TRANSNUCLEAR, INC.

TN-32 DRY STORAGE CASK SYSTEM

SAFETY EVALUATION REPORT

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INTRODUCTION

This Safety Evaluation Report (SER) documents the review and evaluation of Revision 11A to the Safety Analysis Report (SAR) for the Transnuclear, Inc. (TN) TN-32 Dry Storage Cask System.[1] The SAR, submitted by TN, follows the format of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.[2] This SER uses essentially the same Section-level format, with some differences implemented for clarity and consistency.

The review of the SAR addresses the handling and dry storage of spent fuel in a single dry storage cask design, the TN-32. 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 TN-32 cask is unloaded and subsequently decontaminated before disposition or disposal.

References

1. TN-32 Dry Storage Cask Safety Analysis Report, Rev. 11A, Transnuclear Inc., January 1999.

2. NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."

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.

1.0 GENERAL DESCRIPTION

The objective of the review of the general description of the TN-32 dry storage cask system is to ensure that Transnuclear, Inc. has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the cask.

1.1 System Description and Operational Features

The TN-32 cask accommodates 32 intact pressurized water reactor (PWR) fuel assemblies, with or without burnable poison rod assemblies (BPRAs), and/or thimble plug assembles (TPAs), and consists of the following components (see Figure 1-1):

- A basket assembly which locates and supports the fuel assemblies.
- An inner confinement vessel (and lid) which comprises the primary confinement barrier.
- A carbon steel gamma shield structure surrounding the primary confinement vessel.
- Neutron shielding material (jacketed) exterior to the gamma shield.
- A protective cover which provides weather protection for the closure lid and seal components, the top neutron shield, and the overpressure system.
- An overpressure monitoring system which monitors pressure between the two seals of the cask lid. This system allows for early detection of cask seal leakage.
- Sets of upper and lower trunnions for lifting and support of the cask.

There are three versions of the TN-32 cask. The standard TN-32 cask has a standard lid. The TN-32A has a shorter lid assembly and longer cavity. The TN-32B is identical to the TN-32 standard, except that the top lifting trunnions are designed as single failure proof.

TN-32 casks are to be stored at a minimum of 16 ft apart, center to center.

1.2 Drawings

The drawings for the TN-32 associated with the structures, systems, and components (SSCs) important to safety are contained in Section 1.5 of the SAR[1]. The applicant provided sufficiently detailed drawings regarding dimensions, materials, and specifications to 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 TN-32 are specified in the Technical Specifications (TS). The TN-32 cask is designed to store up to 32 intact PWR fuel assemblies manufactured by Westinghouse (W), with or without BPRAs or TPAs. The maximum allowable enrichment of the fuel to be stored is 4.05 wt% U^{235}. A description of the fuel assemblies is provided in Section 2.1 of the SAR.

1.4 Qualification of the Applicant

TN provides the design, analysis, licensing support, and quality assurance (QA) for the TN-32 cask. Fabrication of the cask is done by one or more qualified fabricators under TN's QA program. Section 1.3 of the SAR adequately details TN's technical qualifications and previous experience in the area of dry storage cask licensing.

1.5 Quality Assurance

The quality assurance program (QAP) is addressed in Section 13 of this SER.

1.6 Evaluation Findings

F1.1 A general description and discussion of the TN-32 is presented in Section 1 of the SAR (Rev 11A), 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 SAR. Specific SSCs are evaluated in Sections 3 through 14 of this SER.

F1.3 Specifications for the spent fuel to be stored in the dry storage cask system are provided in Section 2 of the TN-32 SAR.

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

F1.5 The QAP is described in Section 13 of the SAR and is addressed in Section 13 of this SER.

F1.6 The TN-32 cask was not reviewed in this SER for use as a transportation cask.

F1.7 The staff concludes that the information presented in Section 1 of the SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is based on a review that considered the regulation itself, Regulatory Guide 3.61, and accepted dry storage cask practices detailed in NUREG-1536[2].

1.7 References

1. TN-32 Dry Storage Cask Safety Analysis Report, Rev. 11A, Transnuclear Inc., January 1999.

2. NUREG-1536, Standard Review Plan for Dry Cask Storage System, January 1997.

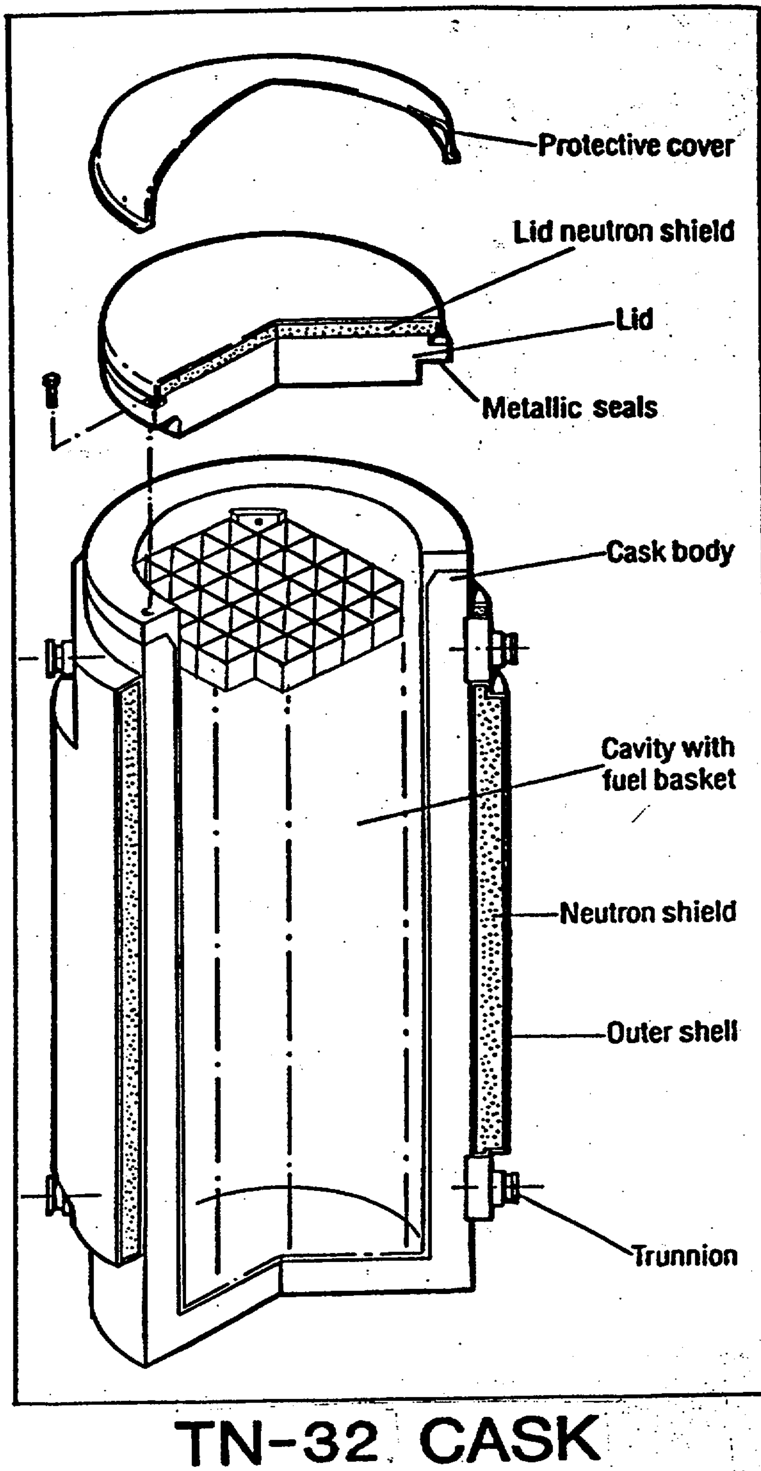

Figure 1-1 TN-32 Dry Storage Cask

2.0 PRINCIPAL DESIGN CRITERIA

The objective of evaluating the TN-32 principal design criteria related to the SSCs important to safety is to ensure that they comply with the relevant general criteria in 10 CFR Part 72.

The TN-32 cask was approved for use by the NRC under a 10 CFR Part 72 site-specific license[1]. That approval was based on the review of Revision 9A of the TN-32 TSAR[2]. Subsequently, the applicant requested a Certificate of Compliance (CoC) under the general license provisions of 10 CFR Part 72. To support this request, TN submitted Revision 11A of the SAR[3], which included changes to the cask design criteria and spent fuel specifications. Therefore, the scope of this review is limited to the changes from the approved TSAR Revision 9A and any other reviews necessary to ensure that the TN-32 meets the 10 CFR Part 72 requirements.

2.1 Changes in the TN-32 Design and Design Criteria

There are three TN-32 cask designs;

- The standard TN-32 cask, which was described in the Topical Safety Analysis Report (TSAR) approved by the NRC for referencing in a 10 CFR Part 72 site-specific license application[2];

- The TN-32A cask, which is identical to the standard TN-32, except that it has a longer cask cavity and a re-designed lid to accommodate longer fuel; and

- The TN-32B cask, which is identical to the standard TN-32, except that the top lifting trunnions are designed as single failure proof.

The major changes to the design criteria include;

- The fuel specifications include a wider variety of fuel assemblies, which may or may not contain BPRAs and TPAs.

- The fuel enrichment specification has been increased from 3.85 wt% to 4.05 wt%.

- The fuel burnup has been increased from 40,000 to 45,000 MWD/MTU.

- The maximum heat load specification has been increased from 27 kW to 32.7 kW.

- The limit for maximum fuel cladding temperature has been decreased from 348°C to 328°C.

- The design-basis temperature limits were changed from a maximum of 115°F and a minimum of -20°F to the following: A maximum average daily ambient temperature of 100°F and a minimum average daily ambient temperature of -20°F.

- The Seismic Design-basis Earthquake has been increased from .12g horizontal and .08g vertical to .26g horizontal and .17g vertical.

2.2 Structures, Systems, and Components Important to Safety

Table 2.3-1 and Drawing 1049-70-2 of the SAR identify the cask SSCs important to safety. For the cask components classified as not important to safety, TN provided justification for their exclusion in Section 2.3 of the SAR.

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

The TN-32 design criteria summary includes the allowed range of spent fuel configurations and characteristics, the enveloping conditions of use, and the bounding site characteristics.

2.3.1 Spent Fuel Specifications

The TN-32 is designed to store 32 intact, unconsolidated, Westinghouse PWR spent fuel assemblies with or without BPRAs and TPAs. Section 2 of the SAR provides detailed fuel assembly parameters which include the fuel type, uranium mass, assembly mass, enrichment, burnup, and cooling time. This section of the SAR also specifies which fuel types are bounding for the criticality, shielding, thermal, and confinement analyses within the SAR.

The fuel characteristic limits are given in TS 2.1. These limits are based on the criticality, shielding, thermal, and confinement analyses which are evaluated in Sections 3 through 14 of this SER.

2.3.2 External Conditions

Section 2.2 of the SAR identifies the bounding site environmental conditions and natural phenomena for which the TN-32 is analyzed. These are evaluated in Sections 3 through 14 of this SER.

Sections 2 and 11 of the SAR identify the normal, off-normal, and accident conditions evaluated. The staff's evaluation of the TN-32 response to the off-normal and accident conditions is in Section 11 of this SER. TS 4.3 identifies the bounding site-specific parameters for the TN-32.

2.4 Design Criteria for Safety Protection Systems

2.4.1 General

Section 2 of the SAR states that the minimum design life of the TN-32 is 40 years. The material mechanical properties analysis in Section 3.3 of the SAR is for a design life of 20 years and is evaluated for 20 years in Section 3 of this SER.

The codes and standards of design and construction are specified in Sections 2.5, 3, and 7 of the SAR. Justification for exceptions to codes and standards is given in TS 4.1.3. SSCs important to safety are designed, fabricated, and tested to quality standards which conform to the criteria of 10 CFR Part 72.

The TN-32 has a pressure monitoring system which meets the intent of the continuous monitoring requirement of 10 CFR Part 72. This is evaluated in Section 7 of this SER.

2.4.2 Structural

Section 3 of the SER evaluates the structural integrity of the TN-32 under the combined normal, off-normal, and accident loads. Loading combinations are classified as Service Conditions, consistent with Section III of the ASME Boiler and Pressure Vessel code[4], and the resulting stresses are evaluated. The TN-32 structural components are designed to protect the cask contents from significant structural degradation, preserve retrievability, and maintain subcriticality and confinement.

2.4.3 Thermal

Section 4 of this SER evaluates the TN-32 thermal design criteria. Normal condition thermal design criteria include confinement of radioactive material and gases, maintaining fuel cladding integrity, and maintaining the neutron shield integrity. The TN-32 is designed to passively reject decay heat and the heat removal mechanisms are independent of intervening actions under normal and off-normal conditions.

2.4.4 Shielding/Confinement/Radiation Protection

Sections 5, 7, and 10 of this SER evaluate the TN-32 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 TN-32 uses a bolted lid closure system, double metallic lid and lid penetration seals, and a combined cover-seal pressure monitoring system to provide confinement. Radiation exposure is minimized by the neutron and gamma shields and by operational procedures.

2.4.5 Criticality

The TN-32 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, minimum basket cell opening, prevention of fresh water entering the cask, and loading/unloading the cask in borated water with a minimum specified boron concentration. The continued efficacy of the neutron absorber plates over a 20-year storage period is assured by the design of the TN-32 cask. 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.

2.4.6 Operating Procedures

The operating procedure descriptions, which are evaluated in Section 8 of this SER, include procedures for loading and unloading. Radiation protection features, including features to facilitate decontamination, are incorporated in both the physical design and the operating procedures.

2.4.7 Acceptance Tests and Maintenance

The TN-32 acceptance tests and maintenance programs are evaluated in Section 9 of this SER.

2.4.8 Decommissioning

The TN-32 decommissioning considerations are presented in Sections 2.4 and 14 of the SAR and evaluated in Section 14 of this SER.

2.5 Review Summary

TN presented general details of the principal design criteria in Section 2 of the SAR and provided appropriate details in the associated Sections of the SAR.

2.6 Evaluation Findings

F2.1 The staff concludes that the principal design criteria of Revision 11A of the TN-32 SAR 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 acceptable engineering practices. More detailed evaluations of the design criteria and assessments of compliance with those criteria are presented in Sections 3 through 14 of this SER.

2.7 References

1. Safety Evaluation Report for the Transnuclear Inc. Dry Storage Cask (TN-32), Docket 72-1021 (M-56), Nov. 1996.

2. TN-32 Dry Storage Cask Topical Safety Analysis Report, Rev. 9A, Transnuclear, Inc., Dec. 1996.

3. TN-32 Dry Storage Cask Safety Analysis Report, Rev. 11A, Transnuclear Inc., January 1999.

4. ASME Boiler and Pressure Vessel Code, Section III, Division I, 1992.

3.0 STRUCTURAL EVALUATION

This section evaluates the structural design of the TN-32 cask. Structural design features and design criteria are reviewed, and analyses related to structural performance under normal, off-normal, accident, and natural phenomena events are evaluated.

3.0.1 Scope

There are three variations of the TN-32. The basic design (designated TN-32) was previously presented in Rev. 9A of the TN-32 TSAR[1] and approved by the NRC[2] for referencing in a Part 72 site-specific license application. Subsequently, the applicant requested an approval for use under the general license provisions of 10 CFR Part 72. To support this request, TN submitted Revision 11A of the SAR[3], which included changes to the cask design criteria and spent fuel specifications. Therefore, this structural evaluation focuses primarily on the "changes" in design and the structural analyses from those evaluated in Reference 2.

3.0.2 Methods of Evaluation

Loads and load combinations are reviewed for the normal, off-normal, accident, and natural phenomena events categorized in NUREG-1536[4]. Structural material specifications are reviewed and compared with acceptable codes and standards. Design assumptions and analytical approaches are reviewed for appropriateness and acceptability. Critical stresses and the construction of the TN-32 cask components are reviewed to ensure they meet the acceptance criteria of the design codes and standards.

3.1 Structural Design

3.1.1 Structural Design Features

The TN-32 cask consists of the following four cask components:

1. Confinement boundary
2. Non-confinement boundary
3. Fuel Basket
4. Trunnions

The confinement boundary of the TN-32, identified in Figure 1.2.1 of the SAR, consists of the inner shell (both the cylindrical portion as well as the bottom plate), the closure flange out to the seal seating surface, and the lid assembly outer plate. The lid bolts and seals are also part of the confinement boundary.

The non-confinement boundary components consist of the gamma shielding, neutron shield outer shell, and trunnions. While these components do not have a containment function, they must react to the confinement or environmental loads, and in some cases, share loading with the confinement components.

The classification of components as "important to safety" and "not important to safety" is contained in Section 2 and summarized in Table 2.3.1 of the SAR. Components considered important to safety include the containment vessel, lid bolts and gasket, lid vent and drain covers and bolting, basket assembly, trunnions, radial neutron shield, and the lid protective cover. Items considered "not" important to safety include the overpressure system, drain tube, Hanson couplings, paint, top neutron shield, and top protective cover seal.

3.1.2 Structural Design Criteria

The SAR uses several design criteria to ensure that the cask design meets the requirements of 10 CFR Part 72. Section 3.1 of the SAR describes the design criteria for the four cask components listed at the beginning of Section 3.1.1 above.

The main design code/standard used for the TN-32 design is the ASME Boiler and Pressure Vessel (BPV) Code[5]. This applies to all three TN-32 designs. The particular ASME portion used is Division 1, Section III, Subsection NB (1992 editions). The SAR also points out that this "criterion" is used "to the maximum practical extent," and exceptions to strict usage are given in Section 7 of the SAR and the TS. The TN-32 cask is not code stamped. The QA requirements of Nuclear Quality Assurance-1 (NQA-1) or 10 CFR 72 Subpart G are imposed in lieu of specific provisions dealing with fabricator qualifications, etc., usually covered by Subsection NCA of Section III, Division 1. Load conditions are categorized as either "Normal" or "Hypothetical;" the former category being compared to Level A service limits and the latter being compared to Level D service limits. It is noted by the staff that the use of the ASME Section III, Division 1, Subsection NB application very closely resembles the more recent ASME Section III, Division 3, specifically designed to address nuclear packaging. In summary, the staff agrees that the following elements of the ASME code are utilized in the design:

1. Material selection and certification.
2. Allowable material stress values.
3. Stress categorization procedures (membrane, membrane plus bending, etc.)
4. Selection of weld types and weld inspection procedures.

3.1.2.1 Individual Loads

Load conditions for both normal and off-normal events are described in Sections 2 and 3 of the SAR. Summary data is presented in SAR Tables 2.2-5 through 2.2-9. Rather than grouping the loading as normal, off-normal, and accident level, the loads were designated either "normal" (Level A) or "accident" (Level D). Cask components under off-normal conditions use the same allowable stresses as for normal conditions.

Normal loading for the cask and contents is described in Section 2 of the SAR. In particular, Tables 2.2-5 and 2.2-6 describe loads categorized as normal. It should be noted that loads specified as design loads for the cask represent a subset of those categorized as level A, or normal loads. The structurally significant normal loading conditions are primarily loadings due to internal and external pressure and lifting loads. Other loads result in only minor structural effect.

Load conditions categorized as "off-normal" and "accident-level" in NUREG-1536 are grouped as Level D loadings in Section 2 of the SAR and are summarized in Table 2.2-7. It is noted that the "fire accident" case, while evaluated in Section 3 of the SAR, is omitted from Table 2.2-7.

In addition, "explosive overpressure", (see Section 3.1.2.1.3 below) is omitted from the table and Section 3. All other accident level natural phenomena load cases listed in NUREG-1536 are listed and treated either in Sections 2 or 3 of the SAR.

SAR Tables 2.2-8 and 2.2-9 list "normal" and "accident" load cases and the series of combinations in which the stress levels for the individual load cases are combined. It is noted by the staff that thermal stresses are omitted from Table 2.2-9 (accident conditions) but evaluated in Section 3.4.6 of the SAR.

3.1.2.1.1 Tipover

The TN-32 cask will not tipover as a result of a postulated natural phenomenon event, including tornado wind, tornado-generated missile, seismic event, or flood. To demonstrate the defense-in-depth features of the design, a non-mechanistic tipover scenario is analyzed. Section 3.3.1 below discusses the tipover analysis performed in the SAR.

3.1.2.1.2 Handling Accident

Handling accidents for the TN-32 cask are considered to be side and end drop events. These are evaluated in Section 3.3.2 below.

3.1.2.1.3 Explosive Overpressure

Explosive overpressure is not addressed in either Sections 2 or 3 of the SAR. The cask is designed to withstand an external pressure of 25 psi. If a credible explosion is identified that would apply more than 25 psi to the outer surface of the cask, each specific site will have to address this issue in its 10 CFR 72.212 evaluation.

3.1.2.1.4 Flood

Flood loading is addressed in Section 2 of the SAR. The TN-32 cask is evaluated for a water level of 57 ft and water velocity of 25 ft/sec.

3.1.2.1.5 Tornado and Tornado Missile

Tornado and tornado missile loadings are addressed in Section 2 of the SAR. The TN-32 cask is evaluated for a design-basis tornado wind velocity of 360 mph and a pressure drop of 3 psi. Tornado missiles are listed in Section 2.2 of the SAR. Stability of the TN-32 cask due to tornado missile impact is evaluated in Section 3.4.4 below.

3.1.2.1.6 Earthquake

The design earthquake for use in the design of an ISFSI must be equivalent to the safe shutdown earthquake (SSE) for the nuclear power plant, the site of which has been evaluated under the criteria of 10 CFR Part 100, Appendix A. The TN-32 cask is evaluated for an applied horizontal acceleration of 0.26g and a vertical acceleration of 0.17g. These earthquake inertia forces are assumed to be applied at the top of the concrete pad. Section 3.4.2 below evaluates the seismic events.

3.1.2.1.7 Snow and Ice

Snow and ice loadings are addressed in Section 2 of the SAR. Section 3.4.5 below evaluates snow and ice loadings of 50 psf on the TN-32 cask.

3.1.2.1.8 Lightning

The effects of lightning on the cask are not directly evaluated in the SAR. Due to the massive size of the cask and the highly conductive carbon steel construction, it is concluded that lightning would not pose a structural concern for the TN-32 cask.

3.1.2.1.9 Fire

Temperatures from the thermal analysis of a fire event performed in Section 4 of the SAR are utilized in Appendix 3B to evaluate the thermal stress response of the cask. These stress values are reported in Tables 3A.2.3-9 and 3A.2.3-10 of the SAR. Due to the low values of stress observed, the staff concurs that thermal stress effects of the fire are acceptable.

3.1.2.2 Loading Combinations

Loading combinations used in the SAR are listed in Table 2.2-8 for normal conditions and Table 2.2-9 for accident conditions. The staff agrees that these combinations simulate the structural events modeled.

3.1.2.3 Allowable Stresses

Allowable stress values for the various cask materials are listed in SAR Tables 3.3-1 and 3.3-4. The staff concludes that these values meet the ASME allowable stresses, based on the appropriate ASME subsections and service levels, and that appropriate considerations to elevated thermal effects were given. Therefore, they are acceptable.

3.1.3 Weights and Center of Gravity

Weights of various cask components for each of the three designs are listed in SAR Table 3.2-1. In addition, the location of the center of gravity of each of the three designs is given. A conservatively high weight is used for most of the structural analyses. A conservatively low weight and high center of gravity are used for the analysis of the stability of the cask.

3.1.4 Materials

The structural materials used for the TN-32 (all designs) are listed in Section 3 of the SAR. A tabulation of the cask components is given in Table 3.3-6, which lists the primary function of each component along with information on drawing number, if applicable; safety class, codes/standards (including welding); coatings; and pertinent service conditions such as stress, temperature, time, pressure, environment, and important mechanical properties. In addition, Section 3.3 of the SAR discusses mechanical properties of materials and other tabulations of pertinent mechanical properties.

Material properties are generally taken from Section III, Part D of the ASME BPV code (1992 edition) when possible. These materials are either Class 1, 2, or 3 materials, or they do not belong to these classes. Materials other than ASME Code materials are permitted as discussed in NUREG-3854, "Fabrication Criteria for Shipping Containers", and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" for the fabrication of casks. The materials used for the cask body (gamma shield, containment shell, bottom, and top) are various grades of carbon steels and are described on page 3.1-1 of the SAR.

The containment shell and bottom plate are designed with SA 203 Grade D, which is not a Class 1 material. The Grade D alloy was selected in part because it has good ductility and adequate strength for the levels of stress to be encountered by this component. The staff notes that this material is likely to have good weldability in comparison with Grade E, which contains a higher allowable carbon content. The lid material selected was SA-350, Grade LF3, which is not listed as a Class 1 material in the 1992 editions (without addenda). It is listed as a Class 2 and Class 3 material. In 1996, a special code case (N-559) gives Class 1 approval for this alloy and provides allowable stress limits.

The cylindrical gamma shield shell is SA-266 Grade 2 material. The top gamma shield plate is SA-105 or SA-516 Grade 70, and the bottom shield plate is fabricated from either SA-266 Grade 2 or SA-516 Grade 70. All of these are Section III, Class 1 materials.

Materials of the TN-32 fuel basket are described in Section 3.1.2.3 of the SAR. No structural consideration is given to the potential load carried by the basket's borated aluminum plates. The aluminum plates (SB-209 6061-T6/T651) are primarily heat conductors and are not used for structural analyses under normal operating loads and accident end drop load. They are assumed to be effective in transmitting the load by contact only during the short duration dynamic loading from a tipover accident. Load-bearing materials are the aluminum basket rails (6061-T6) and the stainless steel square tubing (SA-240 Type 304). The 6061-T6 alloy used in the basket rails is not for Class 1 ASME applications but is for Class 2 and 3 applications. While the temperatures during normal operation exceed Classes 2 and 3 limits, the applicant gives industry data[6] to arrive at allowable stress values at these temperatures. The basis for the allowable stress for the Type 304 stainless steel fuel-compartment box is Section III of the ASME Code.

The trunnion material for all TN-32 designs is SA-105. This material is listed as ASME Class 1 material. SAR Table 3.3-2 gives allowable stress values taken from Table 2A of ASME Section II Part D.

The fracture toughness of ferrous components is assessed in Appendix 3E of the SAR. This is done in the process of determining pre-service and in-service inspection requirements and allowable flaw sizes for various loading conditions and temperatures.

3.1.4.1 Material Compatibility and Durability

Compatibility of materials used in fabrication of the TN-32 is addressed in Section 3.4.1 of the SAR, which reviews chemical, galvanic, and other interactions among the materials and contents for the environmental conditions encountered during the various phases of service (loading, storage, handling, and unloading). Discussions for environmental conditions associated with each phase of service are presented in Section 3.4.1 of the SAR. Components of the system are expected to have excellent corrosion resistance, compatibility with one another, and durability in their respective environments. Periodic maintenance will be done, as needed, on external coatings. A seal replacement can be accomplished if needed during the lifetime of the system.

The interior of the cask contains an aluminum spray that is anodic with respect to other components with which it is in contact. Alterations, with respect to surface damage, are not expected on the aluminum interior coating of the cask in a helium environment, and they are insignificant in borated water. Six days of exposure to borated water would be required to corrode away one micron of the 0.004-inch thick (minimum) aluminum/aluminum oxide coating. Only minimal amounts of hydrogen would be evolved as a result of this corrosion activity. The SAR, using historical data, demonstrates the ability of aluminum to resist corrosion from boron ions in water. Only insignificant corrosion is expected from galvanic, pitting, or crevice effects of the aluminum or the metals that it is expected to sacrificially protect. Other corrosion behaviors, such as intergranular and stress corrosion, are not expected to occur.

The Type 304 stainless steel components and welds of the interior basket assembly are not expected to be significantly affected by the adverse presence of either borated water or the other environmental conditions of temperature and time under service conditions. The staff concurs that no chemical reactions between the stainless steel plates, the aluminum plates, and the borated aluminum plates could affect the areal density of the borated aluminum sheet material, which is sandwiched between the aluminum plates and the stainless plates.

There is one potential exception to the lifetime corrosion resistance of the TN-32 system. Corrosion could occur at the crevice formed where the outer metallic seal contacts the sealing surface. The moisture necessary for this crevice corrosion to occur is not likely to be present because the combined effects of the weather cover and the decay heat from the stored fuel will maintain a low humidity at this seal. Staff notes that this seal material has a very good record of performance and endurance and is not expected to fail during the licensed life of the system. If this seal were nevertheless to somehow fail during this period, there would be no safety significance as the failure would automatically be detected before any adverse effects related to the cask function would occur. A replacement seal would be installed for continued service.

The staff concurs that the factors affecting service performance; chemical reactions, galvanic reactions, or other reactions, and interactions between materials and the environment are not likely to lead to detrimental effects during handling and storage operations of the 20-year licensed service period of the cask.

3.1.4.2 Welds

Weld specification and inspection techniques are discussed in Section 3.1.1 of the SAR. Various standards of the ASME Boiler and Pressure Vessel Code are applied. Confinement boundary weld types are in accordance with Section III, Subsection NB. Acceptance standards are those of Article NB-5000. Test standards are in accordance with Section V, welding standards are in accordance with Section IX, and materials are in accordance with Section II, part C. The staff concurs that sufficient detail to welding has been given in the cask design.

3.1.4.3 Bolting Materials

Lid bolts for the TN-32 cask are SA-320 Grade L43, which is an ASME approved material for Class 1 components. The bolts are expected to be sufficiently durable to serve during the 20-year license.

3.1.4.4 Brittle Fracture of Materials

Fracture toughness of the TN-32 cask confinement boundary, gamma shield, and welds is addressed in Appendix 3E of the SAR. Each is evaluated below.

The TN-32 cask is designed for an ambient temperature of -20°F. The confinement boundary material will be tested to ensure that it is not susceptible to brittle fracture at -20°F. The tests include the following materials acceptance tests in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB:

1. Conduct Charpy V-notch (CVN) tests at -20° F, which is the lowest service temperature, LST, for this application.

2. Conduct tests required to establish the nil ductility transition temperature (T_{NDT}) of -80°F. T_{NDT} = LST - 60°F, where the Lowest Service Temperature, LST, is -20°F.

In addition to the above materials acceptance tests, the allowable flaw sizes in the design of the confinement boundary are calculated using a linear elastic fracture mechanics (LEFM) methodology, from Section XI of the ASME Code (1989 edition). The results of the fracture toughness analysis indicate that the critical flaw sizes in the confinement boundary that would result in unstable crack growth or brittle fracture are larger than those typically observed in the plate or forged steel components.

The plate and forging materials used in the confinement boundary are examined by the ultrasonic method in accordance with ASME Code Section III, Subsection NB, Paragraph NB-2530 and NB-2540, respectively. The external and accessible internal surfaces of the forging materials are examined by the liquid penetrant test or the magnetic particle test in accordance with Paragraphs NB-2546 or NB-2545. The welds are examined by the radiographic and either the liquid penetrant or magnetic particle tests in accordance with Section III, Subsection NB, Paragraphs NB-5210, NB-5220, and NB-5230. Any defects at or above those specified in the TN-32 confinement boundary allowable flaw depth Tables in Appendix 3E of the SAR will be repaired prior to cask use for storage. Lid bolts, which are part of the confinement boundary, will meet the fracture toughness criteria of ASME Code, Section III, Division 1, Subsection NB (Paragraph NB-2333).

The gamma shield is not part of the confinement boundary, however, it provides structural support to the confinement boundary during drop accidents. Cracks in the gamma shield will not propagate into the confinement boundary. The gamma shield will not separate from the confinement boundary, due to the frictional forces between the confinement vessel and the gamma shield which arise as a result of a shrink fit of the gamma shield shell over the containment shell. The results of the fracture mechanics analyses indicate that the critical flaw sizes (flaws large enough to give rise to rapid unstable extension) are larger than those typically observed in forged steel and plate components flaws, and this is true for flaws either in the gamma shield shell or in the top or bottom shield plates. Therefore, no special examination is required of the gamma shield to ensure the absence of flaws that would result in unstable crack growth or brittle fracture.

As for the gamma shield welds, it is noted that failure of either the weld between the gamma shield and top flange or the weld between the top shield plate and lid would have no safety significance. The gamma shield will not separate from the confinement boundary and the top shield will still remain inside the confinement boundary due to the cask arrangement. Therefore, only liquid penetrant or magnetic particle tests of the final surface are specified for these two welds. If the bottom plate weld were to fail, the bottom plate could become detached, which would impact the shielding capability of the cask. At -20°F, the minimum allowable flaw sizes for surface and subsurface are 0.29 in. and 0.58 in., respectively. The following inspections (made prior to placing the cask in service) are required to ensure that large defects (those equal to or larger than the above flaw sizes) are detected and repaired:

1. liquid penetrant or magnetic particle test at base metal
2. liquid penetrant or magnetic particle test at root pass
3. liquid penetrant or magnetic particle test for each 0.375 -inches of weld
4. liquid penetrant or magnetic particle test at final surface

The liquid penetrant or magnetic particle test will be in accordance with Section V, Article 6 of ASME Code.

3.1.4.5 Materials Conclusion

The staff concludes that the materials of construction as specified in the TN-32 cask design are adequate for the service requirements and safety functions (structural, thermal, shielding, criticality, and confinement).

3.1.5 General Standards for Cask

The structural analyses for the cask must ensure positive closure, adequate safety factors for lifting devices, and that there is no adverse effect to the safe storage of the spent fuel due to chemical or galvanic reactions. The most important function of structural analyses is to show sufficient structural capability of the TN-32 system to withstand the postulated worst-case loads under normal, off-normal, accident, and natural phenomena events with adequate margins of safety to preclude the following consequences:

1. unacceptable risk of criticality,
2. unacceptable release of radioactive materials,
3. unacceptable level of radiation, and
4. impairment of retrievability.

The structural analyses presented in Section 3 of the SAR demonstrate that the cask will maintain containment during normal and off-normal operations, accident conditions, and natural phenomena events. Section 2 of the SAR justifies that the cask will maintain containment for natural phenomena events. In addition, results from SAR Section 3 (Appendices 3A, 3B, and

3C) indicate that gross ruptures will not occur in the fuel cladding during accident conditions. In Appendix 3D, a finite element model is used to determine cask response due to a tipover event. The results from that effort demonstrate that fuel damage (sufficient to cause retrieval concerns) will not occur during tipover.

Normal, off-normal, accident and natural phenomena loading will not be sufficiently severe to cause degradation of the gamma shield performance. However, the neutron shield may be damaged by either tornado Missile A or B. Radiological effects due to a loss of the neutron shield are addressed in Section 10 of this SER.

The above mentioned SAR analyses are evaluated in Sections 3.1.4, 3.2, 3.3, and 3.4 of this SER.

3.2 Normal Operating and Off-Normal Conditions

3.2.1 Chemical and Galvanic Reactions

Discussion of potential chemical and galvanic reactions is given in SAR Section 3.4.1. In this SER, these reactions are discussed with other considerations in Section 3.1.4, especially Section 3.1.4.1 on material compatibility and durability. The staff concurs that such reactions have been sufficiently addressed in the design and do not adversely affect cask performance.

3.2.2 Positive Closure

The TN-32 cask lid is bolted directly to the upper ring forging. Access to the lid requires removal of the protective cover. Deliberate loosening of bolts requires extensive effort, using appropriate equipment. The large preload applied to lid bolts prevents inadvertent opening of the cask closure lid from loads such as bottom end drop and thermal expansion. Therefore, the TN-32 cask cannot be opened unintentionally.

3.2.3 Lifting Devices Analysis

Structural effects due to lifting loads passed from the trunnion to the gamma shield are reported in Section 3.4.3.1 of the SAR, in accordance with ANSI N14.6[7]. The TN-32 and TN-32A designs do not have single failure proof lifting trunnions, while the trunnions of the TN-32B do have single failure proof upper trunnions. As required by N14.6, the TN-32 and TN-32A trunnions are designed with a safety factor of 3 against the trunnion material yield stress and a safety factor of 5 against the trunnion material ultimate stress. The TN-32B is designed with a safety factor of 6 against the trunnion material yield stress and 10 against the trunnion material ultimate stress. Stresses in the trunnion material are evaluated using beam shear and bending calculations at several cross-sections of the trunnion. The results for the two designs are reported in SAR Tables 3.4-1A and 3.4-1B.

Stress concentrations caused by the trunnion loads acting on the gamma shield were analyzed using the techniques of Welding Research Council-107 (WRC-107)[8]. These local stresses are superimposed on the stresses of the ANSYS[9] cask body model of SAR Appendix 3A in arriving at the stress values used in evaluations.

3.2.4 Pressure and Temperature Effects

Stress levels in the cask body for normal and off-normal conditions are evaluated in Appendix 3A of the SAR. Finite element modeling was performed using an axisymmetric model to assess stresses due to pressure and temperatures. Temperatures used were taken from thermal models discussed in Section 4 of the SAR. ASME code checks are performed in accordance with Level A load conditions of Subsection NB and were found to be acceptable.

3.3 Accident Conditions

3.3.1 Cask Tipover and Side Drop

The tipover analysis of the TN-32 cask is provided in Appendix 3D of the SAR. The methodology used in performing the analysis was developed by the Lawrence Livermore National Laboratory (LLNL)[10]. This methodology was verified by LLNL through comparison of analyses results with test data.

The TN-32 cask is conservatively assumed to be a rigid body. The peak rigid body accelerations of the TN-32 cask due to a tipover accident are predicted analytically using the LS-DYNA3D[11] finite element program. The TN-32 finite element model is made up of four components: the cask body, cask internals, concrete, and soil. The concrete pad and the soil foundation are assumed to duplicate the concrete pad and soil in Reference 10. Essential parameters of the four components are listed in Section 3D.2.3 of Appendix 3D of the SAR. The finite element models of the cask body and the cask internals are also developed in a similar manner to the model represented in Reference 10. Features on the cask such as the trunnions, neutron shield, and protective cover are neglected in terms of stiffness, but their weight is lumped into the density of the cask body. Mesh sizes of the cask, basket, concrete, and soils are in reasonable agreement with those represented in Reference 10. Contact elements are used between the cask and concrete pad and between the concrete pad and the soil. The result of the analysis indicates that the TN-32 cask has a peak deceleration of 67g at the top end of the cask. Based on the comparisons of this analysis with the LLNL analyses, as well as full scale end drop tests performed by BNFL, the staff concurs that the applicant has adequately validated the finite element modeling technique and the LS-DYNA3D finite element program.

Although the tipover analysis of the TN-32 cask results in a peak deceleration of 67g at the top of the cask, the side drop accident event with the design deceleration of 50g along the length of the cask is used to perform the stress analysis in the TN-32 SAR (also discussed in Section 3 of Reference 2.) The staff agrees that the side drop event stresses will envelop the tipover accident event because the deceleration varies according to the distance from the center of rotation in a tipover accident. Thus, along the axial length of the TN-32 cask, the minimum deceleration (0g) would occur at the bottom end and the maximum deceleration (67g) would occur at the top surface of the lid. This corresponds to a 33.5g uniform load along the axial length that is less than the 50g uniform load resulting from the side drop accident event. While the static analysis due to a 50g uniform load results in a peak stress intensity of about 54,000 psi, the tipover dynamic analysis indicates a peak stress of about 25,000 psi. This shows that the static analysis in the TN-32 SAR has an additional margin of safety of approximately 50% in the cask stresses. The tipover analysis neglects the outer shell and aluminum boxes. These components will deform and absorb energy during the tipover accident.

Therefore, the actual deceleration would be less than the 67g peak deceleration calculated by the tipover analysis.

Because the basket structure is designed using a quasi-static analysis, a dynamic load factor of 1.1 is computed from the transient dynamic analysis. Thus, the load on the basket as a result of tipover can be modeled as a steady-state acceleration equal to 74g (67 x 1.1 = 74). The structural analysis of the basket is performed using 88g for the accident analysis in the TN-32 SAR (also discussed in Section 3 of Reference 2).

3.3.2 Cask Bottom-End Vertical Drop

In SAR Revision 11A, the concrete compressive strength for the reinforced concrete pad used for the evaluation of cask drops has been revised to 6,000 psi rather than the 3,000 psi as indicated in Reference 2. This would increase the upper bound deceleration after the 18-inch bottom end drop from 36g to 42g. However, the end drop analysis in Reference 2 assumed a conservatively high value of 50g deceleration (also discussed in Section 3 of Reference 2.) Therefore, the conclusion reached in Reference 2 is not affected by the revised concrete compressive strength.

3.3.3 Cask Lid Bolt Analysis for Cask Impact

All three TN-32 lid bolt designs use 48 1.5-inch diameter steel bolts. The lid bolt analysis is performed using the TN-32A lid because it is slightly heavier (by about 230 lbs.) than the standard TN-32 lid assembly. The details of the analyses are provided in Section 3A.3 of the SAR. Bolt preload is selected to resist the maximum internal pressure in the cask cavity (100 psi) plus any dynamic loading such as those for the hypothetical bottom end drop and tipover onto the concrete storage pad. Quasi-static analyses are performed using "g" levels from the corresponding cask impact models described in Appendices 3A (bottom end drop) and 3D (tipover) of the SAR. Analyses results indicate that the maximum normal and accident condition stresses are less than allowable values with a substantial margin of safety. Lid gasket compression is maintained at all times since bolt preload is higher than the applied loads during normal and accident condition loads.

3.3.4 Fuel Basket Analysis

In Revision 11A, no changes or revisions were made to the TN-32 fuel basket analysis from that presented in Reference 1. Conclusions reached in Reference 2 are not changed.

3.3.5 Spent Fuel Response due to Cask Impact Events

Appendix 6A of the SAR assesses the response of a typical fuel assembly during end and tipover/side impact events. The analyses are quasi-static and utilize beam models to assess both axial and flexure response. The primary objective of the fuel response modeling was to assess the likelihood of gross fuel failure during such an event.

The methodology used in performing the fuel rod side impact stresses is based on work done at LLNL[12]. The fuel gas internal pressure is assumed to be present and the resulting axial tensile stress is added to the bending tensile stress due to 74g loads, which is taken from SAR

Appendix 3D (Section 3.3.1 above). The stresses for different Westinghouse fuel assembles are provided in SAR Table 6A-1. The results indicate that the stress in the most vulnerable fuel assembly, W-17x17 OFA, is lower than the yield strength of the irradiated zircaloy. Thus, the integrity of the fuel rods will not be breached during the tipover/side drop accident.

For the bottom end drop event, an elastic-plastic stress analysis is performed using the ANSYS Finite Element Program. The analysis uses a three-dimensional finite element model of entire active fuel rod length. The inertial forces load the rod as a column having intermediate supports at each grid support (spacer). In addition, two alternate methods are also performed to verify the finite element analysis result. All three analyses use the irradiated material properties and include the weight of fuel pellets. All three methods of analyses concluded that the fuel cladding tubes will not be damaged during a bottom end drop. The finite element analysis and Alternate Method #1 indicate that the calculated buckling load in terms of "g" is greater than the design-basis "g" loading for the cask bottom end drop of 50g (Section 3.3.2 above). Alternate Method #2 indicates that the calculated maximum stress in the fuel assembly is lower than the yield strength of the irradiated zircaloy.

The staff is in general agreement with the conclusions of SAR Appendix 6A which indicate that gross failure of the fuel is unlikely for the anticipated impact events.

3.4 Extreme Natural Phenomena Events

3.4.1 Flood Condition

No changes or revisions were made to the analysis of flood conditions presented in Reference 1. The conclusion reached in Reference 2 is not changed.

3.4.2 Seismic Events

The TN-32 cask is analyzed for seismic loads in Section 2.2.3 of the SAR. The cask is conservatively considered as a rigid body placed on the concrete pad and equivalent static analysis methods are used to calculate loads and overturning moments. The coefficient of static friction of 0.35 and a lower bound cask weight of 218,000 lbs. are used to calculate the maximum amount of frictional force available to prevent sliding. Based on the analyses, the TN-32 cask will neither slide nor tipover due to a seismic event with an applied horizontal acceleration of 0.26g and vertical acceleration of 0.17g. Because the minimum static coefficient of friction between the steel cask and the concrete pad could be as low as 0.3 in the references cited by the applicant, users of the TN-32 cask will be required to verify that the coefficient of friction for their concrete pads is either greater than or equal to 0.35. This requirement is described in TS 5.2.1.

3.4.3 Tornado and Wind Loadings

No changes or revisions were made to the analysis of tornado and wind loadings presented in Reference 1. Conclusions reached in Reference 2 are not changed.

3.4.4 Tornado Missile Impact

The analysis to determine the cask response to a tornado generated missile impact is provided in Section 2.2.1.2.2 of the SAR. The TN-32 cask stability is analyzed for three types of tornado missile impacts of 126 mph velocity, namely, Missile A - a 4,000 lb. automobile; Missile B - a 276 lb., 8-inch diameter armor piercing artillery shell; and Missile C - 1-inch diameter steel sphere. Based on the analyses, Missile A has the greatest effect on the stability of the TN-32 cask. It has the largest mass and produces the highest cask velocity after impact. The sliding analysis indicates that the TN-32 cask may slide 7.88 inches if Missile A strikes it below the cask center of gravity (CoG). This sliding distance is calculated using the coefficient of dynamic friction of 0.2625. The coefficient of dynamic friction is approximately 25% smaller than the coefficient of static friction and is used when the cask begins to slide. Since the calculated sliding distance of 7.88 inches is much less than the distance between the two casks (approximately 94 inches), this would not cause a collision between the casks. The analyses further indicate that the TN-32 cask will not tipover due to Missile A striking above the cask CoG, nor will there be any damage to the cask body. However, there could be localized damage to the neutron shield, protective cover, or overpressure monitoring system. Missiles B and C may partially penetrate the cask wall if the energy is not first dissipated by the outer shell and neutron shield. The protective cover absorbs all the impact energy, leaving the lid intact. However, the overpressure system could be rendered inoperable. The TN-32 cask will not tipover as a result of Missiles B and C.

3.4.5 Snow and Ice Loading

Due to the heat load of the cask contents, the temperature of the protective cover attached to the top of the cask above the lid will generally stay above freezing. The protective cover is a 0.38-in thick torispherical steel head which can withstand an external pressure of more than 20 psi. By comparison, a 50 psf (0.35 psi) snow or ice load corresponds to approximately 6 ft of snow or 1 ft of ice. This load is insignificant on the protective cover. Therefore, snow and ice loading has little structural consequence on the TN-32 cask.

3.5 Evaluation Findings

F3.1 SSCs important to safety are described in the TN-32 SAR, Revision 11A in sufficient detail to enable an evaluation of their structural effectiveness and are designed to accommodate the combined loads of normal, off-normal, accident, and natural phenomena events.

F3.2 The TN-32 storage system is designed to allow ready retrieval of spent nuclear fuel for further processing or disposal. The staff concludes that no accident or natural phenomena events analyzed will result in damage of the system that will prevent retrieval of the stored spent nuclear fuel.

F3.3 The cask is designed and fabricated so that the spent nuclear fuel is maintained in a subcritical condition under credible conditions. The configuration of the stored spent fuel is unchanged. Additional criticality evaluations are discussed in Section 6 of this SER.

F3.4 The cask and its systems important to safety are evaluated to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.5 The staff concludes that the structural design of the TN-32 dry storage cask 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 TN-32 cask 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 confirmatory analysis.

3.6 References

1. TN-32 Dry Cask Topical Safety Analysis Report, Rev. 9A, Transnuclear Inc., December 1996.

2. Safety Evaluation Report for the Transnuclear Inc., Dry Storage Cask (TN-32) Docket 72-1021 (M-56), November 1996.

3. TN-32 Dry Storage Cask Safety Analysis Report, Rev. 11A, Transnuclear Inc., January 1999.

4. NUREG-1536, Standard Review Plan for Dry Cask Storage System, January 1997.

5. ASME Boiler and Pressure Vessel Code, Section III, Division I, 1992.

6. Aluminum Standards and Data, Volume 1, The Aluminum Association, 1990.

7. ANSI N14.6-1993, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More, June 1993.

8. WRC-107, Local Stresses in Spherical and Cylindrical Shells Due to External Loadings, 1965.

9. ANSYS Engineering Analysis System, User's Manual for ANSYS Rev. 4.4, 1989

10. NUREG/CR-6608, "Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet onto Concrete Pads," February 1998.

11. LS-DYNA3D User's Manual (Nonlinear Dynamic Analysis of Structures in Three Dimensions), August 1995, Livermore Software Technology Corporation.

12. LLNL Report UCID-21246, "Dynamic Impact Effects on Spent Fuel Assemblies," October 1987.

4.0 THERMAL EVALUATION

The thermal review ensures that the cask and fuel material temperatures of the TN-32 cask will remain within the allowable values or criteria for normal, off-normal, and accident conditions. This objective includes confirmation that the temperatures of the fuel cladding (fission product barrier) will be maintained throughout the storage period to protect the cladding against degradation which could lead to gross rupture. This review also confirms that the thermal design of the cask has been evaluated using acceptable analytical and/or testing methods.

The basic TN-32 design was previously presented in Revision 9A of the TN-32 SAR[1] and approved by the NRC[2]. The current thermal analyses presented in Revision 11A of the TN-32 SAR include substantial changes from Revision 9A, with the exception of the fuel basket. As a result, information in SAR Revision 11A forms the basis for the staff's conclusions, except for the thermal behavior of the fuel basket, which relies on Reference 2.

4.1 Spent Fuel Cladding

The staff verified that the analyzed cladding temperatures for each fuel type proposed for storage are below temperatures which could cause cladding damage that would lead to gross rupture. For normal conditions of storage, the applicant calculated a limiting PWR fuel cladding temperature of 622°F (328°C). This limit is based on internal fuel rod pressure according to PNL-6189[3] and is acceptable to the staff. For the short-term accident and loading/unloading operations, the applicant used the temperature limit of 1058°F (570°C) from PNL-4835[4]. This limit is acceptable to the staff for short-term conditions.

In Section 4.6 of the SAR, the applicant considered the effect of cladding integrity during cask reflood operations that quench the hot spent fuel. The applicant provided a quench analysis of the fuel in SAR Section 3.5.2 that concluded the total stress on the cladding, as a result of the quenching process, is below the cladding material's minimum yield strength.

4.2 Cask System Thermal Design

4.2.1 Design Criteria

The design criteria for the TN-32 storage cask have been formulated by the applicant to assure that public health and safety will be protected during dry cask spent fuel storage. These design criteria cover both the normal storage conditions for the 20-year approval period and postulated accidents that last a short time, such as a fire.

Section 4.1 of the SAR defines several primary thermal design criteria for the TN-32 cask:

1. The allowable seal temperatures must be within specified limits to satisfy the leak tight confinement function during normal storage conditions.

2. Maintenance of the neutron shield resin during normal storage conditions; an allowable range of -40 to 300°F is set for the neutron shield.

3. Maximum and minimum temperatures of the confinement structural components must not adversely affect the confinement function.

4. The short-term allowable cladding temperatures that are applicable to off-normal and accident conditions of storage are based on PNL-4835.

5. The allowable fuel cladding temperatures to prevent cladding degradation during long-term dry storage conditions are provided in Section 3.5.1 of the SAR.

The staff concludes that the primary thermal design criteria have been sufficiently defined.

4.2.2 Design Features

To provide adequate heat removal capability, the applicant designed the TN-32 system with the following features:

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

2. Minimal heat transfer resistance through the basket by sandwiching aluminum and neutron absorber (poison) plates between the stainless steel fuel compartments. The compartments are plug-welded together forming paths for heat transfer from the fuel assemblies, along the plates, to the aluminum basket rails;

3. The basket rails are bolted to the steel containment providing a good conduction path to the cask cavity wall;

4. Aluminum boxes filled with a resin compound are placed around the cask gamma shell and enclosed by an outer shell. The boxes provide for neutron shielding and increase the thermal conductance through the neutron shield layer; and

5. High emissivity paint on the exterior cask surface to maximize radiative heat transfer to the environment.

The staff verified that all methods of heat transfer internal and external to the TN-32 are passive. Drawings in Section 1.5 of the SAR, along with the material properties in SAR Section 4.2, Tables a-h, provide sufficient detail for the staff to perform an in-depth evaluation of the thermal performance of the entire package as required by 10 CFR 72.24(c)(3)[5].

4.3 Thermal Load Specifications

The design-basis fuel to be stored in the TN-32 cask is described in Tables 2.1-1, 2.1-2, and 2.1-3 of the SAR for the PWR fuel. SAR Table 2.1-4 contains heat source data for the TPAs and BPRAs. The TN-32 cask is designed to dissipate 32.7 kW or 1.02 kW/assembly. The axial profiles for the design-basis fuels are in SAR Section 4.4.1. The axial peak power in the PWR assemblies is a factor of 1.2 times the average power. Maximum assembly heat load (fuel, TPA, and BPRA) is given in TS 2.1. By review and confirmation using independent analysis, the staff has reasonable assurance that design-basis decay heats were determined properly.

4.3.1 Storage Conditions

To bound the normal storage, off normal, and design-basis natural phenomena conditions, the applicant defined two external environments for storage conditions in Section 4.4 of the SAR. The maximum storage condition considers a 100°F average daily temperature and includes solar insolation equivalent to the total 10 CFR 71.71(c)[6] insolation averaged over a 24-hour period. The total 10 CFR 71.71(c) insolation in a 12-hour period is 2950 BTU/ft^2 and 1475 BTU/ft^2 for horizontal flat and curved surfaces, respectively. The minimum storage condition considers a -20°F average daily temperature and assumes no solar insolation. The staff concludes that the applicant's approach of using maximum and minimum daily average temperatures and insolation for the TN-32 cask is acceptable because cask temperature response to changes in the ambient conditions will be slow due to the large thermal inertia of the cask. Maximum and minimum average daily temperatures are included as siting parameters in the TS that must be evaluated by the cask user.

4.3.2 Accident Conditions - Fire

The fire accident postulated for the TN-32 storage cask is described in Section 4.5.1 of the SAR. The cask initial temperature distribution before the postulated accident is based on the maximum storage conditions.

A 15-minute fire with an average flame temperature of 1550°F, an average convective heat transfer coefficient of 4.5 Btu/hr-ft^2-°F, and an emissivity of 0.9 is hypothesized. This is postulated to be caused by the spillage and ignition of 200 gallons of combustible transporter fuel. The assumed 15-minute duration for the transient evaluation is based on a calculated fire duration of 13 minutes for this amount of fuel. Staff calculations of the fire duration agreed with the applicant.

Following the fire, the outside environment is restored to the maximum storage conditions and the TN-32 cask transient analysis is continued to evaluate temperature peaking of cask components. Based on review, the staff concludes that the thermal loads for the fire accident are acceptable.

4.3.3 Accident Conditions - Buried Cask

The buried cask accident postulated for the TN-32 is described in Section 4.5.2 of the SAR. The cask initial temperature distribution before the postulated accident is based on the maximum storage conditions. The TN-32 cask normally dissipates heat to the environment via radiation and convection. For this accident, the applicant assumed the burial media effectively insulated the cask outer surfaces. The analysis then determines the time to reach limiting temperatures for confinement integrity. Based on review, the staff concludes that the thermal loads for the cask burial accident are acceptable.

4.3.4 Cask Heatup During Loading

For cask heatup during vacuum drying, the cask has been drained of water and filled with air. Initial cask temperatures of 115°F, a building ambient temperature of 115°F, and a maximum allowable cask heat load of 32.7 kW were assumed. The heatup analysis assumed only conduction through air and neglected convection and radiation inside the cask. Based on review, the staff concludes that the thermal loads for cask heatup are acceptable.

4.4 Model Specification

4.4.1 Configuration

A three-dimensional (3-D) model for thermal design of the TN-32 system was developed using the finite element ANSYS[7] computer code. Transport of heat from the fuel assemblies to the outside environment is analyzed using a single large model of the TN-32 cask standing vertical on the concrete pad. The fuel region is modeled as a homogenized material with an effective thermal conductivity for the fuel. All other cask components are modeled in detail. Heat rejection from the outside cask surfaces to ambient air is considered by accounting for natural convection and thermal radiation heat transfer mechanisms from the vertical and top cover surfaces.

The staff reviewed the applicant's use of the ANSYS computer code and the associated inputs, assumptions, material properties, boundary conditions, and initial conditions. The staff has reasonable assurance that the temperatures of the cask components and the cask pressures under normal and accident conditions were determined correctly. Details of the modeling assumptions and approach follow.

4.4.1.1 Fuel Assembly Model

Heat transfer through the fuel assemblies was modeled by treating the fuel region as a homogenized material with effective thermal conductivities (k_{eff}) determined for the transverse and longitudinal directions. First, the applicant used the modified Wooton-Epstein correlation to calculate the k_{eff} of the various fuel assemblies designated for the TN-32 and determine the bounding PWR fuel type. The Westinghouse 17x17 Standard (W-17x17 Std) PWR fuel assembly yielded the highest cladding temperatures and was, therefore, selected as the bounding assembly for detailed analysis to define the fuel k_{eff}.

For the W-17x17 Std PWR fuel assembly, the longitudinal effective thermal conductivity was calculated based on the parallel paths of heat conduction through the cladding and the helium fill gas. Axial conduction through the fuel pellet was neglected. The transverse effective thermal conductivity was determined by using the ANSYS computer code to model a detailed two-dimensional (2-D) quarter symmetry section of the W-17x17 fuel assembly. A series of simulations with varying temperature boundary conditions was performed. The temperature drop across the assembly was then related to the k_{eff} of the fuel. A resultant relationship of k_{eff} of the fuel versus average temperature of the assembly was developed. The effective specific heat and density for the homogenized fuel assemblies were determined using a mass weighted average approach.

4.4.1.2 TN-32 Basket Section Model

The heat rejection capability of the TN-32 design was evaluated by developing a thermal model of the homogenized fuel assemblies, the basket wall geometry, and the layers that form the cask body. The ANSYS model includes the geometry and materials of the basket, the basket rails (peripheral inserts), the cask shells, the neutron shielding (resin in the aluminum containers/top neutron shield), the outer shell, the lid, and the concrete pad.

A detailed full length, quarter slice section of the TN-32 cask was modeled with the appropriate symmetry boundary conditions. The model is shown in SAR Figures 4.4-1, 4.4-2 and 4.4-3. The decay heat from the fuel assemblies was applied to the homogenized fuel elements as volumetric heat generation in the 144-inch active fuel length.

The model includes 8 of the 32 basket stainless steel boxes with one 0.5-inch thick aluminum and one 0.04 -inch thick poison plate placed between the 0.10-inch thick boxes. The boxes are held together by plug welds which pass through the aluminum and poison plates. The thermal model accounts for heat transfer through the aluminum plates and the stainless steel boxes. The thermal conductivity of the aluminum plates was reduced by 10% to account for the effects of the plug welds. No credit was taken for heat transfer through the poison plates. Some of the aluminum plates are interrupted to allow other plates a direct conduction path to the basket periphery. A nominal gap of 0.02 -inches was assumed between the interrupted and continuous basket plates.

The basket bottom and the fuel assemblies rest on the cask bottom during normal storage conditions. However, in lieu of direct contact with the cask bottom, a 0.25-inch gap is assumed between the basket bottom and the cask bottom. Basket rails, bolted to the cask cavity wall, provide both structural support and an increased surface area for heat transfer from the basket plates. A 0.188 -inch gap is assumed between the basket and the basket rails. A 0.01-inch gap between the rails and the cask cavity wall is assumed. Only conduction through the helium gas is modeled across this gap.

4.4.1.3 Cask Body Model

From the inner cavity wall to the exterior cask surface, heat is conducted though an array of concentric layers representing the containment shell, the gamma shield shell, the resin filled boxes that form the neutron shield shell, and the outer shell. Heat is also transferred through the lid, the neutron shield in the lid, and through the cask weather protective cover. Heat rejection from the cask exterior surfaces to ambient air includes natural convection and thermal radiation heat transfer from the vertical and top cover surfaces. Heat transfer modeled through the cask bottom includes a containment plate, a 0.125-inch gap, and a bottom gamma shield. The bottom gamma shield is assumed to be in perfect thermal contact with a 36-inch thick slab of concrete which extends 36 -inches around the bottom of the cask. The bottom of the concrete pad is assumed to be in perfect contact with soil, at a constant temperature of 70°F.

4.4.1.4 Radiation from Cask Exterior Surfaces

The applicant considered the thermal radiation interaction among casks in an array. The radiation from the cask was lowered to account for an array of casks. A two-cask wide and infinitely long array was assumed. This assumption resulted in an overall view factor of 0.77 between the cask vertical surfaces and the ambient environment. To ensure the assumed radiative configuration is maintained during storage, a minimum center-to-center cask spacing of 16 ft is included in TS 4.2.

4.4.2 Material Properties

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

4.4.3 Boundary Conditions

The boundary conditions include the total decay heat and the external conditions on the cask surface. The axial peak power for the PWR assemblies is a factor of 1.2 times the average power of 1.02 kW/assembly. The cask external boundary conditions depend on the environment surrounding the cask and are detailed below.

4.4.3.1 Storage Conditions

For storage conditions, the applicant included boundary conditions for ambient temperatures and insolation as described in SER Section 4.3.1. In addition, the bottom of the concrete pad beneath the cask is assumed to be in perfect contact with soil, at a constant temperature of 70°F.

4.4.3.2 Accident Conditions - Fire

For the postulated fire accident conditions, the finite element model described in SAR Section 4.4.1.1 for the storage condition was modified by reducing the geometry into a cross-section model located at the hottest axial location in the cask. In addition, a lid seal region was modeled to examine temperatures at the cask seals. These models are shown in Figures 4.5-1, 4.5-2, and 4.5-3 of the SAR. The boundary conditions include the cask initial temperature distribution before the postulated accident, fire conditions for the fire transient, and ambient temperatures and insolation post-fire as described in SER Section 4.3.2.

4.4.3.3 Accident Conditions – Buried Cask

For the postulated buried cask accident conditions, the finite element model described in SAR Section 4.4.1.1 for the storage condition was modified by reducing the geometry into a cross-section model located at the hottest axial location in the cask. The boundary conditions include the cask initial temperature distribution before the postulated accident and an essentially insulated condition for the transient as described in SER Section 4.3.3.

4.4.3.4 Cask Heatup Analysis

For the cask heatup analysis, the finite element model described in SAR Section 4.4.1.1 for the storage condition was modified by reducing the geometry into a cross-section model located at the hottest axial location in the cask. The boundary conditions include the initial cask and ambient temperatures and the cask environment during the transient as described in SER Section 4.3.4.

4.5 Thermal Analysis

4.5.1 Computer Programs

The thermal analysis was performed using the ANSYS finite element modeling package. ANSYS is capable of general 3-D steady-state and transient calculations. The output from the code work was plotted in SAR Figures 4.4-6 through 4.4-10 and discussed in SAR Section 4.4.

4.5.2 Temperature Calculations

4.5.2.1 Storage Conditions

The TN-32 cask has been analyzed to determine the temperature distribution under long-term storage conditions that envelop normal, off-normal, and design-basis natural phenomena conditions. The basket is considered to be loaded at design-basis maximum heat loads with PWR assemblies. The systems are considered to be arranged in an ISFSI array and subjected to design-basis ambient conditions with insolation. The maximum allowable temperatures of the components important to safety are listed in Table 4.1-1 of the SAR. Low temperature conditions were also considered.

The maximum fuel clad temperatures for zircaloy-clad fuel assemblies are listed in SAR Table 4.4-1. Temperature criteria for the spent fuel cladding are discussed in Section 4.2 of the SER. The fuel cladding temperatures remain below their acceptable temperatures. SER Table 4-1, below, summarizes the temperatures of key components in the cask for various environmental conditions.

Table 4-1 Temperatures of Key Components							
Component	Storage Conditions			Fire Accident		Burial Accident	
	Maximum (°F)	Minimum* (°F)	Allowable Range (°F)	Peak (°F)	Allowable Range (°F)	Time to limit (hours)	Allowable Range (°F)
Outer Shell	240	-20	**	945	**	**	**
Lid (Standard/Type A)	263	-20	**	438	**	**	**
Seal	256	-20	-40 to 536	380	-40 to 536	38	-40 to 536
Top Neutron Shield	256	-20	-40 to 300	N/A	N/A	3	N/A
Radial Neutron Shield	280	-20	-40 to 300	N/A	N/A	3	N/A
Inner Shell	308	-20	**	375	**	**	**
Gamma Shield Shell	303	-20	**	370	**	**	**
Inner Bottom Plate	314	-20	**	***	**	**	**
Outer Bottom Plate	255	-20	**	***	**	**	**
Basket Rail	339	-20	**	398	**	**	**
Basket Plate	527	-20	**	610	**	**	**
Fuel Cladding	565	-20	622 max.	647	1058 max.	93	1058 max.

* Assuming no credit for decay heat and a daily average ambient temperature of -20°F.
** The components perform their intended safety function within the operating range.
*** Not modeled.

4.5.2.2 Accident Conditions - Fire

The applicant analyzed a fire accident on the TN-32 cask using the conditions specified earlier in Section 4.4.3.2 of the SER. The peak temperatures of the key cask components due to a 15-minute fire with a 32.7 kW decay heat are shown in SER Table 4-1. The initial temperatures are based on the maximum storage conditions. All of the fire accident temperatures were below the short-term design-basis temperatures with the exception of the neutron shield material. However, as discussed in SER Sections 5 and 11, the accident condition dose rate limits are shown to remain below the regulatory limit of a total dose of 5 rem assuming complete removal of the neutron shield. Based on these analyses, the staff has reasonable assurance that the cladding integrity and the confinement boundary will not be compromised during the fire or post-fire transient.

4.5.2.3 Accident Conditions – Buried Cask

The results for this accident are summarized in SER Table 4-1. The neutron shield temperature limit is reached at 3 hours, the seal limit at 38 hours, and the cladding limit at 93 hours. Based on review, the staff concludes that the thermal analysis of an adiabatic heatup resulting from cask burial is acceptable. As discussed in SAR Section 11.2.10.4, corrective actions to uncover the cask will need to be taken as soon as possible to protect the seal and cladding integrity.

4.5.2.4 Cask Heatup Analysis

The applicant performed a transient analysis of the cask heatup prior to being filled with helium using the conditions specified earlier in Section 4.4.3.4 of the SER. A maximum cladding steady-state temperature of 935°F was predicted. Consequently, the duration of the cask drying evolution is not constrained by the fuel cladding temperature (short-term limit is 1058°F). However, to maintain cask component temperatures below the temperatures of the fire transient, a maximum time of 36 hours is allowed for design-basis heat load fuel.

TS 3.1.1 and 3.1.2 are written to ensure that the vacuum drying and helium backfill operations are completed within 36 hours or corrective action such as injecting a partial pressure of helium in the cask is required. The applicant concluded that a helium partial pressure of 0.1 atm (the remaining 0.9 atm is assumed to be air) would sufficiently improve heat transfer to maintain temperatures below limits. The staff calculated the effective gas mixture conductivity of the air-helium mixture and determined that the net conductivity was about 25% that of pure helium. Data from Interim Staff Guidance (ISG) 7[8]documents the sensitivity of a different storage cask system to a net gas mixture conductivity of 30% that of pure helium. That data shows that fuel cladding and bulk gas temperatures would increase about 3%. This condition is bounded by the fire transient analysis, which evaluates cask cavity temperatures greater than 10% above those temperatures calculated for steady state during maximum storage conditions. Based on the above, the staff concludes that there is reasonable assurance that the cask component and fuel cladding temperatures will be maintained within limits if a 0.1 atm partial pressure of helium is maintained in the cask for short-term operations.

4.5.3 Pressure Analysis

4.5.3.1 Storage/Off Normal/Accident Conditions

In SAR Sections 7.2.2 and 7.3.2.2, the applicant evaluated internal pressurization for the following conditions:

1) 100°F ambient air temperature and insolation (maximum storage conditions)

2) maximum storage conditions, 10% fuel rod failure, and 10% BPRA failure (off-normal)

3) maximum storage conditions, 100% fuel rod failure, and 100% BPRA failure (accident)

4) maximum storage conditions, 100% fuel rod failure, 100% BPRA failure, and 15-minute external fire (fire accident)

5) 100°F ambient air temperature, 100% fuel rod failure, 100% BPRA failure, and cask burial under debris (burial accident)

The staff reviewed the applicant's calculations and performed confirmatory analyses. The applicant's calculations used appropriate methods and cover gas temperatures determined in SAR Section 4. The highest predicted pressure was 99 psig at a cavity gas temperature of 644°F for the cask burial accident. Staff calculations were in agreement with the applicant's results and confirmed that the expected pressures were below the design internal pressure of 100 psig. Based on review and confirmatory analyses, the staff concludes that internal cask pressures remain below the cask design pressure rating under normal, off-normal, design-basis natural phenomena, and design-basis accident conditions or events.

4.5.3.2 Pressure During Unloading of Cask

In SAR Section 4.6.1, the applicant considered the transient resulting from reflooding the cask with water prior to placement of the cask in the spent fuel pool for fuel unloading. To control cask pressurization, the maximum initial cask reflood rate is controlled to maintain internal pressure below 90 psia (design pressure 100 psig). A TN calculation[9] showed that the maximum saturated steam flow rate from the cask was 0.144 lbm/sec at 90 psia assuming the cask vent discharged into the spent fuel pool as shown in SAR Figure 8.2-1. To limit the water supply for steam production to less than the predicted maximum discharge mass flow rate, operating controls in SAR Section 8.2 reflect a maximum initial inlet flow rate of 0.140 lbm/sec (1.0 gallon per minute). The applicant further ensures overpressure protection of the cask during reflood by placing a check valve in the water inlet line that will restrict cooling water flow if inlet pressure reaches 90 psia. Based on staff review of the reflood evaluation and the proposed operational controls, the staff has reasonable assurance that the cask can be maintained within design pressure limits during reflood.

4.5.3.3 Pressure During Loading of Cask

Although the cask is vented during the draining procedure, the applicant evaluated cask pressurization during loading operations when the cask is filled with borated water and removed from the spent fuel pool. A bounding case, where the maximum allowable cask heat load of 32.7 kW is applied to the boiling of the cask water, was analyzed to determine if the cask design pressure of 100 psig would be reached. The applicant determined that a rate of evaporation of 0.036 lbm/s is much less than the flow rate of 0.144 lbm/s that was calculated for a cask pressure of 90 psia. The staff performed a confirmatory calculation of the evaporation rate, found it to be in agreement with the applicant, and, therefore concludes that there is reasonable assurance that the cask pressure will remain below design pressure during loading operations.

4.5.4 Confirmatory Analyses

The confirmatory analyses of the TN-32 storage cask SAR can be divided into six categories: (1) review of models used in the analyses, (2) review of material properties used in the analyses, (3) review of boundary conditions and assumptions, (4) perform independent analyses to confirm the applicant's analyses, (5) compare the results of the analyses with the applicant's design criteria, and (6) assure that the applicant's design criteria will satisfy the regulatory acceptance criteria and regulatory requirements.

The staff reviewed the approaches used by the applicant in the thermal analyses. The staff performed a confirmatory analysis of the thermal performance of the cask SSCs identified as important to safety. A detailed model of a single fuel compartment was developed using the COBRA-SFS[10] computer code to evaluate the SAR results. The temperature distributions generated by the staff's model displayed good agreement with those values determined by the applicant.

The staff generated a simplified steady-state model of the lid seal region using the HEATING-7[11] computer code. The temperature distributions compared reasonably well with those predicted by the applicant and all values were within design limits.

The staff performed a sensitivity study on the impact of gap sizes in the cask body layers on the thermal performance of the design. The applicant assumed 0.01-inch gaps between the aluminum neutron shield boxes and the adjacent gamma shield shell and surrounding outer shell. Using a linear scale up, the staff projected a peak cladding temperature higher than the long-term storage cladding temperature limit of 622°F (328°C) if fabrication results in gaps of 0.04 -inches or greater between the component layers. To demonstrate that fabrication methods can control gaps within those assumed in the thermal analysis, the applicant will perform a one-time thermal performance test of a cask body with up to 32.7 kW of decay heat. Details of the thermal performance test are provided in Section 9.1.6 of the SER. Performance of this test is included in the TN-32 CoC.

The staff has determined that the thermal SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness. Based on the applicant's analyses, there is reasonable assurance that the TN-32 cask is designed with a heat removal capability having testability and reliability consistent with its importance to safety. The staff further concludes, based on review and confirmatory analysis, that there is reasonable assurance that analysis of the TN-32 cask demonstrates that the applicable design and acceptance criteria have been satisfied.

4.6 Evaluation Findings

F4.1 SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness.

F4.2 The staff has reasonable assurance that the decay heat loads were determined appropriately and accurately reflect the burnup, cooling times, and initial enrichments specified.

F4.3 The staff has reasonable assurance that the temperatures of the cask SSCs important to safety will remain within their operating temperature ranges and that cask pressures under normal and accident conditions were determined correctly.

F4.4 The staff has reasonable assurance that the TN-32 cask is designed with a heat removal capability having testability and reliability consistent with its importance to safety.

F4.5 The staff has reasonable assurance that the TN-32 cask provides adequate heat removal capacity without active cooling systems.

F4.6 The staff has reasonable assurance that the spent fuel cladding will be protected against degradation that leads to gross ruptures by maintaining the clad temperature below maximum allowable limits and by providing an inert environment in the cask cavity.

F4.7 The staff concludes that the thermal design of the TN-32 is in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the TN-32 will allow safe 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.

4.7 References

1. TN-32 Dry Storage Cask Safety Analysis Report, Rev. 9A, Transnuclear, Inc., December 1996.

2. Safety Evaluation Report for the Transnuclear Inc. Dry Storage Cask (TN-32) Docket 72-1021 (M-56), November 1996.

3. Levy, I.S., et al., Pacific Northwest Laboratories, "Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircaloy Clad Fuel Rods in Inert Gas," PNL-6189, May 1987.

4. Johnson, A.B., and E.R. Gilbert, Pacific Northwest Laboratories, "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases," PNL-4835, September 1983.

5. 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.

6. U.S. Code of Federal Regulations, "Packaging and Transportation of Radioactive Material," Title 10 Part 71.

7. ANSYS Engineering Analysis System, "Users Manual for ANSYS, Revision 5.4," ANSYS Inc., Houston, PA.

8. Kane, W.F., "Spent Fuel Project Office Interim Staff Guidance 7," ISG-7. U. S. Nuclear Regulatory Commission, October 1998.

9. Transnuclear Inc., "Mass Flow Rates During Unloading," Calculation No. 1066-70, Revision 0, January 1999.

10. Michener, T.E., et. al., Pacific Northwest Laboratories, "COBRA-SFS: A Thermal-Hydraulic Analysis Code fore Spent Fuel Storage and Transportation Casks," PNL-10782, September 1985.

11. Childs, K.W., Oak Ridge National Laboratories, "Heating 7.2 Users Manual," NUREG/CR-0200, Volume 2, Revision 5, March 1997.

5.0 SHIELDING EVALUATION

The shielding review evaluates the capability of the TN-32 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 TN-32 cask is designed to provide both photon and neutron shielding. There are three versions of the cask; designated as TN-32, TN-32A, and TN-32B. These differ at the top of the cask and the trunnions but the effective shielding of all three models is virtually the same. The principal components of the radial photon shielding are the 1.5-inch thick steel inner shell, the 8.0-inch thick steel body wall, and a 0.5-inch thick steel outer shell. Photon shielding at the bottom of the cask is provided by the 1.5-inch thick steel base of the confinement vessel and a 8.75-inch thick steel bottom plate. The photon shielding at the top of the cask consists of 10.5 inches of steel in the lid for the TN-32 and TN-32B casks. The photon shielding at the top of the TN-32A cask consists of 9.38 inches of steel in the lid plus a supplemental 1.25-inch thick steel plate incorporated with the neutron shield. A 1.12-inch reduction in the lid thickness for the TN-32A cask allows for inclusion of BPRAs and TPAs with upper head injection cups.

In addition to the steel components discussed above, radial neutron shielding is provided by a borated polyester resin compound cast into long slender aluminum containers that surround the cask body. The thickness of the radial neutron shield material is 4.5 -inches. The top neutron shield material is 4.0 -inches of polypropylene encased in a 0.25-inch thick steel shell.

5.1.2 Shielding Design Criteria

The overall design criteria for the TN-32 cask are the regulatory dose limits and requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

The staff evaluated the TN-32 shielding design features and design criteria and found them to be acceptable. The SAR analysis indicates 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 rates are to meet specific limits as described in TS 5.2.3.

An evaluation of the overall radiation protection design features and design criteria of the TN-32 cask is given in Section 10 of the SER.

5.2 Radiation Source Definition

5.2.1 Source Specification

The radiation source specification is presented in Section 5.2 of the SAR. Photon and neutron source terms were generated with the SAS2H and ORIGEN-S modules of SCALE 4.3, using the 45-group coupled neutron-photon cross-section library[3]. The source terms for the fuel assembly designs specified in SAR Table 2.1-1 were examined to determine which assembly gives the maximum source terms. The examination included the photon and neutron sources from the fuel, the photon sources from activated assembly hardware, and an estimate of the dose rates on the cask side at the surface and at 1 meter. The applicant determined that the W-17x17 standard assembly resulted in the maximum source terms and total dose rates. Consequently, design-basis source terms were calculated for Zircaloy clad fuel with Inconel-718 grid spacers in the standard W-17x17 fuel assembly.

Cobalt impurities in Inconel, Zircaloy, and stainless steel were assumed to be 4700, 10, and 800 ppm, respectively. These values were obtained from Reference 4. Measured cobalt impurities in Inconel grid spacers from a W-14x14 assembly range from 890 to 1490 ppm[5]. Another set of measurements resulted in a range of cobalt impurities from 186 to 3600 ppm[6]. The value of 4700 ppm used to estimate the Co-60 source term for the grid spacers bounds these measured values.

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 5. These scaling factors produce calculated source terms which bound measured source terms. The staff performed confirmatory calculations with ORIGEN2[7] and obtained Co-60 source terms which are 15% lower than the values in the SAR. The neutron flux scaling factors from Reference 5 are derived from measurements and are considered to provide bounding values, particularly in relationship to the values calculated in Reference 4.

The characteristics of the Westinghouse fuel assemblies are given in SAR Table 2.1-1. The design-basis 17x17 standard fuel assembly gives the largest source terms and bounding dose rates as shown in SAR Table 5.2-3. A minimum U-235 enrichment of 3.5 wt% at a burnup of 45,000 MWD/MTU and a cooling time of 7 years maximizes the radiation source terms. The source terms for this fuel are listed in SAR Tables 5.2-4, 5.2-8, and 5.2-12. The source terms for the fuel region include photon radiation from activated grid spacers. The limits on burnup and cooling time for BPRAs and TPAs are provided in SAR Section 2 and are given in Figures 2.1-4 and 2.1-5, respectively. Source terms for BPRAs and TPAs are given in Table 5.2-5 of the SAR.

The staff performed confirmatory analyses of the design-basis photon and neutron source terms for the fuel, BPRAs, and TPAs using the ORIGEN2 code. The staff has examined the Department of Energy (DOE) Characteristics Data Base[8] and determined that PWR fuel burned to 45,000 MWD/MTU could have an enrichment as low as 3.3 wt% which would increase the neutron source by 12%. To preclude fuel of this type with a 7-year cooling time from being loaded into the TN-32 cask, a TS in the form of Table 2.1.1-1 is included. Fuel of lower enrichment may be stored in the TN-32 cask provided the burnup is lower and/or the cooling time is increased according to TS Table 2.1.1-1. The staff has reasonable assurance that the design-basis photon source terms are adequate for the shielding analysis.

The applicant calculated a neutron source term assuming all of the fuel was irradiated to the design-basis exposure. Since the neutron source increases exponentially with burnup, neutron peaking factors based on an axial burnup profile were calculated and are listed in SAR Table 5.2-11. The integration of the neutron source as a function of axial position resulted in a 28% larger total neutron source than that given in Table 5.2-3 of the SAR for the average burnup value. Staff calculated a neutron source term that was 9% higher than the applicant's final value. The difference is offset by the applicant's higher photon source term and does not change the conclusions about the cask meeting safety requirements.

The axial burnup profile is given in SAR Table 5.2-11 and shown graphically in Figure 5.2-1. The computational models divided this profile into 7 burnup zones in the top half of the fuel and 10 burnup zones in the bottom half of the fuel.

5.3 Shielding Model Specifications

5.3.1 Model Specifications

The model specifications for shielding are presented in Section 5.3 of the SAR. The applicant's shielding model for normal and accident conditions consists of a 3-D representation of the TN-32 cask using the design drawings in Section 1.5 of the SAR. A description of the shielding configuration is presented in Section 5.3.1 of the SAR. Radial views of the shielding model are depicted in SAR Figures 5.3-3, 5.3-4, and 5.3-7. Axial views of the shielding model are depicted in SAR Figures 5.3-1, 5.3-2, 5.3-5, 5.3-6, 5.3-9, 5.3-10, and 5.3-11.

5.3.1.1 Source Configuration

The radiation source is divided into four axial regions: bottom end fitting, fuel, plenum, and top end fitting. The relative positions of these source term regions are also depicted in the axial view figures identified above. The fuel region is modeled as a totally homogeneous zone, and the end fittings and plenum regions are modeled as homogeneous regions of stainless steel, Inconel, and Zircaloy. The axial distribution of the photon source in the fuel region was developed from actual utility operator data. The photon profile is listed in SAR Table 5.2-11. The axial distribution of the neutron source was determined from a series of SAS2H/ORIGEN-S calculations using the axial burnup profile in Figure 5.2-1 of the SAR. The neutron profile is listed in SAR Table 5.2-11. This profile was used to account for the non-linear buildup of neutron source terms (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.

5.3.1.2 Streaming Paths and Regional Densities

The shielding models included streaming paths for the trunnions. The cask design eliminates other potential streaming paths. Confirmatory calculations show that dose rates at the trunnions are less than dose rates above and below the radial neutron shield.

The composition and densities of the materials used in the shielding analysis are presented in SAR Tables 5.3-1, 5.3-2, and 5.3-3. The applicant did not identify any materials that undergo changes in material density or composition from temperature variations. The bounding accident condition for shielding assumes complete loss of both the radial neutron shield and the axial neutron shield.

The staff evaluated the SAR shielding models and found them to be acceptable. The basket and fuel inserts were homogenized with the fuel rods and assembly hardware for the radial calculations but not for the axial calculations. The model dimensions and material specifications are consistent with the drawings in Section 1 of the SAR and provide the basis for reasonable assurance that the TN-32 cask was adequately modeled in the shielding analysis.

The staff notes that the aluminum tubes containing the neutron shield material were homogenized with the neutron shield. Staff concludes that since the aluminum tubes have a wall thickness of only 1/8 -inch and actual measurements have not detected streaming, neutron streaming through the aluminum is not significant.

Appendix 5A of the SAR provides measured dose rates for previously licensed TN-32 casks that have been loaded with PWR spent fuel. Source terms for the loaded casks are lower than the source terms for this license application. Since it was provided for informational purposes only and is not significant to the application, Appendix 5A was not reviewed.

5.4 Shielding Analyses

5.4.1 Shielding Analyses

The shielding analyses are presented in Section 5.4 of the SAR. The 3-D Monte Carlo transport code, SAS4[9], was used for the cask surface and 1 meter shielding analysis. The cross-section library used has 27 neutron groups and 18 photon groups and is based on ENDF/B-V cross-section data. The Monte Carlo N-Particle (MCNP) code[10] was used to calculate direct and skyshine dose rates at large distances. The SAR uses the ANSI/ANS Standard 6.1.1-1977 flux-to-dose-rate conversion factors to calculate dose rates in the shielding analysis.

5.4.1.1 Normal Conditions

The SAR presents calculations for normal condition dose rates of the design-basis fuel without inserts for the TN-32 cask. Calculated dose rates listed in SAR Table 5.1-2 are averages over the specified surface area or 1 meter from the surface. The higher dose rates on the side of the cask occur above and below the radial neutron shield. The average values for these regions without inserts are 337 mrem/hr and 283 mrem/hr, respectively. The SAR also presents incremental dose rates for the fuel inserts (BPRAs and TPAs) in Table 5.1-2.

Confirmatory calculations for the TN-32 cask were made with the MCNP code. The MCNP model included BPRAs which were homogenized with the basket, fuel rods, and assembly hardware in the fuel, plenum, and top end fitting regions. The MCNP model is similar to the SAS4 model used by the applicant and is based on the drawings in Section 1.5 of Rev. 11A of the SAR. A comparison between the applicant's results and the confirmatory calculations showed a variation in the results which is expected when two different codes are used for shielding calculations. The applicant's dose rates 1 meter from the cask are in good agreement with the confirmatory calculations. Overall, the differences between the applicant's and confirmatory results fell within acceptable bounds. A TS has been included which specifies a maximum allowable dose rate on the surface of the cask at specified locations based on the applicant's calculations.

Staff used the MCNP code to evaluate the effect of homogenizing the Inconel grid spacer with the fuel. Using an impurity of 4700 ppm cobalt in the grid spacers, it was determined that the variation in the Co-60 photon dose rate on the side of the cask from discretely modeled grid spacers was much less than the maximum dose rate on the cask surface. The procedure of homogenizing the grid spacers with the fuel and basket was determined to be acceptable.

5.4.1.2 Accident Conditions

Table 5.1-2 of the SAR also contains results of calculations for accident-condition dose rates of the design-basis fuel on the cask side, 1 meter from the cask side, the top of the cask, and 1 meter from the top of the cask. Maximum dose rates on the surface and at 1 meter are approximately 1690 mrem/hr and 623 mrem/hr, respectively. These dose rates are without fuel inserts. MCNP code confirmatory calculations for the accident condition were made with fuel inserts. The confirmatory dose rate on the side of the cask is 1900 mrem/hr. The additional photon source from the BPRAs accounts for the increased dose rate. The accident condition dose rates given in SAR Table 5.1-2 are acceptable.

5.4.1.3 Occupational Exposures

Design-basis fuel at 45,000 MWD/MTU burnup and 7-year cooling time with design-basis BPRAs, was used to estimate occupational exposures during cask operations. Section 10 of the SAR presents estimated occupational exposures using the calculated dose rates for the locations shown in Figure 5.1-2.

5.4.1.4 Off-Site Dose Calculations

Direct-path off-site dose rates are presented in Table 5.1-3 of the SAR for a single cask, but do not include the BPRA source term. Direct-path dose rates for off-site locations assumed a design-basis fuel loading, level topography, and a 100% occupation time. MCNP code confirmatory calculations of the direct dose rates include the BPRA source term. The applicant's dose rate at 250 meters is very close to the confirmatory dose rate. Beyond 250 meters, the applicant's dose rates are higher than the confirmatory dose rates but are determined to be within acceptable bounds. Confirmatory skyshine dose rates when a berm or shield is present also were calculated with MCNP code and are discussed in SER Section 10.

Section 10 of the SER evaluates the overall off-site dose rates from the TN-32 cask. The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved. The general licensee using the TN-32 cask 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 TN-32 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 TN-32 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 TN-32 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. Petrie, L.M., et al., "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200, Vol. 1-4, Revision 4, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1995.

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

5. 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.

6. 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.

7. Croff, A.G., "ORIGEN2: A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code," ORNL/TM-5621, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1980.

8. TRW Environmental Safety Systems, Inc., "DOE Characteristics Data Base, User Manual for the CDB_R," November 16, 1992.

9. 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, Oak Ridge, Tennessee, 1997.

10. Briesmeister, J.F., ed., "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4B," LA-12625-M, Los Alamos National Laboratory, Los Alamos, New Mexico, 1997.

6.0 CRITICALITY EVALUATION

The staff reviewed the TN-32 cask 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 TN-32 cask 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 11A of the SAR was also reviewed to determine whether the TN-32 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, and
- more than 75% for fixed neutron absorbers when subject to standard acceptance tests. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are needed.

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 TN-32 cask design features relied upon to prevent criticality during cask loading and unloading are the basket geometry, fixed neutron poisons in the basket, and required soluble boron concentration in the spent fuel pool water. For the basket, TS 4.1.1 requires a minimum basket fuel cell opening of 8.64 inches square and a minimum Boron-10 areal density of 10 mg/cm^2. The applicant took credit for 90% of the minimum specified Boron-10 areal density in the basket poison material in all calculations. The fabrication requirements and acceptance criteria for the fixed neutron poison, which justify the use of 90% credit of basket's fixed neutron poison, are outlined in SAR Section 9.1.7. For the spent fuel pool, TS 3.3.1 requires the boron concentration in the pool to be at least 2300 ppm. The TS surveillance requires two independent measurements of the boron concentration prior to cask loading or unloading which meets the requirements of 10 CFR 72.124(a). During storage, the TN-32 cask is designed to prevent water from entering the cask cavity, which maintains subcriticality.

The staff reviewed the TN-32 cask design criteria and features discussed in Sections 1.2, 2.3.4, and 6 of Revision 11A of the 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 postulated 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 TN-32 cask design meets the double contingency requirements of 10 CFR 72.124(a).

6.2 Fuel Specification

The TN-32 dry storage cask is designed to store a maximum of 32 intact Westinghouse PWR spent fuel assemblies. The assembly types are limited to 14x14 standard ZCA and ZCB or OFA, 15x15, or 17x17 standard or OFA spent fuel assemblies. The fuel assemblies are described in Sections 2.1 and 6.2 of the SAR and the fuel characteristic limits are given in TS 2.1. The maximum uranium mass, listed in TS 2.1, conservatively bounds the allowed fuel assemblies. The maximum initial enrichment is 4.05 wt% U-235 and the maximum assembly average burnup is 45,000 MWD/MTU. The fuel may contain BPRAs or TPAs. The applicant performed calculations which verify that criticality safety is maintained for each of these fuel types in the TN-32 cask with and without BPRAs. Fuel containing TPAs may also be loaded into the cask. The analysis of fuel with TPAs has not been performed as it is bounded by the analysis of fuel containing BPRAs.

Specifications on the fuel condition are also included in Section 6.2 of the SAR and TS 2.1. Fuel with structural defects greater than pinhole leaks and hairline cracks may not be loaded into the TN-32 cask. Fuel bundles with missing pins are not allowed unless the missing pin is replaced by a fuel pin or dummy pin that displaces an equivalent volume.

In Appendix 6A of the SAR, the applicant has shown that the increase in fuel assembly burnup to 45,000 MWD/MTU will not result in fuel cladding failure during the cask drop accidents, which bounds all storage conditions. In Section 3 of this SER, the staff reviewed these analyses and agree that the criticality models need only consider intact fuel pins.

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

6.3 Model Specification

6.3.1 Configuration

A single TN-32 cask with full water reflection was modeled in all cases. The fuel assembly hardware above and below the active fuel region was modeled as unborated water and borated water was used above the fuel assembly hardware in the cask. The active fuel region was modeled explicitly. The applicant did not take credit for the burnup of the fuel. The aluminum spacers between the fuel basket and the cask wall were modeled as a homogeneous mixture of 2000 ppm borated water and aluminum as was done in the previously approved TSAR Revision 9A (i.e., the boron content was not increased). No credit was taken for burnable absorbers, instead, the BPRAs were modeled as pure aluminum filling the guide tube.

A number of parametric cases were analyzed to determine the most reactive model for normal conditions. First, the reactivities of 32 14x14 standard, 14x14 OFA, 15x15, 17x17 standard, and 17x17 OFA assemblies in the cask, with and without BPRAs, were compared. The 17x17 standard with BPRAs was found to be most reactive and thus used for the remainder of the calculations. The assemblies were shifted toward the center of the cask, as this was found to be more reactive than the case with the assemblies centered in the basket compartments.

The tolerances on the compartment size and the thickness of the neutron poison plate were also varied. As expected, the case with the minimum values on each of these dimensions was found to be the most reactive, because this increases the interaction between the assemblies in the basket. A 2-inch downward shift of the fuel, which results in an offset of the fuel below the neutron absorber plate, also resulted in a slight increase in reactivity.

The density of the borated water in the cask was also varied. Borated water at 95% of theoretical density was slightly more reactive than full density borated water. The interior of the TN-32 cask does not allow for preferential or uneven flooding of the cask. Unborated water was modeled in the fuel rod cladding-pellet annulus, which also increased k_{eff}.

The scenario where the cask is partially filled with borated water and partially filled with steam was not analyzed, as this will not increase reactivity in this cask design. However, the borated water above the fuel assembly hardware was removed to reduce the absorption of neutrons in the reflector. The reactivity was statistically unchanged.

The normal condition model combined the most reactive conditions from the parametric studies. Thus, the normal condition model contains 17x17 standard assemblies with BPRAs off-center in the basket, with reduced basket compartment sizes, reduced neutron poison thickness, and fuel shifted 2 inches downward below the poison plates. The pellet-clad annulus of the pins contains fresh water and the cask contains 2300 ppm borated water at 95% of theoretical density. The borated water above the fuel assemblies was removed.

The accident condition model substituted a single 5 wt% enriched fuel assembly for one of the central 4.05 wt% assemblies in the normal condition model to represent a misloading accident.

The staff reviewed the applicant's models and agrees that they are consistent with the description of the cask and contents given in Sections 1 and 2, including engineering drawings.

The staff also reviewed the applicant's methods, calculations, and results for determining the worst-case manufacturing tolerance. Based on the information presented 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 Nos. 1049-70-5, Revision 4 and 1049-70-6, Revision 6. The staff's fuel assembly models were based on the fuel assembly parameters given in Section 6 of the SAR, TS 2.1, and "Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation" [3]. The uranium masses and enrichment were taken from Revision 11A of the SAR as these are the values used in the TS. The staff's results were comparable 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 the SAR. The minimum required areal density of the Boron-10 (^{10}B) in the fixed neutron poison plates is 10 mg/cm^2. The calculations modeled 90% of the ^{10}B, or 9 mg/cm^2. In SAR Section 9.1.7, the justification for the use of 90% credit is given, along with acceptance tests for the fabrication of the neutron absorber sheet materials.

The continued efficacy of the neutron absorber plates over a 20-year storage period is assured by the design of the TN-32 cask. Justification for this is given in SAR Section 9.1.7. The neutron absorber is a borated aluminum alloy material that is sandwiched between aluminum and steel plates that meet all structural and thermal requirements. The fabricated plates meet the thermal requirements and they can be expected to have no significant erosion or corrosion under ISFSI service. A structural analysis was performed which demonstrates that the basket plates will remain in place during all accident conditions. The neutron flux in the dry cask over the storage period is also very low such that ^{10}B 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 utilized the CSAS modules of the SCALE computer codes and the accompanying 27-group cross-section library for the TN-32 cask analysis and the benchmark calculations.

The staff performed confirmatory analysis with the CSAS/KENO.Va modules of SCALE developed at Oak Ridge National Laboratory[4]. The 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 system.

6.4.2 Multiplication Factor

Results of the applicant's criticality analysis show that k_{eff} of the TN-32 cask will remain below 0.95 for all allowed fuel loadings. The staff reviewed the applicant's calculated k_{eff} values and Upper Subcritical Limit (USL) and agrees that these values have been appropriately calculated to include all biases and uncertainties at a 95% confidence level or better.

The staff performed independent calculations using SCALE to ensure the staff understood the reactivity behavior of the cask with respect to the increase in fuel enrichment and the addition of BPRAs. The SAR indicates that the reactivity of the TN-32 cask filled with 2300 ppm borated water decreases as the fuel pin pitch is reduced. However, the staff expected k_{eff} to increase in this scenario because the amount of poison between the pins is greatly reduced. An analysis of infinitely reflected pincells revealed that this was in fact the case; k_{eff} increases as the pitch decreases. This result indicates that the borated water outside the assembly but inside the basket compartment plays an important role in isolating the assemblies from one another. As the fuel pin pitch is reduced in the full cask model, the amount of borated water in this region increases and further isolates the assemblies from one another, which results in a decrease in reactivity. The staff also expected k_{eff} to increase with the addition of the BPRAs, as these components displaced highly borated water as was shown by the applicant in Revision 11A of the SAR. Thus, the confirmatory analysis performed by the staff is in close agreement with the applicant's results for the TN-32 cask.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff concludes that the TN-32 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 comparisons on selected critical experiments which were chosen to bound the variables in the TN-32 cask design. The parameters in the benchmarks bounded the parameters in the analysis with respect to fuel enrichment, fuel pin pitch, boron areal density in the separator plates, hydrogen to U-235 atom ratio, water to fuel volume ratio, assembly separation, and soluble boron concentration. There were no trends in the bias.

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 benchmark comparisons in the SAR and agrees that the CSAS module of the SCALE computer codes used for the analysis was adequately benchmarked to representative critical experiments.

An USL of 0.9341 was calculated by the applicant. 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.

The staff reviewed the applicant's method for determining the USL and found it to be acceptable and conservative. The staff also verified that only biases that increase k_{eff} have been applied.

6.5 Supplemental Information

The following fuel types can be loaded into the TN-32 cask without compromising criticality safety requirements:

Fuel Type (PWR)	Maximum U/assembly (kg)
Westinghouse 17x17 standard	467.1
Westinghouse 17x17 OFA	428.2
Westinghouse 15x15	467.1
Westinghouse 14 X 14 standard ZCA and ZCB	414.4
Westinghouse 14x14 OFA	361.1

The maximum initial enrichment (prior to irradiation) of the assemblies is 4.05 wt%. The assemblies may contain BPRAs or TPAs. Fuel pins with cladding defects greater than pinhole or hairline cracks are not allowed in the TN-32 cask. These fuel pins must be removed from the assembly. All missing fuel pins must be replaced with a fuel pin or a dummy rod with the same external dimensions as the fuel before the assembly can be loaded into the cask.

All supportive information has been provided in the SAR, primarily in Sections 1, 2, and 6 and the TS.

6.6 Evaluation Findings

Based on the staff's review of Revision 11A of the TN-32 SAR and the staff's own confirmatory analyses, the staff concludes that the TN-32 cask meets the acceptance criteria specified in NUREG-1536. In addition, the staff finds the following:

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

F6.2 The TN-32 cask is designed to be subcritical under all credible conditions.

F6.3 The criticality design is based on favorable geometry, fixed neutron poisons, and soluble poisons in the spent fuel pool. 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 them.

F6.4 The analysis and evaluation of the criticality design and performance have demonstrated that the cask will provide for the safe 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 TN-32 cask 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 TN-32 cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted 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.

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

3. Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation, Volume 3, December 1987.

4. Scale 4.4, A Modular Code System for Performing Standarized Computer Analyses for Licensing Evaluation, Oak Ridge National Laboratory, September, 1998.

7.0 CONFINEMENT EVALUATION

Confinement systems must be designed to ensure that the annual dose equivalent, from normal operations and anticipated occurrences, to individuals beyond the controlled area is less than the limits set forth in 10 CFR 72.104(a)[1]. For design-basis accidents, radiation doses to individuals at or beyond the controlled area must be less than the limits given in 10 CFR 72.106(b). The cask design must also protect the spent fuel cladding against degradation that might lead to gross ruptures as required in 10 CFR 72.122(h)(1). The conclusions in this SER section are based on information provided in TN-32 SAR Revision 11A.

7.1 Confinement Design Characteristics

The TN-32 SAR contains a description of the confinement boundary in Sections 1.2.1, 2.3.2, and 7.1 and Figure 1.2-1. The confinement boundary includes the inner shell, the shell bottom plate, shell flange, the cask lid, the lid bolts, vent and drain port cover plates and bolts, and the inner metallic seals on the lid and the vent and drain ports. Confinement welds include a circumferential bottom closure weld, a circumferential weld attaching the top flange to the vessel shell, and longitudinal and circumferential welds needed to construct the cylindrical vessel. The cask lid is bolted to the shell flange on the cask body with 48 bolts. The lid has penetrations for the vent and drain ports that are closed by cover plates attached to the lid by 8 bolts each. The confinement vessel is designed, fabricated, and tested as closely as possible in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB[2]. Exceptions to the ASME Code are discussed in Section 7.1.1 of the SAR and listed in Table 4.1-1 of the TN-32 TS. The staff concludes that the description of the confinement boundary satisfies the requirements of 10 CFR 72.24(c)(3).

Cask closures, the lid, the vent port cover, and the drain port cover, are sealed by double metallic seals. The double metallic seals, seal materials, the sealing configuration, and seal reliability data are described in Sections 2.3.2.1 and 7.1.3 of the SAR. Each seal is designed to limit leakage rates to much less than the allowable leakage rate (both seals combined) of 1×10^{-5} ref cm^3/sec. Based on the design and test data provided by the applicant, the staff has reasonable assurance that the double metallic seals will provide a reliable and effective seal for spent fuel storage. The staff evaluated information in SAR Section 7.1.3 to demonstrate that the lid seals perform as separate seals when the lid bolts are properly torqued. The staff concludes that the seal design satisfies the requirements of 10 CFR 72.236(e) for redundant sealing of the confinement boundary.

A cask seal overpressure monitoring system (OMS) maintains helium between the inner and outer lid seals and the vent and drain port cover seals (the interseal) at a pressure higher than the cask cavity pressure and atmospheric pressure. As long as the interseal pressure is higher than the cask cavity pressure, cavity gas cannot leak out of the cask and air cannot leak into the cask cavity. The OMS provides continuous monitoring as discussed in SER Section 7.2.

The applicant provided procedures for draining and vacuum drying the cask interior during loading operations. TS 3.1.1 requires that a pressure of less than 4 millibar (3 Torr) be sustained for at least 30 minutes with the vacuum pump isolated from the system. Removal of water and potentially oxidizing material from the cask is necessary to protect the fuel cladding from degradation during storage. The cask is then backfilled with helium as required by TS 3.1.2. The helium cover gas protects the cladding from oxidation during storage and was also credited in the criticality and thermal analyses. After helium backfill, the cask is sealed and leak tested as required by TS 3.1.3 and 3.1.4 to demonstrate that the total cask leakage rate does not exceed 1 x 10^{-5} ref cm^3/sec. The staff concludes that these procedures provide reasonable assurance that residual water and other potentially oxidizing material are removed from the cask and that the fuel will be protected from severe degradation during storage.

The TN-32 uses multiple barriers provided by the fuel cladding and the cask confinement system to assure that there is no release of radioactive material to the environment. Section 3 of the SER concludes that all confinement boundary components, including the lid bolts, are maintained within their code-allowable stress limits during normal, off-normal, and hypothetical accident conditions. SER Section 4 concludes tha the peak confinement boundary component temperatures and pressures are within the design-basis limits for normal, off-normal, and hypothetical accident conditions. Leakage rate testing of the lid and vent and drain port cover plate metallic seals assure the integrity of the TN-32 closure. TN described the inspection and test acceptance criteria in SAR Section 9.1. The construction of the TN-32 with the redundant metallic seals, the OMS, and extensive inspection and testing provides reasonable assurance that release of radioactive material will not occur during normal storage and transfer conditions and that an inert atmosphere will be maintained in the cask cavity over the storage period.

7.2 Confinement Monitoring

The OMS is placed into service at the time of initial fuel storage and maintained over the storage lifetime. The OMS pressurizes the cask interseal region with helium to a pressure above the cask internal pressure. This ensures that helium would leak into the cask cavity if an unanticipated failure of the inner seal occurs. This system of metallic seals and helium pressurization also ensures that air does not leak into the cask cavity, protecting the fuel from degradation.

Pressure transducers or switches in the OMS are designed to be connected to a monitoring panel, provided by the licensee, to signal a low-pressure condition (e.g., OMS pressure below 3.2 atmospheres, absolute (atm abs)). Since the cask cavity is initially backfilled with helium to 2.1 to 2.3 atm abs at equilibrium temperature, any leakage would be from the OMS to either the cask cavity or the atmosphere. The minimum 3.2 atm abs limit allows sufficient time to detect and correct problems with cask seals before any potential leakage from the cask cavity occurs. TS 3.1.5 requires monitoring the cask interseal pressure at least once each 7 days and provides for periodic testing of OMS instrumentation.

In SAR Section 7.1.5, the applicant presented an analysis of the OMS pressure verses time, assuming leakage at the tested leakage rate (1x10^{-5} ref cm^3/sec) and temperature change due to decay heat decrease. This analysis demonstrated, and the NRC staff confirmed, that the OMS pressure would remain above the cask cavity pressure, and atmospheric pressure. However, if the seals leaked at the assumed tested leak rate, the analysis indicates that the OMS will need to be repressurized after about 9 to 12 years to avoid reaching the minimum 3.2 atm abs limit. Periodic testing of the OMS components per TS Surveillance Requirement 3.1.5.2 provides for repressurization of the OMS with inert gas at 3-year intervals. This provides additional assurance that the interseal pressure will be maintained above the cask cavity pressure and thus preclude leakage under normal conditions.

Given the high reliability of the double metallic seals, the simple, reliable design of the OMS, and the required surveillance of the cask interseal pressure, the staff concludes that the OMS meets the requirements of 10 CFR 72.122(h)(4) for continuous monitoring.

7.3 Nuclides with Potential for Release

The quantities of the radionuclides postulated to be released to the environment under off-normal and accident conditions were assessed using the methods and data given in NUREG/CR-6487[3] and ANSI N14.5-1997[4]. The fuel source term used in the calculations consists of all radionuclides that comprise greater than or equal to 0.1% of the total radioactivity in the fuel pins plus iodine, in accordance with NRC Interim Staff Guidance (ISG)-5[5]. The source term for Co-60 in the fuel crud was calculated using the method for calculating the inventory of crud on spent fuel surfaces given in NUREG/CR-6487.

The failed fuel and release fractions used in the analysis were taken from NUREG/CR-6487. For off-normal conditions, it was assumed that 10% of the fuel cladding fails and for accident conditions, 100% of the fuel cladding was assumed to fail. The release fractions given in Tables 7.3-1 and 7.3-2 of the SAR were consistent with the release fractions given in NUREG/CR-6487.

The applicant applied an additional 0.10 fraction for release of fuel fines based on the fraction expected to remain airborne after ejection from the fuel rod. This fraction was based on recommendations from Sandia Report SAND90-2406[6]. The staff reviewed this report and experimental results from rod burst tests reported in NUREG/CR-0722[7]. The staff performed particle buoyancy calculations to determine the maximum particle size that would remain airborne. The staff concludes that there is reasonable assurance that the 0.10 fraction bounds the fraction of fuel fines expected to remain airborne in the cask and, therefore, available for release after ejection from the fuel rod.

Leakage rates calculated in Section 7.3 of the SAR were calculated using the methods in ANSI N14.5-1997. The analysis was based on a confinement boundary tested leakage rate of $1x10^{-5}$ ref cm^3/sec as adjusted for applicable off-normal and accident temperatures and pressures and helium gas properties. The staff performed independent calculations that confirmed the off-normal and accident condition release rates given in the SAR.

7.4 Confinement Analysis

For normal conditions, the staff concludes that no discernable leakage during normal operations is credible and, therefore, dose at the controlled area boundary from atmospheric releases is not calculated because:

- the TN-32 confinement boundary is sealed and leak tested at the time of cask loading,
- the temperature and pressure of the cask are within design-basis limits, and
- the OMS functions to prevent leakage from the cask cavity to the environment.

The applicant evaluated the doses from off-normal and hypothetical accident conditions in the SAR to demonstrate compliance with the applicable requirements for off-normal operations and design-basis accidents. In these evaluations, the OMS pressurization function is assumed to have failed and a leak rate from the cask was calculated for the respective off-normal and hypothetical accident temperatures and pressures using the methodology discussed in SER Section 7.3. Other inputs and assumptions are summarized in SER Table 7-1.

Table 7-1 Summary of TN-32 Confinement Dose Calculation Inputs

Case	% Rods failed	Leak/ exposure duration	Pasquill stability class	wind speed (m/sec)	χ/Q method	Distances to site boundary, meters
Off-normal	10	1 year	D	5	RG 1.145[8]	100, 500
Accident	100	30 days	F	1	RG 1.25[9]	100, 500

TN used dose conversion factors (DCF) from EPA Federal Guidance Reports 11[8] and 12[9]. The staff noted that the applicant used the bounding DCF except for strontium 90 (Sr-90) and plutonium isotopes. For Sr-90, the applicant provided an acceptable justification for the use of a lower DCF based on the expected chemical solubility of compounds containing Sr-90. For the plutonium isotopes, the staff's confirmatory dose calculations demonstrated that the overall differences between the applicant's and NRC's estimated doses were small and primarily affected dose predictions for lung dose. Regardless of the assumed chemical form of the plutonium isotopes and the assumed DCF, the estimated doses were well within the allowable dose limits for the released plutonium isotopes. Estimated doses for off-normal and accident conditions assuming a 100 meter site boundary are summarized in SER Table 7-2.

Table 7-2 Summary of TN-32 Confinement Dose Estimates (mrem)

Off-normal Case 100 m	TEDE (limit 25 mrem)	Thyroid (limit 75 mrem)	Lung (limit 25 mrem)	Bone Surface (limit 25 mrem)	Other Organs* (limit 25 mrem)		
TN	1.9	0.26	4.6	19	1.1		
NRC	1.9	0.26	5.9	19	1.1		
Accident Case 100 m	**TEDE (limit 5000 mrem)**	**Thyroid (limit 50000 mrem)**	**Lung (limit 50000 mrem)**	**Bone Surface (limit 50000 mrem)**	**Other Organs* (limit 50000 mrem)**	**Skin (limit 50000 mrem)**	**Lens of Eye (limit 15000 mrem)**
TN	59	6.7	112	664	33	1	**
NRC	58	6.9	158	660	**	1	37

* Includes remaining organ tissue except skin and lens of the eye (e.g., liver, spleen, brain, and intestines (large and small)).
** Not calculated.

By review of the applicant's calculations and independent confirmatory calculations, the staff confirmed that the applicant's results and methods for estimating doses from postulated releases were consistent with the SRP, NUREG-1536[10], and ISG-5. The estimated doses for the minimum site boundary of 100 meters and beyond are within the limits of 10 CFR 72.104 for off-normal conditions and 10 CFR 72.106 for accident conditions, and are acceptable.

7.5 Latent Seal Failure Evaluation

As previously discussed, the TN-32 seals are designed for high reliability and analyzed to maintain integrity during normal, off-normal, and design-basis accident conditions during the licensed lifetime. The OMS allows for detection of postulated gross seal leakage within the frequency of the surveillance requirements. However, for postulated seal leakage rates greater than the tested rate, but not gross leakage, there could be a lag time before OMS pressure decays to 3.2 atm abs and indicates a low pressure condition. This degraded seal leakage is considered a "latent" condition and should be presumed to exist concurrently with other off-normal and design-basis events.

For the off-normal case, the OMS will limit leakage to the tested rate and, therefore, the results of the off-normal analysis address the latent condition. The OMS system is not designed to withstand design-basis accident conditions and, therefore, its function is not considered for the hypothetical accident concurrent with a latent degraded seal condition.

The applicant provided the results of sensitivity studies in SAR Section 7.3.3 to determine (1) the delay time from the onset of the latent condition to the point where the OMS would indicate system leakage and (2) the dose consequences if an accident occurred concurrent with the latent condition. For a leak rate of 1×10^{-3} ref cm^3/sec (100 times the tested leak rate) the delay time to indication of a degraded seal was 19 days. At this leak rate, the applicant's dose assessment concluded that the dose limits of 10 CFR 72.106(b) could be exceeded after the leak condition existed for 22 days.

The staff has reasonable assurance that the design of the TN-32 for latent seal leakage conditions is acceptable based on the following considerations:

- The possibility of the occurrence of a design-basis event that would remove the OMS concurrent with a degraded seal condition is judged to be very remote. The short delay time for detection of the latent condition further reduces the possibility of occurrence.

- If the accident were to occur, the staff expects that actions to mitigate the event would occur in less than 22 days. In SAR Section 8.4, the applicant has described recovery actions including installation of a blank flange on the OMS port to mitigate this event.

- In its dose assessment, TN calculated atmospheric dispersion factors (χ/Q) using the guidance of RG 1.25[11]. While the χ/Q model used in RG 1.25 provides a bounding estimate of the χ/Q, the model is applicable to short-term release durations (the assumed duration was 2 hours). An updated model for estimating χ/Q for longer release periods is provided in RG 1.145[12]. The staff performed independent dose calculations using χ/Q values based on the RG 1.145 model and found that the dose limits would not be exceeded for a 30 day accident with leak rates of 100 times the tested leak rate.

7.6 Evaluation Findings

F7.1 Sections 1, 2, and 7 of the SAR describe confinement SSCs important to safety in sufficient detail to permit evaluation of their effectiveness.

F7.2 The design of the TN-32 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 TN-32 provides redundant sealing of the confinement system closure joints using double metallic seals and a bolted lid. Penetrations into the cask cavity include a vent and drain port, both of which are in the cask lid. Both penetrations are sealed with double metallic seals and bolted closures.

F7.4 The confinement system is monitored with an OMS system as described in Section 7.2 of the SER . No instrumentation is required to remain operational under accident conditions.

F7.5 The quantity of radioactive materials 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 dose from direct radiation to demonstrate that the TN-32 design satisfies the requirements of 10 CFR 72.104(a) and 72.106(b).

F7.6 The cask confinement system has been evaluated by analysis to demonstrate that it will reasonably maintain confinement of radioactive material under normal, off-normal, and hypothetical accident conditions.

F7.7 The staff concludes that the design of the confinement system of the TN-32 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 TN-32 will allow safe storage of spent fuel. This finding is reached based on reviews that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, the applicant's analysis and the staff's confirmatory analysis, and accepted engineering practices.

7.7 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.

2. ASME Boiler and Pressure Vessel Code, Section III, Division I, 1992.

3. Anderson, B.L., R.W. Carlson, and L.E. Fischer, "Containment Analysis for Type B Packages Used to Ship Various Contents," NUREG/CR-6487, Lawrence Livermore National Laboratory, Livermore, California, November 1996.

4. 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.

5. Kane, W.F., "Spent Fuel Project Office Interim Staff Guidance 5," ISG-5, U. S. Nuclear Regulatory Commission, October 1998.

6. Sanders, T. L., et al., "A Method for Determining the Spent Fuel Contribution to Transport Cask Containment Requirements," SAND90-2406, Sandia National Laboratories, Albuquerque, New Mexico, November 1992.

7. Lorentz, R. A., et al., "Fission Product Release from Highly Irradiated LWR Fuel," NUREG/CR-0722, Oak Ridge National Laboratory, Oak Ridge, Tennessee, May 1980.

8. U.S. Environmental Protection Agency (EPA), "Limiting Values of Radionuclide Intake and Air Concentration Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report No. 11, EPA 520/1-88-020, Washington, DC, September 1988

9. U.S. Environmental Protection Agency (EPA), "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report No. 12, EPA 402-R-93-081, Washington, DC, September 1993.

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

11. U.S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating Accidents in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors," Regulatory Guide 1.25, March 1972.

12. U.S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1983.

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: cask loading and handling, cask storage operations, and cask unloading.

The information provided in Section 8 of the SAR, Revision 11A, forms the basis of the staff conclusions in this SER Section.

8.1 Cask Loading and Handling

The TN-32 SAR presents generic cask loading procedures. Detailed cask loading procedures must be developed by each cask user. Based on the information in SAR Section 8, as discussed below, the staff concludes that the general cask 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 TN-32 cask is compatible with wet loading. The staff also concludes that the cask 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 cask loading procedures presented in SAR Section 8 include important prerequisite, preparation, and receipt inspection provisions to prepare the cask for loading. Preparations include visual inspections of important components for damage such as the cask sealing components and closure bolts. The procedures include steps to replace the lid, vent port, and drain port seals. To meet TS requirement 3.3.1, the procedures specify that the cask fill water will have a minimum boron content of 2300 ppm.

8.1.2 Fuel Specifications

The loading procedures in the SAR state that pre-selected fuel assemblies may be loaded into the cask basket. Table 8.1-1 of the SAR also states that a procedure shall be developed by the user to ensure that the fuel loaded into the cask meets the fuel specifications and that the identity of the fuel assemblies loaded into the cask shall be verified. The fuel assembly specifications for the TN-32 cask are provided in TS 2.1. 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 TN-32 cask will meet the fuel-related assumptions (e.g., inventory, heat load, criticality-related parameters) made in the TN-32 SAR analyses.

8.1.3 ALARA

The staff concludes that the TN-32 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 5.2.3. The smooth external surfaces of the TN-32 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 cask 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 cask. The main intent of these procedures is to remove water and oxidizing impurities from the cask cavity to protect the fuel cladding from degradation.

The TN-32 lid is placed on the cask while it is submerged in the pool. Procedures require that at least six bolts be installed and hand tightened prior to fully removing the cask from the pool. After the cask is removed from the pool, the remaining lid bolts are installed and torqued to the values specified in SAR Drawing 1049-70-1. Verification of proper bolt torque is also required. Similarly, bolt torque requirements for the vent and drain port covers are provided on Drawing 1049-70-1.

When the lid is placed on the cask, the procedures ensure that the cask remains vented to preclude inadvertent pressurization as water in the cavity heats up. The bulk of the water is pumped from the cask via the drain line, and a vacuum drying system is then used to remove residual water from the cask cavity. Precautions are given to either control the evacuation rate, or provide a heat source on the evacuation line, to prevent blockage of the line by ice. Cask pressure is reduced to 4 millibar (3 Torr) and held for at least 30 minutes to verify appropriate levels of dryness are achieved. If the pressure increases above 4 millibar during the 30-minute holding time, the vacuum pumping process is repeated until this criterion is met. These steps are consistent with TS 3.1.1.

8.1.5 Filling and Pressurization

The TN-32 cask is backfilled with helium to slightly above atmospheric pressure and the vacuum drying adapter is replaced by a quick disconnect fitting. The cask is then re-evacuated to a pressure of 100 millibar and refilled with helium to a minimum pressure of 2230±100 millibar as specified in TS 3.1.2. A minimum helium purity of 99.99% is specified in TS 4.1.4. Calculations presented by the applicant indicated this process will leave about 0.17 gram-mole of oxidizing impurities in the cask cavity. This is less than the 1 gram-mole/cask recommendation given in PNL-6365[1] and is therefore sufficient to prevent severe fuel degradation during the 20-year storage period. Independent staff calculations of the residual impurity levels agree with the applicant's results. The procedure also states that the evacuation and backfill process must be repeated if the cask cavity is exposed to the atmosphere.

8.1.6 Cask Sealing

The operating procedures provide the steps to properly seal the cask, including helium backfill, necessary bolt torque, and leak testing. Section 8 of the SAR describes the steps for properly placing and tightening the lid, drain port, and vent port cover bolts that are consistent with the analyses presented in SAR Sections 2 (design criteria), 3 (structural evaluation), and 9 (acceptance tests and maintenance program).

The cask 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 SAR Section 9. The combined leak rate for all closure seals and the overpressure system is required by TS 3.1.3 and 3.1.4 to be less than 1×10^{-5} ref cm^3/sec. The procedures provide for final installation and testing of the OMS and corrective actions for a failed leak test, up to and including returning the cask to the pool to replace the lid seal. The staff concludes that the cask 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 TN-32 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. TS 3.1.6 limits cask lifting if the outer surface of the cask is below -20°F. For cask transport operations, the cask lift height above the transport surface will generally be limited to less than 18 inches. This lift height is reflected in TS 4.2.2. In addition, TS 5.2.2 requires that a site-specific transport evaluation program be developed to evaluate transport route conditions to ensure that design-basis drop limits are met. Consistent with TS 4.2.1, the procedures ensure that the casks are spaced a minimum of 16 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 SAR Section 8.3 in sufficient detail to permit cask users to develop detailed procedures. Appropriate procedures and precautionary statements provide for inspecting, maintaining, and testing the overpressure system. Maintenance operations, discussed in SAR Section 9, are anticipated to be minimal over the lifetime of the cask. Verification that the interseal pressure exceeds 3.2 atm is performed every 7 days and periodic testing of the overpressure instrumentation is performed in accordance with TS 3.1.5. 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 metallic seals and overpressure system. 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 Cask Unloading

As with the cask loading procedures, each cask user will be required to develop site-specific cask unloading procedures. The basis for the detailed user-developed cask unloading procedures is provided in Section 8.2 and Table 8.2-1 of the SAR. The general actions to be taken during unloading include transferring the cask to the spent fuel building, sampling the cask cavity gas, connecting fill and drain lines, lowering the cask into the pool, reflooding the cask with borated water (minimum of 2300 ppm boron), removing the cask lid, 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 TN-32 cask is compatible with wet unloading. In addition, the staff concludes that the generic cask unloading procedures presented in the TN-32 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 cask. 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 to prevent personnel contamination or exposure to airborne radioactive materials. The procedures indicate that the special precautions are to be planned, reviewed, and approved by the cask user's designated approval authority. 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 cask cavity is depressurized to atmospheric pressure, fill and drain lines are attached to the fill and drain ports in the cask lid, and the cask is lowered into the spent fuel pool. A typical vent and fill arrangement is shown in SAR Figure 8.2-1. The unloading procedure cautions cask users to ensure that the fill and drain lines are designed for steam at 100 psig to help protect against failures that could result in radiological exposures as well as personnel hazards (e.g., steam burns). Water with a minimum of 2300 ppm boron content, per TS 3.3.1, is slowly added through the drain port to fill the cask and gradually cool the fuel. The use of water with a minimum of 2300 ppm boron ensures the fuel will remain subcritical.

An analysis of cask pressure during reflood operations was presented in SAR Section 4 to demonstrate that cask pressures remain below the 100 psig design pressure limit. This analysis is the basis for controlling cask inlet water flow rates to 1 gallon per minute or less during the initial phase of cask fill. As shown in SAR Figure 8.2-1, a check valve on the water supply line will shut off flow if the pressure reaches 75.3 psig. 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 TN-32 generic procedures incorporate precautions and procedural steps to prevent or mitigate the potential dispersal of fuel crud particulate material. These include a required cover gas sample prior to lid removal and sampling the water ejected from the vent line during reflood operations. The applicant provided a note in the unloading procedures to alert cask users to the possibility that fuel crud could cause an airborne radiation condition due to floating particulates on the pool surface. The applicant provided suggested crud contamination control measures, including enhanced fuel pool filtration, increased area ventilation, and increased monitoring. The procedures and cautions regarding fuel crud were acceptable to the staff.

8.3.4 ALARA

The TN-32 cask unloading procedures incorporate general ALARA principles. ALARA practices include provisions to sample cask cavity gases to identify potential fuel cladding damage, monitoring of the water/steam ejected from the vent line during reflood, temporary radiation shielding, and respiratory protection, 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 TN-32 cask unloading procedures.

Any radioactive effluents generated during cask unloading will be governed by the 10 CFR Part 50 license conditions.

8.4 Evaluation Findings

F8.1 The TN-32 is compatible with wet loading and unloading. General procedure descriptions for these operations are summarized in Section 8 of the SAR. Detailed procedures shall be developed on a site-specific basis.

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

F8.3 The smooth surface of the cask is designed to facilitate decontamination. Only routine decontamination will be necessary after the cask 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 cask 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 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 TN-32 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 E.R. Gilbert, "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 staff reviewed the acceptance tests and maintenance program in the applicant's SAR to ensure they are appropriate for the TN-32 and that the applicable acceptance criteria have been satisfied in compliance with 10 CFR Part 72[1]. The principal objective of the acceptance tests and maintenance programs is to support commitments for TN-32 dry storage casks. A clear, specific listing of acceptance test and maintenance program commitments helps avoid ambiguities concerning design, fabrication, and operational testing requirements when the NRC staff conducts subsequent inspections.

9.1 Acceptance Tests

9.1.1 Visual and Nondestructive Examination Inspections

Visual inspections are performed at the fabricator's facility to ensure that the cask materials and components conform to the drawings and specifications, all specified coatings are applied, and the cask is free of defects. The casks are also visually inspected upon arrival at the user's facility to ensure the casks were not damaged during shipment and that casks are in conformance with the drawings and specifications. Any defects detected at the user's facility will be repaired or evaluated with respect to the effects of the defect on the component's ability to perform it's intended safety function. Cask design, fabrication, and testing are performed in accordance with the requirements of 10 CFR 72, Subpart G–Quality Assurance.

The TN-32 confinement boundary welds are designed, fabricated, tested, and inspected in accordance with the ASME Boiler and Pressure Vessel Code Subsection NB[2]. Nondestructive examination (NDE) requirements for welds are specified on the drawings provided in the SAR Section 1 using standard NDE symbols and notations in accordance with a standard of the American Welding Society (AWS 2.4)[3]. Exceptions to the ASME Code are specified in Section 7 of the SAR and Section 4 of the TS. Circumferential and longitudinal confinement boundary welds are examined volumetrically by radiography and liquid penetrant or magnetic particle methods and accepted in accordance with ASME NB-5000 standards. The bottom inner plate weld will be inspected using ultrasonic examination methods if the weld is applied before the outer and inner shells are assembled. This inspection is done radiographically and with either liquid penetrant or magnetic particle methods if the weld is applied after assembly. Non-confinement welds are inspected in accordance with the ASME Code, Subsection NF. Additional inspections will also be performed on the gamma shield shell to the bottom shield weld and the lid to the shield lid weld as specified in the SAR, Section 9. NDE personnel are qualified in accordance with American Society for Nondestructive Testing Recommended Practice SNT-TC-1A[3].

Basket welds are inspected using 100% visual inspection methods, assisted by remote visual inspection using mirrors and auxiliary lighting for basket welds that are not directly visible. Mechanical testing will be performed on at least one coupon from each welding machine to verify proper machine settings and operation prior to each working shift. Acceptance criteria for each weld test are based on failure of the base metal, prior to failure of the weld area, and visual verification that the fused weld zone is 0.5-inch in diameter. In addition, bubble leak tests are performed at 3 psi or greater on the resin enclosure to identify leak passages on the weld enclosures.

All structural materials are chemically and physically tested to ensure that the required properties are met. The confinement vessel materials are impact tested in accordance with the ASME Code Section III, Subsection NB, paragraph NB-2300, and meet the acceptance standards in paragraph NB-2330. Ultrasonic examinations of the closure flange and other forgings that form part of the confinement boundary are performed in accordance with paragraph NB-2542 and the acceptance standards provided in paragraph NB-2542.2. All external and accessible internal surfaces are tested using the liquid penetrant or magnetic particle methods in accordance with paragraph NB-2546 or NB-2545. Acceptance standards presented in paragraphs NB-2546.3 and NB-2545.3 are applied. Lid bolts, vent and drain cover bolts, and holes for the bolts are visually inspected in accordance with NB-2582. The lid bolts are also dye penetrate tested in accordance with NB-2520.

Fracture toughness of the TN-32 confinement boundary components is ensured by material selection and testing. The required nil-ductility transition temperature (T_{NDT}) of TN-32 cask materials is -80°F, which is 60°F below a service temperature of -20°F. Confinement boundary components will be tested in accordance with ASME Code, Section III, Division I, Article NB-2330. In addition to determining the T_{NDT}, Charpy V-notch testing will be performed at a temperature no greater than 60°F above the T_{NDT} temperature. Acceptance criteria are that at this temperature the material shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lbs absorbed energy.

The plate and forging materials of the confinement boundary will be examined by ultrasonic methods in accordance with ASME Code, Section III, Subsection NB, paragraphs NB-2530 and NB-2540, respectively. External and accessible internal surfaces of the forging materials will be examined by liquid penetrate or magnetic particle methods in accordance with NB-2546 or NB-2545. Welds will be examined by radiographic methods and either liquid penetrate or magnetic particle methods in accordance with Subsection NB, paragraphs NB-5210, NB-5220, and NB-5230. Allowable surface and subsurface flaw sizes are given in Appendix 3E of the SAR. The NRC staff concludes that this material selection and testing provides reasonable assurance that the confinement boundary materials will not be susceptible to brittle fracture at -20°F, which corresponds to the lowest (averaged throughout the day) temperature for which the cask has been approved for service.

The applicant conducted a similar analysis to determine testing requirements and acceptance criteria for the gamma shield material. Preliminary Charpy data provided by the manufacturer indicates the gamma shield material has relatively good Charpy impact properties at -20°F. The applicant determined allowable flaw sizes for the gamma shield material (see Appendix 3E) and committed to perform dye penetrate or magnetic particle testing on the final welds. No special examination requirements are specified for the gamma shield forged steel and plate components because the allowable flaw size is larger than the flaws generally observed in forged steel and plate components.

The NRC staff concludes that the visual and nondestructive examination inspections to be performed on the TN-32 cite appropriate standards, including ASME Code Section III, Subsections NB, NF, and NG, as stated in the SAR Sections 7 and 9. The welds and NDE requirements are clearly stated on the drawings in Section 1 of the SAR using appropriate American Welding Society (AWS) symbols.

9.1.2 Structural and Pressure Tests

Structural and pressure tests are subdivided into two sections: lifting trunnions and hydrostatic testing. Other tests, (i.e., tests related to leaks, shielding, neutron absorbers, and thermal), are discussed separately.

9.1.2.1 Lifting Trunnions

To ensure that the lifting trunnions perform satisfactorily, the trunnions on the TN-32 and TN-32A are load tested at 1.5 times the design lift load for 10 minutes in accordance with ANSI N14.6[4]. For the TN-32B, which is designed for non-redundant (single failure proof) lifting, load tests at 3 times the design lift load will be performed. The different load test conditions for the different trunnion designs are acceptable to the NRC staff. Following the load tests, the trunnion weld and bearing surfaces are examined using liquid penetrate or magnetic particle examination methods. Acceptance standards for these inspections are in accordance with ASME Code, Section III, Articles NF-5340 and NF-5350. NDE personnel are qualified in accordance with SNT-TC-1A[5]. These tests are acceptable to the NRC staff.

9.1.2.2 Hydrostatic Testing

A hydrostatic test of the confinement vessel will not be performed because the method of fabricating the cask precludes an effective hydrostatic test. Hydrostatic testing would need to be performed after the gamma shield and confinement vessel are assembled, so a hydrostatic test would not be meaningful. All confinement welds are fully radiographed in accordance with ASME Code, Subsection NB requirements. The stresses due to internal pressure are small when compared with the confinement boundary loads caused by the drop and tipover events. The NRC staff, therefore, agrees that the results of structural analysis and radiographic examinations of all confinement boundary welds provide reasonable assurance that the confinement boundary components can adequately withstand the effects of internal pressure.

9.1.3 Leak Tests

The applicant states that leakage tests are performed on the confinement system and overpressure system at the fabricator's facility using the helium mass spectrometry method or other method that provides the required sensitivity. Leakage tests are performed in accordance with ANSI N14.5[6]. The total leak rate, across the lid, vent, and drain port seals, must be shown to be less than 1×10^{-05} ref cm^3/sec at standard conditions and the sensitivity of the leak test procedure must be at least 5×10^{-06} ref cm^3/sec. Similar leakage rate criteria and sensitivities are applied to the overpressure system. The NRC staff concludes that the applicant's leak test requirements are in accordance with established requirements and are acceptable.

9.1.4 Shielding Tests

The neutron shield consists of a poured resin material, a proprietary borated reinforced polymer. Qualification testing of the procedures and personnel used for mixing and pouring the neutron shield will be performed to ensure that its properties meet the required specifications. During fabrication, both composition (chemical analysis) and density of the resin will be periodically tested to ensure consistency. The process controls include appropriate measures to ensure the absence of large voids. External dose rate surveys are performed on the loaded casks, as a final verification of neutron shield performance. TS 5.2.3 is included to limit the dose rate at the cask surface. The NRC staff concurs that together, these measures and specifications provide reasonable assurance that the cask will provide adequate shielding.

9.1.5 Neutron Absorber Tests

Fixed plates encase a neutron absorbing material in the form of borated aluminum sheets that contain 4.5 wt% boron enriched to 95 wt% ^{10}B. The borated aluminum sheets are used to ensure subcriticality during loading and unloading operations that use borated water inside the vessel. Within the borated aluminum sheets, the forms of the boron are precipitates of AlB_2 and TiB_2. These boride precipitates form within and are contained within the matrix of a standard (apart from these precipitates) aluminum alloy (series 1100, 6351, or 6063). These precipitates are very stable, second-phase particles that are very finely dispersed, with diameters in the range of 1 to 10 micrometers. The effects that these second-phase particles are expected to have on the physical properties of the parent alloy are the effects normally associated with uniform, finely dispersed, inert, equiaxed, second-phase particles. Further, the cask safety analysis does not rely upon the conductivity or mechanical properties of this material. Hence, the effects of the precipitates on properties of interest to service requirements are minimal. As these precipitates are stable and durable, the durability of the borated aluminum sheet material is expected to be governed by, and to be similar to, that of the matrix of the aluminum alloy in which the precipitates have been formed. The service conditions (radiation and thermal) are not so severe as to promote significant alterations of aluminum alloys used for this neutron absorbing material. Therefore, durability of these neutron absorbing materials is regarded to meet or exceed the service requirements of this application.

It is noted that, in production of the sheet material, finished sheets are visually examined for significant imperfections and these can be removed if the removal does not result in a dimensional non-conformance. Such a non-conformance could render the effective boron content of the sheet to be inadequate, i.e., to levels below that of the 10 mg/cm^2 required for this application.

The significant variable in the borated aluminum sheet material, the one that relates to service performance in this application, is the areal density of ^{10}B, as it determines the effectiveness of the sheet material as a neutron absorber. Samples taken from these sheets are used to measure both the boron content and its uniformity. Coupon samples of the sheet material are taken, as described in Section 9.1.7 and as shown in Figure 9.1-1 of the SAR. On these coupons, the ^{10}B content is measured by neutron transmission tests. The results of these measurements are compared with appropriate standards (e.g., ZrB_2 or TiB_2) and used to verify that the ^{10}B content meets the value required for the application. The effective boron content of each coupon, minus 3σ based on neutron counting statistics for that coupon, must be greater than or equal to the required minimum value of 10 mg $^{10}B/cm^2$. Adequate procedures have been established to handle the case in which a coupon fails this test. In addition to the verification of the ^{10}B content, the uniformity of the distribution of the boron is verified by neutron radioscopy or radiography of the coupons. Acceptance is based on uniform luminance across the coupons. These methods of testing the effective boron content and the uniformity of boron in the sheet material are regarded to be sufficient for verification that the neutron absorbing material is present at the required level and in a distribution that will satisfy the requirements of this application. The applicant takes credit for 90% of the 10 mg $^{10}B/cm^2$ and the staff concurs that this is justified based upon data on the effective ^{10}B content and the uniformity of the boron in the sheet material.

9.1.6 Thermal Tests

The applicant will perform a thermal acceptance test of one TN-32 production cask to verify the capabilities of the cask design to dissipate heat as predicted in SAR Section 4. This test will verify the radial heat transfer properties of the cask confinement shell, gamma shell, neutron shield shell, and outer shell, and thereby ensure that the cask fabrication process can control gaps between these various cask components within the range assumed in the thermal analysis.

The test will be conducted on a production cask in an upright position, without the basket. Helium will be maintained inside the cask during the test. A heat source will be placed inside the cask cavity to uniformly distribute the heat over the corresponding inner surfaces of the cask body. Insulation will be applied to the lid and bottom of the cask to reduce axial heat flow. A temporary barrier will be set up outside the cask to isolate the cask from the influence of air currents. Temperature readings will be taken on the interior surface of the cask, on the cask exterior surface, and in the ambient. A radial value of thermal conductance will be calculated from the test data and then compared with an analytically determined value.

The NRC staff concludes that this test will provide an acceptable means of verifying the heat transfer properties of the TN-32 cask. Should changes be made to the TN-32 method of manufacture in the future, the applicant must evaluate whether or not the design change could affect heat transfer and, if so, the test may need to be repeated.

9.1.7 Cask Identification

Section 1.2.1 of the SAR states that each cask will be marked with the empty weight and each cask will be identified with an alphanumeric mark that contains the specific model (TN-32, TN-32A, or TN-32B) and a sequential number corresponding to a specific cask (e.g., TN-32A-XX). This method of identification is an acceptable means of providing a unique, permanent, and visible number to permit identification of the cask.

9.2 Maintenance Program

The TN-32 storage cask requires little maintenance over its lifetime. All safety-related functions (e.g., confinement, shielding, criticality control, etc.) are provided by passive systems and components. Typical maintenance tasks identified in the SAR include occasional recalibration of seal monitoring instrumentation, verification of overpressure system tank pressure, and repainting. A procedure for calibration of the pressure transducers/switches is provided in Section 8.3 of the SAR. The staff concludes that maintenance and inspection programs are acceptable.

9.3 Evaluation Findings

F9.1 The applicant's proposed program for pre-operational testing and initial operations of the TN-32 are described in Sections 7.1 and 9.1 and Appendix 3E of the SAR. Sections 8.3 and 9.2 of the SAR discuss the proposed maintenance program.

F9.2 SSCs important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. The safety important SSCs are identified in Section 2.3 of the SAR. The applicable standards for their design, fabrication, and testing are given under Sections 2.2 and 2.5 of the SAR.

F9.3 The applicant will examine and/or test the TN-32 to ensure the absence of defects that could significantly reduce its confinement effectiveness. Sections 7.1, 9.1, and Appendix 3E of the SAR describe this inspection and testing.

F9.4 The applicant will mark the cask with a data plate indicating its model number, unique identification number, and empty weight. Section 1.2.1 of the SAR describes the data plate.

F9.5 The staff concludes that the acceptance tests and maintenance program for the TN-32 are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The acceptance tests and maintenance program are accepted as providing 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 applicable regulations, 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, 1992.

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

4. 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.

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

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 TN-32 to ensure that the cask meets regulatory dose requirements.

10.1 Radiation Protection Design Criteria and Features

10.1.1 Design Criteria

The applicant's SAR, Rev. 11A, lists four major sources of radiation protection design criteria, including 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), 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 the SAR 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 walls of the TN-32 cask body provide shielding from gamma radiation.
- The cask is surrounded by a borated resin-filled layer that provides shielding from neutron radiation.
- The confinement system includes double metallic seals and an overpressure system that prevent 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 cask body consists of smooth surfaces to facilitate decontamination prior to transfer to the ISFSI, 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 TN-32 cask 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 TN-32 cask 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 applicant's SAR. Chapter 12 of the SAR contains TSs 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 TN-32 cask 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 the SAR. 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 the SAR. The occupational dose calculations assume no temporary shielding is used. Occupational dose estimates are given in SAR Table 10.3-1 for cask loading, transport, and emplacement and in Table 10.3-2 for cask maintenance. The estimated total dose for cask loading, transport, and emplacement is as high as 4.25 person-rem per cask. Annual maintenance doses were calculated for four maintenance activities. The highest maintenance exposure calculated was 0.368 person-rem per cask for instrument operability verification and calibration. TSs are provided that include surface dose rates (see Administrative Control 5.2.3) and surface contamination limits (see LCO 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

An 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 TN-32 cask 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 TN-32 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 the SAR present the analysis of radiation doses during normal and off-normal operations for the TN-32 cask. 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. SAR Section 5.1 presents the results of the direct-path radiation dose calculations at distances of 100 to 500 meters from the cask. Section 10.2.2 of the SAR presents the skyshine dose rates for single casks and arrays.

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. The staff's review of the skyshine dose calculations is presented below.

Skyshine dose rates for a single bermed cask containing a design-basis fuel source were calculated by the applicant using MCNP as described in Section 5 of the SAR. The dose rates are given in SAR Table 10.2-1. The berm was assumed to be 4.2 meters high and located 20 meters from the cask centerline. Confirmatory calculations of skyshine dose rates were made with MCNP. The confirmatory calculations give a dose rate which is comparable to or lower than the applicant's calculations at a range of 250 meters and beyond. This comparison is within expected uncertainties and the NRC accepts the applicant's calculated skyshine dose rates. This conclusion also considered the fact that the nearest real person is typically 300 meters or more from the storage pad.

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 250 meters is necessary to meet the 25 mrem/yr limit in 10 CFR 72.104(a). If a berm is placed around the cask, effectively reducing the direct radiation dose to insignificant levels, a minimum distance of about 175 meters is necessary to ensure the dose rate is below the 10 CFR 72.104(a) limits. For a typical array of 48 casks placed inside a bermed area, a minimum distance of approximately 450 meters to the nearest real person is necessary to meet the regulatory limits.

The applicant's results and staff's confirmatory analysis 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 5.2.3) 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 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 the SAR. Accident conditions include hypothetical cask drop and tipover events, cask burial accidents, and possibly severe natural phenomena that could lead to simultaneous loss of the neutron shield and loss of one confinement barrier. The bounding dose is the sum of the direct radiation dose from loss of the neutron shield and the atmospheric dose from the loss of one confinement barrier 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 rates from direct exposures to a loss of the neutron shield were calculated by scaling the normal condition dose rate at 100 meters (SAR Table 5.1-3) to the ratio of the accident to normal dose rates at the surface (SAR Table 5.1-2). The analysis of public doses from atmospheric releases caused by loss of one confinement barrier and 100% fuel cladding failure accidents is presented in SAR Section 7.3. 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 calculations, in Section 11.2.5.3 of the SAR, to estimate the dose rate at 100 meters from loss of the neutron shield were also evaluated and confirmatory analyses were performed. The time-integrated direct radiation dose at 100 meters after the assumed loss of neutron shielding was calculated to be about 660 mrem, assuming an individual is present for an entire 30-day period. The staff's review of the doses from loss of one confinement barrier and 100% fuel failure is presented in SER Section 7. The total effective dose equivalent (TEDE) from this event was calculated to be 59 mrem at 100 meters from the cask. The total dose of about 720 mrem 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 TN-32 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 TN-32 and found it to be acceptable. TSs are provided that include surface dose rates (see Administrative Controls 5.2.3) and surface contamination limits (see LCO 3.2.1) to ensure that occupational exposures are maintained ALARA.

10.5 Evaluation Findings

F10.1 The TN-32 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 TN-32 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 TN-32 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 analysis 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

The conclusions in this SER section are based on information provided in TN-32 SAR Revision 11A.

11.1 Off-Normal Events

Section 11.1 of the SAR examines the causes, radiological consequences, and corrective actions for the off-normal events described below. The SAR determined that the confinement function of the TN-32 is not affected by off-normal conditions. However, because the cask lid seals are not demonstrated to be leak tight as defined in ANSI N14.5-1997[1], SAR Section 7 presented an analysis of the release of radioactive materials at the tested seal leakage rate as corrected for off-normal conditions. The radiological doses from this continuous atmospheric release are evaluated in Section 7.4 of the SER.

The staff reviewed the off-normal event analysis, performed confirmatory calculations, and found the estimated dose consequences were within the allowable limits. Therefore, staff has reasonable assurance that the dose to any individual beyond the controlled area boundary will not exceed the limits in 10 CFR 72.104(a)[2] during off-normal conditions and anticipated occurrences.

11.1.1 Loss of Electric Power

The applicant analyzed a loss of electric power as an off-normal event in SAR Section 11.1.1. Electric power provides area lighting and power to the OMS instrumentation. Neither area lighting nor the OMS instrumentation are important to safety. Because loss of electric power has no effect on confinement boundaries, there would be no radiological consequences from this event.

11.1.2 Cask Seal Leakage or Leakage of the OMS

SAR Section 11.1.2 presents analysis for cask seal leakage or leakage of the OMS. If the OMS is functioning, leakage from the cask is not expected as discussed in SER Section 7.2. As a bounding off-normal case, the OMS pressurization function is assumed to have failed and leakage of radioactive material from the cask is assumed for a source term and leak rate calculated using the methodology discussed in SER Section 7.3 for off-normal conditions. Radiological dose estimates from off-normal conditions presented in Section 7.3 of the SAR demonstrate that the TN-32 design meets the requirements of 10 CFR 72.104(a).

11.1.3 Off-Normal Pressures

Internal cavity pressures from off-normal conditions are presented in SAR Section 7.2.2 and reviewed by the staff in SER Section 4.5.3. Based on this review, the staff concludes that there is reasonable assurance that the TN-32 cask will maintain confinement under off-normal pressure conditions.

11.2 Design-Basis Accidents and Natural Phenomena Events

Section 11.2 of the SAR examines the causes, radiological consequences, and corrective actions for the identified design-basis accidents and natural phenomena events. The SAR demonstrated that the TN-32 would reasonably maintain its confinement function during and after design-basis accidents. However, SAR Section 11 evaluates the radiological doses from a combination of unlikely events, including events that result in loss of neutron shielding, loss of one confinement barrier, and simultaneous loss of neutron shielding and one confinement barrier. The applicant determined that the radiological dose at 100 meters would not exceed the dose limits specified in 10 CFR 72.106(b)[3].

The staff reviewed the design-basis accident analyses, performed confirmatory analyses, and found the estimated dose consequences to be within the allowable limits. Therefore, the staff has reasonable assurance that the dose to any individual beyond the controlled area boundary will not exceed the limits in 10 CFR 72.106(b) for credible hypothetical accident conditions. Sections 5, 7, and 10 of the SER provide further evaluations of the radiological doses during accident conditions.

11.2.1 Earthquake

11.2.1.1 Cause of Earthquake

An earthquake at the ISFSI site is postulated.

11.2.1.2 Consequences of Earthquake

The applicant performed a seismic event analysis to determine the effects of a design-basis earthquake on the TN-32 storage cask. This analysis demonstrated the stability of the TN-32 under application of vertical and horizontal seismic loading conditions. SAR Section 2.2.3 qualified the cask for an applied acceleration up to 0.26 g horizontal and 0.17 g vertical loading conditions. The TN-32 will not tipover or slide under the equivalent seismic loading conditions. The staff concludes that the cask would maintain confinement under these applied loading conditions and that no radiological or safety consequences result from this event.

11.2.2 Extreme Wind

11.2.2.1 Causes of Extreme Wind

The TN-32 cask will be placed on an unsheltered concrete pad at an ISFSI and will be subject to extreme weather conditions which could include extreme winds from a tornado.

11.2.2.2 Consequences of Extreme Wind

High-velocity winds from passing tornadoes will exert an external pressure load on the cask and could also generate large missiles that have the potential for striking and damaging the cask. The potential effects include cask tipover or penetration of the cask confinement boundary.

The analysis of cask stability under extreme wind loading conditions was presented in SAR Section 2.2.1. Cask stability was evaluated for a design-basis wind velocity of 360 mph and a pressure drop of 3 psi. The cask was shown to not tipover or slide as a result of the 360 mph wind. The external pressure drop of 3 psi, when combined with other internal pressure loads, was shown to be small when compared to the 100 psi design internal pressure of the cask.

Section 2.2.1.2.2 of the SAR evaluates the stability of the cask when subjected to various types of missiles. SAR Section 2.2.1.3 evaluates the potential damage to cask structures from impacts of various missiles. These analyses showed that the loaded TN-32 cask may slide a short distance as a result of a missile impact. The applicant determined that the cask will remain upright under simultaneous tornado wind and tornado missile loadings and that tornado missiles will not breach the cask confinement boundary. Staff review of tornado missile impacts is in SER Section 3.4.4.

A tornado missile may damage the OMS and the neutron shield. To determine the bounding radiological dose consequences from this accident, the applicant combined the doses from the loss of neutron shielding, in SAR Section 11.2.5.3, with the TEDE from the loss of one confinement barrier with 100% fuel cladding failure, in SAR Section 11.2.9.3. The estimated combined dose from this accident of about 720 mrem, is below the limits in 10 CFR 72.106(b).

11.2.3 Flood

11.2.3.1 Causes of Floods

A flood at an ISFSI caused by external events such as unusually high water from a river, dam break, seismic event, tsunami, and severe weather (e.g., hurricanes) is postulated.

11.2.3.2 Consequences of Floods

The analysis of flood effects on the TN-32 is presented in SAR Section 2.2.2. The cask was evaluated for a 57 ft static head of water (25 psi). The cask confinement boundary would not be compromised for static heads less than 57 ft. The cask was also evaluated for a water drag force of 57,160 lbs. This is equivalent to a stream of water flowing past the cask at 25 ft/sec. The analysis demonstrates that the cask would not tipover or slide at this water velocity. However, a flood could damage the OMS or deposit debris around the cask. The radiological dose consequences from this accident would be bounded by doses from the loss of one confinement barrier with 100% fuel cladding failure discussed in SAR Section 11.2.9.3. The impacts of flood debris are bounded by the cask burial analysis in SAR Section 11.2.10.

11.2.4 Explosion

11.2.4.1 Causes of Explosions

An explosion caused by combustion of the cask transporter fuel is credible. Explosions involving combustible materials shipped to reactor sites and on transportation links near nuclear power plant sites are also possible.

11.2.4.2 Consequences of Explosions

The external pressure wave generated in a credible explosion accident is on the order of a few psig. This is bounded by the design-basis external pressure of 25 psig. The structural evaluation in Section 3 of the TN-32 SAR demonstrated that the stresses on confinement boundary components from the design-basis external pressure are within allowable levels. The analysis demonstrates there will be no effect on the integrity of the confinement boundary as a result of a credible explosion event.

11.2.5 Fire

11.2.5.1 Causes of Fire

A rupture of the transporter vehicle fuel tank and subsequent ignition of a 200 gallon pool of spilled fuel is postulated.

11.2.5.2 Consequences of Fire

As described in SAR Section 4.5.1, a bounding, hypothetical fire is assumed that engulfs the cask and burns for 15 minutes. After the fire burns out, the post-fire thermal transient is evaluated. The thermal analysis in Section 4.5 of the SAR demonstrated that this fire would not compromise the integrity of the TN-32 confinement systems, that no melting of cask components would occur, and that the fuel cladding remains below its maximum short-term temperature limit. The applicant's internal pressurization analysis in SAR Section 7.3.2.2 demonstrates that the cask cavity pressure remains below the design pressure of 100 psig. Based on this analysis, the applicant concluded there would be no release of radioactive material from the cask. Staff review of the temperature and pressure analysis in SER Section 4 concludes that the fire accident component temperatures and cask internal pressure were within limits and, therefore, would not compromise the integrity of the cask.

However, the applicant concluded that the neutron shield would offgas during the hypothetical fire. As a result, a shielding analysis in Section 5 of the SAR calculated the radiation dose rate assuming the neutron shield resin is removed. Staff review of the shielding calculations is presented in Section 5 of the SER. The dose at the site boundary was calculated to be about 660 mrem, assuming the off-site receptor is located 100 meters away from a cask continuously for a 30-day period (no berm was assumed). The applicant's estimate of the site boundary accident dose was an approximation. This approximation scaled the differences in surface dose rates for the normal and accident case and used that value to multiply the normal dose rate at 100 meters to derive an accident dose rate. The staff concludes that this simplified approximation method was acceptable because the resulting dose was only a small fraction of the 5 rem limit of 10 CFR 72.106(b).

11.2.6 Inadvertent Loading of A Newly-Discharged Fuel Assembly

11.2.6.1 Causes of Loading a Newly-Discharged Fuel Assembly

An operator error or failure of administrative controls governing fuel handling operations is postulated.

11.2.6.2 Consequences of Loading A Newly-Discharged Fuel Assembly

The SAR evaluated the loading of a fuel assembly with a decay heat load greater than the design-basis 1.02 kW/assembly. As discussed below, the loading with spent fuel with a higher than design-basis decay heat load is not a credible accident.

The TN-32 SAR specifies the parameters for spent fuel assemblies allowed to be stored in the cask. In addition, TS 2.1 specifies the allowable fuel parameters. Section 8 of the SAR describes the general procedures for fuel loading. Specifically, Table 8.1-1 states that (1) preselected fuel assemblies will be loaded into the cask, (2) procedures will be developed by the cask users to ensure that the fuel to be loaded meets the fuel specifications, and (3) the fuel assembly identities are to be verified after they are loaded into the cask, including, as stated in SAR Section 11.2.6.2, a final verification of the fuel loaded into the cask and comparison with fuel management records. Fuel loading and measures to ensure that the TS requirements are met will be conducted under the cask user's QAP. This provides the staff with reasonable assurance that this condition will be detected and appropriate corrective actions will be taken prior to sealing the cask.

11.2.7 Inadvertent Loading of a Fuel Assembly with a Higher Initial Enrichment than the Design-basis Fuel

11.2.7.1 Causes of Improper Cask Loading

An operator error or failure of administrative controls governing fuel handling operations is postulated.

11.2.7.2 Consequences of Improper Cask Loading

The SAR evaluated the loading of a fuel assembly with an initial enrichment of 5 wt% U-235, which is greater than the design-basis initial enrichment of 4.05 wt% U-235. All fuel is modeled as fresh fuel. In Section 6 of the SAR, the applicant demonstrated that the cask remains subcritical under these conditions.

As discussed in SER Section 11.2.6.2, there are sufficient controls to detect and correct this loading error prior to sealing the cask. Therefore, there are no consequences from this event and no adverse effects on the cask system.

11.2.8 Hypothetical Cask Drop and Tipping Accidents

11.2.8.1 Causes of Cask Drop and Tipover Accidents

Although a handling accident event is unlikely, a cask drop from the handling height limit during transport is regarded as a credible event. The analysis of the TN-32 has shown that the cask does not tipover as a result of severe natural phenomena, such as earthquakes, tornadoes, tornado missiles, and floods. However, a cask tipover is evaluated as a bounding event to demonstrate the defense-in-depth of the design.

11.2.8.2 Consequences of Cask Drop and Tipover Accidents

Cask drop and tipping accidents are analyzed in Section 11.2.8 of the TN-32 SAR. The effects of drop (18-inch) and tipping accidents on the cask structures were evaluated in Section 3 of the SAR. Staff review of the structural analyses of these hypothetical accidents are located in Section 3 of this SER.

Sections 2.2, 3, and 11 of the TN-32 SAR qualify the cask for a bottom end drop deceleration of 50 g and side deceleration of 50 g, to simulate bounding loads for the end drop and tipover accidents, respectively. The cask was shown to maintain confinement after these deceleration conditions were applied. TS 4.2.2 and 5.2.2 are provided to control cask lift height or take other actions to ensure that decelerations are maintained below these levels. This control assures that the consequences of cask drop and tipover accidents are limited to those analyzed.

A drop accident could damage the OMS and the neutron shield. To determine the bounding radiological dose consequences from this accident, the applicant combined the doses from the loss of neutron shielding, in SAR Section 11.2.5.3, with the TEDE from the loss of one confinement barrier with 100% fuel cladding failure, in SAR Section 11.2.9.3. The estimated combined dose from this accident of about 720 mrem, is below the limits in 10 CFR 72.106(b).

11.2.9 Loss of Confinement Barrier

11.2.9.1 Causes of a Loss of Confinement Barrier

A loss of OMS integrity and leakage from the confinement is postulated.

11.2.9.2 Consequences of a Loss of Confinement Barrier

The dose calculations for a loss of confinement are presented in Section 7.3 of the SAR. The cask is assumed to leak for 30 days at the tested leak rate ($1x10^{-5}$ ref cm^3/sec) as adjusted for containment temperatures and pressures assuming 100% fuel cladding failure and internal temperatures at the maximum post fire level. Staff review of this analysis is presented in SER Section 7.4. As summarized in SER Table 7-2 for an assumed minimum site boundary of 100 meters, the calculated TEDE was 59 mrem and the maximum organ (bone surface) dose was 664 mrem. These calculated doses are well within the allowable limits and provide reasonable assurance that the doses at the controlled area boundary will not exceed the maximum dose criteria in 10 CFR 72.106(b).

11.2.10 Buried Cask

11.2.10.1 Cause of Buried Cask

The SAR analyzed the effects of cask burial that may result from an earthquake or other natural phenomena that could lead to burial of the cask under man-made or earthen material.

11.2.10.2 Consequences of a Buried Cask

Thermal analyses were performed in Section 4.5.2 of the SAR to evaluate cask temperatures assuming it is completely buried in a medium that interferes with natural convection cooling and unrestricted radiative heat transfer to the environment. The results indicated that the neutron shield would begin to degrade after 3 hours and that cask seals would reach their long-term maximum temperature limit after about 38 hours. Accordingly, actions to retrieve the cask should begin as soon as possible to prevent seal failure. The staff has reasonable assurance that a minimum of 38 hours before seal temperature limits are met provides sufficient time to implement corrective actions to prevent seal failure.

The dose consequences from this event would be represented by loss of the neutron shield combined with loss of one confinement barrier. To determine the bounding radiological dose consequences from this accident, the applicant combined the doses from the loss of neutron shielding, in SAR Section 11.2.5.3, with the TEDE from the loss of one confinement barrier with 100% fuel cladding failure, in SAR Section 11.2.9.3. The estimated combined dose from this accident of about 720 mrem, is the same as the bounding radiological dose from the extreme wind and cask drop and tipover accidents analyzed in Sections 11.2.2 and 11.2.8 of the SER. The resulting doses are well below the accident dose limits in 10 CFR 72.106(b).

11.2.11 Latent Seal Failure

11.2.11.1 Causes of Latent Seal Failure

Although extremely unlikely, it is possible for a seal to leak at a rate greater than the tested leak rate during the casks's 20-year storage period. For postulated seal leakage rates greater than the tested rate, but not gross leakage, there could be a lag time before OMS pressure decays to 3.2 atm abs and indicates a low pressure condition. This degraded seal leakage is considered a "latent" condition and should be presumed to exist concurrently with other design-basis events. If the outer seal has the latent failure or the OMS leaks, the inner seal still functions to provide confinement and there will be no release from the cask cavity. Of interest is a postulated latent failure of the inner seal concurrent with a postulated loss of OMS integrity from a design-basis accident.

11.2.11.2 Consequences of Latent Seal Failure

The applicant provided the results of sensitivity studies in SAR Section 7.3.3 to determine (1) the delay time from the onset of the latent condition to the point where the OMS would indicate system leakage, and (2) the dose consequences if an accident occurred concurrent with the latent condition. For a leak rate of 1×10^{-3} ref cm^3/sec (100 times the tested rate), a maximum delay time to indication of a degraded seal was 19 days. At this leak rate, the applicant's dose assessment concluded that the dose limits of 10 CFR 72.106(b) would be exceeded after the leak condition existed for 22 days. The staff concludes that the analysis of the latent seal failure was acceptable as discussed in Section 7.5 of the SER.

11.3 Criticality

Nuclear criticality evaluations are presented in Section 6 of the SAR. Criticality during the dry storage period is not credible as long as fresh water is prevented from entering the cask cavity. Confinement is maintained during credible hypothetical accidents and natural phenomena events, which will prevent fresh water from entering the cask cavity. The staff concludes that subcriticality will be maintained after credible accidents and natural phenomena events that could occur during dry storage.

As discussed in Section 6 of the SER, the applicant has shown that the irradiated fuel remains subcritical ($k_{eff} < 0.95$) under all credible normal, off-normal, and postulated accident conditions. The design-basis off-normal and credible accident conditions do not adversely affect the design features important to criticality safety. Based on the assessment provided in Section 6 of the SER, the staff concludes that the TN-32 meets the "double-contingency" requirements of 10 CFR 72.124(a).

11.4 Post-Accident Recovery

Section 11.2 of the SAR discusses corrective actions for each accident identified in Section 11.2. The SAR did not identify a design-basis accident that would affect the 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 TN-32 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 the TN-32 is a passive storage system, no instrumentation and control systems are required to remain operational under accident conditions. The SAR demonstrated that the confinement boundary integrity would be maintained during normal, off-normal, and design-basis accident and natural phenomena conditions. However, cask seal monitoring provided by the OMS may not be functional following an accident or severe natural event. The applicant evaluated the radiological consequences for seal leakage without the OMS function and demonstrated that estimated doses would only be a fraction of the 10 CFR 72.106 (b) limits. The applicant demonstrated that doses could be maintained within limits in the extremely unlikely event of an undetected, or latent, seal failure and a concurrent accident that breaches the OMS boundary. Post-accident recovery actions include verification and/or restoration of proper OMS function to ensure that technical specification surveillance requirements are met. The staff concludes that no TN-32 instrumentation is required to remain operational under accident conditions.

11.6 Evaluation Findings

F11.1 The SSCs of the TN-32 storage cask are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that may occur.

F11.2 The spacing of casks, discussed in Section 4.4.1.4 of the SER and included as TS 4.2.1, will ensure accessibility of the equipment and services required for emergency response.

F11.3 Table 12-1 of this SER lists the TS for the TN-32. These TS are further discussed in Section 12 of the SER.

F11.4 The applicant has evaluated the TN-32 to demonstrate that it will reasonably maintain confinement of radioactive material under credible accident conditions.

F11.5 An accident or natural phenomena event will not preclude the safe recovery of the TN-32 spent fuel cask.

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

F11.7 Neither off-normal nor accident conditions will result in a dose to an individual outside the controlled area that exceeds the limits of 10 CFR 72.104(a) or 72.106(b), respectively.

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 TN-32 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.

11.7 References

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

2. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Waste," Title 10, Part 72.104, as revised Federal Register, Vol. 63, No. 197, p. 54559, October 1998

3. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Waste," Title 10, Part 72.106, as revised Federal Register, Vol. 63, No. 197, p. 54559, October 1998.

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 TN-32 are fully defined in the CoC and the TS which are appended to it.

12.2 Technical Specifications

Table 12-1 lists the TS for the TN-32 dry storage cask system. The staff has appended these TS to the CoC for the TN-32.

12.3 Evaluation Findings

F12.1 Table 12-1 of the SER lists the TS for the TN-32. 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 TN-32 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 TN-32 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	Fuel to be Stored in the TN-32 Cask
2.2	Functional and Operating Limits Violations
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
3.0	SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	CASK INTEGRITY
3.1.1	Cask Cavity Vacuum Drying
3.1.2	Cask Helium Backfill Pressure
3.1.3	Cask Helium Leak Rate
3.1.4	Combined Helium Leak Rate
3.1.5	Cask Interseal Pressure
3.1.6	Cask Minimum Lifting Temperature
3.2	CASK RADIATION PROTECTION
3.2.1	Cask Surface Contamination
3.3	CASK CRITICALITY CONTROL
3.3.1	Dissolved Boron Concentration
4.0	DESIGN FEATURES
4.1	STORAGE CASK
4.1.1	Criticality
4.1.2	Structural Performance
4.1.3	Codes and Standards
4.1.4	Helium Purity
4.2	STORAGE PAD
4.2.1	Storage Location for Casks
4.2.2	Pad Properties to Limit Cask Gravitational Loadings Due to Postulated Drops
4.3	SITE SPECIFIC PARAMETERS AND ANALYSES

Table 12-1 TN-32 Technical Specifications (Continued)

Number	Technical Specification
5.0	ADMINISTRATIVE CONTROLS
5.1	TRAINING MODULE
5.2	PROGRAMS
5.2.1	Cask Sliding Evaluation
5.2.2	Cask Transport Evaluation Program
5.2.3	Cask Surface Dose Rate Evaluation Program

13.0 QUALITY ASSURANCE

Part 72 of Title 10 of the Code of Federal Regulations (10 CFR), 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 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 TN 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 TN-32 dry storage cask 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 TN-32 is provided in Section 14 of the SAR. TN presents two decommissioning options. In Option 1, the TN-32, including the spent fuel in storage, is shipped to either a monitored retrievable storage system or geological repository for final disposition. In Option 2, the spent fuel is removed from the cask and shipped in an NRC approved cask. Table 14.1-2 of the SAR provides the activity concentrations of the major radiation sources in the cask which TN has determined would exist after 20 years of irradiation by 32 design-basis PWR fuel assemblies stored in the TN-32 system and 30 days decay. The material activation results presented in Table 14.1-2 confirm that total system activation is low for all components. While the applicant has not kept the methods in this section current with the methods used in Section 5 of the SAR, the staff accepts this analysis because of the large margins in the results and the conceptual nature of the decommissioning plan.

TN determined that the TN-32 cask can be decommissioned using standard industry practices. Activated steel components can be decontaminated using existing mechanical or chemical methods.

14.2 Evaluation Findings

F14.1 The TN-32 system design includes adequate provisions for decontamination and decommissioning. As discussed in Section 14 of the SAR, these provisions include facilitating decontamination of the TN-32, 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 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 TN-32 are in compliance with 10 CFR Part 72. This evaluation provides reasonable assurance that the TN-32 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.

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