TRANSNUCLEAR
AN AREVA COMPANY

December 8, 2006
E-24269

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: Revision 4 to Application for Amendment 1 to TN-68 CoC 72-1027
Docket 72-1027, TAC L23802

Gentlemen:

Transnuclear, Inc. herewith submits Revision 4 to its application for Amendment 1 to TN-68 CoC 72-1027. Based on recent discussions with the NRC staff, this revision changes the Technical Specifications and Updated Final Safety Analysis Report to clarify that only damaged fuel with an average bundle burnup less than or equal to 45 GWd/MTU may be stored in the TN-68 cask.

Should you or your staff require additional information, please do not hesitate to contact me at 410-910-6930 or Mr. Don Shaw at 410-910-6878.

Sincerely,

Robert Grubb
Senior Vice President - Engineering

cc: Mr. Jose Cuadrado (NRC SFST), with eight printed copies of Enclosures 1 and 2 and one compact disc containing Enclosures 1 and 2

cc: (without enclosures)

Jeff Gagne, Transnuclear
David Shortes, Exelon

Enclosures:

1. List of Enclosed Technical Specification and UFSAR pages
2. Replacement Technical Specifications and UFSAR pages

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NMSS01

List of Enclosed Technical Specification and UFSAR pages

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Technical Specification Pages:

- 2.0-1, **changed** regarding high burnup damaged fuel

UFSAR Pages:

- 1.2-4, Rev 1, unchanged (front side of page 1.2-5)
- 1.2-5, Rev 4, **changed** regarding high burnup damaged fuel (back side of page 1.2-4)

- Table of Contents page with List of Tables and List of Figures, Rev 0, unchanged (front side of page 2.1-1)
- 2.1-1, Rev 4, **changed** regarding high burnup damaged fuel (back side of page showing Table of Contents page with List of Tables and List of Figures)

- 5.2-1, Rev 0, unchanged (front side of page 5.2-2)
- 5.2-2, Rev 4, **changed** regarding high burnup damaged fuel (back side of page 5.2-1)

- 6B.1-1, Rev 4, **changed** regarding high burnup damaged fuel (front side of page 6B.2-1)
- 6B.2-1, Rev 4, **changed** regarding high burnup damaged fuel (back side of page 6B.1-1)

- 6B.3-1, Rev 4, **changed** regarding high burnup damaged fuel (front side of page 6B.3-2)
- 6B.3-2, Rev 4, **changed** regarding high burnup damaged fuel (back side of page 6B.3-1)

- 6B.3-3, Rev 4, **new page**, regarding high burnup damaged fuel (front side of page 6B.3-4)
- 6B.3-4, Rev 4, **new page**, regarding high burnup damaged fuel (back side of page 6B.3-3)

- 6B.3-5, Rev 4, **changed and re-paginated**, regarding high burnup damaged fuel (front side of page 6B.4-1)
- 6B.4-1, Rev 0, unchanged (back side of page 6B.3-5)

- 6B.10-2, Rev 3, unchanged (front side of 6B.10-3)
- 6B.10-3, Rev 4, **changed** regarding high burnup damaged fuel (back side of page 6B.10-2)

- 7.3-1, Rev 4, **changed** regarding high burnup damaged fuel (front side of page 7.3-1)
- 7.3-1a, Rev 1, unchanged (back side of page 7.3-1)

Replacement Technical Specifications and UFSAR pages

2.0 FUNCTIONAL AND OPERATIONAL LIMITS

2.1 Functional and Operational Limits

2.1.1 Fuel to be Stored in the TN-68 Cask

The spent nuclear fuel to be stored in the TN-68 cask shall meet the following requirements:

A. Fuel shall be unconsolidated INTACT FUEL ASSEMBLIES except that up to 8 fuel assemblies with damage consisting of known or suspected cladding defects greater than pinholes or hairline cracks *may be stored subject to the following limitations:*
   - i they must be HANDLED BY NORMAL MEANS
   - ii they must be stored in a basket configured for damaged fuel, in the designated compartments shown in Figure 2.1.1-1, with end caps installed top and bottom,
   - iii there must be no missing fuel pins or fuel pin segments, *and*
   - *iv assembly average burnup is limited to ≤45 GWd/MTU.*

B. Fuel shall be limited to fuel with Zircaloy cladding. Fuel having stainless steel replacement rods may be stored provided that a shielding analysis demonstrates that the dose rate contribution from such rods is bounded by the design basis fuel rods.

C. Fuel shall be limited to the following fuel types or equivalents by other manufacturers with the following unirradiated specifications:

Assembly Type		Designation	#of Fuel Rods	Max Rod Pitch	Min Rod OD	Max Uranium Content (MTU/assy)
GE	7x7	2,2A,2B	49	0.738	0.563	0.1977
GE	7x7	3,3A,3B	49	0.738	0.563	0.1923
GE	8x8	4,4A,4B	63	0.640	0.493	0.1880
GE	8x8	5,6,6B,7,7B	62	0.640	0.483	0.1876
GE	8x8	8,8B	62	0.640	0.483	0.1885
GE	8x8	8,8B,9,9B,10	60	0.640	0.463	0.1824
GE	9x9	11,13	74	0.566	0.440	0.1757
GE	10x10	12	92	0.510	0.404	0.1857

Fuel designs 6, 6B, 7 and 7B may also be designated as P, B or BP. Fuel designs may be C, D or S lattice only.

D. Fuel assemblies may be channeled or unchanneled. Channel thickness up to 0.120 inches thick are acceptable.

E. 7x7 fuel assemblies shall have the bounding characteristics specified in Table 2.1.1-1.

tight. Venting/draining may occur while lifting the cask out of the pool. The cask is moved from the cask pit/spent fuel pool to the decontamination area. The remaining lid bolts are installed. The cask cavity is then evacuated and dried by means of a vacuum system and then back-filled with helium. The lid seals and penetration cover seals are leak tested. The top neutron shield is installed on the lid. The external surface radiation levels are checked to assure that they are within acceptable limits.

The overpressure system is installed and the overpressure system and seal interspace is pressurized with helium. The protective cover may be installed either in the decontamination area or at the ISFSI.

The cask is transferred to the ISFSI by a transport vehicle. The cask is set in its storage position, and connected to the site storage cask monitoring system. A channel operational test to verify proper functioning of the pressure switch/transducer is performed.

To unload the cask, these steps are performed in reverse. The cask is brought back to the reactor building. The protective cover, pressure monitoring system, overpressure tank and top neutron shield are removed. Prior to opening the cask, the cavity gas is sampled through the vent or drain port. The cavity is depressurized and the cask is lowered into the spent fuel pool. The cask is slowly filled with pool water or demineralized water through the vent or drain port. The cask is vented during this process. The water/steam mixture from the vent line may contain some radioactive gas. If the gas is radioactive, protective measures shall be imposed in accordance with ALARA such as routing the gas through the plant gaseous radwaste system. Pressure and temperature should be monitored during this operation. When the cask is full of water, the lid is removed and the fuel is accessible for unloading.

1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

1.2.2.3.1 Criticality Prevention

Criticality is controlled by utilizing neutron absorption materials in the fuel basket. These features are only necessary during the loading and unloading operations that occur in the cask loading pool (underwater). During storage, with the cavity dry and sealed from the environment, criticality control measures within the installation are not necessary because of the low reactivity of the fuel in the dry cask and the assurance that no water can enter the cask during storage.

1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the TN-68 dry storage cask.

1.2.2.3.3 Operation Shutdown Modes

The TN-68 dry storage cask is a totally passive system so that consideration of operation shutdown modes is unnecessary.

1.2.2.3.4 Instrumentation

The only instrumentation pertinent to storage is the pressure transducers/switches which monitor the cask seals for leakage. The transducers/switches monitor the pressure in an interspace between the inner and outer seals to provide an indication of seal failure before any release is possible.

An initial functional check of the transducers/switches is performed at the manufacturer's plant and a channel operational test is performed at the commencement of storage. Two identical transducers/switches are provided to assure a functional system through redundancy.

1.2.2.3.5 Maintenance Techniques

Because of their passive nature, the storage casks will require little, if any, maintenance over their lifetime. Typical maintenance would be limited to external paint touch-up and repressurizing the overpressure system. No special maintenance techniques are necessary.

1.2.3 Cask Contents

The TN-68 cask is designed to store up to 68 Boiling Water Reactor (BWR) fuel assemblies with or without fuel channels. The maximum allowable initial lattice-average enrichment varies from 3.7 to 4.7 wt% U235 depending on the B10 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 7x7 fuel, 60 GWd/MTU, 0.441 kW/assembly, and 7 years for all other fuel. Damaged fuel assemblies are limited to bundle average 45 GWd/MTU burnup. The cask is designed for a maximum heat load of 30 kW.

In addition to satisfying these limits, the fuel to be stored must also meet the fuel qualification requirements developed in Chapter 5. This assures that the radioactive source for shielding and confinement is bounded by the design basis fuel assembly, which is an 8x8 lattice with 63 fuel rods and with burnup, bundle average enrichment, and cooling time of 48 GWd/MTU, 2.6 wt % U235, and 7 years, respectively.

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. A description of the fuel assemblies is provided in Section 2.1.

The quantity and type of radionuclides in the spent fuel assemblies are described and tabulated in Chapter 5. Chapter 6 covers the criticality safety of the TN-68 cask and its contents, listing material densities, moderator ratios, and geometric configurations.

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CHAPTER 2

PRINCIPAL DESIGN CRITERIA

This chapter provides the principal design criteria for the TN-68 casks. Section 2.1 presents a general description of the spent fuel to be stored. Section 2.2 provides the design criteria for environmental conditions and natural phenomena. This section presents the analysis which shows that the casks will not tip over or move significant distances under the design basis seismic, tornado, wind and missile loadings, design basis earthquake or extreme floods. This section also contains an assessment of the local damage due to the design basis environmental conditions and natural phenomena and the general loadings and design parameters used for analysis in subsequent chapters. Section 2.3 provides a description of the systems which have been designated as important to safety. Section 2.4 provides a general discussion regarding decommissioning considerations. This is further elaborated on in Chapter 14. Section 2.5 summarizes the cask design criteria.

2.1 Spent Fuel To Be Stored

The TN-68 cask is designed to store 68 General Electric (GE) Boiling Water Reactor (BWR) spent fuel assemblies with or without fuel channels. The maximum allowable lattice-average initial enrichment varies from 3.7 to 4.7 wt% U235 depending on the B10 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 7x7 fuel, 60 GWd/MTU, 0.441 kW/assembly, and 7 years for all other fuel. The damaged fuel assembly is limited to bundle average burnup $\leq$ 45 GWd/MTU. The cask is designed for a maximum heat load of 30 kW. The fuel shall have no damage to fuel grid spacers that would render the fuel outside its licensing basis for use in the reactor. 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 is defined as fuel with known or suspected cladding defects greater than pinholes or hairline cracks. Damaged fuel to be stored in the TN-68 must be capable of being handled by normal means. There must be no missing fuel pins or fuel pin segments. Fuel with missing pins that have been replaced with dummy rods that displace a volume equal to or greater than that of the original rods is not regarded as damaged. This definition is based on analysis in Appendix 6B which shows that damaged fuel so limited would be retrievable under normal and off-normal conditions. Thermal and criticality analyses assume that such fuel undergoes further damage under accident conditions.

Scoping calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, heat load and confinement. The fuel assemblies considered are listed in Table 2.1-1. The design basis fuel for decay heat, shielding and confinement is an 8x8 lattice with 63 fuel rods and with burnup, bundle average enrichment, and cooling time of 48 GWd/MTU, 2.6 wt% U235, and 7 years, respectively. The fuel qualification screening which is developed based on this fuel is conservative when applied to other fuel types with lower mass of uranium. Of the acceptable contents, only 7x7 fuel has a greater mass of uranium, but it is not bounding because it is restricted to lower burnup, longer

5.2 Source Specification

There are five principal sources of radiation associated with cask storage that are of concern for radiation protection:

- Primary gamma radiation from spent fuel;
- Primary neutron radiation from spent fuel (both alpha-n reactions and spontaneous fission);
- Gamma radiation from activated fuel structural materials;
- Capture gamma radiation produced by attenuation of neutrons by shielding material of the cask; and
- Neutrons produced by sub-critical fission in fuel.

The first three sources of radiation are evaluated using SAS2H/ORIGEN-S modules of the SCALE[1] code with the 44 group ENDF/B-V library. The capture gamma radiation and sub-critical multiplication are handled as part of the shielding analysis which is performed with MCNP4C.

The fuel assemblies acceptable for storage in the TN-68 are listed in Table 2.1-1. This listing of fuel assemblies was collapsed into seven basic designs provided below. The various fuel assembly designs were separated according to fuel assembly array, the maximum metric tons of uranium, and the number of water rods. These three parameters are the significant contributors to the SAS2/ORIGEN-S model. The largest uranium loading results in the largest source term at the design basis enrichment and burnup.

Fuel Array Type	Number of Fueled Rods	Number of Water Rods	Metric Tons Uranium per Assembly
7 x 7	49	0	0.1977
8 x 8	63	1	0.1880
8 x 8	62	2	0.1856
8 x 8	60	4	0.1825
8 x 8	60	1	0.1834
9 x 9	74	2	0.1766
10 x 10	92	2	0.1867

Table 5.2-1 provides additional fuel assembly design characteristics for the seven basic fuel designs.

The 8x8 fuel assembly with 63 fuel rods is the design basis fuel for shielding purposes because it has the highest initial heavy metal loading of the 8x8, 9x9, and 10x10 fuels, and therefore results in the highest radioactive source terms for a given irradiation history. Initial enrichment of 2.6 wt% U235, assembly average burnup of 48 GWd/MTU, and cooling time of 7 years complete the specification of the design basis 8x8 fuel. A conservative three-cycle irradiation at a constant specific power of 6 MW/assy is utilized with a 30 day down time between each cycle to calculate the source terms.

For the purpose of developing the 7x7 fuel qualification table, this fuel is specified at 3.3 wt% U235 assembly average enrichment, 40 GWd/MTU burnup, 10 years cooling, and irradiation at a constant power of 5 MW/assy over three cycles.

The TN-68 cask is capable of storing damaged assemblies at the eight peripheral locations of the basket. Fuel damage is limited in accordance with SAR Section 2.1, Table 8.1-1 step A14, and Technical Specification 2.1.1. Even though the maximum bundle average burnup for damaged fuel assemblies is limited to 45 GWd/MTU, the design basis source terms and the shielding models for intact and damaged fuel assemblies are conservatively assumed to be identical.

The source terms are generated for the active fuel regions, the plenum region, and the end regions. Irradiation of the fuel assembly structural materials (including the channel, plenum, and end fittings) are included in the irradiation of the fuel zone. The fuel assembly hardware materials and masses on a per assembly basis are listed in Table 5.2-2. Table 5.2-3 provides the material composition of fuel assembly hardware materials. Cobalt impurities are included in the SAS2H model. In particular, the cobalt impurities in Inconel, Zircaloy and Stainless Steel are 0.649%, 0.001% and 0.08%, respectively.

Table 5.3-2 contains the masses of the various hardware materials present in the four principal regions of the fuel assembly with and without channels. Table 5.3-2a breaks these materials down into their elemental components, using the material compositions provided in Table 5.2-3.

The masses for the materials in the top end fitting, the plenum, and the bottom fitting regions are multiplied by the activation ratios 0.1, 0.2 and 0.15, respectively[6], to correct for the spatial and spectral changes of the neutron flux outside of the active fuel zone. These compositions, before and after the application of these activation ratios, are shown in Table 5.3-2a.

As an example, the cobalt mass in the top end region is calculated here:

Co = 1.88 kg Zircaloy (from Table 5.3-2)* 0.001 % (from Table 5.2-3) + 2.715 kg (stainless steel)*0.08 % + 0.556 kg Inconel * 0.649% = 0.005799 kg

effective Co (after applying the 0.1 flux factor) = 0.0005799 kg as shown in Table 5.3-2a

The material compositions of the fuel assembly hardware are included in the SAS2H/ORIGEN-S model on a per assembly basis. The cobalt content for each fuel assembly region utilized in the source term calculation is obtained from Table 5.3.2a reduced by the activation ratios and is repeated in Table 5.2-8.

Axial variation in the moderator density along the BWR fuel assembly was considered by including a volume averaged density for the moderator around the fuel pins. The following axial variation of temperatures and moderator densities were used to calculated the volume average moderator density for use in the BWR source term models[3].

APPENDIX 6B

DAMAGED FUEL CLADDING STRUCTURAL EVALUATION

6B.1 Introduction

The purpose of this appendix is to demonstrate structural integrity of the damaged fuel cladding in the TN-68 basket following normal and off-normal loading conditions of storage and onsite transfer (required for Part 72 License) and normal condition of offsite transport (required for Part 71 License: included here for information only).

Note: Although this appendix discusses low burnup and high burnup scenarios, only damaged fuel assemblies are limited to maximum bundle average burnup of $\leq$ 45 GWd/MTU.

In this appendix, the damaged fuel is defined as fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks and pinhole leaks as defined in Technical Specification Section 2.1.1. Damaged fuel must be capable of being handled by normal means in order to be stored in the TN-68.

This appendix evaluates stresses in the damaged fuel cladding associated with normal and off-normal conditions of on-site transfer/storage and off-site transport. It also presents a fracture mechanics assessment of the cladding using conservative assumptions regarding defect size geometry and amount of oxidation in the cladding material. These evaluations demonstrate the structural integrity of the damaged fuel cladding under normal and off-normal conditions.

The TN-68 cask and fuel basket is designed to store 68 intact fuel assemblies, or no more than 8 damaged and the remainder intact, for a total of 68 standard BWR fuel assemblies per canister. All the fuel assemblies, intact or damaged, consist of BWR fuel assemblies with Zircaloy cladding. Damaged fuel assemblies may only be stored in eight peripheral compartments of the TN-68 fuel basket fitted with end caps to retain and retrieve damaged fuel fragments.

6B.2 Design Input / Data

The design inputs are summarized in the following table. The design inputs and assumptions are the same as described in Appendix 6A.

Design Parameters of BWR Fuel Assemblies

Tube Arrays	**7 x 7**	**8 x 8**	**8 x 8**	**8 x 8**	**8 x 8**	**9 x 9**	**10 x 10**
GE Designation	GE2, GE3	GE4	GE5	GE8	GE9, GE10	GE11, GE13	GE12
No. of Full Length Fuel Rods	49	63	62	60	60	66	78
Maximum Active Fuel Length (in.)	144	146	150	150	150	146	150
Fuel Tube OD (in.)	0.563	0.493	0.483	0.483	0.483	0.44	0.404
Corroded Fuel Tube OD (in.)	0.5576	0.4876	0.4776	0.4776	0.4776	0.4346	0.3986
Clad Thickness (in.)	0.032	0.034	0.032	0.032	0.032	0.028	0.026
Corroded Clad Thickness (in.)	0.0293	0.0313	0.0293	0.0293	0.0293	0.0253	0.0233
Fuel Tube I.D. (in.)	0.499	0.425	0.419	0.419	0.419	0.384	0.352
Fuel Tube Radius, mid-thickness (in.)	0.2642	0.2282	0.2242	0.2242	0.2242	0.2047	0.1877
Corroded Fuel Tube Area (in^2)	0.0486	0.0449	0.0413	0.0413	0.0413	0.0325	0.0275
Corroded. Fuel Tube M.I. (in^4)	1.702 $x10^{-3}$	1.173 $x10^{-3}$	1.041 $x10^{-3}$	1.041 $x10^{-3}$	1.041 $x10^{-3}$	0.684 $x10^{-3}$	0.486 $x10^{-3}$
Irradiated Yield Stress at 750°F (1) (psi)	69,000	69,000	69,000	69,000	69,000	69,000	69,000
Young's Modulus E at 750°F (1) (psi)	1.21E+ 07	1.21E+ 07	1.21E+ 07	1.21E+ 07	1.21E+ 07	1.21E+ 07	1.21E+ 07

Note:
1. Values are calculated from PNNL report [36] with very small strain input.

6B.3 Loads

6B.3.a Part 72 Normal and Off-normal Condition Loads

The damaged fuel (≤ 45 GWd/MTU burnup) inside the TN-68 fuel basket is subjected to following normal and off normal condition Part 72 loads:

- Dead Weight
- Internal Pressure
- Thermal
- Transfer Load (Inertia Loads associated with moving the TN-68 cask in vertical position from the fuel loading area to the ISFSI site), which consists of 1*g* in the longitudinal, 1*g* in the transverse and 1*g* in the vertical direction.

The stresses due to the dead weight are insignificant. No internal pressure is assumed for the damaged fuel. The cladding is assumed to be able to expand due to thermal loads and thus no thermal-induced stresses are considered. However, the temperature of the cladding is considered for selection of allowable stresses. Therefore, the structural integrity of the damaged fuel (≤ 45 GWd/MTU burnup) is evaluated in this section only for the following transfer/storage loads.

1g Vertical Loads

The maximum g load acting on the damaged fuel rods subjected to 1g vertical load is conservatively taken as 2g (1g deadweight + 1g external load). The damaged fuel rod structural integrity under this 2g load is assessed by computing the compressive stress in the cladding by ratioing the stresses from the 15g end drop (2/15) given Section 6B.5. The results of the compressive stress are shown in the following table.

Axial Compressive Stresses due to Vertical Loads

Tube Arrays	**7 x 7**	**8 x 8**	**8 x 8**	**8 x 8**	**8 x 8**	**9 x 9**	**10 x 10**
GE Designation	GE2, GE3	GE4	GE5	GE8	GE9, GE10	GE11, GE13	GE12
No. of Full Length Fuel Rods	49	63	62	60	60	66	78
Fuel Assembly Weight (lb)	705	705	705	705	705	705	705
Sectional Area (in^2)	0.0486	0.0449	0.0413	0.0413	0.0413	0.0325	0.0275
Compressive Stress (ksi)	0.59	0.50	0.55	0.57	0.57	0.66	0.66

The axial stresses in the fuel rod are compressive stresses and are significantly less than the irradiated yield stress of the cladding material (69.0 ksi, at 750°F, data is calculated from PNNL report [36] with very small strain rate input). Therefore, the fuel rods will maintain their structural integrity when subjected to the 2g applied load in the vertical direction.

1g Longitudinal/Transverse Load

The maximum g load acting on the damaged fuel rods under longitudinal/transverse load is conservatively taken as 2g lateral load. The damaged fuel rod structural integrity under this load is assessed by computing the bending stress in the cladding by ratioing the stresses from the 35g side drop (2/35) given Section 6B.6.(1) The results of the maximum bending stress are shown in the following table.

Note 1: The maximum bending stresses in the 35g side drop analysis is due to overhang (top portion of the fuel rods). The stress outside the overhang location is much lower. During the transfer condition the cask is vertical, the fuel rod is supported at the bottom of the cask, and there will be no overhang of the fuel rod at the top of the basket. The 35g side drop analysis is nonlinear (Contact Element), most of the nonlinearity occurred at the overhang location, and the center and bottom end of the fuel rod remain mostly linear. Since the stress at the center and bottom portion is lower than the overhang location, ratioing the stress from the overhang location is conservative.

Summary of Stress Results for the Longitudinal/Transverse Load

	7x7	8x8	8x8	8x8	8x8	9x9	10x10
GE Designation	GE2, GE3	GE4	GE5	GE8	GE9, GE10	GE11, GE13	GE12
Fuel Cladding O.D. (in.)	0.5576	0.4876	0.4776	0.4776	0.4776	0.4346	0.3986
Fuel Cladding I.D. (in.)	0.499	0.425	0.419	0.419	0.419	0.384	0.352
Fuel Cladding thickness (in.)	0.0293	0.0313	0.0293	0.0293	0.0293	0.0253	0.0233
Max Bending Stress, S_b (ksi)	2.31	2.39	2.75	2.75	2.75	2.33	2.50

The maximum bending stresses in the fuel rod are significantly less than the irradiated yield stress of the cladding material (69.0 ksi).

The same fracture mechanics evaluation methodology described in Section 6B.8 is also used to evaluate the maximum cladding bending stresses due to the 2g lateral load case. The fuel cladding material fracture toughness (K_{IC}) is taken from the lower-bound value presented in reference [13] as described below.

1. Through-wall Axial Crack in the Cladding

As described in Section 6B.8.d, the maximum stress intensity factor results indicate that any existing axial through-wall crack in the spent fuel cladding would not sustain further damage from any additional load. From the linear elastic fracture mechanics point of view, the axial through-wall crack will not cause fracture.

2. Through-wall Circumferential Crack in the Cladding Under Bending

Section 6B.8.c evaluates this crack model using the model shown in Figure 6B-11. The calculation was performed using the computer code **pc-CRACK** [34]. Bending stresses at 1 ksi, 5 ksi, and 10 ksi to 80 ksi at 10 ksi intervals were calculated for a parametric fracture mechanics evaluation.

Figure 6B-13 presents the applied stress intensity factor versus half crack length for GE fuel cladding. The stress intensity factors (K) were presented for several levels of applied bending stress, from 1 ksi to 80 ksi.

Using the results in Figures 6B-13, for a given applied stress level along with selected through-wall circumferential crack length, the applied K value can be determined.

The maximum calculated bending stress from table above is 2.75 ksi for GE 8x8; conservatively using a stress level of 5 ksi and a half crack length of 0.05 in. (this is equivalent to crack length of 0.1 in., reference [26] indicates that typical crack length is less than

0.039 in.), the applied K value from Figure 6B-13 is determined to be approximately 2.0 ksi-in$^{1/2}$.

This calculated stress intensity factor (K) is compared with experimentally obtained fracture toughness, K_{IC}, of irradiated Zircaloy cladding material. Reference [13, pg. 4-1] suggests a typical lower-bound value of K_{IC} (PWR or BWR) for end-of-life burnup at 20°C (68°F) with relatively high hydrogen concentration (≈ 750 ppm) is in the range of 18-20 MPa m$^{1/2}$ (16.36-18.18 ksi in$^{1/2}$). Therefore, a K_{IC} value of 16.36 ksi in$^{1/2}$ is used for the fracture evaluation. The K_{IC} = 16.36 ksi in$^{1/2}$ fracture toughness value is considered conservative since it is measured at relatively low temperatures. Also, the stress intensity ratio is on the order of 0.12 (2/16.36) which translates into a factor of safety on the order of 8 which accounts for any unknown effects.

This evaluation demonstrates that the damaged fuel assemblies (≤ 45 GWd/MTU burnup) in the TN-68 cask will retain their structural integrity when subjected to normal and off-normal condition transfer and storage loads. Therefore, the retrievability of the damaged fuel assemblies is assured when subjected to any of these normal and off-normal transfer and storage loads.

6B.3.b Part 71 Normal Condition Loads

The evaluations of the 1 foot end drop and 1 foot side drop are for information only for Part 72 application. The structural integrity of the damaged fuel cladding due to 1 foot end drop and 1 foot side drop will be addressed and analyzed in the future Part 71 application.

The damaged fuel is evaluated for the following normal condition 10CFR Part 71 off-site transportation loads:

- 1 foot end and side drop loads
- Vibratory loads
- Shock load
- Lifting and Tie-down loads

During one-foot end and side drops, fuel assemblies are subjected to 15g and 35g loads respectively [10].

Vibratory loads of 0.30g in longitudinal direction, 0.30g in the transverse direction and 0.60g in the vertical direction, taken from Reference [4] are considered representative for a truck loaded cask. The vibration load of 0.19g in the longitudinal direction, 0.19g in the transverse direction and 0.37g in the vertical direction, taken from Reference [4], are considered representative for a rail car loaded cask [5].

The shock load of 4.7g in the longitudinal and 4.7g in the lateral and vertical directions for a rail car loaded cask (bounding values between rail and truck transport) during off-site transport are also taken from Reference [5].

Lifting load of 6g vertical is taken fromPart71-45(a). Tie-down loads 2g (vertical)/5g (lateral)/10g (longitudinal) are taken from Part71-45(b).

All of the above loads however are bounded by 1 foot end drop (15g) and 1 foot side drop (35g) transport load. Therefore, structural integrity of the damaged fuel for the normal conditions of Part 71 is evaluated only for the one-foot end and side drop conditions.

6B.4 Evaluation Criteria

The retrievability of the damaged fuel in the TN-68 Cask is assured if the damaged fuel cladding retains its structural integrity when subjected to normal and off normal loads. Per the damaged fuel definition in Section 6B.1, the damaged fuel rods loaded in the TN-68 basket may have cladding defects greater than hairline cracks or pinhole leaks. However, under normal and off-normal loads, the original defects (such as cracks or pinholes or missing grid) should not change significantly so that the damaged fuel can be retrieved.

The damaged fuel cladding needs to meet the following criteria to ensure their structural integrity and thus be retrievable:

- Fuel cladding stresses under normal and off-normal load conditions are less than the irradiated yield strength of the cladding material.
- Stability of the cladding tube is maintained (i.e., no buckling occurs).
- The stress intensity factor, K_I, of the fuel cladding tube geometry considering through-wall flaw is less than experimentally determined fracture toughness, K_{Ic}, considering temperature and irradiation effects.

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20. (not used)

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7.3 Confinement Requirements for Hypothetical Accident Conditions

7.3.1 Source Terms for Confinement Calculations

Section 5.2.4 provides the definitions and source terms for three combinations of burnup, enrichment, and cooling time for 8x8 fuel: design basis (DBF-68), medium burnup (MBF-68) and high burnup (HBF-68). These represent bounding combinations of fuel characteristics allowed for storage under the fuel qualification flowchart in TN-68 Technical Specification Figure 2.1.1-2. The evaluation here of these three combinations verifies that the fuel qualification flowchart provides a basis for selecting fuel that is appropriate not only for thermal and shielding limits, but also for confinement limits.

Fuel Description	Burnup	Enrichment	Cooling Time
Design Basis Fuel **(DBF-68)**	48 GWd/MTU	2.6 wt % U235	7 Years
Medium Burnup Fuel **(MBF-68)**	55 GWd/MTU	2.5 wt % U235	10 Years
High Burnup Fuel **(HBF-68)**	60 GWd/MTU	3.2 wt % U235	12 Years

The TN-68 cask is also authorized to load up to eight damaged assemblies with bundle average burnup ≤ 45 GWd/MTU at the peripheral locations of the cask. Fuel damage is limited in accordance with SAR Section 2.1, Table 8.1-1 step A14, and Technical Specification 2.1.1. The source terms for these damaged fuel assemblies are identical to those of the intact fuel assemblies.

Table 5.2-10 lists the activity representing the fission gases, volatiles, and fines contributing more than 0.1% of the activity contained in the 68 fuel assemblies, plus Iodine 129.

The releasable source term is first determined. The release fractions applied to the source term are provided below (developed from References 3 and 4).

Variable	Off-Normal Conditions	Accident Conditions
Fraction of crud that spalls off rods, f_C	0.15	1.0
Fraction of Rods that develop cladding breaches, f_B	0.10	1.0
Fraction of Gases that are released due to a cladding breach, f_G	0.3	0.3
Fraction of Fines that are released due to a cladding breach, f_F	3×10^{-5}	3×10^{-5}
Fraction of Released fines that remain airborne following a cladding breach, $F_{f,a}$ *	0.10	0.10
Fraction of Volatiles that are released due to a cladding breach, f_V	2×10^{-4}	2×10^{-4}

* 0.003% of the fuel in a rod is released from the rod during a cladding failure in the form of fines. However, only 10% of the fuel fines ejected from the rod during a cladding failure remain airborne (Reference 10).

The releasable source term also depends on the leak rate from the TN-68. Under off-normal conditions, it is assumed that the overpressure system is not functioning properly. In this case, the cask cavity gas is free to leak out at a rate of 1 x 10^{-5} std cc /sec. Assuming the cask cavity gas acts like helium (including the gases, volatiles, fines and crud), the leak rate is adjusted to a helium leak rate at cask cavity conditions using the equations of ANSI N14.5. This calculation is shown below.

P_u = 2.47 atm abs , 36.3 psig (off-normal cask cavity pressure assuming 10% of the fuel rods have failed – Section 4.7.5)
P_d = 1.0 atm abs
D = 4.825 x 10^{-4} cm
a = 0.5 cm
μ = 0.0279 cP (for helium at 479K)
T = fluid absolute temp = average cavity gas temp = 402°F = 479 K
M = 4.0
$P_a = ½ (P_u + P_d)$ = 1.735 atm abs

Substituting:

F_c = 9.674E-06
F_m = 5.399E-06