Top Closure Plate	2.0-in Stainless steel (all Types)							
Inner Closure Plate	1.0-in Stainless steel (all Types)							
Shield Plug (Top Sheet)	0.12" SS	N/A	0.12" SS	N/A	0.12" SS	N/A	0.12" SS	N/A
Shield Plug	2.1" DU	7.25" SS	1.3" DU	7.25" SS	3.4" Lead	7.25" SS	3.8" Lead	7.25" SS
Shield Plug (Bottom Sheet)	1.6" Steel	N/A	3.6" Steel	N/A	1.6" Steel	N/A	1.6" Steel	N/A
Bottom Closure Details								
Closure Plate	1.0" Steel	1.0" Steel	1.6" Steel	1.0" Steel	1.0" Steel	1.0" Steel	1.0" Steel	1.0" Steel
Shield Plug	2.1" DU	5.8" Steel	1.9" DU	5.8" Steel	3.1" Lead	5.8" Steel	3.1" Lead	5.8" Steel
End Plate	1.8" Steel	1.8" Steel	1.8" Steel	1.8" Steel	1.0" Steel	1.8" Steel	1.8" Steel	1.8" Steel

The transfer cask has different materials to provide gamma and neutron shielding. The radial gamma shielding is provided by a composite structure of steel-lead-steel. The inner steel shell is 0.75-in thick and the outer steel shell is 1.5-in thick. The lead shielding is 3.4-in thick. The transfer cask radial neutron shielding is provided by 3 inches of water. The shielding at the top and bottom of the transfer cask consists of 2.75-in of either RX-277 or NS-3 neutron shielding and a 3.0-in thick steel plate.

5.1.2 Shielding Design Criteria

The overall design criteria for the FuelSolutions™ Storage System are the regulatory dose limits and the requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

The staff evaluated the FuelSolutions™ Storage System shielding design features and design criteria and found them to be acceptable. The SAR analyses provide reasonable assurance that the shielding design features and design criteria can meet the regulatory requirements in

10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Cask surface dose rate limits are specified in TS 5.3.5.

An evaluation of the overall radiation protection design features and design criteria of the FuelSolutions™ Storage System is given in Section 10 of the SER.

5.2 Radiation Source Definition

5.2.1 Cooling Tables

The radiation source specification is presented in Section 5.2 of the WSNF-200. Generic photon and neutron source terms were generated with the ORIGEN-2.1[3] computer code and four ORIGEN libraries[4] (PWR-US, PWR-UE, BWR-US and BWR-UE). The neutron and gamma source terms were re-binned into a 67 group structure, for use with the BUGLE-93[5] cross section set. The transfer functions used to rebin the neutrons and gammas were based on logarithmic interpolation and conservation of energy.

The applicant performed generic source term calculations for the Westinghouse (W) standard 17x17 PWR and General Electric (GE) -5 8X8 BWR fuel assemblies. The applicant has shown that they yield the largest source term per metric ton of initial heavy metal (MTIHM) for each reactor type. A generic source term, in units of gammas/sec-MTIHM, was determined for each burnup and enrichment combination (state point) in the fuel cooling tables. The source term for each state point on the cooling table can be determined by multiplying the maximum amount of uranium for a given fuel assembly by the generic source term for each energy group. The generic source term includes photons from activated assembly hardware located within the fuel region (i.e., grid spacers and in-core control components).

The applicant added the source term due to in-core hardware to the generic source term for both the PWR and BWR fuel assemblies. For PWR fuel assemblies, the applicant included the per-assembly source term from 50 grams of cobalt for the high cobalt cooling table and 11 grams for the low cobalt cooling table. The applicant included the source term from 2.9 grams of cobalt in the fuel region hardware for the Big Rock Point BWR cooling table. The TS Tables 2.1-5 through 2.1-8 of WSNF-201 and TS Table 2.1-3 of WSNF-203 require that the maximum amount of cobalt in the fuel region be less than the values listed above.

To determine the amount of Co-60 activation, the applicant performed a depletion calculation to determine the source term from one gram of cobalt. The applicant then multiplied the per curie source term by the maximum amount of cobalt in the fuel region allowed by the applicable cooling table and decayed the source to the appropriate decay time from the cooling tables. This source term was added to the gamma group with energies between 1.0 and 1.25 MeV.

5.2.2 Adjoint Source Term

The source term for the adjoint calculations is the flux-to-dose conversion factors used to determine the dose rates in the forward shielding calculations. The conversion factors were normalized using the sum of the flux-to-dose conversion factors.

5.2.3 Forward Bulk Shielding Analysis Source Term

The applicant performed the forward bulk shielding calculations for PWR fuel. The source term is determined from the generic PWR source term using the low-cobalt cooling table and multiplied by a fuel loading of 0.471 MTU. The applicant determined this to be the maximum amount of fuel in any PWR fuel assembly. The applicant stated that PWR fuel provides the maximum dose rates when the total dose rate is dominated by gammas.

The applicant chose two combinations of burnup, enrichment, and cool times to perform forward calculations. The applicant stated that these combinations on the cooling table yield bounding and typical values expected for the dose rates. The two state points on the PWR cooling table are 36,000 MWD/MTU, 4.5 weight percent enriched, with a 4-year cool time for the bounding analysis, and 36,000 MWD/MTU, 3.5 weight percent enriched, and 10-year cooled fuel for the typical analysis. The bounding analysis state point was chosen because the 4 year cool time bounds the minimum cool time (4.6 years) given in the cooling table. By using the response function approach and varying the cool time to produce either a maximum dose rate or decay heat, lower burnups produce higher dose rates. Fuel assemblies with higher burnups and longer cool times will have a lower average gamma energy than fuel assemblies with shorter cool times, since the fission products that produce the higher energy gammas have shorter cool times. Longer cooled fuel assemblies will produce a higher decay heat and lower dose rate than fuel assemblies with a shorter cool time. Therefore, 36,000 MWD/MTU fuel with a 4 year cool time, which is shorter than the minimum cool time in the cooling tables, will have higher average energy gammas than fuel with a higher burnup and a longer relative cool time.

The variation of the axial distribution of neutron and gamma sources was taken from Reference 6, and is shown in Table 5.2-12 of WSNF-200 for the forward shielding calculations. The neutron profile was used to account for the non-linear buildup of a neutron source term (primarily Cm-244) as a function of burnup. The photon source distributions within the plenum, top end fittings, and bottom end fittings were assumed to be uniform.

Variation of the maximum peaking factor as a function of the maximum burnup was developed from Reference 6 for PWRs. The variation of the maximum peaking factors with burnup is shown in Tables 5.2-7 and 5.2-8 of WSNF-201 for the W21 canister. Variation of the maximum peaking factor with burnup for the W74 canister was determined from actual utility operator data. The variation of the maximum peaking factors with burnup is shown in Tables 5.2-4 and 5.2-5 of WSNF-203 for the W74 canister.

To determine the amount of activated hardware in the non-fuel region, including in-core control components, the applicant performed a survey of the Office of Civilian Radioactive Waste Management Spent Fuel Assembly Database[7] for all fuel assemblies to be stored in the cask. The applicant assumed cobalt impurities in Inconel-718, Inconel-X750, Zircaloy, and stainless steel of 4700, 6500, 10, and 800 ppm, respectively. Measured cobalt impurities in Inconel grid spacers from a W14x14 assembly range from 890 to 1490 ppm[8]. Another set of measurements resulted in a range of cobalt impurities from 186 to 3600 ppm[9]. The value of 4700 ppm used to estimate the Co-60 source term for the grid spacers bounds these measured values.

The applicant determined that the Babcock & Wilcox (B&W) 15x15 Mark B fuel assembly with a B&W 15x15 thimble plug assembly (TPA) has the maximum cobalt activation level for the top nozzle and the bottom nozzle regions. The applicant determined that the B&W 17x17 Mark C fuel assembly produced the maximum cobalt quantities for the gas plenum region. Although the B&W fuel assembly produced the maximum quantity of cobalt activation in the top nozzle region, it was not the fuel assembly with the largest quantity of cobalt prior to irradiation. Three Combustion Engineering (CE) fuel assemblies have cobalt quantities twice that of the B&W fuel assembly, prior to irradiation, but since the top nozzle is farther from the core region in the CE than the B&W fuel assembly, the amount of cobalt activation is less for the CE than the B&W fuel assembly. The applicant conservatively used the cobalt quantity of the CE fuel assembly, prior to irradiation, and applied to it the activation level of the B&W fuel assembly. The applicant stated that this produces an activated cobalt quantity that is almost a factor of two higher than the actual maximum possible. The applicant used the cobalt activation of the B&W 15x15 for the bottom nozzle and the B&W 17x17 Mark C fuel assembly for the gas plenum region in the storage cask.

The cobalt quantities for the transfer cask are all based upon the B&W 15x15 Mark B fuel assembly. The applicant stated that the B&W 17x17 Mark C was not used for the gas plenum region in the transfer cask, since there are only four fuel assemblies of this type in existence and this design is not being produced any longer.

To correct for spatial and spectral changes of the neutron flux outside the fuel zone during irradiation in the reactor core, the masses of the materials in the bottom end fitting, plenum, and top end fitting were multiplied by scaling factors of 0.2, 0.2, and 0.1, respectively. These are the factors recommended in Reference 8. These scaling factors produce calculated source terms which bound measured source terms. The neutron flux scaling factors from this reference are derived from measurements and are considered to provide bounding values, particularly in relationship to the values calculated in Reference 10.

The staff performed confirmatory analyses of the bounding photon and neutron source terms for the fuel region. The staff performed calculations using the SAS2H module within the SCALE4.4[11] System. The staff's bounding source term is consistent with the applicants. The staff also determined the maximum quantity of irradiated cobalt in the fuel assembly hardware. The applicant's source term due to irradiated hardware was conservatively larger than the staff's. The staff also found that the W74 cooling table provides lower gamma sources than the PWR fuel assemblies per metric ton of uranium. The staff has reasonable assurance that the applicant has adequately determined the bounding photon and neutron source terms used in the shielding analysis.

5.3 Shielding Model Specifications

The model specifications for shielding are presented in Section 5.3 of the WSNF-200. The models for normal and accident conditions consist of 2-D representations of the storage and transfer casks using the design drawings in Section 1.5 of the WSNF-200. The composition and densities of the materials used in the shielding analysis are presented in Tables 5.3-1 through 5.3-18 of WSNF-200. The applicant did not identify any materials which undergo changes in material density or composition from temperature variations.

5.3.1 Storage Cask

The applicant presents four models for the shielding evaluation for the storage cask. The four shielding calculations performed for the storage cask are (1) the adjoint analysis, (2) the forward bulk shielding analysis, (3) an evaluation of streaming from the inlet and outlet vents, and (4) the dose evaluation at the site boundary.

Adjoint Model

The adjoint model is a two-dimensional model of the storage cask and basket. Each fuel assembly and fuel tube is homogenized as either 21 or 64 discrete locations, for the W21 or the W74 canisters, respectively. The adjoint model conservatively neglects the basket spacers. The adjoint model is shown in Figure 5.3-1 of WSNF-200.

Forward Bulk Shielding Model

The radiation source is divided into four axial regions: bottom end fitting, fuel, gas plenum, and top end fitting. The relative positions of these source term regions are also depicted in the figures identified below. The entire fuel assembly region in the forward bulk shielding calculations is modeled as one homogeneous zone. The end fittings and plenum regions are modeled as homogeneous regions of stainless steel, Inconel, and Zircaloy.

The spacer plates in the homogenous fuel region were conservatively neglected. Outside the fuel region the applicant explicitly modeled the spacer plates. The storage cask forward bulk shielding model is depicted in Figures 5.3-2 and 5.3-4 of WSNF-200.

Inlet/Outlet Vent Model

The storage cask shielding models included streaming paths for the inlet and outlet vents. The cask design eliminates other potential streaming paths. The applicant performed an evaluation of the potential for streaming through the inlet and outlet vents using MCNP[12]. For the vent forward shielding calculation, the ducts and the materials surrounding them were explicitly modeled. The canister and internals were neglected. The source for this model is the flux at the opening of the inlet and outlet vents from the forward bulk shielding calculations.

Storage Cask Array Model

The applicant's evaluation for the dose at the site boundary considered an 8x8 array of casks. The applicant modeled a three-dimensional representation of the outer 2 inches of each concrete cask and applied the source to the side and top surfaces of the cask and on the surface of the inlet and outlet vents. The sources for the side, top and vent surfaces are surface flux data taken from both the forward bulk shielding evaluation and the inlet/outlet vent shielding evaluation.

5.3.2 Transfer Cask

The model for the transfer cask is a two-dimensional representation of the transfer cask with various shields in place. This model is used to estimate the occupational exposure around the transfer cask during canister closing and reopening. The canister model is similar to the one used for the forward bulk shielding calculations. The applicant explicitly modeled the transfer cask and its lids. The bounding accident condition for the transfer cask shielding assumes complete loss of the radial neutron shield. The transfer cask shielding model is depicted in Figure 5.3-3 of WSNF-200.

The model dimensions and material specifications are consistent with the drawings in Section 1 of WSNF-200 and provide the basis for reasonable assurance that the FuelSolutions™ Storage System was adequately modeled in the shielding analysis. The staff evaluated the shielding models and found them to be acceptable.

5.4 Shielding Analyses

The shielding analyses are presented in Section 5.4 of WSNF-200. The applicant used DORT[13] and MCNP with the BUGLE-93 cross-section library to determine the dose rates from the storage and transfer casks. The BUGLE-93 cross-section library has 47 neutron groups and 20 photon groups. The applicant uses the ANSI/ANS Standard 6.1.1-1977 flux-to-dose conversion factors to calculate dose rates in the shielding analysis.

5.4.1 Storage Cask

The applicant performed four sets of shielding analyses for the storage cask; adjoint calculations, forward bulk shielding calculations, forward shielding calculations at inlet and outlet vents, and off-site dose rate calculations. The adjoint calculations were used to construct the cooling tables. The forward shielding calculations (both bulk shielding and at the inlet and outlet vents) were used to show that the dose rate from a limiting state point on the cooling table with a cool time less than the minimum will remain within the TS limits and that an array of casks can meet the off-site dose limits. The MCNP code was used to calculate direct and skyshine dose rates at large distances from the array of casks.

Adjoint Calculations

The applicant performed adjoint calculations using the DORT computer code. The adjoint calculations were used to determine the neutron and gamma importance functions (units of mrem/hr/particle/sec-cm). Multiplying the importance functions by a neutron and gamma source term per unit length yields dose rates on the surface of the cask. Using the importance functions, the applicant determined the minimum cooling time required to meet both the decay heat limit and the TS 5.3.5 maximum dose rate limit of 50 mrem/hr on the side of the concrete cask.

Forward Bulk Shielding Calculations

The applicant performed two sets of forward shielding calculations, bulk shielding calculations and streaming calculations at inlet and outlet vents. The bulk shielding calculations determine the dose rates on the side and top of the cask, except for the vents which are evaluated as described below. The bulk shielding calculations are also performed with the DORT computer code. WSNF-200 presents calculations for normal condition dose rates for both bounding and typical source terms. Calculated dose rates are summarized in SAR Table 5.1-1.

Figures 5.4-1 through 5.4-3 of WSNF-200 show the dose rates over the surface of the storage cask. The maximum dose rates in the fuel region on the side and top, are 32.6 mrem/hr and 55 mrem/hr. The average dose rate over the top of the cask is 18.9 mrem/hr.

The applicant determined accident condition dose rates around the storage cask in Section 11. The maximum dose rate around the storage cask under accident conditions is 125 mrem/hr.

Inlet/Outlet Vent Calculations

The applicant performed an MCNP calculation to determine the dose rates at the inlet and outlet vents. The applicant calculated the dose at the inlet and outlet vents to be 14 mrem/hr and 510 mrem/hr, respectively. The applicant stated that most of the dose through the outlet vents is due to the top nozzle source term, which the applicant stated is conservatively overestimated by approximately a factor of two.

Confirmatory Calculations

The staff performed evaluations to check other state points on both PWR cooling tables. The staff used the applicant's importance functions to perform a comparison of the source terms from different state points on the cooling tables. The staff agreed with the applicants conclusion that locations limited by decay heat had lower dose rates than locations limited by dose rate.

Confirmatory shielding calculations for the FuelSolutions™ Storage System were made with the SAS4 module in the SCALE 4.4 system. The staff homogenized each fuel assembly, but explicitly modeled the spacer plates, both inside and outside the fuel region. A comparison between the applicant's results and the staff's confirmatory calculations showed a variation in the results which is expected when two different codes are used for shielding calculations. The surface dose rate calculated by the staff for the bounding source term is 29.4 mrem/hr on the side of the cask. The staff's dose rate for the bounding source term is less than the applicants dose rate of 32.7 mrem/hr.

The applicant's dose rates 1 meter from the cask are in good agreement with the confirmatory calculations. The staff's dose rate 1 meter from the surface of the storage cask is 15.7 mrem/hr. The staff's dose rate is lower than the applicant's value of 17.7 mrem/hr. A TS has been included which specifies a maximum average allowable dose rate on the surface of the cask, both at the cask longitudinal midplane and along the top surface of the cask.

5.4.2 Transfer Cask

The maximum dose rates for the transfer cask are shown in Table 5.1-2 of WSNF-200. These dose rates have been adjusted to account for transfer cask locations or conditions which yield higher dose rates than those calculated using the W21 canister bounding analyses. Locations where the neutron dose rates are limiting may yield higher total dose rates for the W74 canister.

The applicant calculated adjustment factors based on the W21 canister as a reference. If either the W74 canister or high burnup W21 fuel provides higher dose rates than the bounding W21 source term, then the dose rates calculated in the shielding calculations for the transfer cask were increased based upon the adjustment factors. Figures 5.4-4 through 5.4-6 of WSNF-200 show the dose rates over the surface of the transfer cask, for various cases described in Section 5.3.3. The maximum adjusted dose rates on side and top surface of the transfer cask are 219 mrem/hr and 101 mrem/hr with all the lids in place and the transfer cask cavity and annulus full of water. The transfer cask dose rates vary depending on the amount and location of water in the cask and which lids (top or bottom) are in place. Maximum dose rates on the radial surface and at 1 meter with all of the water drained from the transfer cask except for the neutron shield, are 826 mrem/hr and 342 mrem/hr respectively.

Table 5.1-2 of WSNF-200 also contains results of calculations for accident condition dose rates of the bounding fuel on the transfer cask side, top and bottom surfaces and 1 meter from the side, top and bottom surfaces. Maximum dose rates on the surface and at 1 meter from the transfer cask occur for the side of the cask and are 6,776 mrem/hr and 2,255 mrem/hr, respectively.

The staff performed confirmatory calculations for the FuelSolutions™ transfer cask using the SAS4 module in the SCALE 4.4 system. The staff's model of the canister is the same one used in the storage cask calculations. The staff performed two sets of analyses to determine the radial dose rates at the fuel midplane. The first being with all the lids in place and the transfer cask cavity, annulus, and neutron shield full of water. The staff used the applicant's adjustment factors to determine the maximum dose rates. The staff's adjusted surface dose rate using the staff's bounding source term is 237 mrem/hr on the surface and 105 mrem/hr at 1 meter from the surface.

The staff also performed calculations with the cask cavity and annulus dry, and the neutron shield filled with water. The staff's adjusted dose rate using the staff's bounding source term is 829 mrem/hr on the surface and 331 mrem/hr 1 meter from the surface.

The staff's total dose rates are approximately 7 to 9 % higher than the applicants for the first set of calculations and either within 1% or lower than the applicants for the second set of calculations. The breakdown of the staff's calculations on the surface show that the adjusted gamma and neutron dose rates are 224 mrem/hr and 13.5 mrem/hr, respectively. The gamma dose rate which is the major contributor to the total dose is approximately 5% higher than the applicants (213 mrem/hr). Variations in the results, such as these, are expected when two different codes are used for shielding calculations. Overall, the differences between the applicant's and confirmatory results fell within acceptable bounds.

5.4.3 Occupational Exposures

Design-basis fuel at 36,000 MWD/MTU burnup and 4-year cooling time with design-basis BPRAs, was used to estimate occupational exposures during cask operations. Section 10 of WSNF-200 presents estimated occupational exposures using the calculated dose rates for the locations shown in Figures 5.1-1 and 5.1-2.

5.4.4 Off-Site Dose Calculations

Direct-path off-site dose rates are presented in Section 10 of WSNF-200 for a single cask and an array of casks. Direct-path dose rates for off-site locations are for a bounding fuel loading, level topography, and a 100% occupation time. The applicant determined the off-site dose from an 8x8 array of casks. The surface fluxes from both the forward bulk shielding model and the outlet vent models are used as the source term for the off-site dose calculations. Table 10.4-2 of WSNF-200 shows the dose for one year exposure (8760 hours) at specified distances from the array of storage casks. Section 10 of the SER evaluates the overall off-site dose rates from the FuelSolutions™ Storage System. The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved.

The general licensee using the FuelSolutions™ Storage System must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate operational compliance with 10 CFR 72.104(a). The actual doses to individuals beyond the controlled area boundary depend on site-specific conditions such as cask-array configuration, topography, demographics, and use of engineered shielding features (e.g., berm). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities in the region such as reactor operations. Consequently, final determination of compliance with 72.104(a) is the responsibility of each general licensee.

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

5.5 Evaluation Findings

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

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

F5.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The FuelSolutions™ Storage System cask shielding features are designed to assist in meeting these requirements.

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

5.6 References

1. U.S. Code of Federal Regulations, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,” Title 10, Part 72.

2. U.S. Code of Federal Regulations, “Standards for Protection Against Radiation,” Title 10, Part 20.

3. ORIGEN2.1 Isotope Generation and Depletion Code Matrix Exponential Method. RSICC Code Package CCC-371, Radiation Shielding Information Code Center, Oak Ridge, Tennessee.

4. Ludwig, S. B., Renier, J. P.. Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2.1 Computer Code, ORNL/TM-11018, Oak Ridge, Tennessee, December 1989.

5. BUGLE-93: Coupled 47 Neutron, 20 Gamma Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications, RSICC Data Library DLC-175, Radiation and Shielding Information Code Center, Oak Ridge, Tennessee.

6. DOE/RW-0495 Depletion and packaging Modeling Assumptions for Actinide-Only Burnup Credit, Office of Civilian Radioactive Waste, U.S. Department of Energy, May 1997.

7. DOE/RW-0184-R3, Characteristics of Potential Repository Wastes, Oak Ridge National Laboratory, Oak Ridge, Tennessee, December 1987.

8. Luksic, A.T., “Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal,” PNL-6906 Vol. 1, Pacific Northwest Laboratory, Richland, Washington, 1989.

9. Luksic, A.T., et al., “The Role of Trace Impurities in Classification of In-Core Reactor Components,” EPRI TR-102800, Battelle Pacific Northwest Laboratories, Richland, Washington, 1993.

10. Croff, A.G., et al., “Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code,” ORNL/TM-6051, Oak Ridge , Tennessee, 1978.

11. Oak Ridge National Laboratory, “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations,” NUREG/CR-0200, Vol. 1-3, Revision 5, 1997.

12. MCNP4B: Monte Carlo N-Particle Transport Code System, RSICC Code Package CCC-660, Radiation Shielding Information Code Center, Oak Ridge, Tennessee.

13. TORT-DORT-PC: Two- and Three-Dimensional Discreet Ordinates Transport, Version 2.7.3, RSICC Code Package CCC-543, Radiation Shielding Information Code Center, Oak Ridge, Tennessee.

6.0 CRITICALITY EVALUATION

The staff reviewed the FuelSolutions™ Storage System criticality analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that storage of spent fuel in the FuelSolutions™ Storage System meets the following regulatory requirements: 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c), and 72.236(g).[1] Revision 4 of the SAR was also reviewed to determine whether the FuelSolutions™ storage cask fulfills the following acceptance criteria listed in Section 6 of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems:[2]

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

b. 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.

c. 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.

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

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

6.1 Criticality Design Criteria and Features

The design criterion for criticality safety is that the effective neutron multiplication factor, k_{eff}, including statistical biases and uncertainties, shall not exceed 0.95 for all postulated arrangements of fuel within the cask under normal, off-normal, and accident conditions.

The FuelSolutions™ Storage System design features relied upon to prevent criticality are the basket geometry and fixed neutron poisons in the basket. For the W21 canister basket, TS 4.1.3 in the WSNF-201 requires a minimum basket cell opening of 8.90 inches square and a minimum ^{10}B areal density of 0.02 gm/cm^2 in the basket poison material. The applicant took credit for 75% of the minimum specified ^{10}B areal density. For the W74 canister basket, TS 4.1.3 in the WSNF-203 requires a minimum basket cell opening of 6.85 inches square and a minimum boron content of 1.0 wt% natural boron in the basket poison material. The applicant took credit for 75% of the minimum specified natural boron content. The fabrication requirements and acceptance criteria for the fixed neutron poison, which justify the use of 75% credit of each basket's fixed neutron poison, are outlined in each canister SAR Section 9.1.4.

The staff reviewed the design criteria and features discussed in Sections 1.2, 2.1.2, and 6 of Revision 4 of each canister SAR and verified that the design features important to criticality safety are clearly identified and adequately described. The staff verified that the SAR contains engineering drawings, figures, and tables that are sufficiently detailed to support an in-depth staff evaluation.

The staff also verified that the design basis off-normal and accident events would not have an adverse effect on the design features important to criticality safety. Therefore, based on the information provided in the SAR, the staff concludes that the FuelSolutions™ Storage System design meets the double contingency requirements of 10 CFR 72.124(a)

6.2 Fuel Specification

The FuelSolutions™ Storage System is designed to store a maximum of 21 intact PWR spent fuel assemblies in the W21 canister, or 64 intact Big Rock Point BWR spent fuel assemblies in the W74 canister.

The fuel assemblies that are approved for storage in the W21 canister are described in Sections 2.2 and 6.2 of the WSNF-201 and the fuel characteristic limits are given in TS 2.1. The W21 canister can be configured to accommodate 21 fuel assemblies (Loading Specification W21-1) with the maximum uranium masses and initial enrichments listed in Table 2.1-3 of TS 2.1, or 20 fuel assemblies (Loading Specification W21-2) with the maximum uranium masses and initial enrichments listed in Table 2.1-4 of TS 2.1. The maximum initial enrichment is five weight percent for either the W21-1 or W21-2 loading specification. The W21 canister may also include radial and axial assembly spacers to accommodate fuel assemblies with smaller fuel assembly widths and lengths. The criticality analysis conservatively neglects these spacers and assumes all assemblies are at the maximum active fuel length and that they are free to relocate within the guide tube toward the center of the canister. The fuel assemblies may be stored with or without control components. Control components are not considered in the criticality analysis, due to the fact that they would displace water in PWR assemblies which are already undermoderated, resulting in lower system k_{eff}.

The fuel assemblies approved for storage in the W74 canister are described in Sections 2.2 and 6.2 of the WSNF-203 and the fuel characteristic limits are given in TS 2.1. These assemblies are GE 9x9 or Siemen's 9x9 or 11x11 Big Rock Point BWR assemblies. The maximum uranium mass, listed in Table 2.1-2 of TS 2.1, conservatively bounds the allowed fuel assemblies. The maximum initial enrichment is 4.10 wt% ^{235}U and the maximum assembly average burnup is 40,000 MWD/MTU. Big Rock Point BWR spent fuel assemblies are stored without flow channels.

Specifications on the fuel condition are also included in Section 6.2 of the SAR and TS 2.1 for each canister. Fuel with structural defects greater than pinhole leaks and hairline cracks may not be loaded into the FuelSolutions™ Storage System with either canister. Fuel assemblies with missing pins are not allowed unless the missing pin is replaced by a dummy pin that displaces an equivalent volume.

The staff reviewed the fuel specifications considered in the criticality analysis and verified that they bound the specifications given in Sections 1 and 2 of the SAR and TS for each canister.

The staff verified that all fuel assembly parameters important to criticality safety have been included in the TS for each canister.

6.3 Model Specification

6.3.1 Configuration

For all calculations, the criticality analysis for the FuelSolutions™ Storage System considers infinite arrays of W21 and W74 canisters inside of a bounding, representative overpack that consists of an inner steel shell, depleted uranium shield, outer steel shell, and a steel jacket which contains water for neutron shielding. The actual storage overpack consists of a steel liner surrounded by a thick steel reinforced concrete shield, and the actual transfer cask consists of an inner steel shell, lead shield, outer steel shell, and a steel jacket containing water for neutron shielding. As shown in Section 6.4 of the SAR, the representative overpack configuration is more reactive than the storage and transfer configurations, due to the greater neutron reflection provided by the depleted uranium in the representative overpack walls.

Both normal and accident conditions are considered in the criticality analysis. Normal conditions include: complete flooding of the canister, including fuel-clad gaps, with an unborated water density that produces optimum moderation; worst case asymmetric assembly placement within the guide tubes; and worst case material and fabrication tolerances. Case studies are presented in Section 6.4 of each canister SAR to determine the worst case conditions. The case studies for optimum moderation showed that a water density of 1.0 g/cm^3 is the most reactive for both the W21 and W74 canisters. The worst case material and fabrication tolerances, and asymmetric assembly placement patterns are presented in Section 6.3 of each canister SAR.

The accident conditions include all of the normal conditions, plus the conditions expected to result from the hypothetical accident conditions defined for transportation in 10 CFR 71.73.[3] Although the hypothetical accident conditions from 10 CFR 71.73 are for transportation packages, the results of these tests bound the results from the off-normal and accident conditions structural and thermal analyses in Sections 3, 4, and 11 of the storage system and canister SARs. The results of the accident conditions include a 0.08 inch permanent deformation throughout the entire length of the guide tubes, detachment of the guide tubes from the spacer plates, and the loss of the outer neutron shield from the representative transportation package.

6.3.1.1 FuelSolutions™ Storage System with W21 Canister

The W21 canister consists of an array of guide tubes, Boral neutron absorber plates, support rods, and spacer plates arranged to provide structural integrity and to prevent criticality under normal, off-normal, and accident conditions. There are two canister basket and shell assembly types, designated W21M and W21T, which differ with respect to the materials used for the support rods, vent and drain port covers, outer closure plates, inner closure plates, cylindrical shell, and spacer plates. There are also six basket configurations, to accommodate the dimensions of the range of fuel assemblies to be stored, which differ with respect to overall length, spacer plate separation, guide tube and neutron absorber plate length, and support rod sleeve length. The applicant performed case studies, discussed in Section 6.4 of the SAR, to determine which canister basket and shell assembly configuration is the most reactive. The

W21M canister with the long canister length, with shorter internal cavity and steel end plug shielding material, was determined to be the most reactive configuration.

The fuel assemblies modeled in the W21 canister are the representative bounding fuel assemblies given in Table 6.1-1 of the SAR. All assemblies are modeled as UO_2 rods with a 96.5% theoretical density and with no pellet dishing. No credit is taken for burnup or for any isotope other than ^{235}U or ^{238}U in the fuel. The maximum initial enrichments given in Table 6.1-1 of the SAR for each loading specification (W21-1 and W21-2) are assumed to be uniform over the entire assembly. No spacer grids, spacer sleeves, top and bottom end fittings, or any other assembly hardware, are modeled. The Boral neutron absorber plates are modeled with a ^{10}B areal density at 75% of the specified 0.02 g/cm^2.

The W21 canister, and representative overpack, are modeled axially from the top of the canister bottom inner closure plate to the bottom of the top shield plug. The start of the active fuel stack is assumed to be 1.97-in above the top of the bottom closure plate for all assemblies modeled, which is representative of the shortest bottom end fitting of any allowed fuel assembly. The fuel is modeled to a height of 151.97-in in all cases, resulting in an active fuel length of 150-in, representative of the longest fuel assembly allowed in the W21 canister. Under normal conditions the guide tubes begin at a height of 1.0-in from the canister bottom and the Boral plates begin at a height of 2.5-in from the canister bottom.

Under off-normal and accident conditions, the guide tube assemblies separate from the spacer plates and are modeled resting against the bottom of the upper shield plug. In addition, it is assumed that the tabs at the top of the guide tubes collapse and allow a further 0.75-in upward displacement. This results in the position of the bottom of the Boral plates being increased to 5.25-in above the bottom of the canister, allowing for a greater length of unpoisoned fuel.

6.3.1.2 FuelSolutions™ Storage System with W74 Canister

The W74 canister consists of an array of guide tubes, borated stainless steel neutron absorber plates, support tubes, and spacer plates arranged to provide structural integrity and to prevent criticality under normal, off-normal, and accident conditions. There are two canister basket and shell assembly types, designated W74M and W74T, which differ with respect to the materials used for the alignment bars, vent and drain port covers, outer closure plates, inner closure plates, canister shells, and engagement spacer plates. The applicant performed case studies, discussed in Section 6.4 of the SAR, which determined that the W74T basket and shell assembly type is the most reactive.

The fuel assemblies modeled in the W74 canister are the representative bounding fuel assemblies given in Table 6.1-1 of the SAR. All assemblies are modeled as UO_2 rods with a 96.5% theoretical density and with no pellet dishing. No credit is taken for burnup or for any isotope other than ^{235}U or ^{238}U in the fuel. The maximum initial enrichment of 4.10 wt% is assumed to be uniform over the entire assembly. No spacer grids, spacer sleeves, top and bottom tie plates, or any other assembly hardware, are modeled. The borated stainless steel neutron absorber plates are modeled with a nominal natural boron loading at 75% of the specified 1.0 wt%. The manufacturer's minimum specified boron content, verified at manufacturing, is 1.25 wt% natural boron.

The W74 canister, and representative overpack, are modeled axially from the middle of the canister bottom end shield plug to the bottom of the top shield plug. The start of the active fuel stack in the lower basket is assumed to be 1.895-in above the top of the bottom closure plate for all assemblies modeled, which represents a minimum 1.25-in high bottom tie plate plus a 0.645-in high bottom end plug. The fuel is modeled to a height of 71.895-in for all assemblies, resulting in an active fuel length of 70-in, representative of the longest fuel assembly allowed in the W74 canister. Under normal conditions the borated stainless steel neutron absorber plates are modeled beginning at an elevation of 1.5-in above the bottom closure plate, whereas the actual elevation will be 0.375-in. Similarly, the bottom of the lower basket guide tubes is modeled at an elevation of 1.5-in from the bottom closure plate, whereas the guide tube faces with borated stainless steel plates on them will begin at the surface of the bottom closure plate. The relative placement of fuel, guide tubes, and borated stainless steel in the upper basket is the same, with the reference point being the top of engagement spacer plate.

Under off-normal and accident conditions, the guide tube assemblies separate from the spacer plates and are modeled resting against the bottom of the engagement spacer plate in the lower basket, and the bottom of the upper shield plug in the upper basket. In addition, it is assumed that the tabs at the top of the guide tubes collapse and allow a further 0.5-in upward displacement. This results in the position of the bottom of the borated stainless steel plates being increased to 2.5-in above the bottom of the canister in the lower basket and 3.525-in above the engagement spacer plate in the upper basket, allowing for a greater length of unpoisoned fuel in both basket sections.

6.3.1.3 Staff Review of Models

The staff reviewed the applicant's models and agrees that they are consistent with the description of the cask system and contents given in Sections 1 and 2 of each canister SAR, including engineering drawings. The staff also reviewed the applicant's methods, calculations, and results for determining the worst-case materials and fabrication tolerances. Based on the information presented in the SAR, the staff agrees that the most reactive combination of cask parameters and dimensional tolerances was incorporated into the calculation models.

The staff performed confirmatory analyses using the information provided in the SAR and TS. Specifically, the staff used Drawing No. W21-120, Revision 2, and W74-120, Revision 1. The staff's fuel assembly models were based on the fuel assembly parameters given in Section 6 of the SAR and TS 2.1. The uranium masses and enrichments are the values used in the TS. The staff's results were consistent with those of the applicant.

6.3.2 Material Properties

The compositions and densities for the materials used in the computer models are provided in Section 6 of each canister SAR. The minimum required areal density of the ^{10}B in the Boral neutron absorber plates for the W21 canister is 0.02 gm/cm^2. The minimum required natural boron content in the borated stainless steel neutron absorber plates for the W74 canister is 1.0 wt%. The calculations for each canister modeled 75% of the minimum required ^{10}B areal density or natural boron content. Each canister SAR Section 9.1.4 discusses the acceptance tests for the fabrication of neutron absorber plate materials.

The continued efficacy of the neutron absorber plates over a 20-year storage period is assured by the design of the FuelSolutions™ Storage System. The Boral neutron absorber plates for the W21 canister consist of a core of boron carbide (B_4C)-aluminum mixture surrounded by 1100F aluminum alloy cladding. These plates are attached to the guide tubes by a welded Type 316 stainless steel wrapper, and are therefore encased in and completely supported by the stainless steel guide tube assembly. A structural analysis in Section 3 of the W21 WSNF-201 demonstrates that the neutron absorber plates remain in the guide tube assembly under the bounding accident conditions. The borated stainless steel neutron absorber plates in the W74 canister consist of natural boron alloyed with AISI type 304 stainless steel. These neutron absorber plates are attached to the guide tubes by seven 20-gage stainless steel neutron absorber sheet retainers, which are welded to the guide tube through small holes in the absorber plates. A structural analysis is provided in Section 3 of the W74 WSNF-203 which demonstrates that the neutron absorber plates will remain in place under the bounding accident conditions. Also, the plates meet thermal requirements and can be expected to have no significant erosion or corrosion under ISFSI service. The neutron flux in either canister over the storage period is also very low such that boron depletion during 20 years of ISFSI service is negligible. Thus, the staff agrees with the SAR conclusion that the neutron poison will remain effective for the 20-year storage period.

The compositions and densities for the materials in the computer models were reviewed by the staff and determined to be acceptable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

6.4 Criticality Analysis

6.4.1 Computer Programs

The applicant used the MCNP 4a code package for all criticality calculations. MCNP is a general purpose Monte Carlo code that can be used for neutron, photon, electron or coupled neutron/photon/electron transport, and has the capability to calculate k_{eff} for critical systems. MCNP uses a continuous energy cross-section library developed from the Evaluated Nuclear Data File system (ENDF/B-V).

The NRC staff performed confirmatory calculations using KENO V.a in the SCALE 4.4 code system[4] with the 44 group cross section library. The KENO V.a code is a standard in the industry for performing criticality analyses.

The staff agrees that the codes and cross-section sets used are appropriate for this particular application and fuel storage system.

6.4.2 Multiplication Factor

The applicant calculated a maximum k_{eff} for each fuel assembly class to be stored in the W21 and W74 canisters. In Section 6.5 of each canister SAR, the applicant calculated the limiting upper subcritical limit (USL) for each assembly class, using USL Method 1 from NUREG/CR-6361.[5]

Results of the applicant's criticality analyses show that the k_{eff} for the FuelSolutions™ Storage System will remain below 0.95 for all fuel loadings. The results of the applicant's MCNP 4a criticality calculations for the bounding assemblies are given in Table 6.4-6 of the W21 WSNF-201 and Table 6.4-8 of the W74 WSNF-203. The maximum k_{eff} calculated for each canister are summarized in the table below and compared to the limiting USL for that assembly class.

Canister	Most Reactive Fuel Assembly Class/Type	Maximum k_{eff} + 2σ	Limiting USL
W21	W17x17A	0.94211	0.94224
W74	Siemen's 11x11, 121 Fuel Rods	0.94007	0.94286

The staff reviewed the applicant's calculated k_{eff} values and agrees that they have been appropriately adjusted to include all biases and uncertainties at a 95% confidence level or better.

The staff performed independent criticality calculations for the most reactive fuel assembly class for the W21 and W74 canisters, using KENO V.a in the SCALE 4.4 code system with the 44-group cross-section library. The staff's models were similar to the applicant's, and consisted of explicitly modeled canisters inside of an infinite array of representative transportation packages. The results of the staff's confirmatory calculations were in close agreement with the applicant's results for both canisters.

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

6.4.3 Benchmark Comparisons

The applicant performed benchmark calculations on selected critical experiments, chosen, as much as possible, to bound the range of variables in the design. A set of 49 critical experiments, described in detail in NUREG/CR-6361, were analyzed using the MCNP 4a code system to demonstrate its applicability to the criticality analysis. Results from these analyses were used to establish a set of USL equations using USL Method 1, Confidence Band with Administrative Margin, described in Section 4 of NUREG/CR-6361. The USL incorporates the biases and uncertainties of the model and computer code into a value that has a 95% confidence level such that any k_{eff} less than the USL is less than 0.95, which is the design criterion. Four USL equations were established for each canister analysis based on the following critical experiment system parameters: enrichment, water-to-fuel ratio, hydrogen-to-^{235}U ratio, and pin pitch. The appropriate values for each assembly class were entered into the USL equations, resulting in a set of four USLs. The most limiting of the four USLs for each assembly class, listed in Table 6.4-6 of the W21 WSNF-201 and Table 6.4-8 of the W74 canister SAR, were used in the criticality analysis.

The applicant stated that the benchmark calculations were performed with the same computer codes and cross-section data and on the same computer hardware used in the criticality calculations.

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

6.5 Supplemental Information

All supportive information has been provided in Revision 4 of the W21 and W74 canister SARs, primarily in Sections 1, 2, and 6.

6.6 Evaluation Findings

Based on the staff's review of Revision 4 of the FuelSolutions™ Storage System SAR and the staff's own confirmatory analyses, the staff concludes that the FuelSolutions™ Storage System meets the acceptance criteria specified in NUREG-1536. In addition, the staff finds the following:

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

F6.2 The FuelSolutions™ Storage System and its spent fuel transfer systems are designed to be subcritical under all credible conditions.

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

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

F6.5 The staff concludes that the criticality design features for the FuelSolutions™ Storage System are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the FuelSolutions™ Storage System will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

6.7 References

1. U.S. Code of Federal Regulations, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,” Title 10, Part 72, January 1, 1999.

2. U.S. Nuclear Regulatory Commission, “Standard Review Plan for Dry Cask Storage Systems,” NUREG-1536, January 1997.

3. U.S. Code of Federal Regulations, “Packaging and Transportation of Radioactive Material,” Title 10, Part 71, January 1, 1999.

4. U.S. Nuclear Regulatory Commission, “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” NUREG/CR-0200, Vol. 1-5, Rev. 5, March, 1997.

5. U.S. Nuclear Regulatory Commission, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,” NUREG/CR-6361, March, 1997.

7.0 CONFINEMENT EVALUATION

The review of the confinement features and capabilities ensures that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures.

7.1 Confinement Design Characteristics

The applicant has clearly identified the confinement boundary. The confinement boundary includes the canister cylindrical shell, the bottom end closure plate, the top end inner and outer closure plates, and the vent, drain, instrument, and leak test ports with their associated covers. They are designed, fabricated, and tested in accordance with the applicable requirements of ASME Section III, Subsection NB, using ASME Section II, Part D, austenitic stainless steel, as discussed in Section 2.1.2 and 2.5.1 of the SAR. The canister is sealed using redundant closure welds, one at the outer top closure plate and the second at the inner top closure plate. The canister is designed, fabricated, and tested in accordance with the applicable requirements of the ASME Code, Section III, Subsection NB and Code Case N-595 as discussed in Section 2.1.2 of the SAR. The FuelSolutions™ canister cylinder shell seam welds are full penetration groove welds designed and RT inspected per Subsection NM. The shell assembly includes redundant confinement welds at the top end at the weld joints between the cylindrical shell, and the top inner and top outer closure plates. All canister top closure welds are examined with liquid penetrant; the inner closure weld is inspected at the root and final weld passes, and the outer closure weld is inspected at the root, intermediate, and final weld passes. The full penetration weld between the bottom closure plate and the canister shell is examined by both liquid dye penetrant and radiographic examination. The welds forming the confinement boundary are described in detail in the SAR. The redundant closures of the canister satisfy the requirements of 10 CFR 72.236(e) for redundant sealing of confinement systems.

The applicant provided procedures for drying and evacuating the cask interior during loading operations. The procedures for the vacuum drying process are established to ensure that the peak cladding temperatures do not exceed 400 ^{o}C. That temperature limit is established to ensure compliance with the maximum allowable temperature limits (based on a 1% strain limit) for normal storage conditions. The staff reviewed these procedures and finds that this design, fabricated in accordance with the SAR, will maintain the confinement boundary and fuel conditions within the analyzed constraints. Maintaining the stable pressure of 3 torr or less for 30 minutes assures that an acceptably low quantity of water remains in the canister.

For a welded canister, no leakage is expected. However, in the unlikely event that a leak does develop, the applicant tests the performance of the canister to ensure that the leak rate will not exceed a hypothetical equivalent leak rate of 8.52×10^{-6} ref·cm^3/s. The applicant used a design-basis leak rate of 8.52×10^{-6} ref•cm^3/s, as defined in ANSI N14.5-1997 Standard. This Standard is derived under the following postulated conditions for air: P_u = 1.0 atm upstream pressure, P_d = 0.01 atm downstream pressure, and 298.15 ^{o}K upstream temperature. Based on this hypothesized leak rate, the applicant calculated the offsite consequences. In addition, the applicant confirmed that the amount of helium lost from the canister over the approved period due to the hypothetical accident condition leakage rate is limited to less than 1% of the gas volume in the canister. This ensures that an adequate inventory of helium remains in the canister to maintain an inert atmosphere and to support the heat transfer over the lifetime of the cask.

Section 9 of the SAR addresses the inspection and acceptance tests performed prior to the use of each FuelSolutions™ canister with the associated FuelSolutions™ Storage System components. These inspections and tests provide added assurance that the canister is fabricated and operated in accordance with the requirements set forth in the SAR. The canister is classified as important to safety. The testing and inspection acceptance criteria applicable to each canister are listed in Table 9.1-1 in the SAR. These inspections and tests demonstrate that a canister has been fabricated and examined in accordance with the criteria contained in Section 2 of the SAR.

Following the pressurization with helium of the cavity formed by the bottom closure plate, the canister shell, and a temporary closure (as described in Section 9 of the SAR), the canister shell circumferential and longitudinal full-penetration butt welds and the bottom closure plate-to-shell welds are helium leak rate tested to verify that an equivalent maximum upstream air leak rate of 8.52×10^{-6} ref•cm^3/s (as defined in ANSI N14.5-1997) is not exceeded. This leak rate is based on an equivalent leak diameter of 6.598×10^{-4} cm, as identified by the applicant.

Following the installation of the vent and drain block assemblies, the installed vent/drain port quick connect fittings are soap bubble leak tested.

Prior to internal basket insertion, all canister shell radiographs are reviewed to assure that they meet the design and code of construction requirements. Radiography is required for the canister shell longitudinal and circumferential seam and shell-to-bottom closure plate welds only. The inner and outer top end closure plate welds do not require radiography since the canister shell top end uses redundant closure welds with liquid penetrant examination, and helium leak testing of the inner closure plate welds is performed in accordance with the technical specification requirements contained in Section 12.3 of the SAR.

7.2 Confinement Monitoring Capability

For cask systems using canisters with seal weld closures, continuous monitoring of the weld closures is not necessary because there is no known plausible, long-term degradation mechanism that would cause the seal welds to fail. Continuous monitoring of the cask, including periodic surveillance, inspection, and survey requirements, as well as existing licensee radiological and environmental monitoring programs, are such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions.

7.3 Nuclides with Potential for Release

The quantity of radioactive nuclides postulated to be released to the environment and the applicable bounding calculation method have been assessed as discussed in NUREG/CR-6487, “Containment Analysis of Type B Packages Used to Ship Various Contents” and ANSI N14.5-1997, “Leakage Tests on Packages for Shipment.”

The limiting source term for the W21 canister is shown to be that for a B&W 17x17 assembly. The applicant documented the limiting consequences calculated for fuel with 1.5 wt% and 5.0

wt% initial enrichment, irradiated to 60,000 MWD/MTU, and cooled three years. The limiting source term for the W74 is shown to be that for an 11x11 assembly; the applicant documented the limiting consequences for fuel with 1.5 wt% and 5 wt% initial enrichment, irradiated to 46,000 MWD/MTU and cooled three years. The applicant bounded the 40,000 MWD/MTU radionuclide activity (source terms) by assuming 46,000 MWD/MTU burnup. The applicant evaluated the source terms, based on a full loading of both assemblies described above, using the ORIGEN2 source term and depletion code. The staff performed confirmatory source term calculations using the SAS2H/ORIGEN-S sequence in the SCALE 4.4 code package with the 44-group cross-section library. The staff used assumptions similar to the applicant's, and obtained comparable results.

In accordance with the NRC staff guidance, ISG-5, the fractions of radioactive materials available for release from spent fuel are provided in Table 7-1 for PWR fuel and BWR fuel for normal and off-normal occurrences, and accident conditions. These values were used in the confinement analysis to demonstrate compliance with 10 CFR Part 72 requirements. These fractions account for radionuclides trapped in the fuel matrix and radionuclides that exist in a chemical or physical form that are not releasable to the environment under credible normal, off-normal, and accident conditions. In addition, the applicant justified the use of a 0.1 reduction factor of the mass fraction of fuel fines that can be released from the cask.

Table 7.1 Fractions Available for Release

Variable	PWR AND BWR FUEL	
	Normal and Off-normal Conditions	Hypothetical Accident Conditions
Fraction of gases released due to a cladding breach, f_G†	0.3	0.3
Fraction of volatiles released due to a cladding breach, f_V†	2×10^{-4}	2×10^{-4}
Mass fraction of fuel released as fines due to cladding breach, f_F	3×10^{-5}	3×10^{-5}
Fraction of crud that spalls off cladding, f_C	0.15#	1.0#

The source of radioactivity in crud is ^{60}Co on fuel rods. At the time of discharge from the reactor, the specific activity, S_c, is estimated to be 140 µCi/cm^2 for PWRs and 1254 µCi/cm^2 for BWRs. Total ^{60}Co activity is this estimate times the total surface area of all rods in the cask. Decay of ^{60}Co to determine activity at the minimum time before loading was used.

The staff has accepted the following rod breakage fractions for the confinement evaluations:

1% for normal conditions,
10% for off normal conditions, and
100% for design basis accident and extreme natural phenomena.

For the source term, the applicant used the activity from the Co^{60} in the crud, the activity from iodine, fission products that contribute more than 0.1% of design basis fuel activity, and actinide activity that contributes more than 0.01% of the design basis activity. The total activity of the design basis fuel was based on the cask design loading that yields the bounding radionuclide inventory (considering initial enrichment, burnup, and cool time).

The calculations for determining the atmospheric dispersion factors (χ/Q) for various downwind distances followed the guidance presented in Regulatory Guide 1.145. For normal and off-normal conditions, the atmospheric dispersion factors are based on neutral atmospheric conditions (Pasquill D) with an assumed wind speed of 5 m/s. The factors for the accident conditions are based on moderately stable atmospheric conditions (Pasquill F) for a wind speed of 1 m/s. The evaluations were performed for a leakage duration of one year for normal and off-normal conditions, and 30 days for accident events, consistent with the staff guidance outlined in ISG-5.

7.4 Confinement Analysis

Since the confinement boundary is welded and the temperature and pressure of the canister are within the design-basis limits, no discernable leakage is credible. However, to demonstrate that the canister meets the requirements of 10 CFR 72.104(a), The applicant performed detailed analyses using the assumptions listed in Table 7-2.

Table 7-2 Analytic Assumptions for Calculating Offsite Radiological Consequences

	Normal Operating Conditions	Off-Normal Operating Conditions	Accident Conditions
Postulated Leak Diameter (cm)	6.598×10^{-4}	6.598×10^{-4}	6.598×10^{-4}
% Failed Fuel	1%	10%	100%
Breathing Rate	3.30×10^{-4} m^3/sec	3.30×10^{-4} m^3/sec	3.30×10^{-4} m^3/sec
χ/Q (at 100 m)	1.244×10^{-3}	1.244×10^{-3}	8.65×10^{-3}
Wind Speed	5 m/sec	5 m/sec	1 m/sec
Dispersion Factor	D-Stability Diffusion	D-Stability Diffusion	F-Stability Diffusion

The staff's independent analyses confirmed the calculated results listed in the SAR. Good agreement was obtained between the staff's and The applicant's calculations and confirmed compliance with the requirements of 10 CFR 72.104(a). Tables 7-3 and 7-4 compare the NRC and The applicant results at 100 meters.

Table 7-3 NRC and BFS Results for the W21 Cask						
60,000 MWD/MTU Burnup	NRC Calculated Dose Rates (mrem) (W21 at 100m)			FuelSolutions™ Calculated Dose Rates (mrem) (W21 at 100m)		
	Normal	Off-Normal	Accident	Normal	Off-Normal	Accident
Whole Body	0.89	5.01	67.7	0.80	5.03	81.2
Thyroid	0.19	0.50	5.72	0.18	0.50	6.83
Lung	4.26	13.7	169.	4.30	22.42	353.
Bone	2.67	42.9	426.	2.41	43.13	751.
Skin	7.81×10^{-3}	3.60×10^{-2}	0.47	5.79×10^{-3}	3.44×10^{-2}	0.55

Table 7-4 NRC and BFS Results for the W74 Cask						
46,000 MWD/MTU Burnup	NRC Calculated Dose Rates (mrem) (W74 at 100m)			FuelSolutions™ Calculated Dose Rates (mrem) (W74 at 100m)		
	Normal	Off-Normal	Accident	Normal	Off-Normal	Accident
Whole Body	3.98	9.60	125	3.98	9.59	125.
Thyroid	1.08	2.13	25.3	1.08	2.13	25.3
Lung	22.9	51.1	644.	22.9	52.1	663.
Bone	2.25	27.2	463.	2.26	27.2	464.
Skin	2.91×10^{-2}	6.62×10^{-2}	0.843	2.94×10^{-2}	7.07×10^{-2}	0.921

Table 7.5 identifies the dose calculated by the applicant at 200 meters for the bounding FuelSolutions™ canister.

Table 7.5 Maximum Dose Calculated at 200 meters (mrem)			
	TEDE / Reg Limit	Thyroid / Reg Limit	Other Organ / Reg Limit
Normal Condition	1.11 / 25	0.30 / 75	6.40 / 25
Off-Normal Condition	2.68 / 25	0.60 / 75	14.6 / 25
Accident Condition	36.0 / 5000.	7.29 / 50,000.	216.0 / 50,000.

The staff finds the applicant's analyses acceptable and concludes that the requirements of 10 CFR 72.104(a) and 10 CFR 72.106(a) are met.

7.5 Evaluation Findings

F7.1 Section 2 of the SAR describes confinement SSCs important to safety in sufficient detail to permit evaluation of their effectiveness.

F7.2 The design of the FuelSolutions™ Storage System adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of the SER discusses the staff's relevant temperature considerations.

F7.3 The design of the FuelSolutions™ Storage System provides redundant sealing of the confinement system closure joints using dual welds on the canister lids (e.g., inner and outer closure plates).

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

F7.5 The quantity of radioactive nuclides postulated to be released to the environment has been assessed as discussed above. In Section 10 of the SER, the dose from these releases is added to the direct dose to show that the FuelSolutions™ Storage System satisfies the regulatory requirements of 10 CFR 72.104(a) {e.g., during normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area does not exceed 25 mrem to the thyroid and 25 mrem to any other critical organ} and 10 CFR 72.106(b) {e.g., any individual located on or beyond the nearest boundary of the controlled area will not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem. The lens dose equivalent will not exceed 15 rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem}.

F7.6 The cask confinement system has been evaluated by analysis. Based on successful completion of specified leakage tests and examination procedures, the staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F7.7 The staff concludes that the design of the confinement system of the FuelSolutions^{TM} Storage System is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the FuelSolutions^{TM} Storage System will allow safe storage of spent fuel. This finding considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

8.0 OPERATING PROCEDURES

The staff reviews the content of the operating procedures to ensure that the applicant's SAR presents acceptable operating sequences, guidance, and generic procedures for three key operations: canister loading and handling, cask storage operations, and canister unloading.

The information provided in Section 8 of WSNF-200, Revision 4, forms the basis of the staff conclusions in this SER Section.

8.1 Canister Loading and Handling

The FuelSolutions™ Storage System SAR presents generic canister loading procedures. Detailed canister loading procedures must be developed by each user. Based on the information in SAR Section 8, as discussed below, the staff concludes that the general canister loading procedures provide an adequate basis for the development of the more detailed site-specific operations and test procedures. In addition, the staff concludes that the FuelSolutions™ Storage system is compatible with wet or dry loading (dry loading specifically for the W74 canister). The staff also concludes that the canister loading procedures presented in the SAR are in the proper sequence and are of sufficient detail that cask users will be able to develop detailed site-specific procedures that adequately protect the workers, public, and the environment and will protect the fuel from significant damage or degradation.

8.1.1 Cask Preparation

The canister loading procedures presented in Section 8 of WSNF-200 include important inspections and preparation to prepare the cask for loading. Preparations include visual inspections for damage, trial fits of important components such as the closure plates and shield plugs, and the automatic welding/opening system (AW/OS) system. In addition, the canister is placed in the transfer cask and either placed in the spent fuel pool for wet loading or staged near the pool for dry loading.

8.1.2 Fuel Specifications

The loading procedures in the SAR state that fuel assemblies meeting the requirements in the TS, Section 12.3, of the appropriate canister SAR may be loaded into the canister. The site-specific procedures to be developed by each cask user are subject to evaluation at each site through the inspection process. The staff concludes that the procedures and TS requirements provide an acceptable means to ensure that fuel loaded in the FuelSolutions™ Storage System will meet the fuel-related assumptions (e.g., inventory, heat load, criticality-related parameters) made in the FuelSolutions™ Storage System SAR analyses.

8.1.3 ALARA

The staff concludes that the FuelSolutions™ Storage System generic cask loading procedures adequately incorporate general as low as reasonably achievable (ALARA) principles and practices. The procedures provide for a radiation survey to ensure the external gamma and neutron dose rates are below limits and for decontamination of the external surfaces of the cask until acceptable levels of contamination are obtained. These procedure actions are in conformance with TS 3.2.1 and 5.3.5. The smooth external surfaces of the FuelSolutions™ Storage System transfer cask and canisters facilitate decontamination. The procedures incorporate notes to indicate elevated dose rates, provisions for temporary shielding, and other ALARA practices during loading. Any radioactive effluents generated during canister loading will be governed by the 10 CFR Part 50 license conditions.

8.1.4 Draining and Drying

Based on the discussion below, the staff concludes that the SAR provides acceptable procedures for draining and drying the canister. The main intent of these procedures is to (1) remove water and oxidizing impurities from the canister cavity to protect the fuel cladding from degradation, and (2) ensure no significant annealing of the cladding occurs by maintaining the cladding temperature to less than or equal to 400 ^{o}C (see Section 4 for additional detail).

Once the shield lid has been welded in place and the dye penetrant examination of the inner closure plate weld is performed, compressed gas is used to force water from the canister cavity through the drain line. After the bulk of the water is removed a pump is used to draw a vacuum on the canister cavity. Precautions are given to control the evacuation rate and time at which the canister remains under vacuum conditions. Canister pressure is reduced to less than 3 torr and held for at least 30 minutes to verify the appropriate level of dryness is achieved. For the W21 canister, should the vacuum drying criteria not be met within twelve hours, then helium gas would be injected and maintained for four hours for canister cooldown. After that four-hour cooldown period, vacuum drying is repeated for another 8-hour period. The four hour cooldown and eight hour vacuum process can be repeated until the vacuum drying criterion is met (30 minutes at 3 torr). For the W74 canister, should the vacuum drying criteria not be met within seven hours, then helium gas would be injected and maintained for four hours for canister cooldown. After that four-hour cooldown period, vacuum drying is repeated for another 4-hour period. The four hour cooldown and four hour vacuum process can be repeated until the vacuum drying criterion is met (30 minutes at 3 torr). The procedures are outlined in TS 3.1.2 and 5.3 of the Administrative Controls of the respective canister SARs.

8.1.5 Filling and Pressurization

The FuelSolutions™ Storage System is backfilled with helium to slightly above atmospheric pressure. The canister is backfilled to a pressure of approximately 10 psig of helium, which is consistent with the helium backfill quantities in TS 3.1.1 of the appropriate canister SAR. A minimum helium purity of 99.995% is specified in the Operating Procedures. This will minimize contaminants in accordance with the recommendations of PNL-6365[1]. The procedure also states that the evacuation and backfill process must be repeated if the canister cavity is exposed to the atmosphere.

8.1.6 Canister Welding and Sealing

The operating procedures provide the steps to properly seal the canister, including helium backfill, and leak testing. Section 8 of WSNF-200 describes the steps for properly placing and welding the lid, drain port, and vent port covers that are consistent with the analyses presented in Sections 2 (design criteria), 3 (structural evaluation), and 9 (acceptance tests and maintenance program) of WSNF-200.

The canister is leak tested using helium mass spectrometry after being backfilled with helium. Leak test procedures are in accordance with ANSI/ANS N14.5-1997[2], as stated in Section 9 of WSNF-200. The combined leak rate for all closure seals is required by TS 3.1.3 to be less than 8.52×10^{-6} ref-cm^3/sec. The staff concludes that the canister sealing, leak test, and corrective actions described in the SAR provide an acceptable basis for development of site-specific procedures.

8.2 Cask Handling and Storage

8.2.1 Cask Handling

All accidents applicable to the transfer of the cask to the storage location are bounded by the design events identified and evaluated in Sections 2 and 11 of the FuelSolutions™ Storage System SAR. The structural (Section 3) and thermal (Section 4) evaluations presented in the SAR bound conditions that could potentially be created during cask lifting and transfer operations. Consistent with TS 4.2.1, the procedures ensure that the casks are spaced a minimum of 15 ft apart, center-to-center. The staff concludes that the procedures for cask handling provide a sufficient basis for development of detailed site-specific procedures.

8.2.2 Cask Storage

Inspection, surveillance, and maintenance requirements during the storage period are described in Section 9.2 of WSNF-200 in sufficient detail to permit cask users to develop detailed procedures. Maintenance operations, discussed in Section 9 of WSNF-200, are anticipated to be minimal over the lifetime of the cask. The staff concludes that the inspection, surveillance, and maintenance procedures provide an adequate basis for development of detailed procedures by cask users.

There will be no routine radioactive effluents generated during storage operations. Gaseous, liquid, and particulate releases from the cask cavity are not anticipated due to the welded canister. The external surfaces of the cask are decontaminated before it is transported to its storage location, so no significant contamination of the storage area is anticipated. Routine surveillance and maintenance activities do not introduce the potential for radioactive contamination. As a result, the staff concludes that no significant radioactive effluents are generated during storage operations.

8.3 Canister Unloading

As with the canister loading procedures, each cask user will be required to develop site-specific canister unloading procedures. The basis for the detailed user-developed canister unloading procedures is provided in Section 8.2 of WSNF-200. The general actions to be taken during unloading include transferring the cask to the spent fuel building, sampling the canister cavity gas, connecting fill and drain lines, reflooding the canister with borated water, removing the canister lids, lowering the cask into the pool, removing the shield plug, and removing the fuel assemblies from the storage basket. Several precautions are described to ensure that personnel are adequately protected during unloading operations. The staff concludes that the FuelSolutions™ Storage System is compatible with wet or dry unloading (dry unloading specifically for the W74 canister). In addition, the staff concludes that the generic cask unloading procedures presented in the FuelSolutions™ Storage System SAR will provide a sufficient basis for development of safe and effective detailed site-specific procedures.

8.3.1 Damaged Fuel

The SAR describes appropriate contingency actions to be taken prior to lid removal to detect damaged or degraded fuel in the canister. Degraded fuel would be detected via a cavity gas sample taken from the vent port. If degraded fuel conditions are suspected, additional measures are to be taken, by following the site-specific 10 CFR Part 50 procedures for handling damaged fuel, to prevent personnel contamination or exposure to airborne radioactive materials. The requirement for cover gas sampling prior to lid removal, and the special precautions provided are acceptable to the staff.

8.3.2 Cooling, Venting, and Reflooding

If the cover gas sample indicates the fuel is not degraded, the helium in the canister cavity is depressurized to atmospheric pressure, fill and drain lines are attached to the fill and drain ports in the canister lid, and the cask is lowered into the spent fuel pool. The unloading procedure cautions cask users to ensure that the fill and drain lines are designed for 100 psig to help protect against failures that could result in radiological exposures as well as personnel hazards (e.g., steam burns). Water is slowly added through the drain port to fill the canister and gradually cool the fuel.

An analysis of canister pressure during reflood operations is presented in SAR Section 4 to demonstrate that canister pressures remain below the 100 psig maximum reflood pressure limit. This analysis is the basis for controlling canister inlet water flow rates to 10 gallon per minute or less during the initial phase of canister fill. Cask users must develop site-specific reflood procedures that control fill rates to ensure that the design pressure of the cask is not exceeded. The staff concludes that actions to ensure subcriticality and prevent cask overpressurization were acceptable.

8.3.3 Fuel Crud

The FuelSolutions™ Storage System generic procedures incorporate precautions and procedural steps to prevent or mitigate the potential dispersal of fuel crud particulate material. These include a required vent and drain port gas sample prior to lid removal. The applicant provided a note in the unloading procedures to alert cask users to wait until any loose particles have settled, and to slowly move the fuel assemblies to minimize crud dispersion in the spent fuel pool. The applicant provided suggested crud contamination control measures, including auxiliary fuel pool filtration and enhanced airborne detection systems. The procedures and cautions regarding fuel crud were acceptable to the staff.

8.3.4 ALARA

The FuelSolutions™ Storage System unloading procedures incorporate general ALARA principles. ALARA practices include provisions to sample canister cavity gases to identify potential fuel cladding damage, monitoring of the water/steam ejected from the vent line during reflood, and temporary radiation shielding, where necessary. ALARA principles are also reflected in various warnings and notes included in the procedures. Each cask user will need to develop detailed unloading procedures that reflect the ALARA objectives of their site-specific radiation protection programs. The staff concludes that ALARA principles were adequately addressed in the FuelSolutions™ Storage System unloading procedures.

Any radioactive effluents generated during canister unloading are processed in accordance with the site-specific10 CFR 50 procedures as applicable.

8.4 Evaluation Findings

F8.1 The FuelSolutions™ Storage System is compatible with wet and dry loading and unloading (dry loading and unloading specifically for the W74 canister). General procedure descriptions for these operations are summarized in Section 8 of the SAR. Detailed procedures shall be developed and approved on a site-specific basis.

F8.2 The welded lids of the canister allow ready retrieval of the spent fuel for further processing or disposal as required.

F8.3 The smooth surface of the transfer cask and canister are designed to facilitate decontamination. Only routine decontamination will be necessary after the transfer cask and canister is removed from the spent fuel pool.

F8.4 No significant radioactive waste is generated during operations associated with the ISFSI. Contaminated water from the spent fuel pool will be governed by the 10 CFR Part 50 license conditions.

F8.5 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during canister loading and unloading will be governed by the 10 CFR Part 50 license conditions.

F8.6 The content of the general operating procedures described in the SAR are adequate to protect health and minimize damage to life and property. Detailed procedures will need to be developed and approved on a site-specific basis.

F8.7 Section 10 of this SER assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.

F8.8 The staff concludes that the content of the generic procedures and guidance for the operation of the FuelSolutions™ Storage System are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurances that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

8.5 References

1. Knoll, R.W., and Gilbert, E.R., "Evaluation of Cover Gas Impurities and their Effects on the Dry Storage of LWR Spent Fuel," PNL-6365, Pacific Northwest Laboratory, Richland, Washington, November 1987.

2. American National Standards Institute (ANSI), "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," ANSI N14.5-1997, New York, New York, February 1997.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The objective of the review of the acceptance tests and maintenance program is to ensure that the respective SARs includes the appropriate acceptance tests and maintenance programs for the FuelSolutions™ Storage System.

9.1 Acceptance Tests

The acceptance tests and inspections to be performed on the FuelSolutions™ Storage System are discussed in detail in Sections 9.1 of the respective SARs. These inspections and tests are intended to demonstrate that the FuelSolutions™ Storage System has been fabricated, assembled, and examined in accordance with the design criteria given in Section 2 of the respective SARs.

9.1.1 Visual and Nondestructive Examination Inspections

The visual and nondestructive examination criteria for the W150 Storage Cask and the W100 Transfer Cask are listed in SAR Tables 9.1-1 and 9.1-2 of the FuelSolutions™ Storage System SAR, WSNF-200. The visual and nondestructive examinations to be performed on the W21 and W74 storage canisters are discussed in detail in the Sections 9.1 and Tables 9.1-1 of the respective Canister Storage SARs, WSNF-201 and WSNF-203. These inspections and tests are intended to demonstrate that the FuelSolutions™ Storage System has been fabricated, assembled, and examined in accordance with the design criteria given in Section 2 of WSNF-200.

9.1.2 Structural

The structural performance of the FuelSolutions™ Storage Systems, i.e., the W150 Storage Cask, W100 Transfer Cask, and the W21 and W74 canisters, can be assured through adequate verification of the material properties, the dimensions, and the quality of construction. Materials and the material properties are verified to be in compliance with the design code through the receipt inspections. Nondestructive examinations, strength or leakage tests in accordance with the design codes are performed to demonstrate that the components are constructed to the required high quality.

9.1.2.1 W150 Storage Cask

The inspections and tests performed to assure W150 storage casks structural performance is described below:

1. Reinforcement strength and reinforcement placement are verified for each storage cask concrete segment. Representative samples of the concrete for each segment are tested to verify that the concrete property meets or exceed the minimum requirements for concrete quality in accordance with ACI318, Chapters 3 and 4. Ongoing testing for concrete mixture is to include taking of slumps, temperature, density, and air entrainment. In addition, the concrete compressive strength is based on tests performed on concrete cylinders aged 28 days.

2. The storage cask tie-rods, which joins the individual concrete segments together, are tensioned to 110,000 LBS. after the grout has cured.

3. Inspection and examinations of structural steel components are in accordance with Table 9.1-1 of WSNF-200.

4. The material and the dimensions of the storage cask impact limiter are verified. The compressive strength of the foam material is tested in accordance with Table 9.1-3 of WSNF-200.

9.1.2.2 W100 Transfer Cask

The inspections and tests performed for the transfer cask are the following:

1. The transfer cask steel components (cask body and covers) are fabricated and examined in accordance with ASME Code, Subsection NF as specified in Table 9.1-2 of WSNF-200.

2. Prior to installing the neutron shield, the upper trunnions are tested to 300 percent of the design load in accordance with NUREG-1536 and ANSI14.6. After sustaining the test load, critical areas are inspected by nondestructive surface examination.

3. A hydrostatic pressure test of the neutron shield cavity is performed during fabrication.

9.1.2.3 W21 and W74 Canisters

The FuelSolutions™ canisters are subjected to the following inspections and examinations:

1. Prior to basket insertion, all canister radiographs are reviewed to assure that they meet the design code construction requirements. Radiographic examination is required for the canister shell longitudinal and circumferential seam and shell-to-bottom closure plate welds in accordance with ASME Code, Subsection NB. The inner and outer top end closure plate welds, however, receive only progressive PT in accordance with ISG-4, Rev. 1, with the inner closure plate welds helium leak tested.

2. Following attachment of the bottom closure plate to the canister shell during shop fabrication, the cavity formed by the bottom plate, the shell and a temporary top closure are filled with helium to 12.5 psig. All circumferential and longitudinal full-penetration butt welds in the shell and bottom closure plate-to-shell welds are then pressure tested in accordance with Subarticle NB-6300 of the ASME Code.

3. Following the placement of the inner top closure plate and vacuum drying of the canister following fuel loading, the canister cavity is backfilled with helium to 12.5 psig. The inner top closure plate weld to the canister shell and the welds to the drain and vent port bodies are then pressure tested, as described in Section 8.1.8 of WSNF-200.

4. Following the successful pressure and leak rate tests of the inner closure plate welds and a second vacuum drying cycle of the canister cavity, the canister cavity is backfilled with helium in accordance with the technical specification in Section 12.3 of the respective canister SAR's. The

vent and drain port covers are seal-welded in place after the placement of the outer top closure plate.

5. As discussed in Section 2.1.2 of the respective FuelSolutions™ Canister SAR's, the canister internal basket assembly is designed and constructed as a core support structure in accordance the applicable requirements of Section III, Subsection NG of the ASME Code. Thus, appropriate inspections and examinations are performed for the canister basket assembly.

9.1.3 Leak Tests

The W150 Storage Cask and the W100 Transfer Cask are not confinement casks. Thus, they do not have any designated leak test requirements. The confinement function is provided by the W21 and W74 Canisters. The leak tests performed for the canisters are as the following:

1. Following the pressurization with helium of the canister cavity formed by the bottom closure plate, the shell, and a temporary closure, the canister shell circumferential and longitudinal full penetration butt welds and the bottom closure plate-to-shell welds are helium leak rate tested to a maximum leak rate of 8.52×10^{-6} ref-cc/s, as defined in ANSI N14.5-1997.

2. Following installation of the vent and drain block assemblies, the installed vent and drain port quick connect fittings are soap bubble leak tested.

3. Following the pressurization with helium of the cavity formed by the inner top closure plate and the canister shell, the inner top closure plate welds are helium leak rate tested in accordance with the technical specification requirements in Section 12.3 of WSNF-200.

9.1.4 Shielding Tests

Fabrication and testing controls for each shielding material are described in Section 9.1.2.5 and 9.1.3.5 of WSNF-200. The concrete utilized in the construction of the FuelSolutions™ storage cask shall be mixed, poured, and tested in accordance with the applicable provisions of ACI-318, and dimensions in the drawings. Concrete testing shall be performed for each lot of concrete and comply with ACI-318. The dimensions and chemical composition of the lead will be verified against the design drawings.

The effectiveness of the lead pours in the transfer cask body shall be verified during fabrication by performing gamma scanning of the cask in the lead pour region. Gamma scanning shall be performed in accordance with written and approved procedures. After first loading of each storage cask and transfer cask, radiation measurements will be performed to verify shielding effectiveness and to verify compliance with dose limits in the TS. The density of the solid neutron shield materials used in the top and bottom of the cask will be verified.

The staff reviewed the shielding fabrication testing and controls and effectiveness tests and found them acceptable. Each cask user will need to develop site-specific, detailed tests that incorporate the shielding effectiveness tests described in this section.

9.1.5 Neutron Absorber Tests

Fixed neutron absorber plates are used to ensure subcriticality during loading and unloading operations that use water inside the FuelSolutions™ W21 and W74 canisters. The W21 canister uses Boral plates and the W74 canister uses borated stainless steel plates.

After manufacturing, each batch of Boral is tested using wet chemistry and/or neutron attenuation techniques to verify presence, proper distribution, and minimum ^{10}B content. The test shall be representative of each Boral panel. The minimum allowable ^{10}B content is 0.02 gm/cm^2 for all plates in the W21 canister. Any panel with a ^{10}B loading less than the minimum allowed will be rejected.

Each batch of borated stainless steel for the W74 canister is also tested using wet chemistry and neutron attenuation analysis to verify the presence, proper distribution, and minimum ^{10}B content. The minimum allowable boron content in the stainless steel is 1.25 wt% natural boron. Any panel with a boron content less than the minimum allowed will be rejected.

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

Installation of the Boral and borated stainless steel panels on the W21 and W74 guide tubes shall be performed in accordance with written and approved procedures. Quality control procedures shall be in place to ensure that the canister guide tube walls contain a Boral or borated stainless steel panel as specified in each canister SAR Section 1.5 license drawings.

9.1.6 Thermal Acceptance

For the first system in place, the heat transfer characteristics of the cask system will be recorded by temperature measurements of the first storage cask placed in service with a heat load equal to or greater than 10 kW. In accordance with 10CFR72.4, 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 22 kW for the W21 canister and 24.8 kW for the W74 canister), the calculation and measured temperature data shall be reported to the NRC at every 2 kW increase. The calculation and comparison need not be reported to the NRC for Storage Casks that are subsequently loaded with lesser loads than the latest reported case.

9.1.7 Cask Identification

10 CFR 72.236(k) states that storage casks must be conspicuously and durably marked with (1) A model number; (2) A unique identification number; and (3) An empty weight. Sections 9.1.2.7 and 9.1.3.7 of the storage system SAR describe the cask identification nameplate, which contains the above information plus optional additional information, for the storage and transfer cask, respectively. Additionally, Section 9.1.7.2 of each canister SAR describes the identification method for the W21 and W74 canisters. The methods of identification described in the aforementioned SAR sections are an acceptable means of providing a unique, permanent, and

visible number to permit identification of the storage cask, transfer cask, and W21 and W74 canisters.

9.2 Maintenance Program

The maintenance programs are for FuelSolutions™ Storage System components that are classified as important to safety. Non-compliances encountered during the required maintenance activities will be dispositioned in accordance with the BFS Quality Assurance Program, discussed in Chapter 13 of the FuelSolutions™ Storage System SAR, or the licensee's NRC-approved Quality Assurance Program. The maintenance programs are intended to demonstrate that the FuelSolutions™ Storage System continues to perform properly, and to comply with regulatory requirements and TS contained in Chapter 12 of the FuelSolutions™ Storage System SAR.

9.2.1 W21 and W74 Canisters

The maintenance program for the W21 and W74 canisters is discussed in Section 9.2 of each respective canister SAR. The canisters rely on no mechanical components or moving parts. Exposed materials are corrosion resistant stainless steel. No inspection of the loaded canister during storage is required since the integrity of the canister is verified during fabrication, acceptance testing, and the canister closure procedure. Periodic temperature measurement of the FuelSolutions™ Storage Casks is required by the TS. The staff agrees that this is acceptable.

9.2.2 W150 Storage Cask

The FuelSolutions™ storage cask is a passive system requiring a minimal amount of maintenance. The licensee is to maintain records that include evidence that all maintenance and testing performed on a storage cask is in compliance with the Certificate and maintained under an NRC-approved quality assurance program. The maintenance program is summarized in Table 9.2-1 of WSNF-200.

9.2.3 W100 Transfer Cask

The W100 Transfer Cask is used for loading each canister into a storage cask and requires only a limited amount of periodic maintenance to properly perform its intended functions. The maintenance program is summarized in Table 9.2-2 of the WSNF-200 SAR. The licensee is to maintain records that include evidence that all maintenance and testing performed on the transfer cask is in compliance with the Certificate and maintained under an NRC approved quality assurance program.

9.3 Evaluation Findings

F9.1. Section 9.1 of the respective SARs describes the applicant's proposed program for preoperational testing and initial operations of the FuelSolutions™ Storage System. Section 9.2 of the respective SARs discusses the proposed maintenance programs.

F9.2 SSCs important to safety of the FuelSolutions™ Storage System will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. The respective SARs identify the safety importance of SSCs. The respective SARs present the applicable standards for their design, fabrication, and testing.

F9.3 The certificate holder/licensee will examine and/or test the FuelSolutions™ Storage System to ensure that it does not exhibit any defects that could significantly reduce its confinement and shielding effectiveness. Section 9.1 of the respective SARs describes the inspection and testing.

F9.4 The certificate holder/licensee will mark the cask with a data plate indicating its model number, unique identification number, and empty weight. The information to be placed on the data plates is described in Chapter 9 of the respective SARs and there are Engineering Drawings in Chapter 1 of the respective SARs that call out the location and required information for the data plates.

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

9.4 References

1 U.S. Code of Federal Regulations, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste,” Title 10, Part 72.

2. ASME Boiler and Pressure Vessel Code, Section III, Division I, 1995, including 1996 addenda.

3. American Welding Society, AWS 2.4, Standard Symbols for Welding, Brazing, and Nondestructive Examination, 1986.

4. American Society for Nondestructive Testing Recommended Practice SNT-TC-1A, Personnel Qualification and Certification in Nondestructive Testing, 1984.

5. American National Standards Institute, ANSI N14.6-1993, American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials, New York, June 1993.

6. American National Standards Institute, ANSI N14.5-1997, American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials, New York, February, 1997.

10.0 RADIATION PROTECTION EVALUATION

The NRC staff reviewed the radiation protection capabilities of the FuelSolutions™ Storage System to ensure that the system meets regulatory dose requirements.

10.1 Radiation Protection Design Criteria and Features

10.1.1 Design Criteria

The SAR lists four major sources of radiation protection design criteria, including 10 CFR Part 20, 10 CFR 72.104, 10 CFR 72.106, and Regulatory Guide 8.8[1]. This is consistent with NRC guidance. The cask users are responsible for demonstrating site-specific compliance with these requirements.

10.1.2 Design Features

Sections 10.1 and 10.2.1 of WSNF-200 describe the various radiological design features that provide radiation protection to operational personnel and members of the public. These radiation protection design features are summarized below:

- The thick concrete walls of the FuelSolutions™ storage cask provide shielding from gamma and neutron radiation.

- The confinement system includes a welded canister that prevents atmospheric releases of radioactive material. The confinement system is designed to maintain confinement of radioactive materials during normal, off-normal, hypothetical accident conditions, and severe natural phenomena events.

- The canister body consists of smooth surfaces to facilitate decontamination prior to transfer to the concrete storage cask, to minimize the time spent decontaminating a cask, and to reduce the quantity of radioactive waste generated during decontamination.

- ALARA principles are incorporated into cask design and operating procedures to minimize the occupational exposures.

Additional radiation protection features of the FuelSolutions™ Storage System include minimal maintenance and inspection requirements, location of cask monitoring instruments in an easily-accessible location, and adequate cask spacing in the ISFSI to facilitate surveillance activities.

The NRC staff evaluated the radiation protection design features and criteria for the FuelSolutions™ Storage System and found they provide reasonable assurance that the cask can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). In addition, all of the ALARA design considerations presented in Regulatory Guide 8.8 are addressed satisfactorily in Sections 8, 10.1.2, and 10.1.3 of the WSNF-200. Chapter 12 of WSNF-200 contains TS's on the maximum allowable surface dose rates and external surface contamination levels for the cask. Sections 5, 7, and 8 of the SER discuss the staff's evaluations of the shielding capabilities, confinement features, and operating procedure descriptions, respectively. Sections 11.1 and 11.2 of the SER discuss the NRC staff's evaluation of the FuelSolutions™ Storage System under off-normal and accident conditions, respectively.

10.2 Occupational Exposures

General operating procedure descriptions that each cask user will follow for cask loading, operation, unloading, and maintenance are presented in Section 8 of WSNF-200. Section 10.3 of the SAR presents estimates of: (1) the time and personnel requirements for these operations, (2) the dose rates in occupied areas where these operations occur, and (3) the doses received by personnel. Operational dose rates were taken from Section 5 of WSNF-200. The occupational dose calculations assume no temporary shielding is used. Occupational dose estimates for cask loading, and transfer to the storage cask are given in Tables 10.3-1 and 10.3-2 of WSNF-200, for vertical and horizontal canister transfer, respectively. The estimated total dose for cask loading, and transfer is as high as 2.16 person-rem per cask.

Annual maintenance doses were calculated for four maintenance activities. The highest maintenance exposure calculated was 0.457 person-rem per year for cask damage and vent obstruction inspections. TSs are provided that include surface dose rates (see TS 5.3.5) and surface contamination limits (see TS 3.2.1) to ensure that occupational exposures are within regulatory limits.

The staff reviewed the occupational dose estimates and determined that the analysis provides reasonable assurance that use of the cask can meet the occupational exposure requirements in 10 CFR Part 20. Actual occupational exposures will depend on site-specific operating procedures and special precautions (e.g., use of temporary shielding) taken to maintain exposures ALARA. Each licensee will have an established radiation protection program required by 10 CFR Part 20 Subpart B and will also be required to demonstrate compliance with occupational limits given in 10 CFR Part 20 Subpart C and other site-specific 10 CFR Part 50 license requirements.

10.3 Public Exposures

A SAR for a dry storage cask system provides an analysis of public exposures to facilitate site-specific analyses by a cask user. The SAR for the FuelSolutions™ Storage System provides estimates of the public exposures assuming the distance to the controlled area boundary is 100 to 500 meters. The staff's evaluation of the applicant's analysis of public exposures during normal (SER Section 10.3.1) and hypothetical accident conditions (SER Section 10.3.2) is summarized below. Based on the following review, the NRC staff concludes that the FuelSolutions™ Storage System design, along with appropriate site characteristics, can provide the required radiation protection for members of the public.

10.3.1 Normal and Off-normal Conditions

Sections 5.1, 5.4, 7.2, and 10.2.2 of WSNF-200 present the radiation dose analyses and results during normal and off-normal operations for the FuelSolutions™ Storage System. The analysis shows that the confinement functions of the cask are not affected by normal and off-normal conditions. In addition, the applicant performed an analysis of a continuous, non-mechanistic release of airborne radioactive material at the tested leakage rate of the confinement system.

Section 10.4 of WSNF-200 presents the results of the direct-path radiation dose calculations at distances of 100 to 500 meters from the cask. The total dose to a member of the public at the controlled area boundary is the sum of the contributions from atmospheric releases, direct-path radiation, and skyshine. The NRC staff's review of the atmospheric release calculations is presented in SER Section 7.3 and the evaluation of the applicant's direct-path (i.e., line-of-sight) radiation dose calculations is presented in SER Section 5. The analyses were determined to be acceptable. Skyshine dose rates for a single cask and array of casks containing a bounding fuel were included in the direct dose estimates using MCNP as described in Section 5 of the SAR. The dose rates are given in Tables 10.4-1 through 10.4-8 of WSNF-200.

The results of the applicant's site boundary analysis show that for a single cask with design-basis fuel and no berm, a minimum distance of approximately 225 meters is necessary to meet the 25 mrem/yr limit in 10 CFR 72.104(a). For a typical array of 64 casks a minimum distance of approximately 375 meters to the nearest real person is necessary to meet the regulatory limits.

The applicant's results provide reasonable assurance that a cask user can meet the requirements of 10 CFR 72.104(a). Each cask user or general licensee must perform a site-specific analysis as required by 10 CFR 72.212(b) to demonstrate compliance with 10 CFR 72.104(a) for normal operations and anticipated occurrences. The general licensee may consider site-specific conditions, such as actual distances to the nearest real person, topography, array configurations, characteristics of stored fuel, and use of engineered features, such as berms or walls, in their analysis of public doses. The site-specific analysis must also include the doses received from other fuel cycle activities (e.g., reactor operations) in the region.

A TS that requires measured dose rates to meet established limits (see Administrative Controls 5.3.5) is included in the SAR. The dose rate limits are used to identify casks which may cause the regulatory limits to be exceeded.

TS 4.3.2 has been included regarding engineered features used for radiological protection. The TS states that engineering features (e.g., berms and shield walls) used to ensure compliance with 10 CFR 72.104(a) are to be considered important to safety and must be evaluated to determine the applicable QA Category.

10.3.2 Accident Conditions and Natural Phenomena Events

The radiation exposures from accidents are presented by the applicant in Section 11.2 of WSNF-200. Accident conditions include hypothetical cask drop and tipover events, cask burial accidents, and possibly severe natural phenomena that could lead to a two-inch reduction in the thickness of the concrete shield and loss of one confinement barrier. The bounding dose is the sum of the direct radiation dose from loss of a portion of the concrete shielding and the atmospheric dose from the loss of confinement barrier integrity with 100% fuel cladding failure.

Time-integrated exposures were calculated by the applicant assuming an individual is located 100 meters from the cask for 30 days. The dose rate increase factors for direct exposures due to a loss of a portion of the concrete shield were determined and are shown in Figure 5.4-3 of WSNF-200. The dose rate increase factors were determined in the forward DORT shielding evaluation. The analysis of public doses from atmospheric releases caused by loss of confinement barrier integrity and 100% fuel cladding failure accidents is presented in Section 7.3of WSNF-200. The accident-related doses are the sum of the time-integrated direct dose and the dose from atmospheric releases.

The NRC staff's review of the direct dose rate calculations is presented in Section 5 of the SER. The results in Tables 10.4-7 and 10.4-8 of WSNF-200, were used to estimate the dose rate at 100 meters from the storage cask. The applicant also evaluated the transfer cask for accident conditions, assuming a loss of the neutron shield. The time-integrated radiation dose at 100 meters was calculated to be about 2900 mrem for an array of storage casks, assuming an individual is present for a year. The applicant determined that the loss of the neutron shield in the transfer cask results in a maximum dose rate of 25.3 mrem per 24 hours. The staff's review of the doses from a leaking confinement barrier and 100% fuel failure is presented in Section 7 of the SER. The total effective dose equivalent (TEDE) from this event was calculated to be 125 mrem at 100 meters from the cask and 751 mrem for the bone. The total dose of about 2900 mrem for the storage cask was found to be well below the 5 rem limit set forth in 10 CFR 72.106(b). The staff concludes there is reasonable assurance that the combined doses from direct radiation and atmospheric releases from bounding design-basis accidents and natural phenomena will be below the 5 rem regulatory limit specified in 10 CFR 72.106(b).

10.4 ALARA

The FuelSolutions™ Storage System shielding design incorporates a number of features to maintain radiation exposures ALARA. Operational ALARA policies, procedures, and practices are the responsibility of the site licensee as required by 10 CFR Part 20. The staff evaluated the ALARA assessment of the FuelSolutions™ Storage System and found it to be acceptable. TSs are provided that include surface dose rates (see TS 5.3.5) and surface contamination limits (see TS 3.2.1) to ensure that occupational exposures are maintained ALARA.

10.5 Evaluation Findings

F10.1 The FuelSolutions™ Storage System provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104 and 72.106.

F10.2 Occupational radiation exposures satisfy the limits of 10 CFR Part 20 and meet the objective of maintaining exposures ALARA.

F10.3 The staff concludes that the design of the radiation protection system of the FuelSolutions™ Storage System is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the FuelSolutions™ Storage System will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

10.6 References

1. Regulatory Guide 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable,” Revision 3, U.S. Nuclear Regulatory Commission, June 1978.

11.0 ACCIDENT ANALYSES

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

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

11.1 Off-Normal Events

Section 11.1 of the storage system and canister SARs examines the causes, radiological consequences, system performance, and corrective actions for off-normal conditions, as defined in ANSI/ANS 57.9-1992. These events can be expected to occur with moderate frequency or on the order of once per year. Section 2.3 of the SARs describes the design loadings for evaluating the combined load effects on the structural performance of the FuelSolutions™ Storage System. Table 2.3-1 of each canister SAR lists the load combinations for the W21 and W74 canisters. In addition to the environmental conditions and natural phenomenon events, the loads considered for the canisters include the dead weight, handling load, internal pressure, and thermal effects. Tables 2.3-6, 2.3-7, and 2.3-8 list the load combinations for the storage and transfer casks. In addition to the environmental conditions and natural phenomenon events, the loads considered for the storage and transfer casks include the dead weight, live load, drop or tip-over, and thermal effects. Loadings are combined in accordance with the load combinations identified in NUREG-1536[1] for reinforced concrete and structural steel components. The NRC staff reviewed the analyses for these conditions and found them to be acceptable. There is no adverse impact on the cask integrity from any off-normal event.

11.1.1 Severe Environmental Conditions (125°F and -40°F)

The applicant evaluated the FuelSolutions™ Storage System for a severe environmental temperature of 125°F and an insolation load of 62 Btu/hr-ft^2 for the sides of the storage transfer casks in the horizontal position, 123 Btu/hr-ft^2 for the top of the storage cask in the vertical position, and 31 Btu/hr-ft^2 for the top of the transfer cask in the horizontal position. The maximum decay heat for each canister (25.1 kW for the W21 and 26.4 kW for the W74) was modeled to determine maximum canister temperatures, and the storage and transfer cask thermal rating of 28 kW was modeled to determine maximum storage and transfer cask temperatures.

The applicant also evaluated the FuelSolutions™ Storage System for conditions with ambient temperatures of -40°F, no insolation, and the maximum decay heat load for each canister. For determining the storage and transfer cask temperatures, the thermal rating of 28 kW was used. The staff concurs with this approach since the largest radial thermal gradient would exist with the maximum decay heat load and thus produce the largest thermal stresses. Also, since the material of the canister is ductile stainless steel, it would not be susceptible to brittle fracture associated with the colder temperatures.

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

11.1.2 Cask Misalignment or Interference

The applicant evaluated the FuelSolutions™ Storage System for the effects of a cask misalignment or interference during horizontal canister transfer. Three scenarios are evaluated: 1) misalignment resulting in the canister jamming against the storage cask annulus shielding ring; 2) excessive friction between the canister and the storage or transfer cask rails; and 3) continued application of hydraulic ram pressure after the canister contacts the storage cask bottom end canister support tubes or the transfer cask top lid. The analyses provided in Section 3.6.3.1 of the storage system SAR and in Section 3.6.3 of each canister SAR show that all components of the storage system remain within the appropriate allowable values under the above scenarios. The estimated increase in occupational exposure is 456 person-mrem for bounding fuel, according to the personnel duration times in Section 10 of the storage system SAR. The staff concludes that the effects and consequences of this off-normal event are in compliance with the radiological dose limits from normal operations and anticipated occurrences provided in 10 CFR 72.104(a).

11.1.3 Hydraulic Ram Failure During Horizontal Transfer

The applicant evaluated the FuelSolutions™ Storage System for the effects of failure of the hydraulic ram during horizontal canister transfer operations. It is possible that a mechanical failure of the hydraulic ram could occur during canister transfer with the transfer only partially completed. A thermal analysis of the cask in the horizontal position for off-normal conditions, showing that no cask components exceed allowable temperatures, is performed in Section 4.5.2 of the storage system SAR. A similar analysis is performed in Section 4.5.2 of each canister SAR to show that no canister components exceed allowable temperatures. The estimated increase in occupational exposure is 104 person-mrem for bounding fuel, according to the personnel duration times in Section 10 of the storage system SAR. The staff concludes that the effects and consequences of this off-normal event are in compliance with the radiological dose limits from normal operations and anticipated occurrences provided in 10 CFR 72.104(a).

11.1.4 Off-Normal Internal Pressure

The applicant evaluated the FuelSolutions™ Storage System for the effects of off-normal canister internal pressure. The design basis off-normal pressure for each canister is 16 psig. The maximum canister internal pressure is calculated in Section 4.5 of each canister SAR assuming the required helium backfill, in accordance with the TS, elevated to the maximum off-normal canister cavity gas temperature. The non-mechanistic failure of 10% of the fuel rods, which release all of their fill gas and 30% of their fission gases, is also considered in the calculation. The structural response of the cask to the calculated pressure rise is discussed in Section 3.6.2 of each canister SAR, and all stresses are shown to be within the allowable values. There are no radiological consequences for this event.

11.1.5 Canister Reopening/Reflood

The applicant evaluated the FuelSolutions™ Storage System for the effects of reflooding the canister after the canister cavity has been drained and dried. Section 3.6.4 of each canister SAR evaluates the canister structural response to the internal pressure rise from the reflood and shows that all stresses are within allowable values. Section 4.4.2.3 of each canister SAR evaluates the thermal stress effects in the fuel cladding due to the reflood. Section 6 of each canister SAR shows that the most reactive configuration of fresh fuel and unborated water will remain subcritical, thereby bounding the conditions expected during reflood. The occupational radiation exposure incurred during the reflood operation is estimated to be equivalent to that incurred during canister draining, drying, and closure operations.

11.2 Accident and Natural Phenomenon Events

Section 11.2 of the storage system and canister SARs, determines the radiological dose consequences for the identified design basis accidents and natural phenomena events. The SAR determined that the FuelSolutions™ Storage System has adequate design margins and would reasonably maintain its confinement function during and after design basis accidents. The staff concurs that all appropriate accident and natural phenomena events have been identified and all potential safety consequences considered.

11.2.1 Fully Blocked Storage Cask Inlet and Outlet Vents

Although it is considered unlikely that all eight storage cask inlet and outlet vents will be completely blocked, the applicant considers complete blockage of all vents as a result of unlikely phenomena such as a tornado or flood. Vent blockage results in increased temperatures inside the storage cask from the loss of air flow. Section 4.6 of the storage system and canister SARs provides a transient analysis of the storage cask with blocked vents, starting at the normal steady-state temperatures. The analysis shows that the concrete is the first component to reach its short-term allowable temperature of 350°F, at a time of 43 hours after the start of the transient. Section 4 of each canister SAR shows that the canister shell, basket, and fuel cladding temperatures are well below their short-term allowable temperatures at 43 hours into the transient. Section 4 of each canister SAR also shows that the pressure increase due to the temperature rise associated with the vent blockage remains below the design basis accident pressure of 69 psig.

Since the FuelSolutions™ Storage System retains its shielding performance, the radiological consequences of this event are low. Personnel dose associated with recovery actions to restore the air flow path is the most significant consequence. Assuming the debris removal operation does not require more than two individuals for one hour, the additional occupational exposure is estimated by the applicant to be about 20 person-mrem. The consequences of this event are acceptable since the occupational exposure remains within 10 CFR Part 20 limits and there is no associated increase in dose to the public.

11.2.2 Storage Cask Drop

The applicant evaluated the storage cask for a non-mechanistic 36-inch vertical end drop onto the bottom end of the cask. The only credible drop scenarios are failure of the vertical cask transporter, resulting in a six-inch end drop onto the roadway, or a drop while lifting the cask onto the upender/downender J-skid, resulting in a thirty-inch end drop. The evaluated 36-inch end drop bounds both scenarios.

Section 3.7.3 of the storage system SAR evaluates the response of the concrete storage cask to this event. The results of the stress evaluation show that there will be loss of concrete around the site of the impact, but there will be no compressive failure of the concrete, no permanent deformation of the top cover plate, and the cask will allow for recovery or repair after the end drop. Section 3.7.3 of each canister SAR provides the structural evaluation of the canister shell and basket assemblies during the 36-inch cask end drop. The resulting stresses for all canister components are within allowable values.

The storage cask, canister, and fuel cladding temperatures may increase as a result of damage to the cask inlet and outlet vents or heat shield. This condition is bounded by the blockage of all cask inlet and outlet vents discussed in Section 11.2.1 of this SER. The loss of concrete shielding in the area local to the area of the cask which contacted the storage pad is expected to have little or no effect on the shielding performance of the cask. Occupational exposure will increase due to the recovery of the canister from the damaged cask. The additional occupational exposure is expected to be equivalent to that of retrieving a canister from an undamaged storage cask, which was evaluated and found to be acceptable in Section 10 of this SER.

11.2.3 Storage Cask Tip-over on J-Skid

The applicant evaluated the storage cask for a tip-over while being up-ended or down-ended on the J-skid. This event could occur as a result of a failure of the cask upender/downender during rotation between the vertical and horizontal positions. The analysis of the cask structural response to tip-over is discussed in Section 3.7.4 of the storage system SAR. This analysis indicates that cracking or spalling of the outer concrete may occur where the cask rests on the J-skid, but this will not affect the structural integrity and the cask will allow for recovery or repair after the event. The structural effects of the cask tip-over event on the canister shell and basket assembly, guide tubes, and fuel cladding are bounded by those for the transfer cask side drop, discussed in Section 11.2.4 of this SER.

The storage cask, canister, and fuel cladding temperatures may increase as a result of damage to the cask inlet and outlet vents or heat shield. This condition is bounded by the blockage of all cask inlet and outlet vents discussed in Section 11.2.1 of this SER. The thermal effects due to the cask being in the horizontal position after the tip-over event are bounded by those for the normal conditions horizontal position analysis. The loss of concrete shielding in the area local to the area of the cask which contacts the J-skid is expected to have little or no effect on the shielding performance of the cask. Occupational exposure will increase due to the recovery operations, consisting of uprighting the cask and transferring the canister from the damaged cask to a transfer cask. The additional occupational exposure is expected to be approximately equal to that of retrieving a canister from an undamaged storage cask, which was evaluated and found to be acceptable in Section 10 of this SER.

11.2.4 Transfer Cask Drop

The applicant evaluated the transfer cask for a horizontal side drop of 72-in. Although unlikely due to the fact that the transfer cask is very stable in its horizontal transfer configuration and the trailer and skid are robust heavy industrial equipment with high safety margins, the 72-in side drop is considered in order to bound any postulated transfer cask drop or trailer tip over scenarios. The analysis of the transfer cask structural response to the side drop is discussed in Section 3.7.5 of the storage system SAR. This analysis shows that the resulting stresses are below allowable levels and that the transfer cask satisfies the applicable stability requirements. The neutron shield is lost during this event, but this result has no effect on the structural integrity of the cask. Section 3.7.5 of each canister SAR discusses the effect of the side drop on the canister shell and basket assembly and shows that the basket guide tubes receive a small permanent deformation as a result of this event.

The loss of the neutron shield will result in decreased heat transfer and a rise in cask, canister, and fuel cladding temperatures. Section 4.6.2 of the storage system and canister SARs show that the maximum temperatures remain below allowable values. The loss of the neutron shield will also result in a significant decrease in neutron shielding, along with a minor decrease in gamma shielding. The resulting dose rates for bounding fuel are 6.8 rem/hour at the cask surface, 2.3 rem/hour at 1 meter, and approximately 1 mrem/hour at 100 meters.

Section 3 of each canister SAR evaluates the effects of the transfer cask side drop on each canister basket. The small permanent deformation of the guide tubes is evaluated in Section 6 of each canister SAR to determine its effect on reactivity. The analysis in Section 6 shows that the system remains subcritical.

Occupational exposure is expected to increase significantly due to the recovery operations. The total additional dose for the recovery is 2500 person-mrem, assuming that five individuals work in close proximity to the cask for two hours, temporary neutron shielding is in place, work is performed mainly at the ends of the cask where the radiation fields are lower, and general dose fields are reduced to an estimated 250 mrem/hour. The consequences of this event are acceptable since the occupational exposure remains within 10 CFR Part 20 limits and there is no associated increase in dose to the public.

11.2.5 Fire

The applicant evaluated the storage cask and transfer cask for their performance in a hypothetical accident conditions fire. This event is unlikely, due to the absence of significant combustion sources within twenty feet of the storage casks. However, it is considered that storage casks may be affected by fires in nearby foliage, manmade structures, vehicles used for ISFSI operations and maintenance, or fuel carried by those vehicles. Section 3.7.6 of the storage system SAR provides a discussion of how the storage cask is affected by the fire. Some spalling of the concrete will occur, but the cask will remain largely intact and allow for normal unloading of the canister into a transfer cask. Section 3.7.6 also provides an evaluation of the transfer cask response to the fire event. Some of the transfer cask structural materials will reach the temperature of the fire for a short period of time, but will not significantly degrade as a result of the transient. All structural and gamma shielding materials remain below their allowable temperatures. The neutron shield is assumed to be lost as a result of this accident scenario. Section 4.6 of each canister SAR discusses the maximum internal pressure resulting from the fire accident and shows that it is less than the design basis accident pressure for each canister. This section also shows that canister shell, basket, and fuel cladding temperatures all remain below their allowable temperatures.

The surface dose rates will increase from 50 mrem/hour to 125 mrem/hour due to the spalling of exterior concrete from the storage cask. The loss of the transfer cask neutron shield will cause a significant increase in neutron dose. This shielding effect is evaluated in Section 11.2.4 of the storage system SAR. Occupational exposures will increase as a result of the recovery operations for the storage cask. Assuming two individuals working one eight-hour day per cask, the occupational exposure will be 400 person-mrem for each cask affected. As discussed in Section 11.2.4 of this SER, the occupational exposure due to transfer cask recovery with no neutron shield is estimated to be 2500 person-mrem. The consequences of this event are acceptable since the occupational exposure remains within 10 CFR Part 20 limits and there is no associated increase in dose to the public.

11.2.6 Explosive Overpressure

The effects of the design basis tornado wind load are assumed to bound any credible overpressure loads on the FuelSolutions™ Storage System. The effects of the design basis tornado on the storage and transfer casks are evaluated in Section 11.2.8 of the storage system SAR.

11.2.7 Flood

The applicant evaluated the storage cask for the effects of the design basis flood. The probability of a flood event is specific to each ISFSI, and should consider tsunamis and seiches, as well as high water from a river or broken dam. Section 3.7.8 of the storage system SAR shows that the storage cask remains upright and does not slide under the design basis flood event. The structural consequences of the flood event are considered to be minor compared to those of the tornado, earthquake, and drop accidents. The effects of the flood accident on the canisters are bounded by the accident internal pressure effects discussed in Section 11.2.8 of each canister SAR.

Under flood conditions, the storage cask vents and annular space around the canister are filled with water. This situation precludes heat transfer by air flow through the storage cask annulus, but water provides for far better heat transfer than air and results in lower temperatures than under normal conditions. If flood waters cover only the inlet vents, or they are blocked by debris resulting from the flood, then this scenario is bounded by the consideration of blocked inlet and outlet vents discussed in Section 11.2.1 of the SARs.

11.2.8 Tornado Winds and Missiles

The applicant evaluated the FuelSolutions™ Storage System for the effects of design basis tornado wind and tornado wind-driven missile loads. Tornado probability is dependent on the geographic location of the ISFSI. It is assumed that a tornado could occur during storage and transfer operations. The responses of the storage cask and the transfer cask are discussed in Section 3.7.9 of the SAR. These analyses show that the storage cask and transfer cask remain upright during the tornado accident. The storage cask can slide if a missile impact is combined with the tornado wind load, but contact with another cask as a result of sliding would not occur due to the spacing between the casks.

The storage cask will not experience any thermal effects due to the tornado event. The transfer cask will lose its neutron shield, due to tornado wind-driven missile impact. This will result in decreased heat transfer to the outside of the cask, as discussed in Section 4.6.2 of the storage system SAR. This section shows that the transfer cask temperatures will remain below their allowable values during this event.

The storage cask will also not experience any reduction in shielding effectiveness due to the tornado event. The loss of the transfer cask liquid neutron shield, however, will result in a significant increase in neutron dose rate, and a small increase in gamma dose rate. The resulting dose rates are the same as those for the transfer cask drop accident. Occupational exposure would increase as a result of the recovery operation. The total occupational exposure of 2500 person-mrem is obtained using the assumptions discussed in Section 11.2.4 of this SER. The consequences of this event are acceptable since the occupational exposure remains within 10 CFR Part 20 limits and there is no associated increase in dose to the public.

11.2.9 Earthquake

The applicant evaluated the FuelSolutions™ Storage System for the effects of a design basis earthquake. Earthquakes are naturally occurring events that can strike without warning and vary in magnitude with geographic location. Section 3.7.10 of the SAR shows that the storage cask and transfer cask remain upright and do not slide under the design basis earthquake event. Section 3.7.8 of each canister SAR shows that all canister stresses due to earthquake event are below maximum allowable values. There are no other effects on the FuelSolutions™ Storage System as a result of this event.

11.2.10 Accident Internal Pressure

The applicant evaluated the W21 and W74 canisters for an accident internal pressure based on the required helium backfill elevated to the extreme off-normal canister cavity gas temperature, concurrent with a non-mechanistic failure of 100% of the fuel rods which release all of their fill gas and 30% of their fission gases. Section 3.7.9 of each canister SAR evaluates the stresses produced in each canister as a result of the design basis internal pressure. The evaluation shows that all stresses are within allowable values.

Section 4.6 of each canister calculates the internal pressure that would result from the temperature and fuel rod rupture scenario described above. The design basis pressure used in the structural evaluation bounds that calculated for this event.

Section 7.3 of each canister SAR evaluates the effects of the increased internal pressure on the leak rate for each canister. The increased release from each canister will result in a small increase in the dose to the public, calculated in Section 7.3 of each canister SAR. The dose rates to the public remain less than the limit in 10 CFR 72.106(b).

11.3 Criticality

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

11.4 Post-Accident Recovery

Section 11.2 of the storage cask and canister SARs discusses corrective actions for each accident identified in Section 11.2. There are no credible design basis accidents that would affect the canister confinement boundary or significantly damage the cask system at a level that could result in undue risk to public health and safety.

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

11.5 Instrumentation

Because of the passive nature of the FuelSolutions™ Storage System, no instrumentation and control systems are needed to monitor SSCs important to safety. Therefore, there are no instrumentation and control systems that must remain operational under accident conditions. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection. Since the W21 and W74 canisters both use an entirely welded redundant closure system and, under normal and off-normal conditions, there are no anticipated mechanisms that would cause weld failure, no direct monitoring of the closure is required.

11.6 Evaluation Findings

F11.1 The SSCs of the FuelSolutions™ Storage System are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that do occur.

F11.2 The spacing of casks, discussed in storage system SAR Section 1.4, ensures accessibility of the equipment and services required for emergency response.

F11.3 Table 12-1 of this SER lists the TS, Approved Contents and Design Features for the FuelSolutions™ Storage System. These are further discussed in Section 12 of the SER.

F11.4 The applicant has evaluated the FuelSolutions™ Storage System to demonstrate that it will reasonably maintain confinement of radioactive material under credible accident conditions.

F11.5 A design basis accident or a natural phenomenon event will not prevent the retrieval of spent fuel for further processing or disposal.

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

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

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

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

12.0 CONDITIONS FOR CASK USE - TECHNICAL SPECIFICATIONS

The conditions for cask use are reviewed to ensure the applicant has fully evaluated the TS and that the SER incorporates any additional operating controls and limits that the staff determines are necessary.

12.1 Conditions for Use

The conditions for use of the FuelSolutions™ Storage System are fully defined in the CoC and the TS which are appended to it.

12.2 Technical Specifications

SER Table 12-1 lists the TS for the FuelSolutions™ Storage System, which includes the W150 storage cask, the W100 transfer cask, the W21 canister, and the W74 canister. The staff has appended these TS to the CoC for the FuelSolutions™ SFMS.

12.3 Evaluation Findings

F12.1 Table 12-1 of the SER lists the TS for the FuelSolutions™ Storage System. These TS are further discussed in Section 12 of the SAR and are part of the CoC.

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

Table 12-1 FuelSolutions™ Storage System Technical Specifications

Number	Technical Specification
1.0	USE AND APPLICATION
1.1	Definitions
1.2	Logical Connectors
1.3	Completion Times
1.4	Frequency
2.0	FUNCTIONAL AND OPERATING LIMITS
2.1	Functional and Operational Limits
2.2	Functional and Operating Limits Violations
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
3.0	SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	CANISTER INTEGRITY
3.1.1	Canister Helium Backfill Density
3.1.2	Canister Vacuum Drying Pressure
3.1.3	Canister Leak Rate
3.1.4	Hydraulic Ram Force During Horizontal Canister Transfer
3.1.5	Canister Vertical Time Limit in Transfer Cask
3.2	CANISTER RADIATION PROTECTION
3.2.1	Cask Surface Contamination
3.3	STORAGE CASK INTEGRITY
3.3.1	Storage Cask Air Inlet and Outlet Openings
3.3.2	Storage Cask Temperatures During Storage
3.3.3	Storage Cask Temperatures During Horizontal Transfer
3.4	STORAGE CASK RADIATION PROTECTION
3.4.1	Storage Cask Dose Rates
3.5	TRANSFER CASK INTEGRITY
3.5.1	Transfer Cask Structural Shell Temperature
3.6	TRANSFER CASK RADIATION PROTECTION
3.5.1	Transfer Cask Surface Contamination
4.0	DESIGN FEATURES
4.1	Storage System
4.2	Storage Pad
4.3	Site Specific Parameters and Analyses

Table 12-1 FuelSolutions^{TM} Storage System Technical Specifications

Number	Technical Specification
5.0	ADMINISTRATIVE CONTROLS
5.1	Training Modules
5.2	Preoperational Testing and Training Exercises
5.3	Programs
5.4	Special Requirements for the First System in Place

List of Tables

13.0 QUALITY ASSURANCE

Part 72 of Title 10 of the Code of Federal Regulations provides for “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.”[1] Subpart G of 10 CFR Part 72 describes Quality Assurance (QA) requirements applying to ISFSIs.

The FuelSolutions™ Storage System SAR section on QA states that all quality related activities will be controlled under an NRC approved quality assurance program meeting the requirements of 10 CFR Part 72. The BFS QA Program was reviewed and approved by staff under separate correspondence.

13.1 References

1. U.S. Code of Federal Regulations, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste,” Title 10, Part 72.

14.0 Decommissioning

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

14.1 Decommissioning Considerations

The conceptual decommissioning plan for the FuelSolutions™ Storage System is provided in Section 14 of the storage system SAR. Section 14.1 provides an analysis of storage cask, transfer cask, and canister activation after the 20-year service life. The results of this analysis indicate that storage system components will become activated to a relatively low level. The applicant states that the preferred approach to decommissioning would be to decontaminate the storage system components using conventional techniques, and to reuse the casks and canisters to the maximum extent practicable, provided they have not reached the end of their design life. An alternative approach would be to keep the storage system components at the licensee's ISFSI site and allow them to decay until they can be released for unrestricted use. In addition, since Section 14 of the SAR shows that none of the storage system components will be activated beyond 10 CFR Part 61 Class A waste limits, they can be disposed of as Class A waste, with no additional decay required.

14.2 Evaluation Findings

F14.1 The FuelSolutions™ Storage System design includes adequate provisions for decontamination and decommissioning. As discussed in Section 14 of the SAR, these provisions include facilitating decontamination of the FuelSolutions™ Storage System, if needed; storing the remaining components, if no waste facility is expected to be available; and disposing of any remaining low-level radioactive waste.

F14.2 Section 14 of the SAR also presents information concerning the proposed practices and procedures for decontaminating the cask system and disposing of residual radioactive materials after all spent fuel has been removed. This information provides reasonable assurance that the applicant will conduct decontamination and decommissioning in a manner that adequately protects public health and safety.

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

CONCLUSION

The staff reviewed Revision 4 to the Safety Analysis Report for the FuelSolutions^{TM} Storage System. Based on the statements and representations contained in the SAR and the conditions in the Certificate of Compliance, the staff concludes that the FuelSolutions^{TM} Storage System meets the requirements of 10 CFR Part 72.

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