ENCLOSURE 4

M190138

Amended Sections and Pages for NEDO-33866 GE2000 SAR
Revision 4

Non-Proprietary Information

**IMPORTANT NOTICE**

This is a non-proprietary version of Enclosure 3 to M190138, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

# 1 GENERAL INFORMATION

## 1.1 Introduction

The Model 2000 Radioactive Material Transport Package was developed at Vallecitos Nuclear Center. The primary use of the packaging is to provide containment, shielding, impact resistance, criticality safety, and thermal resistance for its contents during normal and hypothetical accident conditions. The packaging is designed to transport Type B quantities of radioactive materials. It complies with the Nuclear Regulatory Commission (NRC) regulations contained in the Code of Federal Regulations, Title 10, Part 71 (10 CFR 71). The package is to be shipped in all modes of transportation, except air. The Model 2000 Transport Package may only be shipped exclusive use, as discussed in Section 5.1.2. The Criticality Safety Index (CSI) is determined to be 50, as discussed in Section 6.1.3.

Calculations, engineering logic, and all related documents that demonstrate compliance with regulations are presented in subsequent sections of this report.

The GEH Quality Assurance Program (QAP-1) (Reference 1-1) controls design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and modification of the packages. The NRC has approved QAP-1 under Docket Number 71-0171 upon demonstration that the QA plan meets the requirements of Subpart H of 10 CFR 71.

## 1.2 Package Description

The Model 2000 Transport Package, shown in Figure 1.2-1, is transported exclusive use, in the upright position. The approximate overall packaging dimensions are 131.5 inches in height and 72 inches in diameter. The approximate total weight of the package (packaging plus the contents) is 33,550 lb. Table 2.1-3 shows the breakdown of the component weights for the Model 2000 Transport Package.

The Model 2000 Transport Package and contents are described below:

Packaging

- Cask
- Overpack
- High performance insert (HPI)
- HPI material basket

Contents

- Solid radioactive materials

#### 1.2.1.4. HPI Material Basket

The material basket is shown in Figure 1.2-5 with an example of supplemental dunnage. The material basket is constructed of [[ ]] pattern and are identified as Item 1 on Drawing 001N8424. See Figure 1.2-6 for material basket details. The outer [[ ]] of the material basket form a composite section with the addition of [[ ]] The center location of the material basket is a developed cell, which is created by the surrounding [[ ]] To allow for the proper insertion of supplemental dunnage and facilitate fabrication, [[ ]] are inserted at the top and bottom of the developed cell and are identified as Item 2 on Drawing 001N8424. Therefore, the exterior view of the material basket shows [[ ]] facilitate loading and positioning of the material basket within the HPI cavity. Parts List 001N8424G001 is provided in Section 1.3.

### 1.2.2. Contents

#### 1.2.2.1. Radioactive Material Contents

The Model 2000 Transport Package is designed to transport Type B quantities of radioactive materials. This may include irradiated hardware and byproducts, Co-60 isotope rods, or irradiated fuel. The following are requirements for all shipments:

a) The maximum quantity of material per package shall not exceed 5,450 lb, including all cask internals and contents (defined as "payload" for purposes of this report – see Table 2.1-3).
b) All contents shipped shall be in solid form.
c) All configurations require the use of the HPI.
d) The decay heat for shipping all contents shall be limited to no more than 1500 W. However, a decay heat of 3000 W is conservatively used as the design basis for the Model 2000 Transport Package, where applicable. There are a few exceptions as noted within this SAR where 1500 W forms the basis; while a 1500 W decay heat is used in these sections, it is demonstrated that the 3000 W design basis is bounding.

The specific radioactive contents transported in the Model 2000 cask are:

1. Irradiated Hardware and Byproducts
    a. Irradiated hardware components composed of stainless steels, carbon steels, nickel alloys, and zirconium alloys.
    b. Irradiated byproducts such as control rods and/or blades composed of hafnium and boron carbide.
    c. Minimum decay time shall be at least 30 days prior to shipment.
    d. Refer to loading table provided in Section 7.5.1
2. Cobalt-60 Isotope Rods
    a. Must be shipped with the HPI material basket in the upright position and confined per 2.b and demonstrated to meet NCT.

    b. Content shall be in the form of pellets or cylindrical solid rods with the source(s) evenly distributed and encapsulated in normal or special form.
    c. Total activity in any axial 1-inch increment in the HPI cavity must be ≤ 17,000 Ci (see Section 7.5.2).
3. Irradiated fuel
    a. Fuel type is GNF BWR 10x10 fuel.
    b. Minimum cooling time of 120 days.
    c. The active fuel length of any segment must be at least 5.3 inches.
    d. Must be shipped with the HPI material basket in the upright position.
    e. Maximum initial U-235 enrichment of 6 wt%.
    f. Maximum U-235 mass of 1750 g.
    g. Maximum burnup of 72 GWd/MTU.
    h. Refer to loading table provided in Section 7.5.3.

Shipment of combined contents is allowed.

### 1.2.3. Special Requirements for Plutonium

All contents in the Model 2000 Transport Package are in solid form. Thus, any plutonium in excess of 0.74 TBq (20 Ci) per package shall be in solid form.

### 1.2.4. Operational Features

The Model 2000 Transport Package description in Section 1.2.1 shows that the packaging is not a complex system. There are no valves or items that require specialized knowledge for proper operation, and cooling is provided through natural convection and radiation. [[ ]] during installation, and only normal practices for seal handling (e.g., cleanliness) are required.

The Model 2000 Transport Package operation is described in Chapter 7. The loading operation is a dry or wet-loaded operation. If wet-loaded, the cask and cask internals contain features to allow easy drainage of water for underwater loading. To vacuum dry the cask, its cavity pressure is reduced below the vapor pressure of water and maintained at or below this pressure level for a period of time.

Content shoring may include components such as the rod [[ ]] holders shown in Figure 1.2-5. This example shoring is designed to fit into the HPI material basket (Drawing 001N8424), but other shoring components may be placed directly into the HPI cavity (Drawing 001N8423). The HPI material basket is loaded into the HPI cavity (Figure 1.2-4) if required for a specific content.

When the HPI top plug is installed (Drawing 001N8427), additional shoring may be added, as necessary, to ensure the [[ ]] between the bottom of the cask lid and the top of the HPI does not exceed 0.25 inches. However, no credit for shoring is given in the Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) evaluations. The required evaluations are included in this application to demonstrate safe transport of the Model 2000 Transport Package for the included contents with specified required internals.

### 2.6.7.3. Material Basket Evaluation

This section evaluates the material basket for NCT. Factors of safety for the basket are calculated based on the criteria for Service Level 'A' limits from ASME Section III-NF (NF-3221).

**End Drop Case**

During the end drop the material basket is loaded by inertia loads acting on the end of the [[ ]] Depending on the orientation of the outer cask when dropped, the basket contents will either load the material basket or the lid of the high performance insert (HPI). There is a washer welded to the bottom of the basket [[ ]] that holds the rod holders. Nothing prevents the rod holders from exiting the top of the material basket. If the outer cask is dropped while in the upright position, the material basket will be loaded by the contents. The worst-case condition of upright end drop is evaluated. The inertial loading will load the [[ ]] bundle in compression. There is no bending or shear stress present. For this evaluation, all 18 full length [[ ]] are loaded.

Stresses at bottom of [[ ]]:

$$\sigma_{\text{membrane}} = \frac{P}{A} = 837 \text{ psi compression}$$

$$\sigma_{\text{bending}} = 0 \text{ psi}$$

$$\tau_{\text{shear}} = 0 \text{ psi}$$

where $P = W \times G = 4913.5$ lbs inertial load on 18 [[ ]] bundle

$W =$ [[ ]] lbs basket plus contents weight

$G = 15.5$ G NCT end drop acceleration

$A =$ [[ ]] $\text{in}^2$ Cross section area of [[ ]], Table 2.6.7-18)

$S_y = 17700$ psi Yield Strength, 316 stainless steel, 800°F (Table 2.2-2)

Minimum Margin of Safety:

The minimum margin of safety for the NCT end drop case is:

$$MS = \frac{S_y}{\sigma_m} - 1 = \frac{17700}{837} - 1 = +20.1$$

**Side Drop Case**

During the side drop the steel [[ ]] provides a close fit with the high performance insert inner shell, which distributes the inertial load as three beam segments along the length of the basket assembly. The basket is assembled using short [[ ]] at each end of the basket starting at the center location. To provide strength to the basket assembly, [[ ]] are added between the [[ ]] at the outside of the assembly forming a [[ ]] shape. For this evaluation it is assumed that only the outer [[ ]] carry the load. Additionally, no credit is taken for the [[ ]], which is significantly stiffer than the individual [[ ]] The basket is analyzed using classical hand calculations for a 55.1 g side drop inertia load and a bounding weight of [[ ]] pounds. Assuming one-third of the inertial load is carried by one of the equivalent beam segments, the bending stress in the basket is:

#### 2.7.1.2.3. Corner Drop

Results of the LS-DYNA analysis presented in Section 2.12.1 shows that the side drop accelerations bound the corner drop.

#### 2.7.1.2.4. Oblique Drops

Results of the LS-DYNA analysis presented in Section 2.12.1 shows that the side drop accelerations bound the oblique drop angles.

#### 2.7.1.2.5. Cask Overpack Bolt Evaluation

Bolt Torque

Per Model 2000 cask overpack drawings 101E8719 and 105E9521 (Table 1.3-1), the overpack bolt torque is 100±5 lb-ft dry. The following overpack evaluation assumes a maximum torque of 105 lb-ft.

Bolt Evaluation Procedure

This analysis is based on the procedure outlined in NEDE-31581, Subsection 2.10.7, which was developed to account for the overpack fastener failure during the quarter-scale model side drop test. Once the procedure was satisfactorily developed to explain the fastener failure, it was used to redesign the fastening system. This section presents the steps and results of this analysis as applied to the Model 2000 with the HPI.

Bolt Stresses – HAC Side Drop

The Model 2000 transport package overpack is fastened together with 15 equally spaced ASTM A-540 Grade B22, Class 3 or equivalent 7/8-9 UNC socket head shoulder bolts. The adequacy of these fasteners is determined by comparing the service loads (from the HAC) to the allowable loads, using the criteria given in the ASME Code, Section III, Division 1, Appendix F.

Bolts: 7/8-9 UNC-2A, ASTM A540 Grade B22, Class 3, 15 equally spaced

Tensile area of threaded portion = 0.462 $in^2$

Proof Strength = Minimum Yield Strength x 85% = 115700 (0.85) = 98345 psi

Loading: The highest stresses for the overpack fasteners occur during the HAC side drop accident condition. The maximum load is calculated for an impact acceleration of 161.9 g's.

For the side drop case, the load is applied to the overpack junction as shown in Figure 2.7.1-12. The overpack is modeled as a simple beam with the force of the cask and contents as a distributed load and the neutral axis at the side of the overpack opposite the side of impact.

### 2.7.5. Immersion - Fissile Material

Subpart F of 10 CFR 71 requires performing an immersion test for fissile material packages in accordance with the requirements of 10 CFR 71.73(c)(5). The criticality evaluation presented in Chapter 6.0 assumes optimum hydrogenous moderation of the contents, thereby conservatively addressing the effects and consequences of water in-leakage.

### 2.7.6. Immersion - All Packages

According to the requirements of 10 CFR 71.73(c)(6), a package must be subjected to water pressure equivalent to immersion under a head of water of at least 15 meters (50 feet) for a period of 8 hours, which is equivalent to 21.7 psig. The cask closure including the lid and bolts are designed to survive puncture loads, which exceed the load experienced during immersion (Sections 2.12.1 and 2.12.4). From ASME Section III-NB, A-2221, when subjected to 21.7 psig the 1.0-inch thick outer shell of the cask with a mean radius of 18.75 inches, produces a primary membrane stress intensity 418 psi that is much less than the material yield strength. Therefore, the Model 2000 Transport Package satisfies all of the immersion requirements for a package that is used for the international shipment of radioactive materials.

### 2.7.7. Deep Water Immersion Test (for Type B Packages Containing More than $10^5$ $A_2$)

The contents specified in this application is less than $10^5$ $A_2$. Therefore, this is not applicable for the Model 2000 Transport Package with HPI and material basket.

### 2.7.8. Summary of Damage

The analytical results reported in Sections 2.7.1 through 2.7.7 indicate that the damage incurred by the Model 2000 Transport Package during the hypothetical accident is minimal, and such damage does not diminish the cask ability to maintain the containment boundary. A 30-foot side drop followed by the 40-inch pin puncture accident may damage the overpack and inflict local damage on the outer shell of the cask. However, the shielding remains intact and satisfies the accident shielding criteria. Additionally, the HPI and material baskets maintain structural integrity during all postulated HAC events, which supports the criticality and shielding analysis assumptions. Based on the analyses of Sections 2.7.1 through 2.7.7, the Model 2000 Transport Package fulfills the structural and shielding requirements of 10 CFR 71.73 for all of the hypothetical accident conditions.

## 2.8 Accident Conditions for Air Transport of Plutonium

This section does not apply for the Model 2000 Transport Package with HPI and material basket.

## 2.9 Accident Conditions for Fissile Material Packages for Air Transport

This section does not apply for the Model 2000 Transport Package with HPI and material basket.

$F_6\,56.8 + (0.8898\,F_6)x169.1 = 2.134\,(10^7)$

$F_5 = F_6 = 102{,}960.00$
$F_7 = F_8 = 0.8898\ F_6 = 91{,}614.00$
$RF = F5y + F6y + F7y + F8y - Wa$
$= 0.774F5 + 0.774F6 + 0.654F7 + 0.654F8 - 335{,}500 = -56{,}287$ lb

5g Transverse

This time the 5g acceleration will cause the package to rotate at a point 90° clockwise from point "o". This will cause ropes 2, 4, 6 and 8 to go slack, and tension ropes 1, 3, 5 and 7. From symmetry the following assumptions can be made with reference to Figure 2.12.3-13.

$F_1 = F_7$
$F_3 = F_5$

The component forces for these ropes are:

$F_3x = F_5x = F_5 \cos 23.2° \sin 32.7° = 0.497\,F_3$
$F_3z = F_5z = F_5 \sin 23.2° = 0.394\,F_3$
$F_1x = F_7x = F_7 \cos 46.3° \sin 18.69° = 0.221\,F_1$
$F_1z = F_7z = F_7 \sin 46.3° = 0.723\,F_1$

The reaction forces from chocking and friction (RF) and bearing on the package base (RB) are:

$RF = F_5x + F_3x + F_7x + F_1x - Wa$
$RB = F_5z + F_3z + F_7z + F_1z + Wg$

RB is calculated assuming $F_2 = F_4 = F_6 = F_8 = 0$

$\Sigma Mox = 0 = -\,Wa\,63.60 + Wg\,24.25 - RB\,24.25 + (F_5x + F_3x)25.5 + (F_7x + F_1x)$

$105.0 + (F_5z + F_3z + F_7z + F_1z)\,40.12$

$\Sigma Mox = -\,Wa\,63.60 + Wg\,24.25 - (F_5z + F_3z + F_7z + F_1z + Wg)24.25 + \ldots$

$\ldots (F_5x + F_3x)25.5 + (F_7x + F_1x)105.0 + (F_5z + F_3z + F_7z + F_1z)40.12$

$= -\,Wa\,63.60 - [2 \times 0.394F_3 + 2 \times 0.723F_1]24.25 + 2 \times 0.497F_3(25.5) + \ldots$

$\ldots 2 \times 0.221F_1\,(105.0) + [2 \times 0.394F_3 + 2 \times 0.723F_1]40.12$

$= -Wa\,63.60 - 19.1F_3 - 35.1F_1 + 25.3F_3 + 46.41F_1 + 31.6F_3 + 58F_1$

$= -Wa\,63.60 + (-19.1 + 25.3 + 31.6)\,F_3 + (-35.1 + 46.41 + 58)\,F_1$

$= -\,(63.60)\,Wa + (37.8)F_3 + (69.31)F_1$

$\Rightarrow 1.063\,(10^7) = (37.8)F_3 + (69.3)F_1$*

From equation** $F_5$ and $F_7$ are related as:

$F_7 = 0.8898F_5$
$= 0.8898F_3 = F_1$
$\Rightarrow 1.063\,(10^7) = (37.8)F_3 + (69.3)(0.8898F_3)$*
$F_3 = 106{,}874$

# 3 THERMAL EVALUATION

This section presents the thermal evaluation of the Model 2000 Transport Package and high performance insert (HPI) with a contents thermal loading of 1500 W and a contents thermal loading of 3000 W, both under Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) as prescribed by 10 CFR 71 (Reference 3-1). The 3000 W results are presented in Sections 3.3 and 3.4, and the 1500 W results are presented in Section 3.5.1.

The decay heat limit for shipping all contents shall be conservatively limited to 1500 W. However, a decay heat of 3000 W is evaluated as the basis for the Model 2000 Transport Package in Chapter 2.

Specifically, the following requirements of 10 CFR 71 are addressed:

1) General standards for all packages, 10 CFR 71.43(g)
2) Normal Conditions of Transport—heat, 10 CFR 71.71(c)(1)
3) Hypothetical Accident Conditions—thermal, 10 CFR 71.73(c)(4)

## 3.1 Description of Thermal Design

### 3.1.1. Design Features

The Model 2000 Transport Package, described in Section 1.2, is designed with a thermally passive system. The cask is enclosed in an overpack that serves as a fire shield. The overpack is designed to reduce heat flow from the fire environment into the cask structure by the use of enclosed air spaces. It is composed of two concentric cylindrical SS304 shells approximately 83 inches long with an OD of 48.5 inches and an ID of 40.5 inches. The shells are separated radially by eight equally spaced [[ ]] along the length of the shells, and horizontally by two [[ ]] sections to provide closed air spaces. A 24-inch diameter toroidal shell is attached at both ends of the outer shell with a circular plate enclosing the inner regions of the torus. The internal shell is also closed at each end by a circular plate. All materials are SS304. The [[

]] Attached at both ends of the overpack inner surface are aluminum honeycomb pads.

The cask is designed with lead shielding on three sides and a 6-inch thick stainless steel forging at the base that functions as a heat sink that allows the heat to flow through the bottom of the package. When the cask is placed in the overpack during assembly, air gaps of 1.0-inch radially and 1.0-inch at the top separate the cask from the overpack inner surfaces.

The cask lid seal design, which includes an [[ ]] and metal retainer component, is based on a 1500 W content decay heat. The cask lid seal is a [[ ]] retainer with four Parker Compound No. [[ ]] rings, two concentric [[ ]] seals on the top and two [[ ]] seals on the bottom. See Chapter 4 for further discussion. The cask lid is secured to the cask body by fifteen (15) 1¼-inch diameter socket head screws.

The HPI is described in Section 1.2.1.3.

### 3.1.2. Content's Decay Heat

The derivations of the decay heats for the different contents of the Model 2000 Transport Package are presented in Chapter 5. The decay heat for irradiated hardware and by-product, cobalt-60 isotope rod, and irradiated fuel contents is determined using watt-per-Curie conversion factors listed in Section 5.5.4 and the radionuclide inventory of the contents.

### 3.1.3. Summary Tables of Temperatures

Thermal design criteria are specified for regions throughout the cask, cask cavity, and the outside overpack wall. The cask lid seal and port O-ring is limited to the temperature listed in Table 3.1.3-1, and this serves as the thermal criteria for the region associated with the seal area. The maximum allowable internal pressure is 30 psia, which corresponds to air of 100% humidity heated to 600°F at constant volume.

Table 3.1.3-1 presents the maximum design temperatures of the components or materials that affect structural integrity, containment, and shielding under both NCT and HAC for 3000 W of decay heat. Where available, temperature limits for the Model 2000 Transport Package components are obtained from manufacturers' literature. Otherwise, component temperature limits are defined as the melting temperature of the material of construction.

**Table 3.1.3-1. Temperature Limits**

| Component or Material | Temperature Limit (°F) |
|---|---|
| Stainless Steel Components | 2546 |
| Lead Shielding[a] | 622 |
| Depleted Uranium Shielding[a] | 2071 |
| Aluminum Honeycomb[b] | 350 |
| Cask Lid Seal | 5 to 508[c] |
| Cask Ports | -40 to 612[c] |
| Accessible Surfaces of Package | < 185[d] |

Notes:
a. Temperature limit is melting temperature (Reference 3-3).
b. Maximum operating temperature (Reference 3-4).
c. See Chapter 4 for additional discussion.
d. Exclusive use requirement per 10 CFR 71.43(g).

### 3.1.3.1. NCT Temperature Summary

Per the requirements of 10 CFR 71.71(c)(1) (Reference 3-1), the 3000 W case is evaluated for NCT. Specifically, a steady-state thermal analysis is performed simulating exposure of the package to a 100°F ambient temperature in still air and insolation as specified in 10 CFR 71.71. The results of the analysis are presented in Section 3.3. The temperatures of several key package components are summarized and compared with their allowable temperatures in Table 3.1.3-2.

**Table 3.1.3-2. NCT Temperature Summary and Comparison with Allowable Temperatures**

| Item | NCT Temperatures (°F) | Allowable Temperature (°F) |
|---|---|---|
| Material Basket | 1,001 (max) | 2,546 |
| HPI Shielding (Depleted Uranium) | 601 (max) | 2,071 |
| Cask Lid Seal | 432 (max) | 508[c] |
| Cask Shielding (Lead) | 449 (max) | 622 |
| Honeycomb Impact Limiters | 359 (max)[a] / 334 (avg) | 350 |
| Cask Drain Port (Bottom) | 370 | 612[c] |
| Cask Test Port (Top) | 426 | |
| Cask Vent Port (Lid) | 442 | |
| Overpack Outer Surface | 215 | 185[b] |

Notes:
a. The maximum honeycomb impact limiter temperature of 359°F exceeds the allowable temperature of 350°F. However, this maximum temperature occurs in a very limited area of the impact limiter and is based on steady-state boundary conditions for the hot case, which ignores the removal of solar insolation during the night cycle. The majority of the impact limiter temperatures are below 350°F. Therefore, the average temperature of 334°F is appropriate to compare to the allowable temperature.
b. Limit specified in 10 CFR 71.43(g), which requires the addition of a personnel barrier to satisfy this requirement. Refer to Section 7.1.4, Preparation for Transport.
c. See Chapter 4 for additional discussion.

## 3.2 Material Properties and Component Specifications

### 3.2.1. Material Properties

The thermal properties of the materials of construction used in the analyses for the thermal evaluation are presented in Table 3.2.1-1. When available from the open literature, temperature-dependent properties are used in the analyses. Additionally, the thermal properties of the cask fill gas (helium) and overpack gas (air) are presented in Table 3.2.1-2 (Reference 3-3).

### 3.2.2. Component Specifications

The Model 2000 Transport Package component materials are primarily stainless steel, lead, and aluminum. The maximum allowable temperatures of these materials are given in Table 3.1.3-1. The temperatures resulting from normal and accident thermal conditions fall within these temperatures.

The only component material that is temperature sensitive is the [[ ]] material in the cask lid seal and port plug O-rings. The material used is [[ ]] that offers an operating temperature range adequate for 1500 W decay heat (Reference 3-8). The cask lid seal design includes an aluminum [[ ]]; temperatures resulting from normal and accident thermal conditions fall within the material limits. See Chapter 4 for additional discussion.

## 3.3 Thermal Evaluation under Normal Conditions of Transport

Thermal performance for 3000 W decay heat is analyzed for NCT (with and without insolation) by performing steady-state heat transfer analyses on a finite element representation of the package. Specifically, the general-purpose finite element code ANSYS, Release 14.0 (Reference 3-2), is used to model and analyze the Model 2000 Transport Package with a content heat load of 3000 W for NCT. Several ANSYS macros are created in order to build the model, apply boundary conditions, and perform the steady-state analyses.

Assumptions made for this evaluation are:

- The Model 2000 Transport Package is assumed to be in an upright (vertical) orientation during NCT.
- The cask and HPI are backfilled with Helium at 70°F and 14.7 psia.
- Natural convection within the package cavities is neglected.
- The contents of the HPI are assumed to generate a maximum of 3000 W that is uniformly distributed among the [[ ]].
- During NCT, the package is assumed to have an emissivity consistent with the material of construction at temperature.

As mentioned above, for the NCT analysis, a steady-state thermal analysis was performed simulating exposure of the package to a 100°F ambient temperature in still air and insolation as specified in 10 CFR 71.71. The results of the analysis are presented below. The temperatures of the key package components are summarized and compared with their allowable temperatures in Table 3.1.3-2.

NCT sensitivity studies were also performed to evaluate the thermal performance of the package using boundary conditions applied as both steady state and as constant boundary conditions solved as a transient. Because the solutions are radiation-dominated, the transient solution results in better convergence and slightly higher temperatures. To achieve steady-state conditions with the transient solver, a simulated time of 2000 hours was used. The 2000-hour duration of the transient analyses is sufficiently long enough for the temperatures within the package to reach steady-state

[[

]]

**Figure 3.3-8. Contents Heat Flux Applied to Material Basket** [[ ]]

### 3.3.1. Heat and Cold

#### 3.3.1.1. Hot Case

##### 3.3.1.1.1. NCT Solar Heat Flux (Insolation)

Per the requirements of the regulations for NCT, the Model 2000 Transport Package is exposed to an ambient temperature of 100°F and insolation according to 10 CFR 71.71. The solar heat fluxes specified in 10 CFR 71.71 are per 12-hour period. This 12-hour period represents a 12-hour long "day" in a 24-hour day/night cycle. Because the solar heat flux is constant, the insolation value should be time averaged over 24 hours in order to maintain the proper total heat flux to the package over the full day/night cycle. Therefore, to simulate a day-night cycle, these heat fluxes are time-averaged over a 24-hour period as follows:

Flat surfaces (other than transported horizontally base)

$$q" = \frac{800\ cal/cm^2}{24\ h}\left(\frac{4.1868\ J}{cal}\right)\left(\frac{100\ cm}{m}\right)^2\left(\frac{1\ h}{3600\ s}\right)\left(\frac{1\ W}{1\ J/s}\right)\left(\frac{0.3171\frac{Btu}{h-ft^2}}{1\ W/m^2}\right)\left(\frac{1\ ft^2}{144\ in^2}\right)$$

# 5 SHIELDING EVALUATION

This chapter outlines the Model 2000 cask shielding analysis and demonstrates compliance with the external radiation requirements of 10 CFR 71, "Packaging and Transportation of Radioactive Material" (Reference 5-1). This shielding evaluation was performed to demonstrate that the Model 2000 Transport Package with the high performance insert (HPI) provides sufficient shielding such that the external radiation limits are satisfied under Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC).

## 5.1 Description of Shielding Design

### 5.1.1. Design Features

The Model 2000 cask is a cylindrical lead lined cask used for transporting Type B quantities of radioactive materials and solid fissile materials. For any shipments of radioactive material in the Model 2000 cask, the use of the HPI is required, and all contents must be confined inside the HPI cavity. The radiation shielding design features of the Model 2000 with the HPI are the lead and Stainless Steel (SS) in the Model 2000 cask and the depleted uranium (DU) and SS in the HPI. Narrative descriptions of the HPI, Model 2000 cask, and Model 2000 overpack are provided in Section 1.2. The radiation shielding design features of the Model 2000 with the HPI are provided in Table 5.1-1, including nominal dimensions, materials of construction, and densities of the materials that provide gamma shielding.

**Table 5.1-1. Model 2000 Transport Package Shielding Design Features**

| Model 2000 Component | Part | Component | Thickness (in) | Thickness (cm) | Material of Construction | Material Density (lb/in$^3$) | Material Density (g/cm$^3$) |
|---|---|---|---|---|---|---|---|
| HPI | Top Plug | Inner Shell | [[ | | [[ ]] | 0.29 | 8.000 |
| | | DU | | | DU | [[ | ]] |
| | | Outer Shell | | | [[ ]] | 0.29 | 8.000 |
| | HPI Body | Inner Shell | | | [[ ]] | 0.29 | 8.000 |
| | | DU | | | DU | [[ | ]] |
| | | Outer Shell | | | [[ | 0.29 | 8.000 |
| | Bottom Plug | Inner Shell | | | ]] | 0.29 | 8.000 |
| | | DU | | | DU | [[ | ]] |
| | | Outer Shell | | ]] | [[ ]] | 0.29 | 8.000 |
| Cask | Cask Lid | Lid Flange | 1.75 | 4.445 | SS304 | 0.29 | 8.000 |
| | | Lead | 5.37 | 13.64 | Lead | 0.41 | 11.34 |
| | | Inner Plate | 1.50 | 3.810 | SS304 | 0.29 | 8.000 |
| | Cask Body (Side) | Cavity Shell | 1.00 | 2.540 | SS304 | 0.29 | 8.000 |
| | | Lead | 4.00 | 10.16 | Lead | 0.41 | 11.34 |
| | | Cask Shell | 1.00 | 2.540 | SS304 | 0.29 | 8.000 |

| Model 2000 Component | Part | Component | Thickness (in) | Thickness (cm) | Material of Construction | Material Density ($lb/in^3$) | Material Density ($g/cm^3$) |
|---|---|---|---|---|---|---|---|
| | Cask Body (Bottom) | Cask Bottom[a] | 5.88 | 14.92 | SS304 | 0.29 | 8.000 |
| Overpack[b] | Overpack (Top) | Top Plate | 0.50 | 1.270 | SS304 | 0.29 | 8.000 |
| | | End Plate | 0.50 | 1.270 | SS304 | 0.29 | 8.000 |
| | Overpack (Side) | Inner Shell | 0.50 | 1.270 | SS304 | 0.29 | 8.000 |
| | | Outer Shell | 0.50 | 1.270 | SS304 | 0.29 | 8.000 |
| | Overpack (Bottom) | Support Plate | 0.50 | 1.270 | SS304 | 0.29 | 8.000 |
| | | Bottom Plate | 0.50 | 1.270 | SS304 | 0.29 | 8.000 |
| | | End Plate | 0.50 | 1.270 | SS304 | 0.29 | 8.000 |

Notes: [a] Due [[ ]], the minimum thickness is used.
[b] Credit for shielding provided by the cask overpack is only taken for NCT analyses.
General: All dimensions are based on component licensing drawings in Section 1.3.1.

### 5.1.2. Summary Table of Maximum Radiation Levels

Table 5.1-2 and Table 5.1-3 present the maximum calculated NCT and HAC dose rates at the appropriate locations for exclusive use shipment of the Model 2000 Transport Package with the HPI. The calculated NCT and HAC dose rates are reported for each of the three content types described in Section 5.2, as well as the overall maximum dose rates from all contents. The 1-meter transportation index dose rate limits are not applicable as the Model 2000 cask will only be shipped as exclusive use. The Model 2000 cask will only be shipped in the upright position, thus the 2-meter and occupied position (cab) dose rates are calculated at the appropriate distances from the side of the cask. Dose rates are limited to 90% of the regulatory limit at each location to provide additional assurance that any small uncertainties in the source term or cask modeling will not result in external dose rates exceeding the respective regulatory limit.

**Table 5.1-2. Maximum NCT Dose Rates**

| Contents | Radiation | Package Surface mSv/hr (mrem/hr) | | | 2-meter mSv/hr (mrem/hr) | Cab mSv/hr (mrem/hr) |
|---|---|---|---|---|---|---|
| | | Top | Side | Bottom | Side | Side |
| 1 | Gamma + Neutron | 0.3141 (31.41) | 1.8000 (180.00) | 1.0677 (106.77) | 0.0434 (4.34) | 0.0078 (0.78) |
| 2 | Gamma | 0.1026 (10.26) | 1.7999 (179.99) | 0.1651 (16.51) | 0.0251 (2.51) | 0.0044 (0.44) |
| 3 | Gamma | 0.1799 (17.99) | 0.8578 (85.78) | 0.3832 (38.32) | 0.0162 (1.62) | 0.0028 (0.28) |
| Overall Maximum | | 0.3141 (31.41) | 1.8000 (180.00) | 1.0677 (106.77) | 0.0434 (4.34) | 0.0078 (0.78) |
| 10 CFR 71.47(b) Limits | | 2 (200) | 2 (200) | 2 (200) | 0.1 (10) | 0.02 (2) |

Contents:
1 – Irradiated fuel
2 – Irradiated hardware and byproducts
3 – Cobalt-60 isotope rods

**Table 5.1-3. Maximum HAC Dose Rates**

| Contents | HAC Radiation | 1 Meter from Package Surface mSv/hr (mrem/hr) Top | Side | Bottom |
|---|---|---|---|---|
| 1 | Gamma + Neutron | 0.3342 (33.42) | 0.5363 (53.63) | 0.3112 (31.12) |
| 2 | Gamma | 0.1335 (13.35) | 0.3421 (34.21) | 0.0841 (8.41) |
| 3 | Gamma | 0.5843 (58.43) | 1.6951 (169.51) | 0.3454 (34.54) |
| Overall Maximum | | 0.5843 (58.43) | 1.6951 (169.51) | 0.3454(34.54) |
| 10 CFR 71.51(a)(2) Limit | | 10 (1000) | 10 (1000) | 10 (1000) |

Contents:
1 – Irradiated fuel
2 – Irradiated hardware and byproducts
3 – Cobalt-60 isotope rods

## 5.2 Source Specification

The allowable contents for the Model 2000 cask are: 1) irradiated fuel, 2) irradiated hardware and byproducts and 3) cobalt-60 isotope rods. The irradiated fuel contents have photon and neutron source terms for determining package external dose rates. The irradiated hardware and byproduct and cobalt-60 isotope rod contents have photon source terms for determining package external dose rates. Due to the thick layers of shielding provided by the HPI and Model 2000 cask, external dose rate contributions from charged particles (alpha and beta particles) and their secondary particles from interactions (e.g., bremsstrahlung) are negligible. The exception to the above statement is that the neutron source from alpha-n reactions in the irradiated fuel contents is considered, as explained in Section 5.5.1.

### Irradiated Fuel

The irradiated fuel content is GE BWR 10x10 fuel, which is segmented and placed into the HPI. The required parameters for the irradiated fuel that are relevant to the shielding analysis include:

1. Cooling time: Minimum of at least 120 days.
2. Length: Minimum active fuel length of at least 5.3 inches for each segment.
3. Arrangement: Confined and placed into the HPI material basket in the upright position with or without additional shoring component that ensures the fuel remain upright.
4. Initial enrichment U-235: Minimum of 1.5 wt%, maximum of 6 wt%.
5. Fuel exposure: Maximum of 72 GWd/MTU.
6. If the irradiated fuel is encapsulated (e.g., in cladding), the encapsulation material is treated as irradiated hardware and must be composed of an approved irradiated hardware material as described in the next paragraphs.

**Irradiated Hardware and Byproducts**

The irradiated hardware and byproduct contents are irradiated components from typical reactor operation. These contents include:

1. Hardware: Irradiated metals composed of materials such as SS, carbon steels, nickel alloys, and zirconium alloys. Examples include:
   - Bundle components: fuel cladding, water rods, spacers, and upper/lower tie plates
   - Reactor internals: jet pump components, core shroud samples
2. Irradiated Byproducts: Irradiated control rod blades with the following neutron poison materials:
   - Hafnium
   - Boron Carbide

**Cobalt-60 Isotope Rods**

The radioactive material in the cobalt-60 isotope rod contents is in the form of pellets or cylindrical solid rods with the source(s) evenly distributed and encapsulated in normal or special form. The isotope rods are loaded into a commercial or research reactor to irradiate the cobalt source pellets. After discharge from the reactor, the isotope rods are loaded into the Model 2000 cask for transport. These [[ ]] prior to loading into the HPI. Herein for the cobalt-60 isotope rod contents, the term 'rod' refers to a full-length rod, in its form as it is irradiated in a reactor; and the term [[ ]] in its form as it is loaded and shipped in the Model 2000 Transport Package.

## 5.2.1. Gamma Source

### 5.2.1.1. Irradiated Fuel

To calculate gamma source strengths, ORIGEN-ARP is used, which implements the ORIGEN-S module with the GE BWR 10x10 cross section library (ge10x10-8) distributed in the SCALE6.1 code package (Reference 5-2). With the ORIGEN-ARP methodology, a problem dependent cross section library is generated by interpolating between cross sections in the SCALE6.1 pre-generated libraries. The pre-generated GE BWR 10x10 library covers initial uranium enrichments from 1.5 to 6 wt%, with burnups from 0 to 72 GWd/MTU, and moderator densities from 0.1 to 0.9 g/cm$^3$. Any mention of enrichment refers to the initial U-235 enrichment of the fuel. ORIGEN-ARP has been validated extensively for light water reactor spent fuel, as documented in the Oak Ridge National Lab report ORNL/TM-13584 (Reference 5-9).

[[

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The dose rate contribution from a given fuel segment at a regulatory dose rate location is calculated by multiplying the burnup- and enrichment-dependent dose rate per gram of initial U-235 by the initial U-235 mass of that fuel segment. The total dose rate from a payload of irradiated fuel is calculated by summing the dose rate contributions from each fuel segment included in the shipment. The burnup- and enrichment-dependent dose rate per gram of initial U-235 includes contributions from both gamma rays and neutrons. Details of the parameters used for the ORIGEN-S neutron and photon irradiated fuel source term calculations are provided in Section 5.5.1.

#### 5.2.1.2. Irradiated Hardware and Byproducts

For the irradiated hardware and byproduct contents, the gamma source strength and spectra are based on the individual radionuclides in a given shipment. Multiple ORIGEN-S irradiation calculations were used to identify the radionuclides that could be in a shipment of irradiated hardware and byproduct. Table 5.2-1 provides a list of radionuclides that may be present in irradiated hardware and byproduct contents that contribute to external dose rates. Other radionuclides which may be present in irradiated hardware and byproducts but do not emit significant gammas were excluded from Table 5.2-1. However, all radionuclides that may be present in irradiated hardware and byproducts are considered when determining the total decay heat of the payload as described in Section 5.5.4.

External dose rates are calculated individually for 1 Ci of activity with the energy spectrum from each of the listed radionuclides. The energy spectrum for each radionuclide is from the ORIGEN-S Data Library origen.rev04.mpdkxgam.data (Reference 5-2). The dose rate contribution from a specific radionuclide at a regulatory dose rate location is calculated by multiplying the total activity for the radionuclide by its respective dose rate per curie multiplier. The total dose rate from a payload of irradiated hardware and byproduct is calculated by summing the dose rate contributions from each radionuclide included in the shipment. Details of the ORIGEN-S irradiated hardware and byproduct source term calculations and the energy spectra for each radionuclide of interest are provided in Section 5.5.2.

**Table 5.2-1. Irradiated Hardware and Byproduct Radionuclides Significant to External Dose Rates**

| Radionuclides |
|---|
| Sc-46 |
| Cr-51 |
| Mn-54 |
| Co-58 |
| Fe-59 |
| Co-60 |
| Zn-65 |
| Nb-92m |
| Nb-94 |
| Zr/Nb-95 |
| Sb-124 |
| Sb-125 |
| Sb-126 |
| Cs-134 |
| Cs-137 (Ba-137m) |
| Hf-175 |
| Hf-181 |
| Ta-182 |

#### 5.2.1.3. Cobalt-60 Isotope Rods

The primary gamma source in the cobalt-60 isotope rod content is from the cobalt-60 source pellets. Dose rate contributions from the small quantities of radionuclides in crud that has built up on the rods while in the reactor is negligible due to insignificant gammas emitted. The cask external dose rates are dominated by the quantity of cobalt-60 in the isotope rods, and any dose rate contributions from any radionuclides in the rod cladding can be accounted for as irradiated hardware (see Section 5.4.4.4 for further explanation). Table 5.2-2 provides the energy spectrum and gamma source strength for cobalt-60 used for dose rate calculations. The energy spectrum is from the ORIGEN-S data library origen.rev04.mpdkxgam.data (Reference 5-2). All energy lines less than 0.1 MeV are considered negligible and are neglected from the energy spectrum. The source strength is based on the cobalt-60 activity equivalent to the thermal limit of 1500 W. The watt/curie (W/Ci) conversion factor is based on the ORIGEN-S decay library origen.rev03.decay.data (Reference 5-2). The values from this library and the calculation of a W/Ci conversion factor for multiple radionuclides are presented in Section 5.5.3. Using the W/Ci conversion factor presented in Section 5.5.4, the equivalent activity for 1500 W is 97,250 Ci of cobalt-60.

**Table 5.2-2. Isotope Rod Source Term (97,250 Ci Cobalt-60)**

| Energy (MeV) | Relative Intensity | Source Strength (γ/sec) |
|---|---|---|
| 0.347 | 7.500E-05 | 2.699E+11 |
| 0.826 | 7.600E-05 | 2.735E+11 |
| 1.173 | 9.985E-01 | 3.593E+15 |
| 1.333 | 9.998E-01 | 3.598E+15 |
| 2.159 | 1.200E-05 | 4.318E+10 |
| 2.506 | 2.000E-08 | 7.197E+07 |
| **Total** | 1.998E+00 | 7.191E+15 |

### 5.2.2. Neutron Source

#### 5.2.2.1. Irradiated Fuel

The neutron source strengths for the irradiated fuel contents are calculated with the same method as the gamma source term. The ORIGEN-S source term calculations detailed in Section 5.5.1 generate both the gamma and neutron source terms for the irradiated fuel contents.

In Section 5.2.1.1 it was described how the neutron contribution is already included in the total dose rate per gram of initial U-235.

#### 5.2.2.2. Irradiated Hardware and Byproducts / Cobalt-60 Isotope Rods

There is no applicable neutron source term for the irradiated hardware and byproduct or cobalt-60 isotope rod contents.

## 5.3 Shielding Model

### 5.3.1. Configuration of Source and Shielding

The following subsections describe the shielding model geometry and source configuration for the dose rate calculations of each of the described content types of the Model 2000 cask.

#### 5.3.1.1. Source Distribution

An individual source geometry is used in the shielding model for each of the Model 2000 cask contents. The source geometry for each content type is based on the respective content specifications and the source term calculation.

**Irradiated Fuel**

For the segmented irradiated fuel content, the NCT source geometry is a single 5.3-inch line source across which the photon and neutron sources are distributed uniformly. The irradiated fuel source term specification requires that the active fuel length when loaded into the Model 2000 Transport Package must be greater than or equal to 5.3 inches. Using the minimum allowable segment length for the line source ensures a bounding dose rate calculation, as greater distribution of the source activity (e.g., a longer line source) results in lower calculated maximum external dose rates.

The axial distribution of activity along the irradiated fuel is a function of different initial U-235 enrichment axially and variations in moderator density during irradiation. The bounding gamma and neutron source strength considered for any given segment is based on the minimum enrichment and maximum burnup in the segment. During the irradiation of the fuel, lower moderator densities results in higher source strengths (Reference 5-10). By calculating all gamma and neutron source strengths at the minimum moderator density (0.1 g/cm$^3$) available in the library, the calculated source strengths are bounding for any expected changes in axial moderator density. Thus, a uniform line source is acceptable despite variations in the irradiated fuel activity profile because the source term calculation results in bounding gamma and neutron source strengths.

For the HAC shielding model source geometry, it is conservative to assume that all activity is concentrated into a single point. The source locations for the NCT model line source and the HAC model point source are in the locations shown in Figure 5.3-1.

**Irradiated Hardware and Byproducts**

Due to the uncertainty in the form and activity distribution of irradiated hardware or byproduct contents, both the NCT and HAC shielding models conservatively assume that all the activity is concentrated into a single point. Therefore, the use of the HPI material basket is not required for irradiated hardware and byproduct shipments. However, use of the HPI material basket for shipments of irradiated hardware and byproducts is optionally allowable because the material basket 12-inch line source is bounded by the shielding results obtained from the point source model, as long as all dose rates and thermal limits are satisfied. The source locations of the point sources in the shielding models for the irradiated hardware and byproduct dose rate calculations are shown in Figure 5.3-1.

**Cobalt-60 Isotope Rods**

For the cobalt-60 isotope rod content, the NCT source geometry is a single 12-inch line source, across which the photon source activity is distributed uniformly. There is variation in the distribution of cobalt-60 activity in the HPI cavity with a shipment of cobalt-60 isotope [[ ]] and loading of the rods into the HPI. Section 5.5.3 provides a discussion of the distribution of activity in the HPI cavity for the cobalt-60 isotope rod contents, and the basis for a 12-inch line source for NCT dose rate calculations.

For the HAC shielding model source geometry, the structural components in the cask cavity were conservatively assumed to fail. Therefore, all source activity was concentrated into a single point. The source locations for the NCT model 12-inch line source and the HAC model point source are shown in Figure 5.3-1. The 12-inch line source modeling limitation imposed during the NCT evaluations requires that the HPI material basket be present for cobalt-60 isotope rod shipments.

### 5.3.1.2. Source Locations

The sources for the dose rate calculations are modeled in the HPI cavity in the position that results in the highest dose rate for the respective regulatory dose rate location. This limiting source position changes based on the geometry of the source and the direction of interest. Figure 5.3-1 provides two depictions of the Model 2000 cask with the HPI. This figure shows the positions for any point or line sources in the HPI cavity for all dose rate calculations.

The source positions for side dose rate locations are located at the bottom corner of the HPI cavity, at the interface of the HPI body and the HPI bottom plug. This is the most restrictive location for side dose rates because in this area the [[ ]], due to the step at this interface. For the HAC side 1-meter dose rate, the calculated dose rate is higher with a point source in the bottom corner than in the top corner of the HPI cavity, despite the slump in the lead column.

The line source positions for top and bottom dose rate locations are centered in the HPI cavity so that particles emitted at any location along the line source can travel at any angle in the direction of interest, unimpeded before entering the respective plug (top or bottom). For a line source pushed to the side against the HPI body, there is a reduction in the calculated dose rates for the top and bottom.

[[

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**Figure 5.3-1. Point / Line Source Locations**

#### 5.3.1.3. NCT Shielding Model Geometry

The NCT model geometry used for the dose rate calculations in this shielding analysis is a detailed three-dimensional model of the HPI, the Model 2000 cask, and the overpack. Table 5.3-1 provides the relevant dimensions of the shielding model including the modeled thicknesses of each material. This table along with Table 5.1-1 allow for a quick review of the most significant dimensions of the shielding model geometry. All HPI shield dimensions are at the minimum (except cavity radius which is nominal), per the respective licensing drawings, with the fabrication tolerances subtracted from the nominal values. The model dimensions for the Model 2000 cask and overpack use predominantly nominal dimensions with some areas of reduced thickness. For example, for the [[ ]], the cask bottom is considered to be flat at the minimum thickness. The majority of the material thicknesses prescribed by the Model 2000 cask and overpack licensing drawings have tolerances based on American Society for Testing and Materials (ASTM) specifications, as the component dimensions are based on ASTM SS stock plate. Per ASTM A480 (Reference 5-3), for plates up to 10 inches in thickness, the tolerance under the specified thickness is 0.01 inches. These plate thicknesses are modeled at the specified nominal plate value.

**Table 5.3-1. Relevant Shielding Model Dimensions**

| Model 2000 Component | Part | Parameter | Dimension (cm) | Dimension (in) |
|---|---|---|---|---|
| Cask | Cask Lid | Lid Flange ($t_{SS1}$) | 3.810 | 1.500 |
| | | Lead ($t_{Pb}$) | 13.64 | 5.370 |
| | | Inner Plate ($t_{SS2}$) | 4.445 | 1.750 |
| | Cask Side | Cavity Radius ($r_{cavity}$) | 33.66 | 13.25 |
| | | Cavity Shell | 2.540 | 1.000 |
| | | Lead ($t_{Pb}$) | 10.16 | 4.000 |
| | | Cask Shell ($t_{SS2}$) | 2.540 | 1.000 |
| | | Lead ($h_{pb}$[c]) | 141.9 | 55.87 |
| | Cask Bottom | Cask Bottom ($t_{SS}$) | 14.94[a] | 5.880[a] |
| | | Cavity Height ($h_{cavity}$) | 137.5 | 54.13 |
| HPI | HPI Top Plug | Inner Shell ($t_{SS1}$) | [[ | |
| | | DU ($t_{DU}$) | | |
| | | Outer Shell ($t_{SS2}$) | | |
| | HPI Body Side | Cavity Radius ($r_{cavity}$) | | |
| | | Inner Shell ($t_{SS1}$) | | |
| | | DU ($t_{DU}$) | | |
| | | Outer Shell ($t_{SS2}$) | | |
| | HPI Bottom Plug | Inner Shell ($t_{SS1}$) | | |
| | | DU ($t_{DU}$) | | |
| | | Outer Shell ($t_{SS2}$) | | ]] |
| Overpack | Top | Top Plate ($t_{SS1}$) | 1.270 | 0.500 |
| | | End Plate ($t_{SS2}$) | 1.270 | 0.500 |
| | Side | Inner Shell ($t_{SS1}$) | 1.270 | 0.500 |
| | | Outer Shell ($t_{SS2}$) | 1.270 | 0.500 |
| | Bottom | Support Plate ($t_{SS1}$) | 1.270 | 0.500 |
| | | Bottom Plate ($t_{SS2}$) | 1.270 | 0.500 |
| | | End Plate ($t_{SS3}$) | 1.270 | 0.500 |

Notes: [a] Cask Bottom modeled flat, with thickness equal to the 6.13” height [[ ]]
[b] Minimum DU thicknesses considered with tolerance gaps explicitly modeled
[c] Lead column height
[d] Nominal value

There are two different NCT shielding models. One is for photon dose rate calculations and the other is for neutron dose rate calculations. For both models the geometry is the same. However, for the photon dose rate model the materials for the HPI, cask, and overpack are defined as prescribed in Section 5.3.2. For the neutron dose rate NCT model, all shielding provided by the materials of the packaging are neglected. Taking no credit for shielding of neutrons provided by the HPI, the Model 2000 cask, and the Model 2000 overpack results in bounding calculated dose rates. The dose rates calculated through void bound the dose rates calculated by crediting the shielding provided by all cask components, any additional neutrons from subcritical multiplication, and the additional (secondary) photons from neutron interactions in the cask. The NCT shielding models are shown in Figure 5.3-2. The materials for the HPI, cask, and overpack are defined as prescribed in Section 5.3.2.

[[

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**Figure 5.3-2.  NCT Shielding Models**

The photon NCT model conservatively neglects the additional shielding provided by the HPI material basket and rod holders for irradiated fuel and the cobalt-60 isotope rod contents.  Due to the use of vertical line sources, these contents must be shipped in the upright position. The HPI material basket may be used to position these contents in the upright position. The material basket is not required for shipments of irradiated hardware and byproducts because a point source was used for the shielding analysis.

#### 5.3.1.4. HAC Shielding Model Geometry

For HAC, the shielding model only includes the HPI and the Model 2000 cask, with dimensions as prescribed in Table 5.3-1. This model conservatively assumes the removal of the overpack. The HAC model also includes the slump in the lead column of the Model 2000 cask body. In Section 2.12.2, the maximum deformation in the lead column is calculated to be 3.56 mm. This value is rounded up to 4 mm for this analysis. It is determined in Chapter 2 that the overpack provides adequate protection from HAC to the cask body. More specifically, in Section 2.12.1 it is stated that when the cask is dropped 30 feet followed by a drop of 40 inches onto a rigid pin 6 inches in diameter, no gross deformations of the cask are predicted. As with the NCT models, there are two different HAC shielding models, as show in Figure 5.3-3. These models have the same geometry but the photon model includes the materials of the HPI and the Model 2000 cask, and the neutron model neglects all materials.

[[

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**Figure 5.3-3. HAC Shielding Models**

#### 5.3.1.5. MCNP6 Tallies

To calculate the particle flux at the regulatory dose rate locations of interest, multiple arrangements of cell tallies are modeled at each location. The void cells that are added to the model for particle tallying allow for dose rates to be calculated at the multiple locations of interest, without having an effect on the calculated flux. All of the tally cells are modeled as small 1 cm thick volumes, to ensure that the calculated flux is not averaged over too large of a region.

Figures 5.3-4 and 5.3-5 provide depictions of the tally cells used in the MCNP6 shielding models, with the tally cells highlighted in yellow.

[[

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**Figure 5.3-4. NCT MCNP6 Tallies with 10% Margin to the Regulatory Limit**

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**Figure 5.3-5. HAC MCNP6 Tallies with 10% Margin to the Regulatory Limit**

### 5.3.2. Material Properties

The material compositions used for photon dose rate calculations are listed in Tables 5.3-2 through 5.3-5. There is negligible difference between the two types of SS in terms of shielding effectiveness. However, both types are included for accuracy to the actual materials of construction. The densities and material compositions for both stainless steel types are from Pacific Northwest National Lab (PNNL) report PNNL-15870 Revision 1 (Reference 5-4). The densities of the lead and DU materials are based on the minimum specified densities for these materials in the respective component licensing drawings in Section 1.3.1. All materials are modeled as void for neutron dose rate calculations, so the isotopic composition of materials is not required.

**Table 5.3-2. Type 304 Stainless Steel Material Composition**

| | Element | ZAID | Mass Fraction |
|---|---|---|---|
| **Elemental Composition** | C | 6000 | 4.00E-04 |
| | Si | 14000 | 5.00E-03 |
| | P | 15000 | 2.30E-04 |
| | S | 16000 | 1.50E-04 |
| | Cr | 24000 | 1.90E-01 |
| | Mn | 25000 | 1.00E-02 |
| | Fe | 26000 | 7.02E-01 |
| | Ni | 28000 | 9.25E-02 |
| **Density ($g/cm^3$)** | 8.0 | | |

**Table 5.3-3. [[ ]] Material Composition**

| | Element | ZAID | Mass Fraction |
|---|---|---|---|
| **Elemental Composition** | C | 6000 | 4.10E-04 |
| | Si | 14000 | 5.07E-03 |
| | P | 15000 | 2.30E-04 |
| | S | 16000 | 1.50E-04 |
| | Cr | 24000 | 1.70E-01 |
| | Mn | 25000 | 1.01E-02 |
| | Fe | 26000 | 6.69E-01 |
| | Ni | 28000 | 1.20E-01 |
| | Mo | 42000 | 2.50E-02 |
| **Density ($g/cm^3$)** | 8.0 | | |

**Table 5.3-4. Lead Material Composition**

| | Element | ZAID | Mass Fraction |
|---|---|---|---|
| **Elemental Composition** | Pb | 82000 | 1.00E+00 |
| **Density ($g/cm^3$)** | 11.34 | | |

**Table 5.3-5. Depleted Uranium Material Composition**

| | Element | ZAID | Mass Fraction |
|---|---|---|---|
| **Elemental Composition** | U | 92000 | 1.00E+00 |
| **Density ($g/cm^3$)** | [[ ]] | | |

## 5.4 Shielding Evaluation

### 5.4.1. Methods

#### 5.4.1.1. Computer Codes

The shielding calculations for this analysis were completed using MCNP6 Version 1.0 (Reference 5-5) for the irradiated hardware and byproducts and cobalt-60 isotope rod contents, and MCNP6 Version 2.0 (Reference 5-11) for the irradiated fuel content. MCNP6 is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code. MCNP6 was used in the photon only transport mode to calculate external dose rates for the Model 2000 cask for each of the content types considered. Photon dose rate calculations used the MCNP6 photoatomic data library MCPLIB84, which compiles data from the ENDF/B-VI.8 data library (Reference 5-6).

Since the neutron models are voided and unshielded, MCNP6 is not needed. The neutron flux around a point or line source in a vacuum can be calculated analytically.

#### 5.4.1.2. MCNP6 Variance Reduction

Due to the thick layers of photon shielding provided by the Model 2000 cask and the HPI, multiple variance reduction techniques are used for the MCNP6 photon dose rate calculations. MCNP6 variance reduction parameters for weight windows, exponential transform, and source biasing were used as necessary to aid in the statistical convergence of the photon dose rate calculations.

#### 5.4.1.3. Irradiated Fuel Dose Rate Calculation

The flux, $\phi(\mathrm{r,p,e})$, at regulatory dose rate location of interest, r, is calculated for a photon or neutron particle, p. [[

]] The photon flux is calculated in MCNP6 and the neutron flux is calculated analytically using Equation 5-1 for a point source and Equation 5-2 for a line source. [[
]]

$$\phi(\mathrm{r,p,e}) = \frac{1}{4\pi \mathrm{r}^2} \tag{5-1}$$

$$\phi(\mathrm{r,p,e}) = \frac{1}{4\pi \mathrm{x}(l_1 + l_2)}\left[\tan^{-1}\frac{l_1}{\mathrm{x}} + \tan^{-1}\frac{l_2}{\mathrm{x}}\right] \tag{5-2}$$

By applying the appropriate flux-to-dose-rate conversion factors (Section 5.4.3), $\mathcal{R}(\mathrm{p,e})$[[
]] the dose rate response, $\mathrm{R(r,p,e)}$, and associated standard deviation, $\sigma_{\mathrm{R}}(\mathrm{r,p,e})$, are calculated following Equations 5-3 and 5-4.

$$\mathrm{R(r,p,e)}\left[\frac{\mathrm{mrem}}{\mathrm{hr}}\cdot\frac{\mathrm{sec}}{\mathrm{emitted\ p}}\right] = \phi(\mathrm{r,p,e})\left[\frac{\frac{\mathrm{p}}{\mathrm{cm^2}}}{\mathrm{emitted\ p}}\right]\cdot\mathcal{R}(\mathrm{p,e})\left[\frac{\frac{\mathrm{mrem}}{\mathrm{hr}}}{\frac{\mathrm{p}}{\mathrm{cm^2\cdot sec}}}\right] \tag{5-3}$$

$$\sigma_{\mathrm{R}}(\mathrm{r,p,e}) = \mathrm{R(r,p,e)}\cdot \mathrm{fsd(r,p,e)} \tag{5-4}$$

The quantity $R_\sigma(r,p,e)$ accounts for the statistical uncertainty. Two standard deviations are added to the calculated dose rate response in Equation 5-5. Note that there is no statistical uncertainty associated with the neutron flux calculation since it was calculated analytically ($\sigma_R = 0$).

$$R_\sigma(r,p,e)\left[\frac{\text{mrem}}{\text{hr}}\cdot\frac{\text{sec}}{\text{emitted p}}\right] = \left(R(r,p,e) + 2\cdot\sigma_R(r,p,e)\right)\left[\frac{\text{mrem}}{\text{hr}}\cdot\frac{\text{sec}}{\text{emitted p}}\right] \tag{5-5}$$

Equation 5-6 shows that $\ddot{DR}(r,B|E)$ [[

]] is calculated per gram of U-235.

[[

]]

The total dose rate, $DR(r,B|E)$, at a given burnup and enrichment is determined by multiplying Equation 5-6 with the mass of U-235 in the irradiated fuel content at the respective burnup and enrichment, $m(B|E)$, as shown in Equation 5-7.

$$DR(r,B|E)\left[\frac{\text{mrem}}{\text{hr}}\right] = \ddot{DR}(r,B|E)\left[\frac{\frac{\text{mrem}}{\text{hr}}}{\text{gU235}}\right]\cdot m(B|E)[\text{gU235}] \tag{5-7}$$

where

| | | | |
|---|---|---|---|
| $\phi$ | Calculated flux (MCNP6 or analytical) | r | Regulatory dose rate location/distance |
| p | Particle (neutron or gamma) | [[ | ]] |
| x | Perpendicular distance of r from line source | $l_1, l_2$ | Partial length of line source on either side of perpendicular line x |
| R | Calculated dose rate response | $\sigma$ | Standard deviation |
| $\mathcal{R}$ | Flux-to-dose-rate conversion factor | $R_\sigma$ | Dose rate response with 2σ uncertainty |
| fsd | MCNP6 fractional standard deviation | S | Calculated source strength (ORIGEN-S) |
| $\ddot{DR}$ | Dose rate per gram U-235 | DR | Total dose rate |
| B\|E | Burnup/Enrichment pairing | | |
| m | Mass of U-235 | | |

#### 5.4.1.4. Irradiated Hardware, Byproduct, and Cobalt-60 Isotope Rod Dose Rate Calculation

To calculate a dose rate response $R(r, X)$ for an individual radionuclide X, the MCNP6 calculated photon flux $\phi(r, X)$ is multiplied by the dose rate conversion factor $\mathcal{R}$ as well as a per-curie multiplier and the total number of gammas per decay of the respective radionuclide $I(X)$.

$$R(r,X)\left[\frac{\frac{\text{mrem}}{\text{hr}}}{\text{Ci}}\right] = \phi(r,X)\left[\frac{\frac{\gamma}{\text{cm}^2}}{\text{emitted }\gamma}\right]\cdot 3.7\text{e}10\left[\frac{\frac{\text{decays}}{\text{sec}}}{\text{Ci}}\right]\cdot I(X)\left[\frac{\text{emitted }\gamma}{\text{decay}}\right]\cdot \mathcal{R}\left[\frac{\frac{\text{mrem}}{\text{hr}}}{\frac{\gamma}{\text{cm}^2\cdot\text{sec}}}\right] \tag{5-8}$$

$$\sigma_R(r,X) = R(r,X)\cdot \text{fsd}(r,X) \tag{5-9}$$

To account for statistical uncertainty, the two standard deviations are added to the calculated MCNP6 dose rate per curie:

$$R_\sigma(r,X)\left[\frac{\text{mrem}}{\text{hr}}\right] = \left(R(r,X) + 2\cdot\sigma_R(r,X)\right)\left[\frac{\frac{\text{mrem}}{\text{hr}}}{\text{Ci}}\right] \tag{5-10}$$

The total dose rate $DR(r)$ is calculated by summing the dose rate from the activity of each radionuclide:

$$DR(r)\left[\frac{\text{mrem}}{\text{hr}}\right] = \sum_X R_\sigma(r,X)\left[\frac{\text{mrem}}{\text{hr}}\right]\cdot A(X)[\text{Ci}] \tag{5-11}$$

where

| | | | |
|---|---|---|---|
| R | MCNP6 dose rate per curie | σ | Standard deviation |
| r | Regulatory dose rate location | fsd | MCNP6 fractional standard deviation |
| X | Radionuclide X | $R_\sigma$ | Dose rate per curie with 2σ uncertainty |
| ϕ | MCNP6 calculated flux | DR | Total dose rate |
| I | gammas/decay | A | Activity |
| $\mathcal{R}$ | Flux-to-dose-rate conversion factor | | |

### 5.4.2. Input and Output Data

#### 5.4.2.1. Input Data

Input data will be submitted separately.

#### 5.4.2.2. Output Data

Output data will be submitted separately. The tally fluctuation chart and probability density function plot were studied for each MCNP6 tally to ensure proper tally bin convergence. This along with a check of the reported fsd for each tally bin and the additional statistical information reported for MCNP6 tallies ensured the reliability of all MCNP6 calculated dose rate results.

### 5.4.3. Flux-to-Dose-Rate Conversion

Consistent with NUREG-1609 Section 5.5.4.3 (Reference 5-7), the ANSI/ANS-6.1.1 1977 flux-to-dose-rate conversion factors (Reference 5-8) are used. The gamma and neutron conversion factors used are tabulated in Tables 5.4-1 and 5.4-2, respectively.

**Table 5.4-1. Gamma Flux-to-Dose-Rate Conversion Factors (ANSI/ANS-6.1.1 1977)**

| Gamma Energy (MeV) | Conversion Factor (mrem/hr)/(gammas/cm$^2$-s) |
|---|---|
| 1.00E-02 | 3.96E-03 |
| 3.00E-02 | 5.82E-04 |
| 5.00E-02 | 2.90E-04 |
| 7.00E-02 | 2.58E-04 |
| 1.00E-01 | 2.83E-04 |
| 1.50E-01 | 3.79E-04 |
| 2.00E-01 | 5.01E-04 |
| 2.50E-01 | 6.31E-04 |
| 3.00E-01 | 7.59E-04 |
| 3.50E-01 | 8.78E-04 |
| 4.00E-01 | 9.85E-04 |
| 4.50E-01 | 1.08E-03 |
| 5.00E-01 | 1.17E-03 |
| 5.50E-01 | 1.27E-03 |
| 6.00E-01 | 1.36E-03 |
| 6.50E-01 | 1.44E-03 |
| 7.00E-01 | 1.52E-03 |
| 8.00E-01 | 1.68E-03 |
| 1.00E+00 | 1.98E-03 |
| 1.40E+00 | 2.51E-03 |
| 1.80E+00 | 2.99E-03 |
| 2.20E+00 | 3.42E-03 |
| 2.60E+00 | 3.82E-03 |
| 2.80E+00 | 4.01E-03 |
| 3.25E+00 | 4.41E-03 |
| 3.75E+00 | 4.83E-03 |
| 4.25E+00 | 5.23E-03 |
| 4.75E+00 | 5.60E-03 |
| 5.00E+00 | 5.80E-03 |
| 5.25E+00 | 6.01E-03 |
| 5.75E+00 | 6.37E-03 |
| 6.25E+00 | 6.74E-03 |
| 6.75E+00 | 7.11E-03 |
| 7.50E+00 | 7.66E-03 |
| 9.00E+00 | 8.77E-03 |
| 1.10E+01 | 1.03E-02 |
| 1.30E+01 | 1.18E-02 |
| 1.50E+01 | 1.33E-02 |

**Table 5.4-2. Neutron Flux-to-Dose-Rate Conversion Factors (ANSI/ANS-6.1.1 1977)**

| Neutron Energy (MeV) | Conversion Factor (mrem/hr)/(neutrons/cm$^2$-s) |
|---|---|
| 2.50E-08 | 3.67E-03 |
| 1.00E-07 | 3.67E-03 |
| 1.00E-06 | 4.46E-03 |
| 1.00E-05 | 4.54E-03 |
| 1.00E-04 | 4.18E-03 |
| 1.00E-03 | 3.76E-03 |
| 1.00E-02 | 3.56E-03 |
| 1.00E-01 | 2.17E-02 |
| 5.00E-01 | 9.26E-02 |
| 1.00E+00 | 1.32E-01 |
| 2.50E+00 | 1.25E-01 |
| 5.00E+00 | 1.56E-01 |
| 7.00E+00 | 1.47E-01 |
| 1.00E+01 | 1.47E-01 |
| 1.40E+01 | 2.08E-01 |
| 2.00E+01 | 2.27E-01 |

### 5.4.4. External Radiation Levels

The maximum external radiation levels are determined individually for each of the three content types. The limiting dose rate location for all content types is the NCT side package surface. That is, for each of the three contents, the maximum allowable quantity of material based on external radiation levels is limited by the NCT side surface dose rate. The external radiation levels resulting from each of the three content types are summarized below.

#### 5.4.4.1. Irradiated Fuel

For the irradiated fuel contents, the resulting external dose rates are calculated in two steps. [[

]] The statistical uncertainty associated with the Monte Carlo calculation for photons is added on to the calculated dose rate response as shown in Equation 5-5. [[

]] The dose rates per g U-235 are calculated, as shown in Equation 5-6, [[

]] The dose rates per gram U-235 for all burnup-enrichment pairings, at each regulatory dose rate location are provided in Tables 5.4-3 through 5.4-10.

**Table 5.4-3. NCT Top Surface Dose Rates per g U-235 by Burnup-Enrichment Pairing**

| Top Surface ḊṘ (mrem/hr/gU235) | | | | | | | |
|---|---|---|---|---|---|---|---|
| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 5.936E-03 | 1.904E-02 | 5.181E-02 | 1.098E-01 | 1.963E-01 | 3.218E-01 | 5.635E-01 |
| 2.0≤ E <2.5 | 4.159E-03 | 1.162E-02 | 3.057E-02 | 6.637E-02 | 1.218E-01 | 2.020E-01 | 3.535E-01 |
| 2.5≤ E <3.0 | 3.187E-03 | 7.930E-03 | 1.973E-02 | 4.309E-02 | 8.069E-02 | 1.356E-01 | 2.375E-01 |
| 3.0≤ E <3.5 | 2.580E-03 | 5.860E-03 | 1.363E-02 | 2.948E-02 | 5.593E-02 | 9.514E-02 | 1.668E-01 |
| 3.5≤ E <4.0 | 2.170E-03 | 4.583E-03 | 9.930E-03 | 2.103E-02 | 4.012E-02 | 6.894E-02 | 1.212E-01 |
| 4.0≤ E <4.5 | 1.872E-03 | 3.738E-03 | 7.577E-03 | 1.558E-02 | 2.964E-02 | 5.127E-02 | 9.041E-02 |
| 4.5≤ E <5.0 | 1.648E-03 | 3.146E-03 | 6.015E-03 | 1.193E-02 | 2.249E-02 | 3.900E-02 | 6.898E-02 |
| 5.0≤ E <5.5 | 1.471E-03 | 2.713E-03 | 4.926E-03 | 9.410E-03 | 1.748E-02 | 3.027E-02 | 5.363E-02 |
| 5.5≤ E <6.0 | 1.330E-03 | 2.382E-03 | 4.136E-03 | 7.605E-03 | 1.387E-02 | 2.392E-02 | 4.239E-02 |

**Table 5.4-4. NCT Side Surface Dose Rates per g U-235 by Burnup-Enrichment Pairing**

| Side Surface ḊṘ (mrem/hr/gU235) | | | | | | | |
|---|---|---|---|---|---|---|---|
| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 3.492E-02 | 1.105E-01 | 2.986E-01 | 6.312E-01 | 1.127E+00 | 1.845E+00 | 3.229E+00 |
| 2.0≤ E <2.5 | 2.451E-02 | 6.761E-02 | 1.765E-01 | 3.817E-01 | 6.990E-01 | 1.159E+00 | 2.026E+00 |
| 2.5≤ E <3.0 | 1.880E-02 | 4.627E-02 | 1.141E-01 | 2.480E-01 | 4.635E-01 | 7.782E-01 | 1.361E+00 |
| 3.0≤ E <3.5 | 1.523E-02 | 3.426E-02 | 7.894E-02 | 1.699E-01 | 3.215E-01 | 5.461E-01 | 9.566E-01 |
| 3.5≤ E <4.0 | 1.281E-02 | 2.684E-02 | 5.762E-02 | 1.213E-01 | 2.308E-01 | 3.958E-01 | 6.953E-01 |
| 4.0≤ E <4.5 | 1.106E-02 | 2.192E-02 | 4.405E-02 | 8.999E-02 | 1.706E-01 | 2.945E-01 | 5.188E-01 |
| 4.5≤ E <5.0 | 9.734E-03 | 1.847E-02 | 3.502E-02 | 6.902E-02 | 1.296E-01 | 2.242E-01 | 3.959E-01 |
| 5.0≤ E <5.5 | 8.694E-03 | 1.594E-02 | 2.872E-02 | 5.448E-02 | 1.008E-01 | 1.741E-01 | 3.079E-01 |
| 5.5≤ E <6.0 | 7.860E-03 | 1.401E-02 | 2.415E-02 | 4.409E-02 | 8.006E-02 | 1.376E-01 | 2.435E-01 |

**Table 5.4-5. NCT Bottom Surface Dose Rates per g U-235 by Burnup-Enrichment Pairing**

| Bottom Surface ḊṘ (mrem/hr/gU235) | | | | | | | |
|---|---|---|---|---|---|---|---|
| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 1.571E-02 | 5.799E-02 | 1.684E-01 | 3.657E-01 | 6.607E-01 | 1.089E+00 | 1.916E+00 |
| 2.0≤ E <2.5 | 1.078E-02 | 3.441E-02 | 9.802E-02 | 2.196E-01 | 4.084E-01 | 6.825E-01 | 1.200E+00 |
| 2.5≤ E <3.0 | 8.140E-03 | 2.289E-02 | 6.227E-02 | 1.415E-01 | 2.696E-01 | 4.571E-01 | 8.051E-01 |
| 3.0≤ E <3.5 | 6.523E-03 | 1.652E-02 | 4.230E-02 | 9.596E-02 | 1.860E-01 | 3.198E-01 | 5.647E-01 |
| 3.5≤ E <4.0 | 5.445E-03 | 1.267E-02 | 3.029E-02 | 6.780E-02 | 1.327E-01 | 2.310E-01 | 4.096E-01 |
| 4.0≤ E <4.5 | 4.673E-03 | 1.016E-02 | 2.272E-02 | 4.970E-02 | 9.749E-02 | 1.712E-01 | 3.049E-01 |
| 4.5≤ E <5.0 | 4.096E-03 | 8.428E-03 | 1.774E-02 | 3.764E-02 | 7.349E-02 | 1.297E-01 | 2.321E-01 |
| 5.0≤ E <5.5 | 3.646E-03 | 7.181E-03 | 1.431E-02 | 2.934E-02 | 5.671E-02 | 1.003E-01 | 1.800E-01 |
| 5.5≤ E <6.0 | 3.287E-03 | 6.243E-03 | 1.184E-02 | 2.344E-02 | 4.468E-02 | 7.887E-02 | 1.419E-01 |

**Table 5.4-6. NCT 2-meter Dose Rates per g U-235 by Burnup-Enrichment Pairing**

| 2 Meter ḊṘ (mrem/hr/gU235) | | | | | | | |
|---|---|---|---|---|---|---|---|
| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 6.709E-04 | 2.407E-03 | 6.905E-03 | 1.493E-02 | 2.692E-02 | 4.433E-02 | 7.791E-02 |
| 2.0≤ E <2.5 | 4.623E-04 | 1.436E-03 | 4.029E-03 | 8.975E-03 | 1.665E-02 | 2.779E-02 | 4.883E-02 |
| 2.5≤ E <3.0 | 3.502E-04 | 9.599E-04 | 2.567E-03 | 5.791E-03 | 1.100E-02 | 1.862E-02 | 3.276E-02 |
| 3.0≤ E <3.5 | 2.812E-04 | 6.961E-04 | 1.749E-03 | 3.934E-03 | 7.595E-03 | 1.303E-02 | 2.298E-02 |
| 3.5≤ E <4.0 | 2.351E-04 | 5.357E-04 | 1.257E-03 | 2.784E-03 | 5.425E-03 | 9.420E-03 | 1.668E-02 |
| 4.0≤ E <4.5 | 2.020E-04 | 4.310E-04 | 9.457E-04 | 2.045E-03 | 3.989E-03 | 6.986E-03 | 1.242E-02 |
| 4.5≤ E <5.0 | 1.772E-04 | 3.587E-04 | 7.408E-04 | 1.552E-03 | 3.011E-03 | 5.297E-03 | 9.456E-03 |
| 5.0≤ E <5.5 | 1.578E-04 | 3.064E-04 | 5.992E-04 | 1.212E-03 | 2.326E-03 | 4.097E-03 | 7.337E-03 |
| 5.5≤ E <6.0 | 1.424E-04 | 2.669E-04 | 4.973E-04 | 9.704E-04 | 1.835E-03 | 3.226E-03 | 5.786E-03 |

**Table 5.4-7. NCT Cab Dose Rates per g U-235 by Burnup-Enrichment Pairing**

| Cab ḊṘ (mrem/hr/gU235) | | | | | | | |
|---|---|---|---|---|---|---|---|
| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 1.217E-04 | 4.328E-04 | 1.237E-03 | 2.671E-03 | 4.814E-03 | 7.926E-03 | 1.392E-02 |
| 2.0≤ E <2.5 | 8.397E-05 | 2.586E-04 | 7.224E-04 | 1.606E-03 | 2.978E-03 | 4.968E-03 | 8.727E-03 |
| 2.5≤ E <3.0 | 6.367E-05 | 1.731E-04 | 4.606E-04 | 1.037E-03 | 1.968E-03 | 3.329E-03 | 5.856E-03 |
| 3.0≤ E <3.5 | 5.116E-05 | 1.257E-04 | 3.142E-04 | 7.046E-04 | 1.359E-03 | 2.331E-03 | 4.109E-03 |
| 3.5≤ E <4.0 | 4.279E-05 | 9.687E-05 | 2.259E-04 | 4.990E-04 | 9.710E-04 | 1.685E-03 | 2.981E-03 |
| 4.0≤ E <4.5 | 3.678E-05 | 7.803E-05 | 1.702E-04 | 3.667E-04 | 7.142E-04 | 1.250E-03 | 2.220E-03 |
| 4.5≤ E <5.0 | 3.227E-05 | 6.500E-05 | 1.334E-04 | 2.785E-04 | 5.392E-04 | 9.478E-04 | 1.691E-03 |
| 5.0≤ E <5.5 | 2.875E-05 | 5.555E-05 | 1.080E-04 | 2.177E-04 | 4.167E-04 | 7.333E-04 | 1.312E-03 |
| 5.5≤ E <6.0 | 2.595E-05 | 4.843E-05 | 8.972E-05 | 1.743E-04 | 3.289E-04 | 5.774E-04 | 1.035E-03 |

**Table 5.4-8. HAC Top 1-meter Dose Rates per g U-235 by Burnup-Enrichment Pairing**

| Top 1 Meter ḊṘ (mrem/hr/gU235) | | | | | | | |
|---|---|---|---|---|---|---|---|
| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 1.088E-02 | 2.456E-02 | 5.239E-02 | 9.915E-02 | 1.675E-01 | 2.659E-01 | 4.544E-01 |
| 2.0≤ E <2.5 | 7.930E-03 | 1.630E-02 | 3.276E-02 | 6.182E-02 | 1.058E-01 | 1.688E-01 | 2.871E-01 |
| 2.5≤ E <3.0 | 6.235E-03 | 1.195E-02 | 2.247E-02 | 4.159E-02 | 7.155E-02 | 1.148E-01 | 1.944E-01 |
| 3.0≤ E <3.5 | 5.136E-03 | 9.355E-03 | 1.649E-02 | 2.960E-02 | 5.076E-02 | 8.171E-02 | 1.378E-01 |
| 3.5≤ E <4.0 | 4.372E-03 | 7.665E-03 | 1.274E-02 | 2.203E-02 | 3.738E-02 | 6.018E-02 | 1.012E-01 |
| 4.0≤ E <4.5 | 3.806E-03 | 6.486E-03 | 1.026E-02 | 1.705E-02 | 2.841E-02 | 4.557E-02 | 7.634E-02 |
| 4.5≤ E <5.0 | 3.372E-03 | 5.622E-03 | 8.546E-03 | 1.364E-02 | 2.222E-02 | 3.536E-02 | 5.897E-02 |
| 5.0≤ E <5.5 | 3.026E-03 | 4.964E-03 | 7.300E-03 | 1.122E-02 | 1.782E-02 | 2.804E-02 | 4.647E-02 |
| 5.5≤ E <6.0 | 2.746E-03 | 4.444E-03 | 6.361E-03 | 9.440E-03 | 1.461E-02 | 2.266E-02 | 3.726E-02 |

**Table 5.4-9. HAC Side 1-meter Dose Rates per g U-235 by Burnup-Enrichment Pairing**

| Side 1 Meter $\dot{D}\dot{R}$ (mrem/hr/gU235) | | | | | | | |
|---|---|---|---|---|---|---|---|
| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 2.325E-02 | 4.368E-02 | 7.559E-02 | 1.244E-01 | 1.933E-01 | 2.905E-01 | 4.745E-01 |
| 2.0≤ E <2.5 | 1.719E-02 | 3.056E-02 | 5.013E-02 | 8.093E-02 | 1.255E-01 | 1.881E-01 | 3.039E-01 |
| 2.5≤ E <3.0 | 1.364E-02 | 2.331E-02 | 3.632E-02 | 5.693E-02 | 8.755E-02 | 1.307E-01 | 2.089E-01 |
| 3.0≤ E <3.5 | 1.130E-02 | 1.879E-02 | 2.800E-02 | 4.239E-02 | 6.421E-02 | 9.525E-02 | 1.506E-01 |
| 3.5≤ E <4.0 | 9.659E-03 | 1.573E-02 | 2.257E-02 | 3.298E-02 | 4.896E-02 | 7.197E-02 | 1.126E-01 |
| 4.0≤ E <4.5 | 8.433E-03 | 1.353E-02 | 1.883E-02 | 2.661E-02 | 3.857E-02 | 5.599E-02 | 8.659E-02 |
| 4.5≤ E <5.0 | 7.486E-03 | 1.187E-02 | 1.614E-02 | 2.211E-02 | 3.125E-02 | 4.468E-02 | 6.826E-02 |
| 5.0≤ E <5.5 | 6.729E-03 | 1.058E-02 | 1.411E-02 | 1.882E-02 | 2.594E-02 | 3.645E-02 | 5.495E-02 |
| 5.5≤ E <6.0 | 6.114E-03 | 9.538E-03 | 1.254E-02 | 1.633E-02 | 2.198E-02 | 3.032E-02 | 4.505E-02 |

**Table 5.4-10. HAC Bottom 1-meter Dose Rates per g U-235 by Burnup-Enrichment Pairing**

| Bottom 1 Meter $\dot{D}\dot{R}$ (mrem/hr/gU235) | | | | | | | |
|---|---|---|---|---|---|---|---|
| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 7.553E-03 | 2.050E-02 | 5.058E-02 | 1.029E-01 | 1.805E-01 | 2.928E-01 | 5.086E-01 |
| 2.0≤ E <2.5 | 5.403E-03 | 1.298E-02 | 3.051E-02 | 6.289E-02 | 1.126E-01 | 1.845E-01 | 3.198E-01 |
| 2.5≤ E <3.0 | 4.197E-03 | 9.155E-03 | 2.016E-02 | 4.135E-02 | 7.516E-02 | 1.244E-01 | 2.154E-01 |
| 3.0≤ E <3.5 | 3.429E-03 | 6.954E-03 | 1.428E-02 | 2.870E-02 | 5.251E-02 | 8.767E-02 | 1.518E-01 |
| 3.5≤ E <4.0 | 2.903E-03 | 5.564E-03 | 1.067E-02 | 2.081E-02 | 3.802E-02 | 6.387E-02 | 1.106E-01 |
| 4.0≤ E <4.5 | 2.517E-03 | 4.622E-03 | 8.332E-03 | 1.567E-02 | 2.838E-02 | 4.779E-02 | 8.284E-02 |
| 4.5≤ E <5.0 | 2.223E-03 | 3.949E-03 | 6.758E-03 | 1.221E-02 | 2.177E-02 | 3.661E-02 | 6.346E-02 |
| 5.0≤ E <5.5 | 1.990E-03 | 3.447E-03 | 5.642E-03 | 9.797E-03 | 1.711E-02 | 2.863E-02 | 4.956E-02 |
| 5.5≤ E <6.0 | 1.803E-03 | 3.057E-03 | 4.821E-03 | 8.053E-03 | 1.375E-02 | 2.280E-02 | 3.937E-02 |

For a defined mass of uranium at a given burnup and initial enrichment, the resulting dose rate at any regulatory dose rate location can be calculated by multiplying the mass of initial fissile material (grams U-235) by the dose rates per gram U-235 in the corresponding table, as shown in Equation 5-7. Repeating this dose rate calculation for each loaded fuel segment, then summing the resulting dose rates calculates the total external dose rates for each regulatory dose rate location from a load of segmented irradiated fuel in the Model 2000 Transport Package. This process is completed and recorded in the Irradiated Fuel Loading Table (Section 7.5.3). The use of the Irradiated Fuel Loading Table is described in Section 5.5.5. The maximum possible dose rate for each regulatory location is shown in Table 5.4-11, based on the limiting masses of U-235 shown in Table 5.5-35. The maximum quantity of irradiated fuel is limited by the minimum of the quantity equivalent to the 1500 W thermal limit of the cask or the quantity resulting in an NCT side surface dose rate equal to 90% of the regulatory limit (180 mrem/hr) or the quantity equivalent to the criticality limit of 1750 g.

**Table 5.4-11. Maximum External Dose Rates - Irradiated Fuel**

| Location | NCT Top Surface | NCT Side Surface | NCT Bottom Surface | NCT 2-meter | NCT Cab | HAC Top 1-meter | HAC Side 1-meter | HAC Bottom 1-meter |
|---|---|---|---|---|---|---|---|---|
| **Dose Rate (mrem/hr)** | 31.41 | 180.00 | 106.77 | 4.34 | 0.78 | 33.42 | 53.63 | 31.12 |
| **Regulatory Limit (mrem/hr)** | 200.0 | 200.0 | 200.0 | 10.0 | 2.0 | 1000 | 1000 | 1000 |

It is recognized that fuel burnup is typically tracked by means of a core simulator that uses a nodal three-dimensional neutronics model. In such a case, fuel burnup is known at every axial node, and each node represents a certain axial height of fuel, e.g., 5.5 inches. A fuel segment loaded for shipment may span several core simulator nodes. For example, a 15-inch fuel segment may have been located in three, or even four, of the 5.5-inch computational nodes. The appropriate procedure in this instance to calculate the dose rate from the 15-inch segment is to multiply the mass of U-235 in each node by the dose rate per gram U-235 from Tables 5.4-3 through 5.4-10 that correspond to that node's initial enrichment and burnup, and then to sum the dose rates.

For example, assume that 5.5 inches of this fuel segment were located in node A, 5.5 inches in node B, and 4 inches in node C. Node A has a burnup of 21 GWd/MTU, node B 19 GWd/MTU, and node C 15 GWd/MTU. All nodes have an initial enrichment of 4.2 wt% U-235. To find the NCT side surface dose rate, the mass of node A is multiplied by 4.405E-02 mrem/hr/gU235 from Table 5.4-4, and the mass of nodes B and C is multiplied by 2.192E-02 mrem/hr/gU235 from the same table. These dose rates are then summed to determine the dose rate for the entire 15-inch segment.

- The following condition applies: If the node height of the core simulator is less than 5.3 inches, then the peak exposure over every 5.3 inches shall be conservatively used in order to be consistent with the minimum segment height assumed in the shielding evaluation.

  For example, assume that it is desired to ship a six inch segment and that the burnups for each inch are known to be 28.2, 28.6, 29.0, 29.4, 29.8, and 30.2 GWd/MTU. To find the NCT side surface dose rate, the initial U-235 mass of all six nodes is multiplied by 8.999E-02 mrem/hr/gU235 from Table 5.4-4 so that the limiting exposure band ($30 < B \leq 40$) is represented along the entire segment.

### 5.4.4.2. Irradiated Hardware and Byproducts

For the irradiated hardware and byproduct contents, the resulting external dose rates are calculated by calculating the dose rate per curie in MCNP6 for each radionuclide individually, using the source spectra listed in Tables 5.5-8 through 5.5-25. The $2\sigma$ statistical uncertainty is added on to the calculated dose rate per curie as shown in Equation 5-10. The resulting values from these calculations for NCT and HAC are presented in Tables 5.4-12 and 5.4-13.

**Table 5.4-12. Irradiated Hardware and Byproduct Dose Rate per Curie Results - NCT**

| Radionuclide | Dose Rate (mrem/hr/Ci) | | | | |
|---|---|---|---|---|---|
| | Top Surface | Side Surface | Bottom Surface | 2-meter | Cab |
| Co-58 | 1.356E-05 | 2.444E-04 | 2.169E-05 | 3.375E-06 | 5.945E-07 |
| Co-60 | 4.183E-04 | 9.437E-03 | 6.237E-04 | 1.211E-04 | 2.078E-05 |
| Cr-51 | 3.981E-15 | 1.648E-20 | 5.556E-20 | 2.125E-22 | 3.509E-23 |
| Cs-134 | 1.584E-05 | 3.433E-04 | 2.402E-05 | 4.504E-06 | 7.780E-07 |
| Cs-137 | 9.341E-10 | 3.713E-08 | 2.822E-09 | 4.381E-10 | 7.404E-11 |
| Fe-59 | 1.239E-04 | 2.894E-03 | 1.853E-04 | 3.649E-05 | 6.292E-06 |
| Hf-175 | 1.642E-13 | 1.917E-15 | 3.513E-16 | 1.167E-17 | 1.891E-18 |
| Hf-181 | 1.968E-11 | 2.116E-11 | 3.427E-12 | 2.749E-13 | 4.721E-14 |
| Mn-54 | 2.294E-07 | 9.856E-06 | 4.279E-07 | 1.116E-07 | 1.904E-08 |
| Nb-92m | 4.559E-05 | 8.088E-04 | 7.216E-05 | 1.128E-05 | 1.974E-06 |
| Nb-94 | 5.561E-07 | 2.214E-05 | 9.703E-07 | 2.519E-07 | 4.293E-08 |
| Sb-124 | 1.973E-03 | 3.462E-02 | 3.176E-03 | 4.809E-04 | 8.336E-05 |
| Sb-125 | 1.440E-10 | 3.334E-09 | 3.261E-10 | 4.139E-11 | 7.082E-12 |
| Sb-126 | 6.649E-06 | 1.526E-04 | 1.009E-05 | 1.955E-06 | 3.399E-07 |
| Sc-46 | 4.123E-05 | 1.111E-03 | 6.179E-05 | 1.353E-05 | 2.327E-06 |
| Ta-182 | 1.343E-04 | 3.273E-03 | 1.997E-04 | 4.080E-05 | 6.989E-06 |
| Zn-65 | 2.023E-05 | 5.398E-04 | 3.014E-05 | 6.581E-06 | 1.131E-06 |
| Zr-95 | 5.015E-08 | 2.505E-06 | 1.181E-07 | 2.816E-08 | 4.790E-09 |
| Nb-95 | 5.015E-08 | 2.505E-06 | 1.181E-07 | 2.816E-08 | 4.790E-09 |

**Table 5.4-13. Irradiated Hardware and Byproduct Dose Rate per Curie Results - HAC**

| Radionuclide | Dose Rate (mrem/hr/Ci) | | |
|---|---|---|---|
| | Top 1-meter | Side 1-meter | Bottom 1-meter |
| Co-58 | 1.795E-05 | 4.460E-05 | 1.155E-05 |
| Co-60 | 6.008E-04 | 1.743E-03 | 3.552E-04 |
| Cr-51 | 1.268E-13 | 5.043E-20 | 9.455E-19 |
| Cs-134 | 2.252E-05 | 6.356E-05 | 1.340E-05 |
| Cs-137 | 3.023E-09 | 9.383E-09 | 3.140E-09 |
| Fe-59 | 1.806E-04 | 5.370E-04 | 1.071E-04 |
| Hf-175 | 1.676E-12 | 4.820E-16 | 7.744E-16 |
| Hf-181 | 1.493E-10 | 6.868E-12 | 5.073E-12 |
| Mn-54 | 4.057E-07 | 2.046E-06 | 3.326E-07 |
| Nb-92m | 5.907E-05 | 1.450E-04 | 3.671E-05 |
| Nb-94 | 9.502E-07 | 4.548E-06 | 7.240E-07 |
| Sb-124 | 2.568E-03 | 6.144E-03 | 1.618E-03 |
| Sb-125 | 6.258E-10 | 9.213E-10 | 4.006E-10 |
| Sb-126 | 9.516E-06 | 2.875E-05 | 5.919E-06 |
| Sc-46 | 6.274E-05 | 2.115E-04 | 3.852E-05 |
| Ta-182 | 1.976E-04 | 6.138E-04 | 1.176E-04 |
| Zn-65 | 3.066E-05 | 1.026E-04 | 1.860E-05 |
| Zr-95 | 9.528E-08 | 5.459E-07 | 1.039E-07 |
| Nb-95 | 9.528E-08 | 5.459E-07 | 1.039E-07 |

The resulting dose rate at any regulatory dose rate location can be calculated, for a cask loading of irradiated hardware or byproducts with a defined radionuclide inventory, by multiplying the activity of each radionuclide by the respective dose rate per curie for the given location and summing the dose rate contributions from each radionuclide, as shown in Equation 5-11. Repeating this dose rate calculation for each regulatory dose rate location determines the total external dose rates for the Model 2000 Transport Package. This process is completed and recorded in the Irradiated Hardware and Byproduct Loading Table. The use of the Irradiated Hardware and Byproduct Loading Table is described in Section 5.5.6.

The maximum activity of each radionuclide, individually, is limited by the minimum of either the activity equivalent to the 1500 W thermal limit of the cask or the activity resulting in an NCT side surface dose rate equal to 90% of the regulatory limit (180 mrem/hr). The maximum activity limit for each radionuclide individually is presented in Table 5.4-14. These limits are based on the dose rate per curie limits in Tables 5.4-12 and 5.4-13, and the decay heat W/Ci values in Table 5.5-29. The maximum possible dose rates for each regulatory location are summarized in Table 5.4-15. These values are calculated using the activity limits in Table 5-4.14 and the dose rate per curie values in Tables 5.4-12 and 5.4-13 for each radionuclide.

**Table 5.4-14. Maximum Activities for Irradiated Hardware and Byproduct Individual Radionuclides**

| Radionuclide | Activity Limit (Ci) | Basis[a,b] |
|---|---|---|
| Co-58 | 2.508E+05 | Thermal |
| Co-60 | 1.907E+04 | Dose Rate |
| Cr-51 | 6.899E+06 | Thermal |
| Cs-134 | 1.472E+05 | Thermal |
| Cs-137 | 3.009E+05 | Thermal |
| Fe-59 | 6.219E+04 | Dose Rate |
| Hf-175 | 6.369E+05 | Thermal |
| Hf-181 | 3.465E+05 | Thermal |
| Mn-54 | 3.011E+05 | Thermal |
| Nb-92m | 2.225E+05 | Dose Rate |
| Nb-94 | 1.438E+05 | Thermal |
| Sb-124 | 5.199E+03 | Dose Rate |
| Sb-125 | 4.742E+05 | Thermal |
| Sb-126 | 8.108E+04 | Thermal |
| Sc-46 | 1.192E+05 | Thermal |
| Ta-182 | 5.499E+04 | Dose Rate |
| Zn-65 | 3.334E+05 | Dose Rate |
| Zr-95 | 1.525E+05 | Thermal |
| Nb-95 | 3.127E+05 | Thermal |

Notes: [a] Thermal – 1500 W thermal limit.
[b] Dose Rate – 180 mrem/hr NCT side surface dose rate limit.

**Table 5.4-15. Maximum External Dose Rates - Irradiated Hardware and Byproducts**

| Location | NCT Top Surface | NCT Side Surface | NCT Bottom Surface | NCT 2-meter | NCT Cab | HAC Top 1-meter | HAC Side 1-meter | HAC Bottom 1-meter |
|---|---|---|---|---|---|---|---|---|
| Dose Rate (mrem/hr) | 10.26 | 179.99 | 16.51 | 2.51 | 0.44 | 13.35 | 34.21 | 8.41 |
| Regulatory Limit (mrem/hr) | 200 | 200 | 200 | 10 | 2 | 1000 | 1000 | 1000 |

### 5.4.4.3. Cobalt-60 Isotope Rods

For the cobalt-60 isotope rod contents, the resulting external dose rates are calculated using the dose rate per curie in MCNP6 for cobalt-60, with the cobalt-60 source energy spectrum listed in Table 5.2-2. The 2σ statistical uncertainty is added on to the calculated dose rate per curie as shown in Equation 5-10. The resulting values from these calculations for NCT and HAC are presented in Tables 5.4-16 and 5.4-17.

**Table 5.4-16. Cobalt-60 Isotope Rod Dose Rate per Curie Results – NCT**

| Radionuclide | Dose Rate (mrem/hr/Ci) | | | | |
|---|---|---|---|---|---|
| | Top Surface | Side Surface | Bottom Surface | 2-meter | Cab |
| Co-60 | 1.850E-04 | 8.821E-04 | 3.940E-04 | 1.664E-05 | 2.921E-06 |

**Table 5.4-17. Cobalt-60 Isotope Rod Dose Rate per Curie Results - HAC**

| Radionuclide | Dose Rate (mrem/hr/Ci) | | |
|---|---|---|---|
| | Top 1-meter | Side 1-meter | Bottom 1-meter |
| Co-60 | 6.008E-04 | 1.743E-03 | 3.552E-04 |

The resulting dose rate at any regulatory dose rate location can be calculated, for a cask loading of cobalt-60 isotope rod [[ ]], by filling out the cobalt-60 isotope rod loading table.

To determine the maximum possible dose rate at each regulatory location, the dose rate per curie values in Tables 5.4-16 and 5.4-17 are multiplied by the cobalt-60 activity equivalent to the 1500 W thermal limit (97,250 Ci). The results of this calculation are presented in Table 5.4-18.

**Table 5.4-18. Maximum External Dose Rates – Cobalt-60 Isotope Rods**

| Location | NCT Top Surface | NCT Side Surface | NCT Bottom Surface | NCT 2-meter | NCT Cab | HAC Top 1-meter | HAC Side 1-meter | HAC Bottom 1-meter |
|---|---|---|---|---|---|---|---|---|
| Dose Rate (mrem/hr) | 17.99 | 85.78 | 38.32 | 1.62 | 0.28 | 58.43 | 169.51 | 34.54 |
| Regulatory Limit (mrem/hr) | 200 | 200 | 200 | 10 | 2 | 1000 | 1000 | 1000 |

#### 5.4.4.4. Combined Contents

There is the possibility of a shipment that includes combined contents. For example, some shipments of irradiated fuel will also include irradiated hardware. This is due to the possibility of being encapsulated in fuel cladding. For this reason, shipments of irradiated fuel may also include segments of irradiated bundle hardware. In order to demonstrate compliance with the 10 CFR 71 external dose rate limits for a shipment of combined content types, the external dose rate contributions from each content type are calculated separately. The total thermal and dose rates for the shipment of combined contents are calculated as the sum of the thermal and dose rate contributions from each content type. This process is completed and recorded in the Combined Contents Loading Table. The use of the Combined Contents Loading Table is described in Section 5.5.7, including an example of a hypothetical shipment with irradiated fuel and bundle hardware.

Another example is a combined content of cobalt-60 isotope rods with irradiated hardware. For this case the Cobalt-60 Isotope Rod Loading Table (in Chapter 7) is confirmed for the isotope rod contents and any radionuclide activity in the rod cladding or additional irradiated hardware shipped with the isotope rods is confirmed in the Irradiated Hardware and Byproduct Loading Table (in Chapter 7). The resulting thermal and dose rate contributions from radionuclides in the hardware and cladding are summed with the thermal and dose rate contributions from the cobalt-60 isotope rods in the Combined Contents Loading Table (in Chapter 7).

## 5.5 Appendices

### 5.5.1. ORIGEN-S Irradiated Fuel Source Term Calculation

Per the recommendations for spent fuel specifications provided in NUREG/CR-6716 (Reference 5-10), the principal parameters for spent fuel source term generation are burnup, enrichment, and cooling time. To generate a bounding source term, the maximum burnup should be considered, along with the minimum enrichment and cooling time. To provide flexibility for future use of the Model 2000 cask, a wide range of enrichments and burnups are considered. The irradiated fuel source term calculation is performed for several enrichment bands and burnup bands. In the ORIGEN-S (Reference 5-2) source term analysis, for each initial enrichment band the minimum enrichment is considered, and for each burnup band the maximum burnup is considered. This generates a bounding source term for each burnup-enrichment pairing. Table 5.5-1 shows the burnup bands and Table 5.5-2 shows the initial enrichment bands for which a separate source term is calculated. The source terms are taken at a cooling time of 120 days. Any irradiated fuel contents are required to have at least 120 days of cooling time prior to shipment in the Model 2000 cask.

**Table 5.5-1. Burnup Bands and Analyzed Values**

| Burnup Band (GWd/MTU) | Analyzed Burnup (GWd/MTU) |
|---|---|
| 60< B ≤72 | 72 |
| 50< B ≤60 | 60 |
| 40< B ≤50 | 50 |
| 30< B ≤40 | 40 |
| 20< B ≤30 | 30 |
| 10< B ≤20 | 20 |
| 0< B ≤10 | 10 |

**Table 5.5-2. Initial Enrichment Bands and Analyzed Values**

| Initial Enrichment Band (wt%) | Analyzed Initial Enrichment (wt%) |
|---|---|
| 1.5≤ E <2.0 | 1.5 |
| 2.0≤ E <2.5 | 2.0 |
| 2.5≤ E <3.0 | 2.5 |
| 3.0≤ E <3.5 | 3.0 |
| 3.5≤ E <4.0 | 3.5 |
| 4.0≤ E <4.5 | 4.0 |
| 4.5≤ E <5.0 | 4.5 |
| 5.0≤ E <5.5 | 5.0 |
| 5.5≤ E <6.0 | 5.5 |

Table 5.5-3 lists the values used for the secondary parameters for the ORIGEN-S irradiated fuel source term calculation. While these parameters are not as significant to the irradiated fuel source term calculation as the principal parameters, they are selected to generate a bounding source term. The additional parameters include the fuel assembly type analyzed, the presence of burnable poisons, the specific power analyzed, and the moderator density considered. The values used for each parameter are selected to be appropriate, or bounding, for the irradiated fuel contents outlined in Section 5.2.

**Table 5.5-3. Secondary Source Term Calculation Parameters**

| Parameter | Value |
|---|---|
| Fuel Assembly Type | GE BWR 10x10 |
| Burnable Poison | 5 wt% $Gd_2O_3$ [1] |
| Specific Power | 40 MW/MTU [2] |
| Moderator Density | 0.1 $g/cm^3$ |

Notes: [1] Contained in $UO_2$-$Gd_2O_3$
[2] Conservative value for BWRs and consistent with the maximum value used in NUREG/CR-6716, Table 8 (Reference 5-10)

The source strengths calculated for each enrichment band are normalized to 1 gram of U-235, so that the total source strength for a shipment of segmented irradiated fuel can be calculated by multiplying the source strength for a respective burnup-enrichment pairing by the total mass of U-235.

The ORIGEN-S analysis calculates the source term using the parameters listed above, resulting in the gamma and neutron source strength values per gram of initial U-235 for each burnup-enrichment pairing [[

]] The source strength values used include 120 days of cooling time. Extended cooling times beyond 120 days will only reduce the calculated source strengths.

An example [[ ]]is included in Table 5.5-4 for the $4.0 \leq E < 4.5$ enrichment band and $30 < B \leq 40$ burnup band, [[

]]

**Table 5.5-4. Example [[ ]]for a Single Enrichment and Burnup Pair**

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The neutron source term is from radionuclides that emit neutrons through spontaneous fission (SF) and emitted alphas that generate neutrons through alpha-n reactions in the fuel (α-n). The neutron source terms are dominated by Cm-244(SF) and Cm-242(SF), with significant contributions at higher burnups from Cf-252(SF), and at lower burnups from Pu-238(α-n).

For each burnup-enrichment pairing, the total wattage per gram U-235 is listed in the outputs from the ORIGEN-S calculations. The total wattage results from all ORIGEN-S outputs are listed in Table 5.5-5. These values can be used to ensure that the thermal limit of the cask will not be exceeded from a load of irradiated fuel.

**Table 5.5-5. Irradiated Fuel Total Radionuclide Decay Heat (W/gU235)**

| Enrichment (wt%) | Burnup (GWd/MTU) | | | | | | |
|---|---|---|---|---|---|---|---|
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 1.088E+00 | 1.453E+00 | 1.728E+00 | 1.963E+00 | 2.163E+00 | 2.335E+00 | 2.512E+00 |
| 2.0≤ E <2.5 | 8.215E-01 | 1.086E+00 | 1.282E+00 | 1.452E+00 | 1.601E+00 | 1.731E+00 | 1.866E+00 |
| 2.5≤ E <3.0 | 6.604E-01 | 8.664E-01 | 1.016E+00 | 1.147E+00 | 1.264E+00 | 1.368E+00 | 1.478E+00 |
| 3.0≤ E <3.5 | 5.527E-01 | 7.210E-01 | 8.403E-01 | 9.453E-01 | 1.041E+00 | 1.128E+00 | 1.220E+00 |
| 3.5≤ E <4.0 | 4.751E-01 | 6.171E-01 | 7.160E-01 | 8.026E-01 | 8.823E-01 | 9.557E-01 | 1.036E+00 |
| 4.0≤ E <4.5 | 4.170E-01 | 5.395E-01 | 6.233E-01 | 6.963E-01 | 7.643E-01 | 8.275E-01 | 8.975E-01 |
| 4.5≤ E <5.0 | 3.716E-01 | 4.793E-01 | 5.518E-01 | 6.144E-01 | 6.729E-01 | 7.282E-01 | 7.902E-01 |
| 5.0≤ E <5.5 | 3.352E-01 | 4.310E-01 | 4.950E-01 | 5.496E-01 | 6.004E-01 | 6.492E-01 | 7.044E-01 |
| 5.5≤ E <6.0 | 3.053E-01 | 3.916E-01 | 4.485E-01 | 4.969E-01 | 5.416E-01 | 5.849E-01 | 6.345E-01 |

### 5.5.2. ORIGEN-S Irradiated Hardware and Byproduct Source Term Calculation

The radionuclides that are significant to the irradiated hardware and byproduct dose rate calculations, were determined with multiple ORIGEN-S (Reference 5-2) irradiation calculations. For the irradiation case there are two significant inputs; the composition of the material that is being irradiated and the neutron flux that the material is exposed to. For determining the source term, the quantity of material is irrelevant for the determination of which radionuclides are generated. A generic thermal neutron flux of 1E+14 n/s·cm$^2$ is assumed for the irradiation cases. For the material compositions of the irradiated hardware/byproducts, there are six materials considered. These materials along with their compositions are listed in Table 5.5-6. The materials selected include multiple SS, a nickel alloy, a zirconium alloy, as well as hafnium and boron carbide. The materials listed in parentheses in Table 5.5-6 are included as they are similar in composition to the material listed. The materials listed contain elements expected in any irradiated hardware or byproduct contents. Thus, the resulting total radionuclide inventory from the ORIGEN-S calculations is comprehensive. The basis for each ORIGEN-S input is 1 kg of the respective material being irradiated. Because elements for each material are entered into the ORIGEN-S input in grams, Table 5.5-6 lists the gram amount of each element per kilogram of the material. While an increase or decrease in the flux or a variation in the material composition entered for the irradiation case would result in a change to the relative activity of the radionuclides generated, the purpose of the ORIGEN-S source term calculations is not to determine the inventory of each radionuclide, but simply to identify which radionuclides may be present in irradiated hardware/byproduct contents. The quantity of each radionuclide that is significant to dose rate calculations must be entered into the Irradiated Hardware and Byproduct Loading Table to calculate the maximum external dose rates. The quantity of each radionuclide that is significant to the thermal calculations must also be entered into the Irradiated Hardware and Byproduct Loading Table to calculate the total thermal content.

The radionuclides calculated from the ORIGEN-S irradiation cases are listed in Table 5.5-7. This table also includes some radionuclides that may be included on the hardware or byproduct contents in the form of surface contamination, as these contents may be exposed to a reactor environment. Radionuclides in cells that are highlighted are considered significant to dose rate calculations. The selection of significant radionuclides is based on the energy of the gamma emissions and half-lives. A radionuclide is considered insignificant to dose rate calculations if it has no gamma emissions greater than 0.3 MeV, or if it has a half-life less than 3 days. All shipments of irradiated hardware and byproducts are required to include a decay time of 30 days prior to shipment. Thus, for any radionuclide with a half-life less than 3 days, there are more than 10 half-lives of decay time prior to shipment.

Tables 5.5-8 through 5.5-25 provide the energy spectra for all radionuclides considered significant to the irradiated hardware and byproduct dose rate calculations. These radionuclide energy spectra are from the ORIGEN-S Data Library origen.rev04.mpdkxgam.data (Reference 5-2). Any gamma lines under 0.1 MeV are neglected from the listed radionuclide spectra. Though Cs-137 does not emit any significant gammas, the gamma emission of its short-lived daughter Ba-137m is used as its representative spectrum. Also, because Nb-95 is the daughter of Zr-95, the energy spectra of the two radionuclides are combined and only one set of dose rate calculations is performed for both radionuclides. Thus, the dose rates calculated for this combined spectrum account for one decay of each radionuclide. This calculates an appropriate dose rate for Zr-95, as it accounts for the decay of its daughter Nb-95. However, this spectrum results in a conservative dose rate for Nb-95, as the calculated dose rate includes the contribution from its parent radionuclide as well. The resulting dose rates from this combined spectrum are used to calculate external dose rates for activities of both Zr-95 and Nb-95, individually.

**Table 5.5-6. Irradiated Hardware and Byproduct Irradiation Materials**

| Material | Symbol | Element ID No. | wt % | g/kg material | # nuclides |
|---|---|---|---|---|---|
| SS304 (SS302, SS304L) | C | 60000 | 0.0800 | 0.8000 | 10 |
| | N | 70000 | 0.1000 | 1.0000 | |
| | Si | 140000 | 0.7500 | 7.5000 | |
| | P | 150000 | 0.0450 | 0.4500 | |
| | S | 160000 | 0.0300 | 0.3000 | |
| | Cr | 240000 | 19.000 | 190.00 | |
| | Mn | 250000 | 2.0000 | 20.000 | |
| | Fe | 260000 | 67.495 | 674.95 | |
| | Co | 270000 | 0.0800 | 0.8000 | |
| | Ni | 280000 | 10.420 | 104.20 | |
| SS CF3M (SS316) | C | 60000 | 0.0300 | 0.300 | 8 |
| | Si | 140000 | 2.0000 | 20.000 | |
| | Cr | 240000 | 19.000 | 190.00 | |
| | Mn | 250000 | 1.5000 | 15.000 | |
| | Fe | 260000 | 62.970 | 629.70 | |
| | Co | 270000 | 0.0800 | 0.8000 | |
| | Ni | 280000 | 11.920 | 119.20 | |
| | Mo | 420000 | 2.5000 | 25.000 | |
| SS348H | C | 60000 | 0.0700 | 0.7000 | 11 |
| | Si | 140000 | 1.0000 | 10.000 | |
| | P | 150000 | 0.0450 | 0.4500 | |
| | S | 160000 | 0.0300 | 0.3000 | |
| | Cr | 240000 | 18.000 | 180.00 | |
| | Mn | 250000 | 2.0000 | 20.000 | |
| | Fe | 260000 | 64.555 | 645.55 | |
| | Co | 270000 | 0.2000 | 2.0000 | |
| | Ni | 280000 | 13.000 | 130.00 | |
| | Nb | 410000 | 1.0000 | 10.000 | |
| | Ta | 730000 | 0.1000 | 1.0000 | |
| Inconel-718 (Inconel X-750) | B | 50000 | 0.0060 | 0.0600 | 16 |
| | C | 60000 | 0.0800 | 0.8000 | |
| | Al | 130000 | 0.5000 | 5.0000 | |
| | Si | 140000 | 0.3500 | 3.5000 | |
| | P | 150000 | 0.0150 | 0.1500 | |
| | S | 160000 | 0.0150 | 0.1500 | |
| | Ti | 220000 | 0.9000 | 9.0000 | |
| | Cr | 240000 | 19.000 | 190.00 | |
| | Mn | 250000 | 0.3500 | 3.5000 | |
| | Fe | 260000 | 14.934 | 149.34 | |
| | Co | 270000 | 1.0000 | 10.000 | |
| | Ni | 280000 | 54.000 | 540.00 | |
| | Cu | 290000 | 0.3000 | 3.0000 | |
| | Nb | 410000 | 2.7500 | 27.500 | |
| | Mo | 420000 | 3.0500 | 30.500 | |
| | Ta | 730000 | 2.7500 | 27.500 | |

| Material | Symbol | Element ID No. | wt % | g/kg material | # nuclides |
|---|---|---|---|---|---|
| Zircaloy-2 (Zircaloy-4) | O | 80000 | 0.1200 | 1.2000 | 6 |
| | Cr | 240000 | 0.1000 | 1.0000 | |
| | Fe | 260000 | 0.2000 | 2.0000 | |
| | Ni | 280000 | 0.0800 | 0.8000 | |
| | Zr | 400000 | 97.800 | 978.00 | |
| | Sn | 500000 | 1.7000 | 17.000 | |
| Boron Carbide ($B_4C$) | B | 50000 | 78.261 | 782.61 | 2 |
| | C | 60000 | 21.739 | 217.39 | |
| Hafnium | Hf | 720000 | 100.00 | 1000.0 | 1 |

## Table 5.5-7. Irradiated Hardware and Byproduct Radionuclides

| | | | | | |
|---|---|---|---|---|---|
| H-3 | Co-58m | Sr-91 | Tc-99[a] | **Sb-125** | Lu-177m |
| C-14 | **Co-60** | Y-89m | Tc-99m | **Sb-126** | Yb-175 |
| Na-24 | Co-60m | Y-90 | Tc-101 | Te-125m | Yb-177 |
| Si-31 | Co-61 | Y-90m | Ru-106a | I-129[a] | Ta-180 |
| P-32 | Ni-57 | Y-91 | In-113m | **Cs-134[a]** | **Ta-182** |
| P-33 | Ni-59 | Y-91m | In-114 | **Cs-137 (Ba-137m) [a]** | Ta-183 |
| S-35 | Ni-63 | Y-92 | In-114m | La-140[a] | W-181 |
| Ca-45 | Ni-65 | Nb-91m | In-115m | Ba-140[a] | W-183m |
| **Sc-46** | Fe-55 | **Nb-92m** | Sn-113 | Ce-144[a] | W-185 |
| Sc-47 | **Fe-59** | **Nb-94** | Sn-113m | Hf-173 | Re-186 |
| Sc-48 | Cu-64 | **Nb-95** | Sn-117m | **Hf-175** | Np-237[a] |
| V-49 | Cu-66[a] | Nb-96 | Sn-119m | Hf-177m | Pu-238[a] |
| V-52[a] | **Zn-65** | Nb-95m | Sn-121 | Hf-180m | Pu-239[a] |
| **Cr-51** | Zr-89 | Nb-97 | Sn-121m | **Hf-181** | Pu-240[a] |
| Cr-55[a] | **Zr-95** | Nb-97m | Sn-123 | Lu-173 | Pu-241[a] |
| **Mn-54** | Zr-97 | Mo-93 | Sn-123m | Lu-174 | Am-241[a] |
| Mn-56 | Sr-87m | Mo-93m | Sn-125 | Lu-174m | Cm-242[a] |
| Co-57 | Sr-89 | Mo-99 | Sb-122 | Lu-176m | Cm-243[a] |
| **Co-58** | Sr-90[a] | Mo-101 | **Sb-124** | Lu-177 | Cm-244[a] |

Notes: [a] Radionuclides not calculated in ORIGEN-S calculations, but included from previous shipments. Only present in small quantities in surface contamination.

## Table 5.5-8. Sc-46 Gamma Emission Energy Spectrum

| Total Photons/Disintegration | |
|---|---|
| 2.000E+00 | |
| **Energy** | **Intensity** |
| 0.889 | 1.00E+00 |
| 1.121 | 1.00E+00 |
| 2.010 | 1.30E-07 |

**Table 5.5-9. Cr-51 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 9.910E-02 | |
| **Energy** | **Intensity** |
| 0.320 | 9.91E-02 |

**Table 5.5-10. Mn-54 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 9.998E-01 | |
| **Energy** | **Intensity** |
| 0.511 | 5.600E-09 |
| 0.835 | 9.998E-01 |

**Table 5.5-11. Co-58 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 1.305E+00 | |
| **Energy** | **Intensity** |
| 0.511 | 2.98E-01 |
| 0.811 | 9.95E-01 |
| 0.864 | 6.86E-03 |
| 1.675 | 5.17E-03 |

**Table 5.5-12. Fe-59 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 1.041E+00 | |
| **Energy** | **Intensity** |
| 0.143 | 1.02E-02 |
| 0.189 | 9.00E-06 |
| 0.192 | 3.08E-02 |
| 0.335 | 2.70E-03 |
| 0.382 | 1.80E-04 |
| 1.099 | 5.65E-01 |
| 1.292 | 4.32E-01 |
| 1.482 | 5.90E-04 |

**Table 5.5-13. Co-60 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 1.998E+00 | |
| **Energy** | **Intensity** |
| 0.347 | 7.5000E-05 |
| 0.826 | 7.6000E-05 |
| 1.173 | 9.9850E-01 |
| 1.333 | 9.9983E-01 |
| 2.159 | 1.2000E-05 |
| 2.506 | 2.0000E-08 |

**Table 5.5-14. Zn-65 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 5.289E-01 | |
| **Energy** | **Intensity** |
| 0.511 | 2.84E-02 |
| 0.345 | 2.53E-05 |
| 0.771 | 2.68E-05 |
| 1.116 | 5.00E-01 |

**Table 5.5-15. Nb-92m Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 1.02E+00 | |
| **Energy** | **Intensity** |
| 0.511 | 1.28E-03 |
| 0.449 | 1.63E-05 |
| 0.561 | 2.23E-05 |
| 0.913 | 1.78E-02 |
| 0.934 | 9.91E-01 |
| 1.132 | 5.15E-05 |
| 1.848 | 8.52E-03 |

**Table 5.5-16. Nb-94 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 1.997E+00 | |
| **Energy** | **Intensity** |
| 0.703 | 9.98E-01 |
| 0.871 | 9.99E-01 |

**Table 5.5-17. Zr/Nb-95 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 1.99E+00 | |
| **Energy** | **Intensity** |
| 0.204 | 2.80E-04 |
| 0.562 | 1.50E-04 |
| 0.724 | 4.43E-01 |
| 0.757 | 5.44E-01 |
| 0.766 | 9.98E-01 |

**Table 5.5-18. Sb-124 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | | | | | |
|---|---|---|---|---|---|
| 1.878E+00 | | | | | |
| **Energy** | **Intensity** | **Energy** | **Intensity** | **Energy** | **Intensity** |
| 0.148 | 3.91E-05 | 0.766 | 1.21E-04 | 1.566 | 1.37E-04 |
| 0.190 | 6.36E-05 | 0.775 | 9.39E-05 | 1.580 | 3.81E-03 |
| 0.210 | 5.48E-05 | 0.791 | 7.39E-03 | 1.622 | 4.09E-04 |
| 0.254 | 1.61E-04 | 0.817 | 7.29E-04 | 1.691 | 4.76E-01 |
| 0.292 | 8.70E-05 | 0.857 | 2.38E-04 | 1.721 | 9.51E-04 |
| 0.336 | 7.43E-04 | 0.899 | 1.72E-04 | 1.852 | 6.45E-05 |
| 0.371 | 3.81E-04 | 0.968 | 1.88E-02 | 1.919 | 5.45E-04 |
| 0.400 | 1.39E-03 | 0.977 | 8.32E-04 | 2.016 | 9.49E-04 |
| 0.444 | 1.89E-03 | 1.045 | 1.83E-02 | 2.040 | 6.42E-04 |
| 0.469 | 4.99E-04 | 1.054 | 4.89E-05 | 2.080 | 2.05E-04 |
| 0.481 | 2.37E-04 | 1.087 | 3.78E-04 | 2.091 | 5.49E-02 |
| 0.526 | 1.38E-03 | 1.264 | 4.13E-04 | 2.099 | 4.57E-04 |
| 0.530 | 4.21E-04 | 1.301 | 3.43E-04 | 2.108 | 4.33E-04 |
| 0.572 | 1.90E-04 | 1.326 | 1.58E-02 | 2.172 | 2.05E-05 |
| 0.603 | 9.78E-01 | 1.355 | 1.04E-02 | 2.183 | 4.24E-04 |
| 0.632 | 1.05E-03 | 1.368 | 2.62E-02 | 2.284 | 8.02E-05 |
| 0.646 | 7.42E-02 | 1.376 | 4.83E-03 | 2.294 | 3.20E-04 |
| 0.662 | 2.93E-04 | 1.385 | 6.26E-04 | 2.324 | 2.44E-05 |
| 0.709 | 1.35E-02 | 1.437 | 1.22E-02 | 2.455 | 1.47E-05 |
| 0.714 | 2.28E-02 | 1.445 | 3.30E-03 | 2.682 | 1.65E-05 |
| 0.723 | 1.08E-01 | 1.489 | 6.72E-03 | 2.694 | 3.03E-05 |
| 0.736 | 5.57E-04 | 1.526 | 4.09E-03 | 2.808 | 1.47E-05 |
| 0.736 | 7.14E-04 | | | | |

**Table 5.5-19. Sb-125 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | | | |
|---|---|---|---|
| 8.628E-01 | | | |
| **Energy** | **Intensity** | **Energy** | **Intensity** |
| 0.111 | 1.04E-05 | 0.408 | 1.84E-03 |
| 0.117 | 2.63E-03 | 0.428 | 2.96E-01 |
| 0.133 | 8.58E-06 | 0.444 | 3.06E-03 |
| 0.173 | 1.91E-03 | 0.463 | 1.05E-01 |
| 0.176 | 6.84E-02 | 0.490 | 1.36E-05 |
| 0.179 | 3.37E-04 | 0.491 | 4.74E-05 |
| 0.199 | 1.28E-04 | 0.497 | 3.20E-05 |
| 0.204 | 3.17E-03 | 0.503 | 3.85E-05 |
| 0.208 | 2.48E-03 | 0.539 | 1.39E-05 |
| 0.209 | 4.50E-04 | 0.601 | 1.76E-01 |
| 0.228 | 1.31E-03 | 0.607 | 4.98E-02 |
| 0.315 | 4.03E-05 | 0.617 | 5.33E-05 |
| 0.321 | 4.16E-03 | 0.636 | 1.12E-01 |
| 0.332 | 2.52E-05 | 0.653 | 2.66E-05 |
| 0.367 | 7.99E-05 | 0.671 | 1.79E-02 |
| 0.380 | 1.52E-02 | 0.693 | 4.59E-07 |
| 0.402 | 6.22E-05 | | |

**Table 5.5-20. Sb-126 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | | | |
|---|---|---|---|
| 4.304E+00 | | | |
| **Energy** | **Intensity** | **Energy** | **Intensity** |
| 0.149 | 3.98E-03 | 0.667 | 9.96E-01 |
| 0.209 | 4.98E-03 | 0.675 | 3.69E-02 |
| 0.224 | 1.39E-02 | 0.695 | 9.96E-01 |
| 0.278 | 2.39E-02 | 0.697 | 2.89E-01 |
| 0.297 | 4.48E-02 | 0.721 | 5.38E-01 |
| 0.297 | 4.98E-03 | 0.857 | 1.76E-01 |
| 0.415 | 8.33E-01 | 0.954 | 1.20E-02 |
| 0.415 | 9.96E-03 | 0.958 | 4.98E-03 |
| 0.556 | 1.69E-02 | 0.990 | 6.77E-02 |
| 0.574 | 6.67E-02 | 1.036 | 9.96E-03 |
| 0.593 | 7.47E-02 | 1.061 | 3.98E-03 |
| 0.620 | 8.96E-03 | 1.064 | 8.96E-03 |
| 0.639 | 8.96E-03 | 1.213 | 2.39E-02 |
| 0.656 | 2.19E-02 | 1.477 | 2.79E-03 |

**Table 5.5-21. Cs-134 Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 2.228E+00 | |
| **Energy** | **Intensity** |
| 0.243 | 2.72E-04 |
| 0.327 | 1.62E-04 |
| 0.475 | 1.48E-02 |
| 0.563 | 8.34E-02 |
| 0.569 | 1.54E-01 |
| 0.605 | 9.76E-01 |
| 0.796 | 8.55E-01 |
| 0.802 | 8.69E-02 |
| 0.847 | 3.00E-06 |
| 1.039 | 9.90E-03 |
| 1.168 | 1.79E-02 |
| 1.365 | 3.02E-02 |

**Table 5.5-22. Cs-137 (Ba-137m) Gamma Emission Energy Spectrum**

| Total Photons/Disintegration | |
|---|---|
| 8.990E-01 | |
| **Energy** | **Intensity** |
| 0.662 | 8.99E-01 |

### Table 5.5-23. Hf-175 Gamma Emission Energy Spectrum

| Total Photons/Disintegration | |
|---|---|
| 8.683E-01 | |
| **Energy** | **Intensity** |
| 0.114 | 2.94E-03 |
| 0.161 | 2.27E-04 |
| 0.230 | 6.83E-03 |
| 0.319 | 1.68E-03 |
| 0.343 | 8.40E-01 |
| 0.353 | 2.28E-03 |
| 0.433 | 1.44E-02 |

### Table 5.5-24. Hf-181 Gamma Emission Energy Spectrum

| Total Photons/Disintegration | |
|---|---|
| 1.466E+00 | |
| **Energy** | **Intensity** |
| 0.133 | 4.33E-01 |
| 0.136 | 5.85E-02 |
| 0.137 | 8.61E-03 |
| 0.346 | 1.51E-01 |
| 0.476 | 7.03E-03 |
| 0.482 | 8.05E-01 |
| 0.615 | 2.33E-03 |
| 0.619 | 2.50E-04 |

### Table 5.5-25. Ta-182 Gamma Emission Energy Spectrum

| Total Photons/Disintegration | | | | | |
|---|---|---|---|---|---|
| 1.456E+00 | | | | | |
| **Energy** | **Intensity** | **Energy** | **Intensity** | **Energy** | **Intensity** |
| 0.100 | 1.42E-01 | 0.830 | 1.41E-04 | 1.189 | 1.65E-01 |
| 0.110 | 1.07E-03 | 0.892 | 5.74E-04 | 1.221 | 2.72E-01 |
| 0.114 | 1.87E-02 | 0.928 | 6.14E-03 | 1.224 | 2.36E-03 |
| 0.116 | 4.44E-03 | 0.960 | 3.50E-03 | 1.231 | 1.16E-01 |
| 0.122 | 2.36E-05 | 1.002 | 2.09E-02 | 1.257 | 1.51E-02 |
| 0.152 | 7.02E-02 | 1.036 | 6.70E-05 | 1.274 | 6.60E-03 |
| 0.156 | 2.67E-02 | 1.044 | 2.39E-03 | 1.289 | 1.37E-02 |
| 0.179 | 3.12E-02 | 1.113 | 4.45E-03 | 1.343 | 2.57E-03 |
| 0.198 | 1.46E-02 | 1.121 | 3.52E-01 | 1.374 | 2.22E-03 |
| 0.222 | 7.57E-02 | 1.157 | 7.33E-03 | 1.387 | 7.29E-04 |
| 0.229 | 3.64E-02 | 1.158 | 2.89E-03 | 1.410 | 3.96E-04 |
| 0.264 | 3.61E-02 | 1.181 | 8.74E-04 | 1.453 | 3.07E-04 |
| 0.351 | 1.13E-04 | | | | |

### 5.5.3. Cobalt-60 Isotope Rod Activity Distribution

The cobalt-60 isotope rod shielding analysis utilizes 12-inch long line sources, which distribute the activity of the source uniformly across the line. Distribution of the cobalt-60 activity into a longer line source results in a lower external dose rate, and greater concentration of the cobalt-60 activity into a shorter line source results in a higher external dose rate. Thus, in order to demonstrate compliance with the regulatory dose rate limits for the cobalt-60 isotope rod contents, two requirements must be met. First, it must be shown that the dose rate contribution from all cobalt-60 source activity in a single shipment is less than the regulatory limit. Second, it must also be shown that the distribution of the activity in any single shipment of isotope rod [[ ]] is distributed axially, such that the uniform line source used in the shielding analysis is bounding of the actual axial distribution of activity.

For the cobalt-60 isotope rod shielding analysis, there are two source geometries considered. The first source geometry is referred to as the 'bounding' source geometry, which concentrates all of the cobalt-60 activity into a single 12-inch line source that is located in the most restrictive location for dose rate calculations in the given direction (top, side, or bottom). For side dose rate calculations with the bounding source geometry, dose rates are calculated with the source at both the bottom (Case 1) and top (Case 2) of the HPI cavity to determine the bounding source location.

The second geometry, referred to as the 'Realistic' source geometry distributes the cobalt-60 activity into [[ ]] of the HPI material basket. This source geometry provides a more realistic radial distribution of the source, as rod [[ ]] will be distributed throughout the basket during shipment, while still condensing the source axially to 12-inches. The array of line sources for the realistic arrangement is pushed against the top of the HPI for the top dose rate calculations, and is at the bottom of the HPI for bottom and side dose rate calculations. Figure 5.5-1 shows a cross section of the HPI material basket, with locations of the line sources used in the MCNP6 model.

[[

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**Figure 5.5-1. HPI Material Basket with 'Realistic' Source Geometry Locations**

A list of all source geometries analyzed is provided in Table 5.5-26 and a depiction of each source geometry is presented in Figures 5.5-2 through 5.5-8. The 'realistic' dose rate calculations are only included to quantify the margin in the bounding dose rates. For the demonstration of compliance with the normal and hypothetical accident condition dose rate limits, the reported dose rates are based on the more restrictive 'bounding' source geometries.

**Table 5.5-26. Cobalt-60 Isotope Rod Shielding Analysis Case Summary**

| NCT Dose Rate Calculation Locations | Source Arrangement | Source Arrangement Figure |
|---|---|---|
| Bottom Surface | Realistic | 5.5-2 |
| | Bounding | 5.5-3 |
| Top Surface | Realistic | 5.5-4 |
| | Bounding | 5.5-5 |
| Side Surface | Realistic | 5.5-6 |
| | Bounding – 1 | 5.5-7 |
| | Bounding – 2 | 5.5-8 |
| 2-meter | Realistic | 5.5-6 |
| | Bounding – 1 | 5.5-7 |
| | Bounding – 2 | 5.5-8 |
| Cab | Realistic | 5.5-6 |
| | Bounding – 1 | 5.5-7 |
| | Bounding – 2 | 5.5-8 |

[[

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**Figure 5.5-2. 'Realistic' Source Arrangement for Bottom Dose Rates**

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**Figure 5.5-3. 'Bounding' Source Arrangement for Bottom Dose Rates**

[[

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**Figure 5.5-4. 'Realistic' Source Arrangement for Top Dose Rates**

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**Figure 5.5-5. 'Bounding' Source Arrangement for Top Dose Rates**

[[

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**Figure 5.5-6. 'Realistic' Source Arrangement for Side Dose Rates**

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**Figure 5.5-7. 'Bounding' Source Arrangement for Side Dose Rates – Case 1**

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**Figure 5.5-8. 'Bounding' Source Arrangement for Side Dose Rates – Case 2**

Table 5.5-27 lists the peak dose rate per curie calculated at each NCT regulatory dose rate location for each source geometry included in the cobalt-60 isotope rod shielding analysis, and the overall maximum calculated dose rate for each regulatory dose rate location. All maximum calculated dose rates were calculated with the 'bounding' source geometries, where the activity is concentrated into a single line. Also, the line source at the bottom of the HPI cavity calculated higher side dose rates than at the top.

**Table 5.5-27. Cobalt-60 Isotope Rod Shielding Analysis NCT Dose Rate Results**

| NCT Dose Rate Location | Source Arrangement | Dose Rate (mrem/hr/Ci) | Maximum Location Dose Rate (mrem/hr/Ci) |
|---|---|---|---|
| Bottom Surface | Realistic | 3.555E-04 | 3.940E-04 |
| | Bounding | 3.940E-04 | |
| Top Surface | Realistic | 1.780E-04 | 1.850E-04 |
| | Bounding | 1.850E-04 | |
| Side Surface | Realistic | 4.422E-04 | 8.821E-04 |
| | Bounding | 8.821E-04 | |
| | Bounding | 8.168E-04 | |
| 2-meter | Realistic | 1.254E-05 | 1.664E-05 |
| | Bounding | 1.664E-05 | |
| | Bounding | 1.348E-05 | |
| Cab | Realistic | 2.277E-06 | 2.921E-06 |
| | Bounding | 2.921E-06 | |
| | Bounding | 2.331E-06 | |

Table 5.5-28 lists the maximum calculated dose rate at each NCT regulatory dose rate location using the dose rate per curie values calculated in Table 5.5-27 and the total cobalt-60 activity resulting in the dose rate equal to the 90% of regulatory limit for the respective location. This activity is 204,000 Ci, which results in an NCT side surface dose rate of 180 mrem/hr. Although a cobalt-60 activity of 204,000 Ci is not permitted in the Model 2000 cask due to the thermal limit, this table is included to demonstrate at this activity, no regulatory dose rate limits are exceeded.

**Table 5.5-28. Cobalt-60 Isotope Rod Shielding Analysis Maximum NCT Dose Rates**

| Dose Rate Location | Dose Rate per Curie (mrem/hr/Ci) | Dose Rate[a] (mrem/hr) |
|---|---|---|
| Bottom Surface | 3.940E-04 | 80.4 |
| Top Surface | 1.850E-04 | 37.7 |
| Side Surface | 8.821E-04 | 180.0 |
| 2-meter | 1.664E-05 | 3.4 |
| Cab | 2.921E-06 | 0.6 |

Notes: [a] Based on an activity of 204,000 Ci cobalt-60

In Table 5.5-28 it is demonstrated that external dose rates resulting from any cobalt-60 isotope rod activity up to 204,000 Ci are less than the regulatory dose rate limits. It still must be demonstrated that the activity in any single shipment of isotope rod [[ ]] is distributed axially, such that the uniform line source used in the shielding analysis is bounding of the actual axial distribution of activity. The exact axial activity profile of the cobalt-60 isotope rod [[ ]] is variable due to differences in the neutron flux profiles when irradiated in a commercial or research reactor. To determine that the uniform source in the MCNP6 shielding analysis is bounding of the distribution of activity in a shipment of rod [[ ]], it should first be considered that the source geometry is a single 12-inch line source. Modeling the source in this way assumes that all cobalt-60 activity loaded into the cavity is concentrated into a single line, with a uniform distribution. With all activity in a single line at the most restrictive location of the HPI cavity, any radial distribution of activity is bounded. Thus, the only variation in the source distribution that can cause the external dose rates to increase is in one direction (axially). So, it can be demonstrated that the source distribution of the contents in an actual shipment are bounded, by determining that there is no axial location in the HPI cavity where the concentration of activity is greater than what was analyzed in the MCNP6 analysis.

With the source arrangement of a single 12-inch line, no external dose rates will exceed the regulatory limits for any activity up to 204,000 Ci. By dividing the activity of 204,000 Ci evenly across the 12-inch uniform MCNP6 source, the result is a source that is concentrated in the axial direction to an activity of 17,000 Ci in each inch of the line source. Thus, it can be demonstrated that the activity distribution in the MCNP6 shielding model bounds the total activity distribution in the HPI cavity by determining for the package contents, the total activity in any axial 1-inch increment of the HPI cavity is not greater than 17,000 Ci. If for an actual shipment, there is a total activity of less than or equal to (≤) 17,000 Ci in any axial 1-inch increment of the HPI cavity, there is a greater distribution of the activity in the contents than in the MCNP6 source and the MCNP6 source is bounding.

### 5.5.4. Radionuclide Decay Heat Conversion Factors

In addition to demonstrating compliance with the regulatory dose rate requirements, filling out the Irradiated Hardware and Byproduct Loading Table also demonstrates compliance with the thermal limit of the Model 2000 cask. One characteristic of every radionuclide is a given Q-value, which is the quantity of energy emitted per decay (MeV/Decay). By assuming that all energy emitted is deposited locally in the HPI material basket or the HPI body, the radionuclide decay heat in W/Ci can be calculated. All radionuclides considered in the irradiated hardware and byproduct contents are listed in Table 5.5-7 of Section 5.5.2, regardless of their significance to dose rate calculations. The Q-values for each of these radionuclides are provided in SCALE6.1 ORIGEN-S Decay library origen.rev03.decay.data (Reference 5-2). Table 5.5-29 lists all of the irradiated hardware and by-product radionuclides with their ORIGEN-S library identification number, Q-value, and the calculated decay heat. Radionuclide Q-values are converted to decay heat values as shown in Equation 5-12.

$$\text{Decay Heat}\left[\frac{\text{W}}{\text{Ci}}\right] = \text{Q}\left[\frac{\text{MeV}}{\text{disintegration}}\right] \cdot 1.60217 \cdot 10^{-13}\left[\frac{\text{J}}{\text{MeV}}\right] \cdot 3.7 \cdot 10^{10}\left[\frac{\frac{\text{disintegrations}}{\text{s}}}{\text{Ci}}\right] \quad (5\text{-}12)$$

**Table 5.5-29. Isotope Decay Heat Data**

| Isotope | ORIGEN-S Radionuclide ID | Q-Value (MeV/Decay) | Decay Heat (W/Ci) |
|---|---|---|---|
| H-3 | 10030 | 5.6900E-03 | 3.373E-05 |
| C-14 | 60140 | 4.9470E-02 | 2.933E-04 |
| Na-24 | 110240 | 4.6769E+00 | 2.772E-02 |
| Si-31 | 140310 | 5.9645E-01 | 3.536E-03 |
| P-32 | 150320 | 6.9490E-01 | 4.119E-03 |
| P-33 | 150330 | 7.6430E-02 | 4.531E-04 |
| S-35 | 160350 | 4.8758E-02 | 2.890E-04 |
| Ca-45 | 200450 | 7.6860E-02 | 4.556E-04 |
| Sc-46 | 210460 | 2.1214E+00 | 1.258E-02 |
| Sc-47 | 210470 | 2.7132E-01 | 1.608E-03 |
| Sc-48 | 210480 | 3.5737E+00 | 2.118E-02 |
| V-49 | 230490 | 4.4514E-03 | 2.639E-05 |
| V-52 | 230520 | 2.5137E+00 | 1.490E-02 |
| Cr-51 | 240510 | 3.6680E-02 | 2.174E-04 |
| Cr-55 | 240550 | 1.1017E+00 | 6.531E-03 |
| Mn-54 | 250540 | 8.4017E-01 | 4.981E-03 |
| Mn-56 | 250560 | 2.5226E+00 | 1.495E-02 |
| Co-57 | 270570 | 1.4380E-01 | 8.525E-04 |
| Co-58 | 270580 | 1.0088E+00 | 5.980E-03 |
| Co-58m | 270581 | 2.4744E-02 | 1.467E-04 |
| Co-60 | 270600 | 2.6006E+00 | 1.542E-02 |
| Co-60m | 270601 | 6.3045E-02 | 3.737E-04 |
| Co-61 | 270610 | 5.6391E-01 | 3.343E-03 |
| Ni-57 | 280570 | 2.0927E+00 | 1.241E-02 |
| Ni-59 | 280590 | 6.9156E-03 | 4.100E-05 |
| Ni-63 | 280630 | 1.7425E-02 | 1.033E-04 |
| Ni-65 | 280650 | 1.1863E+00 | 7.032E-03 |
| Fe-55 | 260550 | 5.8421E-03 | 3.463E-05 |
| Fe-59 | 260590 | 1.3060E+00 | 7.742E-03 |
| Cu-64 | 290640 | 3.1188E-01 | 1.849E-03 |
| Cu-66 | 290660 | 1.1645E+00 | 6.903E-03 |
| Zn-65 | 300650 | 5.8284E-01 | 3.455E-03 |
| Zr-89 | 400890 | 3.5256E-01 | 2.090E-03 |
| Zr-95 | 400950 | 8.5013E-01 | 9.835E-03[b] |
| Zr-97 | 400970 | 8.6426E-01 | 5.123E-03 |
| Sr-87m | 380871 | 3.8798E-01 | 2.300E-03 |
| Sr-89 | 380890 | 5.8534E-01 | 3.470E-03 |
| Sr-90 | 380900 | 1.9580E-01 | 1.161E-03 |
| Sr-91 | 380910 | 1.3485E+00 | 7.994E-03 |
| Y-89m | 390891 | 9.0902E-01 | 5.389E-03 |
| Y-90 | 390900 | 9.3302E-01 | 5.531E-03 |
| Y-90m | 390901 | 6.8000E-01 | 4.031E-03 |
| Y-91 | 390910 | 6.0617E-01 | 3.593E-03 |
| Y-91m | 390911 | 5.5554E-01 | 3.293E-03 |
| Y-92 | 390920 | 1.7017E+00 | 1.009E-02 |
| Nb-91m | 410911 | 1.2634E-01 | 7.489E-04 |
| Nb-92m | 410921 | 9.7526E-01 | 5.781E-03 |
| Nb-94 | 410940 | 1.7599E+00 | 1.043E-02 |
| Nb-95 | 410950 | 8.0900E-01 | 4.796E-03 |

| Isotope | ORIGEN-S Radionuclide ID | Q-Value (MeV/Decay) | Decay Heat (W/Ci) |
|---|---|---|---|
| Tc-101 | 431010 | 8.1600E-01 | 4.837E-03 |
| Ru-106 | 441060 | 1.0030E-02 | 5.946E-05 |
| In-113m | 491131 | 3.9159E-01 | 2.321E-03 |
| In-114 | 491140 | 7.7607E-01 | 4.601E-03 |
| In-114m | 491141 | 2.2277E-01 | 1.321E-03 |
| In-115m | 491151 | 3.3436E-01 | 1.982E-03 |
| Sn-113 | 501130 | 2.9753E-02 | 1.764E-04 |
| Sn-113m | 501131 | 7.1749E-02 | 4.253E-04 |
| | | | |
| Sn-117m | 501171 | 3.1563E-01 | 1.871E-03 |
| | | | |
| Sn-119m | 501191 | 8.7589E-02 | 5.192E-04 |
| Sn-121 | 501210 | 1.1582E-01 | 6.866E-04 |
| Sn-121m | 501211 | 3.7987E-02 | 2.252E-04 |
| Sn-123 | 501230 | 5.3006E-01 | 3.142E-03 |
| Sn-123m | 501231 | 6.2147E-01 | 3.684E-03 |
| Sn-125 | 501250 | 1.1357E+00 | 6.732E-03 |
| Sb-122 | 511220 | 1.0098E+00 | 5.986E-03 |
| Sb-124 | 511240 | 2.2351E+00 | 1.325E-02 |
| Sb-125 | 511250 | 5.3352E-01 | 3.163E-03 |
| Sb-126 | 511260 | 3.1205E+00 | 1.850E-02 |
| Te-125m | 521251 | 1.4546E-01 | 8.623E-04 |
| I-129 | 531290 | 7.4338E-02 | 4.407E-04 |
| Cs-134 | 551340 | 1.7185E+00 | 1.019E-02 |
| Cs-137 | 551370 | 1.7945E-01 | 4.985E-03[a] |
| Ba-137m | 561371 | 6.6140E-01 | |
| La-140 | 571400 | 2.8438E+00 | 1.686E-02 |
| Ba-140 | 561400 | 5.0041E-01 | 2.966E-03 |
| Ce-144 | 581440 | 1.1059E-01 | 6.556E-04 |
| Hf-173 | 721730 | 4.4558E-01 | 2.641E-03 |
| Hf-175 | 721750 | 3.9728E-01 | 2.355E-03 |
| Hf-177m | 721771 | 1.5190E+00 | 9.005E-03 |
| Hf-180m | 721801 | 1.1148E+00 | 6.609E-03 |
| Hf-181 | 721810 | 7.3010E-01 | 4.328E-03 |
| Lu-173 | 711730 | 2.3057E-01 | 1.367E-03 |
| Lu-174 | 711740 | 1.5804E-01 | 9.369E-04 |
| Lu-174m | 711741 | 1.6712E-01 | 9.907E-04 |
| Lu-176m | 711761 | 4.9032E-01 | 2.907E-03 |
| Lu-177 | 711770 | 1.8133E-01 | 1.075E-03 |
| Lu-177m | 711771 | 2.4764E-01 | 1.468E-03 |
| Yb-175 | 701750 | 2.0070E-01 | 1.190E-03 |
| Yb-177 | 701770 | 6.2579E-01 | 3.710E-03 |
| Ta-180 | 731800 | 1.0251E-01 | 6.077E-04 |
| Ta-182 | 731820 | 1.5156E+00 | 8.985E-03 |
| Ta-183 | 731830 | 6.3433E-01 | 3.760E-03 |
| W-181 | 741810 | 5.1849E-02 | 3.074E-04 |
| W-183m | 741831 | 2.9876E-01 | 1.771E-03 |
| W-185 | 741850 | 1.2690E-01 | 7.523E-04 |
| Re-186 | 751860 | 3.5696E-01 | 2.116E-03 |

| Isotope | ORIGEN-S Radionuclide ID | Q-Value (MeV/Decay) | Decay Heat (W/Ci) |
|---|---|---|---|
| Nb-96 | 410960 | 2.7140E+00 | 1.609E-02 |
| Nb-95m | 410951 | 2.4933E-01 | 1.478E-03 |
| Nb-97 | 410970 | 1.1330E+00 | 6.716E-03 |
| Nb-97m | 410971 | 7.4336E-01 | 4.407E-03 |
| Mo-93 | 420930 | 1.6143E-02 | 9.570E-05 |
| Mo-93m | 420931 | 2.4158E+00 | 1.432E-02 |
| Mo-99 | 420990 | 5.4317E-01 | 3.220E-03 |
| Mo-101 | 421010 | 1.9735E+00 | 1.170E-02 |
| Tc-99 | 430990 | 5.5202E-02 | 3.272E-04 |
| Tc-99m | 430991 | 1.4222E-01 | 8.431E-04 |

| Isotope | ORIGEN-S Radionuclide ID | Q-Value (MeV/Decay) | Decay Heat (W/Ci) |
|---|---|---|---|
| Np-237 | 932370 | 4.9445E+00 | 2.931E-02 |
| Pu-238 | 942380 | 5.5899E+00 | 3.314E-02 |
| Pu-239 | 942390 | 5.2433E+00 | 3.108E-02 |
| Pu-240 | 942400 | 5.2522E+00 | 3.114E-02 |
| Pu-241 | 942410 | 5.3555E-03 | 3.175E-05 |
| Am-241 | 952410 | 5.6280E+00 | 3.336E-02 |
| Cm-242 | 962420 | 6.2153E+00 | 3.684E-02 |
| Cm-243 | 962430 | 6.1779E+00 | 3.662E-02 |
| Cm-244 | 962440 | 5.9011E+00 | 3.498E-02 |

Notes: [a] Combined decay heat for Cs-137 and Ba-137m
[b] Decay heat calculated using summed Q-values from Zr-95 and Nb-95.

### 5.5.5. Irradiated Fuel Loading Table

In order to demonstrate compliance with the 10 CFR 71 (Reference 5-1) regulatory dose rate limits as well as the thermal and criticality limits of the cask, the Irradiated Fuel Loading Table (Section 7.5.3) must be filled out for every shipment of irradiated fuel in the Model 2000 cask. An example irradiated fuel dose rate calculation for a hypothetical shipment consisting of three irradiated fuel rods is provided. Recall from Section 5.2 that the fuel rod cladding is treated separately as irradiated hardware. An example irradiated hardware dose rate calculation is provided in Section 5.5.6. The relevant information for each of the example fuel rods is presented in Table 5.5-30.

**Table 5.5-30 Hypothetical Irradiated Fuel Rod Shipment Information**

| Rod # | Active Length (cm) | Segments (#) | Fuel Rod Radius (cm) | Total Mass (gU) | Minimum Initial Enrichment[3] (wt% U-235) | Total Mass (gU235) | Maximum Burnup[3] (GWd/MTU) |
|---|---|---|---|---|---|---|---|
| 1 | 381.0 | 10 | 0.4380 | 2218[1] | 3.100 | 68.77[2] | 46 |
| 2 | 381.0 | 13 | 0.4380 | 2218[1] | 2.600 | 57.68[2] | 39 |
| 3 | 381.0 | 10 | 0.4380 | 2218[1] | 4.200 | 93.17[2] | 58 |

Notes: [1] Based on 10.96 g/cm$^3$ $UO_2$ density and approximation of $mU/mUO_2 = 238/(238+2\times16)$
[2] Calculated based on initial enrichment
[3] Uniform initial enrichment and burnup assumed for each of the hypothetical fuel rods

An example dose rate calculation is completed for the NCT side surface location in Table 5.5-31. The total initial mass (in gU235) is from Table 5.5-30 and the side surface dose rate (in mrem/hr/gU235) for the defined example enrichment and burnup is taken from Table 5.4-4. The total dose rate from each fuel rod is calculated by multiplying the initial mass of U-235 by the dose rate per gU235.

**Table 5.5-31. Irradiated Fuel NCT Side Surface Dose Rate Calculation**

| Fuel Rod # | m (gU235) | $\dot{D}\dot{R}$ (mrem/hr/gU235) | DR (mrem/hr) |
|---|---|---|---|
| 1 | 68.77 | 3.215E-01 | 22.11 |
| 2 | 57.68 | 2.480E-01 | 14.30 |
| 3 | 93.17 | 2.945E-01 | 27.44 |

By summing the total dose rate contribution from each fuel rod, the total dose rate at the NCT side surface locations is calculated to be 63.85 mrem/hr. By repeating this calculation for each regulatory dose rate location, it can be demonstrated that this hypothetical group of irradiated fuel rods is acceptable for shipment, as the calculated dose rates do not exceed the regulatory limits. Using the hypothetical group of irradiated fuel rods in Table 5.5-30, an example of the complete loading table filled out for these contents is provided in Table 5.5-32. For this loading table example, it is assumed that each of the fuel rods is segmented to even whole lengths, with the mass of the fuel rod being divided evenly as well. It is demonstrated in this table that the criticality, dose rate, and thermal limit is not exceeded, and the hypothetical fuel rods are acceptable for shipment.

**Table 5.5-32. Hypothetical Irradiated Fuel Rod Shipment Irradiated Fuel Loading Table**

| Segment # | Segment Length (inches) | Initial Enrichment (wt% U-235) | Burnup (GWd/MTU) | Mass U-235 (g) | Decay Heat (W) | NCT | | | | | HAC | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | $DR_{surf}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | | | | | Top | Side | Bottom | | | Top | Side | Bottom |
| 1 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 2 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 3 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 4 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 5 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 6 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 7 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 8 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 9 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 10 | 15 | 3.0≤ E <3.5 | 40< B ≤50 | 6.877 | 7.159 | 0.3846 | 2.211 | 1.279 | 0.0522 | 0.0093 | 0.3491 | 0.4416 | 0.3611 |
| 11 | 11 | 2.5≤ E <3.0 | 30< B ≤40 | 4.230 | 4.852 | 0.1823 | 1.049 | 0.5985 | 0.0245 | 0.0044 | 0.1759 | 0.2408 | 0.1749 |
| 12 | 11 | 2.5≤ E <3.0 | 30< B ≤40 | 4.230 | 4.852 | 0.1823 | 1.049 | 0.5985 | 0.0245 | 0.0044 | 0.1759 | 0.2408 | 0.1749 |
| 13 | 11 | 2.5≤ E <3.0 | 30< B ≤40 | 4.230 | 4.852 | 0.1823 | 1.049 | 0.5985 | 0.0245 | 0.0044 | 0.1759 | 0.2408 | 0.1749 |
| 14 | 11 | 2.5≤ E <3.0 | 30< B ≤40 | 4.230 | 4.852 | 0.1823 | 1.049 | 0.5985 | 0.0245 | 0.0044 | 0.1759 | 0.2408 | 0.1749 |
| 15 | 11 | 2.5≤ E <3.0 | 30< B ≤40 | 4.230 | 4.852 | 0.1823 | 1.049 | 0.5985 | 0.0245 | 0.0044 | 0.1759 | 0.2408 | 0.1749 |
| 16 | 11 | 2.5≤ E <3.0 | 30< B ≤40 | 4.230 | 4.852 | 0.1823 | 1.049 | 0.5985 | 0.0245 | 0.0044 | 0.1759 | 0.2408 | 0.1749 |
| 17 | 12 | 2.5≤ E <3.0 | 30< B ≤40 | 4.614 | 5.293 | 0.1988 | 1.144 | 0.6529 | 0.0267 | 0.0048 | 0.1919 | 0.2627 | 0.1908 |
| 18 | 12 | 2.5≤ E <3.0 | 30< B ≤40 | 4.614 | 5.293 | 0.1988 | 1.144 | 0.6529 | 0.0267 | 0.0048 | 0.1919 | 0.2627 | 0.1908 |
| 19 | 12 | 2.5≤ E <3.0 | 30< B ≤40 | 4.614 | 5.293 | 0.1988 | 1.144 | 0.6529 | 0.0267 | 0.0048 | 0.1919 | 0.2627 | 0.1908 |
| 20 | 12 | 2.5≤ E <3.0 | 30< B ≤40 | 4.614 | 5.293 | 0.1988 | 1.144 | 0.6529 | 0.0267 | 0.0048 | 0.1919 | 0.2627 | 0.1908 |
| 21 | 12 | 2.5≤ E <3.0 | 30< B ≤40 | 4.614 | 5.293 | 0.1988 | 1.144 | 0.6529 | 0.0267 | 0.0048 | 0.1919 | 0.2627 | 0.1908 |
| 22 | 12 | 2.5≤ E <3.0 | 30< B ≤40 | 4.614 | 5.293 | 0.1988 | 1.144 | 0.6529 | 0.0267 | 0.0048 | 0.1919 | 0.2627 | 0.1908 |
| 23 | 12 | 2.5≤ E <3.0 | 30< B ≤40 | 4.614 | 5.293 | 0.1988 | 1.144 | 0.6529 | 0.0267 | 0.0048 | 0.1919 | 0.2627 | 0.1908 |
| 24 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| 25 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| 26 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| 27 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| 28 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| 29 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| 30 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| 31 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| 32 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| 33 | 15 | 4.0≤ E <4.5 | 50< B ≤60 | 9.317 | 7.710 | 0.4777 | 2.744 | 1.595 | 0.0651 | 0.0116 | 0.4246 | 0.5217 | 0.4453 |
| Min? | 11 | | Total | 219.6 | 214.9 | 11.11 | 63.85 | 36.90 | 1.507 | 0.2697 | 10.14 | 12.92 | 10.45 |
| Limit | 5.3 | | Limit | 1750 | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| Criteria Met? | YES | | Criteria Met? | YES | YES | YES | YES | YES | YES | YES | YES | YES | YES |

For every burnup-enrichment pairing, the total allowable mass of U-235 is restricted by either the 1750 gU235 criticality limit, the mass of U-235 corresponding to the thermal limit of 1500 W, or the maximum mass of U-235 corresponding to a dose rate of 180 mrem/hr at the NCT side surface dose location. Tables 5.5-33 and 5.5-34 list the maximum mass of U-235 for each burnup-enrichment pairing corresponding to the NCT side surface dose rate limit and the 1500 W thermal limit, respectively. Table 5.5-35 provides the overall maximum allowable mass of U-235 for each burnup-enrichment pairing in the Model 2000 cask, by listing the minimum value for each pairing between those in Table 5.5-33, Table 5.5-34, and the 1750 g U-235 criticality limit.

**Table 5.5-33. Maximum Allowable Mass (g) of U-235 Based on NCT Side Surface Dose Rate**

| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
|---|---|---|---|---|---|---|---|
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 5154.80 | 1628.96 | 602.82 | 285.17 | 159.76 | 97.54 | 55.74 |
| 2.0≤ E <2.5 | 7344.89 | 2662.49 | 1019.96 | 471.55 | 257.50 | 155.33 | 88.83 |
| 2.5≤ E <3.0 | 9575.63 | 3890.57 | 1578.15 | 725.73 | 388.32 | 231.30 | 132.21 |
| 3.0≤ E <3.5 | 11820.69 | 5254.21 | 2280.34 | 1059.68 | 559.96 | 329.61 | 188.16 |
| 3.5≤ E <4.0 | 14050.91 | 6706.34 | 3123.76 | 1483.58 | 780.06 | 454.72 | 258.88 |
| 4.0≤ E <4.5 | 16278.51 | 8210.98 | 4086.65 | 2000.19 | 1054.98 | 611.18 | 346.98 |
| 4.5≤ E <5.0 | 18492.68 | 9743.59 | 5139.57 | 2608.01 | 1389.28 | 803.02 | 454.65 |
| 5.0≤ E <5.5 | 20704.75 | 11289.89 | 6266.96 | 3303.69 | 1786.40 | 1033.98 | 584.59 |
| 5.5≤ E <6.0 | 22901.28 | 12846.17 | 7454.68 | 4082.72 | 2248.41 | 1307.77 | 739.37 |

**Table 5.5-34. Maximum Allowable Mass (g) of U-235 Based on Thermal Limit**

| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
|---|---|---|---|---|---|---|---|
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 1378.68 | 1032.58 | 868.06 | 764.27 | 693.59 | 642.49 | 597.13 |
| 2.0≤ E <2.5 | 1825.93 | 1381.85 | 1170.50 | 1033.41 | 937.21 | 866.80 | 803.86 |
| 2.5≤ E <3.0 | 2271.35 | 1731.30 | 1476.38 | 1307.99 | 1186.71 | 1096.17 | 1014.61 |
| 3.0≤ E <3.5 | 2714.11 | 2080.44 | 1785.01 | 1586.74 | 1440.92 | 1330.18 | 1229.51 |
| 3.5≤ E <4.0 | 3156.95 | 2430.56 | 2094.97 | 1868.99 | 1700.13 | 1569.51 | 1448.28 |
| 4.0≤ E <4.5 | 3597.12 | 2780.35 | 2406.74 | 2154.40 | 1962.71 | 1812.69 | 1671.31 |
| 4.5≤ E <5.0 | 4037.08 | 3129.35 | 2718.49 | 2441.23 | 2229.19 | 2059.81 | 1898.20 |
| 5.0≤ E <5.5 | 4474.94 | 3480.28 | 3030.30 | 2729.26 | 2498.33 | 2310.54 | 2129.47 |
| 5.5≤ E <6.0 | 4913.64 | 3830.08 | 3344.14 | 3018.66 | 2769.39 | 2564.50 | 2363.90 |

**Table 5.5-35. Overall Maximum Allowable Mass (g) of U-235 Based on All Cask Limits**

| Enrichment (wt% U-235) | Burnup (GWd/MTU) | | | | | | |
|---|---|---|---|---|---|---|---|
| | 0< B ≤10 | 10< B ≤20 | 20< B ≤30 | 30< B ≤40 | 40< B ≤50 | 50< B ≤60 | 60< B ≤72 |
| 1.5≤ E <2.0 | 1378.68 | 1032.58 | 602.82 | 285.17 | 159.76 | 97.54 | 55.74 |
| 2.0≤ E <2.5 | 1750.00 | 1381.85 | 1019.96 | 471.55 | 257.50 | 155.33 | 88.83 |
| 2.5≤ E <3.0 | 1750.00 | 1731.30 | 1476.38 | 725.73 | 388.32 | 231.30 | 132.21 |
| 3.0≤ E <3.5 | 1750.00 | 1750.00 | 1750.00 | 1059.68 | 559.96 | 329.61 | 188.16 |
| 3.5≤ E <4.0 | 1750.00 | 1750.00 | 1750.00 | 1483.58 | 780.06 | 454.72 | 258.88 |
| 4.0≤ E <4.5 | 1750.00 | 1750.00 | 1750.00 | 1750.00 | 1054.98 | 611.18 | 346.98 |
| 4.5≤ E <5.0 | 1750.00 | 1750.00 | 1750.00 | 1750.00 | 1389.28 | 803.02 | 454.65 |
| 5.0≤ E <5.5 | 1750.00 | 1750.00 | 1750.00 | 1750.00 | 1750.00 | 1033.98 | 584.59 |
| 5.5≤ E <6.0 | 1750.00 | 1750.00 | 1750.00 | 1750.00 | 1750.00 | 1307.77 | 739.37 |

| Overall Maximum Allowable Mass (g) of U-235 Based on All Cask Limits<br>Table Legend | |
|---|---|
| (yellow) | limited by thermal limit 1500 W |
| (green) | limited by criticality mass of U-235 |
| (blue) | limited by 90% of regulatory dose rate limit 180 mrem/hr |

### 5.5.6. Irradiated Hardware and Byproduct Loading Table

In order to demonstrate compliance with the 10 CFR 71 (Reference 5-1) regulatory dose rate limits and the thermal limit of the cask, the Irradiated Hardware and Byproduct Loading Table must be confirmed for every shipment of irradiated hardware or byproducts in the Model 2000 cask. The use of this loading table is simple: for each of the radionuclides included in a shipment, enter the radionuclide into the table, enter the activity of the radionuclide, then calculate the decay heat and dose rate contribution at each regulatory location based on the dose rate per curie and decay heat values presented in Tables 5.4-12, 5.4-13 and 5.5-29.

Tables 5.5-36 through 5.5-38 provide radionuclide inventories for three hypothetical shipments of irradiated hardware, zirconium-95, and hafnium poison rods. The irradiated hardware radionuclide inventory presented in Table 5.5-39 lists the sample activities and percent-activity of the total content for a list of radionuclides based on a previous shipment of a piece of irradiated 304 SS in the Model 2000 cask with all of the radionuclide activities scaled up to higher activities. The zirconium and hafnium poison rod radionuclide inventories in Tables 5.5-40 and 5.5-41 are hypothetical radionuclide inventories, included only to provide additional examples.

**Table 5.5-36. Example Irradiated SS304 Radionuclide Inventory**

| Nuclide | Ci/sample | % Activity | Total Activity (Ci) |
|---|---|---|---|
| H-3 | 6.75E-06 | 0.00% | 4.605E-05 |
| P-32 | 3.49E-04 | 0.00% | 2.381E-03 |
| S-35 | 8.26E-04 | 0.00% | 5.635E-03 |
| Cr-51 | 3.73E+00 | 2.12% | 2.545E+01 |
| Mn-54 | 5.60E+00 | 3.18% | 3.821E+01 |
| Fe-55 | 5.69E+01 | 32.35% | 3.882E+02 |
| Fe-59 | 3.32E-01 | 0.19% | 2.265E+00 |
| Co-58 | 2.85E+00 | 1.62% | 1.944E+01 |
| Co-60 | 1.01E+02 | 57.42% | 6.891E+02 |
| Ni-59 | 4.14E-02 | 0.02% | 2.825E-01 |
| Ni-63 | 5.42E+00 | 3.08% | 3.698E+01 |
| Zn-65 | 1.06E-02 | 0.01% | 7.232E-02 |
| Nb-93m | 3.25E-04 | 0.00% | 2.217E-03 |
| Mo-99 | 6.13E-14 | 0.00% | 4.182E-13 |
| Tc-99m | 5.94E-14 | 0.00% | 4.053E-13 |
| Total | 175.89 | 100.00% | 1200 |

**Table 5.5-37. Example Zr-95 Radionuclide Inventory**

| Nuclide | Total Activity (Ci) |
|---|---|
| Zr-95 | 80,000 |

**Table 5.5-38. Example Hf Poison Rod Radionuclide Inventory**

| Nuclide | % Activity | Total Activity (Ci) |
|---|---|---|
| Hf-175 | 4.21% | 10,650 |
| Hf-181 | 90.09% | 228,000 |
| Ta-182 | 5.70% | 14,422 |
| Total | 100.00% | 253,072 |

Tables 5.5-39 through 5.5-41 show the respective Irradiated Hardware and Byproduct Loading Tables for each of the hypothetical shipments outlined in Tables 5.5-36 through 5.5-38. These tables show that all three hypothetical shipments of irradiated hardware and byproduct contents comply with all dose rate and thermal criteria and would be acceptable for shipment.

**Table 5.5-39. Example SS304 Irradiated Hardware and Byproduct Loading Table**

| Radio-nuclide | Activity (Ci) | Decay Heat (W) | NCT | | | | | HAC | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | $DR_{surf}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | | Top | Side | Bottom | | | Top | Side | Bottom |
| Cr-51 | 25.40 | 5.53E-03 | 1.01E-13 | 4.19E-19 | 1.41E-18 | 5.41E-21 | 8.93E-22 | 3.23E-12 | 1.28E-18 | 2.41E-17 |
| Mn-54 | 38.20 | 1.90E-01 | 8.77E-06 | 3.77E-04 | 1.63E-05 | 4.26E-06 | 7.27E-07 | 1.55E-05 | 7.82E-05 | 1.27E-05 |
| Fe-55 | 388.20 | 1.34E-02 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| Fe-59 | 2.30 | 1.75E-02 | 2.81E-04 | 6.56E-03 | 4.20E-04 | 8.27E-05 | 1.43E-05 | 4.09E-04 | 1.22E-03 | 2.43E-04 |
| Co-58 | 19.40 | 1.16E-01 | 2.64E-04 | 4.75E-03 | 4.22E-04 | 6.56E-05 | 1.16E-05 | 3.49E-04 | 8.67E-04 | 2.25E-04 |
| Co-60 | 689.10 | 1.06E+01 | 2.88E-01 | 6.50E+00 | 4.30E-01 | 8.34E-02 | 1.43E-02 | 4.14E-01 | 1.20E+00 | 2.45E-01 |
| Ni-63 | 37.00 | 3.82E-03 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| **Total** | - | 10.970 | 0.289 | 6.514 | 0.431 | 0.084 | 0.014 | 0.415 | 1.203 | 0.245 |
| **Limit** | - | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| **Criteria Met?** | - | YES | YES | YES | YES | YES | YES | YES | YES | YES |

**Table 5.5-40. Example Zr-95 Irradiated Hardware and Byproduct Loading Table**

| Radionuclide | Activity (Ci) | Decay Heat (W) | NCT | | | | | HAC | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | $DR_{surf}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | | Top | Side | Bottom | | | Top | Side | Bottom |
| Zr-95 | 80,000 | 787.0 | 4.01E-03 | 2.00E-01 | 9.45E-03 | 2.25E-03 | 3.83E-04 | 7.62E-03 | 4.37E-02 | 8.31E-03 |
| **Total** | - | 787.0 | 4.01E-03 | 2.00E-01 | 9.45E-03 | 2.25E-03 | 3.83E-04 | 7.62E-03 | 4.37E-02 | 8.31E-03 |
| **Limit** | - | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| **Criteria Met?** | - | YES | YES | YES | YES | YES | YES | YES | YES | YES |

**Table 5.5-41. Example Hf Poison Rod Irradiated Hardware and Byproduct Loading Table**

| Radionuclide | Activity (Ci) | Decay Heat (W) | NCT | | | | | HAC | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | $DR_{surf}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | | Top | Side | Bottom | | | Top | Side | Bottom |
| Hf-175 | 10,650 | 25.1 | 1.75E-09 | 2.04E-11 | 3.74E-12 | 1.24E-13 | 2.01E-14 | 1.78E-08 | 5.13E-12 | 8.25E-12 |
| Hf-181 | 228,000 | 986.8 | 4.49E-06 | 4.82E-06 | 7.81E-07 | 6.27E-08 | 1.08E-08 | 3.40E-05 | 1.57E-06 | 1.16E-06 |
| Ta-182 | 14,422 | 129.6 | 1.94E+00 | 4.72E+01 | 2.88E+00 | 5.90E-01 | 1.01E-01 | 2.85E+00 | 8.85E+00 | 1.70E+00 |
| **Total** | - | 1141.5 | 1.94 | 47.20 | 2.88 | 0.59 | 0.10 | 2.85 | 8.85 | 1.70 |
| **Limit** | - | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| **Criteria Met?** | - | YES | YES | YES | YES | YES | YES | YES | YES | YES |

### 5.5.7. Combined Content Shipments

There is the possibility of a shipment that includes multiple content types. To demonstrate compliance with all regulatory and cask requirements, the total thermal power and dose rate contributions from each content type must be determined. Using the procedure in Section 7.5.4, compliance is demonstrated for shipments of multiple content types. For illustration purposes, a shipment of mixed content that combines both irradiated fuel and hardware and byproduct is used as an example. Both the Irradiated Fuel Loading Table and the Irradiated Hardware and Byproduct Loading Table must be filled out for the respective radioactive contents. Then using the Combined Contents Loading Table in Section 7.5.4, the dose rate and thermal power contributions for each are summed, calculating the total thermal power and external dose rates for the shipment.

An example of this process can be demonstrated by using the hypothetical shipment of fuel rods introduced in Section 5.5.5 (see Table 5.5-30) and the example irradiated SS304 radionuclide inventory in Section 5.5.6 (see Table 5.5-36). For this hypothetical shipment of irradiated fuel and hardware, the Irradiated Fuel Loading Table is filled out (see Table 5.5-32) and the Irradiated Hardware and Byproduct Loading Table (see Table 5.5-39) are filled out for the respective contents. Then by following the procedure in Section 7.5.4, the Combined Contents Loading Table is filled out for this shipment, as shown in Table 5.5-42. Based on the total thermal power and external dose rates calculated in Table 5.5-42, this hypothetical shipment of irradiated fuel and hardware is acceptable for shipment in the Model 2000 cask.

**Table 5.5-42. Example Combined Contents Loading Table**

| Content | Thermal Power (W) | NCT | | | | | HAC | | |
|---|---|---|---|---|---|---|---|---|---|
| | | $DR_{surf}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | Top | Side | Bottom | | | Top | Side | Bottom |
| **Fuel** | 214.9 | 11.11 | 63.85 | 36.90 | 1.507 | 0.2697 | 10.14 | 12.92 | 10.45 |
| **Hardware / Byproduct** | 10.97 | 0.289 | 6.514 | 0.431 | 0.084 | 0.014 | 0.415 | 1.203 | 0.245 |
| **Cobalt-60 Isotope Rods** | - | - | - | - | - | - | - | - | - |
| **Total** | 225.87 | 11.399 | 70.364 | 37.331 | 1.591 | 0.2837 | 10.555 | 14.123 | 10.695 |
| **Limit** | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| **Criteria Met?** | YES | YES | YES | YES | YES | YES | YES | YES | YES |

For other combinations of contents, the Loading Table of each content type must be completed and the total contribution to the thermal power and dose rates must be confirmed to be below the limit. Additionally, the requirements for each content type that are defined in Section 1.2.2 must be met.

## 5.6 References

5-1 U.S. NRC, 10 CFR 71, "Packaging and Transportation of Radioactive Material," Washington D.C.

5-2 Oak Ridge National Lab, "SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design, ORNL/TM-2005/39, Version 6, Vols. I-III," ORNL/TM-2005/39, Version 6.1, June 2011.

5-3 American Society for Testing and Materials, "Standard Specification for General Requirements for Flat-Rolled Stainless and Heat-Resisting Steel Plate, Sheet, and Strip," ASTM A480, 2016.

5-4 R.J. McConn et al., "Compendium of Material Composition Data for Radiation Transport Modeling," PNNL-15870, Revision 1, March 2011.

5-5 T. Goorley, et al., "Initial MCNP Release Overview - MCNP6 Version 1.0," Los Alamos National Laboratory, LA-UR-13-22934, April 2013.

5-6 J. Conlin et al., "Listing of Available ACE Data Tables," Los Alamos National Laboratory, LA-UR-13-21822, Revision 4, June 2014.

5-7 U.S. Nuclear Ragulatory Commission, "Standard Review Plan for Transportation Packages for Radioactive Material," NUREG-1609, March 1999.

5-8 ANS 6.1.1 Working Group, M. E. Battat (Chairman), "American National Standard Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," American Nuclear Society, ANSI/ANS-6.1.1-1977, March 1977

5-9 L. C. Leal et al., "ARP: Automatic Rapid Process for the Generation of Problem-Dependent SAS2H/ORIGEN-S Cross-Section Libraries," ORNL/TM-13584, April 1998.

5-10 S.M. Bowman et al., "Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks," NUREG/CR-6716, February 2001.

5-11 C.J. Werner, et al., "MCNP Version 6.2 Release Notes," Los Alamos National Laboratory, LA-UR-18-20808, February 2018.

# 6 CRITICALITY EVALUATION

## 6.1 Description of Criticality Design

### 6.1.1. Design Features

This section describes the design features of the Model 2000 Transport Package that are important for maintaining criticality safety.

The Model 2000 cask is a cylindrical lead lined cask used for transporting Type B quantities of radioactive materials and solid fissile materials. For the fissile contents considered in this analysis, the High Performance Insert (HPI) is required to be used along with the Model 2000 cask. The HPI consists of the insert body and two plugs for the top and bottom. Attached to the insert body is a series of [[ ]] in the Model 2000 cask cavity. Shoring components such as rod holders or the material basket may be present. This analysis is generic by design to allow for the simple loading flexibility of the desired contents into the HPI, then loading of the HPI into the Model 2000 cask. Figure 1.2-1 shows the package configuration. The HPI and material basket are described in Section 1.2.1.3 and Section 1.2.1.4, respectively.

The system consists of the Model 2000 cask, HPI, and other components which ensure that the fuel rod content is shipped upright (e.g., material basket and the fuel rod tube). The term fuel rod is used throughout the context of Chapter 6 to more accurately depict the form of the approved GE BWR 10x10 irradiated fuel content and to reflect how it was modeled in the supporting criticality evaluations.

The Model 2000 cask, HPI, and material basket safety components retain the contents within a fixed geometry relative to other packages in an array. Fuel content rearrangement is limited by the HPI cavity and material basket category B safety components. The fuel pellets can be confined (e.g. encapsulated) within the fuel rod tube; however, the cladding material of the fuel rod is not credited in the criticality analysis. Components such as rod holders or the material basket may provide additional confinement, but are not credited in the criticality analyses.

### 6.1.2. Summary Table of Criticality Evaluation

The demonstration of criticality safety meeting 10 CFR 71 (Reference 6-1) provides assurance of the safe transport of the fissile contents with the Model 2000 and the HPI under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

The contents are GE BWR 10x10 irradiated fuel elements. The configuration of the contents and packaging demonstrate the most reactive configuration for the package.

A summary of most limiting cases is provided in Table 6.1.2-1 for fuel rod content. All limiting cases meet the Upper Subcritical Limit (USL), as defined in Section 6.3.4. The fissile mass limit for fuel rods defined by the criticality safety analyses provide an input to the Irradiated Fuel Rod Loading Table as further discussed in Section 7.5.3. For the fuel rod content, data trends of results in Section 6.4 through Section 6.6 shows that as the fuel rod outer radius increases the overall system reactivity decreases, thus, the fuel pellet outer radius of 0.2 cm is set as the minimum fuel pellet outer radius. Fissile material evaluations and limitations are based on fresh, unirradiated fuel (no credit is applied for burnup).

**Table 6.1.2-1. Fuel Rod Content Summary**

| Case Name | Fuel OR (cm) | Half-pitch (cm) | $k_{eff}$ | σ | $k_{eff}+2\sigma$ | Maximum $k_{eff}$ [1] | H/U-235 | EALF[2] (eV) | M U-235 Limit (g) [3] |
|---|---|---|---|---|---|---|---|---|---|
| Single Package | | | | | | | | | |
| FRLSmh_1_22 | 0.2 | 0.7 | 0.92307 | 0.00021 | 0.92349 | 0.9328 | 570 | 0.31716 | 1750 |
| NCT, 5N Package Array | | | | | | | | | |
| FRLANmh_1_22 | 0.2 | 0.7 | 0.91810 | 0.00023 | 0.91856 | 0.9278 | 570 | 0.31717 | 1750 |
| HAC, 2N Package Array | | | | | | | | | |
| FRLAHmh_1_22 | 0.2 | 0.7 | 0.92350 | 0.00024 | 0.92398 | 0.9333 | 570 | 0.31681 | 1750 |

NOTES: USL is defined as 0.9387 per Section 6.3.4.
[1] Maximum $k_{eff}$ includes the added 1% uncertainty for pitch geometric modeling –see Section 6.9.1.
[2] EALF is defined as energy of average lethargy causing fission.
[3] An administratively reduced fissile mass limit is defined –see Section 6.2.1 for further details.

#### 6.1.3. Criticality Safety Index

The Model 2000 cask is shipped exclusive use only; a single package defines a conveyance. Therefore, per the guidance in 10 CFR 71.59 the criticality safety index is 50 for the Model 2000 cask for any fissile content.

## 6.2 Fissile Material Contents

The purpose of this analysis is to demonstrate that the contents are subcritical for a defined U-235 mass. The criticality analysis demonstrates compliance with 10 CFR 71 of the Model 2000 Transport Package with the HPI containing irradiated GE BWR 10x10 fuel elements.

### 6.2.1. Fuel Rods

The fuel rod contents of the package are restricted to low-enriched uranium dioxide ($UO_2$) fuel. The fissile material in fuel pellets is assumed to be uranium initially enriched up to a maximum of 6.0 wt% U-235 with the remaining 94 wt% modeled solely as U-238. Any U-232, U-234, or U-236 is assumed to be U-238 because these uranium isotopes are not fissile, are present in small amounts, and have total neutron cross sections that tend to be greater than the total neutron cross section for U-238. Additionally, no pellet dishing fraction or chamfering is modeled, which conservatively increases the number of U-235 atoms. Fissile material in the fuel rod contents is only in the form of $UO_2$, and administratively limited to 1750 grams of U-235. The models use the theoretical density, 10.96 g/cm$^3$, for $UO_2$. For the fuel rod content, sensitivity analyses showed that as the fuel rod OR increases the overall system reactivity decreases, thus the minimum fuel pellet OR of 0.2 cm is used in this analysis. Fissile material evaluations and limitations are based on initial, unirradiated fuel, without credit for burnup.

## 6.3 General Considerations

### 6.3.1. Model Configuration

#### 6.3.1.1. Fissile Material Contents Model Configuration

All fissile contents must be in solid form. Theoretical material density is conservatively evaluated for each content. Fissile material evaluations and limitation are based on initial, unirradiated fuel, without credit for burnup.

##### 6.3.1.1.1. Fuel Rods

The fuel rod contents of the package are restricted to low-enriched $UO_2$ fuel. The fissile material in fuel pellets is assumed to be uranium initially enriched up to a maximum of 6.0 wt% U-235 with the remaining 94 wt% modeled solely as U-238. See Section 6.3.2 for material properties.

The fuel rod is modeled as a long cylinder with an axial length of [[ ]] cm, which is near equivalent to the interior height of the HPI cavity (i.e., [[ ]] cm modeled). The modeled axial length bounds the requirement of minimum 5.3-inch rod length segments specified by the shielding analysis. The fuel rod OR is varied from 0.2 to 0.5 cm to encompass a variety of fuel designs. Smaller fuel rod OR results in a higher reactivity, see Sections 6.4 through 6.6 for results of each transport assessment. The materials of the fuel rod cladding or structural components are not modeled. Fuel pellets are assumed to be confined within cylindrical components (e.g., fuel rod holders, and/or the material basket).

The fuel rods are modeled in a hexagonal array with expanding pitch. The addition of more fuel rods through the expansion of the lattice model is evaluated to determine the optimum hydrogen-to-U-235 (H/U-235) ratio. While expansion of the lattice is a condition of HAC, it is also applied to NCT to optimize H/U-235. The fissile mass modeled in the fuel rod array is determined using a mixture of $UO_2$ and $H_2O$, with 1,800 grams of U-235 as the basis, which equates to approximately 34,000 grams of $UO_2$. A circular boundary, which equates to this quantity of U-235, is defined to limit the infinite, heterogeneous lattice to a specific array size. The circular boundary may cut rods radially, thus varying the $UO_2$ mass represented to less than 1,800 grams of U-235. Therefore, an administratively reduced limit of 1,750 grams of U-235 is defined. The equation below displays how the circular boundary radius is calculated for the varying hexagonal pitch sizes. Additionally, as the pitch is expanded to increase H/U-235, the confinement boundary of the HPI cavity will reduce the fissile mass within the boundary.

$$cavity\ OR = \sqrt{\#rods \frac{2\sqrt{3}P^2}{\pi}}$$

where, P is the hexagonal half-pitch

$$\#rods = M_{UO2} / (\pi * OR_{fuel}^2 * H * \rho_{UO2})$$

where,
H is the modeled fuel rod height of [[ ]] cm
$M_{UO2}$ is the mass of $UO_2$ in grams
$\rho_{UO2}$ is the theoretical density of $UO_2$

Table 6.3.1-1 defines the variation of fuel outer radii and pitches evaluated for the fuel rod content; the largest pitches for the smallest rods are not modeled as the $k_{eff}$ trend is already shown to be decreasing. Figure 6.3.1-1 shows the fuel rod model geometry, and Figure 6.3.1-2 shows examples of how the circular boundary defines the fissile mass limit by artificially cutting into the lattices.

Structural features of the rods, shoring components such as rod holders, or the HPI material basket may provide additional confinement of the fuel lattice expansion; however, only the HPI cavity is credited for the confinement boundary. Representation of the fuel structural components as water results in an increase in reactivity due to both a decrease in neutron absorption and an increase in fuel rod lattice moderation.

**Table 6.3.1-1. Fuel Rod Content Model Parameters**

| Parameter | Value (cm) |
|---|---|
| Fuel pellet radius (FROR) | 0.2, 0.3, 0.4, 0.5 |
| Half-pitch | FROR[XX]+0.3, FROR[XX]+0.4, FROR[XX]+0.6, FROR[XX]+0.7, FROR[XX]+0.8, FROR[XX]+0.9, FROR[XX]+1.1, FROR[XX]+1.2, FROR[XX]+1.4, FROR[XX]+1.6, FROR[XX]+2.0 |

[[

]]

**Figure 6.3.1-1. Fuel Rod Content Model Geometry**

[[

]]

**Figure 6.3.1-2. Fuel Rod Content Boundary Model Geometry (Not to Scale)**

For the transport evaluations of fuel rods, the maximum fuel $k_{eff}$ occurs when the fuel lattice is moderated with full density water. For HAC, when leakage during immersion for fuel rod loading activities is possible, moderation in the fuel lattice is assumed present. The full density moderation in the fissile region is conservatively maintained for NCT.

#### 6.3.1.2. Model 2000 and HPI Model Configuration

The MCNP6 model geometry used for criticality safety calculations is a detailed three-dimensional model of the HPI and the Model 2000 cask. Some slight simplifications are made to the MCNP6 geometry to reduce the modeling complexity, such as [[ ]]

The design features of the Model 2000 with the HPI are provided in Section 5.3, including dimensions, materials of construction, and densities of the materials. Table 6.3.1-2 provides the relevant dimensions of the MCNP6 model including the modeled thicknesses of each material used in the geometry. Table 6.3.1-2 along with Section 5.3 allows for a quick review of the most significant dimensions of the criticality model geometry. It can be noted in Table 6.3.1-2 that all HPI dimensions are minimum, with the fabrication tolerances subtracted from the nominal values. The model dimensions for the Model 2000 cask use both nominal and minimum values where it is appropriate. For example, for the [[ ]] at the minimum thickness. However, a number of the dimensions for the cask and overpack are prescribed thicknesses for steel plate that are used for the cask shells (e.g., the 1-inch rolled cask shells). For these instances the prescribed thickness of the steel plate is used for the MCNP geometry.

**Table 6.3.1-2. Relevant MCNP Model Dimensions**

| Model 2000 Component | Part | Dimension | Value (cm) | Value (in) |
|---|---|---|---|---|
| Cask | Cask Lid | $t_{SS1}$ | 3.810 | 1.500 |
| | | $t_{Pb}$ | 13.64 | 5.370 |
| | | $t_{SS2}$ | 4.445 | 1.750 |
| | Cask Side | $r_{cavity}$ | 33.66 | 13.25 |
| | | $t_{SS1}$ | 2.540 | 1.000 |
| | | $t_{Pb}$ | 10.16 | 4.000 |
| | | $t_{SS2}$ | 2.540 | 1.000 |
| | | $h_{pb}$[3] | 141.9 | 55.87 |
| | Cask Bottom | $t_{SS}$ | 14.94[1] | 5.880[1] |
| | | $h_{cavity}$ | 137.5 | 54.13 |
| HPI | HPI Top Lid | $t_{SS1}$ | [[ | |
| | | $t_{DU}$ | | |
| | | $t_{SS2}$ | | |
| | HPI Body Side | $r_{cavity}$ | | |
| | | $t_{SS1}$ | | |
| | | $t_{DU}$ | | |
| | | $t_{SS2}$ | | |
| | HPI Bottom Lid | $t_{SS1}$ | | |
| | | $t_{DU}$ | | |
| | | $t_{SS2}$ | | ]] |

Notes: [1] Cask bottom modeled flat, with thickness equal to the 6.13” height minus [[ ]].
[2] Minimum depleted uranium (DU) thicknesses considered with tolerance gaps explicitly modeled.
[3] Lead column height.

The package has multiple void regions, including within the confinement system. The effect of variations in the package moderation is evaluated by flooding all spaces within the package and varying the light water moderator density from 0.0 to 1.0 g/cm$^3$ (full density). Although leakage during immersion is not credible for HAC, the effect of varying package moderation for HAC was conservatively evaluated to cover normal loading and unloading activities. For NCT, the package cavities, unless otherwise noted, are dry with no additional moderation, as this is representative of this transport condition. Varying the H/U-235 ratio optimizes fissile contents, thus a full spectrum of density variation for each cavity region is not necessary to show the trend toward isolation of a package in an array.

Generally, for package arrays, moderating only the content fissile region with full density water results in the maximum neutron interaction between packages in an array and bounds any variations in the flooding sequence. The voided space in packaging cavities and between packages in an array allows for increased interaction between the packages. The inclusion of interspersed moderation in these regions would increase the isolation of packages within the array, which leads to a decreasing system reactivity approaching that of a single, isolated package. For the single package analysis, the package is fully flooded. Hence, no void regions exist within the model. This generates an increase in reflection, while decreasing particle leakage. However, as the reflection from the depleted uranium (DU) HPI shields provides the dominant increase in neutron interaction within a package, the variation of moderator density provides little additional neutron moderation to increase the package reactivity for the package array and single package. Additionally, the single package and package array model have a 30.48 cm-thick full density water reflector blanket around the exterior. The specification of 30.48 cm (12 inches) of water reflection is selected as a practical value. SSG-26 (Reference 6-2), Section 6.8.1, specifies 20 cm of water reflection as a practical value, as an additional 10 cm of water reflection would add less than 0.5% in reactivity to an infinite slab of U-235.

Chapter 2 structural evaluations show no damage or deformation to the HPI and material basket for NCT or HAC. Model 2000 cask structural evaluations define localized damage for the HAC pin-puncture test and damage to the impact limiters during drop tests, thus, the overpack is conservatively not modeled for HAC. A drop may also allow the content to shift within the HPI cavity. The fuel rod content is defined as the length of the HPI cavity and allowed to shift within the HPI.

Figure 6.3.1-3 shows the HAC MCNP model geometry.

[[

]]

**Figure 6.3.1-3. Package Model Geometry**

#### 6.3.1.2.1. NCT Model

For NCT, the MCNP criticality model includes the HPI and the Model 2000 cask, and conservatively neglects the overpack material and spacing to be consistent with the HAC model, even though structural evaluations show that the damage to the overpack is minimal. The materials for the HPI and cask are defined as prescribed in Section 6.3.2 in the appropriate cells of the model. The NCT 5N package array is represented by seven (7) packages in a hexagonal array as to provide maximum reflection and package neutron interaction. Figure 6.3.1-4 shows the top view of the MCNP model, NCT array. For cases where the fissile content size is limited such that it does not occupy the full HPI cavity radius, the contents are positioned within the HPI cavity toward the centroid of the group of packages, see Figure 6.3.1-5 for example of fuel rod case limited by mass.

Light water moderation is used within the fissile material matrix for each content, optimizing the H/U-235 ratio. No leakage of water is evaluated in the various cavity regions of the packaging, for the evaluation of undamaged packages under NCT (per 10 CFR 71.55(d)). The Model 2000 cask cavity region is void and the HPI cavity region is flooded; this NCT configuration is evaluated for conservatism, and does not imply in-leakage of water during transportation. Full density moderation is maintained between the packages, as $k_{eff}$ results for the single package and package array (NCT and HAC) are very close indicating the shield materials of the HPI and cask provide strong reflection thus neutronically isolating each package within the array.

[[

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**Figure 6.3.1-4. Package Array NCT, 5N Model Geometry**

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**Figure 6.3.1-5. Package Array NCT, 5N Model Geometry, Content Positioning**

#### 6.3.1.2.2. HAC Model

For HAC, the MCNP6 criticality model is the same as the shielding model, and only includes the HPI and the Model 2000 cask, neglecting the material and spacing of the overpack. This model conservatively assumes the complete destruction and removal of the overpack. Additionally, to be consistent with shielding analysis, the HAC model includes the lead slump from which the maximum deformation in the lead column is calculated to be 3.56 mm, which is conservatively rounded up to 4 mm (Section 2.12.2). Figure 6.3.1-6 shows the MCNP model for the HAC array. For cases where the fissile content size is limited such that it does not occupy the full HPI cavity radius, the contents are positioned within the HPI cavity toward the center package. See Figure 6.3.1-7 for an example of a case limited by fissile mass.

Light water moderation is used within the fissile material matrix for each content, optimizing the H/U-235 ratio. Moderation caused by in-leakage is limited to moderators no more effective than water from sources external to the package. Per 10 CFR 71.59(a)(2), the HAC, 2N assessment evaluates the sensitivity of hydrogenous moderation by evaluating water in-leakage into all void spaces of the package cavity regions, including those within the containment system. The moderation space is defined as all available space within the packaging cavities, not including any space occupied by the structural and shielding material design. The moderation density is varied from 0 to 1.0 g/cm$^3$ for the HPI cavity region and the Model 2000 cask region. A full spectrum of density variation for each cavity region is not necessary to show the trend toward isolation of the package in an array. The fissile material matrix region is maintained as fully flooded with full density water for both contents.

[[

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**Figure 6.3.1-6. Package Array HAC, 2N Model Geometry**

[[

]]

**Figure 6.3.1-7. Package Array HAC, 2N Model Geometry, Content Positioning**

### 6.3.2. Material Properties

The material compositions used in the criticality safety analyses are listed in Table 6.3.2-1 through Table 6.3.2-5. The structural components of the Model 2000 cask are constructed of Type 304 stainless steel. The nuclear properties relevant to criticality safety for 304 stainless steel are in Table 6.3.2-1. Type [[ ]] stainless steel comprises the structural components of the HPI. The nuclear properties relevant to criticality safety for [[ ]] stainless steel are in Table 6.3.2-2. It can be noted that there is negligible difference between the two types of stainless steel in terms of absorption properties. Both types are only included for accuracy of the actual materials of construction. Any slight change in the elemental composition of these steels does not result in any significant increase in calculated reactivity. The densities and material compositions for both stainless steel types are from Pacific Northwest National Lab report PNNL-15870 (Reference 6-3). The shielding material of the Model 2000 cask is solely comprised of lead. The nuclear properties relevant to criticality safety for lead are in Table 6.3.2-3. The shielding material of the HPI is solely comprised of DU. The nuclear properties relevant to criticality safety for DU are in Table 6.3.2-4. The densities of the lead and DU materials are based on the minimum specified densities for these materials in the respective component licensing drawings in Section 1.3.1. Isotopic masses are from SCALE6.1 Manual, Table M8.2.1 (Reference 6-4). For the modeling of water moderation, the S(α,β) thermal kernel treatment for hydrogen in the water is applied. The collision kinematics data includes thermal kinematics kernels to describe thermal scattering in moderating materials such as hydrogen in water. Table 6.3.2-5 defines the fissile material properties.

**Table 6.3.2-1. Nuclear Properties of Type 304 Stainless Steel**

| Element | Isotope | ZAID | Mass Fraction |
|---|---|---|---|
| C | C-12 | 6012 | 3.9537E-04 |
| | C-13 | 6013 | 4.6337E-06 |
| Si | Si-28 | 14028 | 4.5933E-03 |
| | Si-29 | 14029 | 2.4168E-04 |
| | Si-30 | 14030 | 1.6499E-04 |
| P | P-31 | 15031 | 2.3000E-04 |
| S | S-32 | 16032 | 1.4207E-04 |
| | S-33 | 16033 | 1.1568E-06 |
| | S-34 | 16034 | 6.7534E-06 |
| | S-36 | 16036 | 1.6825E-08 |
| Cr | Cr-50 | 24050 | 7.9300E-03 |
| | Cr-52 | 24052 | 1.5903E-01 |
| | Cr-53 | 24053 | 1.8380E-02 |
| | Cr-54 | 24054 | 4.6614E-03 |
| Mn | Mn-55 | 25055 | 1.0000E-02 |
| Fe | Fe-54 | 26054 | 3.9617E-02 |
| | Fe-56 | 26056 | 6.4490E-01 |
| | Fe-57 | 26057 | 1.5160E-02 |
| | Fe-58 | 26058 | 2.0529E-03 |
| Ni | Ni-58 | 28058 | 6.2158E-02 |
| | Ni-60 | 28060 | 2.4768E-02 |
| | Ni-61 | 28061 | 1.0946E-03 |
| | Ni-62 | 28062 | 3.5472E-03 |
| | Ni-64 | 28064 | 9.3254E-04 |
| Density (g/cm$^3$) | 8.00 | | |

**Table 6.3.2-2. Nuclear Properties of** [[ ]]

| Element | Isotope | ZAID | Mass Fraction |
|---|---|---|---|
| C | C-12 | 6012 | 4.0525E-04 |
| | C-13 | 6013 | 4.7496E-06 |
| Si | Si-28 | 14028 | 4.6576E-03 |
| | Si-29 | 14029 | 2.4507E-04 |
| | Si-30 | 14030 | 1.6730E-04 |
| P | P-31 | 15031 | 2.3000E-04 |
| S | S-32 | 16032 | 1.4207E-04 |
| | S-33 | 16033 | 1.1568E-06 |
| | S-34 | 16034 | 6.7534E-06 |
| | S-36 | 16036 | 1.6825E-08 |
| Cr | Cr-50 | 24050 | 7.0953E-03 |
| | Cr-52 | 24052 | 1.4229E-01 |
| | Cr-53 | 24053 | 1.6445E-02 |
| | Cr-54 | 24054 | 4.1707E-03 |
| Mn | Mn-55 | 25055 | 1.0140E-02 |
| Fe | Fe-54 | 26054 | 3.7769E-02 |
| | Fe-56 | 26056 | 6.1482E-01 |
| | Fe-57 | 26057 | 1.4453E-02 |
| | Fe-58 | 26058 | 1.9571E-03 |
| Ni | Ni-58 | 28058 | 8.0637E-02 |
| | Ni-60 | 28060 | 3.2131E-02 |
| | Ni-61 | 28061 | 1.4200E-03 |
| | Ni-62 | 28062 | 4.6018E-03 |
| | Ni-64 | 28064 | 1.2098E-03 |
| Mo | Mo-92 | 42092 | 3.5374E-03 |
| | Mo-94 | 42094 | 2.2586E-03 |
| | Mo-95 | 42095 | 3.9322E-03 |
| | Mo-96 | 42096 | 4.1686E-03 |
| | Mo-97 | 42097 | 2.4141E-03 |
| | Mo-98 | 42098 | 6.1715E-03 |
| | Mo-100 | 42100 | 2.5175E-03 |
| Density ($g/cm^3$) | 8.00 | | |

**Table 6.3.2-3. Nuclear Properties of Lead**

| Element | Isotope | ZAID | Mass Fraction |
|---|---|---|---|
| Pb | Pb-204 | 82204 | 1.3781E-02 |
| | Pb-206 | 82206 | 2.3956E-01 |
| | Pb-207 | 82207 | 2.2074E-01 |
| | Pb-208 | 82208 | 5.2592E-01 |
| Density ($g/cm^3$) | 11.34 | | |

**Table 6.3.2-4. Nuclear Properties of Depleted Uranium**

| Element | Isotope | ZAID | Mass Fraction |
|---|---|---|---|
| U | U-235 | 92235 | 7.0000E-03 |
| | U-238 | 92238 | 9.9300E-01 |
| Density ($g/cm^3$) | [[ ]] | | |

**Table 6.3.2-5. Nuclear Properties of Fissile Content**

| Compound | Isotope | ZAID | Mass Fraction | Density ($g/cm^3$) |
|---|---|---|---|---|
| $UO_2$, 6 wt% U-235 | U-235 | 92235 | 5.2257E-02 | 10.96 [1] |
| | U-238 | 92238 | 8.2917E-01 | |
| | O-16 | 8016 | 1.1857E-01 | |

References:
[1] Reference 6-4 Density, Table M8.2.4.
[2] Reference 6-4 Density, Table M8.2.2.
[3] Reference 6-3 Density, Page 235.

### 6.3.3. Computer Codes and Cross Section Libraries

The criticality safety analysis was completed using MCNP6 Version 1.0 (Reference 6-5) with the continuous-energy neutron data library ENDF/B-VII.1 (Reference 6-6). MCNP6 is a general-purpose, continuous-energy, generalized-geometry, time-dependent, Monte Carlo radiation-transport code designed to track many particle types over a broad range of energies. The criticality safety assessment is for low-enriched uranium fuel rods in the Model 2000 cask with the HPI. MCNP6 meets the recommendations in Section 4, Method of Analysis defined in NUREG/CR-5661 (Reference 6-7).

#### 6.3.3.1. Convergence Criteria

Convergence of the cases in criticality safety analysis was verified through inspection of the Shannon entropy of the fission source distribution. In order to determine the Shannon entropy of the problem, MCNP6 divides the fissionable regions of the problem into several bins, which then tally the fission sources during the random walks of each cycle. As the number of cycles completed increases, the fission source distribution will converge to steady state. In the output, MCNP6 prints which cycle was the first cycle to have a value of Shannon entropy within one standard deviation of the average Shannon entropy of the last half of the cycles analyzed; this is the minimum acceptable source convergence. The proper determination of the source convergence requires inspection of the plot of Shannon entropy versus cycle number, from which it can be determined at which cycle number the Shannon entropy converges. At least that many cycles were discarded, and more than 100 additional cycles are run after source convergence. Additionally, to determine adequacy of $k_{eff}$ convergence, the behavior of $k_{eff}$ with cycle number is evaluated to ensure no upward or downward trends are present.

### 6.3.4. Demonstration of Maximum Reactivity

A system is considered acceptably subcritical if a calculated $k_{eff}$ plus calculational uncertainties lies at or below the USL (i.e., $k_{system} + \Delta k_{system} \leq USL$). Thus, the USL is the magnitude of the sum of the biases, uncertainties, and administrative and/or statistical margins applied to a set of critical benchmarks, such that a high degree of confidence defines subcriticality of the system (Reference 6-8):

$$USL = 1 - \Delta k_m + \beta - \Delta\beta$$

where
$\Delta k_m$ is the additional margin to ensure subcriticality (0.05)
$\beta$ is the calculation bias
$\Delta\beta$ is the uncertainty in the bias

Based on a given set of critical experiments, the USL is defined as a function of key system parameters, such as EALF, fuel enrichment, or H/U-235 ratio. Because both $\beta$ and $\Delta\beta$ may vary with a given parameter, the USL is typically expressed as a function of the parameter, within an appropriate range of applicability derived from the parameter bounds. Table 6.8.2-1 displays the USL functions for the Model 2000 Transport Package criticality safety analysis.

The low-enriched lattice system USL function is applicable to the fuel rod content. Results of Section 6.4 through 6.6 shows that the limiting cases for the single package and package array have a H/U-235 value of 570. The value of H/U-235 is calculated based on a ratio of volume for the pitch cell and the estimated number of rods modeled (see equation below). Applying this value to the USL equation in Table 6.8.2-1, results in a rounded USL value of 0.9387 for fuel rod contents (e.g., 0.9473 - 1.5031E-5*570 = 0.93873 for X > 214.9).

$$\frac{H}{X} Ratio = \frac{H}{U^{235}} Ratio = WTF \times \frac{H_2O\ Density}{UO_2\ Density} \times \frac{(1 - Enr.U^{235})MW_{U^{235}} + 2MW_{O_2}}{MW_{H_2O}} \times \frac{2}{1} \times \frac{1}{Enr.U^{235}}$$

where

$$Water\ to\ Fuel\ Volume\ Ratio = \frac{Water\ volume\ in\ fuel\ rod\ cell\ \times \#rods}{Fuel\ volume\ \times \#rods}$$

For FRLA(N/H)_1_22 limiting cases, the #rods is estimated at 203 rods

A fissile mass limit for is selected based on a $k_{eff} + 2\ \sigma$ value that allows for an appropriate USL margin that includes uncertainty and an administrative margin.

## 6.4 Single Package Evaluation

### 6.4.1. Configuration

This single model represents NCT and HAC for the single package evaluations; models are described in Section 6.3. The reference case for the single package is to fill all cavity regions that are normally void space with full density water. The fissile matrix region is moderated with full density water.

### 6.4.2. Results

Peak cases for fuel rod content are provided in Table 6.4.2-1; full results are in Section 6.9.3, Table 6.9.3-2. Figure 6.4.2-1 displays the trends for all data. Based on the most limiting case for the single package, FRLSmh_1_22 (OR=0.2 cm, hex, half-pitch=0.7 cm), the $k_{eff} + 2\sigma$ equals 0.92349. An additional 1% uncertainty is added for the variation of pitch geometric modeling; see Section 6.9.1 for details. Thus, the final, maximum $k_{eff}$ for the fuel rod, single package case is 0.9328 (rounded up for conservatism). Data trends show that as the fuel rod OR increase the overall system reactivity decreases, thus the minimum fuel rod OR of 0.2 cm is a limiting parameter. Thus, an equivalent cylindrical fuel radius of ≥ 0.20 cm is required for the 1750-gram mass limit.

Section 6.3.4 defines the USL value as 0.9387 for fuel rod contents.

**Table 6.4.2-1. Fuel Rod Content, Single Package, Maximum Cases**

| Case Name | Fuel OR (cm) | Half-Pitch (cm) | Estimated No. Rods Modeled | U-235 Mass (g) | $k_{eff}$ | σ | $k_{eff}+2\sigma$ | H/U-235 | EALF (eV) |
|---|---|---|---|---|---|---|---|---|---|
| FRLSmh_1_1 | 0.2 | 0.50 | [[ | 1766 | 0.85526 | 0.00025 | 0.85576 | -- | -- |
| FRLSmh_1_2 | 0.2 | 0.60 | | 1766 | 0.90946 | 0.00023 | 0.90992 | -- | -- |
| FRLSmh_1_21 | 0.2 | 0.65 | | 1766 | 0.92134 | 0.00022 | 0.92178 | -- | -- |
| FRLSmh_1_22 | 0.2 | 0.70 | | 1766 | 0.92307 | 0.00021 | 0.92349 | 570 | 0.31716 |
| FRLSmh_1_23 | 0.2 | 0.75 | | 1766 | 0.91659 | 0.00022 | 0.91703 | -- | -- |
| FRLSmh_1_3 | 0.2 | 0.80 | | 1757 | 0.89990 | 0.00022 | 0.90034 | -- | -- |
| FRLSmh_1_4 [a] | 0.2 | 0.90 | | 1429 | 0.83653 | 0.00021 | 0.83695 | -- | -- |
| FRLSmh_1_5 [a] | 0.2 | 1.00 | ]] | 1219 | 0.78051 | 0.00020 | 0.78091 | -- | -- |

NOTE: [a] Number of rods is limited by HPI cavity size, as described in Section 6.3.1.1.1.

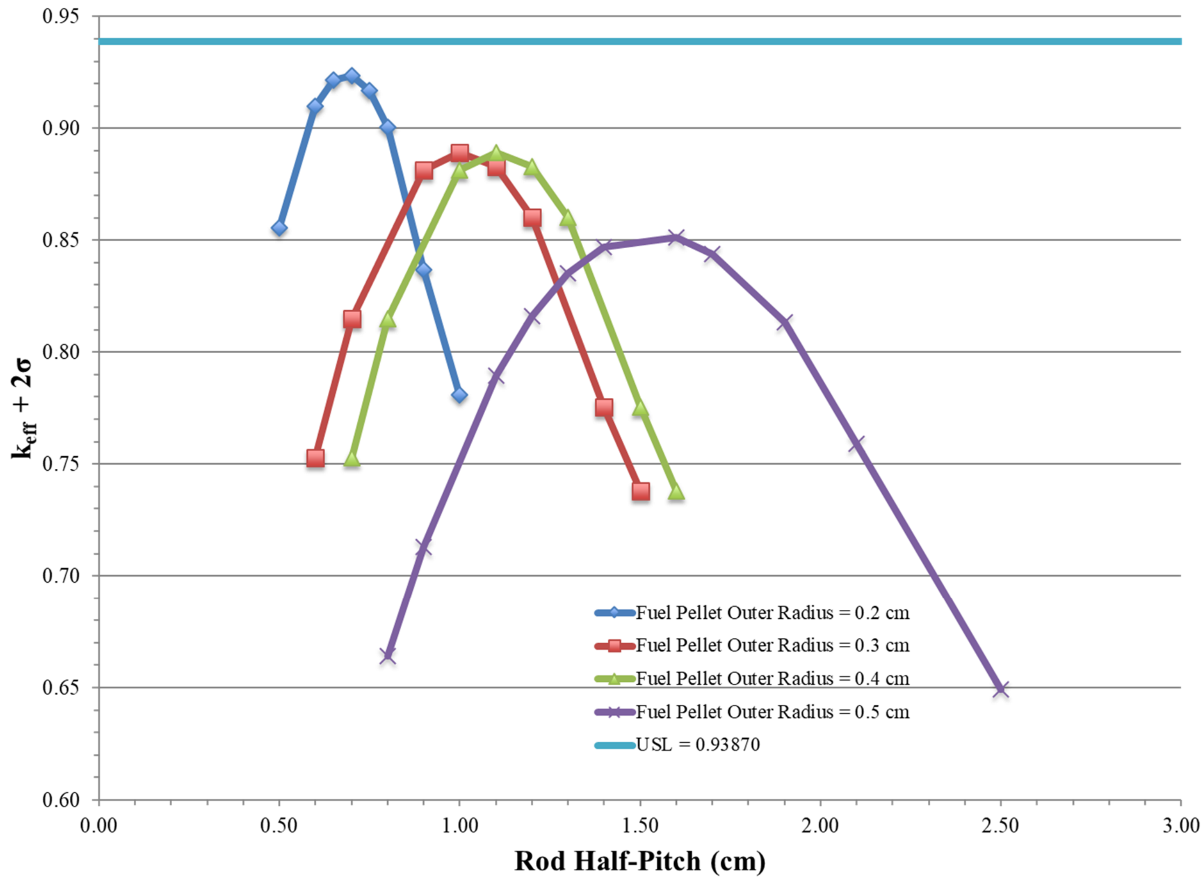


**Figure 6.4.2-1. Fuel Rod Content, Single Package, Results**

A comparison between the nominal GE BWR 10x10 fuel parameters and the fuel parameters used in the criticality evaluation is shown in Table 6.4.2-2.

**Table 6.4.2-2. Nominal vs. Analyzed Fuel Parameters for the GE2000 Criticality Analysis**

| Case | Pellet Outer Diameter (cm) | Initial U-235 Enrichment Range (wt.%) | Pellet Theoretical Density |
|---|---|---|---|
| Typical GE BWR 10x10 | [[ ]] | ≤5.0 | [[ ]] |
| Cases Modeled for Fuel Rod Transport | 0.4-1.0 | ≤6.0 | 100% |

## 6.5 Evaluation of Package Arrays Under Normal Conditions of Transport

### 6.5.1. Configuration

As the Model 2000 cask with HPI is shipped exclusive use, a single package defines a conveyance. Thus, the package array criticality evaluation defines the number N of packages as one. Therefore, for NCT, the package array is modeled as seven (7) packages in a hexagonal array. This evaluation demonstrates that five (5) times N packages is shown to be subcritical with the package arrangement reflected on all sides by 30.48 cm of water. The NCT package array model is described in Section 6.3.1.2.1.

The reference case for the NCT package array is to maintain void in all cavity regions that are normally void space. Full density moderation is maintained between the packages, as $k_{eff}$ results for the single package and package array (NCT and HAC) are very similar, indicating the shield materials of the HPI and cask provide strong reflection, thus neutronically isolating each package within the array. As the reflection from the DU HPI shields provides the dominant impact on neutron interactions within a package, the variation of moderator density provides little additional neutron interaction to increase the package reactivity; see Section 6.9.2.1.2 for comparison. The confinement boundary for NCT and HAC is defined as the HPI cavity. For both contents, the fissile matrix region (HPI cavity) is moderated with full density water. The fuel rod content is described in Section 6.2.1.

### 6.5.2. Results

Result of the HAC 2N array show that the combination of the smallest fuel rod OR and pitch variation produce the highest reactivity in the package array. Therefore, only the three smallest fuel rod OR values (0.2, 0.3, and 0.4 cm) are evaluated for the NCT 5N package array. Peak cases for NCT fuel rod content are provided in Table 6.5.2-1; full results are in Section 6.9.3, Table 6.9.3-3. Figure 6.5.2-1 displays the trends for all evaluated data. For the most limiting case for the NCT package array, FRLANmh_1_22 (OR=0.2 cm, hex, half-pitch=0.7 cm), the $k_{eff} + 2\sigma$ is 0.91856. An additional 1% uncertainty is added for the variation of pitch geometric modeling; see Section 6.9.1 for details. Thus, the final, maximum $k_{eff}$ for the fuel rod, single package case is 0.9278 (rounded up for conservatism). Data trends show that as the fuel rod OR increases the overall system reactivity decreases, thus the minimum fuel rod OR of 0.2 cm is a limiting parameter.

Section 6.3.4 defines the USL value as 0.9387 for fuel rod contents.

**Table 6.5.2-1. Fuel Rod Content, NCT 5N, Maximum Cases**

| Case Name | Fuel OR (cm) | Half-pitch (cm) | Estimated No. Rods Modeled | U-235 Mass (g) | $k_{eff}$ | σ | $k_{eff}+2\sigma$ | H/U-235 | EALF (eV) |
|---|---|---|---|---|---|---|---|---|---|
| FRLANmh_1_1 | 0.2 | 0.50 | [[ | 1766 | 0.83754 | 0.00024 | 0.83802 | -- | -- |
| FRLANmh_1_2 | 0.2 | 0.60 | | 1766 | 0.89350 | 0.00023 | 0.89396 | -- | -- |
| FRLANmh_1_21 | 0.2 | 0.65 | | 1766 | 0.90893 | 0.00024 | 0.90941 | -- | -- |
| FRLANmh_1_22 | 0.2 | 0.70 | | 1766 | 0.91810 | 0.00023 | 0.91856 | 570 | 0.31717 |
| FRLANmh_1_23 | 0.2 | 0.75 | | 1766 | 0.91564 | 0.00020 | 0.91604 | -- | -- |
| FRLANmh_1_3[a] | 0.2 | 0.80 | | 1757 | 0.90089 | 0.00020 | 0.90129 | -- | -- |
| FRLANmh_1_4[a] | 0.2 | 0.90 | | 1429 | 0.83616 | 0.00020 | 0.83656 | -- | -- |
| FRLANmh_1_5[a] | 0.2 | 1.00 | ]] | 1219 | 0.77767 | 0.00019 | 0.77805 | -- | -- |

NOTE: [a] Number of rod is limited by HPI cavity size, as described in Section 6.3.1.1.1

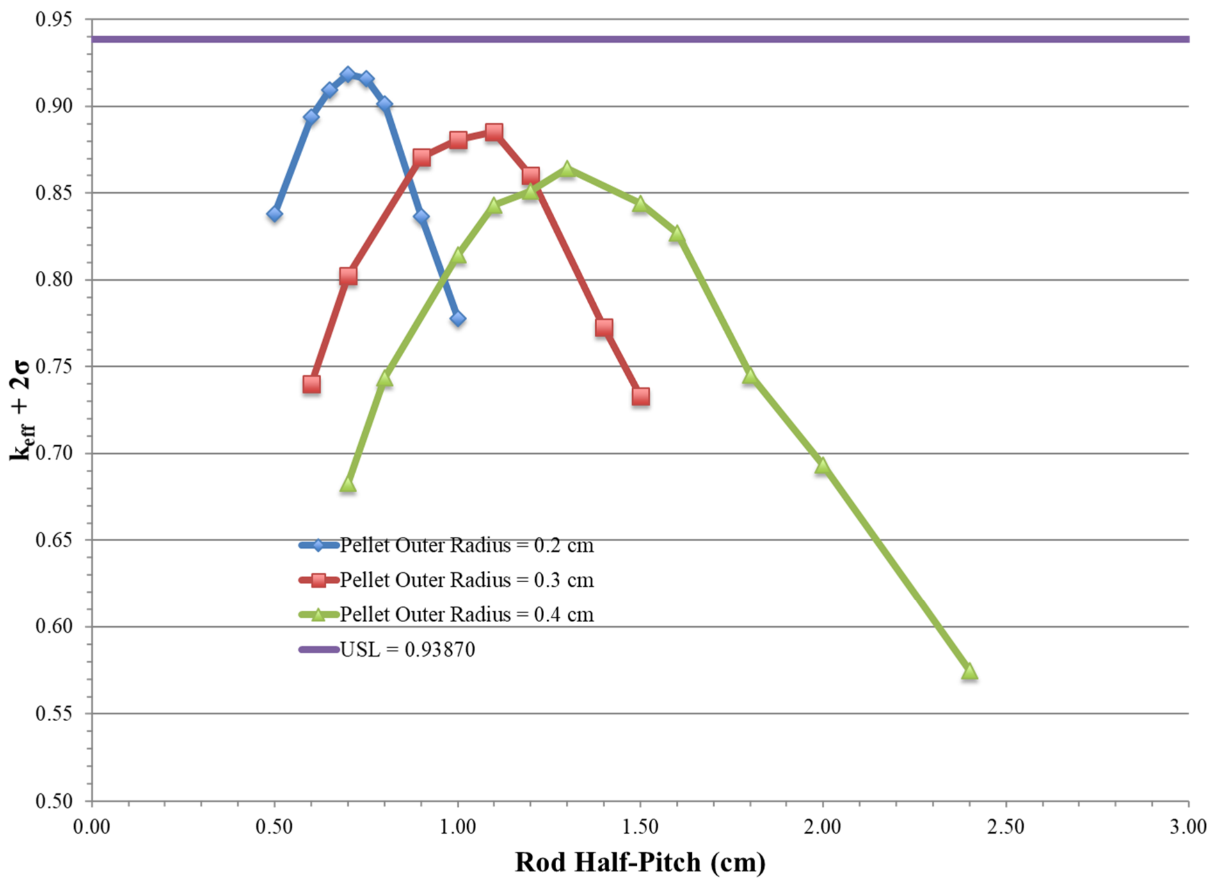


**Figure 6.5.2-1. Fuel Rod Content, NCT 5N, Results**

## 6.6 Package Arrays under Hypothetical Accident Conditions

### 6.6.1. Configuration

As the Model 2000 cask with HPI is shipped exclusive use, a single package defines a conveyance. Thus, the package array criticality evaluation defines the number N of packages as one. Therefore, for HAC, the package array is modeled as two packages side-by-side, evaluating that two times N packages is shown to be subcritical with the package arrangement reflected on all sides by 30.48 cm of water. The HAC, package array model is described in Section 6.3.1.2.2.

The reference case for the HAC package array is to fill all cavity regions that are normally void space with full density water. Full density moderation is maintained between the packages and within the packages. A parametric study assesses varied light water moderator density within the HPI and cask regions. The confinement boundary for HAC is defined as the HPI cavity. The fissile matrix region is moderated with full density water. The fuel rod content is described in Section 6.2.1.

### 6.6.2. Results

Peak cases for HAC package array, fuel rod content is provided in Table 6.6.2-1; full results are in Section 6.9.3, Table 6.9.3-4. Figure 6.6.2-1 displays the trends for all data. For the most limiting case for the HAC package array, FRLAHmh_1_22 (OR=0.2 cm, hex, half-pitch=0.7 cm), the $k_{eff}$ + 2σ is 0.92398. An additional 1% uncertainty is added for the variation of pitch geometric modeling, see Section 6.9.1 for details. Thus, the final, maximum $k_{eff}$ for the fuel rod, HAC package array case is 0.9333 (rounded up for conservatism). Data trends show that as the fuel rod OR increases the overall system reactivity decreases, thus the minimum fuel rod OR of 0.2 cm is a limiting parameter.

Section 6.3.4 defines the USL value as 0.9387 for fuel rod contents.

**Table 6.6.2-1. Fuel Rod Content, HAC 2N, Maximum Cases**

| Case Name | Fuel OR (cm) | Half-pitch (cm) | Estimated No. Rods Modeled | U-235 Mass (g) | $k_{eff}$ | σ | $k_{eff}$+2σ | H/U-235 | EALF (eV) |
|---|---|---|---|---|---|---|---|---|---|
| FRLAHmh_1_1 | 0.2 | 0.50 | [[ | 1766 | 0.85587 | 0.00026 | 0.85639 | -- | -- |
| FRLAHmh_1_2 | 0.2 | 0.60 | | 1766 | 0.90895 | 0.00025 | 0.90945 | -- | -- |
| FRLAHmh_1_21 | 0.2 | 0.65 | | 1766 | 0.92097 | 0.00022 | 0.92141 | -- | -- |
| FRLAHmh_1_22 | 0.2 | 0.70 | | 1766 | 0.92350 | 0.00024 | 0.92398 | 570 | 0.31681 |
| FRLAHmh_1_23 | 0.2 | 0.75 | | 1766 | 0.91616 | 0.00025 | 0.91666 | -- | -- |
| FRLAHmh_1_3[a] | 0.2 | 0.80 | | 1757 | 0.89981 | 0.00021 | 0.90023 | -- | -- |
| FRLAHmh_1_4[a] | 0.2 | 0.90 | | 1429 | 0.83686 | 0.00019 | 0.83724 | -- | -- |
| FRLAHmh_1_5[a] | 0.2 | 1.00 | ]] | 1219 | 0.78005 | 0.00019 | 0.78043 | -- | -- |

NOTE: [a] Number of rods is limited by HPI cavity size, as described in Section 6.3.1.1.1.

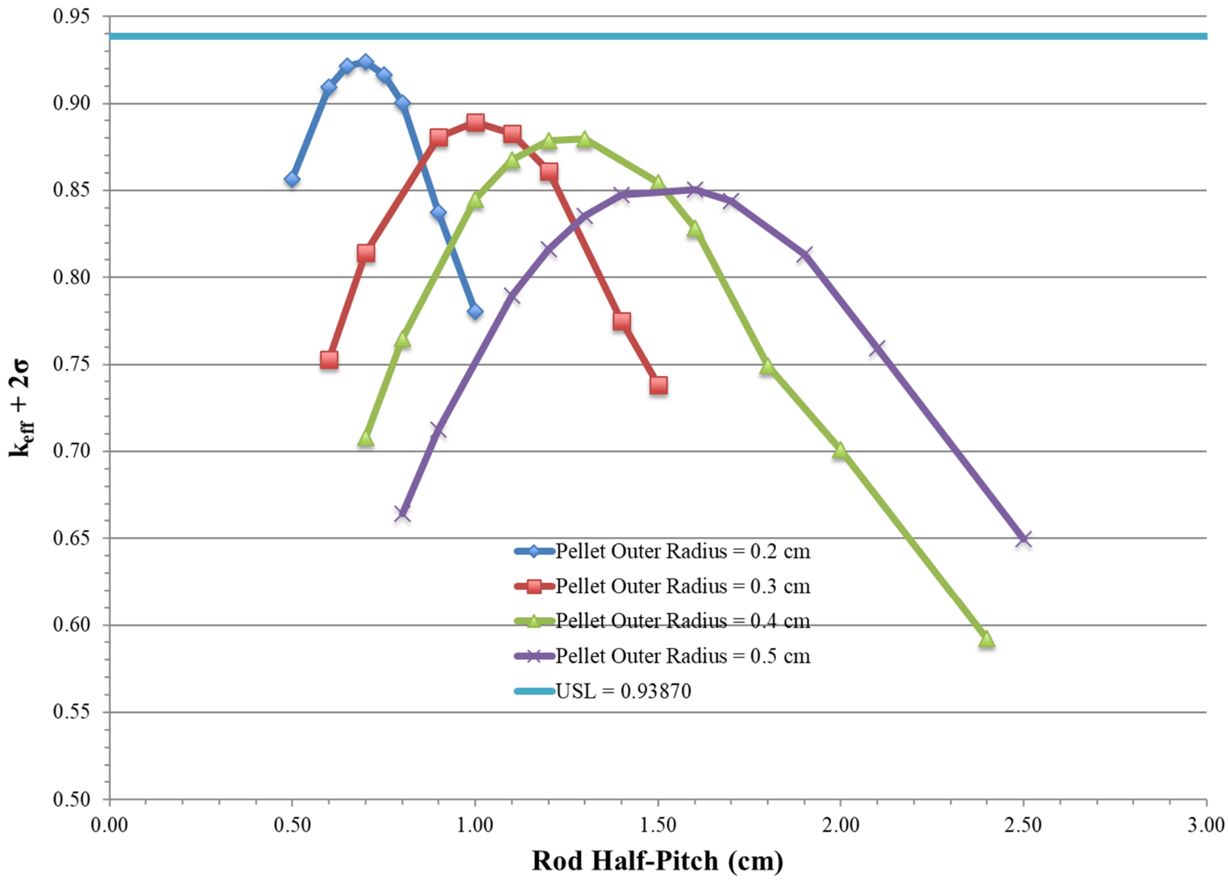


**Figure 6.6.2-1. Fuel Rod Content, HAC 2N, Results**

## 6.7 Fissile Material Packages for Air Transport

The Model 2000 Transport Package will not be transported by air.

## 6.8 Benchmark Evaluations

This section describes the criticality benchmarks for application of MCNP6 (Reference 6-5) with the continuous-energy neutron data library ENDF/B-VII.1 (Reference 6-6) and USLSTATS to the Model 2000 Transport Package criticality safety analysis. The application range is the criticality safety analysis of the proposed contents, consisting of high-enriched free form uranium, free form plutonium, and low-enriched uranium fuel rods in the Model 2000 cask with the HPI.

USLSTATS is used to generate an acceptable USL for the criticality safety analysis. The USLSTATS computer program uses two methods (i.e., (1) confidence band with administrative margin and (2) single-sided uniform width closed interval) to calculate and print USL correlations based on a set of user-supplied $k_{eff}$ values and corresponding values of a single associated parameter X (e.g., lattice pitch, fuel enrichment, average energy group causing fission (AEG)), for a set of criticality benchmark calculations.

### 6.8.1. Applicability of Benchmark Experiments

A total of 69 benchmark experiments were selected to develop the MCNP6 bias and bias uncertainty for the content allowed in the Model 2000 Transport Package.

The low-enriched uranium rod lattice configuration of the Model 2000 Transport Package modeled $UO_2$ rods with 6 wt% U-235 in hexagonally pitched lattices moderated with light water. No cladding was modeled. Experiments that were chosen, had an enrichment range of 2.35 wt% through 4.92 wt% U-235. The benchmark geometries were $UO_2$ rods in square pitched lattices submerged in light water. Reflectors consisted of steel, lead, or uranium with light water.

### 6.8.2. Bias Determination

#### 6.8.2.1. Method

Section 4.1 of NUREG/CR-6361 (Reference 6-8), explains two methods of determining the USL. The first method applies a statistical calculation of the bias and its uncertainty, plus an administrative margin, to a linear fit of critical experiment benchmark data, also known as Method 1: Confidence Band with Administrative Margin. In the second method, statistical techniques with a rigorous basis are applied in order to determine a combined lower confidence band plus subcritical margin, also known as Method 2: Single-Sided Uniform Width Closed Interval Approach. USLSTATS is a program that calculates USL correlations based on these methods. USLSTATS was used in this analysis in order to calculate the USL.

For this analysis, Method 1 is applied and Method 2 is used as a verification of Method 1 such that the USL function of Method 1 ($USL_1$) must be less than the USL function of Method 2 ($USL_2$). If the minimum margin of subcriticality, C*s(p) - W, is less than the administrative margin selected for Method 1, the administrative margin selected is sufficient, as this indicates that the administrative margin is larger than the statistical margin determined by Method 2.

#### 6.8.2.2. Results

NUREG/CR-6361 states that the correlation between the trending parameter and the critical data is the primary criterion to select the parameter that will be utilized to determine the USL. The parameter with the highest correlation coefficient was used to develop the USL, which was H/U-235 for the low-enriched uranium USL function.

For the low-enriched uranium lattice Model 2000 Transport Package contents, the H/U-235 trending parameter correlation coefficient, |r|, is equivalent to 0.3900. Therefore, the USL for low-enriched uranium lattice contents is 0.9387, see Section 6.3.4 for additional details.

The results of the USLSTATS analyses are presented in Table 6.8.2-1. This table includes the USL functions, the applicable trending parameter range, the minimum margin of subcriticality (C*s(p) – W), which is the statistically based subcritical margin from the $USL_2$ calculation, and the correlation coefficient (r) of the trending parameter to the critical data.

**Table 6.8.2-1. Model 2000 Transport Package Criticality Safety USL Functions**

| Trending Parameter | USL Equation (Method 1) | Trending Parameter Range | C*s(p) – W (Method 2) | Correlation Coefficient (r) |
|---|---|---|---|---|
| H/U-235 | 0.9473 - 1.5031E-5*X (X > 214.9)<br>0.9441 (X ≤ 214.9) | 105.5 ≤ X ≤ 256.3 | 9.5216E-3 | -0.3900 |

The administrative margin, $\Delta k_m$, for these analyses was 0.05, which is greater than C*s(p) – W calculated for each trending parameter. This signifies the selected administrative margin is acceptable.

The following figures plot each trending parameter against $k_{norm}$. The first line plots the function $k_c(x)$, the line of fit to the critical data. The second line plots the function $k_c(x) – w(x)$, the line of fit of the critical data with a lower band of 95% confidence margin. The third line plots the function $k_c(x) – W$, the line of best fit of the critical data with the largest band of the 95% confidence margin from the second line, $k_c(x)$ - w(x), applied as a conservatism.

The $USL_1$ and $USL_2$ plots plateau at a certain constant value to not credit for positive biases in $k_c(x) – W$.

The USL function-defining trending parameter for low-enriched uranium lattices was H/U-235, as shown in Figure 6.8.2-1. None of the values were below the lower confidence limit, $k_c(x)$ - W, of the calculated critical values. As the value of H/U-235 increased, the value of $k_{norm}$ decreased. This resulted in a negative correlation between H/U-235 and $k_{norm}$, with a coefficient of r = -0.3900, as shown in Table 6.8.2-1. This is the strongest correlation of the three trending parameters examined for low-enriched uranium lattice systems. Therefore, H/U-235 was selected as the USL function-defining trending parameter for low-enriched uranium lattice systems.

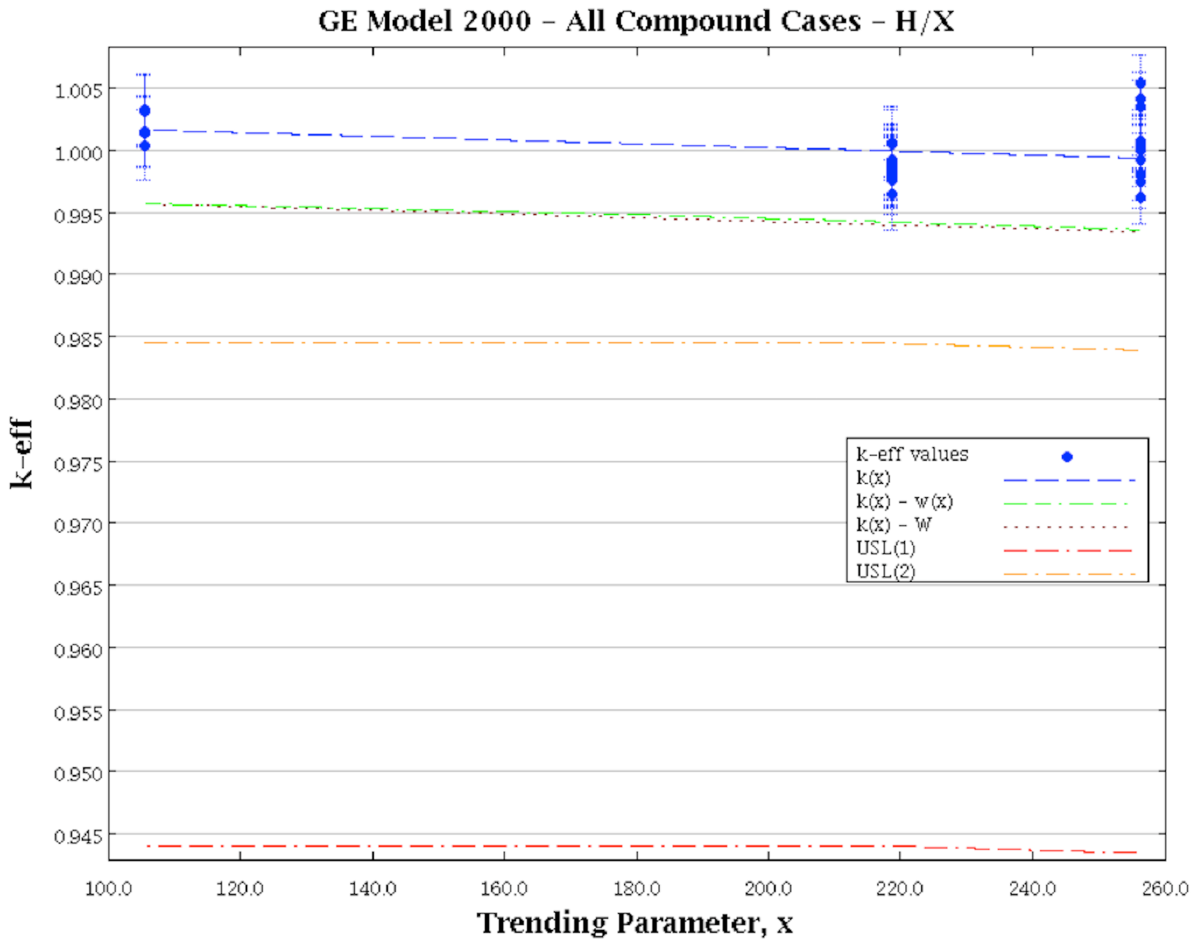


**Figure 6.8.2-1. USLSTATS Trend Plot of H/U-235 versus $k_{norm}$ – Lattice Systems**

## 6.9 Appendices

### 6.9.1. Comparison of Modeled Fuel Rod Pitch

While the fuel rod content is limited by a mass of 1750 grams of U-235 at a maximum of 6 wt% U-235 enrichment, the modeling configuration may vary allowing for a slight variation in the H/U-235 ratio and thus affecting the system criticality. A hexagonal pitch results in tightly packed array of rods, which increases the view factor between rods in the array, while the square pitch results in a slightly higher H/U-235 ratio than a hexagonal pitch. The smallest fuel pellet OR, 0.2 cm, has shown to be the most reactive configuration for fuel rod contents. For a fuel pellet OR of 0.2 cm, the pitch comparison models a square lattice as well as a hexagonal lattice, as shown in Figure 6.9.1-1. Table 6.9.1-1 shows the results of the pitch comparison, and Figure 6.9.1-2 plots the results; full results are in Section 6.9.3, Tables 6.9.3-4 and 6.9.3-5. Table 6.9.1-1 demonstrates that the hexagonal lattice pitch bounds the square lattice pitch at optimum H/U-235 ratio . The maximum difference between the lattice pitch geometry $k_{eff} + 2\sigma$ values is minimal (0.92811 – 0.92398 = 0.00413). However, a conservative 1.0% uncertainty is added to the final maximum $k_{eff} + 2\sigma$ values.

[[

]]

**Figure 6.9.1-1. Fuel Rod Content Pitch Modeling Comparison (Not to Scale)**

**Table 6.9.1-1. Fissile Mass Content, HAC 2N, Maximum Cases**

| Fuel OR | Half-pitch | $k_{eff}$ | σ | $k_{eff} + 2σ$ | H/U-235 | Mass U-235 (g) | $k_{eff}$ | σ | $k_{eff} + 2σ$ | H/U-235 | Mass U-235 (g) |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | HAC 2N, square pitch | | | | | HAC 2N, hexagonal pitch | | | | |
| 0.2 | 0.50 | 0.88665 | 0.00023 | 0.88711 | 317 | 1821.65 | 0.85587 | 0.00026 | 0.85639 | 269 | 1821.65 |
| 0.2 | 0.60 | 0.92539 | 0.00024 | 0.92587 | 477 | 1821.65 | 0.90895 | 0.00025 | 0.90945 | 407 | 1821.65 |
| 0.2 | 0.65 | 0.92769 | 0.00021 | 0.92811 | 568 | 1821.65 | 0.92097 | 0.00022 | 0.92141 | 485 | 1821.65 |
| 0.2 | 0.7 | 0.91849 | 0.00020 | 0.91889 | 665 | 1821.65 | 0.92350 | 0.00024 | 0.92398 | 570 | 1821.65 |
| 0.2 | 0.75 | 0.89758 | 0.00019 | 0.89796 | 771 | 1821.65 | 0.91616 | 0.00025 | 0.91666 | 661 | 1821.65 |
| 0.2 | 0.8 | 0.86402 | 0.00021 | 0.86444 | 883 | 1644.62 | 0.89981 | 0.00021 | 0.90023 | 759 | 1644.62 |
| 0.2 | 0.9 | 0.79759 | 0.00020 | 0.79799 | 1130 | 1299.46 | 0.83686 | 0.00019 | 0.83724 | 972 | 1299.46 |
| 0.2 | 1.0 | 0.74015 | 0.00021 | 0.74057 | 1406 | 1052.56 | 0.78005 | 0.00019 | 0.78043 | 1211 | 1052.56 |

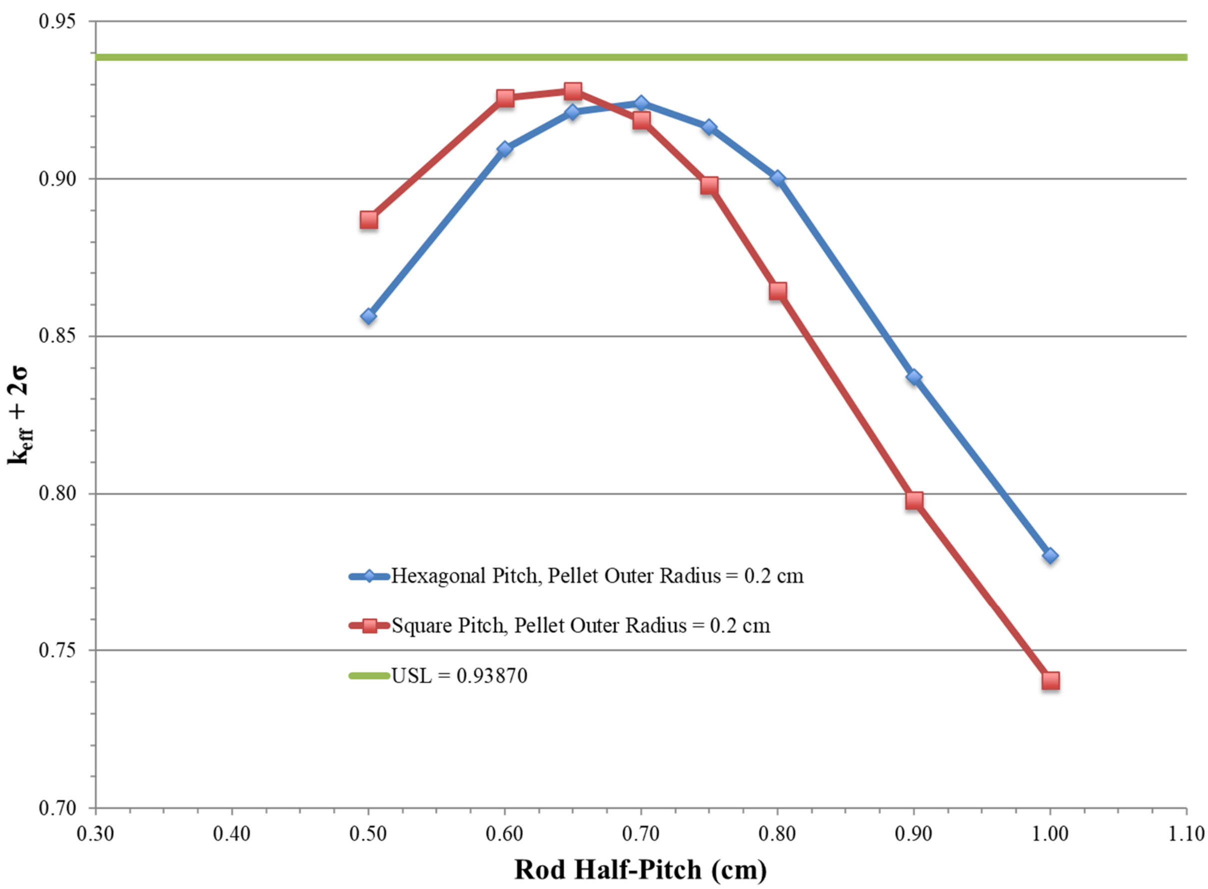


**Figure 6.9.1-2. HAC, Fuel Rod Pitch Comparison**

### 6.9.2. Benchmark Critical Experiments

Table 6.9.2-1 lists the USLSTATS input data for each critical experiment. The critical benchmark experiments were created for MCNP6 using the benchmark specifications in each critical benchmark experiment report (Reference 6-9). The values of EALF are those determined by MCNP6. The ratio H/U-235 and fuel enrichment were either reported in the critical benchmark experiment reports or were determined with hand calculations.

The values of $k_{bench}$ and $\sigma_{bench}$ are the reported effective multiplication factors and the 1σ statistical error from the critical benchmark experiment reports, respectively. The values of $k_{calc}$ and $\sigma_{calc}$ are the effective multiplication factor and the associated Monte Carlo 1σ determined by MCNP6, respectively. The USLSTATS input values of k and $\sigma_{sample}$ for each critical experiment correspond to $k_{norm}$ and $\sigma_{total}$ in Table 6.9.2-1. The normalization of k, $k_{norm}$, is calculated as $k_{calc}$ divided by $k_{bench}$:

$$k_{norm} = k_{calc} / k_{bench}$$

As the benchmark uncertainty and the calculated uncertainty are independent of each other, the total uncertainty, $\sigma_{total}$, was determined by combining the uncertainties using the square root of the sum of the squares:

$$\sigma_{total} = (\sigma_{bench}^2 + \sigma_{calc}^2)^{1/2}$$

**Table 6.9.2-1. USLSTATS Input from Critical Benchmark Lattice Experiments**

| Case | EALF (eV) [1] | H/U-235 | Fissile Enrichment | M/F Ratio [2] | $k_{bench}$ | $\sigma_{bench}$ | $k_{calc}$ | $\sigma_{calc}$ | $k_{norm}$ | $\sigma_{total}$ |
|---|---|---|---|---|---|---|---|---|---|---|
| *LEU-COMP-THERM-010* | | | | | | | | | | |
| 1 | 1.2060E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 1.00421 | 0.00041 | 1.00421 | 0.00214 |
| 2 | 1.1800E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 1.00547 | 0.00043 | 1.00547 | 0.00214 |
| 3 | 1.1616E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 1.00347 | 0.00044 | 1.00347 | 0.00215 |
| 4 | 1.1301E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 0.99625 | 0.00041 | 0.99625 | 0.00214 |
| 5 | 3.5938E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 0.99921 | 0.00039 | 0.99921 | 0.00214 |
| 6 | 2.6608E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 1.00040 | 0.00045 | 1.00040 | 0.00215 |
| 7 | 2.1245E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 1.00059 | 0.00038 | 1.00059 | 0.00213 |
| 8 | 1.8807E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 0.99816 | 0.00041 | 0.99816 | 0.00214 |
| 9 | 1.2527E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 0.99928 | 0.00045 | 0.99928 | 0.00215 |
| 10 | 1.2121E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 1.00071 | 0.00044 | 1.00071 | 0.00215 |
| 11 | 1.1912E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 1.00038 | 0.00043 | 1.00038 | 0.00214 |
| 12 | 1.1533E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 1.00002 | 0.00043 | 1.00002 | 0.00214 |
| 13 | 1.1337E-01 | 256.3 | 4.306 | 3.882 | 1.0000 | 0.0021 | 0.99745 | 0.00043 | 0.99745 | 0.00214 |
| 20 | 2.9977E-01 | 105.5 | 4.306 | 1.597 | 1.0000 | 0.0028 | 1.00328 | 0.00045 | 1.00328 | 0.00284 |
| 21 | 2.9155E-01 | 105.5 | 4.306 | 1.597 | 1.0000 | 0.0028 | 1.00326 | 0.00046 | 1.00326 | 0.00284 |
| 22 | 2.7989E-01 | 105.5 | 4.306 | 1.597 | 1.0000 | 0.0028 | 1.00318 | 0.00044 | 1.00318 | 0.00283 |
| 23 | 2.7305E-01 | 105.5 | 4.306 | 1.597 | 1.0000 | 0.0028 | 1.00147 | 0.00047 | 1.00147 | 0.00284 |
| 24 | 6.0531E-01 | 105.5 | 4.306 | 1.597 | 1.0000 | 0.0028 | 1.00041 | 0.00042 | 1.00041 | 0.00283 |
| 25 | 5.5810E-01 | 105.5 | 4.306 | 1.597 | 1.0000 | 0.0028 | 1.00153 | 0.00045 | 1.00153 | 0.00284 |
| 26 | 5.1886E-01 | 105.5 | 4.306 | 1.597 | 1.0000 | 0.0028 | 1.00143 | 0.00042 | 1.00143 | 0.00283 |
| 27 | 4.8592E-01 | 105.5 | 4.306 | 1.597 | 1.0000 | 0.0028 | 1.00315 | 0.00045 | 1.00315 | 0.00284 |
| *LEU-COMP-THERM-017* | | | | | | | | | | |
| 15 | 1.8165E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99830 | 0.00040 | 0.99830 | 0.00283 |
| 16 | 1.7571E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99839 | 0.00038 | 0.99839 | 0.00283 |
| 17 | 1.7054E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 1.00044 | 0.00041 | 1.00044 | 0.00283 |
| 18 | 1.6926E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99894 | 0.00041 | 0.99894 | 0.00283 |
| 19 | 1.6610E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99882 | 0.00040 | 0.99882 | 0.00283 |
| 20 | 1.6510E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99878 | 0.00039 | 0.99878 | 0.00283 |
| 21 | 1.6365E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99811 | 0.00041 | 0.99811 | 0.00283 |
| 22 | 1.6197E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99761 | 0.00037 | 0.99761 | 0.00282 |
| 23 | 1.7286E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99927 | 0.00041 | 0.99927 | 0.00283 |
| 24 | 1.6843E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 1.00066 | 0.00041 | 1.00066 | 0.00283 |
| 25 | 1.6103E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99879 | 0.00038 | 0.99879 | 0.00283 |
| 26 | 3.8015E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99647 | 0.00039 | 0.99647 | 0.00283 |
| 27 | 3.2488E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99816 | 0.00036 | 0.99816 | 0.00282 |
| 28 | 2.8541E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99897 | 0.00039 | 0.99897 | 0.00283 |
| 29 | 2.5582E-01 | 218.7 | 2.35 | 1.600 | 1.0000 | 0.0028 | 0.99916 | 0.00037 | 0.99916 | 0.00282 |

NOTES: [1] As calculated in MCNP6.
[2] M/F ratio determined through hand calculations.

For all the benchmark experiments, the following key input data for MCNP6 were the minimum values used in order to verify convergence of the cases:

- Neutrons per cycle: 10,000
- Number of skipped cycles: 50
- Total number of cycles: 350

This resulted in a total of 3,000,000 neutron histories analyzed per case. Convergence of the results was verified through inspection of both the Shannon entropy of the fission source distribution and the plot of $k_{eff}$ versus cycle number for each case.

### 6.9.2.2 HAC, 2N

#### 6.9.2.1.1. Moderation

The moderator density is varied from 0 to 1.0 g/cm$^3$ within the HPI cavity region and the Model 2000 cask region. The moderation space is defined as all available space within the cavities, not including any space with lead or DU shielding. The fissile content region is maintained fully flooded with full density water for both contents. The $k_{eff}$ result for the single package and package array (NCT and HAC) are very similar, indicating the shield materials of the HPI and cask provide strong reflection, thus neutronically isolating each package within the array. Therefore, full density water moderation between the packages is maintained to further increase reflection within the package.

#### 6.9.2.1.2. Fuel Rod Content

The moderator density is varied for the peak case of the fuel rod content results in Table 6.6.2-1, FRLAHmh_1_22 (OR=0.2 cm, hex, half-pitch=0.7 cm). The HPI and cask cavity moderator densities are varied independently. Peak cases for fuel rod content are provided in Table 6.9.2-2. Figure 6.9.2-1 displays the trends for all data. Based on the most limiting case, FRLAHmh_1_22w_1_9 (HPI cavity moderator density=1 g/cm$^3$, cask cavity moderator density=0 g/cm$^3$), $k_{eff} + 2\sigma$ equals 0.92413, as compared to the full density flooded HAC, package array base case of $k_{eff} + 2\sigma$ equal to 0.92398 (FRLAHmh_1_22w_9_9). The reflection from the DU HPI shields provides the dominant increase in neutron interaction within a package, while varying the water moderation within the HPI and cask cavity provides statistically similar or reduced package system reactivity. Results of the HAC package array base case (FRLAHmh_1_22) and the most limiting case here (FRLAHmh_1_22w_1_9) are statistically indifferent. Therefore, no additional uncertainty is added to the final, maximum $k_{eff}$ for moderation variation.

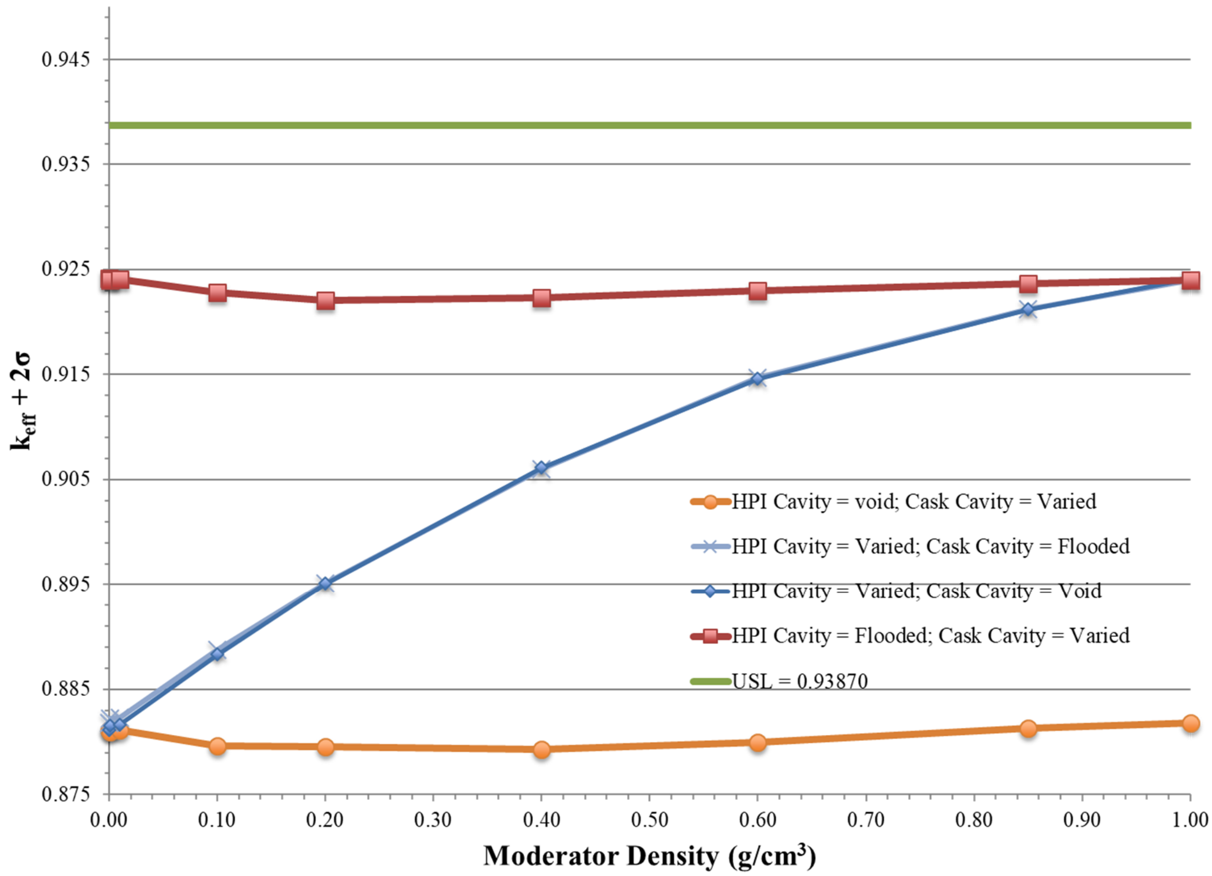


**Figure 6.9.2-1. Fuel Rod Content, HAC 2N, Moderator Variation Study**

**Table 6.9.2-2. Fuel Rod Content, HAC 2N, Moderator Variation Study**

| Case Name | HPI Cavity Water Density (g/cm$^3$) | Cask Cavity Water Density (g/cm$^3$) | $k_{eff}$ | σ | $k_{eff}$+2σ |
|---|---|---|---|---|---|
| Case for HPI cavity at 0 g/cc; cask cavity variation | | | | | |
| FRLAHmh_1_22w_1_1 | 0.000 | 0.000 | 0.88070 | 0.00023 | 0.88116 |
| FRLAHmh_1_22w_2_1 | 0.000 | 0.001 | 0.88048 | 0.00024 | 0.88096 |
| FRLAHmh_1_22w_3_1 | 0.000 | 0.010 | 0.88063 | 0.00024 | 0.88111 |
| FRLAHmh_1_22w_4_1 | 0.000 | 0.100 | 0.87915 | 0.00025 | 0.87965 |
| FRLAHmh_1_22w_5_1 | 0.000 | 0.200 | 0.87906 | 0.00023 | 0.87952 |
| FRLAHmh_1_22w_6_1 | 0.000 | 0.400 | 0.87881 | 0.00023 | 0.87927 |
| FRLAHmh_1_22w_7_1 | 0.000 | 0.600 | 0.87950 | 0.00023 | 0.87996 |
| FRLAHmh_1_22w_8_1 | 0.000 | 0.850 | 0.88086 | 0.00021 | 0.88128 |
| FRLAHmh_1_22w_9_1 | 0.000 | 1.000 | 0.88136 | 0.00024 | 0.88184 |
| Case for cask cavity at 1 g/cc; HPI cavity variation | | | | | |
| FRLAHmh_1_22w_9_1 | 0.000 | 1.000 | 0.88136 | 0.00024 | 0.88184 |
| FRLAHmh_1_22w_9_2 | 0.001 | 1.000 | 0.88181 | 0.00023 | 0.88227 |
| FRLAHmh_1_22w_9_3 | 0.010 | 1.000 | 0.88181 | 0.00024 | 0.88229 |
| FRLAHmh_1_22w_9_4 | 0.100 | 1.000 | 0.88826 | 0.00025 | 0.88876 |
| FRLAHmh_1_22w_9_5 | 0.200 | 1.000 | 0.89467 | 0.00024 | 0.89515 |
| FRLAHmh_1_22w_9_6 | 0.400 | 1.000 | 0.90556 | 0.00024 | 0.90604 |
| FRLAHmh_1_22w_9_7 | 0.600 | 1.000 | 0.91417 | 0.00025 | 0.91467 |
| FRLAHmh_1_22w_9_8 | 0.850 | 1.000 | 0.92079 | 0.00024 | 0.92127 |
| FRLAHmh_1_22w_9_9 | 1.000 | 1.000 | 0.92350 | 0.00024 | 0.92398 |
| Case for cask cavity at 0 g/cc; HPI cavity variation | | | | | |
| FRLAHmh_1_22w_1_1 | 0.000 | 0.000 | 0.88070 | 0.00023 | 0.88116 |
| FRLAHmh_1_22w_1_2 | 0.001 | 0.000 | 0.88111 | 0.00021 | 0.88153 |
| FRLAHmh_1_22w_1_3 | 0.010 | 0.000 | 0.88124 | 0.00022 | 0.88168 |
| FRLAHmh_1_22w_1_4 | 0.100 | 0.000 | 0.88777 | 0.00027 | 0.88831 |
| FRLAHmh_1_22w_1_5 | 0.200 | 0.000 | 0.89460 | 0.00024 | 0.89508 |
| FRLAHmh_1_22w_1_6 | 0.400 | 0.000 | 0.90571 | 0.00022 | 0.90615 |
| FRLAHmh_1_22w_1_7 | 0.600 | 0.000 | 0.91416 | 0.00022 | 0.91460 |
| FRLAHmh_1_22w_1_8 | 0.850 | 0.000 | 0.92073 | 0.00023 | 0.92119 |
| FRLAHmh_1_22w_1_9 | 1.000 | 0.000 | 0.92365 | 0.00024 | 0.92413 |
| Case for HPI cavity at 1 g/cc; cask cavity variation | | | | | |
| FRLAHmh_1_22w_1_9 | 1.000 | 0.000 | 0.92365 | 0.00024 | 0.92413 |
| FRLAHmh_1_22w_2_9 | 1.000 | 0.001 | 0.92353 | 0.00023 | 0.92399 |
| FRLAHmh_1_22w_3_9 | 1.000 | 0.010 | 0.92357 | 0.00024 | 0.92405 |
| FRLAHmh_1_22w_4_9 | 1.000 | 0.100 | 0.92238 | 0.00021 | 0.92280 |
| FRLAHmh_1_22w_5_9 | 1.000 | 0.200 | 0.92162 | 0.00024 | 0.92210 |
| FRLAHmh_1_22w_6_9 | 1.000 | 0.400 | 0.92185 | 0.00023 | 0.92231 |
| FRLAHmh_1_22w_7_9 | 1.000 | 0.600 | 0.92253 | 0.00022 | 0.92297 |
| FRLAHmh_1_22w_8_9 | 1.000 | 0.850 | 0.92320 | 0.00023 | 0.92366 |
| FRLAHmh_1_22w_9_9 | 1.000 | 1.000 | 0.92350 | 0.00024 | 0.92398 |

### 6.9.3. MCNP Results

This section documents an explanation of the criticality analysis MCNP input/output file structure and naming convention. Representative cases are provided for review (Tables 6.9.3-1 through 6.9.3-5).

**Table 6.9.3-1. Fuel Rod Content**

| Key | Description | |
|---|---|---|
| **FRLA(H/N)mh_XX_YY** | | |
| FRL | Fuel Rod | |
| A | A for package array | |
| H / N | H for HAC<br>N for NCT | |
| mh | mh for mass limited, hexagonal pitch | |
| XX<br>[1-4] | Fuel pellet radius (FROR) | 0.2, 0.3, 0.4, 0.5 (centimeters) |
| YY<br>[1-11]<br>(where applicable)<br>[21-23] | Half-pitch | FROR[XX]+0.3, FROR[XX]+0.4, FROR[XX]+0.6, FROR[XX]+0.7, FROR[XX]+0.8, FROR[XX]+0.9, FROR[XX]+1.1, FROR[XX]+1.2, FROR[XX]+1.4, FROR[XX]+1.6, FROR[XX]+2.0<br><br>For XX=1 added cases (Cases 21, 22, and 23, respectively) for FROR[XX]+0.45, FROR[XX]+0.5, FROR[XX]+0.55 |
| **FRLAHm_YY** | | |
| FRL | Fuel Rod | |
| A | A for package array | |
| H | H for HAC | |
| m | m for mass limited, square pitch | |
| -- | Fuel pellet radius (FROR) | 0.2 (centimeters) |
| YY<br>[1-8] | Half-pitch | FROR[XX]+0.3, FROR[XX]+0.4, FROR[XX]+0.45, FROR[XX]+0.5, FROR[XX]+0.55, FROR[XX]+0.6, FROR[XX]+0.7, FROR[XX]+0.8 |
| **FRLSmh_XX_YY** | | |
| FRL | Fuel Rod | |
| S | S for single package | |
| mh | mh for mass limited, hexagonal pitch | |
| XX<br>[1-9] | Fuel pellet radius (FROR) | 0.2, 0.3, 0.4, 0.5 (centimeters) |
| YY<br>[1-11]<br>(where applicable)<br>[21-23] | Half-pitch | FROR[XX]+0.3, FROR[XX]+0.4, FROR[XX]+0.6, FROR[XX]+0.7, FROR[XX]+0.8, FROR[XX]+0.9, FROR[XX]+1.1, FROR[XX]+1.2, FROR[XX]+1.4, FROR[XX]+1.6, FROR[XX]+2.0<br><br>For XX=1 added cases (Cases 21, 22, and 23, respectively) for FROR[XX]+0.45, FROR[XX]+0.5, FROR[XX]+0.55 |

**Table 6.9.3-2. FRLSmh MCNP Results (Section 6.4.2)**

| Input File | $k_{eff}$ | σ |
|---|---|---|
| FRLSmh_1_1_in.inp | 0.85526 | 0.00025 |
| FRLSmh_1_2_in.inp | 0.90946 | 0.00023 |
| FRLSmh_1_21.inp | 0.92134 | 0.00022 |
| FRLSmh_1_22.inp | 0.92307 | 0.00021 |
| FRLSmh_1_23.inp | 0.91659 | 0.00022 |
| FRLSmh_1_3_in.inp | 0.89990 | 0.00022 |
| FRLSmh_1_4_in.inp | 0.83653 | 0.00021 |
| FRLSmh_1_5_in.inp | 0.78051 | 0.00020 |
| FRLSmh_2_1_in.inp | 0.75241 | 0.00022 |
| FRLSmh_2_2_in.inp | 0.81435 | 0.00021 |
| FRLSmh_2_3_in.inp | 0.88079 | 0.00024 |
| FRLSmh_2_4_in.inp | 0.88881 | 0.00023 |
| FRLSmh_2_5_in.inp | 0.88239 | 0.00021 |
| FRLSmh_2_6_in.inp | 0.85986 | 0.00022 |
| FRLSmh_2_7_in.inp | 0.77499 | 0.00021 |
| FRLSmh_2_8_in.inp | 0.73769 | 0.00020 |
| FRLSmh_3_1_in.inp | 0.70705 | 0.00026 |
| FRLSmh_3_10_in.inp | 0.70002 | 0.00020 |
| FRLSmh_3_11_in.inp | 0.59209 | 0.00017 |
| FRLSmh_3_2_in.inp | 0.76489 | 0.00024 |
| FRLSmh_3_3_in.inp | 0.84482 | 0.00024 |
| FRLSmh_3_4_in.inp | 0.86684 | 0.00025 |
| FRLSmh_3_5_in.inp | 0.8784 | 0.00021 |
| FRLSmh_3_6_in.inp | 0.87962 | 0.00022 |
| FRLSmh_3_7_in.inp | 0.85379 | 0.00021 |
| FRLSmh_3_8_in.inp | 0.82821 | 0.00023 |
| FRLSmh_3_9_in.inp | 0.74879 | 0.00020 |
| FRLSmh_4_1_in.inp | 0.66359 | 0.00022 |
| FRLSmh_4_10_in.inp | 0.75879 | 0.00020 |
| FRLSmh_4_11_in.inp | 0.64905 | 0.00018 |
| FRLSmh_4_2_in.inp | 0.71248 | 0.00026 |
| FRLSmh_4_3_in.inp | 0.78917 | 0.00023 |
| FRLSmh_4_4_in.inp | 0.81556 | 0.00026 |
| FRLSmh_4_5_in.inp | 0.83477 | 0.00023 |
| FRLSmh_4_6_in.inp | 0.84636 | 0.00025 |
| FRLSmh_4_7_in.inp | 0.85075 | 0.00023 |
| FRLSmh_4_8_in.inp | 0.84337 | 0.00024 |
| FRLSmh_4_9_in.inp | 0.81279 | 0.00023 |

**Table 6.9.3-3. FRLANmh MCNP Results (Section 6.5.2)**

| Input File | $k_{eff}$ | σ |
|---|---|---|
| FRLANmh_1_1_in.inp | 0.83754 | 0.00024 |
| FRLANmh_1_2_in.inp | 0.89350 | 0.00023 |
| FRLANmh_1_21.inp | 0.90893 | 0.00024 |
| FRLANmh_1_22.inp | 0.91810 | 0.00023 |
| FRLANmh_1_23.inp | 0.91564 | 0.00020 |
| FRLANmh_1_3_in.inp | 0.90089 | 0.00020 |
| FRLANmh_1_4_in.inp | 0.83616 | 0.00020 |
| FRLANmh_1_5_in.inp | 0.77767 | 0.00019 |
| FRLANmh_2_1_in.inp | 0.73940 | 0.00030 |
| FRLANmh_2_2_in.inp | 0.80199 | 0.00027 |
| FRLANmh_2_3_in.inp | 0.87026 | 0.00026 |
| FRLANmh_2_4_in.inp | 0.88039 | 0.00026 |
| FRLANmh_2_5_in.inp | 0.88490 | 0.00023 |
| FRLANmh_2_6_in.inp | 0.86001 | 0.00023 |
| FRLANmh_2_7_in.inp | 0.77229 | 0.00019 |
| FRLANmh_2_8_in.inp | 0.73271 | 0.00021 |
| FRLANmh_3_1_in.inp | 0.68240 | 0.00025 |
| FRLANmh_3_10_in.inp | 0.69325 | 0.00019 |
| FRLANmh_3_11_in.inp | 0.57449 | 0.00018 |
| FRLANmh_3_2_in.inp | 0.74334 | 0.00022 |
| FRLANmh_3_3_in.inp | 0.81419 | 0.00025 |
| FRLANmh_3_4_in.inp | 0.84262 | 0.00023 |
| FRLANmh_3_5_in.inp | 0.85075 | 0.00025 |
| FRLANmh_3_6_in.inp | 0.86365 | 0.00022 |
| FRLANmh_3_7_in.inp | 0.84363 | 0.00024 |
| FRLANmh_3_8_in.inp | 0.82671 | 0.00023 |
| FRLANmh_3_9_in.inp | 0.74479 | 0.00022 |

## Table 6.9.3-4. FRLAHmh MCNP Results (Section 6.6.2)

| Input File | $k_{eff}$ | σ |
|---|---|---|
| FRLAHmh_1_1_in.inp | 0.85587 | 0.00026 |
| FRLAHmh_1_2_in.inp | 0.90895 | 0.00025 |
| FRLAHmh_1_21.inp | 0.92097 | 0.00022 |
| FRLAHmh_1_22.inp | 0.92350 | 0.00024 |
| FRLAHmh_1_23.inp | 0.91616 | 0.00025 |
| FRLAHmh_1_3_in.inp | 0.89981 | 0.00021 |
| FRLAHmh_1_4_in.inp | 0.83686 | 0.00019 |
| FRLAHmh_1_5_in.inp | 0.78005 | 0.00019 |
| FRLAHmh_2_1_in.inp | 0.75220 | 0.00025 |
| FRLAHmh_2_2_in.inp | 0.81358 | 0.00026 |
| FRLAHmh_2_3_in.inp | 0.88044 | 0.00022 |
| FRLAHmh_2_4_in.inp | 0.88897 | 0.00020 |
| FRLAHmh_2_5_in.inp | 0.88203 | 0.00022 |
| FRLAHmh_2_6_in.inp | 0.86057 | 0.00022 |
| FRLAHmh_2_7_in.inp | 0.77474 | 0.00021 |
| FRLAHmh_2_8_in.inp | 0.73764 | 0.00021 |
| FRLAHmh_3_1_in.inp | 0.70760 | 0.00023 |
| FRLAHmh_3_10_in.inp | 0.70081 | 0.00020 |
| FRLAHmh_3_11_in.inp | 0.59202 | 0.00018 |
| FRLAHmh_3_2_in.inp | 0.76450 | 0.00023 |
| FRLAHmh_3_3_in.inp | 0.84442 | 0.00023 |
| FRLAHmh_3_4_in.inp | 0.86718 | 0.00025 |
| FRLAHmh_3_5_in.inp | 0.87828 | 0.00023 |
| FRLAHmh_3_6_in.inp | 0.87916 | 0.00024 |
| FRLAHmh_3_7_in.inp | 0.85409 | 0.00022 |
| FRLAHmh_3_8_in.inp | 0.82809 | 0.00020 |
| FRLAHmh_3_9_in.inp | 0.74868 | 0.00020 |
| FRLAHmh_4_1_in.inp | 0.66372 | 0.00022 |
| FRLAHmh_4_10_in.inp | 0.75882 | 0.00021 |
| FRLAHmh_4_11_in.inp | 0.64908 | 0.00020 |
| FRLAHmh_4_2_in.inp | 0.71195 | 0.00024 |
| FRLAHmh_4_3_in.inp | 0.78927 | 0.00025 |
| FRLAHmh_4_4_in.inp | 0.81556 | 0.00024 |
| FRLAHmh_4_5_in.inp | 0.83475 | 0.00026 |
| FRLAHmh_4_6_in.inp | 0.84683 | 0.00023 |
| FRLAHmh_4_7_in.inp | 0.85022 | 0.00024 |
| FRLAHmh_4_8_in.inp | 0.84335 | 0.00023 |
| FRLAHmh_4_9_in.inp | 0.81274 | 0.00020 |

**Table 6.9.3-5. FRLAHm MCNP Results (Section 6.9.1)**

| Input File | $k_{eff}$ | σ |
|---|---|---|
| FRLAHm_1_1_in.inp | 0.88665 | 0.00023 |
| FRLAHm_1_2_in.inp | 0.92539 | 0.00024 |
| FRLAHm_1_3_in.inp | 0.92769 | 0.00021 |
| FRLAHm_1_4_in.inp | 0.91849 | 0.00020 |
| FRLAHm_1_5_in.inp | 0.89758 | 0.00019 |
| FRLAHm_1_6_in.inp | 0.86402 | 0.00021 |
| FRLAHm_1_7_in.inp | 0.79759 | 0.00020 |
| FRLAHm_1_8_in.inp | 0.74015 | 0.00021 |
| FRLAHm_2_1_in.inp | 0.79426 | 0.00022 |
| FRLAHm_2_2_in.inp | 0.85183 | 0.00026 |
| FRLAHm_2_3_in.inp | 0.87235 | 0.00025 |
| FRLAHm_2_4_in.inp | 0.88724 | 0.00025 |
| FRLAHm_2_5_in.inp | 0.89702 | 0.00022 |
| FRLAHm_2_6_in.inp | 0.90125 | 0.00023 |
| FRLAHm_2_7_in.inp | 0.89588 | 0.00023 |
| FRLAHm_2_8_in.inp | 0.87092 | 0.00020 |

## 6.10 References

6-1 U.S. NRC, "Code of Federal Regulations, Packaging and Transport of Radioactive Material," 10 CFR 71, April 2016.

6-2 International Atomic Energy Agency, "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2012 Edition)," SSG-26 2012.

6-3 R.J. McConn et al., "Compendium of Material Composition Data for Radiation Transport Modeling," PNNL-15870, Revision 1, March 2011.

6-4 “SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,” ORNL/TM-2005/39, Version 6.1, June 2011. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.

6-5 T. Goorley et al., "Initial MCNP 6 Release Overview - MCNP6 Version 1.0," Los Alamos National Laboratory, LA-UR-13-22934, April 2013.

6-6 J. Conlin et al., "Listing of Available ACE Data Tables," Los Alamos National Laboratory, LA-UR-13-21822, Revision 4, June 2014.

6-7 H. R., Parks, C. V. Dyer, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," Oak Ridge National Laboratory, NUREG/CR-5661, 1997.

6-8 J. J. Lichtenwalter, S. M. Bowman, M. D. DeHart, and C. M. Hopper, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages, NUREG/CR-6361, ORNL/TM-13211," Oak Ridge National Laboratory, 1997.

6-9 Organization for Economic Cooperation and Development - Nuclear Energy Agency (OECD-NEA), "International Handbook of Evaluated Criticality Safety Benchmark Experiments, NEA/NSC/DOC(95)03," 2014.

# 7 OPERATING PROCEDURES

Instructions for use of the Model 2000 Transport Package are summarized below, beginning with Section 7.1. Note that the instructions below are limited to operating and regulatory requirements. The instructions for detailed use are implemented administratively via site specific procedures. A pre-shipment engineering evaluation is implemented to ensure that the packaging, with its proposed contents, satisfies the applicable requirements of the package's license, certificate, or equivalent authorization. This evaluation includes, but is not limited to, the review of:

- Proposed contents' isotopic composition, quantities, and decay heat
- Proposed contents' form, weight, and geometry
- Shielding requirements
- Structural requirements
- Thermal requirements
- Shipping hardware (e.g., material basket and shoring devices)
- Compliance with the respective content requirements listed in Section 7.5.

Note any damage or unusual conditions to GEH. If functionality of the part is impaired, do not repair or replace without authorization from GEH.

As discussed in Section 1.2.4, additional shoring may be added, as necessary, to ensure the [[ ]] between the bottom of the cask lid and the top of the HPI does not exceed 0.25 inches.

## 7.1 Package Loading

- Use respective loading tables and guidance provided in Section 7.5 to ensure compliance with the authorized contents.
- The HPI material basket is required for shipping the Co-60 isotope rods and/or irradiated fuel.

### 7.1.1. Preparation for Loading

#### 7.1.1.1. Packaging Receipt and Inspection

#### 7.1.1.2. Removal of the Packaging from the Transport Vehicle

#### 7.1.1.3. Preparing to Load the Cask

a. Note any damage or unusual conditions to GEH. If functionality of the part is impaired, do not repair or replace without authorization from GEH.Torque the lifting ear screws to 600±20 ft-lb.

### 7.1.2. Loading of Contents

#### 7.1.2.1. Cobalt-60 Isotope Rods

This content type must be shipped according to the requirements in Section 7.5.2 or Section 7.5.4.

#### 7.1.2.2. Irradiated Hardware and Byproducts

This content type must be shipped according to the requirements in Section 7.5.1 or Section 7.5.4.

#### 7.1.2.3. Irradiated Fuel

This content type must be shipped according to the requirements in Section 7.5.3 or Section 7.5.4.

### 7.1.3. Closing the Cask and Performing Leakage Tests

#### 7.1.3.1. Removing the Cask from the Loading Area

a. Decontaminate the cask exterior surfaces to a level consistent with 49 CFR 173.443 and 10 CFR 71.87.

#### 7.1.3.2. Securing the Cask Lid

a. Torque the lid bolts to 720±30 ft-lb in a crisscross pattern to ensure equal compression of the seal.

#### 7.1.3.3. Assembly Verification Pre-Shipment Leakage Testing

a. Perform leakage testing of the cask lid closure seal and vent port and drain port threaded pipe plugs in accordance with a procedure developed by an American Society for Nondestructive Testing (ASNT) Level III examiner.

### 7.1.4. Preparation for Transport

#### 7.1.4.1. Preparing the Cask for Transport

a. Torque overpack screws to 100±5 ft-lb (dry) in 15 places (typ).
b. The Model 2000 Transport Package does not have any parts or devices that would need to be rendered inoperable pursuant to 10 CFR 71.87(h).
c. Perform the radiological survey of the package and transport vehicle consistent with 10 CFR 71.47, 71.87 and 49 CFR 173.441, 173.443.
d. Measure and document the temperature of the overpack paying particular attention to the area around the bolting ring. If any temperature reading exceeds 185°F, install the protective personnel barrier around the package, in accordance with 10 CFR 71.43.
e. Apply the security seal to the overpack.

## 7.2 Package Unloading

Operations at the unloading facility are largely the reverse of loading operations. The unloading facility shall be supplied with detailed operating procedures to cover all activities as required by 10 CFR 71.89.

### 7.2.1. Receipt of Package from Carrier

#### 7.2.1.1. Package Receipt and Inspection

Repeat Step 7.1.1.1 and perform a radiological survey in accordance with the requirements of 10 CFR 20.205 or equivalent agreement state regulations.

#### 7.2.1.2. Removal of the Package from the Transport Vehicle

#### 7.2.1.3. Preparing To Unload Contents

a. Torque the cask ear to 600±20 ft-lb.

### 7.2.2. Removal of Contents

#### 7.2.2.1. Co-60 Isotope Rods

#### 7.2.2.2. Unloading Irradiated Hardware

#### 7.2.2.3. Installing the Cask Closure Lid

#### 7.2.2.4. Removing the Cask from the Unloading Area

#### 7.2.2.5. Securing the Cask Lid

a. Repeat Section 7.1.3.2.

## 7.3 Preparation of Empty Packaging for Transport

### 7.3.1. Cask Cavity Inspection

a. Decontaminate the cavity to the limits of 49 CFR 173.428 if the cask is shipped as an empty container as defined in the regulation.

### 7.3.2. Installation of the Cask Closure Lid

a. Install the head bolts and torque to 720±30 ft-lb in a crisscross pattern to ensure equal compression of the seal.

### 7.3.3. Assembly Verification Leakage Testing

Leakage testing is not required to be performed on the empty container. As an option, leakage testing may be performed on an empty container prior to shipment for loading operations at a user facility, to assure a new seal performs as required.

### 7.3.4. Preparing the Empty Cask for Transport

Decontaminate the external surfaces of the cask to a level consistent with 49 CFR 173.427, "Empty Radioactive Materials Packaging".

## 7.4 Other Operations

There are no provisions required for any special operational controls (e.g., route, weather, mode, shipping time restrictions).

## 7.5 Appendix

The offeror is responsible for completing the loading table(s) in Sections 7.5.1 through 7.5.4 as necessary, as part of their pre-shipment evaluation review and approval process/system in advance of releasing the shipment in question.

### 7.5.1. Irradiated Hardware and Byproduct Loading Table

This section is included in order to provide clear instructions for using the Irradiated Hardware and Byproduct Loading Table. Figure 7.5.1-1 shows the Irradiated Hardware and Byproduct Loading Table with cells labeled for clear instruction for data entry. The Irradiated Hardware and Byproduct Loading Table shall be confirmed prior to any shipment of this content type.

It can be noted in this figure that:

- Column 1 is included to record each radionuclide in the Irradiated hardware or byproducts in a single shipment (with activity >1 Ci).
- Column 2 is included to record the activity of each radionuclide listed.
- Column 3 is included to demonstrate compliance with the thermal limits of the cask.
- Columns 4-11 are included to demonstrate compliance with regulatory dose rate limits for each location.
- Row A is filled out individually for each radionuclide in the shipment.
- Row B provides a summed total for each column.
- Row C provides the respective regulatory/cask limit for each column.
- Row D states whether the proposed shipment meets the respective regulatory/cask requirement. Cells in this row should be filled with either 'YES' or 'NO'. Once the Irradiated Hardware and Byproduct Loading Table is filled out entirely, if all cells in Row D say 'YES', the shipment complies with all necessary activity, thermal, and dose rate criteria.
- Row E is included to record the personnel who filled out the loading table.

Column

| Row | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Radionuclide | Activity (Ci) | Thermal Power (W) | NCT | | | | | HAC | | |
| | | | | $DR_{surf}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | | | Top | Side | Bottom | | | Top | Side | Bottom |
| A | A1 | A2 | A3 | A4 | A5 | A6 | A7 | A8 | A9 | A10 | A11 |
| | | | | | | | | | | | |
| | | | | | | | | | | | |
| | | | | | | | | | | | |
| | ⋮ | | | | | ⋮ | | | ⋮ | | |
| | | | | | | | | | | | |
| | | | | | | | | | | | |
| | | | | | | | | | | | |
| B | Total | - | B3 | B4 | B5 | B6 | B7 | B8 | B9 | B10 | B11 |
| C | Limit | - | C3 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| D | Criteria Met? | - | D3 | D4 | D5 | D6 | D7 | D8 | D9 | D10 | D11 |
| E | | | | | | Filled out by: | | | E1 | | |

**Figure 7.5.1-1. Irradiated Hardware and Byproduct Loading Table**

The Irradiated Hardware and Byproduct Table is filled out using the following procedure. Cell labels are from Figure 7.5.1-1.

1. Enter the thermal limit of 1500 W for the shipment in Cell C3.
2. Starting in Cell A1 enter the first radionuclide into the loading table. This column should simply list the radionuclide name or abbreviation (e.g., enter either 'Cobalt-60' or 'Co-60').
    - Only radionuclides (alpha, beta, and gamma emitters) with activity greater than 1 Ci must be entered into the loading table. Any neutron emitting radionuclides are limited to trace amounts, strictly from surface contamination of the hardware or byproducts are permitted for shipment. A list of radionuclides for consideration to include in the loading plan is provided in, but not limited to, Table 5.5-29.
    - Any radionuclide with all gamma emissions less than 0.3 MeV or a half-life less than 3 days is irrelevant to dose rate calculations, but should be entered in the table for thermal contributions. If the radionuclide is not included in Table 5.5-29, the thermal power multiplier can be calculated using Equation 5-12 and the Q-value for the radionuclide in the SCALE6.1 ORIGEN decay library origen.rev03.decay.data (Reference 7-1).
    - Any contents including radionuclides with an activity greater than 1 Ci that are not listed in Table 5.5-29, that also have gamma emissions greater than 0.3 MeV and a half-life greater than 3 days are not allowable for shipment.
3. In Cell A2 enter the activity in curies of the respective radionuclide.
4. In Cell A3, enter the thermal power for the radionuclide (in W). This value is calculated by multiplying the activity of the radionuclide (in Cell A2) by the thermal power multiplier of the radionuclide listed in Table 5.5-29.

5. In Cells A4 through A11, enter the dose rate contribution for the respective radionuclide (in mrem/hr) for the dose rate location of the appropriate column. This value is calculated by multiplying the activity of the radionuclide (in Cell A2) by the dose rate multiplier of the radionuclide for the respective dose rate location. Dose rate multipliers for all radionuclides that are significant to the shielding analysis are provided in Table 5.4-12 for NCT dose rates and Table 5.4-13 for HAC dose rates. Irradiated hardware and byproduct radionuclides listed in Table 5.5-7, but not in Table 5.4-12 or Table 5.4-13, are not relevant to dose rate calculations, thus cells A4 through A11 may be filled with a '0' for those radionuclides.
6. Repeat Steps 2 through 5 in the next row, filling in Columns 1 through 11, for every radionuclide that is included in the irradiated contents.
7. With the top portion of the loading table filled out, in Cell B3, sum the thermal power contributions from all radionuclides entered in Column 3 of the top portion of the loading table.
8. For Cells B4 - B11, sum the dose rate contributions from all radionuclides entered in the top portion of the loading table, for each column (e.g., for Cell B4 sum Column 4, for Cell B5, sum Column 5).
9. For Cells D3 through D11, if the respective value in Row B is less than or equal to the value in Row C, enter 'Yes', if the value in Row B is greater than the value in Row C enter 'No'.
10. If all cells in Row D say 'Yes', the proposed load of irradiated contents meet all thermal and dose rate criteria and are acceptable for shipment. If any cells in Row D say 'No', a limit has been exceeded and the proposed load of irradiated contents is not acceptable for shipment.
11. Upon completion of the Irradiated Hardware and Byproduct Table, the name of the personnel responsible for filling out the table is entered in Cell E1.

## Model 2000 Irradiated Hardware and Byproduct Loading Table

| Radionuclide | Activity (Ci) | Thermal Power (W) | NCT (mrem/hr) | | | | | HAC (mrem/hr) | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | $DR_{surface}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | | Top | Side | Bottom | | | Top | Side | Bottom |
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| | | | | | | | | | | |
| **Total** | - | | | | | | | | | |
| **Limit** | - | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| **Criteria Met?** | - | | | | | | | | | |

**Filled out by:**

### 7.5.2. Verification of Compliance for Cobalt-60 Isotope Rods

Compliance with the cask thermal and regulatory dose rate limits for the cobalt-60 isotope rod contents is demonstrated through a check of the peak activity limit across any rod and using the Cobalt-60 Isotope Rod Loading Table. The Cobalt-60 Isotope Rod Loading Table shall be confirmed prior to any shipment of this content type. It is determined that a batch of cobalt-60 isotope rods is acceptable for shipment in the Model 2000 Transport Package using the following procedure:

1. Verify that the peak cobalt-60 activity in any axial 1-inch increment in the HPI cavity is less than or equal to (≤) 17,000 Ci.
2. Enter 1500W for the thermal power limit into the ‘Limit’ row of the Cobalt-60 Isotope Rod Loading Table.
3. Enter the total cobalt-60 activity of the cobalt-60 isotope rod contents (in Ci).
4. Enter the thermal power for the cobalt-60 isotope rod contents (in W). This value is calculated by multiplying the activity of the isotope rods by the thermal power multiplier for cobalt-60 from Table 5.5-29.
5. In Cells A4 through A11, enter the dose rate contribution for the cobalt-60 isotope rod contents (in mrem/hr) for the dose rate location of the appropriate column. This value is calculated by multiplying the total activity of the isotope rods by the dose rate multiplier for the respective dose rate location. The dose rate multipliers for each dose rate location are provided in Table 5.4-16 for NCT dose rates and Table 5.4-17 for HAC dose rates.
6. Upon completion of the Cobalt-60 Isotope Rod Loading Table, the name of the personnel responsible for filling out the table is entered into the appropriate cell.
7. If the maximum dose rate is less than or equal to the dose rate limit, enter ‘Yes’ in the ‘Criteria Met?’ row, otherwise enter ‘No’.
8. If all cells in the ‘Criteria Met?’ row of the Cobalt-60 Isotope Rod Loading Table say ‘Yes’, the cobalt-60 isotope rod contents meet all regulatory/cask criteria.
9. Use Section 7.5.1 to determine if there is any significant radionuclide activity in the cobalt-60 isotope rod cladding in the shipment. If the use of Section 7.5.1 determines that there is significant radionuclide activity in the cobalt-60 isotope rode cladding, then fill out the Combined Contents Loading Table per the instructions in Section 7.5.4.
10. If the cobalt-60 isotope rod is being shipped with irradiated hardware, then fill out the Combined Contents Loading Table per the instructions in Section 7.5.4.

## Model 2000 Cobalt-60 Isotope Rod Loading Table

| Content | Activity (Ci) | Thermal Power (W) | NCT (mrem/hr) | | | | | HAC (mrem/hr) | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | $DR_{surface}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | | Top | Side | Bottom | | | Top | Side | Bottom |
| **Cobalt-60 Isotope Rod** | | | | | | | | | | |
| **Limit** | - | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| **Criteria Met?** | - | | | | | | | | | |

| **Filled out by:** |
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### 7.5.3. Irradiated Fuel

This section is included in order to provide clear instructions for using the Irradiated Fuel Loading Table. Figure 7.5.3-1 shows the Irradiated Fuel Loading Table with cells labeled for clear instruction for data entry. The Irradiated Fuel Loading Table shall be confirmed prior to any shipment of this content type. Note that Irradiated Fuel encapsulation material (e.g., cladding) must be considered as Irradiated Hardware, which is addressed in Section 7.5.1.

It can be noted in this figure that:

- Columns 1-4 are included to record information on each irradiated fuel segment that is to be sent in a single Model 2000 Transport Package.
- Column 5 is included to demonstrate compliance with the criticality limits of the cask.
- Column 6 is included to demonstrate compliance with the thermal limits of the cask.
- Columns 7-14 are included to demonstrate compliance with regulatory dose rate limits for each location.
- Row A is filled out individually for each irradiated fuel segment (as defined in Section 5.4.4.1) in the shipment.
- Row B provides a summed total for each column.
- Row C provides the respective regulatory/cask limit for each column.
- Row D states whether the proposed shipment meets the respective regulatory/cask requirement. Cells in this row should be filled with either 'YES' or 'NO'. Once the Irradiated Fuel Loading Table is filled out entirely, if all cells in Row D say 'YES', the shipment complies with all necessary criticality, thermal, and dose rate criteria.
- Row E is included to record the personnel who filled out the loading table.

Column

| Row | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 | 13 | 14 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Segment # | Segment Length (inches) | Initial Enrichment (wt% U-235) | Burnup (GWd/MTU) | Mass U-235 (g) | Thermal Power (W) | NCT | | | | | HAC | | |
| | | | | | | | $DR_{surf}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | | | | | | Top | Side | Bottom | | | Top | Side | Bottom |
| A | A1 | A2 | A3 | A4 | A5 | A6 | A7 | A8 | A9 | A10 | A11 | A12 | A13 | A14 |
| | | | | | | | | | | | | | | |
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| B | Minimum | B2 | | Total | B5 | B6 | B7 | B8 | B9 | B10 | B11 | B12 | B13 | B14 |
| C | Limit | 5.3 | | Limit | 1750 | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| D | Meets Criteria? | D2 | | Meets Criteria? | D5 | D6 | D7 | D8 | D9 | D10 | D11 | D12 | D13 | D14 |
| E | | | | | | | | Filled out by: | | | | E1 | | |

**Figure 7.5.3-1. Irradiated Fuel Loading Table**

The Irradiated Fuel Loading Table is filled out using the following procedure. Cell labels are from Figure 7.5.3-1.

1. Starting in Cell A1, enter the segment number identifier for the first segment of the loading table. This number is a label for the segment and can be any unique identifying number, or start with “1”.

2. In Cell A2 enter the active fuel length of the respective irradiated fuel segment. Every segment loaded into the cask, and entered in this loading table must have an active length of at least 5.3 inches.

3. In Cell A3 enter the initial enrichment range, considering the minimum initial enrichment (in wt% U-235) for the respective segment. Depending on the minimum initial enrichment of the segment, this cell should be filled according to Table 7.5.3-1 (e.g., if the initial enrichment for the respective segment is 2.7 wt% U-235, the label ‘2.5≤ e <3.0’ should be entered into cell A3).

**Table 7.5.3-1. Irradiated Fuel Loading Table Column 3 Labels**

| Initial Enrichment Ranges (wt% U-235) |
|---|
| 1.5≤ e <2.0 |
| 2.0≤ e <2.5 |
| 2.5≤ e <3.0 |
| 3.0≤ e <3.5 |
| 3.5≤ e <4.0 |
| 4.0≤ e <4.5 |
| 4.5≤ e <5.0 |
| 5.0≤ e <5.5 |
| 5.5≤ e <6.0 |

4. In Cell A4 enter the burnup range, considering maximum burnup (in GWd/MTU) for the respective segment. Depending on the maximum burnup of the segment, this cell should be filled according to Table 7.5.3-2 (e.g., if the burnup for the respective segment is 45 GWd/MTU, the label '40< b ≤50' should be entered into cell A4).

**Table 7.5.3-2. Irradiated Fuel Loading Table Column 4 Labels**

| Burnup Ranges (GWd/MTU) |
|---|
| 0< b ≤10 |
| 10< b ≤20 |
| 20< b ≤30 |
| 30< b ≤40 |
| 40< b ≤50 |
| 50< b ≤60 |
| 60< b ≤72 |

5. In Cell A5 enter the initial mass of U-235 for the respective rod segment (in gU-235).
6. In Cell A6, enter the thermal contribution for the respective segment (in W). This value is calculated by multiplying the mass of U-235 for the rod (in Cell A5) by the respective thermal power multiplier in Table 5.5-5. The appropriate thermal power multiplier in Table 5.5-5 is determined based on the initial enrichment range (in Cell A3) and burnup range (in Cell A4) for the respective segment (e.g., if the segment's initial enrichment range is '2.5≤ e <3.0' and the burnup range is '40< b ≤50', the thermal power multiplier from Table 5.5-5 is 1.264E+00 W/gU-235).
7. In Cells A7 through A14, enter the dose rate contribution for the respective segment (in mrem/hr) for the dose rate location in the appropriate column. This value is calculated by

multiplying the mass of U-235 for the rod (in Cell A5) by the dose rate multiplier for the respective dose rate location. Table 7.5.3-3 summarizes the information for filling out Cells A7 through A14 and provides an example dose rate contribution multiplier for the example scenario of a segment with an initial enrichment range of ‘2.5≤ e <3.0’ in Cell A3 and a burnup range of ‘40< b ≤50’ in Cell A4.

**Table 7.5.3-3. Irradiated Fuel Loading Table Dose Rate Multipliers**

| Irradiated Fuel Loading Table Cell Label | Dose Rate Location | Multiplier Table (in Chapter 5) | Example Multiplier[1] (mrem/hr/gU-235) |
|---|---|---|---|
| A7 | NCT Top Surface | 5.4-3 | 8.069E-02 |
| A8 | NCT Side Surface | 5.4-4 | 4.635E-01 |
| A9 | NCT Bottom Surface | 5.4-5 | 2.696E-01 |
| A10 | NCT 2-meter | 5.4-6 | 1.100E-02 |
| A11 | NCT Cab | 5.4.7 | 1.968E-03 |
| A12 | HAC Top 1-meter | 5.4-8 | 7.155E-02 |
| A13 | HAC Side 1-meter | 5.4-9 | 8.755E-02 |
| A14 | HAC Bottom 1-meter | 5.4-10 | 7.516E-02 |

Note: [1] Multiplier based on example case of segment with initial enrichment range 2.5≤ e <3.0 and initial burnup range 40< b ≤50

8. Repeat Steps 1 through 7 in the following row, filling in Columns 1 through 14, for each segment that shall be included in the shipment.
9. When the top portion of the loading table is filled out, enter the minimum active fuel length from all the segments in Column 2 in Cell B2.
10. In Cell B5, sum the masses of U-235 from all segments entered in Column 5 of the loading table.
11. In Cell B6, sum the thermal power contributions from all segments entered in Column 6 of the loading table.
12. For Cells B7 - B14, sum the dose rate contributions from all segments entered in the loading table, for each column (e.g., for Cell B7 sum Column 7, for Cell B8, sum Column 8).
13. For Cell D2, if the value in Cell B2 is greater than or equal to the respective value in Row C, write ‘YES’, if not write ‘NO’.
14. For Cells D5 through D14, if the respective value in Row B is less than or equal to the value in Row C, write ‘YES’, if not write ‘NO’.
15. If all cells in Row D say ‘YES’, the proposed load of irradiated fuel segments meets all criticality, thermal, and dose rate criteria and is acceptable for shipment. If any cells in Row D say ‘NO’, a limit has been exceeded and the proposed load of irradiated fuel segments is not acceptable for shipment.

16. Upon completion of the Irradiated Fuel Table, the name of the personnel responsible for filling out the table is entered in Cell E1.
17. If the irradiated fuel is encapsulated (e.g., cladding), that material is treated as irradiated hardware; fill out the irradiated hardware/byproduct (Section 7.5.1) and combined loading (Section 7.5.4) tables as needed. It should be noted that irradiated hardware and irradiated fuel have different cooling time requirements.

The following page provides the Irradiated Fuel Loading Table that is to be filled out prior to any shipment of this content type.

## Model 2000 Irradiated Fuel Loading Table

| **Segment #** | **Segment Length (inches)** | **Initial Enrichment (wt% U-235)** | **Burnup (GWd/MTU)** | **Mass U-235 (g)** | **Thermal Power (W)** | **NCT (mrem/hr)** | | | | | **HAC (mrem/hr)** | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | **$DR_{surf}$** | | | **$DR_{2m}$** | **$DR_{cab}$** | **$DR_{1m}$** | | |
| | | | | | | **Top** | **Side** | **Bottom** | | | **Top** | **Side** | **Bottom** |
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| | | | | | | | | | | | | | |
| **Minimum** | | | **Total** | | | | | | | | | | |
| **Limit** | 5.3 | | **Limit** | 1750 | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| **Meets Criteria?** | | | **Meets Criteria?** | | | | | | | | | | |

**Filled out by:**

### 7.5.4. Combined Contents

The Combined Contents Loading Table shall be confirmed prior to any shipment including multiple content types. For any shipment including multiple content types, compliance with regulatory/cask limits is demonstrated using the following procedure:

1. Fill out the Irradiated Hardware and Byproduct Loading Table per instructions in Section 7.5.1, as applicable.
2. Fill out the Cobalt-60 Isotope Rod Loading Table per instructions in Section 7.5.2, as applicable.
   2.1 Confirm that the total activity of all cobalt-60 in the shipment is less than or equal to (≤) 17,000 Ci per axial inch. Sum the peak cobalt-60 Ci per axial inch value from Step 1 of Section 7.5.2 and the total cobalt-60 point source values from the Irradiated Hardware and Byproduct Loading Table.
3. Fill out the Irradiated Fuel Loading Table per instructions in Section 7.5.3, as applicable. Note that Irradiated Fuel encapsulation material (e.g., cladding) must be considered as Irradiated Hardware, which is addressed in Section 7.5.1.
4. Enter the U-235 mass, thermal power, and dose rate values from the 'Total' row of each applicable loading table from Steps 1, 2, and 3 into the respective 'Hardware/Byproduct', 'Cobalt-60 Isotope Rod,' and 'Irradiated Fuel' rows in the Combined Contents Loading Table.
5. Sum the U-235 mass, thermal power, and dose rate values from all content types in the 'Total' row of the Combined Contents Loading Table.
6. Verify that for each column the value in the 'Total' row is less than or equal to the value in the 'Limit' row. Record this verification by writing 'YES' if the criteria is met, or 'NO' if the criteria is not met.
7. If all cells in the 'Criteria Met?' row of the Irradiated Hardware and Byproduct Loading Table, the Cobalt-60 Isotope Rod Loading Table, the Irradiated Fuel Loading Table, and the Combined Contents Loading Table say 'YES', the proposed load of combined contents meets all regulatory/cask criteria and is acceptable for shipment. If any cells in any of the four Tables say 'NO', a limit has been exceeded and the proposed load of combined contents is not acceptable for shipment.
8. Upon completion of the Combined Contents Loading Table, the name of the personnel responsible for filling out the table is entered into the appropriate cell.

## Model 2000 Combined Contents Loading Table

| Content | Mass U-235 (g) | Thermal Power (W) | NCT (mrem/hr) | | | | | HAC (mrem/hr) | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | $DR_{surf}$ | | | $DR_{2m}$ | $DR_{cab}$ | $DR_{1m}$ | | |
| | | | Top | Side | Bottom | | | Top | Side | Bottom |
| Hardware / Byproduct | - | | | | | | | | | |
| Cobalt-60 Isotope Rods | - | | | | | | | | | |
| Irradiated Fuel | | | | | | | | | | |
| Total | | | | | | | | | | |
| Limit | 1750 | 1500 | 180 | 180 | 180 | 9 | 1.8 | 900 | 900 | 900 |
| Criteria Met? | | | | | | | | | | |

**Filled out by:**

## 7.6 References

7-1 Oak Ridge National Lab, "SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design, ORNL/TM-2005/39, Version 6, Vols. I-III," ORNL/TM-2005/39, Version 6.1, June 2011.