ENCLOSURE 5

M190195

Amended Sections and Pages for NEDO-33866 GE2000 SAR Revision 5

Non-Proprietary Information

IMPORTANT NOTICE

This is a non-proprietary version of Enclosure 4 to M190195, 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 [[ ]].

   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 5 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.

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 5 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).

The [[ ]] irradiated fuel contents is based on the radionuclide inventory generated from the irradiation and decay of various nuclides over time. The gamma source strength is dependent on the enrichment (E) band and burnup (B) band. In the ORIGEN-S 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 strength for each burnup-enrichment pairing. For the calculated source strength for each burnup-enrichment pairing the basis is 1 gram of U-235.

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

Similar to the cobalt-60 isotope rods, the NCT irradiated fuel content source geometry is modeled as a single 5.3-inch line source to represent the entire irradiated fuel content. Within this single 5.3-inch line source the entire irradiated fuel photon and neutron sources are uniformly distributed. 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 single uniform line source is acceptable to represent the entire irradiated fuel content 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 the entire irradiated fuel content 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. See Section 5.4.4.1 for how the external dose rates are determined for irradiated fuel.

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.

5.4 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).

The neutron models are voided and unshielded; therefore, 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. It is dependent on the spectrum [[ ]] and is normalized by its source strength [[ ]]. 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. Note that the source strengths in these equations are normalized to one.

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

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

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.

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

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

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 because 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 $\ddot{D}R(r,B|E)$ is the dose rate per gram of U-235 at a regulatory dose rate location of interest, r, at the specific burnup and enrichment band, B|E, and is calculated [[

]] with the ORIGEN-S calculated source strength,

[[

]] and 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.

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

where

ϕ	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	σ	Standard deviation
$\mathcal{R}$	Flux-to-dose-rate conversion factor	R_σ	Dose rate response with 2σ uncertainty
fsd	MCNP6 fractional standard deviation	S	Calculated source strength (ORIGEN-S)
$\ddot{D}R$	Dose rate per gram U-235	DR	Total dose rate
B|E	Burnup/Enrichment pairing		
m	Mass of U-235		

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. First a dose rate [[ ]] is calculated in MCNP6 for photons and analytically for neutrons. [[ ]] The ORIGEN-S source [[ ]] that leads to the maximum dose rate is always used [[ ]] 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 result is a set of dose rates in mrem/hr [[ ]]

Separately, ORIGEN-S is used to calculate the source strength [[ ]] per g U-235 for a given enrichment/burnup combination for photons and neutrons. The source strength is [[ ]] per g U-235 for every [[ ]] enrichment/burnup combination for photons and neutrons. 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 $\dot{D}\dot{R}$ (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	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

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

Side Surface $\dot{D}\dot{R}$ (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	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

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

Bottom Surface $\dot{D}\dot{R}$ (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	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

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

2 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	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

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

Cab $\dot{D}\dot{R}$ (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	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

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

Top 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	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

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

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

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 5.5-1. 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

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 using the 18-group energy structure for gammas (18GrpSCALE5) and the 44-group energy structure for neutrons (44GrpENDF5). 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 source [[ ]] is included in Table 5.5-4 for the 4.0 ≤ E < 4.5 enrichment band and 30 < B ≤ 40 burnup band, [[

]]

Table 5.5-4. Example Source [[ ]]

[[ ]]	**4.0 wt% U-235, 40 GWd/MTU** [[ ]]
5.00E-02	1.119E+12
1.00E-01	3.550E+11
2.00E-01	4.053E+11
3.00E-01	9.168E+10
4.00E-01	6.905E+10
6.00E-01	4.700E+11
8.00E-01	1.694E+12
1.00E+00	9.553E+10
1.33E+00	3.235E+10
1.66E+00	1.476E+10
2.00E+00	1.916E+09
2.50E+00	7.715E+09
3.00E+00	2.018E+08
4.00E+00	9.983E+06
5.00E+00	5.388E+02
6.50E+00	2.160E+02
8.00E+00	4.235E+01
1.00E+01	8.985E+00

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

5.5.2. 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

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

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

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

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

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 Sections 6.4, 6.5, 6.6, and 6.9.4 show that as the fuel rod outer radius increases the overall system reactivity decreases, thus, the fuel pellet outer radius of 0.392 cm is set as the minimum fuel pellet outer radius.

While Sections 6.4 through 6.6 form the initial basis for the criticality analysis, additional sensitivity studies provided in Section 6.9.4 determine the limitations on the allowed content. The most reactive summary results presented in Table 6.1.2-1 are from the evaluations presented in Section 6.9.4. Table 6.1.2-2 provides justification for maintaining the administratively reduced U-235 mass limit of 1750 grams from Section 6.3.1.1.1.

Fissile material evaluations and limitations are based on fresh, unirradiated fuel (no credit is applied for burnup).

Table 6.1.2-1. Most Reactive Fuel Rod Content Summary at 5 wt.% U-235

Case Name	Fuel OR (cm)	Half-pitch (cm)	k_{eff}	σ	$k_{eff}+2σ$	Maximum k_{eff} [1]	H/U-235	EALF (eV)[2]	U-235 Mass(g)
Single Package									
5p-r392-p070	0.392	0.7	0.91594	0.00025	0.91644	0.92560	684.27	1.08	1500
NCT, 5N Package Array									
5p-r392-p070_NCT_5N	0.392	0.7	0.91127	0.00009	0.91145	0.92056	684.27	1.11	1500
HAC, 2N Package Array									
5p-r392-p070_HAC_2N	0.392	0.7	0.91669	0.00011	0.91691	0.92608	684.27	1.07	1500

NOTES: USL is defined as 0.9370 per Section 6.3.4.
[1] Maximum k_{eff} includes the applied 1% k_{eff} uncertainty for pitch geometric modeling –see Section 6.9.1.
[2] EALF is defined as energy of average lethargy causing fission.

Table 6.1.2-2. Most Limiting Fuel Rod Content Summary using the Administrative U-235 Mass Limit

Case Name	Fuel OR (cm)	Half-pitch (cm)	k_{eff}	σ	$k_{eff}+2σ$	Maximum k_{eff} [1]	H/U-235	EALF (eV)	U-235 Mass Limit (g) [2]
Single Package									
5p-r392-p065	0.392	0.65	0.90973	0.00030	0.91033	0.91943	586.40	1.37	1750
NCT, 5N Package Array									
5p-r392-p065_NCT_5N	0.392	0.65	0.90421	0.00012	0.90445	0.91349	586.40	1.43	1750
HAC, 2N Package Array									
5p-r392-p065_HAC_2N	0.392	0.65	0.90999	0.00014	0.91027	0.91937	586.40	1.37	1750

NOTES: USL is defined as 0.9370 per Section 6.3.4.
[1] Maximum k_{eff} includes the applied 1% k_{eff} uncertainty for pitch geometric modeling –see Section 6.9.1.
[2] An administratively reduced fissile mass limit is defined –see Section 6.3.1.1.1 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. The 6.0 wt% U-235 enrichment was only used for those studies presented in Sections 6.4 through 6.6. Section 6.9.4 reduced the enrichment to 5 wt% U-235 to align with the Section 6.8 benchmarking and allowed content. 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. 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. The 6.0 wt% U-235 enrichment was only used for those initial studies presented in Sections 6.4 through 6.6. Section 6.9.4 reduced the enrichment to 5 wt% U-235 to align with the Section 6.8 benchmarking and allowed content. See Section 6.3.2 for material properties.

The fuel rod was initially 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) in Sections 6.4 through 6.6. Section 6.9.4 provides additional sensitivity studies varying the axial length and OR of the fuel rod. The modeled axial length bounds the requirement of minimum 5.3-inch rod length segments specified by the shielding analysis. The fuel rod OR was initially 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, 6.5, 6.6, and 6.9.4 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. The limiting evaluations in Section 6.9.4 assessed a range of UO_2 masses (105 kg to 4 kg) with a U-235 enrichment of 5 wt%. A circular boundary, which equates to a 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 a range of U-235 mass less than 4,600 grams to 180 grams. 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 in cm
M_{UO2} is the mass of UO_2 in grams
ρ_{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. Initial 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

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Figure 6.3.1-1. Fuel Rod Content Model Geometry

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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.

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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

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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 $S(\alpha,\beta)$ 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/cm³)
UO_2, 5 wt% U-235	U-235	92235	4.4071E-02	10.96 [1]
	U-238	92238	8.3735E-01	
	O-16	8016	1.1857E-01	

References:
[1] Reference 6-4 Density, Table M8.2.4.

6.3.3. Computer Codes and Cross Section Libraries

The criticality safety analysis was completed using MCNP6 Version 1.0 (Reference 6-5) and 2.0 (Reference 6-6) with the continuous-energy neutron data library ENDF/B-VII.1 (Reference 6-7). 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 6-8).

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 6-9):

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

where
Δk_m is the additional margin to ensure subcriticality (0.05)
β 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 β 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. The H/U-235 ratio from the most limiting case for the allowed content in Section 6.9.4 was used to determine the USL. As shown in Table 6.1.2-1, the limiting cases for the single package and package array have a H/U-235 value of 684.27. 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 0.9370 for fuel rod contents (e.g., 0.9473 - 1.5031E-5*684.27 = 0.93701 for X > 214.9).

$$\frac{H}{U^{235}} Ratio = WTF \times \frac{H_2O\ Density}{UO_2\ Density} \times \frac{(1 - Enr.U^{235})MW_{U^{238}} + Enr.U^{235}MW_{U^{235}} + 2MW_{O_2}}{MW_{H_2O}} \times \frac{2}{1} \times \frac{1}{Enr.U^{235}}$$

where

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

6.4 Single Package Evaluation

It should be noted that the following evaluations conducted with a U-235 enrichment of 6 wt% form the initial limiting configuration for the allowed content analyses presented in Section 6.9.4.

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. 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. Section 6.3.4 defines the USL value as 0.9370 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.

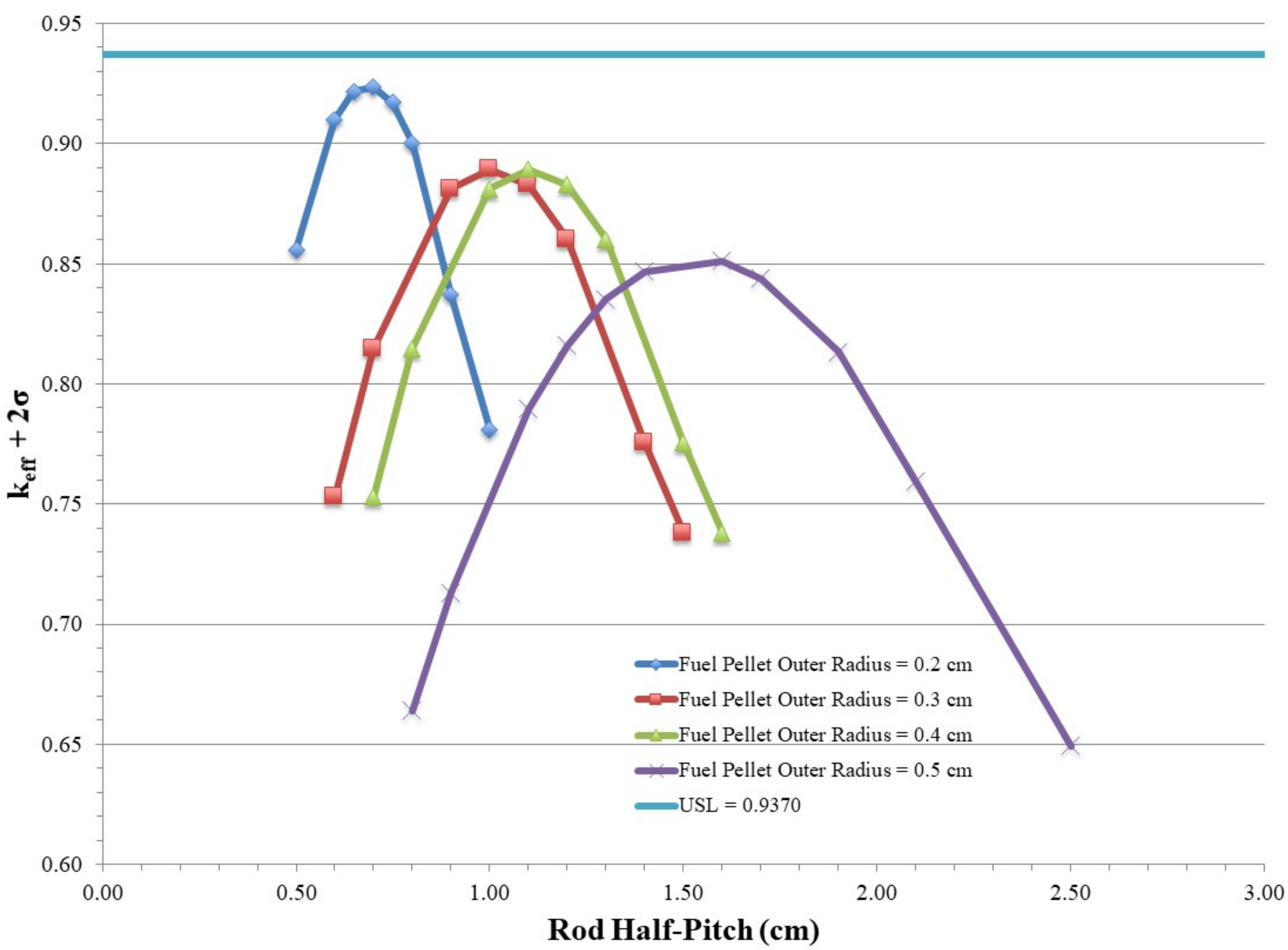

Figure 6.4.2-1. Fuel Rod Content, Single Package, Results

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	≤5.0	100%

6.5 Evaluation of Package Arrays Under Normal Conditions of Transport

It should be noted that the following evaluations conducted with a U-235 enrichment of 6 wt% form the initial limiting configuration for the allowed content analyses presented in Section 6.9.4.

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. Data trends show that as the fuel rod OR increases the overall system reactivity decreases.

Section 6.3.4 defines the USL value as 0.9370 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	$k_{eff}+2σ$	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

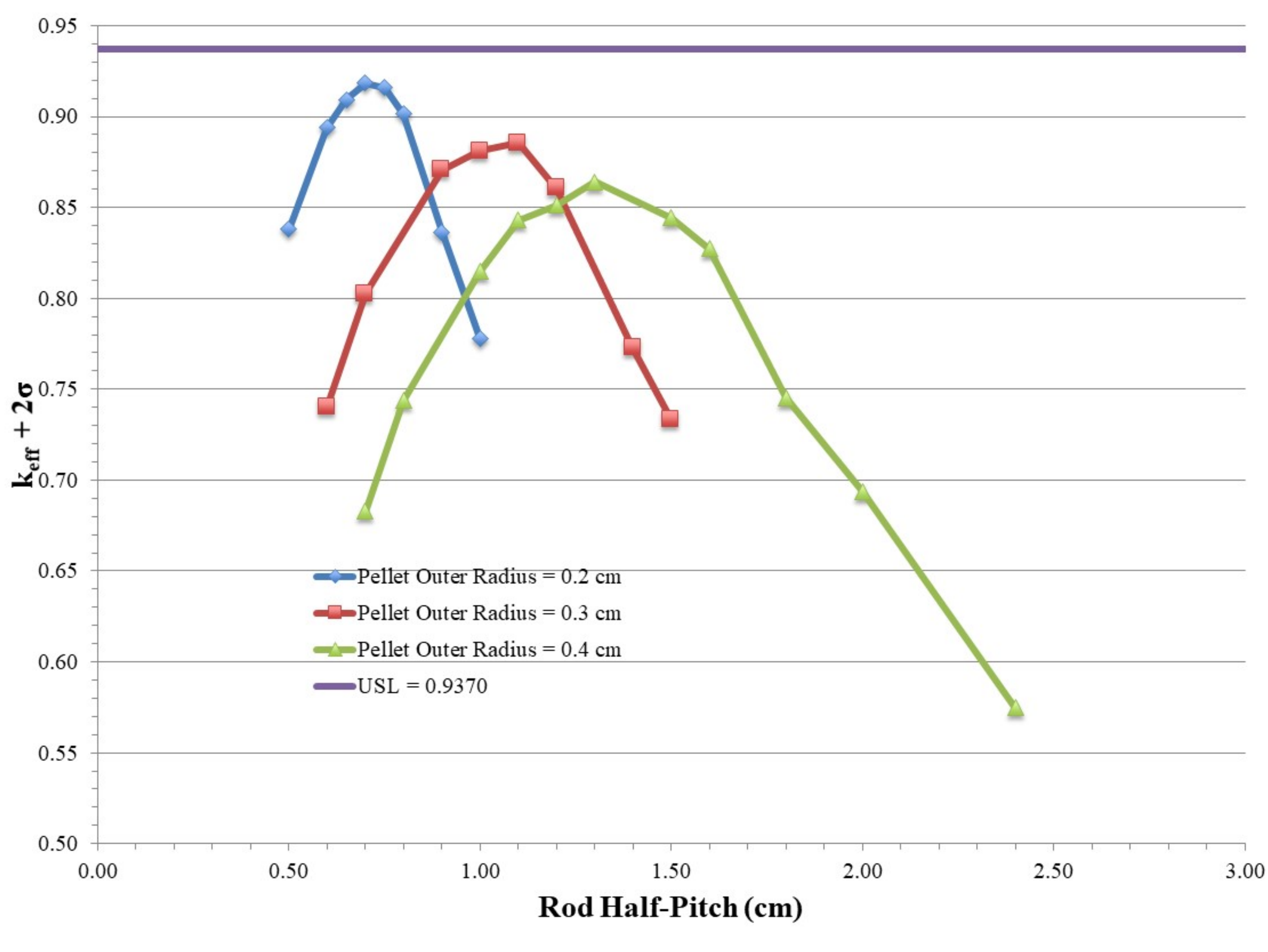

Figure 6.5.2-1. Fuel Rod Content, NCT 5N, Results

6.6 Package Arrays under Hypothetical Accident Conditions

It should be noted that the following evaluations conducted with a U-235 enrichment of 6 wt% form the initial limiting configuration for the allowed content analyses presented in Section 6.9.4.

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 $k_{eff} + 2\sigma$ is 0.92398. Data trends show that as the fuel rod OR increases the overall system reactivity decreases.

Section 6.3.4 defines the USL value as 0.9370 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	$k_{eff}+2\sigma$	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.

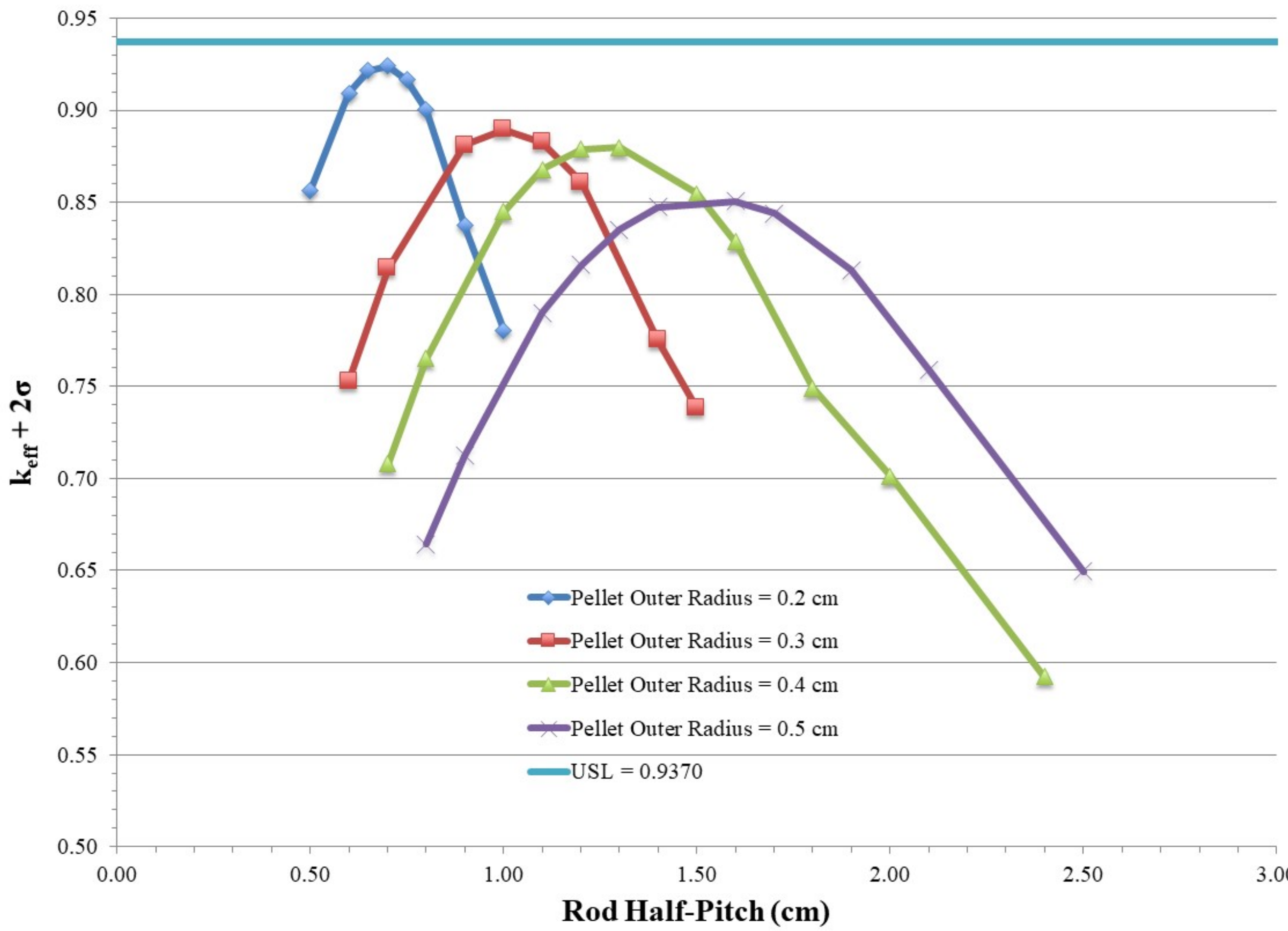

Figure 6.6.2-1. Fuel Rod Content, HAC 2N, Results

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 6-7) 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. It should be noted that the evaluations presented in Section 6.9.4 used MCNP6 Version 2.0 (Reference 6-6), which produce identical results as MCNP6 Version 1.0 (Reference 6-5) as documented via a direct code-to-code comparison by Los Alamos National Laboratory (Reference 6-10). Therefore, the bias and bias uncertainty as determined using MCNP6 Version 1.0 remains applicable to those k_{eff} values calculated using MCNP6 Version 2.0 in Section 6.9.4.

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 36 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 5 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 4.306 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 6-9), 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 0.9370, 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) 0.9441 (X ≤ 214.9)	105.5 ≤ X ≤ 256.3	9.5216E-3	-0.3900

The administrative margin, Δk_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.

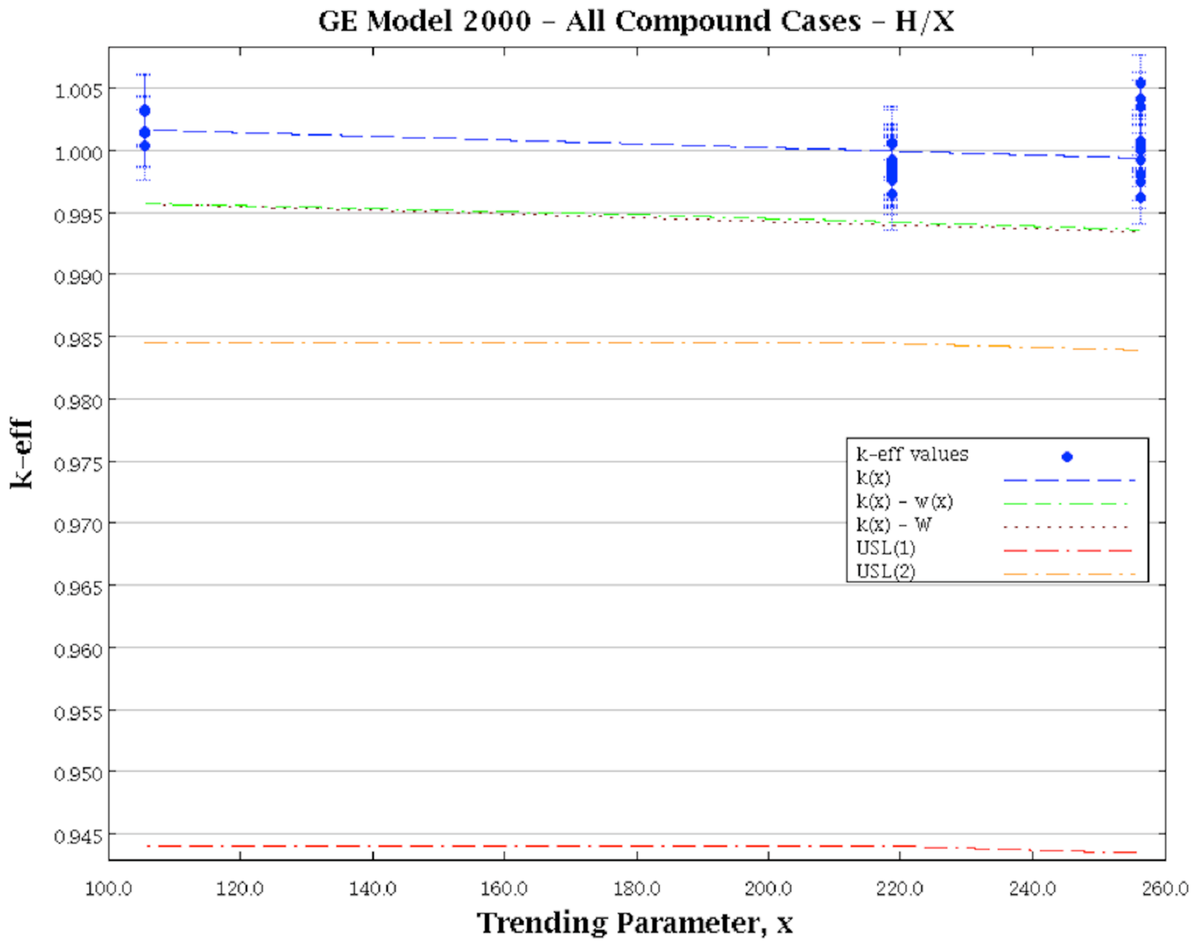

Figure 6.8.2-1. USLSTATS Trend Plot of H/U-235 versus k_{norm} – Lattice Systems

6.8.2.2.1. Test for Normality of Benchmark Data

USLSTATS determined the benchmark experiment data to be normal. USLSTATS tests for normality using the Chi-squared method (Reference 6-11). To verify the USLSTATS result, the Shapiro-Wilk method was used as a secondary test to confirm the normality of the data. This method provided no evidence to reject the null hypothesis that the data (datum) are normal.

The Shapiro-Wilk (Reference 6-12) test is a method to verify normality that is valid for samples between 3 and 50 elements in size. It uses the null hypothesis principle to verify whether a sample of size $n, x_1, x_2, \ldots, x_n$ came from a normally distributed population. In order for this hypothesis to be rejected, the calculated p-value of the data set must be less than the chosen alpha level of the test. For this test, the chosen alpha level is 0.05 and the null hypothesis is that the sample comes from a normally distributed population. In order to determine the p-value, the W test for normality is evaluated:

$$W = \frac{b^2}{S^2}$$

where,

$$b = \sum_{i=1}^{k} a_i \left(x_{n+1-i} - x_i\right)$$

$$S^2 = \sum_{i=1}^{n} (x_i - \bar{x})^2$$

- The values of x_i represent individual values of k_{norm} and are sorted into ascending order.
- $\bar{x}$ is the mean of the values of k_{norm}.
- n is equivalent to the size of the k_{norm} sample.
- k is equivalent to n/2 if n is even and equivalent to (n-1)/2 if n is odd.
- The values of a_i come from Table 6.8.2-2 with n=36.

If the p-value of the W test is less than the alpha level, then the null hypothesis must be rejected, which signifies the data do not come from a normally distributed population. If the p-value of the W test is greater than the alpha level, then there is no evidence to reject the null hypothesis that the data come from a normally distributed population.

Using this method, analyzing all of the critical lattice experiment k_{norm} data resulted in a W test equivalent to 0.95169. Using Table 6.8.2-3 with n=36, this value falls between p-values of 0.1 and 0.5, which correspond to W test values of 0.945 and 0.970, respectively. Linear interpolation was used to determine the p-value corresponding to a W test value of 0.95169, which was determined to be 0.2071. This p-value of 0.2071 associated with a value of W of 0.95169 is greater than the alpha level of 0.05. As a result, no evidence exists to reject the null hypothesis that the sample comes from a normally distributed population; therefore, the results of the USLSTATS test for normality are supported.

Table 6.8.2-2. Shapiro-Wilk a_i Coefficients for the *W* Test (Reference 6-12)

	n = 36
a1	0.4068
a2	0.2813
a3	0.2415
a4	0.2121
a5	0.1883
a6	0.1678
a7	0.1496
a8	0.1331
a9	0.1179
a10	0.1036
a11	0.0900
a12	0.0770
a13	0.0645
a14	0.0523
a15	0.0404
a16	0.0287
a17	0.0172
a18	0.0057

Table 6.8.2-3. Shapiro-Wilk Percentage Points of the *W* Test (Reference 6-12)

n	Level [p-value]								
	0.01	0.02	0.05	0.1	0.5	0.9	0.95	0.98	0.99
36	0.912	0.922	0.935	0.945	0.970	0.984	0.986	0.989	0.990

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 reactivity difference between the lattice pitch geometry is minimal (0.92811 – 0.92398 = 0.00413). However, a conservative 1.0% uncertainty is applied to k_{eff} to determine the 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	$k_{eff} + 2\sigma$	H/U-235	Mass U-235 (g)	k_{eff}	σ	$k_{eff} + 2\sigma$	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

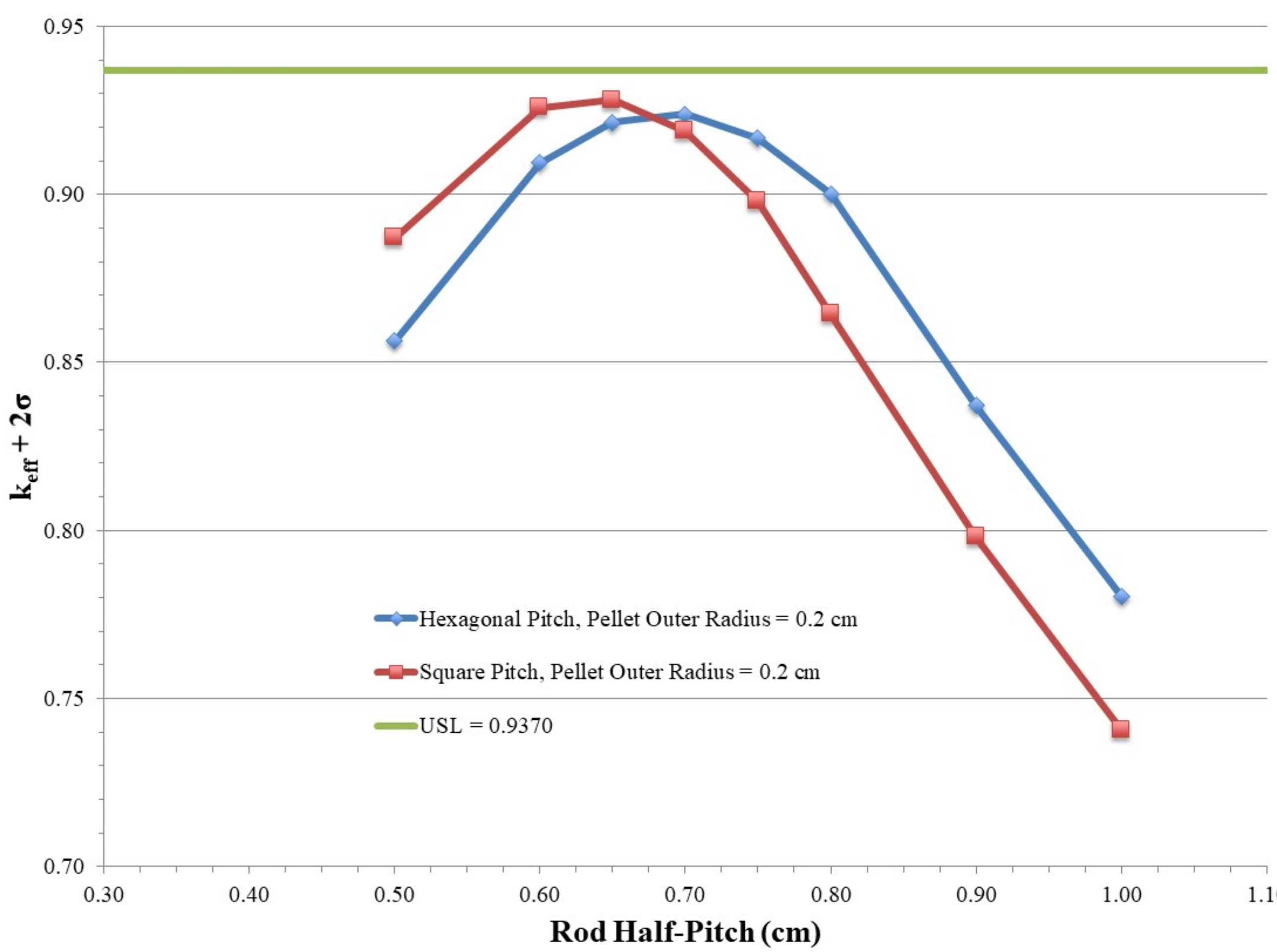

Figure 6.9.1-2. HAC, Fuel Rod Pitch Comparison

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 6-13). 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 σ_{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 σ_{calc} are the effective multiplication factor and the associated Monte Carlo 1σ determined by MCNP6, respectively. The USLSTATS input values of k and σ_{sample} for each critical experiment correspond to k_{norm} and σ_{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, σ_{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}	σ_{bench}	k_{calc}	σ_{calc}	k_{norm}	σ_{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.

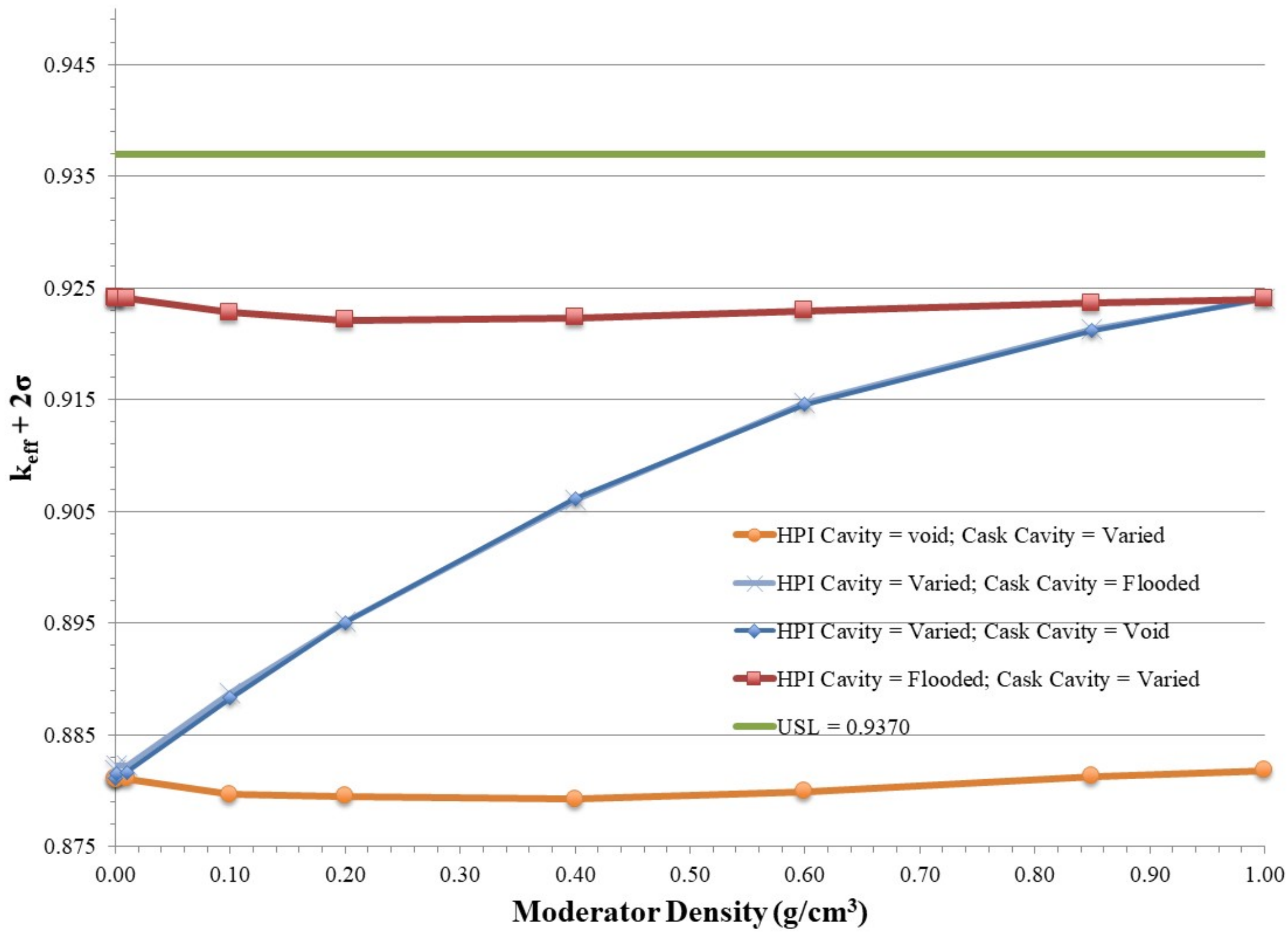

Figure 6.9.2-1. Fuel Rod Content, HAC 2N, Moderator Variation Study

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 6.9.3-7).

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 N for NCT	
mh	mh for mass limited, hexagonal pitch	
XX [1-4]	Fuel pellet radius (FROR)	0.2, 0.3, 0.4, 0.5 (centimeters)
YY [1-11] (where applicable) [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 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 [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 [1-9]	Fuel pellet radius (FROR)	0.2, 0.3, 0.4, 0.5 (centimeters)
YY [1-11] (where applicable) [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 For XX=1 added cases (Cases 21, 22, and 23, respectively) for FROR[XX]+0.45, FROR[XX]+0.5, FROR[XX]+0.55
Xp-rYYY-pZZZ		
X	Enrichment wt%	
p	Pellet	
r	Pellet radius	
YYY	Fuel pellet radius	0.35, 0.392, 0.4, 0.45, 0.55, 0.65 (centimeters)
p	Rod-to-Rod half pitch	
ZZZ	Half-pitch	0.4, 0.5, 0.6, 0.652, 0.7, 0.8, 0.9, 1.0, 1.1, 1.2, 1.4, 1.6, 2.0 (centimeters)

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

Table 6.9.3-6. Limiting Fuel Pellet Radius MCNP Results (Section 6.9.4.2)

Input File	k_{eff}	σ
5p-r300	0.96430	0.00027
5p-r350	0.94933	0.00026
5p-r392	0.91594	0.00025
5p-r400	0.90857	0.00028
5p-r450	0.85074	0.00011
5p-r500	0.78154	0.00030
5p-r550	0.70866	0.00025
5p-r600	0.63906	0.00023
5p-r650	0.57870	0.00021
5p-r700	0.53421	0.00018
6p-r200	0.92378	0.00022
6p-r209	0.88889	0.00026
6p-r219	0.95175	0.00016
6p-r231	0.92122	0.00023
6p-r245	0.97791	0.00018
6p-r262	0.95486	0.00028
6p-r283	0.99438	0.00027
6p-r310	0.98303	0.00025
6p-r346	0.97672	0.00030
6p-r400	0.93065	0.00014
6p-r490	0.81372	0.00026

Table 6.9.3-7. Optimal Fuel Rod Pitch MCNP Results (Section 6.9.4.3)

Input file	Fuel OR (cm)	Half-pitch (cm)	UO_2 Mass with 5 wt% U-235 (kg)	U-235 Mass (g)	k_{eff}	σ	$k_{eff}+2\sigma$	ΔUSL ($k_{eff}+2\sigma$ – 0.9370)
5p-r350-p040	0.350	0.40	105.5	4649	0.76018	0.00014	0.76046	-0.17654
5p-r350-p050	0.350	0.50	67.5	2975	0.89004	0.00025	0.89054	-0.04646
5p-r350-p060	0.350	0.60	46.9	2066	0.94360	0.00025	0.94410	0.00710
5p-r350-p065	0.350	0.65	39.7	1750	0.95334	0.00027	0.95388	0.01688
5p-r350-p070	0.350	0.70	34.4	1518	0.94933	0.00026	0.94985	0.01285
5p-r350-p080	0.350	0.80	26.4	1162	0.93613	0.00027	0.93667	-0.00033
5p-r350-p090	0.350	0.90	20.8	918	0.90085	0.00014	0.90113	-0.03587
5p-r350-p100	0.350	1.00	16.9	744	0.87344	0.00023	0.87390	-0.06310
5p-r350-p110	0.350	1.10	13.9	615	0.83350	0.00021	0.83392	-0.10308
5p-r350-p120	0.350	1.20	11.7	517	0.79050	0.00021	0.79092	-0.14608
5p-r350-p140	0.350	1.40	8.6	380	0.72467	0.00021	0.72509	-0.21191
5p-r350-p160	0.350	1.60	6.6	291	0.64035	0.00018	0.64071	-0.29629
5p-r350-p200	0.350	2.00	4.2	186	0.55747	0.00016	0.55779	-0.37921
5p-r392-p040	0.392	0.40	105.5	4649	0.65809	0.00025	0.65859	-0.27841
5p-r392-p050	0.392	0.50	67.5	2975	0.80860	0.00015	0.80890	-0.12810
5p-r392-p060	0.392	0.60	46.9	2066	0.88961	0.00027	0.89015	-0.04685
5p-r392-p065	0.392	0.65	39.7	1750	0.90973	0.00030	0.91033	-0.02667
5p-r392-p070	0.392	0.70	34.4	1518	0.91594	0.00025	0.91644	-0.02056
5p-r392-p080	0.392	0.80	26.4	1162	0.91849	0.00024	0.91897	-0.01803
5p-r392-p090	0.392	0.90	20.8	918	0.89728	0.00025	0.89778	-0.03922
5p-r392-p100	0.392	1.00	16.9	744	0.87657	0.00026	0.87709	-0.05991
5p-r392-p110	0.392	1.10	13.9	615	0.84334	0.00014	0.84362	-0.09338
5p-r392-p120	0.392	1.20	11.7	517	0.80540	0.00023	0.80586	-0.13114
5p-r392-p140	0.392	1.40	8.6	380	0.74548	0.00022	0.74592	-0.19108
5p-r392-p160	0.392	1.60	6.6	291	0.66147	0.00019	0.66185	-0.27515
5p-r392-p200	0.392	2.00	4.2	186	0.57527	0.00018	0.57563	-0.36137
5p-r400-p040	0.400	0.40	105.5	4649	0.64212	0.00020	0.64252	-0.29448
5p-r400-p050	0.400	0.50	67.5	2975	0.79233	0.00025	0.79283	-0.14417
5p-r400-p060	0.400	0.60	46.9	2066	0.87641	0.00027	0.87695	-0.06005
5p-r400-p065	0.400	0.65	39.7	1750	0.90021	0.00018	0.90057	-0.03643
5p-r400-p070	0.400	0.70	34.4	1518	0.90857	0.00028	0.90913	-0.02787
5p-r400-p080	0.400	0.80	26.4	1162	0.91292	0.00016	0.91324	-0.02376
5p-r400-p090	0.400	0.90	20.8	918	0.89405	0.00014	0.89433	-0.04267
5p-r400-p100	0.400	1.00	16.9	744	0.87591	0.00015	0.87621	-0.06079
5p-r400-p110	0.400	1.10	13.9	615	0.84349	0.00016	0.84381	-0.09319
5p-r400-p120	0.400	1.20	11.7	517	0.80666	0.00021	0.80708	-0.12992

Input file	Fuel OR (cm)	Half-pitch (cm)	UO_2 Mass with 5 wt% U-235 (kg)	U-235 Mass (g)	k_{eff}	σ	k_{eff}+2σ	ΔUSL (k_{eff}+2σ – 0.9370)
5p-r400-p140	0.400	1.40	8.6	380	0.74811	0.00022	0.74855	-0.18845
5p-r400-p160	0.400	1.60	6.6	291	0.66462	0.00021	0.66504	-0.27196
5p-r400-p200	0.400	2.00	4.2	186	0.57807	0.00019	0.57845	-0.35855
5p-r450-p050	0.450	0.50	67.5	2975	0.68794	0.00010	0.68814	-0.24886
5p-r450-p060	0.450	0.60	46.9	2066	0.79519	0.00011	0.79541	-0.14159
5p-r450-p065	0.450	0.65	39.7	1750	0.83069	0.00010	0.83089	-0.10611
5p-r450-p070	0.450	0.70	34.4	1518	0.85074	0.00011	0.85096	-0.08604
5p-r450-p080	0.450	0.80	26.4	1162	0.87207	0.00011	0.87229	-0.06471
5p-r450-p090	0.450	0.90	20.8	918	0.86905	0.00011	0.86927	-0.06773
5p-r450-p100	0.450	0.00	16.9	744	0.85940	0.00011	0.85962	-0.07738
5p-r450-p110	0.450	1.10	13.9	615	0.83648	0.00010	0.83668	-0.10032
5p-r450-p120	0.450	1.20	11.7	517	0.80687	0.00011	0.80709	-0.12991
5p-r450-p140	0.450	1.40	8.6	380	0.75590	0.00010	0.75610	-0.18090
5p-r450-p160	0.450	1.60	6.6	291	0.67707	0.00009	0.67725	-0.25975
5p-r450-p200	0.450	2.00	4.2	186	0.59144	0.00008	0.59160	-0.34540
5p-r550-p060	0.550	0.60	46.9	2066	0.62707	0.00022	0.62751	-0.30949
5p-r550-p065	0.550	0.65	39.7	1750	0.67324	0.00022	0.67368	-0.26332
5p-r550-p070	0.550	0.70	34.4	1518	0.70866	0.00025	0.70916	-0.22784
5p-r550-p080	0.550	0.80	26.4	1162	0.75918	0.00025	0.75968	-0.17732
5p-r550-p090	0.550	0.90	20.8	918	0.78359	0.00026	0.78411	-0.15289
5p-r550-p100	0.550	1.00	16.9	744	0.79202	0.00015	0.79232	-0.14468
5p-r550-p110	0.550	1.10	13.9	615	0.78706	0.00026	0.78758	-0.14942
5p-r550-p120	0.550	1.20	11.7	517	0.77313	0.00014	0.77341	-0.16359
5p-r550-p140	0.550	1.40	8.6	380	0.73852	0.00024	0.73900	-0.19800
5p-r550-p160	0.550	1.60	6.6	291	0.67533	0.00013	0.67559	-0.26141
5p-r550-p200	0.550	2.00	4.2	186	0.59986	0.00021	0.60028	-0.33672
5p-r650-p070	0.650	0.70	34.4	1518	0.57870	0.00021	0.57912	-0.35788
5p-r650-p080	0.650	0.80	26.4	1162	0.63785	0.00021	0.63827	-0.29873
5p-r650-p090	0.650	0.90	20.8	918	0.67916	0.00026	0.67968	-0.25732
5p-r650-p100	0.650	1.00	16.9	744	0.70393	0.00024	0.70441	-0.23259
5p-r650-p110	0.650	1.10	13.9	615	0.71191	0.00024	0.71239	-0.22461
5p-r650-p120	0.650	1.20	11.7	517	0.71243	0.00023	0.71289	-0.22411
5p-r650-p140	0.650	1.40	8.6	380	0.69509	0.00024	0.69557	-0.24143
5p-r650-p160	0.650	1.60	6.6	291	0.65053	0.00023	0.65099	-0.28601
5p-r650-p200	0.650	2.00	4.2	186	0.58790	0.00022	0.58834	-0.34866

6.9.4. Bounding Fuel Rod Parameters for Approved Content

This section documents the most limiting (bounding) fuel rod configurations for justification of the allowed content (GE BWR 10x10 fuel rods with a maximum U-235 enrichment of 5 wt% with a U-235 mass limit of 1750 g).

6.9.4.1. Method Overview

Starting with the most limiting fuel rod configuration from Sections 6.4 through 6.6 (FRLSmh_1_22), the fuel pellet height and diameter were varied while holding the system H/U-235 ratio constant (same U-235 mass). The U-235 enrichment was then reduced to 5 wt% and a new optimal fuel rod-to-rod pitch was determined. The evaluations in the subsections below were assessed using MCNP6 Version 2.0 (Reference 6-6).

The fissile mass modeled in the fuel rod array used a range of UO_2 masses (105 kg to 4 kg) with a U-235 enrichment of 5 wt% at various rod diameters (heights) and rod-to-rod pitches. This is approximately 4,600 grams to 180 grams of U-235.

6.9.4.2. Limiting Fuel Pellet Radius

Starting with the most limiting fuel rod configuration from Sections 6.4 through 6.6 (FRLSmh_1_22), the fuel pellet height and diameter were varied while holding the system H/U-235 ratio constant (same U-235 mass). Both the fuel rod-to-rod pitch (half-pitch of 0.7 cm), total rods number of rods, and U-235 enrichment (6 wt%) were the same from the limiting HAC single package case.

As shown in Figure 6.9.4-1, certain fuel column heights/radii combinations will exceed the 0.9370 USL with U-235 enrichments at 6 wt%. Therefore, a similar study was repeated using a U-235 enrichment of 5 wt% to determine potential pellet heights/radii that would be below the USL. Figure 6.9.4-2 shows the results of the fuel column heights/radii study with a U-235 enrichment of 5 wt%.

The results of these studies are in Section 6.9.3, Table 6.9.3-6.

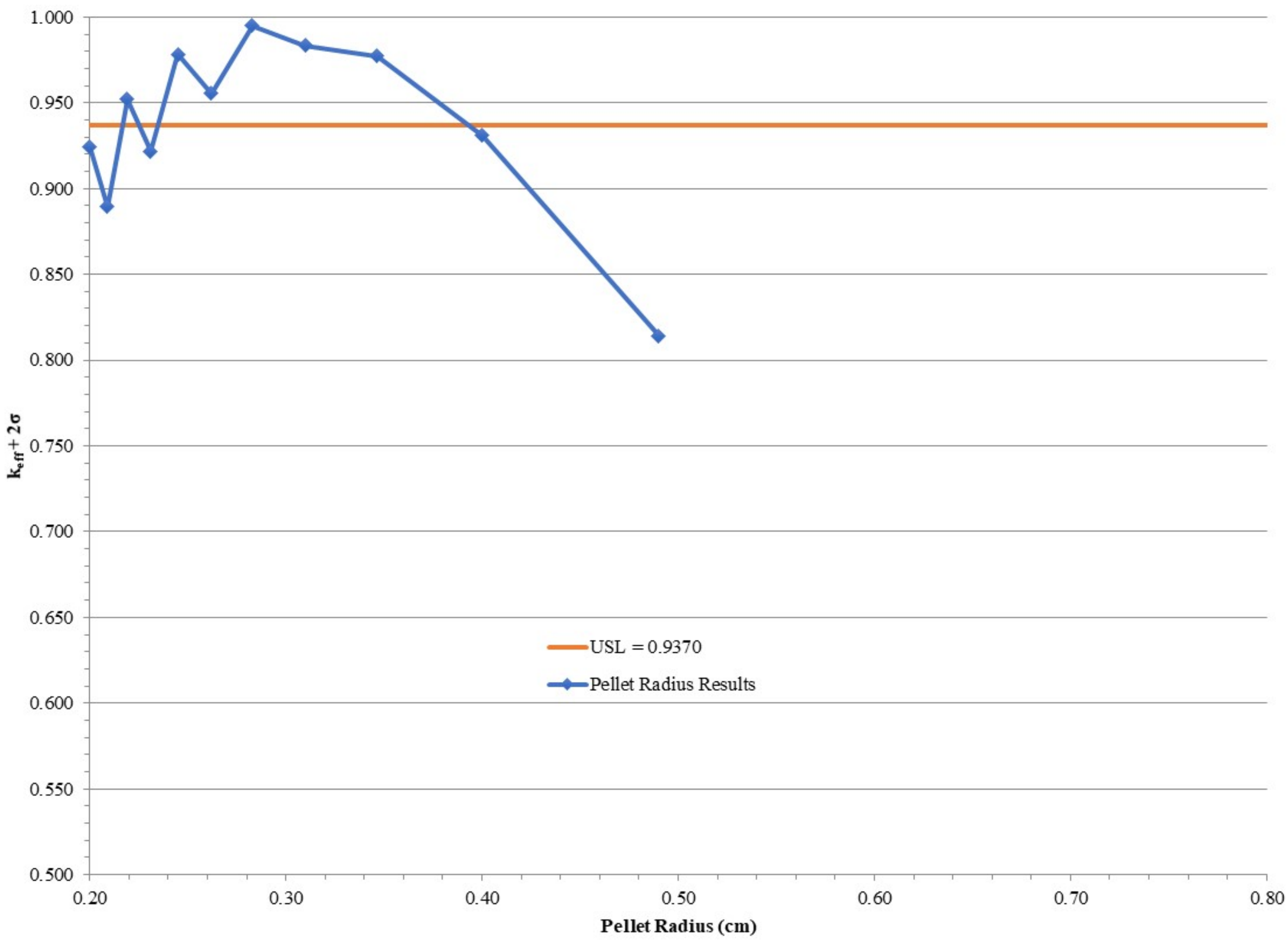

Figure 6.9.4-1. Fuel Column Evaluation at same H/U-235 Ratio (6 wt% U-235)

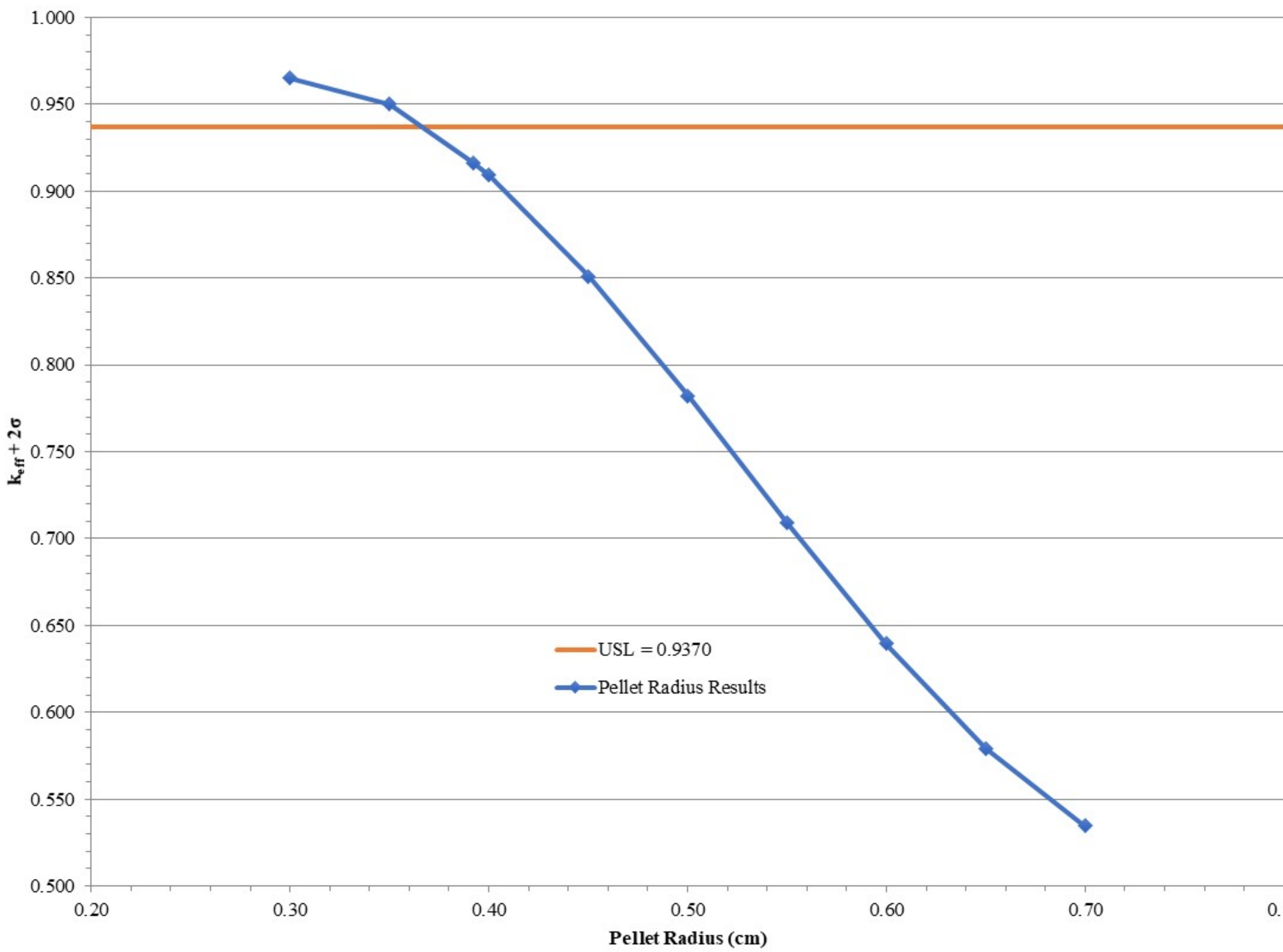

Figure 6.9.4-2. Fuel Column Height/Radius Evaluation at 5 wt% U-235

6.9.4.3. Optimal Fuel Rod Pitch

To confirm that the fuel column heights/radii would remain below the 0.9370 USL, a rod-to-rod pitch study was conducted (i.e. determine optimal system H/U-235 ratio for a given fuel column). This evaluation covered a large range of UO_2 masses (105 kg – 4 kg) with a U-235 enrichment of 5 wt%; see Table 6.9.3-7. Note that the 1750 U-235 mass limit correlates to a rod-to-rod half-pitch of approximately 0.65 cm and remains below the USL for the allowed fuel rod content.

Figure 6.9.4-3 provides a summary of this study. The results of these studies are in Section 6.9.3, Table 6.9.3-7.

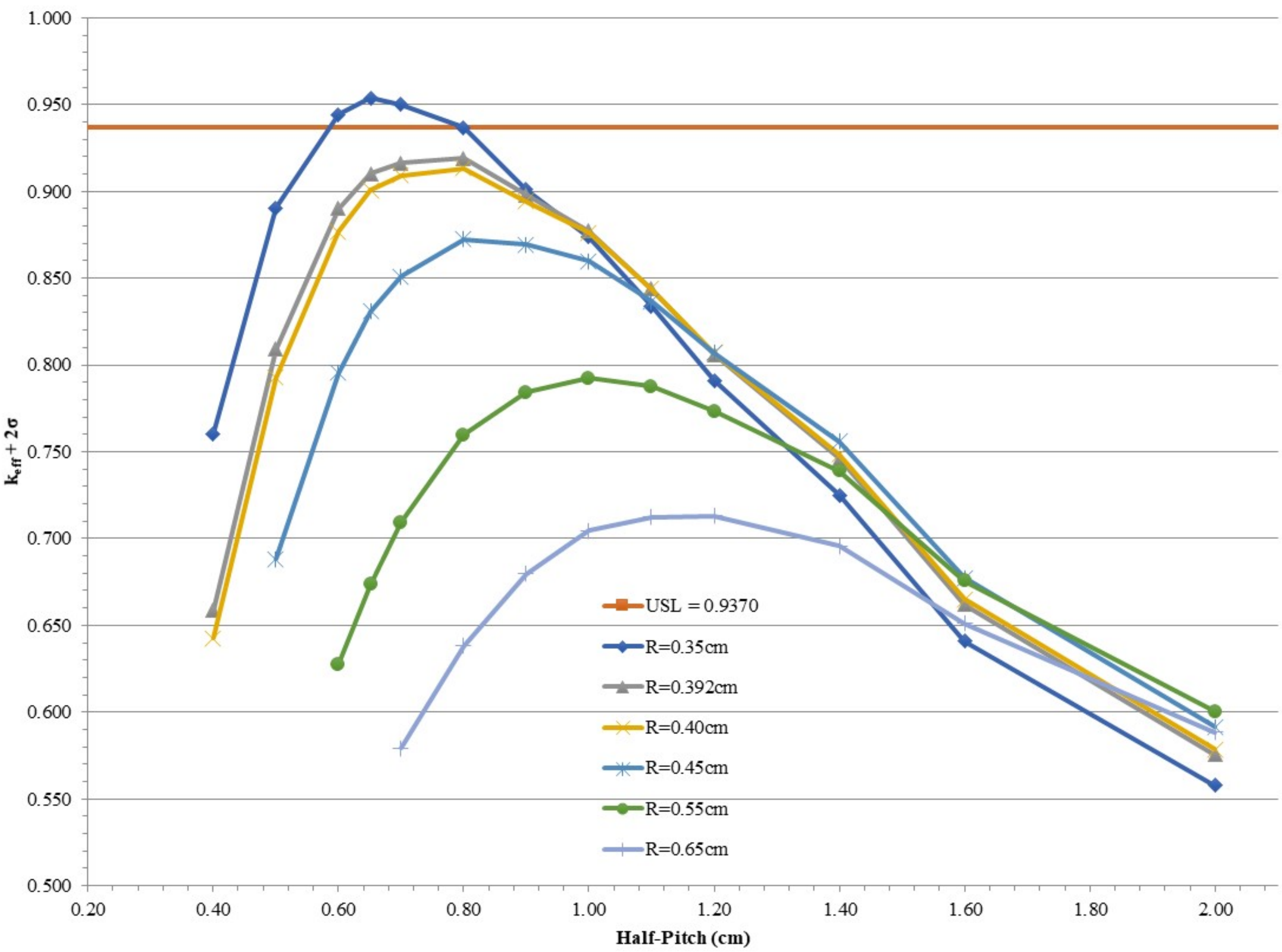

Figure 6.9.4-3. Optimal Fuel Rod Pitch Evaluation at 5 wt% U-235

6.9.4.4. Conclusions

Based on the studies presented in Section 6.9.4, the following limitations apply to irradiated fuel rods to maintain criticality safety:

a) Fuel rod pellet OD: ≥ 0.784 cm (0.392 cm radius).
b) U-235 enrichment: ≤ 5 wt%
c) U-235 mass: ≤ 1750 g

The Section 6.9.4 analysis used the single HAC model to determine the most limiting fuel rod configuration with 5 wt% U-235. Therefore, both the 2N HAC and 5N NCT arrays were assessed to confirm that the results would met the USL. Table 6.1.2-1 provides the bounding fuel rod content results for the single HAC, 5N NCT array, and 2N HAC array. Table 6.1.2-2 provides the results of the most limiting fuel rod content when applying the U-235 administrative mass limit of 1750 grams. Note that both Tables 6.1.2-1 and 6.1.2-2 apply the 1% k_{eff} uncertainty for pitch geometric modeling as described in Section 6.9.1.

6.10 References

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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 "MCNP Version 6.2 Release Notes," Los Alamos National Laboratory, LA-UR-18-20808.

6-7 J. Conlin et al., "Listing of Available ACE Data Tables," Los Alamos National Laboratory, LA-UR-13-21822, Revision 4, June 2014.

6-8 H. R., Parks, C. V. Dyer, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," Oak Ridge National Laboratory, NUREG/CR-5661, 1997.

6-9 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.

6-10 "Verification of MCNP6.2 for Nuclear Criticality Safety Applications," Los Alamos National Laboratory, LA-UR-17-24406, June 2017.

6-11 J.R. Taylor, "An Introduction to Error Analysis," page 268-271, 2nd Edition, University Science Book, 1997.

6-12 S.S. Shapiro and M.B. Wilk, "An Analysis of Variance Test of Normality (Complete Samples)," General Electric Co. and Bell Telephone Laboratories, Biometrika, Vol. 52, No. 3/4, 1965.

6-13 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.

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

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