DRAFT
SAFETY EVALUATION REPORT

Docket No. 72-1027
TN-68 Dry Storage Cask
Certificate of Compliance No. 1027
Amendment No. 1

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SUMMARY

By letter dated January 14, 2005, as supplemented on July 15, and November 15, 2005; September 22, and December 8, 2006; and February 22, March 16, and March 23, 2007; Transnuclear, Inc. (TN) submitted a request to amend Certificate of Compliance (CoC) No. 1027 for the TN-68 Dry Storage Cask System. TN requested approval of several changes, which included increasing the allowable fuel burnup, minimum cooling times, decay heat, and fuel enrichment. The amendment also requested to include damaged fuel as authorized contents of the cask and to reduce the cask spacing on the storage pad. This safety evaluation report addresses only those changes made to the previously approved design.

The application, as supplemented, included the necessary engineering analyses and proposed Safety Analysis Report (SAR) page changes. The proposed SAR revisions will be incorporated into the Final Safety Analysis Report (FSAR).

The U.S. Nuclear Regulatory Commission (NRC) staff performed a detailed safety evaluation of the proposed amendment request which is documented in this safety evaluation report (SER). The staff's evaluation and conclusions are based on the information submitted by TN on January 14, 2005, as supplemented, requesting an amendment to CoC No. 1029 for the TN-68 Dry Storage Cask System. The staff determined that the proposed amendment of the TN-68 Dry Storage Cask System, meets the requirements of 10 CFR Part 72.

1.0 GENERAL INFORMATION

The objective of the general description review of Amendment 1 to the TN-68 Dry Storage Cask is to ensure that TN has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system, as amended.

1.1 Cask Description and Operational Features

The applicant revised the general description of the TN-68 Dry Storage Cask to incorporate changes requested in the amendment application. These changes involved minor revisions to the cask description. The most significant revision of the cask description involves the reduction of the spacing between casks on the storage pads from 16 ft to 14 ft.

1.2 Drawings

The applicant provided revised non-proprietary versions of the cask drawings with its amendment application. Section 1.5 of the SAR contains the revised drawings, which properly identify all structures, systems, and components (SSCs) that are important to safety, and specifies all changes from the previous drawing revision. The applicant included two additional drawings (Drawing No. 972-70-7, and No. 972-70-8) to the drawings list. The staff determined that the drawings contained sufficient detail to perform its review of the amendment request.

1.3 Cask Contents

The applicant requested to change the parameters of the fuel to be stored in the TN-68 Storage Cask. The TN-68 cask is designed to store up to 68 BWR fuel assemblies with or without fuel channels. The maximum allowable initial lattice-average enrichment varies from 3.7 to 4.7 wt% U-235, depending on the B-10 areal density in the basket neutron absorber plates. The maximum bundle average burnup, maximum decay heat, and minimum cooling time are 40 GWd/MTU, 0.312 kW/assembly, and 10 years for 7 x 7 fuel, 60 GWd/MTU, 0.441 kW/assembly, and 7 years for all other fuel. The cask is designed for a maximum heat load of 30 kW. Fuel specifications are detailed in Section 2.1 of the Technical Specifications (TS).

The applicant also requested to include damaged fuel as contents of the TN-68 cask. Damaged fuel that can be handled by normal means may be stored in eight peripheral compartments fitted with damaged fuel end caps designed to retain gross fragments of fuel within the compartment.

1.4 Qualifications of the Applicant

Section 1.3 of the SAR contains reference to the applicant's qualifications and has not changed from the previous FSAR.

1.5 Evaluation Findings

F1.1 A general description of the TN-68 Dry Storage Cask, as amended, is presented in Chapter 1 of the SAR.

F1.2 Drawings for the TN-68 Dry Storage Cask, identifying SSCs important to safety are presented in Section 1.5 of the SAR.

F1.3 Specifications for the spent fuel to be stored in the TN-68 Dry Storage Cask, as amended, are discussed in Section 1.2.3 of the SAR.

F1.4 The technical qualifications of the applicant are identified in Section 1.3 of the SAR, which are unchanged from the FSAR.

F1.5 The quality assurance program for the TN-68 Dry Storage Cask remains unchanged, and is referenced in Chapter 13 of the SAR.

F1.6 The TN-68 Dry Storage Cask has been certified for transport use under the general license provisions of 10 CFR 71.17, as authorized by Certificate of Compliance (CoC) No. 9293. However, as of the approval date of this amendment, the applicant has not requested to amend its Part 71 CoC to include the modifications requested by this amendment.

F1.7 The staff concludes that the information presented in this section of the SAR satisfies the general description requirements of 10 CFR Part 72.

2.0 PRINCIPAL DESIGN CRITERIA

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

2.1 Structures, Systems, and Components Important to Safety

The SSCs important to safety are summarized in Table 2.3-1. The applicant did not request any changes to the list of SSCs important to safety for this amendment request for the TN-68 Dry Storage Cask.

2.2 Design Basis for SSCs Important to Safety

The TN-68 Dry Storage Cask design criteria summary includes the range of spent fuel types and configurations to be stored, and design criteria for environmental conditions and natural phenomena. This summary discusses the changes requested by the applicant's amendment application.

2.2.1 Spent Fuel Specifications

TN requested to revise the spent fuel specifications for the TN-68 Dry Storage Cask. The TN-68 Dry Storage Cask is designed to store 68 General Electric (GE) BWR fuel assemblies with or without fuel channels. The maximum allowable lattice-average initial enrichment varies from 3.7 to 4.7 wt% U-235, depending on the B-10 areal density of the basket neutron absorber plates. The maximum bundle average burnup, maximum decay heat, and minimum cooling time are 40 GWd/MTU, 0.312 kW/assembly, and 10 years for 7 x 7 fuel, 60 GWd/MTU, 0.441 kW/assembly, and 7 years for all other fuel. The cask is designed for a maximum heat load of 30 kW. The bounding fuel parameters for each type of fuel assembly are discussed in Table 2.1-2 of the SAR. Table 2.1-4 and Figure 2.1-4 provide the minimum cooling times required for all combinations of maximum fuel burnup and initial lattice-average enrichment.

The amendment made two spent-fuel related requests: (1) to store damaged fuel as well as intact fuel, and (2) to increase the burnup limit to 60 GWd/MTU maximum bundle average.

The dimensions and characteristics of the spent fuel, such as cladding thickness and pellet diameter, were spot checked and found to be accurate. For damaged fuel, only 8 of the 68 available spaces may be used in a single cask and the assembly average burnup for these damaged fuel assemblies shall be limited at 45 GWd/MTU.

Instead of defining damaged fuel, a definition is provided for intact fuel. By default, fuel not defined as intact can be assumed to be damaged. This corollary definition of damaged fuel meets the guidance of ISG-1, Rev. 1. The definitions for damage, both to fuel rods and fuel assemblies, are given in the Technical Specifications. Steps have been included in the operating procedures to ensure that fuel is checked for damage and loaded into the correct basket type.

Damaged fuel that can be handled by normal means may be stored in eight peripheral compartments fitted with damaged fuel end caps designed to retain gross fragments of fuel within the compartment. Damaged fuel must not have any missing fuel pins or fuel pin segments. Missing fuel pins or fuel pins segments must be replaced with dummy rods that displace a volume equal to or greater than that of the original rods.

2.2.2 External Conditions

Section 2.2 of the SAR identifies the bounding site environmental conditions and natural phenomena for which the TN-68 Dry Storage Cask is analyzed. No changes to these conditions are necessary as a result of the amendment request.

2.3 Design Criteria for Safety Protection Systems

Section 2.3 of the SAR identifies the design criteria for all safety protection systems on the TN-68 Dry Storage Cask. Minor changes to this section were made for this amendment request, and do not change or revise any of the safety protection systems of the cask.

2.4 Evaluation Findings

F2.1 The staff concludes that the changes requested to the principal design criteria for the TN-68 Dry Storage System are acceptable with regard to meeting the regulatory requirements of 10 CFR Part 72.   A more detailed evaluation of the changes to the design criteria and an assessment of the compliance with those criteria is presented in Section 3 through 12 of the SER.

3.0 STRUCTURAL EVALUATION

By application dated January 14, 2005, Transnuclear, Inc. (TN) requested an amendment to Certificate of Compliance (CoC) No. 1027 for the TN-68 Dry Storage Cask. The amendment requested, among several changes, to include "damaged" fuel as well as "intact" fuel as authorized contents, and add high burnup, Zircaloy clad 8 x 8, 9 x 9, and 10 x 10 BWR fuel assemblies to the approved inventory of the cask. To assess compliance with 10 CFR Part 72 requirements, the staff requested additional information on July 15, 2005. TN submitted responses to this first round of Request for Additional Information (RAI) to staff on November 15, 2005. The staff reviewed the completely revised Appendices 6A and 6B presenting new structural analyses of the fuel in this revised amendment application. A second round of RAI was generated on April 24, 2006. TN responded to this RAI on September 22, 2006. The staff found the fracture mechanics data used to justify the adequacy of the cladding for high burnup damaged fuel in the response to be unacceptable. The staff requested TN to update Section 6.B to omit high burnup damaged fuel and resubmit the application. On December 8, 2006, TN submitted Revision 4 of the application. The staff's structural review of this amendment application for the TN-68 Dry Storage Cask is limited to verifying compliance with the requirements of 10 CFR Part 72, and does not constitute approval of any of the package performance requirements required by 10 CFR Part 71.

3.1 Evaluation Results

Appendix 6A, "Evaluation of Undamaged Fuel Under Accident Accelerations," evaluates the structural integrity of fuel rod cladding during side drop, and bottom end drop, using material properties of high burnup fuel assemblies.

For the side drop analysis, one vertical row of each fuel assembly type was modeled in an ANSYS finite element model. The model was subjected to inertia loads due to cladding tube mass, fuel pellets mass, and the fuel assembly end fitting mass. The weight of the fuel pellets was incorporated by adjusting the equivalent density of the fuel cladding. Fuel cladding thickness was reduced by 0.0027 inch to account for oxidation. The loads were entirely taken by the cladding and no credit was taken for fuel pellet moment of inertia. The fuel cladding axial tensile stress due to internal pressures was added to the bending stress from the side drop analyses and compared with the dynamic yield strength of fuel cladding material at the operating temperature to demonstrate that fuel cladding will not fail during the side drop. The maximum overhang length of the fuel assembly (from top end of the end fitting to first support grid spacer) used for the analyses was 29.255 inches (GE12 10 x 10).

The staff noted that for all fuel types, the maximum bending stresses occur at the top edge of the basket. The fuel cladding temperature at this location was less than 400°F. An ANSYS modal analysis was conducted to calculate lateral natural frequencies of the bounding case to determine the dynamic load factor for side drop g-loading. The accident side drop loads for the TN-68 cask (a dual-purpose cask for storage and transportation) include tip-over and a 30 foot side drop onto an unyielding surface with an impact limiter. A value of 75g was used for static analysis including dynamic load factor. The calculated maximum stress (fuel cladding axial tensile plus the bending stress from the side drop) of 75,764 psi for this assembly was less than the cladding material (Zircaloy-2) dynamic yield stress of 101,431 psi at 400°F. The staff finds the approach used in this analysis and the results acceptable.

For the bottom end drop analysis, a transient dynamic analysis was performed using the ANSYS finite element code and elastic-plastic material properties to determine the stresses and/or total strains in the fuel rods. The strain results were compared to permissible strains for irradiated Zircaloy-2 cladding at 650°F under the dynamic loads for an end-drop condition. All of the acceleration values from the response curve obtained from the TN-68 cask bottom end drop scale model test were scaled by a factor of 1/3 and all of the time values were scaled by a factor of 3. All of the acceleration values were scaled such that the maximum g-load was 75g. An initial deflection (bowing) value of 0.039 inch was introduced in the bottom two spans to account for initial out-of-straightness and for distortion/twisting from radiation growth. The staff finds the estimated initial bowing of 0.039 inch to be conservative.

The ANSYS transient dynamic analyses results show that the cladding yield stress exceeded the allowable yield stress for the 9 x 9 (GE11), and 10 x 10 (GE12) assembly rods. Based on limited data from ongoing work at PNNL, the applicant used a failure strain for high burnup fuel in the range of 1.7% to 3%. The actual total strain for the 9 x 9 assembly was 1.17%, and 1.07% for a 10 x 10 assembly. The applicant concluded that, although some of the fuel rods could yield, they would not fail. The staff acknowledges the conservative assumptions used in this analyses, and concurs with the applicant's conclusion.

Appendix 6B, "Damaged Fuel Cladding Structural Evaluation," provides a definition of damaged fuel, and evaluates structural integrity of the damaged fuel cladding in the TN-68 basket following normal and off-normal loading conditions of storage and on-site transfer, and normal condition of offsite transportation. A damaged fuel assembly is defined as fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks and pinhole leaks. An intact fuel assembly is redefined in the revised Technical Specification Section 2.1.1 as a spent nuclear fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means; and fuel assembly that do not have any damage to spacer grids that renders the fuel outside its design and licensing basis for use in the reactor. The staff has reviewed the definition of "damaged fuel" and "intact fuel," as cited above, and accepts these definitions, as they are defined based on the properties the fuel and the cladding must exhibit to meet the requirements of storage (and/or transportation).

The structural integrity of the damaged fuel for the loads under normal conditions for both Part 72 and Part 71 were evaluated. The one-foot end drop (15g) and one-foot side drop (35g) during transport conditions bounds loads, such as, vibratory loads, shock loads, and lifting and tie-down loads.

The ANSYS models in Appendix 6A, mentioned above, were used for the 35g side-drop analyses. The maximum computed stress in the damaged fuel rod was found to be 48,170 psi for 8 x 8 fuel, which was less than the irradiated Zircaloy-2 cladding material allowable yield stress of 69,000 psi at 750°F. This allowable value for the low burnup damaged fuel that is proposed to be stored in the eight peripheral compartments of the TN-68 fuel basket fitted with end caps to retain and retrieve damaged fuel fragments, was taken from a report dated November 2005, "PNNL Stress/Strain Correlation for Zircaloy," by Geelhood K. J., and Beyer, C. E.

The staff noted that the maximum allowable dynamic yield stress of 101,431 psi at 400°F was used for the high burnup undamaged fuel in Appendix A, mentioned above. This is acceptable as the fuel temperature at the location where the maximum bending stress occurs due to side drop, was at 400°F. The overall stability of the fuel cladding tube was demonstrated by verifying that no buckling occurred. The staff concurs with the overall approach used in this analyses.

For the one-foot side drop, the applicant demonstrated, by using fracture mechanics procedures, that for the low burnup damaged fuel rods, (i.e., fuel rods with the maximum peak rod average burnup < 45 Gwd/MTU), the fuel rods will maintain their structural integrity. The computed stress intensity of 2.0 ksi $in^{1/2}$ (8 x 8 fuel) was less than the critical (crack initiation fracture toughness) stress intensity of 16.36 ksi $in^{1/2}$. The allowable value of 16.36 ksi $in^{1/2}$ was taken from the lower-bound value presented in the EPRI Report 1001281, “Fracture Toughness Data for Zirconium Alloys, Application for Spent Fuel Cladding in Dry Storage,” dated January 2001. The staff considers this allowable fracture toughness value to be conservative since it is measured at relatively low temperatures. The through-wall circumferential crack in the cladding under bending was evaluated using the computer code “pc-CRACK.” The staff agrees with the approach used in the calculations presented in Section 6B.3.a, “Part 72 Normal and Off-normal Condition Loads.”

3.2 Evaluation Findings

F3.1 The staff concludes that the analyses presented in Appendices 6A and 6B of Revision 4 of the application for Amendment No. 1 to the TN-68 Dry Storage Cask comply with requirements of 10 CFR 72.236(b), (c), (d), (h) and (I).

4. THERMAL EVALUATION

The objective of the thermal review is to ensure that the cask components and fuel material temperatures of the system will remain within the allowable values for normal, off-normal and accident conditions. This objective includes confirmation that the temperatures of the fuel cladding will be within acceptable limits throughout the transfer and storage periods 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 methods.

4.1 Spent Fuel

The applicant is seeking approval of the use of the TN-68 cask for the storage of up to 68 BWR spent fuel assemblies (including up to 8 damaged assemblies) with a maximum total decay heat load of 30 kW.

4.2 Cask System Thermal Design

4.2.1 Thermal Analysis Objectives

Section 4.1 of the SAR defines several thermal analysis objectives, including the determination of the:

1. Acceptable minimum and maximum temperatures of all cask materials.

2. Temperature distributions for thermal stress analyses.

3. Maximum cask internal pressures for all conditions.

4.2.2 Design Criteria

The design criteria for the system 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 the normal storage conditions for the 20-year approval period and postulated off-normal and accident conditions and include:

1. Containment component temperatures must be within acceptable limits.

2. Fuel cladding temperatures must be consistent with ISG-11 recommendations.

3. Maximum seal and neutron shield temperatures must be acceptable.

4. Maximum cask internal pressure must not exceed allowable limits.

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

4.3 Thermal Load Specifications

The applicant selected bounding fuel configurations (including damaged fuel assemblies) as input to the thermal models. The staff has reviewed this selection and has reasonable assurance that these loads are bounding.

4.3.1 Normal and Off-Normal Conditions

These conditions are described in Section 4.3 of the SAR. The storage conditions consider a maximum average daily temperature of 100°F and include solar insolation as recommended in 10 CFR Part 71. The minimum temperature storage condition considers a -20°F temperature and assumes no solar insolation.

The staff concludes that the applicant's approach of using these temperatures 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 in TS Section 4.3 as siting parameters that must be evaluated by the storage system user.

4.3.2 Accident Conditions

4.3.2.1 Buried by Dirt and/or Debris

Accident conditions are postulated by the applicant. One accident is a complete burial of the cask. The burial boundary conditions are stated in SAR Section 4.4.1.2 and are appropriately applied.

4.3.2.2 Fire

Another postulated accident is a fire (SAR Section 4.4.1.1). A 15 minute fire with an average flame temperature of 1475°F, an average convective heat transfer coefficient of 4.5 Btu/hr-ft^2-°F, and radiative heat transfer as recommended in 10 CFR 71.73 is hypothesized. This is postulated to be caused by the spillage and ignition of 200 gallons of combustible transporter fuel forming a circular pool around the cask. The assumed 15-minute duration for the transient evaluation is conservative based on a calculated fire duration of 13 minutes for this amount of fuel, at a consumption rate of 0.15 in/min and a pool size of roughly 176 inches in diameter (which is 1 meter beyond the cask surface when in a vertical orientation). The staff finds that this hypothetical accident condition is reasonable.

4.3.3 Fuel Loading and Unloading Conditions

The applicant's description of the effects of loading and unloading conditions on the system is provided in Section 4.5 of the SAR. The loading conditions evaluated by the applicant include vacuum drying and reflooding. The staff reviewed the modeling conditions and found them to be acceptable.

4.4 Model Specification

4.4.1 Configuration

The applicant developed thermal models of the cask using a finite element code (ANSYS) as discussed in SAR Section 4.3 (normal and off-normal conditions), Section 4.4 (accident conditions) and Section 4.5 (loading and unloading). The staff reviewed the details of these models and determined that the models are adequate to determine the thermal performance of the cask in the stated conditions.

4.4.2 Material Properties

The material properties used in the thermal analysis of the cask are listed in SAR Section 4.2. The applicant provided a summary of the material compositions and thermal properties for all components used in the system. 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 fuel assemblies, the applicant described the source from which the effective thermal properties were derived.

4.5 Thermal Analysis

4.5.1 Temperature Calculations

To determine the temperatures of the TN-68 cask components for various storage configurations the applicant provided analyses using the ANSYS® general purpose finite element analysis (FEA) modeling approach, which indicated that adequate margin existed to the limits for the fuel cladding and other cask components; however, temperatures for the radial neutron shield exceeded operating limits, based on a revised analysis provided by the applicant at the staff's request (See Section 4.5.3.1). Calculation results are summarized in Section 4.5.1.5 below.

4.5.1.1 Normal and Off-Normal Conditions

SAR Table 4.3-1 lists the maximum temperatures for these storage conditions with an ambient temperature of 100°F. No material temperatures exceeded stated limits.

4.5.1.2 Accident Conditions- Fire

SAR Table 4.4-1 lists the maximum temperatures for the hypothetical fire accident. No material temperatures exceeded stated limits (the applicant assumes the neutron shields have been replaced with air).

4.5.1.3 Accident Conditions - Buried Cask

SAR Table 4.4-2 lists the maximum temperatures for the postulated buried cask accident. No material temperatures exceeded stated limits (the applicant assumes the neutron shields have been replaced with air).

4.5.1.4 Cask Heatup Analyses

Average heat up rates are stated in SAR Table 4.5-1. SAR Tables 4.5-2 and 4.5-3 list the temperatures (and time to achieve) for the vacuum drying process with and without helium backfill. The calculated times are used to determine limits in the Technical Specifications and Procedures.

4.5.1.5 Summary of Thermal Analysis Results

Normal Conditions (100°F ambient)			
Component	**Temperature (°F)**	**Time (hours)**	**Limit (°F)**
Fuel Cladding	622	Steady State	752
Radial Neutron Shield	303[1]	Steady State	300
Top Neutron Shield	217	Steady State	220
Hypothetical Accident Condition Fire			
Fuel Cladding	737	Steady State	1058
Buried Cask Transient			
Fuel Cladding	1058	90	1058
Radial Neutron Shield	300	0.6	300
Top Neutron Shield	300	38	300
Vacuum Drying Without Helium Backfill			
Fuel Cladding	752	37	752
Vacuum Drying With Helium Backfill (at 30 Hours)			
Fuel Cladding	711	30	752
Fuel Cladding	606	Steady State[2]	752

[1] See discussion in Section 4.5.3.1
[2] Maximum temperature following vacuum drying and helium backfill

4.5.2 Pressure Analyses

The pressure analyses are documented in SAR Section 4.7. The staff reviewed the calculations and found them to be acceptable.

4.5.3 Confirmatory Analyses

Confirmatory analyses were conducted by the staff in both the ANSYS® general purpose FEA code and Cobra-SFS finite difference codes. The analyses conducted by the staff indicated that adequate margin to the limits existed for the fuel cladding and other cask components; however, temperatures for the radial neutron shield, determined using the ANSYS® FEA modeling approach, exceeded allowable limits.

The Cobra-SFS modeling approach indicated a greater margin between the determined component temperatures and their established limits, giving the staff additional confidence that the critical components of the TN-68 cask would perform their necessary safety functions.

4.5.3.1 View Factor Calculation Sensitivity Study

The staff reviewed the applicant's approach for determining the view factor for radiative exchange from the cask surface to the environment (which used an engineering textbook correlation rather than an actual calculation) and determined that it was a non-conservative approach.

To more precisely evaluate the level of conservatism in the results obtained with the proposed methodology, the staff constructed a view factor computation model for a fully loaded TN-68 cask in a typical storage array (as illustrated in Figure 4.10-1 of 72-1027 TN-68 Amendment 1 SAR, Rev. 0, 01/05). This model was developed using RadCad® v5.0, which determines view factors between specified geometric shapes by means of ray-tracing. A graphical representation of the RadCad® v5.0 model is shown in Figure 1 below.

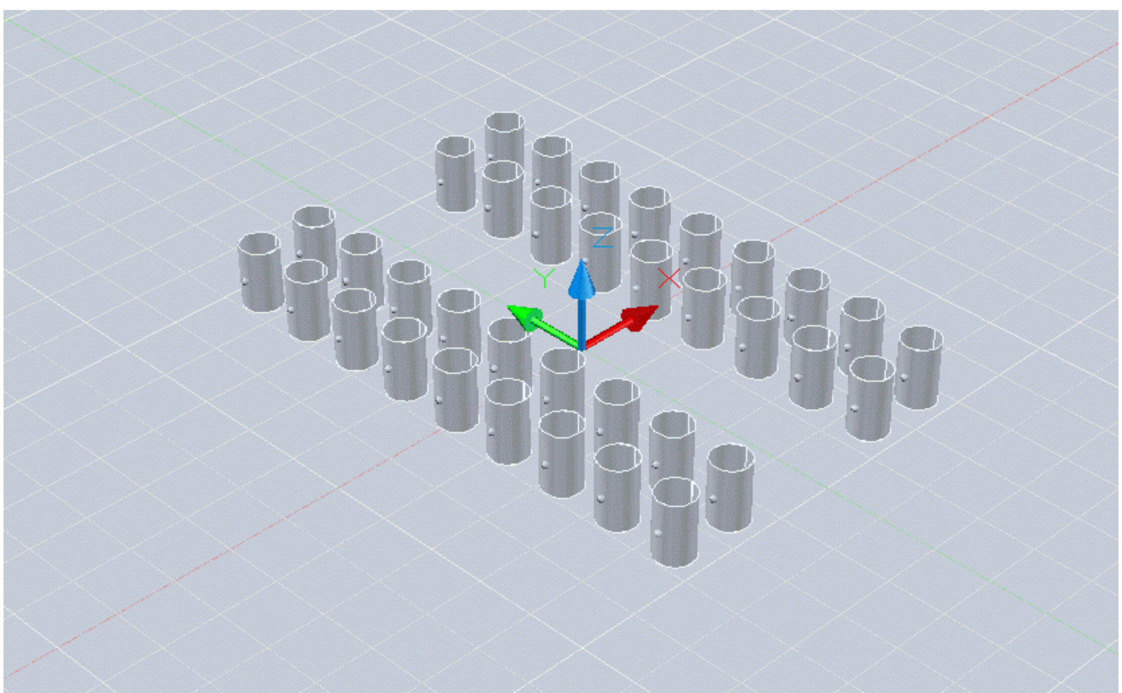

Figure 1 – Example TN-68 Cask Storage Configuration (Based on SAR Figure 4.10-1)

The staff determined that the view factor of the emitting cask in Figure 1 to the environment, with the concrete storage pads and access road conservatively considered to have no communication with the cask surface, was approximately 0.33. The applicant's original value of 0.62 for the view factor was considered to be non-conservative.

The staff requested that the applicant submit a calculation of the view factor in order to more appropriately capture the actual behavior of radiative exchange between a cask in an array of casks situated on a storage pad. The applicant submitted a view factor calculation that utilized the radiosity method of the ANSYS FEA code, and completed a sensitivity study of the effect of variations of the view factor on the thermal performance of the cask. The applicant documented this effort in Sections 4.10.1.1 and 4.10.1.3 of the SAR.

The staff reviewed the ANSYS model that was submitted by the applicant and found some minor errors related to the mirror plane and the resolution of the view factor calculation in the submitted model. The staff discussed the issues with the applicant, and, by letter dated March 16, 2007, the applicant provided a revised model that the staff found acceptable.

The applicant should use the modeling approach accepted by the staff for any future calculations related to the exchange of radiation between the cask and the surrounding environment for the TN-68 cask design. The applicant must consider the geometry of the cask storage pad when evaluating site specific application of the TN-68 cask design at a storage facility, since their view factor calculation method is dependent on the pad geometry.

Section 4.10.1.4 of the SAR describes the potential degradation of the neutron shield due to temperatures potentially exceeding the service temperature limit of 300ºF. The staff finds that the uncertainties related to the potential increase in thermal conductivity of the neutron shield material due to exposure to temperatures in excess of the recommended service temperatures could result in degradation exceeding that described in this section. The staff finds that additional degradation of the neutron shield is acceptable given the thermal margin demonstrated in the staff's confirmatory analyses, as well as the practices of cask users under approved 10 CFR Part 20 radiation protection programs.

4.5.4 Conclusion

The staff accepts the applicant's thermal analysis for storage of fuel as stated in Section 4.1 above, within the limits described in the Technical Specifications.

4.6 Evaluation Findings

F4.1 The staff finds that the thermal SSCs important to safety are described in sufficient detail in the SAR to enable an evaluation of their effectiveness. Based on the applicant's analyses, there is reasonable assurance that the system is designed with a heat removal capability consistent with its importance to safety. The staff also finds that there is reasonable assurance that analyses of the systems demonstrate that the applicable design and acceptance criteria have been satisfied for the storage of the authorized fuel assemblies.

F4.2 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, off-normal, and accident conditions were determined correctly.

F4.3 The staff has reasonable assurance that the system provides adequate heat removal capability without active cooling systems.

F4.4 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.

F4.5 The staff finds that the thermal design of the system 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 system 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.

5.0 SHIELDING EVALUATION

The shielding review evaluated the capability of the TN-68 Dry Storage Cask to provide adequate protection against direct radiation from its contents. The review considered dose rate calculations from both neutron and photon radiation at locations near the storage casks and at specific distances from a single cask and a representative array of storage casks. 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, 72.106(b), 72.212(b), and 72.236(d). An assessment of compliance with the dose limits in 10 CFR Part 72 for members of the public from a generic ISFSI is discussed in Section 10 (Radiation Protection) of this SER. Section 10 also includes an estimate of the dose to workers for the loading and storage operations of the TN-68 Dry Storage Cask. This application was also reviewed to determine whether the TN-68 Dry Storage Cask fulfills the acceptance criteria listed in Section 5 of NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems.”

5.1 Shielding Design Description

5.1.1 Design Criteria

The overall radiological protection design criteria are the regulatory requirements in 10 CFR Part 20 for occupational exposures and maintaining occupational exposures as-low-as-reasonably-achievable (ALARA), 10 CFR 72.104(a) and 10 CFR 72.106(b) (via 72.226(d) for certificate of compliance holders) for doses around ISFSIs. To show compliance with these regulations, the applicant evaluated the TN-68 cask loaded with design-basis spent fuel with the radiological characteristics shown in Table 2.1-2 of the SAR.

The analyses in the SAR provide reasonable assurance that the TN-68 can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). The applicant provided a radiation protection program in Section 5.2.3 of the TS, which limits the maximum dose rates on the top and sides of the storage cask, which are consistent with the dose calculations in Chapter 5. Based on the evaluation of an array of storage casks, the applicant has shown that the TN-68 storage casks can comply with the dose requirements in 10 CFR Part 72.

5.1.2 Design Features

The TN-68 cask is designed to provide both photon and neutron shielding. The principal components of the radial photon shielding are the 1.5-inch-thick steel inner shell, 6.0-inch-thick steel gamma shield and a 0.75-inch-thick steel outer shell. Photon shielding on the top of the storage cask is provided by a 4.5-inch-thick steel shield plate and a 5.0-inch-thick steel lid. Photon shielding on the bottom of the cask is provided by the 1.5-inch-thick steel base of the confinement vessel and an 8.25-inch-thick steel bottom plate. In addition, there is a provision for a 1-inch-thick steel shield ring above the radial neutron shield. However, as specified in the Technical Specification (TS) 5.2.3, this shield is only required when needed to meet the dose-rate limits in the TS.

Radial neutron shielding is provided by a borated polyester resin compound that is cast into long slender aluminum containers that surround the cask body. The radial neutron shield is

6.0-inches thick. Neutron shielding on the top of the storage cask is provided by 4.0 inches of polypropylene.

The TN-68 is designed to store up to 68 intact boiling-water reactor (BWR) fuel assemblies or up to 8 damaged BWR fuel assemblies with the balance of the storage locations storing intact BWR fuel assemblies. The eight damaged fuel assemblies can only be located in the outer two fuel assembly locations in each quadrant of the basket. The storage locations containing damaged fuel must be closed with specially designed end caps to contain the failed fuel on both the top and bottom of the basket.

5.2 Contents and Source Specification

5.2.1 Contents

A detailed description of the contents for the TN-68 is located in Section 2.1 of the SAR. The contents consist of up to 68 high burnup General Electric 7 × 7, through 10 × 10 fuel assemblies. The fuel assemblies may be surrounded by channel spacers. Additionally the storage cask can include up to eight damaged BWR fuel assemblies located in the outer two fuel assembly locations in each quadrant of the basket. Damaged fuel includes rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks, but the damaged assemblies may not be missing fuel pins or fuel rod segments. The extent of damage will be limited such that fuel must be handled by normal means, as defined in the TS definitions. Basket locations containing damaged fuel must be sealed with top and bottom end caps to confine any loose material.

Table 2.1-4 and figure 2.1-4 describe the enrichment, burnup, decay heat load and cooling times of the fuel assemblies, while Tables 5.2-1 and 5.2-1a describe the fuel assembly characteristics important for determining the source term. The maximum allowable burnup is 40 GWd/MTU for 7 × 7 fuel assemblies and 60 GWd/MTU for all other fuel assemblies. The minimum cooling time varies from 10 years to 12 years, based on the burnup and enrichment for 7 × 7 fuel assemblies, while the minimum cooling time for the 8 × 8, 9 × 9 and 10 × 10 fuel assemblies is 7 years, as shown in Figure 2.1-4.

5.2.2 Source Specification

The source specification is presented in Section 5.2 of the SAR. The gamma and neutron source terms were calculated with the SAS2H/ORIGEN-S module and the 44-group ENDF/B-V cross section set in the SCALE 4.4 computer code. The applicant generated a source term for both the 7 × 7 and 8 × 8, which are the two fuel assemblies with highest uranium loading of the fuel assemblies to be stored, to ensure that the bounding source term was used in the shielding evaluation. The characteristics of the two design basis source terms that the applicant generated are shown in Table 5-1. The source term generated for the 7 × 7 fuel assemblies used the same characteristics as during the original licensing. Since only the amount of uranium was changed, these characteristics are still appropriate.

The design basis burnup attributes for the 8 × 8 fuel assemblies were compared against several other sets of burnup characteristics to ensure that the design basis fuel assembly burnup characteristics are bounding. The applicant performed a simple one-dimensional analysis based on the sidewall characteristics of the TN-68 to determine both neutron and gamma dose rate for 14 variations on the burnup, minimum enrichment and cooling time. Dose rates at both the surface and 2 meters were determined. The burnup characteristics with the highest total dose rate were used as the design basis source term.

Table 5-1 Fuel Assembly Characteristics for Design Basis Source term Calculations

Specification	7 × 7	8 × 8
Burnup (GWd/MTU)	40	48
Cool time (yrs)	10	7
Enrichment (w/o)	3.3	2.6
Number of Water Rods	0	1
Specific Power (MW/assy)	5	6

5.2.2.1 Gamma Source

Table 5.2-8 of the application provides the SAS2H calculated gamma source terms for the active fuel region as well as the top and bottom regions of the fuel assembly. The hardware activation analysis considered the cobalt impurities in the assembly hardware. The quantity of impurities evaluated in the analysis are presented in Tables 5.2-2, 5.2-3, 5.3-2, and 5.3-2a. Although cobalt impurities can vary, the applicant's assumed values are reasonable and acceptable. To correct for changes in the neutron flux outside the fuel zone during irradiation, the masses of the materials in the bottom end fitting, top plenum, and top end fitting, were multiplied by scaling factors of 0.15, 0.2, and 0.1, respectively. These scaling factors are consistent with the values used in the original licensing basis and are considered to provide appropriate values.

5.2.2.2 Neutron Source

The SAS2H calculated neutron source term for the fuel assemblies is provided in Table 5-9 of the application. The applicant determined the design-basis neutron source term using the 8 × 8 fuel assemblies, because it has the highest burnup of the two design basis source terms. The neutron source term is more dependent on burnup than cooling time, since after cooling in the spent fuel pool most of the neutrons are due to decay of ^{244}Cm, which has an 18 year half-life.

5.2.2.3 Confirmatory Analyses

The staff reviewed the proposed contents and the hardware cobalt impurities provided by the applicant. The staff has reasonable assurance that the design basis gamma and neutron source terms are acceptable for the TN-68 storage cask shielding analysis. The staff also reviewed the neutron flux scaling factors for the hardware source terms and found them to be appropriate. The staff performed confirmatory calculations of the source terms for the specified fuel types, burnup conditions, and cooling times. The staff also used the SAS2H module of SCALE 4.4 with the 238-group cross section library. The staff's source term calculations were in agreement with the applicant's.

5.3 Shielding Model Specifications

The shielding analyses to determine dose rates around a single the TN-68 were performed with MCNP. MCNP is a three-dimensional Monte Carlo code for determining transport of neutrons and gammas. The applicant determined the dose rates using three-dimensional models of the TN-68 storage cask.

5.3.1 Shielding and Source Configuration

The applicant explicitly modeled the compartments for the fuel assemblies as well as the aluminum rails, steel structural supports and the channel fuel compartment. Each of the fuel assemblies were homogenized within their own fuel compartment. Each fuel assembly is divided into 14 axial regions, as shown in Table 5.2-6 of the application. The axial distribution of the gamma and neutron sources is assumed to follow the relative burnup profile derived from in-core data from five different fuel assemblies during operations. Outside of the fuel region the applicant modeled the steel shells and the borated resin explicitly. Dose rates were determined at distances ranging from the surface, one, and two meters from the side, top and bottom of the storage cask.

5.3.2 Material Properties

The composition and densities of the materials used in the shielding analysis are presented in Tables 5.1-1 and 5.3.3 through 5.3-5. The homogenized fuel assembly regions account for the uranium dioxide, cladding and spacers, but the applicant does not take credit for the shielding effectiveness of the fuel assembly channels, but does include their contribution to the source term.

The materials used in modeling the storage cask, basket, and fuel assemblies were reviewed and accepted by the staff. The material compositions and densities used were appropriate and provide reasonable assurance that the material densities were adequately modeled.

5.4 Shielding Analyses Results

5.4.1 Computer Programs

The applicant's shielding analysis was performed with MCNP and is presented in Section 5.4 of the SAR. MCNP uses essentially point wise cross sections to determine the gamma and neutron dose rates on the surface, one and two meters from the storage cask. Additionally the applicant determined these dose rates with and without the auxiliary shield ring.

5.4.2 Flux-to-Dose-Rate Conversion

The applicant used the ANSI/ANS Standard 6.1.1-1977 flux-to-dose conversion factors to calculate dose rates. The values listed in this standard are provided in Table 5.3-1 of the SAR.

5.4.3 Normal Conditions

Tables 5.4-1 through 5.4-5 of the SAR present the maximum and average dose rates for both normal and off-normal conditions during storage. Based on the assumptions used in the analyses, the source term and cooling time of the design basis contents, and the administrative programs established in Section 5.4 of the technical specifications; the staff has reasonable assurance that the user will be able to maintain normal-condition doses ALARA and meet the dose requirements of 10 CFR Part 72.

The calculated dose rates for the storage cask shown in Table 5.4-1 and 5.4-2 of the SAR are generally dominated by the gamma component. The maximum radial dose rate on the surface of the storage cask at the fuel midplane is 147 mrem/hr. The peak dose rate around the storage cask is 2010 mrem/hr on the surface of the storage cask just above the trunnion. The average axial dose rate in the location above the trunnion to the top of the storage cask is 1460 mrem/hr. The maximum dose rates on the top and bottom of the fuel are 361 and 1850 mrem/hr, respectively. The top and bottom dose rates were determined using ring detectors of varying radii to determine five different dose rates as well as an average top and bottom dose rates. The maximum dose rates on both the top and bottom occurred in the smallest ring of 25 cm. The fluxes causing these dose rates are utilized in Chapter 10 to determine off-site doses.

5.4.4 Off-Normal Conditions

Chapter 11 of the SAR does not identify any off-normal event that degrades the shielding effectiveness of the TN-68 storage beyond normal conditions. For loading and unloading operations, the stresses on the storage cask shell are demonstrated in Chapter 3 of the SAR to be within ASME Code stress limits. Therefore, there is no permanent deformation of the shell. Thus, there is no potential for breach of the confinement pressure boundary or release of radioactive material. Similarly, the stress levels for the TN-68 during the extreme ambient conditions are demonstrated in Chapter 3 of the SAR to be within the ASME Code stress limits. The dose rates around the storage cask are the same for both normal and off-normal conditions.

5.4.5 Accident Conditions

For accident conditions, no event is postulated that can impact the confinement boundary of the TN-68. Accidents such as tornado missiles are not postulated to affect the effectiveness of the confinement boundary of the storage cask. The only difference between the shielding models for normal/off-normal conditions and accident conditions is the loss of neutron shield due to a hypothetical fire accident. The applicant assumed that the neutron shielding will be lost or ineffective after a fire accident

The applicant estimated the dose rates for accident conditions to be 4300 mrem/hr at the radial surface, 1680 mrem/hr at a distance of 1 meter from the surface of the storage cask. Additionally, in Section 11.2.5.3, the applicant assumed that the accident conditions dose rate 100 meters from the storage cask drops off similarly to the normal conditions dose rate, yielding a dose rate of 0.32 mrem/hr. Based on the radial dose rate 100 meters from the storage cask during accident conditions, the storage cask will be able to meet the dose rate criterion of 5 rem (total effective dose equivalent) in 10 CFR 72.106, for a postulated accident that last 30 days and an individual is 100 meters from the accident for 24 hours per day over those 30 days.

5.4.6 Occupational Exposures

The analysis in the SAR used the design basis 8 × 8 fuel assembly to estimate occupational exposures for the TN-68 storage cask. Chapter 10 of the SAR presents the estimated occupational exposures that are based on dose rate calculations in Chapter 5 of the SAR. The staff's evaluation of the occupational exposures is in Chapter 10 of this SER.

5.4.7 Confirmatory Calculations

The staff performed confirmatory analyses of selected dose rates using the SAS4 module of the SCALE system. The main differences between the staff's and the applicant's calculations are the fuel assembly homogenization and the treatment of localized areas such as the trunnions. The staff homogenized all the fuel assemblies into one cylindrical region and did not model the steel of the basket. The staff also did not model localized areas such as the trunnions and vent regions. The staff's evaluation is based on the design features and specifications presented in the SAR. Limiting fuel assembly characteristics as well as the burnup and cooling time were similar to the applicant's. The staff's calculated dose rates were in general agreement with the SAR values.

5.5 Evaluation Findings

F5.1 Chapters 2, 5, and 10 of the SAR sufficiently describe the radiation protection design bases and design criteria for the structures, systems, and components important to safety.

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

F5.3 The TN-68 Dry Storage Cask is designed to provide redundant sealing of the confinement system.

F5.4 The staff concludes that the design of the radiation protection system of the TN-68 is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the TN-68 will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

6.0 CRITICALITY EVALUATION

The staff reviewed the proposed amendment to the TN-68 Dry Storage 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-68 cask meet the regulatory requirements of 10 CFR Part 72. The factors affecting criticality safety proposed in the amendment include higher enrichment fuel and damaged fuel added to the TN-68 cask. Both the revised SAR and the applicant's responses to RAI questions were used to determine the acceptability of the proposed amendment.

6.1 Criticality Design Criteria and Features

The design criterion for criticality safety is that the effective neutron multiplication factor, k_{eff}, remains below a value of 0.95 for all postulated arrangements of fuel within the cask under normal, off-normal and accident conditions, including statistical biases and uncertainties. The TN-68 cask relies primarily on the basket geometry and the fixed neutron poisons in the basket to maintain subcriticality of the cask. The maximum lattice-average enrichment of U-235 varies with the basket type, which is determined by the 10B areal density in the fixed neutron absorber and can vary between 27 to 63 mg B10/cm^2. Each level of areal density corresponds to a different basket type and an associated fuel enrichment limit. For intact fuel the limit is based on the lattice-averaged enrichment, while for damaged fuel (as added under the proposed amendment) is based on the maximum pellet enrichment.

6.2 Fuel Specification

No new fuel assembly types were added in this amendment. However, enrichment was increased to a maximum of 4.70 wt% U-235 for both intact assembly lattice average enrichment and for damaged assembly peak pellet enrichment.

6.3 Model Specification

6.3.1 Configuration

The TN-68 cask was modeled using the appropriate geometry options in KENO V.a of the CSAS25 module in SCALE-4.4. A KENO model was generated in the analysis to determine the initial enrichment for intact and damaged fuel assemblies as a function of fixed neutron poison loading. The active fuel region was modeled explicitly except for axially, which was modeled as infinite. The applicant did not take credit for the burnup of the fuel or for burnable absorber in the fuel. The fixed poison modeled in the calculation is modeled as aluminum and boron, which is adequate to represent the borated aluminum alloy or boron carbide/aluminum composites.

The applicant performed a number of parametric cases similar in procedure and underlying assumptions to the originally approved license application to determine the most reactive model for normal conditions and used this calculation model to determine the maximum allowable enrichment as a function of the fixed neutron poison loading. Staff reviewed the applicant's models and agrees that they are consistent with the description of the cask and contents given in the proposed amendment. The staff also reviewed the applicant's methods, calculations, and results and agrees that the most reactive combination of cask parameters and dimensional tolerances were incorporated into the calculation models.

The most reactive configuration for intact fuel assemblies was based on the most reactive assembly type (GE 10x10) and found to be off-center "inward" positioning of the fuel. The maximum initial enrichment for intact assemblies was found to be 4.70 wt% U-235. Reconstituted fuel assemblies where the fuel pins are replaced with non-fuel pins are also considered intact assemblies provided they displace an equivalent amount of moderator.

The TN-68 cask was also analyzed with up to 8 damaged assemblies located in the peripheral locations using the limiting intact fuel configuration determined by the applicant. This configuration consists of a 0.200-inch thick 54 mg B10/cm^2 poison plate with an initial enrichment of 4.50 wt% U-235.

As demonstrated in the structural analysis, undamaged fuel remains intact under accident accelerations; however, damaged fuel is assumed to sustain some further damage. This was analyzed by the applicant by modeling both single-ended and double-ended shearing of fuel rods with moderator intrusion in order to bound the credible criticality effect of any fuel reconfiguration. The single-ended shear case indicated that it is bounded by the axial rod shift case. The double-ended shear case showed a small positive change in reactivity.

The highest calculated k_{eff} for the damaged assembly calculations occurred for the GE 10 x 10 damaged fuel assembly under double-ended shearing conditions with an initial enrichment of 3.95 wt% U-235 and a poison loading of 31.5 mg B10/cm^2. In all instances the resulting k_{eff} remained below the USL.

The staff performed confirmatory calculation analyses for the intact and damaged fuel configurations using the information provided in the revised SAR and Technical Specifications and the results were comparable with those of the applicant.

6.3.2 Material Properties

All materials used in the TN-68 criticality analysis are from the Oak Ridge National Laboratory (ORNL) SCALE code package which contains a standard material data library for common elements, compounds, and mixtures. The boron credit used in the analysis has not changed from the previously approved certificate.

6.4 Criticality Analysis

6.4.1 Computer Programs

The applicant utilized the CSAS25 modules in SCALE-4.4 computer codes and the accompanying 27-group cross-sectional library for the TN-68 cask amendment analysis and the benchmark calculations.

The staff performed confirmatory analyses using the CSAS25 modules in the SCALE V.a computer codes with the 27-group cross-sectional library. The SCALE group of 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 the k_{eff} of the TN-68 cask with the proposed amended changes 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.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff concludes that the proposed changes in the amendment to the TN-68 cask will remain subcritical with an adequate margin of safety under all credible normal, off-normal, and accident conditions.

6.4.3 Benchmark Comparisons

The applicant performed benchmark comparisons on selected critical experiments similar to those performed in the original approved license. The staff reviewed the applicant's method for determining the USL and found it to be acceptable and conservative.

6.5 Supplemental Information

The allowable fuel types that may be loaded into the TN-68 cask are largely unchanged with the exception of GE 7 x 7 fuel assemblies 3, 3A, and 3B which have increased the maximum uranium per assembly from 0.1896 MTU per assembly to 0.1923 MTU per assembly.

6.6 Evaluation Findings

F6.1 Based on the staff's review of the amended SAR, the applicant's responses to the RAI questions regarding criticality safety, and the staff's own confirmatory analyses, the staff concludes that the proposed changes to the TN-68 are acceptable.

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. SCALE4.4, “A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” Oak Ridge National Laboratory, March 1997.

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

7.0 CONFINEMENT EVALUATION

The confinement review for the TN-68 Dry Storage Cask, Amendment 1, was based on the application dated January 14, 2005, as supplemented. TN has not changed any of the physical containment boundary nor has it changed the leak testing values. This amendment changes the source term to include a higher burnup to 60 GWD/MTU, a shorter cooling time (from 10 years to 7 years), a higher heat load per canister (from 21.8 kW to 30 kW), a maximum heat load per assembly of 0.441 kW, a higher maximum enrichment (from 3.7 % wt. U-235 to 4.7 % wt. U-235) and up to 8 damaged fuel assemblies. The amount of uranium per assembly and fuel array type has not changed.

Based on the changes described above, the only elements of the confinement evaluation that are directly affected by this amendment are the source term that is released, its associated dose effect and cask maximum pressure.

7.1 Nuclides with Potential for Release

The staff reviewed the applicants methodology for determining source term releases and found them to be acceptable and in accordance with current regulatory guidance, including ISG-5, Revision 1. The applicants's design basis source term is based upon an assembly heat load limit of 0.441 kW (30 kW cask heat limit divided by 68 assemblies) which effectively limits the amount of radionuclides from the high-burnup fuel. To ensure that the correct fuel is loaded the applicant utilizes a flow chart, Figure 2.1.1-2 in the technical specifications, which, among other parameters, ensures that the assembly heat load limit is maintained. The staff reformed a bounding source term evaluation and determined that the releases still resulted in a reference leak rate of 10^{-5} standard cubic centimeters per second.

For damaged fuel, the applicant considered it the same as intact fuel for the purpose of determining the releasable source term. The staff agrees with this approach since it is conservative and produces a releaseable source term with a higher radiological activity. From the staff's perspective, damaged fuel is depressurized and lacks the energy to aerosolize any of the actinide fines in the fuel. In addition, damaged fuel fission gases are released in the spent fuel pool or reactor long before the damaged fuel is loaded into the cask. Finally, the volatile radionuclides have been reduced or removed via the vacuum drying operation, which heats them into the gaseous phase and removes them along with the water vapor. Consequently, using the assumption that all fuel is treated as intact bounds the release from the 8 damaged fuel assemblies per cask.

7.2 Confinement Analyses

The staff reviewed the applicants pressurization calculations (now located in the thermal section) and noted that the cask pressure increased due to an increase in cask temperature (resulting from a higher heat load) and increase in fission gases (resulting from a higher burnup). The maximum off-normal pressure increased from 20 psig to 21.6 psig, the maximum pressure under the fire accident scenario increased from 48 psig to 71.7 psig , and the maximum pressure under the buried cask accident scenario increased from 63 psig to 96.7 psig. All values remain below the cask design pressure of 100 psig.

7.3 Evaluation Findings

F7.1 The staff concludes that the TN-68 Spent Fuel Dry Storage Cask, as amended, meets the regulatory requirements for a confinement system contained in 10 CFR Part 72.

8.0 OPERATING PROCEDURES

The review of the technical bases for the operating procedures is to ensure that the applicant's SAR presents acceptable operating sequences, guidance, and generic procedures for key operations. The procedures for the TN-68 Dry Storage Cask, as described in Chapter 8 of the SAR, have been amended to conform with the revised contents of the cask.

8.1 Cask Loading

Several minor changes to the cask loading procedures were made as a result of this amendment request. These changes are reflected in Table 8.1-1, "Sequence of Operations-Loading." Table 8.1-1 was revised to add or revise steps as necessary by the amendment request.

8.1.1 Fuel Specifications

The applicant revised the procedures described in Table 8.1-1 of the SAR to include steps for verification of the fuel basket type and neutron absorber, and verification of the fuel assembly type according to the Technical Specification (TS) 2.1.1. The applicant also included steps for users to verify the condition of each fuel assembly and for classifying these as either intact or damaged. It outlines the specific locations where damaged fuel assemblies may be loaded, and verifies that bottom end caps are installed in these damaged fuel compartments.

8.1.2 ALARA

Additional ALARA practices beyond those previously approved for the TN-68 Dry Storage Cask are discussed in Chapter 10 of this SER. No major changes to these practices were requested in the amendment application.

8.1.3 Draining, Drying, Filling, and Pressurization

Section 8.1.3 and Table 8.1-1 of the SAR clearly describe draining, drying, filling, and pressurization procedures for the TN-68 Dry Storage Cask that will provide reasonable assurance that the cask is properly dried and backfilled, and the fuel is stored in an inert atmosphere. The applicant proposed revised TS 3.1.1 and 3.1.2 to specify new time limits for completion of draining and backfill activities and these are discussed in Chapter 12 of this SER. In addition, minor changes to the procedures for draining, drying, filling, and pressurization were requested in the amendment application and the staff has found them acceptable.

The proposed operating procedures call for the vacuum pump to be isolated by a valve but to run continuously during the pressure rise test. The staff requested that the applicant revise the proposed configuration used during vacuum drying operations to ensure that the pressure rise test would not be affected if the cask isolation valve leaks. A leaking isolation valve could result in a false measurement of negligible pressure rise due to the vacuum pump removing any water vapor from the cask. The applicant proposed an alternate option for the vacuum drying configuration that ensures that a leaking isolation valve could not cause a false compliance with the pressure rise requirements.

8.1.4 Welding and Sealing

The applicant did not request major changes to the sealing operations of the TN-68 Dry Storage Cask, and these remain similar to those previously approved by the staff. The staff reviewed minor changes to these procedures and found them acceptable.

8.2 Cask Handling and Storage Operations

All handling and transportation events applicable to moving the TN-68 Dry Storage Cask to the storage location are similar to those previously reviewed by the staff in the original approval of CoC No. 1027. Minor changes to the procedures were made to incorporate the reduction of spacing between casks in the storage pad. The staff reviewed these and found them acceptable.

8.3 Cask Unloading

No changes to the cask unloading procedures for the TN-69 Dry Storage Cask were requested in the amendment application.

8.4 Evaluation Findings

F8.1 The staff concludes that the generic procedures and guidance for the operation of the TN-68 Dry Storage Cask, as amended, is in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The purpose of the review of the acceptance tests and maintenance procedures for Amendment 1 to the TN-68 Dry Storage Cask is to ensure that the changes requested by the applicant meet the requirements of 10 CFR Part 72. The changes requested by the applicant in its amendment request include the addition of new types of neutron absorber materials for use in the TN-68 cask basket. Only those sections revised by this amendment request are discussed below.

9.1 Acceptance Tests

9.1.1 Structural/Pressure Tests

The applicant made minor revisions to the SAR to state the specific acceptance criteria used for the shield shell and lid shield plate. The staff has reviewed these changes and found them acceptable.

9.1.2 Neutron Absorber Tests

The amendment request includes the use of Boral® and boron carbide/aluminum metal matrix composites (MMC) as neutron absorber materials for the TN-68 cask basket.

Sections 9.1.7.1 and 9.1.7.2 specify a direct chill (DC) or permanent mold casting process for manufacturing selected neutron absorber materials. The arguments made in the SAR for qualification of material produced through this process are regarded as acceptable. The acceptance testing specified for any lot of materials to be used in a service application and produced under this process are acceptable and adequately supplement the specified qualification testing.

Sections 9.4.2 and 9.4.3.5 specify various acceptance tests for all neutron absorber materials, which, in the criticality calculations, are given 90% credit for the boron added to these. The applicant states that the specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence. The requirements of these sections ensure quality and safety of the materials proposed for use under this amendment.

The staff requested that selected portions of the acceptance tests for the neutron absorber materials be incorporated by reference into the technical specifications. The specific sections referenced in the technical specifications are discussed in Section 12 of this SER. These sections have been reviewed and found to be consistent with current accepted practices for neutron absorber materials used in storage and transport cask systems.

9.2 Evaluation Findings

F9.1 The staff concludes that the acceptance tests for the TN-68 Dry Storage Cask, as amended, are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the acceptance tests provides reasonable assurance that the neutron absorber materials will allow safe storage of spent fuel throughout its licensed or certified term.

10.0 RADIATION PROTECTION EVALUATION

The staff reviewed the radiation protection design features, design criteria, and the operating procedures for the TN-68 Dry Storage Cask to ensure that its use will meet the regulatory dose requirements of 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). The application was also reviewed to determine whether the TN-68 fulfills the acceptance criteria listed in Section 10 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."

10.1 Radiation Protection Design Criteria and Design Features

10.1.1 Design Criteria

The radiological protection design criteria are the limits and requirements of 10 CFR Part 20, and 10 CFR Part 72. The technical specifications also establish an administrative program which controls the dose rates around the storage cask, which are based on the calculated dose rate values used to determine occupational and off-site exposures. The technical specifications also establish exterior contamination limits for the DSC to keep contamination levels below 1,000 dpm/100 cm^2 for beta and gamma sources, and 20 dpm/100 cm^2 for alpha sources.

10.1.2 Design Features

Chapters 5 and 10 of the SAR define the radiological protection design features which provide radiation protection to operational personnel and members of the public. The radiation protection design features include the following:

- Thick walls of the TN-68 cask body provide shielding from gamma radiation.
- The cask is surrounded by a borated resin-filled layer that provides shielding from the neutron radiation.
- The confinement system includes double metallic seals and a pressure monitoring system to detect leaks in the confinement boundary and preclude atmospheric releases of radioactive material. The confinement system is designed to maintain confinement of radioactive materials during normal, off-normal, hypothetical accident conditions, including severe natural phenomena.
- The cask body consists of smooth surfaces to facilitate decontamination prior to transfer to the ISFSI, minimize the time spent decontaminating a cask and reduce the quantity of radioactive waste generated during decontamination.

The staff evaluated the radiation protection design features and design criteria for the TN-68 and found them acceptable. The SAR analysis provides reasonable assurance that use of the TN-68 storage cask can meet the regulatory requirements in 10 CFR Part 20 and 10 CFR Part 72. Sections 5, 7, and 8 of this SER discuss staff's evaluations of the shielding features, confinement systems, and operating procedures, respectively. Section 11 of this SER discusses staff evaluations of the capability of the shielding and confinement features during off-normal and accident conditions.

10.2 Occupational Exposures

Chapter 8 of the application discusses the generic operating procedures that general licensees will use for fuel loading, cask operations, transferring the storage cask to the pad and fuel unloading. Table 10.3-1 of the SAR provides the estimated number of personnel, time, tasks involved and the estimated dose to load a TN-68 storage cask. The estimated occupational doses are based on the dose rates from the design basis 8 × 8 fuel determined in Chapter 5 of the application. The dose estimates indicate that the total occupational dose in loading a single canister with design basis fuel is approximately 4 person-rem. Additionally the applicant included the measured operational doses from previous loadings of the TN-68 storage casks at Peach Bottom Atomic Power Station. Although the cask heat load was approximately one half the heat load requested in this amendment, the operational doses were all a factor of 10 less than the doses estimated in Table 10.3-1. Additionally, Table 10.3-2 estimates the annual exposure from high burnup fuel for ISFSI maintenance operations to be less than 400 mrem/yr for any task. .

10.3 Public Exposures from Normal and Off-Normal Conditions

Section 10.3 of the SAR presents the calculated direct radiation doses at distances from 1 meter to 600 meters from a single storage cask and a sample 20-cask array configuration both loaded with design basis fuel. Table 10.2-8 and figures 10.2-6 and 10.2-7 provide the estimated dose as a function of distance for a sample 2 × 10 array of storage casks. The table shows that the regulatory design limit of 25 mrem/yr can be achieved at approximately 500 meters from the array, assuming 100 percent occupancy for 365 days.

The staff evaluated the public dose estimates during normal and off-normal conditions and found them acceptable. The primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions is from direct radiation (including skyshine). The cask is sealed by double o-rings and the confinement function is not affected by normal or off-normal conditions. A discussion of the staff’s evaluation and confirmatory analysis of the shielding calculations is presented in Section 5 of this SER.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by each general licensee. The general licensee using the TN-68 storage casks must perform a site-specific evaluation, as required by 10 CFR 72.212(b) to demonstrate compliance with 10 CFR 72.104(a). The actual doses to an individual beyond the controlled area boundary depend on several site-specific conditions such as fuel characteristics, cask-array configurations, topography, demographics, and use of engineered features (e.g., berm). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities such as reactor operations. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of each applicant for a site license.

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 in 10 CFR Part 20, Subpart D, by evaluations and measurements.

10.4 Public Exposures from Design-Basis Accidents and Natural Phenomena Events

Table 5.1-2 of the SAR summarizes the calculated dose rates to individuals beyond the controlled area for the accident conditions listed in Chapter 11 of the SAR. The confinement function of the storage is not affected by design-basis accidents or natural phenomena thus there is no additional release of contents.

The SAR analysis indicates the worst-case shielding consequences results in a dose at the controlled area boundary that will meet the regulatory requirements of 10 CFR 72.106(b). Chapter 11 of the SAR discusses corrective actions for each design-basis accident. The staff evaluated the public dose estimates from direct radiation under accident conditions and natural phenomena events and found them acceptable. A discussion of the staff's evaluation and any confirmatory analysis of the shielding and confinement analysis is presented in Sections 5 and 7 of this SER. A discussion of the staff's evaluation of the accident conditions and recovery actions are presented in Chapter 11 of this SER. The staff has reasonable assurance that the effects of direct radiation from bounding design basis accidents and natural phenomena will be below the regulatory limits in 10 CFR 72.106(b).

10.5 ALARA

Chapters 5, 7, and 10 of the SAR present evidence that the TN-68 radiation protection design features and design criteria address ALARA requirements, consistent with 10 CFR Part 20 and Regulatory Guides 8.8[1] and 8.10[2]. Each general licensee will apply its existing site-specific ALARA policies, procedures, and practices for cask operations to ensure that personnel exposure requirements in 10 CFR Part 20 are met. Each plant will have to consider the use of this canister with respect to their particular ALARA implementation philosophy.

The staff evaluated the ALARA assessment of the TN-68 storage cask and found it acceptable. Section 8 of this SER discusses the staff's evaluation of the operating procedures for the TN-68. Operational ALARA policies, procedures, and practices are the responsibility of the site licensee as required by 10 CFR Part 20. In addition, the technical specifications establish an administrative program which controls dose limits dose and surface contamination limits to ensure that occupational exposures are maintained ALARA.

10.6 Evaluation Findings

F10.1 The SAR sufficiently describes the radiation protection design bases and design criteria for the structures, systems, and components important to safety.

F10.2 The TN-68 storage cask provides radiation shielding and confinement features that are sufficient to meet the requirements of 10 CFR 72.104, and 10 CFR 72.106.

F10.3 The occupational radiation exposures provided in the SAR satisfy the limits in 10 CFR Part 20 and meet the objective of maintaining exposures ALARA.

F10.4 The staff concludes that the design of the radiation protection system of the TN-68 storage cask is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the TN-68 will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

10.7 References

1. U.S. Nuclear Regulatory Commission, "Information Relevant to Ensuring that Occupational Radiation Exposures Will Be As Low As Reasonably Achievable," Regulatory Guide 8.8, Revision 3, June 1978.

2. U. S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable," Regulatory Guide 8.10, Revision 1-R, May 1977.

11.0 ACCIDENT ANALYSES

11.1 Staff Evaluation

The applicant made several revisions to selected sections of the accident analyses evaluation. The staff evaluated these changes and found that the changes were made to ensure consistency with the changes requested for Chapter 4, Chapter 5, Chapter 7, and Chapter 10 of the SAR. The technical bases for the changes requested in this chapter are discussed individually in the above mentioned chapters.

11.2 Evaluation Findings

F11.1 The staff concludes that the accident design criteria for the TN-68 Dry Storage Cask, as amended, are in compliance with 10 CFR Part 72 and the accident design and acceptance criteria have been satisfied.

12.0 CONDITIONS FOR CASK USE - TECHNICAL SPECIFICATIONS

The purpose of the review of the technical specifications for the cask is to determine whether the applicant has assigned specific controls to ensure that the design basis of the cask system is maintained during loading, storage, and unloading operations.

12.1 Conditions for Use

The applicant requested an additional condition in the Certificate of Compliance (CoC) authorizing the use of previously approved amendments of the certificate. The applicant's states that the purpose of the condition is to ensure that previous revisions of the certificate will not be superceded by this amendment and cask users may load under the provisions of the original certificate (i.e., Amendment "0"). The staff has no objections with the applicant's request and finds the addition of the condition acceptable.

The applicant also requested the removal of Condition No. 9 of the CoC. Condition No. 9 outlined the requirements for alternate composite materials used as neutron absorbers in the TN-68 cask basket. The requirements for new neutron absorber materials in the TN-68 cask basket have been incorporated in Section 4.1.1 of the Appendix 1 to the CoC (Technical Specifications). The staff finds the proposed deletion of Condition No. 9 of the CoC acceptable.

12.2 Technical Specifications

The technical specifications have incorporated, by reference, SAR Sections 9.1.7.1, 9.1.7.2, 9.1.7.3, as well as 9.4.2, 9.4.3.5, and 9.4.4.3. These sections describe, in part, the important-to-safety processing steps that are taken for the absorber materials to be used in the TN-68 Cask under Amendment No. 1. The staff requested the inclusion of these sections in the technical specifications. The staff believes that the neutron absorber materials are important-to-safety components of the cask system, and changes to their manufacturing process and qualification testing should be approved by NRC.

The staff also requested that the applicant include a condition in the technical specification restricting the storage of damaged fuel to assemblies with burnups lower than 45 GWd/MTU. The staff's rationale for this restriction is discussed in Section 3.0 of this SER.

The applicant requested additional revisions to the Technical Specifications as discussed in Part B of its amendment application. The staff reviewed these changes and has found them acceptable. The staff evaluated the changes made to the technical specification bases corresponding to these proposed changes and found that they are consistent with those made to the technical specifications.

12.3 Evaluation Findings

F12.1 The staff concludes that the conditions for use of the TN-68 Dry Storage Cask identify necessary technical specifications to satisfy 10 CFR Part 72 and that the applicant acceptance criteria have been satisfied. The amended technical specifications provide reasonable assurance that the cask will provide for safe storage of spent fuel.

13.0 QUALITY ASSURANCE

The purpose of this review and evaluation is to determine whether TN has a quality assurance program that complies with the requirements of 10 CFR Part 72, Subpart G. The staff has previously reviewed and accepted the TN quality assurance program in the TN-68 Dry Storage Cask FSAR.

14.0 DECOMMISSIONING

The decommissioning evaluation was previously reviewed and approved in the TN-68 Dry Storage Cask FSAR. There were no significant changes proposed by the applicant under this amendment request.

CONCLUSIONS

The staff performed a detailed safety evaluation of the proposed CoC amendment request and found that the proposed amendment does not reduce the ability of the TN-68 Dry Storage Cask to meet 10 CFR Part 72. Based on the statements and representations contained in the applicant's SAR, and the conditions of the CoC, the staff concludes that Amendment No. 1 to the TN-68 Dry Storage Cask meets the requirements of 10 CFR Part 72.

Issued with Certificate of Compliance No. 1027, Amendment 1,
on _______________________.