

SAFETY EVALUATION REPORT

Docket No. 71-9293
Model No. TN-68 Package
Certificate of Compliance No. 9293
Revision No. 0

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SUMMARY

By application dated May 19, 1999, as supplemented, Transnuclear, Inc. (TN), requested that the Nuclear Regulatory Commission approve the Model No. TN-68 as a Type B(U)F-85 package. Based on the statements and representations in the application, as supplemented, and the conditions listed below, the staff concludes that the package meets the requirements of 10 CFR Part 71.

References

TN application dated May 19, 1999.

Supplements dated March 2, October 18, and November 13, 2000 and January 12, 2001.

1.0 GENERAL INFORMATION

1.1 Packaging

The TN-68 is predominantly a steel package that is used to transport up to 68 intact BWR fuel assemblies with or without channels. The overall dimensions of the package are 271 inches long and 144 inches in diameter with the impact limiters installed.

The package generally consists of four components, the fuel basket assembly, a containment vessel within a forged steel cask body, a radial neutron shield, and impact limiters.

The basket assembly locates and supports the fuel assemblies, transfers heat to the cask body wall and provides neutron absorption to satisfy sub-criticality requirements. The basket structure consists of an assembly of stainless steel cells, joined by fusion welding of 1.75 inch wide stainless steel plates. Above and below the plates are slotted borated aluminum (or boron carbide/aluminum) metal matrix composite neutron poison plates which form an egg-crate structure. This construction forms a honey-comb like structure of cell liners which provides compartments for 68 fuel assemblies. The nominal dimensions of each cell is 6.0 inches x 6.0 inches.

A thick-walled (6.0 inch), forged steel cask body for gamma shielding surrounds the containment vessel, by an independent shell and bottom plate of carbon steel. The gamma shield completely surrounds the containment vessel inner shell and bottom closure. The thickness of the bottom of the cask body is 8.25 inches. A 4.5 inch thick steel gamma shield is also welded to the inside of the containment lid.

The containment boundary consists of an inner shell and bottom plate, shell flange, lid outer plates, lid bolts, penetration cover plate and bolts and the inner metallic O-rings of the lid seal and the two lid penetrations (vent and drain). The containment vessel length is approximately 189 inches with a wall thickness of 1.5 inches. The cylindrical cask cavity has a nominal diameter of 69.5 inches and a length of 178 inches. The containment lid is 5 inches thick and is fastened to the cask body with 48 bolts. Double metallic O-ring seals are provided for lid closure. To preclude air in-leakage, the cask cavity is pressurized with helium to above atmospheric pressure. There are two penetrations through the containment vessel which are located in the lid. These penetrations are for draining and venting. Double metallic seals are also used on these two lid penetrations. The OP port provides access to the interspace lid seals for leak testing purposes. The OP transport cover is not part of the containment boundary.

Neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield. The resin compound is cast into long, slender aluminum containers. The total thickness of the resin and aluminum is approximately 6 inches. The array of resin-filled containers is enclosed within a smooth 0.75 inch outer steel shell constructed of two half cylinders.

The package has impact limiters at each end of the cask body. The impact limiters consist of balsa wood and redwood blocks, encased in sealed stainless steel shells that maintain the wood in a dry atmosphere and provide wood confinement when crushed during a free drop. The impact limiters have internal radial gussets for added strength and confinement. The impact limiters are attaching to each other using 13 tie rods and to the cask by eight bolts attaching to brackets welded to the outer shell in eight locations (four bolting locations per impact limiter).

The approximate dimensions and weights of the package are as follows:

Overall length (with impact limiters, in)	271
Overall length (without impact limiters, in)	197
Impact Limiter Outside diameter, (in)	144
Outside diameter (without impact limiters, in)	98
Cavity diameter (in)	69.5
Cavity length (in)	178
Containment shell thickness (in)	1.5
Containment vessel length (in)	184
Body wall thickness (in)	7.5
Containment lid thickness (in)	5
Overall lid thickness (in)	9.5
Bottom thickness (in)	9.75
Resin and aluminum box thickness (in)	6
Outer shell thickness (in)	0.75
Overall basket length (in)	164
Maximum weight of package (pounds)	272,000
Maximum weight of BWR fuel contents (pounds)	47,900
Maximum weight of impact limiters and attachments (pounds)	32,000

1.2 Type and Form of Material

Fuel is limited to 68 unconsolidated intact irradiated GE BWR fuel assemblies with zircalloy cladding. An intact fuel assembly is a spent nuclear fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks. Partial fuel assemblies (i.e. spent fuel assemblies from which fuel rods are missing), shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water equal to that displaced by the original rod(s).

Spent nuclear fuel may be transported with or without channels. Any fuel channel thickness up to 0.120 is acceptable on any of the fuel designs shown below. The maximum initial rod pressurization is 155 psig. The maximum fuel assembly length is 176.2 inches and the maximum fuel assembly width is 5.44 inches.

Permissible fuel assemblies are limited as stated in table 1-2-1 (fuel types may be C, D, or S lattice):

Table 1-2-1, Fuel characteristics

GE fuel generation	model	array	rod pitch	fuel rods	rod od	clad thick	pellet dia.	water rods	water rod od	water rod id	U content (MTU/ Assembly)	Max active fuel length
2A	2a	7x7	0.738	49	0.570	0.036	0.488	0	x	x	0.1977	144
2, 2B	2	7x7	0.738	49	0.563	0.032	0.487	0	x	x	0.1977	144
3, 3A, 3B	3	7x7	0.738	49	0.563	0.037	0.477	0	x	x	0.1896	144
4, 4A,4B	4	8x8	0.640	63	0.493	0.034	0.416	1	0.493	0.425	0.1880	146
5	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
6, 6B	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
7, 7B	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	0.1876	150
8, 8B -2w	82	8x8	0.640	62	0.483	0.032	0.411	2	0.591	0.531	0.1885	150
8, 8B-4W*	84	8x8	0.640	60	0.483	0.032	0.411	4	0.591	0.531	0.1824	150
8, 8B-4W**	84	8x8	0.640	60	0.483	0.032	0.411	4	0.483	0.431	0.1824	150
9, 9B	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	0.1824	150
10	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	0.1824	150
11	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	0.1757	146 full, 90 partial
13	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	0.1757	146 full, 90 partial
12	12	10x10	0.510	92	0.404	0.026	0.345	2	0.98	0.92	0.1857	150 full, 93 partial

*2 large water rods

**2 small water rods

Notes on table 1-2-1:

1. All dimensions in inches.
2. All fuel channels 5.278 inches inside, and from 0.065 to 0.120 inches thick.
3. All fuels are evaluated with 96.5% theoretical density and 3.7 wt% U-235 average enrichment.
4. The fuel pitch is for C and D lattice designs. The S lattice fuels have a smaller pitch, which is less reactive.
5. The fuel designs designated by GE as 6, 6B, 7, and 7B are sometimes referred to as "P" (pressurized) and "B" (barrier).

Provided all of the requirements of this section are met, the bounding fuel characteristics are: a) maximum initial lattice-average enrichment is 3.7%; b) the minimum initial bundle average enrichment is 3.3%; c) the maximum assembly average burnup is 40,000 MWD/MTU; d) the minimum cool time is 10 years; and e) the maximum heat load per assembly is 0.313 Kw.

Fuel assemblies are categorized into three types, Type I, Type II and Type III. There are two basic loading configurations for the package. The first configuration is a mixture of Type I and Type II fuel assemblies. The second configuration is Type III fuel assemblies. The maximum burnup, minimum initial enrichments and cooling times for each of the three fuel assembly types is contained in the tables below.

In the mixed Type I and Type II configuration, Type I assemblies shall be placed only into the interior compartments of the fuel basket as shown in figure 5.3-3 of the application. Type II fuel assemblies may be placed in any basket fuel compartment.

In the second configuration, Type III fuel assemblies may be placed in any basket fuel compartment.

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)
TYPE I BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	10	10	X	X	X	X	X	X	X	X	X	X
1.1	10	10	X	X	X	X	X	X	X	X	X	X
1.2	10	10	X	X	X	X	X	X	X	X	X	X
1.3	10	10	X	X	X	X	X	X	X	X	X	X
1.4	10	10	X	X	X	X	X	X	X	X	X	X
1.5	10	10	10	10	11	11	11	X	X	X	X	X
1.6	10	10	10	10	10	11	11	11	X	X	X	X
1.7	10	10	10	10	10	11	11	11	12	X	X	X
1.8	10	10	10	10	10	11	11	11	11	12	X	X
1.9	10	10	10	10	10	11	11	11	11	12	X	X
2.0	10	10	10	10	10	10	11	11	11	12	12	X
2.1	10	10	10	10	10	10	11	11	11	12	12	12
2.2	10	10	10	10	10	10	11	11	11	12	12	12
2.3	10	10	10	10	10	10	11	11	11	11	12	12
2.4	10	10	10	10	10	10	10	11	11	11	12	12
2.5	10	10	10	10	10	10	10	11	11	11	12	12
2.6	10	10	10	10	10	10	10	11	11	11	12	12
2.7	10	10	10	10	10	10	10	10	11	11	11	12
2.8	10	10	10	10	10	10	10	10	10	11	11	12
2.9	10	10	10	10	10	10	10	10	10	11	11	12
3.0	10	10	10	10	10	10	10	10	10	10	11	12
3.1	10	10	10	10	10	10	10	10	10	10	11	12
3.2	10	10	10	10	10	10	10	10	10	10	10	11
3.3	10	10	10	10	10	10	10	10	10	10	10	10
3.4	10	10	10	10	10	10	10	10	10	10	10	10
3.5	10	10	10	10	10	10	10	10	10	10	10	10
3.6	10	10	10	10	10	10	10	10	10	10	10	10
3.7	10	10	10	10	10	10	10	10	10	10	10	10

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)
TYPE II BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	18	21	X	X	X	X	X	X	X	X	X	X
1.1	17	20	X	X	X	X	X	X	X	X	X	X
1.2	17	20	X	X	X	X	X	X	X	X	X	X
1.3	17	20	X	X	X	X	X	X	X	X	X	X
1.4	17	20	X	X	X	X	X	X	X	X	X	X
1.5	16	19	25	26	26	X	X	X	X	X	X	X
1.6	16	19	25	26	26	X	X	X	X	X	X	X
1.7	16	19	25	25	26	26	27	X	X	X	X	X
1.8	16	19	24	25	26	26	27	27	X	X	X	X
1.9	16	19	24	25	25	26	27	27	X	X	X	X
2.0	16	18	24	25	25	26	26	27	28	X	X	X
2.1	15	18	23	25	25	26	26	27	27	X	X	X
2.2	15	18	23	25	25	25	26	27	27	X	X	X
2.3	15	18	23	24	25	25	26	26	27	27	X	X
2.4	15	18	22	24	24	25	26	26	27	27	X	X
2.5	15	17	22	24	24	25	25	26	26	27	X	X
2.6	15	17	22	24	24	24	25	26	26	27	X	X
2.7	15	17	22	24	24	24	25	26	26	26	27	27
2.8	14	17	22	23	24	24	25	25	26	26	27	27
2.9	14	17	22	23	23	24	24	25	26	26	27	27
3.0	14	17	21	23	23	23	24	25	25	26	27	27
3.1	14	17	21	23	23	23	24	25	25	26	27	27
3.2	13	16	21	23	23	23	24	24	25	25	26	27
3.3	13	16	21	23	22	23	23	24	25	25	26	26
3.4	13	16	21	23	22	23	23	24	25	25	26	26
3.5	13	16	21	22	22	23	23	24	25	25	26	26
3.6	13	16	21	21	22	22	23	24	25	25	26	26
3.7	12	15	20	21	22	22	23	24	25	25	25	26

Acceptable cooling time as a function of maximum burnup and minimum initial enrichment and BWR Cooling times (years)
TYPE III BWR Fuel

Burnup (GWd/MTU)

Initial Enrichment (bundle ave %w)	15	20	30	32	33	34	35	36	37	38	39	40
1.0	10	11	X	X	X	X	X	X	X	X	X	X
1.1	10	11	X	X	X	X	X	X	X	X	X	X
1.2	10	10	X	X	X	X	X	X	X	X	X	X
1.3	10	10	X	X	X	X	X	X	X	X	X	X
1.4	10	10	X	X	X	X	X	X	X	X	X	X
1.5	10	10	15	16	16	17	17	X	X	X	X	X
1.6	10	10	14	16	16	17	17	17	X	X	X	X
1.7	10	10	14	15	16	16	17	17	17	X	X	X
1.8	10	10	14	15	15	16	16	17	17	18	X	X
1.9	10	10	14	15	15	16	16	17	17	18	X	X
2.0	10	10	14	15	15	16	16	16	17	17	18	X
2.1	10	10	14	15	15	15	16	16	16	17	18	18
2.2	10	10	13	14	15	15	16	16	16	17	17	18
2.3	10	10	13	14	15	15	16	16	16	17	17	18
2.4	10	10	13	14	15	15	15	16	16	17	17	18
2.5	10	10	13	14	14	15	15	16	16	16	17	18
2.6	10	10	13	14	14	15	15	16	16	16	17	17
2.7	10	10	13	14	14	15	15	15	16	16	17	17
2.8	10	10	13	13	14	14	15	15	16	16	17	17
2.9	10	10	13	13	14	14	15	15	15	16	16	17
3.0	10	10	12	13	14	14	14	15	15	16	16	17
3.1	10	10	12	13	14	14	14	15	15	15	16	16
3.2	10	10	12	13	14	14	14	15	15	15	16	16
3.3	10	10	12	13	13	14	14	14	15	15	16	16
3.4	10	10	12	13	13	13	14	14	15	15	16	16
3.5	10	10	12	13	13	13	14	14	14	15	15	16
3.6	10	10	12	12	13	13	14	14	14	15	15	15
3.7	10	10	12	12	13	13	14	14	14	15	15	15

1.2.2 Maximum Quantity of Material Per Package

1.2.2.1 Weight

The maximum contents weight is 75,600 pounds. The maximum weight of the irradiated fuel contents is 47,900 pounds.

1.2.2.2 Decay Heat Limit

Maximum decay heat per package not to exceed 21.2kW. The maximum heat load per assembly is 0.313 kW/assembly.

1.3 Conclusions

The applicant adequately described the contents of the package as required by 10 CFR 71.33(b). The staff agrees with the applicant's conclusion that the package meets the requirements of 10 CFR Part 71.

1.4 Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control:	0.0
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1.5 Drawings

The package is constructed and assembled in accordance with TN drawings:

- 972-71-1, Revision 1
- 972-71-1, Revision 2
- 972-71-3, Revision 4
- 972-71-4, Revision 2
- 972-71-5, Revision 1
- 972-71-6, Revision 1
- 972-71-7, Revision 3
- 972-71-8, Revision 2
- 972-71-9, Revision 2
- 972-71-10, Revision 1
- 972-71-11, Revision 1
- 972-71-12, Revision 0
- 972-71-13, Revision 0
- 972-71-14, Revision 1

2.0 STRUCTURAL EVALUATION

The objective of the structural review is to determine and verify that the information presented in the Safety Analysis Report (SAR), under normal conditions of transport and hypothetical accident conditions, are acceptable, complete, and meet the requirements of 10 CFR Part 71.

2.1 Description of Structural Design

The TN-68 transportation package consists generally of four components. These components are the cask body, the fuel basket, the neutron shield and the impact limiters.

The cask body is further divided into containment and non-containment structures. The containment boundary of the TN-68 consists of the inner shell (both the cylindrical portion as well as the bottom plate), the closure flange out to the seal seating surface, and the lid assembly outer plate. The lid bolts and seals are also part of the containment boundary. The non-containment boundary components consist of the steel gamma shielding, and the trunnions. While these components do not have a containment function, they must react to the structural loads, and in some cases, share loading with the containment system components.

The ASME Boiler and Pressure Vessel Code (ASME Code), Section III, 1995 including 1996 addenda is the governing code for the structural design of the package. The containment boundary components are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NB for Class 1 components. In addition, the allowable stress limits, the fracture toughness criteria, and design loading for package containment boundary components are shown to be consistent with the requirements of Regulatory Guides 7.6 and 7.8, and Subsection WB of the ASME Section III, Division 3. The non-containment boundary components are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF for support components. The fuel basket is designed and constructed as a core support structure in accordance with Section III, Subsection NG of the ASME Code. There are a few deviations from the ASME Code such as code stamping, imposing the QA requirements of Nuclear Quality Assurance-1 (NQA-1) or 10 CFR Part 71 in lieu of specific provisions dealing with fabricator qualifications, etc. These deviations are listed as exemptions to the code for the TN-68 transportation package in Table 2.11 of the SAR.

The impact limiter and attachments are designed to withstand the applied loads and to prevent separation of the limiters from the cask during impact. To prepare a cask for off-site transport, the impact limiters are bolted to the ends of the cask and further restrained with tie-rods. The impact limiters are constructed of two types of wood glued into assemblies and encased in a weather tight stainless steel sheet metal cover. The fasteners are made of ferritic steel. A principle design requirement for the impact limiters is their performance at the design minimum temperature of -20 °F. To verify the performance of the impact limiter, a drop test was conducted at -20 °F.

A one-third scale model cask was subjected to a series of 30 foot drop tests to verify both the overall impact limiter design and the performance of the wood and glue components of the impact limiters. The tests confirmed that the glued wood assemblies did not suffer any adverse change in material properties due to low temperatures.

The austenitic stainless steel used for the impact limiter covers is known for its immunity to changes in physical properties at cryogenic temperatures, thus, its selection and use

in this application is appropriate without further tests. The fasteners and tie-rods are made from SA-193 and SA-540 ferritic steels. Ferritic steels have temperature dependent properties. A specific grade of ferritic steel will have a transition temperature region where, with lowering temperatures, the impact resistance falls substantially. This reduction in impact resistance makes the material behave in a brittle fashion. As long as the impact strength at the design temperature is measured and found to meet the design requirements, then the material is suitable for use.

Since there are no code requirements for the impact strength of these two materials at -20 °F, the applicant has committed to test each lot of bolts and stay-rods to ensure they exceed the minimum design requirements. Impact testing will be performed at the design minimum temperature before any of these materials are accepted for use.

Neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield. The resin compound is cast into long, slender aluminum containers. The total thickness of the resin and aluminum is approximately 6 inches. The array of resin-filled containers is enclosed within a smooth 0.75 inch outer steel shell constructed of two half cylinders.

Full details of the TN-68 transportation packaging design are provided on the drawings in Appendix 1.4 of the SAR. The staff reviewed the package description presented in the General Information and Structural Evaluation sections of the application and found that the regulatory requirements of 10 CFR 71.33 have been adequately addressed.

2.2 Materials

The structural materials used for the TN-68 transportation package are delineated on the drawings in Appendix 1.4 of the SAR. The material properties that are used in the structural analysis of the packaging are provided in Tables 2.7 and 2.8 of the SAR.

The confinement shell and bottom plate are designed with SA 203 Grade E material. The lid material selected is either SA-350 Grade LF3 or SA 203 Grade E. Both steels are Section III, Class 1 materials.

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

Material properties of the TN-68 fuel basket are provided in Tables 2.10 and 2.11 of the SAR. No structural consideration is given to the potential load carried by the basket's poison plates (either borated aluminum, or a boron carbide / aluminum metal matrix composite). Structural detail of basket fabrication is given on drawings 972-71-5 and 972-71-6 in the SAR. Axial rails support the basket at the inside cask wall. These are fabricated from 6061-T6 aluminum. Load-bearing materials are the aluminum basket rails (6061-T6), the stainless steel square tubing and the stainless steel plate (SA-240 Type 304). The basis for the allowable stress for the Type 304 stainless steel fuel-compartment box, plate and 6061-T6 alloy is Section III of the ASME Code.

The upper trunnion material for the TN-68 design is SA-182 Grade F6NM. The rear trunnion material for the TN-68 design is SA-105. Material properties of these materials are provided in Tables 2.13 and 2.15 of the SAR.

The impact limiter materials consisted of mainly balsa and redwood blocks. Carbon steel cylinders, gussets, and end plates are designed to position and confine the balsa and redwood blocks so that the impact energy is properly absorbed. Austenitic stainless steel is used for the impact limiter covers. The fasteners and tie-rods are made from SA-193 and SA-540 ferritic steels. Material properties of the TN-68 impact limiter and attachment are provided in Appendix 2.10.8 of the SAR.

The fracture toughness of ferrous components is assessed in Appendix 2.10.4 of the SAR. This assessment is performed in the process of determining pre-service and in-service inspection requirements and allowable flaw sizes for various loading conditions and temperatures.

The neutron shielding material consists of a borated polyester resin. The castability of the resin and borated additive depends upon the viscosity of the mixed resin and filler and the size of the channel to be filled. For this package, it was determined that the viscosity of the resin and filler mixture was fluid enough to ensure no difficulties would occur. Additionally, the channels that form the in-place mold for the resin mixture are of sufficient size to avoid bridging of the resin and allow air to escape. These two factors permit the resin to completely fill the channel without fear of resulting voids in the shield.

Long term chemical stability of the polymeric neutron shield was examined by the applicant. For the filled polyester polymer employed, the radiation stability has been tested to radiation doses far exceeding that which would be achieved in cask service. The useful life of the filled resin was found to greatly exceed the cask service life. Further, even if the polymer were to degrade, there would be no loss of material, (i.e., no loss of neutron shield material) that would result in a reduction of neutron shielding ability. The primary degradation mechanism for polyester under the conditions experienced on the cask is embrittlement. Since the material is completely encased within aluminum channels, there is no requirement for the polymer to retain strength or impact resistance. Thus even if the neutron shield material became embrittled, it will remain in place and retain its neutron shielding ability.

The staff found that the material properties provided in the SAR are consistent with acceptable industry standards and are acceptable.

2.3 Fabrication and examination

Fabrication and examination specifications of the package are prescribed mainly by Section III of the ASME Boiler and Pressure Vessel Code. Specific sections, divisions, subsections, and articles of the ASME Code for fabrications and examinations of different transportation components are delineated on the drawings in Appendix 1.4 and listed in Section 2.1.2 of the SAR. The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.31(c).

2.4 General Standards for All Packaging (10 CFR 71.43)

2.4.1 Minimum Package Size

The overall length of the package with the impact limiter is approximately 271 inches. The diameter of the package without the impact limiters is 98 inches. The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.43(a) for minimum size.

2.4.2 Tamperproof Feature

A security wire seal is installed in the upper impact limiter attachment tie-rod prior to each shipment. It would be necessary to remove the upper impact limiter to have access to the closure lid. The presence of this seal demonstrates that unauthorized entry into the package has not occurred. The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.43(b) for a tamperproof feature.

2.4.3 Positive Closure

The package is positively closed by the bolts which secure the lid to the upper ring forging of the cask. Deliberate loosening of bolts requires extensive effort, using appropriate equipment. The large pre-load applied to the lid bolts prevents inadvertent opening of the cask closure lid from loads such as bottom-end drop and thermal expansion. The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.43(c).

2.4.4 Chemical and Galvanic Reaction

There are no materials that would produce significant adverse chemical, galvanic or other chemical reactions that would result in compromising the design, design margins, or any components ability to perform its containment or transport function. The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.43(d).

2.5 Lifting and Tie-Down Standards For All Packages

2.5.1 Lifting Devices (10 CFR 71.45(a))

Lifting of the package is accomplished by using two upper trunnions which are each attached to the gamma shield with twelve 1.5-inch bolts. Although 10 CFR 71.45(a) requires a minimum factor of safety of three against yield, the package is designed to the requirements of ANSI N14.6 which require a safety factor of 6 against the trunnion material yield stress and 10 against the trunnion material ultimate stress. Global stresses in the cask wall due to the effect of a 6g lifting load were simulated by applying 6g vertical acceleration in the ANSYS finite element model of the package. A dynamic amplification factor of 1.15 was also used on the dead weight load. Since the local stresses in the cask body outer shell at the trunnion locations due to the loadings applied through the trunnions were not included in the ANSYS stress results, they are

superimposed on the results when the stress values are evaluated. The local stresses were calculated using the techniques of Welding Research Council-107 which analyze local stresses in cylindrical shells due to external loading. The results indicated that, even with a safety factor of 6, the combined stress intensities in the cask were less than the yield stresses. Stresses in the trunnion material and trunnion flange bolts were analyzed using conventional textbook methods. The results indicated that calculated stresses were below the allowable stresses. Therefore, the TN-68 trunnion design is adequate.

By comparing the local stresses at upper trunnion/containment vessel interface, the minimum margin of safety (shear stress) was found to occur at the trunnions' shoulders. Under excessive loads, the trunnion shoulder would fail before the trunnion bolts or containment vessel. This failure mode does not compromise the ability of the package to meet the other requirements of 10 CFR Part 71.

In the transport configuration, it is possible that the front trunnions could accidentally be used as tie-down devices. Analysis were performed to evaluate the condition in which the front and rear trunnions share equally the tie-down load without taking structural credit for the front support saddle. The analysis results indicated that, even if the upper trunnions were accidentally used as tie-down devices, the trunnions could withstand the tie-down loads and would not exceed the allowable stresses.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirement of 10 CFR 71.45(a) for lifting devices.

2.5.2 Tie-Down Devices (10 CFR 71.45(b))

In the transport configuration, the tie-down loads of the package are shared by the rear trunnions and the outer surface of the cask at the front end which contacts the front saddle of the shipping frame. The rear trunnions were analyzed by assuming that the longitudinal load (10g) was taken by the two rear trunnions. The vertical load (2g) was taken 50 percent by the rear trunnions and 50 percent by the front saddle support. The 5g lateral load was taken 50 percent by one rear trunnion and 50 percent by the front saddle support. Global stresses in the cask wall due to the effect of a 2/5/10g tie-down load on the trunnions were calculated by the ANSYS finite element computer code. Stress concentrations (local stresses) caused by the trunnion loads acting on the gamma shield were analyzed using the techniques of Welding Research Council-107. The local stresses and global stresses were conservatively added together. The results of the analysis indicated that the package tie-down system is capable of withstanding the applied force specified in 10 CFR 71.45(b)(1) without generating stress in package components in excess of material strength.

By comparing the local stresses at rear trunnion/containment vessel interface, the minimum margin of safety (shear stress) was found to occur at the trunnions' shoulders. An excessive load would damage the trunnion, but the cask would not lose its structural integrity. The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.45(b)(3) for tie-down devices.

2.6 Normal Conditions of Transport (10 CFR 71.71)

2.6.1 Heat

The applicant's evaluation of the package to meet the normal conditions of transport heat condition was based on the results of computer simulation using the ANSYS finite element computer code. The temperatures and temperature distributions (due to hot environment condition at 100°F ambient temperature) applied to the package were calculated in Chapter 3 of the SAR. The finite element analysis conservatively assumed a design internal pressure of 100 psig applied to the cask cavity. In addition to the internal pressure, the load combinations also included the lid bolt pre-loads, fabrication loads, and 1g unit deceleration load at the trunnions resulting from braking of the vehicle transporting the package. The results (shown in Tables 2-19 and 2-20 in the SAR) indicated that the combined stresses due to thermal expansion, internal pressure, bolt pre-loads, fabrication loads, and 1g unit deceleration load were below the ASME code allowables.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(1).

2.6.2 Cold

The applicant's evaluation of the package to meet the normal conditions of transport cold condition was based on the results of computer simulation using the ANSYS finite element computer code. Thermal stresses due to -40°F ambient temperature and -20°F ambient temperature were calculated. Thermal stresses for cold environment at -20°F ambient temperature were used to combine stresses due to normal and hypothetical accident load conditions as per Regulatory Guide 7.8. Thermal stresses due to -40°F ambient temperature were used to assess the package for the effects of a ambient temperature of -40°F. Since the internal pressure generally reduce thermal stresses due to contraction, the cask internal pressure was conservatively neglected in the cold environment load combinations. This resulted in a net external pressure loading of 14.7 psig. However, a 25 psig external pressure loading was conservatively used in the cold environment load combinations calculation. The lid bolt pre-loads, fabrication stresses, gravity and trunnion local loads were included. The results (shown in Tables 2-21 and 2-22 of the SAR) indicated that the combined stresses due to the cold environment load combinations at -40°F ambient temperature were within the allowable limits.

Fracture toughness of the package containment boundary, gamma shield, and welds is evaluated in Appendix 2.10.4 of the SAR.

The package is designed for ambient temperatures as low as -20°F. The containment boundary materials (including lid bolts) were selected to meet the fracture toughness criteria of ASME Code, Section III, Division 3, Subsection WB. While the cask is designed to meet NB requirements, it is noted that the use of WB requirements for fracture toughness indicates that both the requirements of NB and the two brittle fracture Regulatory Guides recommended in NUREG-1536 have been met. ASME Table WB-2331.2-1 of Section III, Subsection WB (Para. WB-2330) is used to

determine the nil ductility transition temperature (TNDT) of the containment boundary design. The results indicate that TNDT for the 1.5-inch thick containment shell and bottom plate is -80°F ; TNDT for the 7.5-inch thick flange is -133°F ; and TNDT for the 5-inch thick lid plate is -126°F . In addition, Charpy V-Notch (CVN) testing of the containment boundary materials will also be conducted at a temperature no greater than 60°F above the TNDT to ensure that they will not be susceptible to brittle fracture at -20°F . The acceptance criteria is 35 mil lateral expansion and 50 ft-lb. absorbed energy. The fracture toughness requirements of the lid bolts will meet the criteria of ASME Code, Section III, Division 3, Subsection WB (Para. WB-2333). CVN testing will be performed at -20°F . The acceptance criteria is that the material exhibit at least 25 mils lateral expansion as per ASME Table WB-2333-1. The containment boundary welds will be examined by radiographic and either liquid penetrant or magnetic particle examinations in accordance with Section III, Subsection NB, Paragraphs NB-5210, NB-5220, and NB-5230.

The gamma shield is not part of the containment boundary, but provides structural support to the containment boundary during hypothetical accident conditions. Because of the package geometry, cracks in the gamma shield will not propagate into the containment boundary. The gamma shield will not separate from the containment boundary, due to the frictional forces between the containment vessel and the gamma shield which arise as a result of a shrink fit of the gamma shield shell over the containment shell. The allowable flaw sizes in the gamma shield design are calculated using a linear elastic fracture mechanics (LEFM) methodology, from Section XI of the ASME Code (1989 editions). The results of the fracture toughness analysis indicate that the critical flaw sizes (flaws large enough to give rise to rapid unstable extension) are larger than those typically observed in forged steel and plate components' flaws. Therefore, no special examination is required of the gamma shield to ensure the absence of flaws that would result in unstable crack growth or brittle fracture.

If the bottom plate weld were to fail, the bottom plate could become detached, which would impact the shielding capability of the cask. At -20°F , LEFM analysis indicates that the minimum allowable flaw sizes for surface and subsurface are 0.23 in. and 0.48 in., respectively. The applicant will inspect the package prior to first use as stated in Chapter 8 of the SAR.

Failure of the weld between the gamma shield and top flange would not have any safety significance since the gamma shield will not separate from the containment boundary. Failure of the weld between the top shield plate and lid could result in a drop of the top shield plate into the cask cavity. However, the top shield plate will still remain inside the containment boundary due to the package arrangement and would not lose its shielding capability. The inspection requirements in Chapter 8 of the SAR, specified for this location, are the same as that specified for the bottom plate weld.

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

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(2) for cold conditions.

2.6.3 Reduced External Pressure

The maximum normal operating pressure (MNOP) for the TN-68 transportation package is 30 psig. A reduced external pressure of 3.5 psia would produce a net effective internal pressure of 41.2 psig. However, the package is designed to withstand a maximum internal pressure of 100 psig. The applicant performed analyses using the ANSYS finite element model. In addition to the 100 psig internal pressure, lid bolt pre-loads, fabrication stress, gravity and trunnion local loads were also included. The thermal stresses for the hot thermal condition were also included in the load combination. The results indicated that the combined stresses due to reduced external pressure load combinations were within acceptable limits.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(3).

2.6.4 Increased External Pressure

The applicant performed an ANSYS finite element analyses to verify that an increased external pressure of 20 psia will have no adverse effect on the package. The applicant calculated that the minimum cask cavity pressure of 0 psia would result in a net external pressure of 20 psig. However, a 25 psi external pressure loading was used in the load combinations calculations. Lid bolt pre-loads, fabrication stresses, gravity and trunnion local loads were included. In addition, the thermal stresses for the -20°F minimum temperature were also included in the load combination calculations. The results indicated that the combined stresses due to increased external pressure load combinations were within acceptable limits.

The package would not buckle under increased external pressure of 20 psia because the package is capable of withstanding up to 670 psig without exceeding the allowable buckling stress as demonstrated under 10 CFR 71.61, special requirement for irradiated nuclear fuel shipments.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(4).

2.6.5 Vibration

The applicant evaluated the stresses due to both transport truck and transport rail vibrations. The peak inertia values used to evaluate transport truck vibration were obtained from truck bed accelerations in ANSI N14.23. The peak inertia values used to evaluate transport rail vibration were obtained from NUREG 76-6510. Since the peak inertia values for rail vibration loadings were less than the truck vibration loadings, the stresses obtained from the truck vibration analysis were used to perform both truck and rail car transport vibration load combinations. In addition, to stresses due to the truck vibration, transport truck vibration calculations included load combination due to lid bolt pre-loads, fabrication stresses, internal pressure, thermal stresses, and the trunnion effects. Transport truck vibration load combination calculations in a cold environment included lid bolt pre-loads, fabrication stresses, external pressure, thermal stresses, and

the trunnion effects. The results indicated that a positive margin of safety exists for vibratory effects.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(5).

2.6.6 Water Spray

All exterior surfaces of the TN-68 transportation package are metal. The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(6) and that the water spray will have no impact on package performance.

2.6.7 Free Drop

The TN-68 transportation package was analyzed by computer modeling for one foot top end, bottom end, and side drops. For the three drop cases, the evaluations were performed for both the high temperature environment and at the -20°F minimum transport temperature. In all cases, bolt pre-load effects and fabrication stress were included. For the hot environment condition, thermal stress load, 100 psig internal pressure, and impact load cases were combined. For the cold environment condition, thermal stresses, 25 psi external pressure, and impact load cases were combined. The design g-load for the one-foot end drops and side drop were calculated to be 12g and 32g respectively. However, the applicant used 15g for the end drop and 35g for the side drop in the stress calculations. The computer code used to determine the inertial g-loads is explained under the hypothetical accident conditions section below. The results indicated that the combined stresses in the cask body due to the one foot free drop load combinations were within acceptable limits.

The basket was also evaluated for 15g end drop and 35g side drop. The stress analysis of the basket due to inertial loading analysis is described in detail in Appendix 2.10.5 of the SAR. The results of the analyses indicated that various structural components for the basket assembly were structurally adequate for loads up to 40g. This g load is higher than the 15g used for one foot end drop and the 35g used for one foot side drop. Thus, the basket is structurally adequate and it will properly support and position the fuel assemblies under normal conditions of transport.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(7).

2.6.8 Corner Drop

This requirement, as delineated in 10 CFR 71.71(c)(8), is not applicable because the weight of the TN-68 transportation package exceeds 220 pounds and neither wood or fiber board are used as materials of construction.

2.6.9 Compression

This requirement, as delineated in 10 CFR 71.71(c)(9), is not applicable because the weight of the TN-68 transportation package exceeds 11,000 pounds.

2.6.10 Penetration

A 13-pound steel cylinder drop from a height of 40 inches, as delineated in 10 CFR 71.71(c)(10), would have no significant effect on the TN-68 transportation package since the more limiting case of a 40-inch drop of the entire package onto a puncture bar resulted in no detrimental consequence. The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(10).

2.6.11 Summary

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71.

2.7 Hypothetical Accident Conditions (10 CFR 71.73)

2.7.1 Thirty-Foot Free Drop

The package was analyzed by computer modeling for a free drop from a height of 30 feet onto an unyielding surface in the following orientations: (1) top end, (2) bottom end, (3) side, (4) center of gravity (C.G.) over the bottom corner, (5) C.G. over the top corner, and (6) 15° slap down impact at lid end. One-third scale model drop tests were also conducted for the bottom end drop, side drop, and 15° slap down impact at lid end to verify that the inertia g forces calculated by the computer model analyses were conservative.

The 30-foot drop orientations were selected to inflict the greatest possible cumulative damage to the package. Orientations 1 and 2 were chosen because they would generate the highest axial g load. Orientation 3 was chosen because it would generate the highest normal transverse g load as well as significant deformation. Orientations 4 and 5 were chosen because C.G. over the corner drop generates large stresses at the corner of impact. In addition, the C.G. over the top corner drop would also generate the largest stresses in the lid bolts. Orientation 6 was chosen because it would generate a high normal as well as rotational g-load at the ends of the cask (secondary impact). This orientation would also put the highest g-load on the impact limiter tie rods and attachment bolts and it is the limiting loading condition for the impact limiter attachments.

The impact force and the deceleration (g-load) for the 30-foot drop tests were determined by the ADOC (Acceleration Due To Drop On Covers) computer code. The ADOC code utilized the force-deflection curve for each drop orientation to predict analytically the rigid body responses of the package to a 30-foot drop impact in that orientation. These force-deflection relationships (curves) for the various drop

orientations were developed based on the crush geometry of the impact limiter and the wood mechanical properties. The force-deflection relationships for various impact angles are presented in Tables 2.10.8-5 through 2.10.8-10 of the SAR and were validated by actual force-deflection measurement during static crush tests of impact limiters with similar construction and size. The ADOC computer code output values have been compared with: (1) actual measured dynamic impact testing results of the one-third scale model of the package, (2) independent analytical calculation based on classical mechanics methods, and (3) other recognized computer codes, such as the SCANS. The cask rigid body accelerations and maximum impact limiter crush distances of the TN-68 transportation package as predicted by the ADOC computer code compared very well with those of the one-third scale model 30-foot drop test results and independent analytical calculations. Excellent agreement between results from ADOC and SCANS code were also obtained. Therefore, it can be concluded that the ADOC computer code was validated by the above comparisons.

Based on the results of analyses by the ADOC computer code, as supplemented by the 1/3 scale model tests, the maximum rigid body deceleration of the package is 75g from the 30-foot end drop orientation. The applicant used 80g in the evaluation of the package for the 30-foot end, side, and C. G. over the corner drops. It should be noted that the 80g is applied vertically for C. G. over the corner drop. Thus, a 69g load is applied along the longitudinal axis of the package, and a 40g load is applied perpendicular to the package longitudinal axis. The 15° slap down impact at lid end (second impact) generates 25.4g normal acceleration and 46.8g rotational acceleration. Both accelerations have been increased by roughly 66% from the calculated values to 42.22g normal and 77.78g rotational for the structural analysis.

The stress analysis of the cask body was performed using an ANSYS finite element model. The stress intensities for the cask body were obtained by combining or superimposing the stresses from appropriate individual load cases, specified in 10 CFR 71.73 and Regulatory Guide 7.8. In all cases, bolt pre-load effects and fabrication stresses were included. For the hot environment evaluation, thermal stresses, 100 psig internal pressure, and impact load cases were combined. For the cold environment evaluation, thermal stress, 25 psig external pressure, and impact load cases were combined. The combined stress intensities are presented in Tables 2-59 through 2-98 of the SAR. The staff noted that the thermal stresses and primary stresses were conservatively added together. The thermal stresses due to the hot and cold environments are usually considered to be secondary stresses which could be evaluated using higher allowable values than for primary stresses. However, the applicant evaluated the combined stresses using the primary stress allowables. Even with this conservative assumption, the results indicated that the highest stress intensities in the containment vessel and gamma shield were well below the allowable stress limits. In addition, the inner shell was also evaluated for buckling in accordance with ASME Code Case N-284. The results indicated that the allowable buckling stress of the inner shell is much higher than the stresses resulting from the 30-foot drop load. Therefore, there is no potential of buckling of the inner shell structure under the 30-foot drop condition.

Analysis of the fuel basket under hypothetical accident condition loads using ANSYS were provided in Appendix 2.10.5 of the SAR. Stress levels due to various 30-foot drop orientations were evaluated for the stainless steel boxes and plates, stainless steel

fusion welds, and the aluminum side rails. The stainless steel members were the primary structural components for the basket assembly. Aluminum side rails were attached to the periphery of the basket to establish and maintain basket orientation and to prevent twisting of the basket assembly. The strength of the poison plate in the basket assembly was neglected in the analyses. However, its weight was distributed on all four sides of stainless steel boxes. Stresses were compared to allowable limits of ASME Section III, Division 1, Subsection NG (Core Support Structures). The results indicated that the maximum allowable g-load for the stress in fuel basket (limited by stainless steel plate) was 107g. The maximum allowable g-loads for the buckling was 101g (limited by 30° buckling). Both of these g loads were higher than the 80g used for the 30-foot drop evaluation. Therefore, the basket will remain in place and maintain separation of the adjacent assemblies during the 30-foot drop condition.

The effect of 30-foot side and end drops on the integrity of fuel rod cladding were evaluated in Appendix 2.10.7 of the SAR. The methodology used in performing the 30-foot side drop analysis was based on LLNL report UCID-21246, "Dynamic Impact Effects on Spent Fuel Assemblies." The results indicated that stresses for different GE fuel due to 30-foot side drop were all less than the yield stress of the irradiated fuel cladding. For the 30-foot end drop, large displacement finite element analyses using ANSYS finite element code were employed. The analyses used a three-dimensional finite element model of the entire active fuel rod length. Intermediate transverse supports were placed along the fuel to simulate grid support effects. Results from these analyses indicated that the lowest buckling load for GE fuel assemblies was about 95g. The maximum g-load from 30-foot end drop was calculated to be 80g and the staff agrees with the applicant's evaluation that the fuel cladding tube will remain intact and retain the fuel pellets during the 30-foot accident drop conditions.

Closure bolts were evaluated in Appendix 2.10.2 of the SAR using the methodology of NUREG/CR-6007. The bolt preload condition was calculated to withstand the worst case load combination and to maintain a clamping force on the closure joint under both normal conditions of transport and hypothetical accident conditions. The preload and high temperature loads were used for bolt stress calculations in both load cases. The results of the analysis indicated that the maximum stresses were less than allowable values. Lid separation or loss of gasket compression would not occur during normal conditions of transport and accident condition loads since bolt preloads are higher than all applied loads. A fatigue analysis indicated that the cask lid bolts would not fail due to fatigue during transport if they are replaced after every 150 round trip shipments.

The behavior of impact limiter attachments during and after the 30-foot drop orientations were evaluated in Appendix 2.10.8. Based on the analyses using the ADOC computer code, a 15° shallow angle slap down 30-foot drop has the highest potential for detaching the impact limiter. The shallow angle drop would apply the greatest overturning moment onto the impact limiter attachments at the slap down end which tend to pull the limiter away from the cask. Other drop orientations were not critical cases for the impact limiter attachments because the impact force tends to push the impact limiter onto the cask in those orientations. Analysis using text book equations and one-third scale model testing were conducted for the 15° drop orientation to demonstrate that the affected impact limiter would remain in place to insulate the cask during the subsequent hypothetical accident conditions thermal test. The results

indicated that the impact limiter attachments were structurally adequate to ensure that the impact limiters remain attached in the 15° slap down drop orientation. Scale model tests of other orientations also verified that impact limiter attachments would hold the impact limiters on the ends during other drop orientations.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(c)(1).

2.7.2 Crush

Since the package weight exceeds 1,100 lbs, the dynamic crush test is not required.

2.7.4 Puncture

The ability of the package to adequately withstand the 40-inch puncture test condition was demonstrated by analysis using textbook equations and 1/3 scale model test.

An analysis of the 40-inch puncture bar test assumed that the puncture bar impinged directly on the center of the side walls of the gamma shield cylindrical shell, between the impact limiters. The outer shell and neutron shield would provide protection to the gamma shield cylindrical shell, but were conservatively neglected in the analysis. The results of the analyses indicated that the stresses at the center of the cask resulting from the inertial forces were negligible. The puncture pin would not penetrate the 7.5 inches thick cask wall.

A one-third scale model, with the impact limiter that was previously crushed during the 30-foot end drop test, was dropped in the 90° end drop orientation onto the 6 inch diameter puncture bar. The test unit C.G. was centered over the puncture bar. This orientation was chosen because it would assure that the puncture impact location absorbs all of the drop energy. Furthermore, the center of the impact limiter outer plate is the weakest portion of the impact limiter since there are no gussets in this location. The result of the test indicated that the impact limiter wood remains confined, and the maximum puncture depth represented only one-third the thickness of the impact limiter.

The vent and drain ports are small and located in solid steel forgings. They are protected by the impact limiters and solid steel cover. Closure of the ports is accomplished by bolts tightened to a prescribed torque. The 1/3 scale model puncture test onto the center of the impact limiter has demonstrated that puncture bar would not impinge on the ports. Thus, the ports are adequately protected from the puncture bar and no structural evaluation is performed.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(c)(3).

2.7.5 Thermal

The applicant performed an ANSYS transient thermal analysis to show the package can withstand the 30-minute fire test. The initial steady state assumed an ambient temperature of 100°F and maximum decay heat. The initial steady state condition is

followed by a 30-minute thermal transient which is then followed by a cool down period. The temperatures through the cross section of the cask, at the time of the maximum thermal gradient, was used for input to the cask model for thermal stress analysis. The stress component results of these analyses were presented in Tables 2.10.1-39 and 2.10.1-40 of the SAR and were also combined with those due to the lid bolt pre-load, the internal pressure, and fabrication stress. Although the maximum internal pressure during the fire was calculated to be 47 psig, 100 psi internal pressure was used for the pressure stress calculations. The combined stress intensities were presented in Tables 2-99 and 2-100 of the SAR. These combined stress intensities were all within the ASME code allowable levels. The applicant concluded that the TN-68 transportation package would maintain containment during the hypothetical accident conditions thermal test. The neutron shield is assumed to be lost after the fire event. Post-fire shielding analysis is provided in Chapter 5 of the SAR.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(c)(4).

2.7.6 Immersion - Fissile Material

The applicant's criticality evaluation assumed water in-leakage and optimum hydrogenous moderation of the contents. The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(c)(5).

2.7.7 Immersion - All Packages

The cask body stresses for the 50-foot head of water, 36.4 psi external pressure immersion condition is bounded by the special requirement for irradiated nuclear fuel shipments (water pressure of 290 psi) described below.

2.7.8 Special Requirement for Irradiated Nuclear Fuel Shipments (10 CFR 71.61)

The ability of the package to adequately withstand an external water pressure of 290 psig for a period of not less than one hour without collapse, buckling, or inleakage of water was demonstrated by using ANSYS finite element code. The 290 psig external pressure was applied to the outer surface of the containment vessel, although the containment vessel is completely enclosed by the gamma shield and is not exposed to an external pressure due to immersion. The results indicated that stresses in the inner cylinder shell, bottom plate, the lid, and flange were all within the ASME code allowable levels. Additional analysis was also performed to evaluate the buckling of the inner cylindrical shell using the ASME Code Case N-284. The results indicated that the 1.5 inches thick inner cylindrical shell can withstand up to 670 psi without exceeding the allowable buckling stress.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.61.

2.7.9 Summary

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73.

2.8 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the structural design has been adequately described and evaluated and that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The objective of this review is to verify that the thermal performance of the package has been adequately evaluated for the tests specified under normal conditions of transport and hypothetical accident conditions and that the package design satisfies the thermal requirements of 10 CFR Part 71.

3.1 Description of the Thermal Design

3.1.1 Thermal Packaging Design Features

The thermal design features of the package include the following:

- A. Helium backfill gas for heat conduction which also provides an inert atmosphere to prevent fuel cladding oxidation and degradation;
- B. Minimal heat transfer resistance through the basket by sandwiching aluminum neutron poison plates between the stainless steel fuel compartments. The compartments are fusion welded to 1.75 in. wide stainless steel plates. Above and below the plates are slotted poison plates, which form an egg crate structure providing good paths for heat transfer from the fuel assemblies, along the plates, to the aluminum basket rails.
- C. The basket rails are bolted to the basket periphery providing a good conduction path from the basket to the cask cavity wall.
- D. Aluminum boxes filled with a resin compound are placed around the cask gamma shell and enclosed by an outer shell. The boxes provide for neutron shielding and increase the thermal conductance through the neutron shield layer.
- E. High emissivity paint on the exterior cask surface to maximize radiative heat transfer to the environment.

3.1.2 Codes and Standard

Where appropriate, codes and standards were referenced by the applicant. For standard materials, the ASME Code is referenced by the applicant.

3.1.3 Content Heat Load Specification

The applicant analyzed the transportation cask for BWR fuel assembly types. The package was analyzed based on a maximum decay heat of 0.313 kW per fuel assembly from 68 BWR fuel assemblies, for a maximum decay heat of 21.2 kW for the whole cask. The maximum assembly average burnup is 40,000 MWD/MTU and the minimum cool time is 10 years. The minimum cool time varies with enrichment and burnup, as shown in Tables 1-2, 1-3 and 1-4 of the SAR.

The applicant used the SAS2H/ORIGEN-S code to determine the assembly decay heat load using burnup, enrichment, and cooling time of the fuel. The method the applicant used to determine heat load was reviewed by the staff and found to be adequate.

3.1.4 Summary Tables of Temperatures

The summary tables of the temperatures of package components, contained in table 3-1 of the SAR, was verified to include the impact limiters, containment vessel, seals, shielding, and neutron absorbers. The temperatures were consistent with those presented throughout the SAR for both the normal conditions of transport and hypothetical accident conditions. The staff also confirmed that the summary tables contained the design temperature limits for each of the critical components of the package. For hypothetical accident conditions, the applicant accounted for the pre-fire, fire and post-fire component temperatures. With the exception of the impact limiters and the neutron shield, which are not critical components needed to function after the hypothetical accident condition fire test, all components remain below their material property temperature limits.

3.1.5 Summary Tables of Pressures in the Containment System

The pressure calculation of the containment system under the normal conditions of transport and hypothetical accident conditions were reviewed and found consistent with the pressures presented in the General Information, Structural Evaluation, and Thermal Evaluation sections of the SAR. The design basis pressure was reported along with the Maximum Normal Operating Pressure (MNOP) and the hypothetical accident condition pressure.

3.2 Material Properties and Component Specifications

3.2.1 Material Properties

The application provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the cask. Conservative thermal emissivities were used to model the radiative heat transfer to and away from the

package. The thermal properties used for the analysis were appropriate for the materials specified. Additionally, the fluid properties of the surrounding air were provided in the evaluation of thermal convection parameters. These properties were appropriate for the conditions of the cask required by 10 CFR Part 71.

3.2.2 Technical Specifications of Components

References for the technical specifications of pre-fabricated package components for O-rings, impact limiters and neutron absorber materials were provided by the applicant. All components were shown to satisfactorily perform under normal conditions of transport with an ambient temperature of -40°F .

3.2.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of cask containment, radiation shielding, and criticality were specified.

The applicant proposed a maximum allowable fuel cladding temperature of 716°F , which is well below the maximum allowable fuel cladding temperature of 1058°F from the Pacific Northwest National Laboratory report, PNL-4835, which has been accepted by the NRC staff.

3.3 Thermal Evaluation under Normal Conditions of Transport

3.3.1 Heat

Under normal conditions of transport, all of the materials used remain below their respective failure temperatures. The applicant performed three steady-state calculations for normal conditions of transport using the ANSYS computer code. These calculations provided steady-state temperature distributions for the following combined boundary conditions: (1) an ambient temperature of 100°F with solar insolation and maximum decay heat, and (2) an ambient temperature of -40°F with no solar insolation and no decay heat.

The applicant modeled the package using a three-dimensional 30° symmetric section of the cask body and impact limiters. The model includes the cask body, neutron shield, lid, impact limiters and thermal shield.

The applicant also modeled the basket using a 90° symmetric section of the basket and cask. The applicant explicitly modeled the packaging and homogenized the fuel assemblies. A schematic of the model is shown in Figures 3-1 through 3-3 of the SAR.

For normal conditions of transport, the applicant performed a steady-state evaluation of the entire cask. This analysis produced a maximum cladding temperature of 490°F which remains well below the limit of 1058°F . The maximum seal temperature under normal conditions is 366°F , which is substantially below the extended exposure limit of 536°F .

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(1).

3.3.2 Cold

With no decay heat and an ambient temperature of -40° F, the entire package will maintain a steady-state temperature of -40° F. Cask components, including the containment system seals, would not be adversely affected by this temperature.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71(c)(2).

3.3.3 Maximum Normal Operating Pressure (MNOP)

The applicant calculated the MNOP, assuming that 100% of the fuel rods fail and that 30% of the gaseous fission products are available for release. The total gas volume considered the gaseous fission products, the helium fill gas, and the cavity back-fill gas. The gaseous fission products were based on the maximum fuel burnup.

The average gas temperature was calculated to be 366°F. Based on this gas temperature, the MNOP was determined to be 45.6 psig.

3.3.4 Evaluation of Accessible Surface Temperature

Under normal conditions of transport, the package is enclosed by a protective screen to ensure that the accessible surface remains well below a temperature of 185° F. No solar insolation was applied to the package in making this determination.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.43(g) and must be shipped as exclusive use.

3.3.5 Confirmatory Analyses

A COBRA-SFS 1/8-symmetry model was developed to independently confirm the applicant's calculated temperatures for the package. A steady-state calculation was performed for the cask in the transportation configuration with the 21.5 kW decay heat distributed equally among the 68 BWR spent fuel assemblies. A 100°F ambient temperature was assumed. A comparison of the applicant's results with the COBRA-SFS predictions is presented in table 3-1 of the Safety Evaluation Report (SER).

There may be several differences between the COBRA-SFS model and the applicant's model. One known difference is that the COBRA-SFS model neglected heat transfer out the top and bottom of the casks, requiring all heat transfer to be through the cask sides. This may result in the 30°F higher value in the cask outer shell temperature predictions. If the COBRA-SFS predictions are altered by 30°F, the differences become much smaller with only the neutron shield (-16°F) being more than an 8°F temperature difference from the applicant's calculation.

The COBRA-SFS results are believed to be conservative and are well within the temperature limits set forth in the application.

3.3.6 Summary

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.71.

3.4 Thermal Evaluation under Hypothetical Accident Conditions

3.4.1 30-Minute Thermal Test

The applicant performed a 30-minute transient thermal analysis using the ANSYS computer code to evaluate the package under hypothetical accident conditions. The evaluation was based on the amount of damage to the impact limiter from the 30-foot drop and the 40-inch puncture tests.

The basket model used in the normal conditions of transport evaluation was modified to determine the maximum temperatures of the packaging internals and the maximum fuel temperature. In order to determine the maximum temperature of the seals from the damaged package, the applicant modeled the damage to the impact limiters from the drop tests using two three-dimensional 180° finite element models. The models included the cask shells, neutron shield, impact limiter spacer, thermal shield and the damaged impact limiters. The two models encompass two configurations of the package after the drop tests. The two configurations include damaging the package to crush the impact limiter to the maximum depth at both the lid seals and the port seals.

Many of the peak temperatures were not realized until the package reached post-fire steady-state conditions. Therefore, some of the hypothetical accident conditions temperatures shown in the table below reflect the peak temperature of a specified component in the period following the 30-minute fire. The post-fire transient was evaluated for a period of approximately 25 hours to observe the cooling of the package to post-fire steady-state temperatures. The peak cladding temperatures were determined using the same method that was used for normal conditions of transport.

For hypothetical accident conditions, the analyses produced a maximum cladding temperature of 423°F, which is below the limit of 1058° F. Under these conditions, the maximum seal temperature was shown to be 383°F. This seal temperature for the fire accident is below the maximum temperature limit of 536°F.

3.4.2 Maximum Pressure

The applicant calculated maximum pressure under hypothetical accident conditions is 47 psig, based on the average cavity gas temperature of 423°F. This pressure is well below the 100 psig design pressure of the package.

3.4.3 Confirmatory Calculations

The COBRA-SFS steady-state predictions were used as initial conditions for the 30-minute fire transient. As was the case for the steady-state calculations, heat transfer into the cask top and bottom was neglected due to the presence of the impact limiters. A summary of the maximum component temperatures is provided in table 3-1. The differences in the predictions are small with the exception of the peak radial neutron shield temperature. The cobra temperature reported is for the center of the neutron shield, which will be at a lower temperature than for the outer radius. This may explain the large temperature difference between the COBRA-SFS calculation and the applicant's calculation.

As in the steady-state predictions, the applicant's predictions of maximum temperatures are very similar to the COBRA-SFS values and within the stated limits.

3.4.4 Effects of Uncertainties

The staff considered the applicant's thermal evaluations and ensured that they addressed the effects of uncertainties in thermal and structural properties of materials, test conditions and diagnostics, and in analytical methods. Because of significant design margins, the staff agreed with the applicant's conclusion that appropriate values were used throughout the application.

3.5 Summary

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(4).

3.6 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the design has been adequately described and evaluated and that the package has meets the thermal performance requirements of 10 CFR Part 71.

Table 3-1, TN-68 Maximum Calculated Temperatures (°F)

Location	Normal Conditions of Transport		Hypothetical Accident Conditions	
	TN	NRC	TN	NRC
Impact Limiter	175	----	N/A	---
Cask Outer Surface	204	234	1000	1225
Neutron Shield	244	258	982	909
Cavity Gas	366	---	423	---
Inner Shell	262	294	371	366
Basket Rail	319	341	390	388
Basket Plate	469	496	537	545
Fuel Cladding	490	513	555	562

4.0 CONTAINMENT REVIEW

The objective of this review is to verify that the package design satisfies the containment requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

4.1 Description of the Containment System

4.1.1 Containment Boundary

The containment system of the package consists of the following components: (1) the inner shell, (2) the bottom end closure plate, (3) welded flange forging, (4) the top closure plate, (5) the lid bolts, (6) the top closure inner metallic O-ring seal, (7) the vent and drain cover plates, (8) the vent and drain cover plate O-rings, and (9) the vent and drain port cover plate bolts. Table 4-1 lists containment boundary components and their material of construction.

Table 4-1: TN-68 Containment System Components		
COMPONENT	MATERIAL	Item No. from 972-71-2 and -4, both REV. 2
Inner Shell	SA-203, Grade E	3
Bottom End Closure	SA-203, Grade E	5
Welded Flange Forging	SA-350, Grade LF-3	35
Top Closure Plate	SA-350, Grade LF-3 or SA-203, Grade 3	2
Closure Lid Bolts	SA 540 Grade B24 Class 1	14
Top Closure Inner O-Ring Seal	Metallic - aluminum jacket, ss304 liner, Nimonic 90 or equivalent spring	16
Drain and Vent Port Closure Plates	SA-240, Type 304	22/23
Drain and Vent Port O-Ring Seal	Metallic - aluminum jacket, ss304 liner, Nimonic 90 or equivalent spring	24
Drain and Vent Port Bolts	SA 193, Grade B-7	25

The containment system is designed to a leakage rate of 1×10^{-5} std-cm³/s or less.

All containment seals are metallic, static face O-ring seals. The top closure, drain and vent port closure plates are equipped with dual O-ring seals. The inner O-rings are the containment seals and are made with an aluminum jacket, a 304 stainless steel liner, and a Nimonic 90 (or equivalent) spring. The metallic outer O-rings facilitate leak testing of the inner containment O-rings. The seals on the vent and drain ports are the same as the lid. All containment seals are leak tested in accordance with ANSI-N14.5 and replaced after each use.

The top closure plate is closed with 48, 2-inch diameter bolts. The vent and drain ports closure plates are closed with 8, 1-inch diameter bolts. Bolt torque or preload values are specified in TN Drawing No. 972-71-2, Rev. No. 2.

4.1.2 Codes and Standards

All containment welds are in accordance with ASME Code Section III, Division 1, Subsection NB Article 3200. The exceptions to the code are listed in Drawing 972-71-14, Rev. 1.

The staff has reviewed the description of the containment system, as shown in Chapters 1 and 4 of the SAR. The staff findings include: (1) the SAR describes the containment system in sufficient detail to provide an adequate basis for its evaluation; (2) the SAR identifies established codes and standards for the containment system; (3) the containment system is securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package; and (4) the containment system is made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction.

4.1.3 Special Requirements for Damaged Spent Nuclear Fuel

Failed fuel is not considered in this review, therefore, this section is not applicable.

4.2 Containment Under Normal Conditions of Transport

4.2.1 Pressurization of Containment Vessel

Within the thermal evaluation, the applicant demonstrated, and the staff confirmed that the pressure under normal conditions of transport would not exceed the package design pressure of 100 psig.

4.2.2 Containment Criteria

The containment system is designed to a leakage rate of 1×10^{-5} std-cm³/sec or less. The applicant calculated the releaseable radiological source term and the maximum allowable leak rate. The applicant assumed a rod failure rate of 3% and a release rate of the fission products and actinide from the pellet to the gap of 30%. In accordance with ANSI 14.5, fabrication verification, periodic verification, and assembly verification leak tests will be performed to verify the containment capability of the containment system.

4.2.3 Compliance with Containment Criteria

Results of the applicant's structural and thermal analyses show that the containment system retains the capability to maintain a seal of 1×10^{-5} std-cm³/sec or less under the conditions specified in 10 CFR 71.71. Therefore, staff agrees with the applicant's conclusion that the loss or dispersal of radioactive material from the cask will be less than 10^{-6} A₂ per hour under normal conditions of transport, as required in 10 CFR 71.51(a)(1).

4.3 Containment Under Hypothetical Accident Conditions

4.3.1 Pressurization of Containment Vessel

Within the thermal evaluation, the applicant demonstrated, and the staff confirmed that the pressure under hypothetical accident conditions would not exceed the package design pressure of 100 pig.

4.3.2 Containment Criteria

The containment system is designed to a leakage rate of 1×10^{-5} std-cm³/sec or less. The applicant calculated the releasable radiological source term and the maximum allowable leak rate under hypothetical accident conditions. The applicant assumed a rod failure rate of 100% and a 30% release rate of the fission products and actinides from the fuel pellets.

4.3.3 Compliance with Containment Criteria

Results of the thermal analysis show that seal temperatures will remain below the seal material temperature limits during and after the 30-minute fire. Results of the structural analysis show that the cask inner shell will not buckle under hypothetical accident conditions.

Overall, results of the structural and thermal analyses also showed that the containment system remained intact under the tests specified in 10 CFR 71.73. Therefore, as required in 10 CFR 71.51(a)(2), the escape of krypton would not exceed 10 A₂ in 1 week, and the escape of other radioactive materials would not exceed A₂ in 1 week under hypothetical accident conditions.

The staff agrees with the applicant's conclusion that the containment system meets the requirements of 10 CFR 71.51(a)(2).

4.4 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the design has been adequately described and evaluated and that the package meets the containment performance requirements of 10 CFR Part 71.

5.0 SHIELDING

The objective of this review is to verify that the package design meets the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

5.1 Shielding Design Description

Gamma-radiation shielding for the TN-68 shipping cask is provided around the walls and the bottom of the containment vessel by a forged steel cask body and bottom plate of carbon steel, respectively. The gamma shield completely surrounds the containment vessel inner shell and bottom closure. The 6-inch thick gamma shielding cylinder is SA-266 (Class 2) and the 8.25-inch thick bottom is SA-266 (Class 2) or SA-516 (Grade 70). A 4.5-inch thick gamma shield is also welded to the inside of the containment lid.

Neutron-radiation shielding is provided by a borated polyester resin. The resin compound is cast into long, slender aluminum (6063-T5) containers. The total thickness of the resin and aluminum is six inches. The array of resin-filled containers is enclosed

within a smooth 0.75-inch thick outer steel shell (SA-516, Grade 70) constructed of two half cylinders. This array surrounds the gamma shield and is radially adjacent to the fuel region.

5.2 Radiation Source

The contents for the package are limited to intact 68 General Electric BWR fuel assemblies with zircalloy cladding. The applicant defines an intact fuel assembly as being a spent nuclear fuel (SNF) assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. SNF assemblies with missing fuel rods cannot be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water equal to that displaced by the original rods. The fuel may be transported with or without channels. The permissible fuel assembly types are presented in Section 1.2 of the SER.

Fuel assemblies are categorized into three types, Type I, Type II and Type III. There are two basic loading configurations for the package. The first configuration is a mixture of Type I and Type II fuel assemblies. The second configuration is Type III fuel assemblies.

In the mixed Type I and Type II configuration, Type I assemblies shall be placed only into the interior compartments of the fuel basket as shown in figure 5.3-3 of the application. Type II fuel assemblies may be placed in any basket fuel compartment. In the second configuration, Type III fuel assemblies may be placed in any basket fuel compartment.

A Type I SNF is defined as a bounding GE BWR fuel assembly that produces a source term equivalent to the source term generated via the following combination of parameters: a) a maximum initial lattice-average enrichment is 3.7%; b) the minimum initial bundle average enrichment is 3.3.%; c) the maximum assembly average burnup is 40,000 MWD/MTU; d) the minimum cool time is 10 years; and e) the maximum heat load per assembly is 0.313 Kw. The minimum cooling times required for various combinations of burnup and minimum initial enrichments for Type I fuel assemblies can be found in Table 1-2 of the SAR

A Type II assembly has a combination of burnup, initial enrichment, and cooling time that yields a source term that is considerably less than the source term produced for Type I assemblies. Type II assemblies may be placed in any basket fuel compartment and must be placed in the exterior basket fuel compartments when combined with Type I fuel assemblies. The minimum cooling times required for various combinations of burnup and minimum initial enrichments for Type II fuel assemblies can be found in Table 1-3 of the SAR. The combination of Type I and II assemblies placed into the TN-68 cask shall not yield external radiation levels that exceed the requirements of 10 CFR 71.47.

A Type III assembly has the equivalent source term/dose rate to fuel that has burnup of 40,000 MWT/MTU, a 3.3% initial enrichment, and has been cooled for 16 years. The

minimum cooling times required for various combinations of burnup and minimum initial enrichments for Type III fuel assemblies can be found in Table 1-1 of the SAR.

The applicant calculated the neutron and gamma source strengths for the applicable fuel assemblies using the SCALE-4.3 SAS2H and ORIGEN-S computer modules. The applicant determined that the 7x7 BWR fuel assembly provides the most conservative source term. Thus, the 7x7 BWR fuel assembly source term is considered to be bounding and was utilized in the shielding analysis.

The calculated source terms include radioactive isotopes in both the active fuel and activated hardware. Source terms were evaluated for the active fuel, plenum zone, top-fitting zone, and bottom-fitting zone of the SNF assemblies. Appropriate axial burnup profile parameters were applied for the design basis fuel in the source term modeling.

The applicant's methods for calculating the radiation source terms were reviewed. The staff used the SCALE-4.4 SAS2H and ORIGEN-S computer modules to perform verification analyses and found acceptable agreement with the applicant's reported values.

5.3 Shielding Model

The applicant demonstrates compliance with the external radiation requirements specified in 10 CFR Parts 71.47 and 71.51 by using dose rate modeling. The MCNP-4B computer code was utilized by the applicant to calculate dose rates at various locations around the TN-68 cask. The applicant calculated dose rates for various combinations of burnup, initial enrichment, and cooling time. The dose rate results are presented in Section 5 of the applicant's SAR. All of the applicant's reported dose rate results are below the applicable 10 CFR Part 71 regulatory limits for exclusive-use transport.

The applicant calculated the dose rates for normal conditions of transport and hypothetical accident conditions using two loading scenarios. The first loading pattern used the bounding spent nuclear fuel cooled 16-years (GE 7x7, burnup 40,000 MWD/MTU, with 3.3% enrichment, Type III). The second loading scheme included 44 design basis assemblies, cooled 10 years (Type I) and 24 assemblies, burned 21.5 GWD/MTU and cooled 26 years (Type II). Table 5-1 and 5-2 show the maximum calculated doses for both of these loading scheme.

Table 5-1, Calculated doses for the TN-68 package for Normal Conditions of Transport

Normal Conditions	Package Surface (mrem/h)	Vehicle Edge (mrem/h)	2 meters from Vehicle (mrem/h)			
Radiation	Type III	Type I and II	Type III	Type I and II	Type III	Type I and II
Gamma	104	135	56	72	9	8.2
Neutron	18	15.4	8	6.7	1	0.7
Total	122	151	64	79	10	8.9
Limit	1000	1000	200	200	10	10

Table 5-2, Calculated Doses for the TN-68 package for Hypothetical Accident Conditions

Hypothetical Accident Conditions (either configuration)	1 meter from package surface (mrem/h)
Radiation	
Gamma	78
Neutron	247
Total	325
Limit	1000

5.4 Shielding Evaluation

Utilizing parameters provided in the SAR, the staff performed confirmatory shielding analyses using the MCNP-4C and SCALE-4.4a SAS2H/ORIGEN-S computer codes. These analyses were performed to ensure that the applicant's source term and dose rate calculation methodologies were satisfactory. The staff's dose rate results were found to be slightly lower (within 20%) than those reported by the applicant. The primary reason for the dose rate differences is attributable to the setup of the source term regions in the cask models. The applicant homogenized the fuel and basket regions whereas the staff explicitly modeled these regions. As a result, the applicant's dose rate calculation methodology is considered to be more conservative relative to the staff's methodology and was deemed to be satisfactory. The dose rate results

calculated by the staff and applicant were found to be below the applicable regulatory limits specified in 10 CFR 71.47 and 71.51.

5.5 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the design has been adequately described and evaluated and that the package meets the shielding performance requirements of 10 CFR Part 71.

6.0 CRITICALITY REVIEW

The objective of the review is to ensure that the package design satisfies the criticality safety requirements of 10 CFR Part 71.

6.1 Description of Criticality Design

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

6.1.1 Packaging Design Features

The packaging design features relied upon to prevent criticality are the basket geometry and the fixed neutron poisons in the basket. A minimum basket fuel cell opening of 5.97 inches by 5.97 inches and a minimum Boron-10 (^{10}B) areal density are assumed. The minimum ^{10}B areal density is 0.030 g/cm² for the borated aluminum alloy and 0.036 g/cm² for the B₄C-aluminum composite material. Results of the structural and thermal analyses show that the packaging design features important to criticality safety are not adversely affected by the tests specified in 10 CFR 71.71 and 71.73.

6.1.2 Codes and Standards

The criticality evaluation is consistent with the appropriate codes and standards for nuclear criticality safety. The criticality evaluation is also consistent with the recommendations provided in NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages."

6.1.3 Summary Table of Criticality Evaluations

Tables 6.4-3 through 6.4-5 of the SAR contain a summary of the final analysis results. These results are for a single package and for arrays of damaged and undamaged packages, as required by 10 CFR 71.55 and 71.59. The results illustrate that the package meets the criticality safety criteria of 10 CFR Part 71 and that the package would remain subcritical under normal conditions of transport and hypothetical accident conditions.

The maximum k_{eff} for each condition, as calculated by the applicant, is summarized in Table 6-1 below. The results are below the Upper Subcritical Limit (USL) which is

0.9331. The USL incorporates the biases and uncertainties of the model and computer code into a value that has a 95% confidence level such that any $k_{\text{eff}} + 2\sigma$ less than the USL is less than 0.95. The results of the staff's confirmatory calculations are in close agreement with the applicant's results.

**Table 6-1
Maximum k_{eff} (USL = 0.9331)
SCALE 4.3 Code Results**

Condition	$k_{\text{eff}} + 2\sigma$
Single Package, Flooded 10 CFR 71.55(b), (d), and (e)	0.9256
Infinite Array of Undamaged Packages, Dry 10 CFR 71.59(a)(1)	0.9266*
Infinite Array of Damaged Packages, Flooded 10 CFR 71.59(a)(2)	0.9266*

* The analysis for damaged packages was used to bound the undamaged packages

6.1.4 Transport Index

The criticality analysis shows that an infinite rectangular array of undamaged or damaged packages will remain subcritical with close full-water reflection and optimum interspersed hydrogenous moderation. Therefore, per 10 CFR 71.59(b), the criticality transport index for the TN-68 package is 0.

6.2 Spent Nuclear Fuel Contents

The TN-68 is designed to transport a maximum of 68 General Electric (GE) BWR spent fuel assemblies. The GE designations, along with the lattice specification, describes the mechanical characteristics of the assembly. The maximum assembly average burnup impacts the cladding integrity. The number of fuel rods, pitch, uranium mass, lattice-averaged initial enrichment, and channel thickness are controlled to maintain subcriticality. The lattice-averaged initial enrichment is the average enrichment of the pins across the assembly at any axial plane. Use of an average axial enrichment or the average bundle enrichment is not allowed. The assembly parameters are important to criticality safety are listed in Section 1.2 of the SER.

Specifications on the fuel condition are also included such that fuel with cladding defects greater than pinhole leaks and hairline cracks may not be loaded into the package cask. Assemblies with missing pins are not allowed unless the missing pin is replaced by a fuel pin or dummy pin that displaces an equivalent or greater volume.

The staff reviewed the description of the spent nuclear fuel contents and agrees that all relevant specifications have been provided. The staff also verified that the

specifications used in the criticality safety analysis are consistent with or bound those given in SAR Section 1.2.3.

6.3 General Considerations for Evaluations

6.3.1 Model Configuration

The TN-68 packaging was modeled as infinite in length with full water reflection outside of the gamma shield. The models are based on the engineering drawings in section 1.4 of the SAR and considered the dimensional tolerances except for the plate and tube materials which are very small. The impact of these tolerances on the criticality safety analysis is negligible. The basket's borated aluminum poison material, considered in all scenarios, was generally modeled as the borated aluminum alloy, except as described below. The calculations used 90% of the minimum ^{10}B areal density for the borated aluminum absorber and 75% for the metal matrix composite absorber. Due to modeling constraints of KENO, the aluminum spacers and water between the fuel basket and the cask wall were modeled with more aluminum and less water than is actually present. This is conservative since less water at the perimeter increases the reactivity. The applicant showed that the other minor model simplifications did not statistically change k_{eff} . Hypothetical accident conditions do not adversely affect the cask design with respect to criticality safety therefore, the models for the normal and hypothetical accident conditions are the same.

The active fuel region was modeled explicitly except it was assumed to be infinite in length. The applicant did not take credit for fuel burnup or burnable absorbers in the fuel. In Appendix 2.10.7 of the SAR, the applicant showed that an assembly burnup of 40,000 MWD/MTU will not result in fuel cladding failure during the cask drop accidents. This is bounding for all normal and hypothetical accident conditions since the spent fuel is subjected to maximum g-loads during cask drops.

The staff reviewed the applicant's models and agrees that they are consistent with the description of the cask and contents given in SAR Sections 1 and 6, including engineering drawings. The staff also reviewed the applicant's calculations, and results for determining the worst-case manufacturing tolerance and fuel condition. Based on the information presented in the SAR, the staff agrees that the most reactive combination of cask parameters and dimensional tolerances was incorporated into the calculation models.

6.3.2 Material Properties

The compositions and densities for the materials used in the criticality safety analysis models are provided in Section 6 of the SAR. No credit was taken for burnable absorbers in the fuel. The applicant's calculations take credit for 90% of the minimum required ^{10}B areal density in the borated aluminum and 75% of the metal matrix composite in the basket absorber material. The fabrication requirements and acceptance criteria outlined in SAR Section 8.1.6 justify the use of 90% ^{10}B credit for the borated aluminum alloy. The acceptance criteria require more test samples for the borated aluminum alloy compared to the metal matrix composite. In addition, neutron transmission and neutron radiography or radiography are required on the borated

aluminum coupons. See SER Section 8 for further discussion of the qualification and acceptance tests of the poison material.

The basket materials do not degrade such that there is any impact on criticality safety. The neutron absorber is a borated aluminum alloy or metal matrix composite material sandwiched between aluminum and steel plates that meet all structural and thermal requirements and can be expected to have no significant erosion or corrosion. A structural analysis was performed which demonstrates that the basket plates will remain in place during all accident conditions. The neutron flux in the package is very low such that depletion of the ^{10}B is negligible.

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

6.3.3 Computer Codes and Cross-Section Libraries

The applicant utilized the CSAS modules of the SCALE version 4.3 computer codes and the accompanying 27-group cross-section library for the package analysis and the benchmark calculations. The SCALE code is an industry standard for performing criticality analyses. Thus, the staff agrees that the code and cross-section sets used are appropriate for this particular cask design and contents.

6.3.4 Demonstration of Maximum Reactivity

A number of parametric cases were analyzed to determine the most reactive model for normal conditions of transport (single cask). First, the results of a uniform enrichment model with all pins enriched to 3.7 wt% U-235 and 23 variable enrichment models with an average enrichment of 3.7 wt% U-235 were compared for the 7x7, 8x8 and 9x9 assemblies. The uniform enrichment model results were $.0032 \pm .0037$ higher in k_{eff} . To account for this, the applicant reduced the upper subcritical limit by 0.0042. No positive biases were applied.

Second, the reactivities of 68 7x7, 8x8, 9x9 and 10x10 assemblies were compared. Several of the fuel generations have the same parameters important to criticality safety such that one assembly model bounds more than one fuel generation. Overall, fifteen different assembly models were created. The reactivity of each assembly model was calculated with and without fresh water in the fuel-pellet annulus of the fuel rods. The scenarios with water in the annulus were always more reactive and was included in all subsequent calculations. In addition, the cases were modeled without a fuel channel and with fuel channel thicknesses varying from 0.065" to 0.120". The 10x10 generation 12 model; the 8x8 generations 9, 9b and 10 models; and the 7x7 generations 2 and 2b models had the highest reactivities.

The following parametric calculations were performed for the three fuel types with the highest reactivity:

- A. The assemblies were shifted toward the center of the cask, which was found to be more reactive than the case with the assemblies centered in

the basket compartments due to increased interaction between fuel assemblies,

- B. The use of the minimum tolerances on the basket fuel compartment size was more reactive than the use of nominal dimensions due to increased interaction between the assemblies,
- C. The density of the fresh water in the cask was also varied with full density water resulting in the highest reactivity in all cases,
- D. The fuel channel thickness was varied from 0" to 0.120" with the thicker channel being most reactive in all cases,
- E. One case was also modeled with the metal matrix composite absorber which was slightly less reactive than the equivalent model with the borated aluminum absorber.

The normal condition of transport model combined the most reactive conditions from the parametric studies. Thus, the three most reactive assembly models assume the following; fuel assemblies off-center in the basket compartment, minimum basket compartment sizes, borated aluminum absorber, full density water, and pellet-clad fuel pin annulus contains full density water.

The scenario where the cask is partially filled with water and partially filled with steam was not analyzed as this will not increase reactivity in this package; optimum internal moderation occurs with full density water. The interior of the package does not allow for preferential or uneven flooding, therefore this scenario was also not analyzed.

Based on the applicant's results and the staff's independent confirmatory calculation as discussed below, the staff concludes that the most reactive combination of parameters and dimensional tolerances has been considered.

6.3.5 Confirmatory Analyses

The staff used the Monte Carlo N-Particle (MCNP) code version 4B and its associated cross-section set developed at Los Alamos National Laboratory for the confirmatory analysis. The MCNP code is routinely used for performing criticality analyses and is appropriate for this particular application and fuel system.

The staff's independent calculations were based on the information provided in the SAR. Specifically, the staff used Drawing Nos. 972-71-5, Revision 0 and 972-71-6, Revision 0. Confirmatory calculations with and without the simplifications in the cask model made by the applicant were performed for the 10x10 fuel assembly. The results showed the applicant's model resulted in a higher k_{eff} and therefore is bounding. The staff's analysis also confirmed that the following all increase k_{eff} ; full density water moderation in the cask, the use of minimum compartment sizes and basket tolerances, locating the fuel off-center in the basket compartments towards the center of the cask, and water in the pellet-clad annulus of the fuel. The only variation between the staff's confirmatory analysis and the applicant's analysis concerned the most reactive assembly. The staff's

results indicated that the 7x7 generation 2 and 2b assembly was slightly more reactive than the 10x10 assembly whereas the applicant's results indicated the 10x10 assembly had the highest reactivity. However, this type of variation can be expected between two different computer codes. The applicant performed the accident analysis for both the 7x7 and the 10x10 assemblies to ensure subcriticality of both fuel types in the TN-68 cask. In addition, the applicant adequately benchmarked the SCALE codes against critical experiments and appropriately applied code biases and uncertainties to the SCALE results. Overall, the confirmatory analysis performed by the staff is in close agreement with the applicant's results for the TN-68 cask.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff concludes that the package will remain subcritical, under normal conditions of transport and hypothetical accident conditions.

6.4 Single Package Evaluation

The single package evaluation, discussed in Section 6.4.2.C of the SAR, uses the normal conditions of transport model discussed above which combined the most reactive conditions from the parametric studies. The results of the applicant's criticality analysis show that k_{eff} of a single TN-68 package will remain below the design criterion of 0.95 for all allowed fuel loadings when optimally moderated and fully reflected by water.

The applicant evaluated the package with full water reflection outside the gamma shield (6 inches thick) instead of the containment shell (1.5 inches thick). The staff finds this acceptable because the gamma shield will not separate from the containment shell under normal conditions of transport or hypothetical accident conditions. Additionally, the difference in calculated k_{eff} is expected to be negligible because the evaluated reflector, 7.5 inches of carbon steel surrounded by 12 inches of water, is essentially an infinite neutron reflector.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.55(b), (d), and (e).

6.5 Evaluation of Package Arrays Under Normal Conditions of Transport

The hypothetical accident conditions do not adversely affect the package design with respect to criticality safety. All calculational models for the criticality analysis assume that the neutron shield is missing and that the package is optimally moderated. Therefore, the evaluation for the normal and hypothetical accident conditions are the same and is discussed in SER Section 6.6 below.

6.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

The evaluation of package arrays under hypothetical accident conditions, presented in Section 6.4.2.D of the SAR, uses the combined worst case contents configuration and optimal internal moderation from the single package evaluation. The criticality analysis shows that for the three most reactive assemblies, an infinite rectangular array of packages will remain subcritical with close full-water reflection and optimum

interspersed hydrogenous moderation. Optimum moderation occurs with the packages fully flooded with 100% density water. The interspersed moderation has no significant impact on the reactivity because the thick walled gamma shield precludes neutron coupling between packages.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.59(a)(1), 71.59(a)(2), and 71.59(b).

6.7 Benchmark Evaluations

6.7.1 Experiments and Applicability

The applicant performed benchmark comparisons on selected critical experiments that were chosen to bound the variables in the TN-68 cask design. The benchmark parameters bounded the parameters in the analysis with respect to fuel enrichment, fuel pin pitch, boron areal density in the separator plates, hydrogen to U-235 atom ratio, water to fuel volume ratio, assembly separation, and average lethargy causing fission.

The staff reviewed the benchmark comparisons in the SAR and agrees that the CSAS module of the SCALE computer codes used for the analysis was adequately benchmarked to representative critical experiments relevant to the cask design and contents specified.

6.7.2 Bias Determination

Eight subsets of the benchmark results were analyzed for a trend in bias and no significant trends were found. An USL of 0.9331 was calculated by the applicant. The USL incorporates the biases and uncertainties of the model and computer code into a value that has a 95% confidence level such that any k_{eff} less than the USL is less than 0.95, which is the design criterion.

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

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

6.8 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the design has been adequately described and evaluated and that the package meets the criticality performance requirements of 10 CFR Part 71.

7.0 OPERATING PROCEDURES

Operating procedures for the package are specified in Chapter 7 of the application. The chapter includes sections on package loading, unloading, preparation of an empty package for transport, and preparation of a package used in storage.

The chapter discusses the procedures needed to dry the cask cavity using a vacuum drying system. The torque levels and sequence are provided in the section on preparation for transport.

The inner lid, inner vent, and cover seals are to be leak tested to a leak rate of not greater than 1×10^{-5} ref cm^3/sec in accordance with ANSI N14.5.

The procedures include the installation of the upper and lower impact limiters including impact limiter attachment tie-rods.

Surface doses and temperatures are checked to ensure that the regulatory limits are not exceeded.

Based on the review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure that the package will be operated in a manner consistent with its evaluation for approval.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Acceptance tests and the maintenance program for the package are specified in Chapter 8 of the SAR. The acceptance tests from section 8.1 of the application include visual inspections, structural and pressure tests, components tests, shielding tests, and neutron absorber tests,

The maintenance program is specified in Chapter 8.2 of the SAR and includes structural and pressure tests, leak tests and impact limiter tests.

The metallic containment seals are to be replaced and leak tested after lid or port cover removal to demonstrate that the leak rate is less than 1×10^{-5} ref cm^3/sec .

To ensure that the impact limiters are not damaged or degraded between or during shipments, the applicant has specified an inspection program. A periodic leak test of the stainless steel protective enclosure will be performed to verify the leak tight integrity of this enclosure for excluding moisture from the wood assemblies. Additionally, a visual inspection will be performed prior to each shipment to ensure there are no cracks or other damage that may compromise the leak tightness of the protective enclosure.

The bolts used for the cask structural lid will be reused after each shipment. The applicant has specified a limit to the number of times these bolts may be reused. Periodic inspections of these bolts will be performed to ensure there is no excessive wear, fatigue cracks, or other adverse conditions.

Based on the review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure that the package will be acceptance tested and maintained in a manner consistent with its evaluation for approval.

9.0 CONDITIONS

In addition to the requirements of Subpart G of 10 CFR Part 71:

- (a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, "Operating Procedures," of the application, as supplemented.
- (b) Each package must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program," of the application, as supplemented.
- (c) Known or suspected fuel assemblies with cladding defects greater than pinhole leaks and or hairline cracks are not authorized.

10.0 CONCLUSION

Based on the review of the statements and representations in the application, as supplemented, and the conditions listed above, the staff concludes that the design has been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

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