71-9253

**Department of Energy**

Oak Ridge Operations Office
P.O. Box 2001
Oak Ridge, Tennessee 37831—

June 15, 2001

Mr. E. William Brach, Project Director
U.S. Nuclear Regulatory Commission
Spent Fuel Project Office
One White Flint North
11555 Rockville Pike
Washington, DC 20555

Dear Mr. Brach:

In accordance with the letter received by the Department of Energy (DOE) Oak Ridge Operations Office dated April 4, 2001, the enclosed additional information is provided to support our license application. As discussed by staff, we are providing revised pages for the Safety Analysis Report and a revised Addendum for Configuration 2.

On May 23, 2001, DOE notified the Nuclear Regulatory Commission that we would require additional time to formulate an adequate response. The DOE is appreciative of your efforts to accommodate our request and to maintain our current schedule.

If there are any questions, please feel free to contact Brian DeMonia of my staff at (865) 241-6182.

Sincerely,

Robert C. Sleeman, Group Leader
Environmental Services Group

Enclosure

cc w/o enclosure:
A. Griffith, EM-21, CLVRLF
M. Wangler, EM-5, CLVRLF
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NMSS01 Public

**Responses to NRC Request for Additional Information**
**April 4, 2001**

## CONFIGURATION 1 - "-85" DESIGNATION

1.0 GENERAL INFORMATION

1-1 The packaging drawings provided in Appendix 1.3 are not legible. Provide drawings in which all pertinent information is legible.

**Drawings have been revised to make them legible.**

1-2 Revise the table of contents for the Safety Analysis Report to include the Addendum for the Oak Ridge Container.

**The table of contents has been revised accordingly.**

1-3 The "-85 Compliance Matrix for TN-FSV Package," which was intended to provide assistance and clarity for the review of the package for the "-85" designation under the provisions of 10 CFR 71.13, contains errors and is incomplete. Either revise the matrix to provide useful, complete and accurate information or delete it.

**The Compliance Matrix has been deleted. The package has been reviewed against NUREG-1617.**

1-5 The application should clearly identify any exceptions being taken to the guidance provided in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." If there are none, the application should clearly state that.

**The package was reviewed against NUREG-1617 requirements and the results are shown in the TN-FSV SAR, section 1.1.**

1-6 Revise the application to include a description of the personnel barrier and to clarify the purpose of the personnel barrier.

Operating procedures for the package include steps to install the personnel barrier. However, it is not clear that the personnel barrier is described in the application or in the packaging drawings or what the purpose of the personnel barrier is. Also, show that the personnel barrier does not affect the ability of the package to meet the performance requirements of 10 CFR Part 71. This information addresses the requirements of 10 CFR 71.33.

**The use of the "personnel barrier" is optional. The personnel barrier performs no safety function. If a "personnel barrier" is used for the shipments, it will be an open, loose mesh material such that free air flow is allowed and the thermal analysis is not affected. Chapters 1 and 7 are revised accordingly.**

1.7 Editorial Comment:

Revise Pages 1-11 and 1-12a to clearly identify the new information provided in this application. Pages 1-11 and 1-12a were submitted with the application, however, it is not clear that these pages provide any new information.

**The page 1-11 was mistakenly submitted with the November 2000 submittal, as it contained no new information at that time. It is included with the current submittal, as it contains the corrected top spacer diameter in accordance with RAI comment 2-13(a) (Configuration 2).The revised page 1-12 is now submitted.**

2.0 STRUCTURAL

2-1 Revise Section 2.7.5 of the application to evaluate the appropriate external pressure for the deep water immersion test specified in 10 CFR 71.61

Section 2.7.5 evaluates an external pressure of 284 psi, however, 10CFR 71.61 specifies that the external pressure is 2 MPa (290 psi). Also note that Section 2.7.5 is incorrectly labeled as "Immersion - All Packages," whereas 71.61 applies only to irradiated nuclear fuel shipments. Section 2.7.5 should be revised to also address the requirements of 10 CFR 71.73(c)(6) which is "Immersion – All Packages."

**Revised SAR pages for Section 2.7.5 and Tables 2.7-26 and 2.7-26A are submitted addressing the 290 psi immersion pressure.**

2.2 Verify the validity of the conclusion (Page 2-70) that lead slump will not occur under the 30-foot free drop, and provide the rationale that supports the conclusion.

Lead shielding material is elastic-plastic and will experience deformation and slump under the 30-foot free drop because stresses are expected to be beyond the yield stress of the lead material. Therefore, the statement in the safety analysis report appears incorrect, and should be verified. The degree of lead slump should be considered in the shielding evaluation. This addresses the requirements of 10 CFR 71.33 and 71.47.

**Appendix 2.11.8, TN-FSV Cask Lead Slump Analysis, is added to the revised Addendum. This appendix consists of the following.**

**A non-linear finite element analysis is performed in order to quantify the amount of lead slump that occurs in the TN-FSV Cask lead shielding during a hypothetical accident condition end drop.**

**The two load cases considered are the hypothetical accident condition 30 foot lid and bottom end drops. The maximum axial acceleration of the transport package during a 30 foot end drop impact is 54*g* [TN-FSV SAR, Section 2.10.2].**

3.0 THERMAL

3-1 Revise the hypothetical accident conditions thermal analyses for the package to include the effects of convection that would exist in a fire.

The regulations (10 CFR 71.73) require that the effects of the convection from a fire be considered in a 30-minute thermal test. Note that the convective conditions to be assumed in the analysis of a fire were revised in the 10 CFR Part 71 that became effective April 1, 1996.

**Incorrect Chapter 3 pages were mistakenly submitted previously. The revised pages 3-19, 3-19a, 3-20, and 3-20a show that the convection coefficients used in the previous thermal analysis bound the convection effects of the fire required by 10CFR71.73; therefore, the current SAR thermal analysis for the fire satisfies the requirements of 10CFR71.73.**

4.0 CONTAINMENT

4-1 Revise the containment analysis to calculate the value "$L_R$" as defined in ANSI N14.5-97, "Leakage Tests for Packages for Shipment." Show that (1) the normal conditions leakage rate is still below $1x10^{-3}$ ref-cm$^3$/sec, under reference conditions, or (2) the package will be leak tested to the revised value of $L_R$ under reference conditions.

The value of $L_N$ is given on page 4-9 of the application, however, the leakage hole diameter should be calculated for normal conditions of transport, considering maximum normal operating pressure and temperature. Revise the application to include the complete calculation of the value of $L_R$. Clearly identify the numerical values used for each of the variables in the calculation. Note also that the maximum allowable leakage rate should be consistent with the leak tests specified in Chapters 7.0 and 8.0 of the application. This information addresses the requirements of 10 CFR 71.51.

**The methodology in ANSI N14.5-97 has been used to recalculate the hole diameter and $L_{R,N}$ and the SAR revised accordingly.**

4-2 Revise Page 4-12 of the safety analysis report to correct the calculation of $L_A$.

Page 4-12 uses an incorrect value for the allowable release of krypton-85 under hypothetical accident conditions. Note that the allowable release for krypton-85 was revised in the 10 CFR Part 71 that became effective April 1, 1996. This revised quantity should be considered for packages that have the "-85" designation. This information addresses the requirements of 10 CFR 71.51.

**The revised page 4-12 was mistakenly omitted from the previous submittal. The revised page is provided and shows the correct $A_2$ for Kr-85 is used.**

4.3 Editorial Comment:

Revise Page 4-11 to clearly identify the new information provided in this application. Page 4-11 was submitted with the application, however, it is not clear that this page provides any new information.

**Page 4-11 was not changed and was mistakenly submitted.**

5.0 SHIELDING

5-1 Revise the shielding analysis to address the possibility of lead slump under 30-foot drop conditions.

See Item No. 2-2, above. This addresses the requirements of 10 CFR 71.47.

**The accident dose rate for the cask with the FSC canister is calculated using the MCNP computer code. The details of the MCNP model/calculation for the Oak Ridge Container are in Appendix 5.6.1 of the Addendum. The Container was replaced by the FSV Canister and dose rates calculated for the accident model of the FSV cask. As reported for the Container model, the dose rates at the end of the cask increased due to the lead slump, but the maximum accident dose rate for Configuration 1 is no larger than for normal conditions. Chapter 5 of the SAR has been revised accordingly.**

7.0 OPERATING PROCEDURES

7-1 Revise the operating procedures to clarify which steps are optional.

Section 7.0 implies that operational steps that are not underlined are optional. However, there appear to be some steps that are not underlined that should not be optional (e.g., placing the cask lid on the cask body). It is suggested that optional steps be labeled as such. Note that procedures that assure compliance with the requirements of 10 CFR 71.87 should not be optional.

**Chapter 7 has been revised to remove the underlined items and eliminate any impression that non-underlined items are optional.**

7-2 Revise the operating procedures to specify that prior to loading, the shipper determines that the contents to be loaded are authorized in the Certificate of Compliance.

The provisions of 10 CFR 71.87 specify that the shipper must determine that the package is proper for the contents to be shipped.

**The SAR has been revised as requested.**

## CONFIGURATION 2 – OAK RIDGE CONTAINER

### 1.0 GENERAL INFORMATION

1-1 Revise the application to include the following information:

a. For the Peach Bottom fuel elements: Maximum U-235 content, maximum uranium enrichment, maximum Th-232 loading, maximum burnup, minimum cool time, and maximum decay heat.

b. For the fuel components within an Oak Ridge Canister: Maximum fissile material content, maximum uranium enrichment, maximum burnup, and minimum cool time.

Estimates for some of these parameters are listed in Table 1-3, however, it is not clear that these values are considered bounding. Note that the Certificate of Compliance may be conditioned to include limits for these parameters. This information addresses the requirements of 10 CFR 71.33.

**Each intact Peach Bottom fuel element initially contained up to 0.25 kg of uranium enriched to a maximum of 93.15 weight percent U-235 and up to approximately 1.5 kg of thorium prior to irradiation. The maximum burnup is approximately 72,700 MWd/MTHM, and the minimum cool time is 27 years. The maximum decay heat for any individual element is < 3 watts. The estimated decay heats for the specific fuel elements to be shipped in the Oak Ridge Container are shown in Table 1-3.**

**Each Oak Ridge Canister contains pieces of irradiated nuclear material from various types of reactors including light water reactors, fast reactors, high temperature gas cooled reactors, and the Keuring van Electrotechnische Materialen reactor. The maximum fissile material content for each canister are within the five fissile content groups (shown below) that are based on the maximum pre-irradiation mass of U-235 and fissile plutonium (Pu-239 plus Pu-241). Uranium enrichments for the materials vary greatly depending on the material types. The maximum preirradiation uranium enrichment for any individual fuel component is 98 weight percent U-235, however, only a small amount of material exceeds 94 weight percent enrichment in U-235 and no individual Oak Ridge Canisters are comprised solely of materials with greater than 94 weight percent enrichment in U-235. The maximum burnup for the materials vary greatly depending on the material types. The highest burnup materials are from fast reactor facilities which have a maximum burnup of 15 atom percent, and a minimum cool time of 15 years. The maximum burnup for individual Oak Ridge Canisters and the estimated decay heats are shown in Table 1-3.**

**Fissile Content for Each Group**

| Group | Grams $^{235}U$ (Max) | Grams $^{239}Pu + ^{241}Pu$ (Max) | No. of Canisters | FEM |
|---|---|---|---|---|
| 1 | 475 | 0 | 48 | 475 |
| 2 | 865 | 191 | 8 | 1171 |
| 3 | 200 | 415 | 2 | 864 |
| 4 | 275 | 160 | 11 | 531 |
| 5 | 910 | 0 | 4 | 910 |
| IPB | 250 | 0 | 9 | 250 |

**This information has been provided in Chapter 1 of the Addendum.**

1-2 Revise the engineering drawings for the Oak Ridge Container to ensure they are legible and provide information consistent with the requirements of 10 CFR 71.33.

Parts of the drawings are not legible (e.g., materials specifications). The drawings should also include codes and standards used for design and fabrication of the packaging, and tolerances for all components. Note that NUREG/CR-5502 provides guidance regarding information that should be included in engineering drawings for packagings. This information is required to meet 10 CFR 71.33.

**The design and fabrication standards have been added to the drawings and the material specifications have been enlarged to be legible.**

1-3 Delete the words "or equivalent" in the materials listings with respect to the butyl O-rings.

The application should unambiguously specify the material for the containment system O-rings. Alternatively, identify and justify the use of alternative O-ring materials. This information addresses the requirements of 10 CFR 71.33 and 71.51.

**The words "or equivalent" have been deleted from the drawing.**

1-4 The application should clearly identify any exceptions being taken to the guidance provided in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material." If there are none, the application should clearly state that.

**The addendum has been reviewed against NUREG-1609. The results are shown in the Addendum, section 1.1.**

2.0 STRUCTURAL

2-1 Revise the analysis for the standoffs to consider the weight of the canisters and the fuel assemblies.

Section 2.11.1.2A evaluates the structural adequacy of the standoffs for normal and accident condition events described 10 CFR Part 71. For computing the stresses in standoffs, self-weight, and the weights of poison plates, poison enclosures, and support discs, are used in calculating inertia loads during an end drop accident. However, the weight of the canisters and the spent fuel is not considered. Even though this assumption may be applicable for a perfectly vertical end drop, it will not be correct for a potential drop at any other angle. The inertial loads from the canisters should be considered in the analysis of the standoffs due to an end drop accident. Alternatively, justify why the weight of the canisters and fuel assemblies is not considered in this evaluation. This information addresses the requirements of 10 CFR 71.33.

**During the vertical lid end drop (see Figure 2.11.1-1), the support disc and poison enclosure (including poison plates within the enclosure) nearest to the impact bear directly on the inner surface of the container lid for the lid end drop (or inner surface of the container bottom plate for the bottom end drop – see Figure 2.11.1-2). The standoffs do not carry this inertial load. Therefore, for the end drop analysis, the inertial loads of the support disc and poison enclosure with poison plates nearest to the impact, are not included in the axial compressive load applied to the standoffs.**

**Since the fuel compartments, canisters, and flux traps also bear directly on the inner surface of the container lid during a lid end drop (or inner surface of the compartment spacer, which bears directly on the container bottom plate for the bottom end drop), their loads are also not included in the compressive load applied to the standoffs.**

**In order to bound the potential effects of the inertial load of the fuel compartments, canisters, and flux traps during a near vertical drop, an additional corner drop analysis is added to Appendix 2.11.1 of the application.**

**In addition to the 90° end drop load case, a corner drop load case generating compressive loads in the standoffs is also considered. During a near vertical corner drop, the lateral inertial loads of the fuel compartments, canisters,**

**and flux traps acting on the support discs will generate an axial friction force on the support discs. This additional axial friction force is ultimately transmitted to the standoff nearest to the impact end.**

**The section regarding the compressive stress in the standoffs due to the end drop event (Section 2.11.1.2, of Appendix 2.11.1) has been revised. Note; the outer diameter of the standoffs is changed from 0.75 in. to 0.875 in. Consequently, the weight of the 45 standoffs is increased from 68 lb. to 112 lb.**

**Appendix 2.11.1 has been revised accordingly with the addition of free body diagrams.**

2-2 Revise the analysis of the standoffs to include consideration of a 30-foot side drop.

Section 2.11.1.2A evaluates the structural adequacy of the standoffs. This section addresses the load on the standoffs during an end drop accident, but does not address the behavior during a side drop accident. Alternatively, justify why the side drop accident was not considered. This information addresses the requirements of 10 CFR 71.33.

**During a side drop event, the standoffs and tie rods are subjected to their own inertial load plus the inertial load of the poison enclosures and poison plates (see Figure 2.11.1-3). The tie rods are supported by the transverse reaction force of the spacer discs. This reaction force generates shear stresses in the tie rods at the spacer disc locations, and bending stresses in both the tie rods and standoffs between the spacer discs. Consequently, shear stress generated in the standoffs, during a side drop event, is negligible. Bending stress in both the standoffs and the tie rods is computed in the tie rod analysis section below (see Response to question 2-3).**

2-3 Revise the analysis of the side drop accident to consider bending stresses in the tie rods.

Section 2.11.1.2F evaluates the structural adequacy of the tie rods. This section addresses the loads on the tie rods during a side drop accident, and considers the shear stresses at the support discs. However, the bending stresses in the tie rods are not addressed. Alternatively, justify why the bending stresses in the tie rods are not considered for the side drop accident. This information addresses the requirements of 10 CFR 71.33.

**Figure 2.11.1-3 is a free body diagram of the fuel basket during a side drop event. The figure shows that the only loads applied to the tie rods are the inertial load of the poison enclosures, $F_3$ (acting near the support disc locations), and the inertial loads of the standoffs and tie rods themselves.**

**The tie rod analysis, Section 2.11.1.2.F of the Addendum has been revised to show the additional analysis.**

2-4 Revise Section 2.11.1.2 of the application to justify using the number of poison enclosures and support discs as 8 instead of 9, and the weight of the poison enclosures as a standard 272 lbs. instead of 285 lbs. for the structural evaluation of the standoffs under an end drop accident.

The calculation for the compressive loads on the standoffs during an end drop accident in Section 2.11.1.2 considers the weights of various components, but does not explain the rationale for the numbers. This information addresses the requirements of 10 CFR 71.33.

**During an end drop, the inertial load of the Oak Ridge Container fuel basket is directly supported by the container lid during a lid end drop and by the container bottom during a bottom end drop.**

**Since the support disc nearest to the impact end is in direct contact with the lid/bottom inner surface, the support disc's inertial load is taken by the TN-FSV Cask and not by the standoff nearest to the impact. Also, since the poison enclosure and poison plates nearest to the impact are in direct contact with the support disc nearest to the impact, their inertial loads are also not taken by the standoff. Instead, the inertial load of the poison enclosure and poison plates nearest to the impact are transferred through the support disc nearest to the impact to the container lid or bottom plate and finally to the TN-FSV Cask lid or bottom plate.**

**Since there are a total of 10 support discs and the support disc nearest to the impact is directly supported by the container lid or bottom plate, the inertial load of only 9 support discs is applied to the outer most standoff nearest to the impact. Likewise, since there are a total of 9 poison enclosures with poison plates, and the enclosure nearest to the impact is directly supported by the support plate, the inertial load of only 8 poison enclosures with poison plates is applied to the standoff nearest to the impact.**

**Free body diagrams of the lid and bottom end drops, shown in Figures 2.11.1-1 and 2.11.1-2, clearly show that the support disc nearest to the impact is directly supported by the container lid/bottom plate, which is in turn supported by the TN-FSV Cask lid/ bottom plate.**

**In Section 2.11.1.2 of the Addendum, the inertial loads of 8 poison enclosures with poison plates and 9 support discs are considered for the computation of the applied load to the standoff during a lid end drop. The lid end drop is considered critical, because the bottom disc (1.75 in. thick) is thicker than the**

**top support disc (0.75 in, thick). The formula used to compute the applied load, *P*, has been revised to the following for clarity.**

***P*** **= 16 *gs* × [weight of 45 standoffs, 112 lb. + weight of 80 poison plates, 172 lb. + the weight of 8 poison enclosures, 285 × (8/9) lb. + the weight of 8 support discs plus the 1.75 in. thick disc at the bottom, 217 × (7.75/8.5)lb.] / 5 standoffs = 2,352 lb.**

**The Addendum has been revised accordingly.**

2-5 Justify the basis for installing the nuts in the tie rods as snug-tight only (Ref. Drawing 3044-70-1, Note 3), and verify that the thread engagement of the tie rods at the top support disc is adequate.

It appears that tie rods are assumed to carry no tensile loads, therefore, the nuts are snug-tight only. However, in case the standoffs buckle during an end drop accident, the tie rods may be subjected to tensile loads. Since the nuts are not tightened, the capacity of the tie rods may be limited. To provide a margin of safety in the design, these bolts should be tightened to ensure that the tie rods can carry the loads to its ultimate tensile capacity. Also, the thread engagement should be sufficient at the top plate and in the nuts to ensure that the ultimate tensile capacity of the tie rods can be developed. This information addresses the requirements of 10 CFR 71.33.

**A tie rod nut torque range of 6 to 8 ft. lb. has been selected. Section 2.11.1.2 of the Addendum has been revised to show the evaluation for this torque.**

2-6 Revise the analysis of stresses in the Oak Ridge Container shell and top and bottom plates to include the weight of the canisters and spent fuel due to an end drop accident, as shown in Section 2.11.2.2 (pages 2.11.2-3 and 2.11.2-8) of the application.

During an end drop accident, the canisters and the spent fuel content inertia will load the top or bottom plate of the Oak Ridge Container, and thus will load the container shell. Alternatively justify why this weight should not be considered. This information addresses the requirements of 10 CFR 71.33.

**The inertial load of Oak Ridge Container fuel basket, including the fuel canisters and fuel compartments, is directly supported by the container lid during a lid end drop or by the Oak Ridge Container bottom during a bottom end drop.**

**The flat outer surface of the Oak Ridge Container lid is directly supported by the flat inner surface of the TN-FSV Cask lid. Likewise, the flat outer surface of the Oak Ridge Container bottom is directly supported by the flat inner surface of the TN-FSV Cask bottom plate. Therefore, all inertial loads**

**applied to the Oak Ridge Container lid and bottom are directly transferred to the TN-FSV Cask lid and bottom plate respectively.**

**A free body diagram of the bottom end drop, Figure 2.11.1-2, has been added to the Addendum. The reaction force, $F_R$, shown in Figure 2.11.1-2 represents the uniform support provided by the TN-FSV Cask bottom plate. The inertial load, $F_1$, which represents the inertial loads of the Oak Ridge Container shell, flange, and lid, is used to calculate the maximum compressive stress in the Oak Ridge Container shell during the bottom end drop.**

2-7 Explain the basis for not considering inertia loads from the canisters and the fuel basket structure in evaluating the lid bolts (ref. Section 2.11.4.2).

During an end drop accident, the canisters and the spent fuel content inertia loads will be transferred to the top or bottom plate of the Oak Ridge Container. This will result in the lid bolts being loaded in tension. This information addresses the requirements of 10 CFR 71.33.

**The inertial load of Oak Ridge Container fuel basket, including the SNF, is directly supported by the container lid during a lid end drop. The flat outer surface of the container lid is directly supported by the flat inner surface of the TN-FSV lid. Therefore all inertial loads applied to the container lid are directly transferred to the TN-FSV Cask lid.**

**A free body diagram, Figure 2.11.4-3 of the container lid closure system during a lid end drop has been added to the Addendum. The inertial loads of the container flange, body, and bottom, $F_3$, the internals, $F_2$, and the container lid, $F_1$, are reacted by the support provided by the TN-FSV Cask lid inner surface, $F_R$. Figure 2.11.4-3 shows that the inertial load of the container shell, flange, and bottom, and the inertial load of the container internals is transmitted directly to the container lid, and reacted by the TN-FSV Cask. Consequently, the container lid bolts do not experience any tensile loads during a lid end drop.**

2-8 Explain the basis of the vibration/shock loads in Section 2.6.5 of the application and how ANSI N14.23 is applied considering the frequency of the package.

Sections 2.11.4.6C (page 2.11.4-14), 2.11.5.2B (page 2.11.5-2) and 2.11.5.3C of the application use ANSI N14.23 for vibration/shock loads during transportation. ANSI N14.23, Section 4.2, specifies that the vibration/shock loads be based on the natural frequency of the package and its tie-down system. This information is required to verify that the Oak Ridge Container meets the requirements of 10 CFR 71.71.

**The longitudinal shock accelerations used in sections 2.11.4.6C, 2.11.5.2B, and 2.11.5.3C of the application are taken from ANSI N14.23, Table 1. The accelerations specified in ANSI N14.23, Table 1, are considered a reasonable guide for estimating maximum inertial loading on the cask during a truck shock event based on the most recent version of ANSI N14.32, Section 4.2.**

**The latest version of ANSI N14.23, Section 4.2 reads as follows:**

***4.2 Package Response***

***Acceleration of the package or parts thereof in response to shock depends on the natural frequency of the package and its tie-down system, and on the time duration and shape of the shock pulse. Classical analysis of single shock pulses shows response of a "package" ranging up to twice the amplitude of the driving pulse. Analysis of data from actual shock events in terms of shock response spectra has shown that the "package" can exhibit a peak response several times greater than the peak driving (truck bed) g level. These analyses, however, neglect the interaction between a relatively massive package, such as a nuclear cask, and the truck bed in modifying the truck bed acceleration levels.***

***Recent road simulator tests conducted by Sandia National Laboratories have shown that the truck bed accelerations provide an upper bound on cask accelerations. Therefore it is recommended that the peak acceleration values given in Table 1 be used as a guide in estimating the maximum inertial loading on the cask tie-down system. Since the maximum shock values can occur almost simultaneously, the design of the cask and its tie-down should be based on a vector summation of the vertical, longitudinal and lateral g-loads of Table 1. Note that vertical up in Table 1 forces the package down into the truck bed, while vertical down tend to separate the package from the truck bed.***

**The lid bolt stresses due to vibration/shock are calculated in Section 2.11.4.6C for Oak Ridge Canister lid bolts and Section 2.11.5.3C for TN-FSV Cask lid bolts. Only the longitudinal *g* load generates stress in the lid bolts. Therefore, only maximum longitudinal *g* loads from Table 1 (1.8g – shock, air suspended) and Table 2 (0.3g - vibration) are selected for the bolt stress evaluations. However, the TN-FSV Cask containment stresses, used for the fatigue evaluation (Section 2.11.5.2B), are calculated based on the vector summation of the vertical, longitudinal, and lateral g loads from ANSI N14.23, Table 1.**

2-9 Describe the function of the spacer sleeve (Item No. 30 in the Parts List on Drawing No. 3044-70-1) and the method of attachment to the Oak Ridge Container, if any.

Drawing No. 3044-70-1, Rev.0, shows the spacer sleeve on the outside of the container. The function of the sleeve and the method of attachment to the container shell are not addressed in the application or the packaging drawings. This information addresses the requirements of 10 CFR 71.33.

**The function of the aluminum spacer sleeve is to evenly distribute reaction loads from the TN-FSV Cask interior wall to the Oak Ridge Container shell during transverse loading conditions. Without the spacer sleeve, the nominal radial gap between the container shell and the cask wall in the flange region is 0.32 inches, while in the container shell region the gap is 0.575 inches. The radial thickness of the spacer sleeve is 0.31 inches. Therefore, with the spacer sleeve in place, the total radial gap in the container shell region is, 0.575 – 0.31 = 0.26 inches. Consequently, during a transverse loading event, such as the side drop event, the Oak Ridge Container will be uniformly supported by the interior surface of the spacer sleeve which is in turn supported by the interior surface of the TN-FSV Cask.**

**Figure 1 depicts the location and geometry of the aluminum spacer sleeve. The 0.32 in. gap between the container lid and the TN-FSV Cask wall inner surface, as well as the 0.26 in. gap between the aluminum spacer sleeve and the cask wall inner surface are clearly shown.**

**The spacer sleeve is not rigidly attached to the Oak Ridge Container of TN-FSV Cask. The spacer sleeve is simply placed inside the TN-FSV Cask prior to installment of the Oak Ridge Container.**

2-10 Revise the structural analysis to address the adequacy of the top and bottom plates of the flux trap for normal and accident condition events (10 CFR 71.71 and 10 CFR 71.73).

Section 2.11.1D addresses the shell of the flux trap only. The information addresses the requirements of 10 CFR 71.71 and 10 CFR 71.73.

**The flux traps are loaded in the Oak Ridge Container fuel compartments between Oak Ridge Canisters. The geometry of the flux trap lifting disc (top plate) and the Oak Ridge Canister handling head are very similar. The Oak Ridge Canister Head only provides support along the outer edge of the spacer cap (bottom plate) of the flux trap (see Figure 2.11.1-4). Therefore, during an end drop, the only load applied to the flux trap bottom and top plates is that generated by their own inertial load plus the inertial loads of the flux trap poison plates (item 24E, TN drawing no. 3044-70-3) and poison plate caps (item 24G). Since the flux trap bottom plate is thinner (0.45 in.) than the flux trap top plate (0.667 in.), the bending stresses generated in the bottom plate are greater than the bending stresses generated in the top plate.**

**Therefore, only the flux trap bottom plate is analyzed, since it is the bounding case.**

**The analysis is added to the flux trap spacer section of Appendix 2.11.1 in the Addendum.**

2-11 Justify the use of partial penetration groove welds for connecting the flux trap shell to the top and bottom plates.

Drawing 3044-70-3, section A-A shows the welded connections between the flux trap shell and the top and bottom plates. The welds are partial penetration groove welds, which may not develop the full strength of the shell. Therefore, there is a need to demonstrate structural adequacy of the connections. The information addresses the requirements of 10 CFR 71.71 and 10 CFR 71.73.

**The flux trap weld stress analysis is added to the flux trap analysis Section 2.11.1.2 D of the Addendum. Note that the welds between the top plate and the shell and the bottom plate and the shell are changed from discontinuous to continuous welds.**

2-12 Revise Section 2.2 to include the following: (1) weight of the empty Oak Ridge Container, (2) weight of the empty Oak Ridge Canister, (3) the maximum weight of contents per Peach Bottom fuel element canister (including the fuel element and structural material), (4) the maximum weight of contents per Oak Ridge Canister (including any secondary containers), (5) the maximum weight of contents, including canisters, basket, and other secondary containers, per Oak Ridge Container.

This information addresses the requirements of 10 CFR 71.33.

**Table 2-5A has been added to the Addendum, which contains the information requested. Also Table 2-5 has been revised to reflect the increased standoff weight and the weight of the empty Oak Ridge Container.**

**Note that the Oak Ridge Container consists of the container shell, lid, bottom plate, and fuel basket (see complete component list in Table 2-5, total weight is 3,161 lb.). Therefore, the maximum weight of the contents (payload) per Oak Ridge Container is simply the maximum weight of the Oak Ridge or Peach Bottom Canisters plus flux traps per shipment (1,789 lb.).**

2-13 Editorial Comments:

a. Page 1-11 of the SAR: Top spacer diameter is listed as 18.38 inches, which is not consistent with the Drawing 1090-SAR-5, item 14, dimension of 17.63 inches.

**Page 1-11 of the TN-FSV SAR has been revised to correct the dimension to 17.63 inches.**

b. Drawing 3044-70-1: Material for standoffs is shown as SA-564, Type 630, which is inconsistent with the material shown as SA-693 in the calculation in Section 2.11.1-2 (page 2.11.1-2).

**Section 2.11.1.2 has been corrected to show SA-564, type 630.**

3.0 THERMAL

3-1 Revise the figures in the thermal analysis, for the Oak Ridge and Peach Bottom spent fuel, to indicate which materials have been modeled for each region in the computer analysis and show the boundary conditions that have been applied.

This addresses the requirements of 10 CFR 71.71 and 71.73.

**Figure 3-2 in the Addendum has been revised to include the requested information.**

3-2 Justify why the thermal analysis for the new contents assumed a boundary condition temperature of 167°F (the calculated temperature for the inner shell wall shown in the original safety analysis report for the package) instead of the 183°F calculated temperature of the fuel storage container.

It is not clear that the thermal analyses included the air gap between the inner shell and the fuel storage container. The analysis should assume the appropriate boundary temperature, since the calculated seal temperature under hypothetical accident conditions is close to the maximum allowable service temperature. This addresses the requirements of 10 CFR 71.71.

**The finite element model developed for the thermal analysis includes all the components in the TN-FSV cask body including the cavity between the inner shell and the Oak Ridge Container. Therefore, the maximum temperature of the inner shell is applied on the outermost nodes of the model, which are representing the inner shell of the cask body. The maximum temperature of the TN-FSV cask inner shell is 167°F as indicated in the original safety analysis report. The maximum temperature of the Oak Ridge Container is 183°F, which is a part of the new developed finite element model. Chapter 3 of the Addendum has been revised to better describe this.**

3-3 Justify why the 3-dimensional thermal analyses used to determine the Oak Ridge Container seal region temperatures did not include the thermal effects of the other fuel containers in the package.

It is not clear that the model shown in Figure 3-4 accurately takes into account the temperature effects of the other fuel containers which have temperatures higher than the assumed 167°F (see Figures 3-2 and 3-2). The analyses should be modified to include these effects. This addresses the requirements of 10 CFR 71.71.

**Due to a large thermal resistance in the axial direction the majority of heat transfer will take place radially. This effect is complemented by the horizontal orientation of the packaging. Applying a heat load of 35 Watts as a heat flux directly into the lid bounds the temperature effects of both the cross-section directly adjacent to the lid and the other fuel containers.**

**The 167°F temperature is the maximum cavity wall temperature of the TN-FSV cask as calculated in the TN-FSV SAR. This temperature is calculated from a heat flux much larger than that corresponding to the hottest cross-section of the Oak Ridge Container and is, therefore, conservative. This 167°F temperature is applied to the nodes that comprise the cask cavity wall surface (See Response to 3-2). (Note: there is only one Container, it contains up to 20 fuel canisters.)**

**Chapter 3 of the Addendum has been revised accordingly.**

3-4 Revise the application to include figures that detail the calculated temperatures for the hypothetical accident conditions 30-minute thermal test.

The figures in the application only show the calculated temperatures during normal conditions of transport. This addresses the requirements of 10 CFR 71.73.

**During hypothetical accident conditions, the maximum cavity wall temperature increases to 245°F. This is 78°F hotter than the cavity wall temperature of 167°F that occurs during normal conditions of transport. Since the thermal mass and the thermal conductivity of the components do not change significantly during the hypothetical thermal accident, component temperatures within the Oak Ridge Container during accident conditions are determined by increasing their temperatures under normal transport conditions by 78°F.**

**Figures of the temperature distributions during accident conditions are therefore not available.**

5.0 SHIELDING

5-1 Provide an evaluation of the ability of the package, with the requested contents, to meet the external radiation standards in 10 CFR 71.47.

The shielding evaluation does not include an assessment of the proposed contents to meet the external radiation standards. Although the shielding evaluation provides a useful and informative technical method to assess whether hypothetical contents will result in acceptable dose rates it is not clear that an evaluation of the actual contents proposed for shipment has been performed. Provide a sample calculation for representative or reasonably bounding contents that would provide assurance that the external radiation standards in 10 CFR 71.47 will be met.

**Dose rates for a representative Oak Ridge Container loading have been calculated using the MCNP computer code. This analysis is described in Appendix 5.6.1 in the Addendum. The arrangement of the 20 Oak Ridge canisters in this representative Oak Ridge Container loading result in a total curie content in the twenty canisters that will be approximately 10 percent higher than the maximum anticipated inventory in any of the five planned shipments. While the canister loading plans for the five shipments have not yet been finalized, several possible Oak Ridge Container loading arrangements for the shipments have been identified that will meet all the constraints. As described in the Addendum, to meet the criticality limits, the canisters must be specifically placed in the fuel compartments according to the allowable loading patterns based on fissile group. Also, to meet the thermal limits, the Oak Ridge canisters with higher wattage are constrained to certain combinations of positions within the Oak Ridge Container. (See Section 7.1.2 of the Addendum for the loading constraints.)**

**From the representative Oak Ridge Container loading, three canisters were selected to provide a source spectrum for the LWR, fast reactor, and HTGR fuel types. Each of these three canisters were identified as one of the more heavily loaded canisters for that fuel type, but the canisters are not the absolute bounding canister for each type. Rather, to ensure a conservative representative Oak Ridge Container loading for all twenty canisters, the curie content in the shipment was set at about 110% of the curie loading of the highest expected shipment.**

**The three canisters were used to provide the source spectrum for each fuel type. That is, rather than use several different LWR canisters, the same canister spectrum for CAN-GSF-196 was used, and fractions of the source term (and of the heavy metal content for self-shielding purposes) was used. Since there is only one canister CAN-GSF-196, and few LWR canisters with a higher curie loading, it would have been unrealistic to have included ten full canisters of CAN-GSF-196 in the representative shipment. The HTGR canister (CAN-GSF-182) has close to the same curie content and fissile content as the intact Peach Bottom assemblies, just in a much more concentrated form since the intact Peach Bottom assemblies replace 3 Oak Ridge canisters physically.**

**Since the dose rate at 2 meters for the representative shipment is less than half the limit, all of the proposed certificate and transportation requirements could be met with a shipment that had an even higher curie loading than analyzed.**

5-2 Revise the application to clarify how the equations in Section 5.1 will be used.

It is not clear why the equations were developed, and how they will be used. If they will be used to screen canisters that will potentially be loaded into the cask, that should be clearly stated and should be included in the operating procedures for the package. This addresses the requirements of 10 CFR 71.47.

**The equations were developed as a means to screen the canisters during repackaging and to help plan for loading. Chapter 5 has been revised to state this and Chapter 7 has been revised to indicate that the screening has been used to plan the repackaging and loading.**

5-3 Revise the application to clarify how the equations in Section 5.1 were developed.

The origin of the values in the equations is not clear. For example, it appears that the values for the "top of a canister" equation come from Table 5-3, last column of the table entitled "Dose Rates on Canister Surface from SAS4 (Monte Carlo)," for 1 meter. The values for the "bottom of a canister" and "canister surface (radial direction)," apparently come from the same table for 2 meters and contact, respectively. The derivation and relationship of these values is not clear. This addresses the requirements of 10 CFR 71.47.

**Chapter 5 has been revised to correct errors and to better describe how the equations were developed.**

5-4 Revise the shielding analysis to address the possibility of lead slump under 30-foot drop conditions.

See Item No. 2-2, under Configuration 1, above.

**The revised Chapter 5 addresses the dose rates for the lead slump accident case.**

5-5 Justify why the radionuclides used as contents in the model in Section 5.2 (Co-60 for gamma source, and Cm-244 for neutron source) are bounding for the irradiated fuel contents.

The justification should show how this content is bounding for each category of proposed contents (i.e. LWR, fast reactor, HTGR, Peach Bottom HTGR, or KEMA spent fuel materials). It is unclear if all irradiated fuel types will be bounded by the selected radionuclides. For example, irradiated Th-232 can lead to the production of Tl-208, which emits a 2.6 MeV gamma. This is a significant source in the 2.5 to 3.0 MeV range, not commonly present in depleted uranium fuel.

In addition, the energy correction used for the gamma source is not correct, since dose rate is not directly proportional to energy. This addresses the requirements of 10 CFR 71.47 and 71.51.

**The use of Co-60 is bounding for gamma sources in all 5 categories of SNF contents. The gamma spectra for three different Oak Ridge canisters are shown in a table in the new Appendix 5.6. These canisters are representative of HTGR SNF (GSF-182), LWR SNF (GSF-196), and Fast Reactor SNF (GSF-213). The source term for the Peach Bottom SNF is comparable to the source term for the HTGR SNF in GSF-182. The KEMA SNF is a very low-burnup, long-cooled material and since it does not have a significant dose rate, is bounded by the other types of spent fuel materials.**

**As can be seen from the data in the table, the majority of the photons/sec contributing to the dose rate are in the 0.6 to 0.8 Mev range for these representative canisters. The choice of Co-60 as a bounding spectra is therefore appropriate since the gamma from Co-60, at an energy of 1.25 MeV, is well above the energy of most of the gammas emitted from the actual radionuclides in these canisters. The gamma spectra table shows that only a very small fraction (<0.001) of the total gammas/sec from each canister is in a group with an energy higher than the group with Co-60 (i.e., higher than Group 9, the group from 1.0 to 1.33 MeV).**

**An even smaller fraction is from Tl-208. The gammas from Tl-208 are accounted for in the actual spectra as part of Group 13 (2.50 to 3.00 MeV), but this group is about 24 times smaller than the Group 12 value in gammas/sec emitted per canister. So, although there are photons emitted from the radioactive materials that are above the energy of the gammas emitted by Co-60, the quantity is very small and is not significant enough to have an effect on the calculation for determining the screening value used for measurement of each loaded Oak Ridge canister's surface dose rate.**

**The procedure that adjusts the number of particles in a specific gamma line being placed into an energy bin by the ratio of the gamma line energy to the average energy of the energy bin is well established. Indeed it is the procedure that the ORIGEN-S code uses to bin the discrete gamma line values into various groups. While not strictly rigorous, it has been shown in practice to be the best approach short of refining the number of energy bins.**

**To address the question about Tl-208 for the representative shipment, the HTGR canister, GSF-182, was evaluated since it contained thorium in the initial loading. The SAS4 and SAS1 models were run with the actual spectrum for canister GSF-182 (which included Tl in the spectrum). The**

**resulting gamma dose rate at the surface of the canister in the radial direction increased by a factor of 20 from the value identified in Section 5.4 where the bounding Co-60 spectrum was used.**
**This demonstrates that using the cobalt-equivalent approach to establishing the screening equations is conservative even when Tl is present in the HTGR fuels. This degree of conservatism is expected to be similar for the other type fuels as well, and results from the use of a Co-60 spectrum rather than the actual spectrum.**

**The Cm-244 spectrum is discussed below in response to question 5-6.**

5-6 Provide a reference or justification for the Cm-244 neutron spectrum tabulated in Section 5.2.2. Additionally, explain why the "typical" Cm-244 spectrum is given for 15 years decay.

It is not clear what the basis of the source of the spectrum is or if it is appropriate for the irradiated fuels being considered. This addresses the requirements of 10 CFR 71.47 and 71.51.

**Twenty year decayed neutron source spectra for several representative canisters are shown in Appendix 5.6.1. As can be seen, the spectra presented are very similar to the 15 year decayed Cm-244 spectrum utilized in Section 5.2.2 of the Addendum. Records for the canisters indicate that all the fuel was irradiated more than 20 years ago; therefore, the choice of a fifteen year decayed Cm-244 spectrum is justified for the "screening" evaluation to guide the repackaging and planning for the loading.**

5-7 Justify why twenty Oak Ridge canisters provide the bounding source term for the shielding evaluation, as stated in Section 5.3.1.

The applicant should provide an assessment that shows this configuration has the most limiting source term. This addresses the requirements of 10 CFR 71.47.

**The only configuration with less than twenty canisters is one containing Peach Bottom assemblies. A Peach Bottom assembly essentially replaces three canisters in a fuel compartment. From Table 1-3, the maximum activity in a PB assembly is < 700 Ci. The representative Container loading evaluated in Appendix 5.6.1 contains approximately 37,000 Ci, for an average of approximately 1,800 Ci per canister. Therefore, for every PB assembly placed into the Container, the total activity in the Container would decrease, on the average, by about 4,700 Ci (5400-700)**

5-8 Section 5.3.1 of the application states "The acceptable contact gamma and neutron dose rates calculated for this arrangement will be bounding when applied to the other loading arrangement because the volumetric source for the Peach

Bottom assembly is lower than the Oak Ridge canister." Justify this conclusion, or provide information that would support this conclusion.

The various source volumes are not provided. Additionally, it is unclear why the largest source volume will provide the bounding source term. Rather, the bounding source term is expected to be that arrangement with, the largest burnup, largest initial mass of heavy metal, and the shortest cooling time. It is noted that some of this information is estimated for each canister and is presented in Table 1-3. This addresses the requirements of 10 CFR 71.47.

**Section 5.3.1 has been revised, see the response above.**

5-9 Justify why 200 g of U-238 is selected for the source configuration, and why this is considered "conservative."

Clarify which category of contents this 200 g of U-238 represents (i.e., LWR, fast reactor, HTGR, Peach Bottom HTGR, or KEMA spent fuel materials). It is unclear why this content results in the bounding source term. This addresses the requirements of 10 CFR 71.47.

**This quantity is used for self -shielding in the source volume. The tabular data for the canisters showed that the vast majority of the canisters have post irradiation total heavy metal content over 200g, and for those canisters that had less than 200g, the measured dose rate was very low.**

5-10 Clarify Table 5-3, and correct all errors in the arithmetic for the last column of the table entitled "(1-D SAS1) Oak Ridge Container in the TN-FSV Cask."

The applicant states in Section 5.4, "As shown in Table 5-3, the radial dose at 2 meters is most restrictive." Rather, Table 5-3 is a collection of three tables, with limited explanation of how the values are used in the analysis. Clarify what information Table 5-3 is intended to show. This addresses the requirements of 10 CFR 71.47.

**Table 5-3 has been corrected and split into two tables in the revised Chapter 5 of the Addendum.**

5-11 Clarify the meaning of the footnotes used in Table 5-3 for the last two tables entitled, "Dose Rates on Canister Surface from SAS4 (Monte Carlo)," and "Dose Rates on Container Surface from SAS1."

The footnotes state that respective values are either "based on most restrictive source term determined above," or "based on source term determined for bottom."

The location of "above" and "bottom" is unclear. This addresses the requirements of 10 CFR 71.47.

**The footnotes have been clarified in the revised Chapter 5.**

5-12 Clarify or illustrate the delineation between "radial," "top," and "bottom," with respect to the first table in Table 5-3 entitled, "(1-D SAS1) Oak Ridge Container in the TN-FSV Cask." In addition, clarify the basis for the source volumes used.

It is unclear why the top and bottom of one source have different source volumes. This addresses the requirements of 10CFR 71.47.

**Dose rates are calculated at the top and bottom of the TN-FSV cask and also in the radial direction. For this 1-D calculation, two axial and one radial model are used. The source volumes are discussed in Section 5.3.2. The source volumes in the Table have been corrected.**

5-13 Justify the following statements in Section 5.4, "An additional SAS1 run was made for the neutron dose, substituting U-235 for the U-238 to investigate subcritical multiplication. The result from the U-235 was not significantly different (<1% greater) than the U-238 case."

Provide information that would allow independent confirmation of this conclusion, such as a description of the analysis and the results. This addresses the requirements of 10 CFR 71.47.

**The SAS1 neutron dose input file ( radial and two axial models) was modified to replace the 200 g of U-238 in the source volume with 200 g of U-235 and rerun. This calculation was performed to quantify the increase in the predicted dose rates due to the source multiplication from the small amount of U-235 in the various canisters. The use of U-235 should enhance the source multiplication to some degree relative to the source multiplication from U-238 only. This increase in dose rates is offset by the conservatism in the amount of self-shielding material present. The assumed amount of 200 grams is far less than the typical amount of heavy metal material present. The resulting neutron dose rates were only slightly greater (< 1%) than the runs using U-238. Therefore, the results using U-238 should be conservative.**

## 6.0 CRITICALITY

6-1 Describe the effect on system reactivity if fissile material is released from the Oak Ridge Canisters or the Oak Ridge Container under hypothetical accident conditions.

The release of the fissile materials from the Oak Ridge Canisters can potentially result in increased system reactivity due to fissile material geometry and moderator mixture changes. Alternatively, provide justification for not analyzing this condition. Resolution of this issue is required to determine compliance with the nuclear criticality safety requirements specified in 10 CFR 71.55 and 10 CFR 71.59.

**The release of material from the canister can not happen. Sections 2.11.7.5, 1.2.1.3, and 6.2.2 of the Addendum have been revised to justify this statement.**

6.2 Revise the criticality analyses to include consideration of the fissile radionuclide uranium-233.

Table 1-3 includes uranium-233 in some of the Oak Ridge Canisters, however, uranium-233 is not included in the criticality analyses. The presence of uranium-233 in sufficient quantities within a fissile system could yield a significantly different value of $k_{eff}$ than that for a system without uranium-233. Either include the uranium-233 in the analyses or provide justification for neglecting it. In addition, revise Table 6-2 to include the limits on uranium-233 per Oak Ridge Canister, if appropriate. Note that Table 6-2 may be included as a condition of approval in the Certificate of Compliance. Resolution of this issue is required to determine compliance with the nuclear criticality safety requirements specified in 10 CFR 71.55 and 10 CFR 71.59.

**Uranium-233 is present in both the intact Peach Bottom HTGR fuel and Oak Ridge Canisters that contain sectioned HTGR fuel and KEMA materials. In all cases the presence of uranium-233 resulted from the transmutation of thorium during the irradiation process. None of the fuel materials contained uranium-233 prior to the irradiation process, and all of the fuel materials were subject to breeding ratios of < 1. Consequently, the configuration of the fuel material at the end of life with in-bred uranium-233 is less reactive than the beginning of life configuration prior to irradiation of the material. The criticality analyses is based on the pre-irradiation beginning of life configuration values for added conservatism.**

7.0 OPERATING PROCEDURES

7.1 Revise the operating procedures to clarify which steps are optional.

Section 7.0 implies that operational steps that are not underlined are optional. However, there appear to be some steps that are not underlined that should not be optional (e.g., placing the Oak Ridge Container lid on the container body). It is suggested that optional steps be labeled as such. Note that procedures that assure compliance with the requirements of 10 CFR 71.87 should not be optional.

**Chapter 7 has been revised to remove the underlined items and eliminate the impression that non-underlined items are optional.**

7.2 Justify why the drain O-ring on the TN-FSV cask does not require a pre-shipment leak test.

Procedure nos. 7.1.2.15 and 7.1.2.18 describe how the lid, and vent port cover on the TN-FSV cask are leak tested. However, there is no description of how the drain port O-ring is leak tested.

**The drain port is tested at the fabricators and is not utilized for these shipments. See note in Section 7.1.2.**

7.3 Editorial Comment:

It is suggested that consistent terminology be used throughout the operating procedures. For example, consistently use the phrase "Oak Ridge Container" instead of "Container," to avoid confusion.

**The Addendum has been revised to avoid this confusion.**

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8-1 Revise the application to specify which components and O-rings should be leak tested during fabrication for the TN-FSV cask and the Oak Ridge Container.

It is not clear which components and O-rings on the cask and Oak Ridge Container should be leak tested. This addresses the requirements of 10 CFR 71.51.

**Section 8.1.4 has been revised to identify the components and O-rings/ports that will be leak tested.**

8-2 Revise the application to specify which alternative leak test methods are acceptable and why, for the fabrication and periodic leak tests.

Section 8.1.2 and 8.2.2 state that alternative leak test methods are acceptable. Provide a description of the alternative methods that may be used, and show that the sensitivities are adequate. This addresses the requirements of 10 CFR 71.51.

**The Sections have been revised to state that helium mass spectrometer method will be used.**

8-3 Revise the application to include additional details regarding the fabrication and periodic leakage tests.

Sections 8.1.4 and 8.2.2 state that tests are usually performed using the helium mass spectrometer method. However, the applicant should describe the steps used to confirm that the leak rates are within acceptable limits. This addresses the requirements of 10 CFR 71.51.

**The helium mass spectrometer method will be used to perform the leakage tests. For the Oak Ridge Container, helium will be introduced into the cavity and the mass spectrometer connected to the test port in order to test both the inner lid o-ring and the vent port o-ring, the quick disconnect in the vent port will be removed. The ORC body/lid can be tested by the gas filled envelope method, i.e., placing a helium filled bag around the body and evacuating the cavity through the drain port. The combined leakage rate must be less than $1x10^{-7}$ ref cc/s.**

**For the TN-FSV Packaging, the inner lid o-ring is tested by utilizing the test port connection for the mass spectrometer with helium in the cask cavity. The body, vent, and drain ports can be tested with the gas filled envelope method. The combined leakage rate must be less than $1x10^{-7}$ ref cc/s.**

**The Addendum has been revised to include this information.**

8.4 Describe how the helium mass spectrometer leak test is performed for the vent and drain ports on the TN-FSV cask, since they are single O-ring configuration.

Sections 8.1.4 and 8.2.2 state that tests are usually performed using the helium mass spectrometer method. However, this method is usually used for double O-ring configurations. Clarify how the tests will be performed for the single O-ring configuration. Clarify how it is assured that helium is in the space between the quick-disconnect fitting and the containment seal. This addresses the requirements of 10 CFR 71.51.

**One possibility is to remove the quick disconnect from the vent port, helium bag the lid area, and evacuate the cavity to the helium mass spectrometer, (gas filled envelope method). This tests the vent port o-ring. Reverse the procedure to test the drain port.**

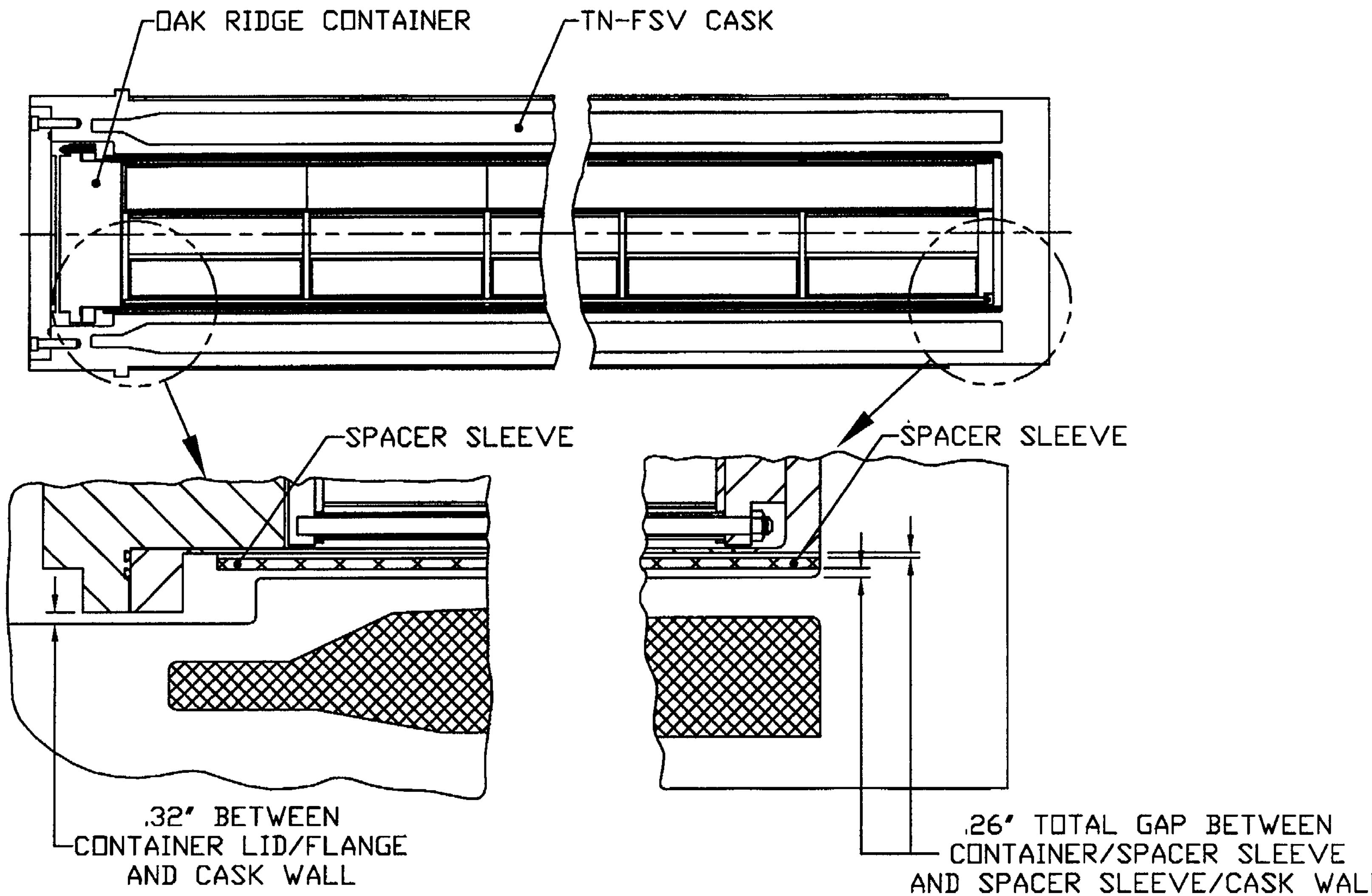
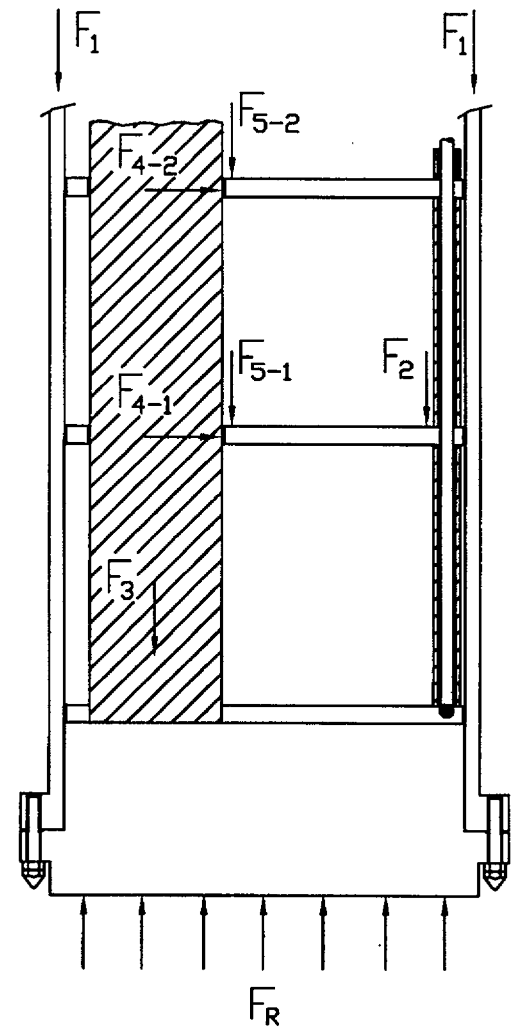



**Figure 1**
**Location and Geometry of Aluminum Spacer Sleeve**

**Figure 2.11.1-1**
**Free Body Diagram of Oak Ridge Container**
**Subjected to a Vertical or Near Vertical Lid End Drop**




FREE BODY DIAGRAM OF LID END DROP

$F_1$ = INERTIA LOADS OF CONTAINER SHELL + FLANGE + BOTTOM

$F_2$ = INERTIA LOADS OF 8 DISCS + 1.75"THK. CANISTER BOTTOM DISC + 8 POISON ENCLOSURES INCLUDING POISON PLATES

$F_3$ = INERTIA LOADS OF CANISTERS + FLUX TRAPS + FUEL COMPARTMENTS

FOR VERTICAL END DROP

$F_4 = F_5 = 0$

$F_R$ = TOTAL REACTION FORCE= $F_1 + F_2 + F_3$

FOR DROP ANGLE OTHER THAN VERTICAL END DROP

$F_4 = F_{4-1} + F_{4-2} + F_{4-3} + F_{4-4} + F_{4-5} + F_{4-6} + F_{4-7} + F_{4-8} + F_{4-9}$

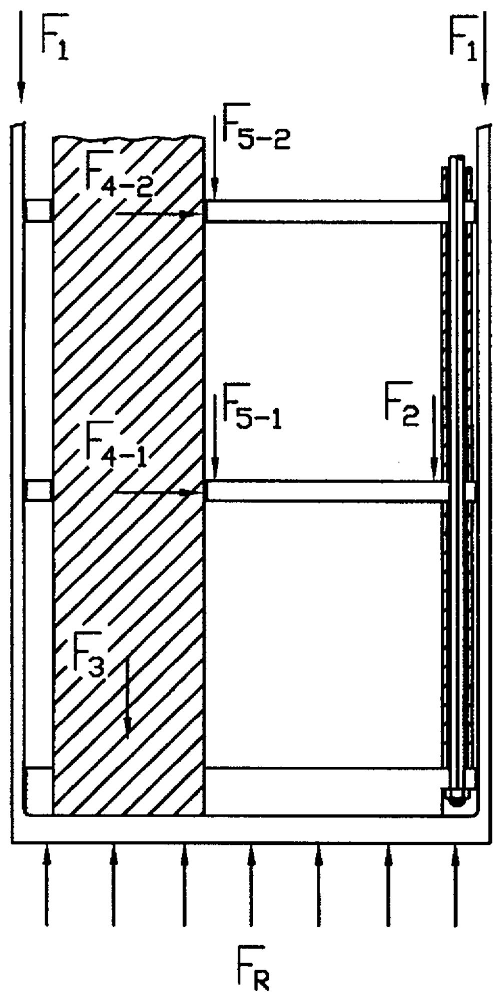
$F_5$ = TOTAL AXIAL FRICTION FORCE

$= F_{5-1} + F_{5-2} + F_{5-3} + F_{5-4} + F_{5-5} + F_{5-6} + F_{5-7} + F_{5-8} + F_{5-9}$

$= (F_{4-1} + F_{4-2} + F_{4-3} + F_{4-4} + F_{4-5} + F_{4-6} + F_{4-7} + F_{4-8} + F_{4-9}) \times \mu$

$F_R$ = TOTAL REACTION FORCE= $F_1 + F_2 + F_3 + F_5$

**Figure 2.11.1-2**
**Free Body Diagram of Oak Ridge Container**
**Subjected to a Vertical or Near Vertical Bottom End Drop**




FREE BODY DIAGRAM OF BOTTOM END DROP

$F_1$ = INERTIA LOADS OF CONTAINER SHELL + FLANGE + LID

$F_2$ = INERTIA LOADS OF 9 DISCS + 8 POISON ENCLOSURES INCLUDING POISON PLATES

$F_3$ = INERTIA LOADS OF CANISTERS + FLUX TRAPS + FUEL COMPARTMENTS

FOR VERTICAL END DROP

$F_4 = F_5 = 0$

$F_R$ = TOTAL REACTION FORCE= $F_1 + F_2 + F_3$

FOR DROP ANGLE OTHER THAN VERTICAL END DROP

$F_4 = F_{4-1} + F_{4-2} + F_{4-3} + F_{4-4} + F_{4-5} + F_{4-6} + F_{4-7} + F_{4-8} + F_{4-9}$

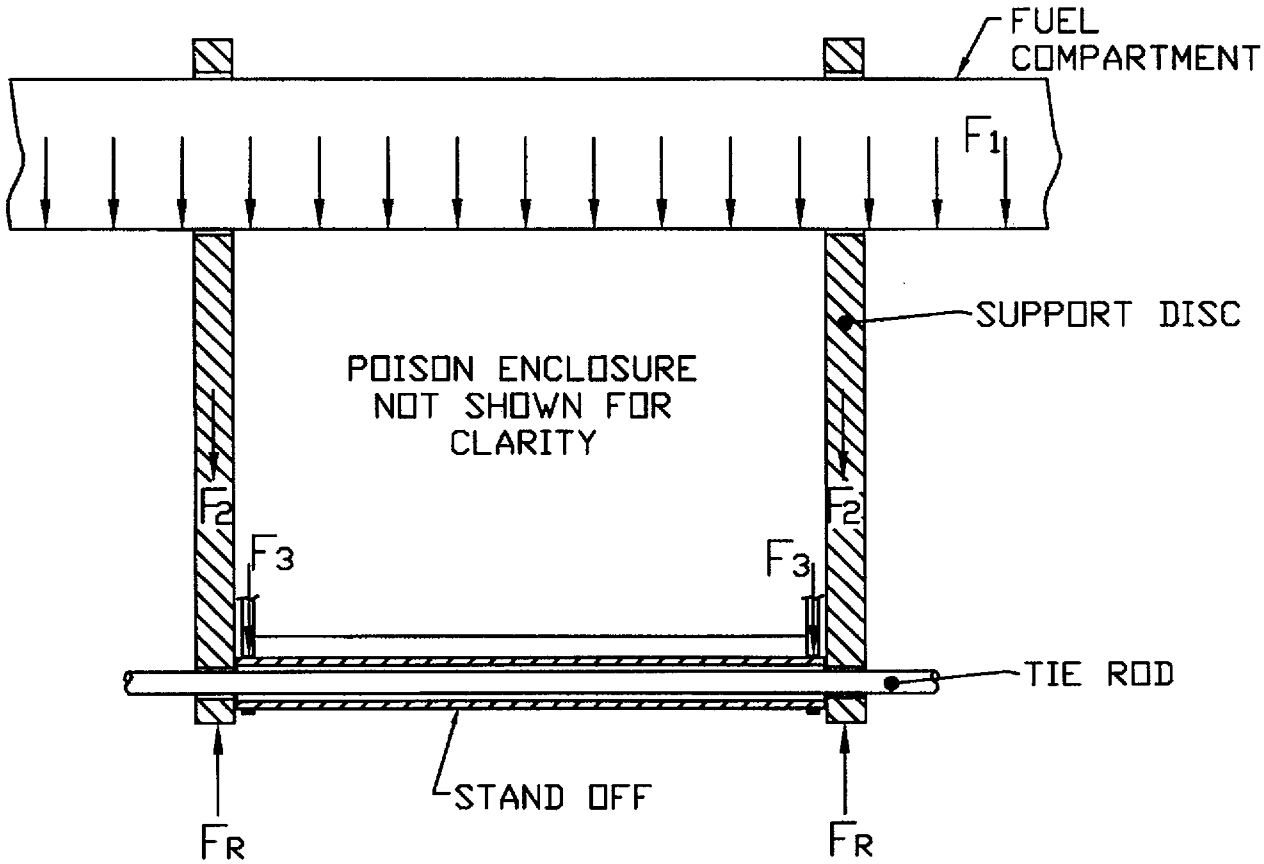
$F_5$ = TOTAL AXIAL FRICTION FORCE

$= F_{5-1} + F_{5-2} + F_{5-3} + F_{5-4} + F_{5-5} + F_{5-6} + F_{5-7} + F_{5-8} + F_{5-9}$

$= (F_{4-1} + F_{4-2} + F_{4-3} + F_{4-4} + F_{4-5} + F_{4-6} + F_{4-7} + F_{4-8} + F_{4-9}) \times \mu$

$F_R$ = TOTAL REACTION FORCE= $F_1 + F_2 + F_3 + F_5$

**Figure 2.11.1-3**
**Free Body Diagram of Oak Ridge Container Fuel Basket Subjected to a Side Drop**




FREE BODY DIAGRAM OF SIDE DROP

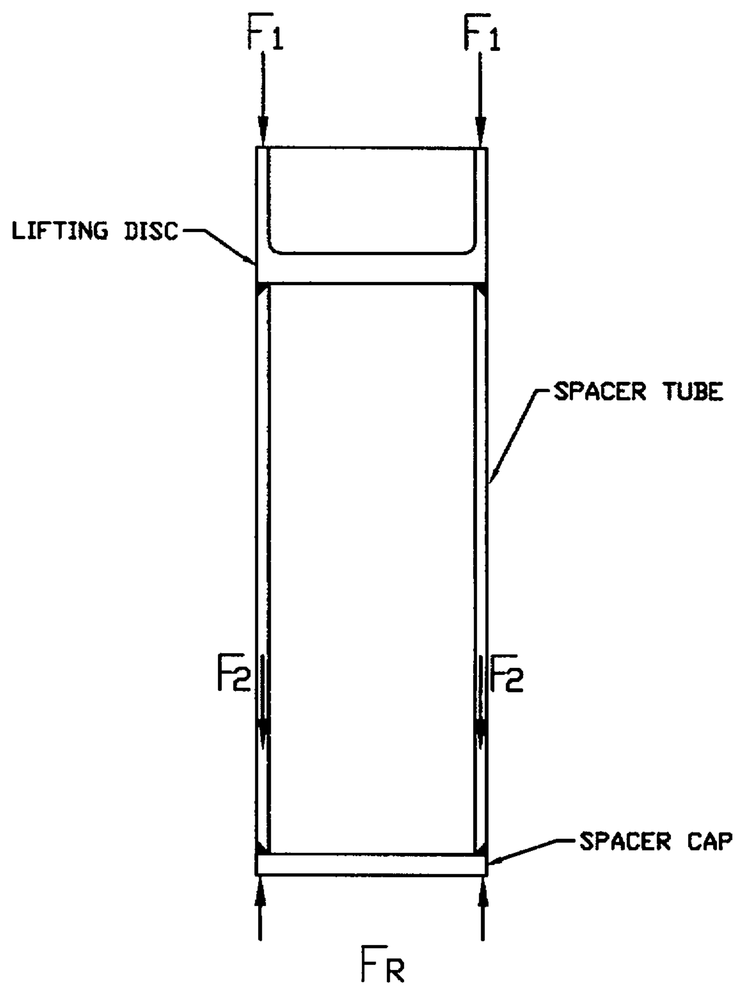
$F_1$ = INERTIA LOADS OF CANISTERS + FUEL COMPARTMENT + FLUX TRAPS

$F_2$ = INERTIA LOADS OF SUPPORT DISC + TIE ROD + STAND OFF

$F_3$ = INERTIA LOADS OF POISON ENCLOSURE + POISON PLATES

$F_R$ = TOTAL REACTION FORCE= $F_1 + F_2 + F_3$

**Figure 2.11.1-4**
**Free Body Diagram of Oak Ridge Container**
**Flux Trap Subjected to a Lid End Drop**




FREE BODY DIAGRAM OF BOTTOM FLUX TRAP DURING BOTTOM END DROP

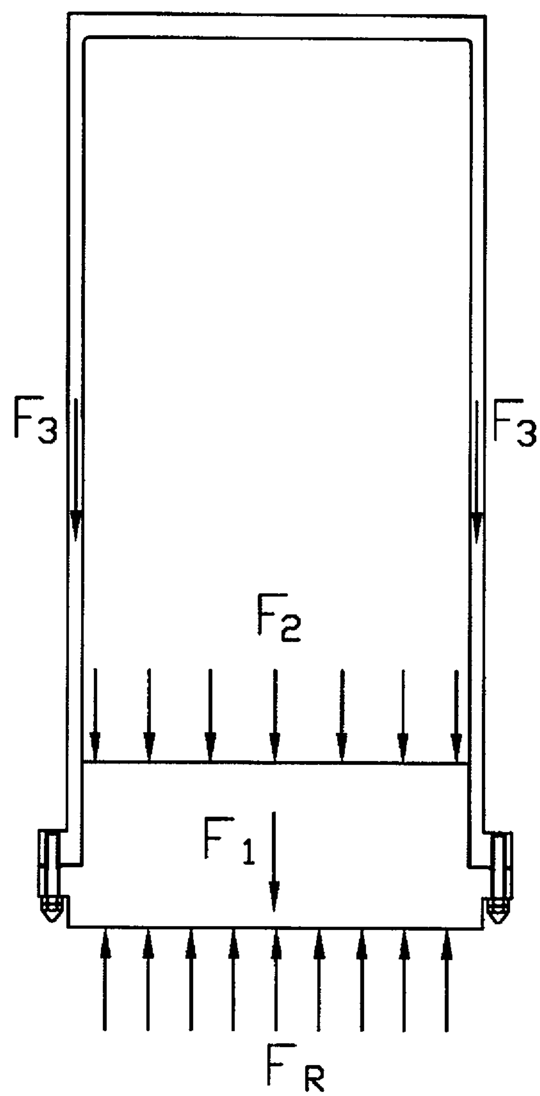
$F_1$ = TOTAL INERTIA LOAD OF CANISTERS + FLUX TRAPS ABOVE

$F_2$ = INERTIA LOAD OF BOTTOM FLUX TRAP

$F_R$ = TOTAL REACTION FORCE= $F_1 + F_2$

**FIGURE 2.11.4-3**

**FREE BODY DIAGRAM OF OAK RIDGE CONTAINER LID CLOSURE SYSTEM SUBJECTED TO A LID END DROP**




FREE BODY DIAGRAM OF OAK RIDGE CONTAINER DUE TO LID END DROP

$F_1$ = INERTIA LOAD OF LID

$F_2$ = INERTIA LOAD OF INTERNALS

$F_3$ = INERTIA LOADS OF CONTAINER FLANGE + SHELL + BOTTOM

$F_R$ = TOTAL REACTION FORCE= $F_1 + F_2 + F_3$