4. CONTAINMENT

Design analysis, full-scale testing, and similarity of the ES-3100 prototypes have been used to demonstrate that the ES-3100 package with highly enriched uranium (HEU) is in compliance with the applicable containment requirements of Title 10 Code of Federal Regulations, Part 71 (10 CFR 71). The containment requirements of 10 CFR 71.51 are shown in Table 4.1. A bounding load case has been established for the ES-3100 package, and it assumes that the maximum HEU content is 35.2 kg (Sect. 1.2.3.6) with a decay heat load of 0.4 W (Sect. 1.2.3.7). This decay heat load and the volumes established for the convenience cans, spacers, silicone rubber pads, and the containment vessel void volume are discussed in Sect. 3.1.2 and Appendix 3.6.4. Sections 2 and 3 of this safety analysis report (SAR) also examine the effects of the lightest weight HEU content [2.77 kg (6.11 lb)]. The evaluations in Sects. 2, 3, and 4 have demonstrated that the ES-3100 shipping package with HEU content weight ranging from 2.77 kg (6.11 lb) to 35.2 kg (77.60 lb) meets the containment requirements specified in 10 CFR 71 for all conditions of transport. A summary of the containment boundary design and fabrication acceptance basis is given in Table 4.2. No credit is taken for the various convenience cans' ability to protect the HEU contents from being released.

Table 4.1. Containment requirements of transport for Type B packages [a]

Condition	Allowable release rate
Normal Conditions of Transport (NCT)	$R_N = 10^{-6}$ A_2 per hour = 2.78×10^{-10} A_2 per second
Hypothetical Accident Conditions (HAC)	$R_A = A_2$ in 1 week = 1.65×10^{-6} A_2 per second
	For ^{85}Kr, a value of 10 A_2 in 1 week is used

[a] From ANSI N14.5-1997, Sects. 5.4.1 and 5.4.2, and 10 CFR 71.51(a)(1) and (a)(2)

Table 4.2. Summary of the containment vessel design and fabrication acceptance basis

Nominal empty weight	15.10 kg (33.29 lb)
Air fill medium temperature at loading	25°C (77°F)
Air fill medium pressure at loading	101.35 kPa (14.70 psia)
Hydrostatic pressure test	1034 ± 34 kPa (150 ± 5) gauge
Helium acceptance leakage rate [a]	$L_T \leq 2.0 \times 10^{-7}$ cm^3/s
Air acceptance leakage rate [a]	$L_T \leq 1 \times 10^{-7}$ ref-cm^3/s
Air preshipment leakage rate	$L_T \leq 1 \times 10^{-4}$ atm-cm^3/s

[a] Acceptance leakage testing includes fabrication, periodic (within 12 months of use), and maintenance testing. According to Sect. 2.1 of ANSI N14.5-1997, *leaktight* is defined as an air leakage rate of 1×10^{-7} ref-cm^3/s; under the same conditions, this air leakage rate is ~ equal to a helium leakage rate of 2×10^{-7} cm^3/s.

The analysis documented in Appendix 4.6.1 was conducted to establish the upper limit for the total activity and the maximum number of A_2s proposed for transport in the ES-3100 package. The maximum activity [3.2427×10^{-1} TBq (8.764 Ci)] of the contents occurs 10 years after initial fabrication. When the maximum activity-to-A_2 value (293.99) is reached at ~70 years from material fabrication, the corresponding activity is 3.2328×10^{-1} TBq (8.737 Ci). These values have been determined using a maximum of 35.2 kg of HEU with isotopic weight percents as shown in Table 4.3. By applying the maximum weight percents of isotopes ^{233}U, ^{234}U, ^{236}U and by incorporating the traces of ^{232}U and the transuranic isotopes, the maximum activity, minimum A_2 value, and the minimum leakage requirements were determined for the proposed contents and are summarized in Tables 4.4, 4.5, and 4.6. The mass and isotopic concentrations used for the proposed content do not take into consideration limits based on shielding and subcriticality.

The initial composition of the content contains several isotopes of uranium (Sect. 1.2.3). As a result of radioactive decay, the ingrowth of uranium daughter products occurs, and these concentrations of daughter products will vary with time. The uranium isotopes and daughter products are considered a mixture of radionuclides, and the method for determining the mixture's A_2 value in Section IV, Appendix A, 10 CFR 71 is applied. The A_2 value for the most conservative set of contents defined in Sect. 1.2.3 has been calculated in Appendix 4.6.1. Since the HEU can be in the form of oxides (U_3O_8-Al, UO_2-Mg, UO_2, UO_3, and U_3O_8), uranyl nitrate crystals (UNX), or metal and alloy, the calculation of the mixture's A_2 used the various uranium isotopic A_2 values for fast, medium, and slow lung absorption criteria shown in Table A-1 of Appendix A of 10 CFR 71.

4.1 DESCRIPTION OF THE CONTAINMENT BOUNDARY

As shown in Table 4.4, the number of A_2s proposed for shipping exceeds 30 but is less than 3000. In accordance with NUREG 1609, the containment vessel is a Category II vessel. Since this vessel may be used for future contents that exceed 3000 A_2, the containment vessel category has been elevated to a Category I vessel. Therefore, the containment vessel is designed (using nominal dimensions for each component), fabricated, and inspected in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*, Sect. III, Division I, Subsection NB and Section IX.

4.1.1 Containment Boundary

The containment boundary consists of the vessel's body, lid assembly, and inner O-ring (Sect. 1, Fig. 1.3). Only the inner O-ring is considered part of the boundary. The outer O-ring is provided to allow a post-assembly verification leak check. Two methods of fabrication may be used to fabricate the containment vessel body as shown on Drawing M2E801580A012 (Appendix 1.4.8). The first method uses a standard 5-in., schedule 40 stainless-steel pipe per ASME SA-312 Type TP304L, a machined flat-head bottom forging per ASME SA-182 Type F304L, and a machined top flange forging per ASME SA-182 Type F304L. The nominal outside diameter of the 5-in. schedule 40 pipe is machined to match the nominal wall thickness of 0.254 cm (0.100 in.). Each of these pieces is joined with full-penetration circumferential weld as shown on sheet 2 of Drawing M2E801580A012 (Appendix 1.4.8). The weld filler material conforms to Sect. II, Part C, of the *ASME Boiler and Pressure Vessel Code*. All full-penetration welds are dye penetrant and radiographically inspected in accordance with Sect. III, Div. I, Sect. NB-5000, of the *ASME Boiler and Pressure Vessel Code*. The top flange is machined to match the schedule 40 stainless-steel 5-in. pipe, to provide two concentric half-dove-tailed O-ring grooves in the flat face, to provide locations for two 18-8 stainless-steel dowel pins, and to provide the threaded portion for closure using the lid assembly. The second method of fabrication uses forging, flow forming, or metal spinning to create the complete body (flat bottom, cylindrical body, and flange) from a single forged billet or bar with final material properties in accordance with ASME SA-182 Type F304L. Final machining of the top flange area is identical to that of the welded forging method. The lid assembly, which completes the containment boundary structure, consists of a sealing lid, closure nut, and external retaining ring (Drawing M2E801580A014, Appendix 1.4.8). The containment vessel sealing lid (Drawing M2E801580A015, Appendix 1.4.8) is machined from Type 304 stainless-steel bar with final material properties in accordance with ASME SA-479. The containment vessel closure nut is machined from a Nitronic 60 stainless-steel bar with material properties in accordance with ASME SA-479. These two components are held together using a WSM-400-S02 external retaining ring made from Type 302 stainless steel. The sealing lid is further machined to accept a ⅜-in.-16 swivel hoist ring bolt, to provide a leak-check port between the elastomeric O-rings, and notched along the perimeter to engage two dowel pins. The lid assembly, with the O-rings in place on the body, are joined together by torquing the closure nut and sealing lid assembly to 162.7 ± 6.78 N·m (120 ± 5 ft-lb). The sealing lid portion of the assembly is restrained from rotating during this torquing operation by the two dowel pins installed in the body flange. This torquing of the closure nut represents the positive fastening device used to satisfy the requirements of 10 CFR 71.43(c). The effectiveness of this closure system has been demonstrated by the NCT and HAC tests, which show that the complete containment system, including welds and O-ring seals, meet the leaktight criterion as defined in ANSI N14.5-1997 after the conclusion of the test series documented in *Test Report of the ES-3100 Package* (Appendix 2.10.7).

10 CFR 71.73(c) requires that the package containment vessel be immersed in 15 m (50 ft) of water, which is equivalent to an external pressure differential of 150 kPa (21.7 psi). The design analyses (Appendix 2.10.1) show that this vessel is conservatively rated for the 150-kPa (21.7-psi) external pressure differential requirement, as well as for an internal pressure differential of 699.82 kPa (101.5 psig). A summary of the containment boundary design and acceptance basis is given in Table 4.2.

The ES-3100 package has no connections, fittings, or tapped holes that penetrate the containment boundary; therefore, the package does not allow continuous venting during transport.

The containment vessel O-rings (Drawing M2E801580A013, Appendix 1.4.8) are manufactured from an ethylene-propylene elastomer in accordance with specifications for 70A Durometer preformed packing developed at Y-12. These O-rings are rated for continuous service as a static face seal in the temperature range of −40 to 150°C (−40 to 302°F) [*Parker O-ring Handbook*, Fig. 2-24]. Tests conducted by Los Alamos National Laboratory (LANL) on a similar compound, ethylene-propylene rubber (EPDM), used as a static face seal, are documented in *SAFKEG 2863B Tests for Verification of O-ring Performance* (TR 96/12/20). The material compound for the LANL tests was certified to ASTM D-2000 as M3BA610A14B13F17. The ES-3100 package O-rings are also certified to ASTM D-2000 as M3BA712A14B13F17. The class of material in the LANL test is identical except that the durometer and tensile strengths are somewhat less than those of the ES-3100 package. Each material was tested in accordance with ASTM D-2137 for brittleness at −40°C (−40°F) without failure. The leak test fixture, as reported in TR 96/12/20, provided a maximum compression of 25.7% in a static face seal configuration. The compression range provided by the flange and lid design of the ES-3100 package is 14.8 to 20.8% or 0.051 to 0.076 cm (0.020 to 0.030 in.) compression due to the half-dovetail design. Furthermore, *Parker O-ring Handbook* states that the minimum squeeze for all seals, regardless of cross-section, should be about 0.018 cm (0.007 in.). Since the minimum compression is 0.051 cm (0.020 in.) and the flange and lid with the closure nut have nearly identical coefficient of thermal expansion, the sealing performance at −40°C (−40°F) should not be degraded. Therefore, the performance of the ES-3100 O-rings should be representative of those documented in TR 96/12/20. These tests demonstrated that the O-rings were leaktight over the temperature range of −40 to 205°C

(−40 to 401 °F), which is greater than the operating temperature range of −40 to 141.22 °C (−40 to 286.2 °F) of the ES-3100 containment vessel (Table 3.17). In addition to component testing, an ES-3100 full-scale test unit (Test Unit-2) was chilled to ≤ −40 °C and later subjected to an NCT drop test and the entire HAC test battery. The containment vessel was leak tested and achieved "leaktight" status. Therefore, the continuous service temperature rating of the ethylene-propylene elastomer has been verified by testing.

4.1.2 Special Requirements for Plutonium

The highly enriched uranium contents have only trace amount of the transuranic isotopes. Therefore, this section is not applicable to the ES-3100 shipping container.

4.2 GENERAL CONSIDERATIONS

4.2.1 Type A Fissile Packages

The A_2 value of the proposed contents exceed the limits established for Type A packages.

4.2.2 Type B Packages

Requirements

(1) A Type B package, in addition to satisfying the requirements of 10 CFR 71.41–71.47, must be designed, constructed, and prepared for shipment so that under the tests specified in:

(a) Section 71.71 ("Normal conditions of transport"): There would be no loss or dispersal of radioactive contents as demonstrated to a sensitivity of 10^{-6} A_2/h, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging; and

(b) Section 71.73 ("Hypothetical accident conditions"): There would be no escape of ^{85}Kr exceeding 10 A_2 in one week, no escape of other radioactive material exceeding a total amount A_2 in one week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in.) from the external surface of the package.

(2) Where mixtures of different radionuclides are present, the provisions of Appendix A, paragraph IV of this part shall apply, except that for ^{85}Kr, an effective A_2 value equal to 10 A_2 may be used.

(3) Compliance with the permitted activity release limits of paragraph (a) of this section may not depend on filters or on a mechanical cooling system.

Analysis. The A_2 value calculated for the ES-3100 shipping package has been determined in accordance with Appendix A of 10 CFR 71, documented in Appendix 4.6.1, and uses the proposed isotopic distribution shown in Table 4.3. Table 4.4 summarizes the results from Appendix 4.6.1 for the proposed contents from 0 to 70 years after original production.

Table 4.3. Isotopic mass and weight percent for the HEU contents [a]

Nuclide	Weight percent	Mass (g)
U-232	0.000004	0.001408
U-233	0.600000	211.200000
U-234	2.000000	704.000000
U-235	54.895996	19323.390592
U-236	40.000000	14080.000000
U-238	0.000000	0.000000
Transuranic	0.004000	1.408000
Np-237	2.500000	880.000000
Total	100.000000	35200.000000

[a]. Weight percent values of individual isotopes are those that generate the largest activity within the allowable ranges presented in Sect. 1.2.3.

Table 4.4. Activity, A_2 value, and number of A_2 proposed for transport

Year	Fast absorption			Medium absorption			Slow absorption		
	Activity (TBq)	A_2 (TBq)	Act / A_2	Activity (TBq)	A_2 (TBq)	Act / A_2	Activity (TBq)	A_2 (TBq)	Act / A_2
0	3.185E-01	1.280E-03	2.489E+02	3.185E-01	1.226E-03	2.599E+02	3.185E-01	1.089E-03	2.926E+02
5	3.238E-01	1.295E-03	2.500E+02	3.238E-01	1.240E-03	2.610E+02	3.238E-01	1.103E-03	2.937E+02
10	3.243E-01	1.296E-03	2.502E+02	3.243E-01	1.241E-03	2.612E+02	3.243E-01	1.104E-03	2.938E+02
20	3.241E-01	1.295E-03	2.503E+02	3.241E-01	1.240E-03	2.613E+02	3.241E-01	1.103E-03	2.938E+02
30	3.239E-01	1.293E-03	2.504E+02	3.239E-01	1.239E-03	2.614E+02	3.239E-01	1.102E-03	2.938E+02
40	3.237E-01	1.292E-03	2.505E+02	3.237E-01	1.238E-03	2.615E+02	3.237E-01	1.101E-03	2.938E+02
50	3.235E-01	1.291E-03	2.506E+02	3.235E-01	1.237E-03	2.616E+02	3.235E-01	1.101E-03	2.939E+02
60	3.234E-01	1.290E-03	2.507E+02	3.234E-01	1.236E-03	2.617E+02	3.234E-01	1.100E-03	2.939E+02
70	3.233E-01	1.289E-03	2.508E+02	3.233E-01	1.235E-03	2.618E+02	3.233E-01	1.100E-03	2.940E+02

4.3 CONTAINMENT UNDER NORMAL CONDITIONS OF TRANSPORT (TYPE B PACKAGES)

Title 10 CFR 71.51(a)(1) specifies that there shall be no loss or dispersal of radioactive contents as demonstrated to a sensitivity of 10^{-6} A_2 per hour, no significant increase in external radiation levels, and no substantial reduction in the effectiveness of the packaging. The initial composition of the HEU contains several isotopes of uranium and transuranic contributors (Table 4.3). As a result of radioactive decay, uranium, transuranics, and daughter product isotopes are present in the HEU contents at varying concentrations depending on the length of decay time. The HEU, with its isotopes and daughter isotopes and contributions from unknown transuranic isotopes, qualifies as a mixture for A_2 determination (Appendix A of 10 CFR 71). The A_2 value and the maximum content activity-to-A_2 value ratio for this mixture have been calculated for several different decay times (Table 4.4). As calculated in Appendix 4.6.1, the A_2 value [1.0997×10^{-3} TBq (2.9720×10^{-2} Ci)] and the maximum content activity-to-A_2 ratio (293.99) used to qualify this package occurs at about 70 years of decay. As previously stated, these values have been determined using a bounding case maximum of 35.2 kg of HEU with isotopic weight percent values as shown in Table 4.3. The specified composition is a very conservative upper bound achieved by using the maximum weight percent values for the higher specific activity isotopes (^{232}U, ^{233}U, ^{234}U, and ^{236}U), including contributions from other transuranics and ^{237}Np.

The maximum activity, minimum A_2, and minimum leakage requirements were determined for this worst case scenario and are presented in Tables 4.4 and 4.5. These masses and isotopic concentrations were used for the proposed contents without regard to limits established based on shielding and subcriticality considerations. The actual mass limits affirmed for this shipping package are established in Sect. 1 (Table1.3). The analyses conducted in Appendix 4.6.2 assumes that the total mass of uranium for each component is available for release as an aerosol (worst case). From experimental tests, the maximum aerosol density containing uranium particulate was reported in *Leakage of Radioactive Powders from Containers* (Curren and Bond 1980) to be 9.0×10^{-6} g/cm^3. This aerosol density is used to calculate the total activity concentration in ANSI N14.5-1997, Section B.15, examples 13, 27, and 29.

The containment criteria for the ES-3100 package will be leaktight (defined in paragraph 2.1 of ANSI N14.5-1997 as having a leakage rate $\leq 1 \times 10^{-7}$ ref-cm^3/s) during the prototype tests. This leaktight criterion satisfies the design verification requirement stipulated in paragraph 7.2.4 of ANSI N14.5-1997. The requirements of ANSI N14.5-1997 are used for all stages of containment verification for the ES-3100 (i.e., design, fabrication, maintenance, periodic and preshipment).

The design, fabrication, maintenance and periodic leakage rate limit is 1×10^{-7} ref-cm^3/s air. The pass criterion for the preshipment leakage rate test, which demonstrates correct assembly of the containment vessels, is 1×10^{-4} ref-cm^3/s, which exceeds the requirements given in ANSI N14.5, paragraph 7.6.4. In accordance with the definition of sensitivity of a leakage test procedure provided in Sections 2 and 7.6.4 of ANSI N14.5-1997, the minimum acceptable leakage rate that the procedure needs to be capable of detecting is 1×10^{-3} ref-cm^3/s. The requirements for the ES-3100 exceed the regulatory criterion by specifying a leakage rate of $\leq 1 \times 10^{-4}$ ref-cm^3/s, and equipment used in accordance with Section 7.6.4 of ANSI N14.5-1997 would not detect this leakage. The preshipment, fabrication, maintenance and periodic leakage rate tests are required to be conducted on each containment vessel in accordance with ANSI N14.5 and are specified in Chapters 7 and 8. These leakage rates are not dependent on filters or mechanical cooling.

Table 4.5. Regulatory leakage criteria for NCT [a]

Verification Activity	Fast Absorption		Medium Absorption		Slow Absorption	
	L_{RN-air} (ref-cm^3/s)	L_{RN-He} (cm^3/s)	L_{RN-air} (ref-cm^3/s)	L_{RN-He} (cm^3/s)	L_{RN-air} (ref-cm^3/s)	L_{RN-He} (cm^3/s)
Design	3.8614E-03	4.1482E-03	3.7003E-03	3.9803E-03	3.2965E-03	3.5587E-03

[a] The procedure used to calculate the above criteria is shown in Appendix 4.6.2. This data has been extracted from Table 1 in Appendix 4.6.2.

Table 4.6. Containment vessel verification tests criteria for NCT

Test Type	Test Values	Leakage test procedure
	Design and compliance leakage testing	
Design verification of O-ring seal (air)	$L_T \leq 1.0 \times 10^{-4}$ ref-cm^3/s	See Appendix 2.10.7
Design verification of containment vessel boundary (helium)	$L_T \leq 2.0 \times 10^{-7}$ cm^3/s	See Appendix 2.10.7
	Verification leakage testing	
Fabrication, periodic, and maintenance (helium)	$L_T \leq 2.0 \times 10^{-7}$ cm^3/s	Y51-01-B2-R-140, Rev. A.1 (Appendix 8.3.1)
	$L_T \leq 1.0 \times 10^{-4}$ ref-cm^3/s	Y51-01-B2-R-074, Rev. A.1 (Appendix 7.5.1)

The complete design verification testing of the ES-3100 package for NCT was conducted on test unit TU-4. Since the containment vessel was assembled at ambient conditions, the pressure was nominally 101.35 kPa (14.70 psia) at 25°C (77°F). In accordance with 10 CFR 71.71(b), the initial pressure inside each containment vessel should be the maximum normal operating pressure (MNOP). As calculated in Appendix 3.6.4, the bounding case MNOP is 137.92 kPa (20.004 psia). The stresses at the maximum normal operating pressure [36.57 kPa (5.304 psig)] are insignificant compared to the allowable stresses (Table 2.21). O-ring grooves are designed and fabricated in accordance with guidance from the *Parker O-ring Handbook*. In accordance with Fig. 3-2 of the *Parker O-ring Handbook*, the durometer of the O-ring, and the tolerance gap from the production drawings, the O-ring should be able to withstand ~800 psig before anti-extrusion devices are required. Therefore, conducting a compliance test with the MNOP in the containment vessel will have little, if any, effect on the results.

Following the design verification testing of paragraphs 10 CFR 71.71(c)(5) through 71.71(c)(10) excluding 71.71(c)(8), Test Unit-4 was subjected to the sequential testing of paragraphs 10 CFR 71.73(c)(1) through (c)(4). Upon removal of the containment vessel from the drum assembly, the cavity between the O-rings was leak checked. This unit recorded a leak rate between the O-rings of 2.4773×10^{-5} ref-cm^3/s.

Following the O-ring leak test, the entire containment boundary of TU-4 was helium leak tested to a value $\leq 2 \times 10^{-7}$ cm^3/s, thereby verifying a leak-tight boundary. The leak-test procedure followed to verify this criteria is documented in the ES-3100 test plan (Appendix 2.10.7). The maximum recorded helium leakage rate for this containment vessel was 2.0×10^{-7} cm^3/s after 20 min of testing. Visual inspection following the testing indicated that neither the vessel body, the O-rings, the seal areas, nor the vessel lid assembly were damaged during the tests. Pictures taken of the containment vessel top following testing showed that the closure nut had rotated a maximum of 0.15 cm (0.060 in.) from its original radial position obtained during assembly. Based on the pitch of the closure nut, this rotation translates into only 0.0013 cm (0.0005 in.) decompression of the O-rings. This compares to the original nominal compression of 0.064 cm (0.025 in.). Therefore, O-ring compression was maintained during compliance testing. Based on these results, the ES-3100 package meets and exceeds the containment criteria specified in 10 CFR 71.51 for NCT when used to ship the contents described in the introductory section of this chapter.

Following fabrication, the containment vessel undergoes hydrostatic pressure testing to 1034 kPa (150 psi) gauge. The hydrostatic test is conducted before the final leakage test. Following the hydrostatic pressure test, and prior to conducting the leakage test, the containment vessel and O-ring cavity must be thoroughly dried. Each vessel is then leak tested with either air or helium to $\leq 1 \times 10^{-7}$ ref-cm^3/s or 2×10^{-7} cm^3/s, respectively. This test ensures the containment vessel's integrity (walls, welds, inner O-ring seal) as delivered for use in accordance with paragraph 6.3.2 of ANSI N14.5-1997.

Following placement of the HEU content inside the containment vessel and joining the body and lid assembly, the volume between the containment vessel's O-ring seals is evacuated and checked to leak $\leq 1 \times 10^{-4}$ ref-cm^3/s. This leak-test procedure is a pressure rise air leak test prescribed in Section 7.1.2. This ensures that each containment vessel has been properly assembled in accordance with paragraph 7.6.4 of ANSI N14.5-1997.

The design verification tests were conducted following compliance tests in accordance with 10 CFR 71.71 and 71.73. The effectiveness of this closure system has been demonstrated by the NCT and HAC tests, which show that the complete containment system, including welds and O-ring seals, meet the leaktight criterion as defined in ANSI N14.5-1997 after the conclusion of the test series documented in *Test Report of the ES-3100 Package* (Appendix 2.10.7).

4.4 CONTAINMENT UNDER HYPOTHETICAL ACCIDENT CONDITIONS (TYPE B PACKAGES)

Requirements. A Type B package, in addition to satisfying the requirements of paragraphs 10 CFR 71.41 through 71.47, must be designed, constructed, and prepared for shipment so that under the tests specified in Sect. 71.73 ("Hypothetical Accident Conditions"), there would be no escape of ^{85}Kr exceeding 10 A_2 in one week, no escape of other radioactive material exceeding a total amount A_2 in one week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in.) from the external surface of the package.

Analysis. Calculations have been conducted in Appendix 4.6.2 to determine the regulatory leakage criteria to satisfy the above requirements. The results are shown in Table 4.7. These analyses assume that the total mass of uranium for each component is available for release as an aerosol (worst case). From experimental tests, the maximum aerosol density containing uranium particulate was reported by Curren and Bond to be 9.0×10^{-6} g/cm^3. This aerosol density is used to calculate the total activity concentration in ANSI N14.5-1997, Section B.15 examples 13, 27, and 29. Design leakage rate verification testing of the containment boundary (Table 4.8) was conducted on Test Units-1 through -6 and documented in test report (Appendix 2.10.7). Since each containment vessel was assembled at ambient conditions, the pressure was nominally 101.35 kPa (14.70 psi) at 25°C (77°F). In accordance with 10 CFR 71.73(b), for these tests, the initial pressure inside each containment vessel should be the maximum normal operating pressure. As shown in Table 2.21, the stresses at the maximum normal operating pressure are insignificant compared to the allowable stresses. Therefore, conducting compliance testing with nominal pressure in the containment vessel would have little, if any, effect on the results. During the structural and thermal tests conducted on the ES-3100 for HAC, the drum experienced plastic deformation, and the insulation and impact limiter material experienced some deterioration, as anticipated (Sect. 2.7). The containment vessels did not exhibit any signs of damage and passed post-test leak tests and the subsequent 10 CFR 71.73(c)(5)–specified 0.9-m (3-ft) water immersion tests except for Test Unit-6. Test Unit-6 was subjected to the test specified by paragraph 10 CFR 71.73(c)(6). After completion of this test, the containment vessel was removed and the lid was drilled and tapped for a helium leak-check port. The entire containment boundary was then helium leak checked and passed the leaktight criteria. Also, no visible water was seen inside the inner O-ring groove of Test Unit-6 and no water was observed inside any of the other test units.

Table 4.7. Regulatory leakage criteria for HAC [a]

Verification activity	Fast absorption		Medium absorption		Slow absorption	
	L_{RA-air} (ref-cm^3/s)	L_{RA-He} (cm^3/s)	L_{RA-air} (ref-cm^3/s)	L_{RA-He} (cm^3/s)	L_{RA-air} (ref-cm^3/s)	L_{RA-He} (cm^3/s)
Design	6.7061E+00	6.4189E+00	6.4258E+00	6.1523E+00	5.7237E+00	5.4839E+00

[a] The procedure used to calculate the above criteria is shown in Appendix 4.6.2.

Table 4.8. Containment vessel design verification tests for HAC

Test Type	Test Values	Leakage test procedure
	Design and compliance leakage testing	
Design verification of O-ring seal (air)	$L_T \leq 1.0 \times 10^{-4}$ ref-cm^3/s	See Appendix 2.10.7
Design verification of containment vessel boundary (helium)	$L_T \leq 2.0 \times 10^{-7}$ cm^3/s	See Appendix 2.10.7

To verify the entire containment boundary to the leaktight criteria, the containment vessels of Test Units-1 through -5 were helium leak tested using the procedure shown in the test report (Appendix 2.10.7). These test units had previously been subjected to the drop test stipulated in 10 CFR 71.71 (c)(6) and the sequential tests stipulated in 10 CFR 71.73 except for Test Unit-4, which had been first subjected to the testing in accordance with 10 CFR 71.71. The maximum recorded helium leak rate for any of these containment vessels was 2.0×10^{-7} cm^3/s after 20 min of testing on Test Unit-4 as documented in Section 5.2 of the test report (Appendix 2.10.7). Test Units-2 and -5 displayed some unusual pulsing action during leak testing. The peak amplitude changed after adding helium in a manner expected for diffusion through the O-rings rather than a rise immediately following the addition of helium that would indicate a leak to the outside of the containment vessel. This is further discussed and graphically presented in Sect. 5.2.2 of the ES-3100 test report (Appendix 2.10.7). These measured leakage rates verify that the containment vessels are leaktight in accordance with ANSI N14.5-1997. Therefore, the containment boundary of the ES-3100 package was maintained during the HAC testing.

The 35.2 kg of HEU content is unirradiated; therefore, only very small quantities of fission gas products will be produced from spontaneous fission and subcritical neutron induced fission. Fission gas products are produced in such small quantities that they have no measurable effect on the releasable content source term or containment vessel pressurization. Fission gas products will not be considered further in this SAR.

4.5 LEAKAGE RATE TESTS FOR TYPE B PACKAGES

The maximum allowable release of radioactive material allowed by 10 CFR 71.51(a)(2) under HAC is A_2 in one week. Title 10 CFR 71.51(a)(2) also specifies that there be no escape of ^{85}Kr exceeding 10 A_2 in one week. ANSI N14.5-1997 specifies the leakage test methods and leakage rates that are accepted in Nuclear Regulatory Commission (NRC) Regulatory Guide 7.4 as demonstrating that a package meets the 10 CFR 71.51(a)(2) requirements for containment. The containment criteria for the ES-3100 package will be leaktight, defined in ANSI N14.5 paragraph 2.1 as having a leakage rate $\leq 1 \times 10^{-7}$ ref-cm^3/s, during the prototype tests. This leaktight criterion satisfies the design verification requirement stipulated in paragraph 7.2.4 of ANSI N14.5-1997. The requirements of ANSI N14.5-1997 are used for all stages of containment verification for the ES-3100 (i.e., design, fabrication, maintenance, periodic and preshipment). The design, fabrication, maintenance and periodic leakage rate limit is 1×10^{-7} ref-cm^3/s air (or 2.0×10^{-7} cm^3/s helium). The pass criterion for the preshipment leakage rate test, which demonstrates correct assembly of the containment vessels, is 1×10^{-4} ref-cm^3/s, which exceeds the requirements given in ANSI N14.5-1997, paragraph 7.6.4. In accordance with the definition of sensitivity of a leakage test procedure provided in Sections 2 and 7.6.4 of ANSI N14.5-1997, the minimum acceptable leakage rate that the procedure needs to be capable of detecting is 1×10^{-3} ref-cm^3/s. The requirements for the ES-3100 exceed the regulatory criterion by specifying a leakage rate of $\leq 1 \times 10^{-4}$ ref-cm^3/s, and equipment used in accordance with Section 7.6.4 of ANSI N14.5-1997 would not detect this leakage. The preshipment, fabrication, maintenance, and periodic leakage rate tests are required to be conducted on each containment vessel in accordance with ANSI N14.5-1997 and are specified in Chapters 7 and 8. These leakage rates are not dependent on filters or mechanical cooling.

The requirements of ANSI N14.5-1997 are used for all stages of containment verification for the ES-3100; the design (HAC test) leakage rate limit is 1×10^{-7} ref-cm^3/s (which is defined as leaktight in ANSI N14.5-1997). The packaging has been shown to maintain containment before and after prototype testing by leakage tests performed for containment verification to the requirements of ANSI N14.5-1997. Test Unit-4's containment vessel was subjected to both the NCT and HAC tests. Test Units-1 through -5 were subjected to the free drop stipulated in 10 CFR 71.71(c)(7) and to the sequential HAC test stipulated in 10 CFR 71.73. Following these tests, each containment vessel was helium leak tested in accordance with the test plan. Again, the test results verified that the containment vessels were leaktight. Thus, there could be no release of radioactive materials from the containment vessels. These leakage rates are not dependent on filters or mechanical cooling. These measured leakage rates verify that the containment vessels are leaktight in accordance with ANSI N14.5-1997.

Therefore, the ES-3100 package meets the containment criteria as specified in 10 CFR 71.73 for HAC when shipping the proposed 35.2 kg of HEU in the containment vessel.

4.6 APPENDICES

Appendix	Description
4.6.1	DETERMINATION OF A_2 FOR THE ES-3100 PACKAGE WITH HEU CONTENTS
4.6.2	CALCULATION OF THE ES-3100 CONTAINMENT VESSEL'S REGULATORY REFERENCE AIR LEAKAGE RATES

Appendix 4.6.1

DETERMINATION OF A_2 FOR THE ES-3100 PACKAGE WITH HEU CONTENTS

Prepared by: Monty L. Goins
BWXT Y-12
November 2006

Reviewed by: G. A Byington
BWXT Y-12
November 2006

Appendix 4.6.1

DETERMINATION OF A_2 FOR THE ES-3100 PACKAGE WITH HEU CONTENTS

Introduction

The containment criteria for radioactive, fissile material packages are given in 10 CFR 71.51(a)(1) for Normal Conditions of Transport (NCT) ($\leq 10^{-6}$ A_2/h) and in 71.51(a)(2) for Hypothetical Accident Conditions (HAC) ($\leq A_2$ in a week). The A_2 value for this mixture of radioisotopes must be determined to establish the content containment criteria and to determine the maximum release quantity that is allowed by the regulations. These values for a mixture of isotopes are determined by the methodology given in 10 CFR 71, Appendix A, "Determination of A_1 and A_2," Sect. IV. The results of these analyses are used to demonstrate compliance of the ES-3100 package with the containment requirements of 10 CFR 71.

Scope

The A_2 value of the highly enriched uranium (HEU) content to be shipped is evaluated based on the mass and weight percents of HEU shown in Table 1 and defined in Sect. 1.2.3. The weight percents shown in Table 1 are the ones that generate the largest activity within the known weight percent ranges. Incorporating the new 10 CFR 71 A_2 values for the uranium isotopes, three different categories have been established based on the absorption rate of the uranium isotopes. The fast lung absorption, medium lung absorption and slow lung absorption categories are addressed in subsequent sections. By applying the maximum weight percents of isotopes ^{232}U, ^{233}U, ^{234}U, ^{236}U and by incorporating the traces of ^{237}Np and the transuranic isotopes, the maximum activity, minimum A_2 value, and the minimum leakage requirements were determined for each category of absorption for the HEU oxides, compounds, and metal pieces. The mass and isotopic concentrations used for the proposed content do not take into consideration limits based on shielding and subcriticality.

Table 1. Isotopic mass and weight percent for the HEU contents [a]

Nuclide	Weight percent	Mass (g)
U-232	0.000004	0.001408
U-233	0.600000	211.200000
U-234	2.000000	704.000000
U-235	54.895996	19323.390592
U-236	40.000000	14080.000000
U-238	0.000000	0.000000
Transuranic	0.004000	1.408000
Np-237	2.500000	880.000000
Total	100.000000	35200.000000

[a] Weight percents values of individual isotopes are those that generate the largest activity within the allowable ranges presented in Sect. 1.2.3.

According to 10 CFR 71, Appendix A, parent and daughter nuclides are considered to be a mixture of different nuclides than those of the parent nuclide. The radioactive decay of uranium (refer to the decay chains presented by Dr. David C. Kocher, *Radioactive Decay Data Tables* [Kocher 1981]) creates isotopes that will accumulate enough activity to exceed their respective criteria for limited quantities (*Shippers—General Requirements for Shipments and Packagings* [49 CFR 173.423], Table A-7, "Activity Limits for Limited Quantities, Instruments, and Articles") and for Type A quantities of radionuclides (10 CFR 71, Table A-1, "A_1 and A_2 Values for Radionuclides"). Furthermore, the A_2 value for the mixture will change over time as a result of radioactive decay. The analysis below shows that the A_2 value for this mixture reaches a minimum at initial fabrication, increases to the 10^{th} year, and declines through the 70^{th} year.

Analysis

Mass Tables. The mass and weight fractions for HEU isotopes used in the containment calculations are presented in Table 1. For conservatism, a small quantity of ^{232}U is assumed in this material at fabrication.

ORIGEN-S Results. The source terms of the isotopes in the mixtures were evaluated using the ORIGEN-S computer program (Parks 1984). The mass values for the parent and daughter products are presented in Table 2 for the time interval of 0–70 years. Contributions from the transuranics and ^{237}Np are held constant at each time interval during the 70-year evaluation.

The A_2 value of each mixture was calculated using the procedure shown in Tables 3, 4, and 5 for the above time intervals. The 70^{th} year of decay represents the smallest value of A_2 and the maximum activity-to-A_2 value ratio (the smallest maximum allowable leakage rate within the time interval examined). A summary of the content activity and the A_2 value of the various configurations for the above time interval is given in Table 6.

From Section 1.2.3, the activity and mass concentrations for the transuranics are 6×10^5 Bq/gU and 40 µg/gU, respectively. Using a maximum uranium mass of 35.2 kg, the mass of transuranics is calculated to be 1.408 g and the activity is calculated to be 2.112×10^{-2} TBq. Dividing this activity by the mass results in a specific activity of 1.5×10^{-2} TBq/g. In accordance with 10 CFR 71, Appendix A, Table A-3 for "contents with no relevant data," an A_2 value of 9.0×10^{-5} TBq is obtained.

Section 1.2.3 gives a maximum concentration for Np-237 of 0.025 g/gU. Using a maximum uranium mass of 35.2 kg, the mass of Np-237 is calculated to be 880 g. This value is used in Tables 3, 4, and 5 of Appendix 4.6.1 for the Np-237 contents.

Results

The containment criteria analysis indicates that the HEU contents must be shipped in a Type B material package since their activities are greater than the A_2 value. The smallest A_2 value of 1.0997×10^{-3} TBq (2.9720×10^{-2} Ci) in conjunction with the maximum activity-to-A_2 value ratio of 293.99 occurs after about 70 years of decay for the assumed maximum 35.2 kg of HEU.

Table 2. Mass values of parent and daughter products for 35.2 kg of HEU

Isotope	0 years	5 years	10 years	20 years	30 years	40 years	50 years	60 years	70 years
Pb-210	0.0000E+00	1.4150E-10	1.0912E-09	8.1664E-09	2.5555E-08	5.6672E-08	1.0349E-07	1.6896E-07	2.5274E-07
Pb-212	0.0000E+00	1.8163E-08	2.0275E-08	1.8867E-08	1.7037E-08	1.5488E-08	1.3996E-08	1.2672E-08	1.1475E-08
Bi-210	0.0000E+00	8.7296E-14	6.7302E-13	5.0054E-12	1.5770E-11	3.4918E-11	6.3923E-11	1.0419E-10	1.5558E-10
Bi-212	0.0000E+00	1.7178E-09	1.9149E-09	1.7882E-09	1.6192E-09	1.4643E-09	1.3277E-09	1.2024E-09	1.0884E-09
Po-210	0.0000E+00	2.4077E-12	1.8586E-11	1.3798E-10	4.3507E-10	9.6448E-10	1.7670E-09	2.8653E-09	4.2944E-09
Rn-222	0.0000E+00	1.4150E-12	5.6602E-12	2.2598E-11	5.0829E-11	9.0112E-11	1.4080E-10	2.0205E-10	2.7526E-10
Ra-223	0.0000E+00	6.5313E-12	2.4734E-11	9.0047E-11	1.8454E-10	2.9951E-10	4.3091E-10	5.7390E-10	7.2463E-10
Ra-224	0.0000E+00	1.5770E-07	1.7600E-07	1.6474E-07	1.4925E-07	1.3489E-07	1.2207E-07	1.1053E-07	1.0011E-07
Ra-225	0.0000E+00	2.1120E-08	4.5619E-08	9.1238E-08	1.3686E-07	1.8227E-07	2.2810E-07	2.7245E-07	3.1891E-07
Ra-226	0.0000E+00	2.2035E-07	8.8000E-07	3.5200E-06	7.8848E-06	1.4010E-05	2.1894E-05	3.1469E-05	4.2803E-05
Ra-228	0.0000E+00	2.0557E-13	6.8992E-13	2.0557E-12	3.6186E-12	5.2378E-12	6.8710E-12	8.5184E-12	1.0166E-11
Ac-225	0.0000E+00	1.6896E-08	3.0835E-08	6.1670E-08	9.2506E-08	1.2313E-07	1.5396E-07	1.8459E-07	2.1542E-07
Ac-227	0.0000E+00	4.6183E-09	1.7565E-08	6.3574E-08	1.3043E-07	2.1256E-07	3.0531E-07	4.0579E-07	5.1207E-07
Ac-228	0.0000E+00	2.5062E-17	8.4198E-17	2.5062E-16	4.4070E-16	6.3923E-16	8.3917E-16	1.0391E-15	1.2404E-15
Th-227	0.0000E+00	1.0724E-11	4.0772E-11	1.4782E-10	3.0338E-10	4.9275E-10	7.0917E-10	9.4298E-10	1.1923E-09
Th-228	0.0000E+00	3.0694E-05	3.4214E-05	3.1962E-05	2.9005E-05	2.6189E-05	2.3795E-05	2.1542E-05	1.9430E-05
Th-229	0.0000E+00	6.3360E-03	9.0394E-03	1.8058E-02	2.7034E-02	3.6115E-02	4.4986E-02	5.4067E-02	6.3149E-02
Th-230	0.0000E+00	9.7856E-03	1.9501E-02	3.9072E-02	5.8573E-02	7.8144E-02	9.7856E-02	1.1686E-01	1.3658E-01
Th-231	0.0000E+00	7.8646E-08	7.8646E-08	7.8646E-08	7.8646E-08	7.8646E-08	7.8646E-08	7.8646E-08	7.8646E-08
Th-232	0.0000E+00	2.0416E-03	4.0973E-03	8.1946E-03	1.2292E-02	1.6333E-02	2.0416E-02	2.4640E-02	2.8723E-02
Th-234	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
Pa-231	0.0000E+00	9.3525E-05	1.8724E-04	3.7487E-04	5.6231E-04	7.4782E-04	9.3525E-04	1.1227E-03	1.3082E-03
Pa-233	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
U-232	1.4080E-03	1.3376E-03	1.2742E-03	1.1546E-03	1.0447E-03	9.4618E-04	8.5747E-04	7.7581E-04	7.0259E-04
U-233	2.1120E+02	2.1119E+02	2.1119E+02	2.1118E+02	2.1117E+02	2.1116E+02	2.1116E+02	2.1115E+02	2.1114E+02
U-234	7.0400E+02	7.0399E+02	7.0398E+02	7.0396E+02	7.0394E+02	7.0392E+02	7.0390E+02	7.0388E+02	7.0386E+02
U-235	1.9323E+04	1.9323E+04	1.9323E+04	1.9323E+04	1.9323E+04	1.9323E+04	1.9323E+04	1.9323E+04	1.9323E+04
U-236	1.4080E+04	1.4080E+04	1.4080E+04	1.4080E+04	1.4080E+04	1.4080E+04	1.4080E+04	1.4080E+04	1.4080E+04
U-238	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
Trans.	1.4080E+00	1.4080E+00	1.4080E+00	1.4080E+00	1.4080E+00	1.4080E+00	1.4080E+00	1.4080E+00	1.4080E+00
Np-237	8.8000E+02	8.8000E+02	8.8000E+02	8.8000E+02	8.8000E+02	8.8000E+02	8.8000E+02	8.8000E+02	8.8000E+02
Total	**3.5200E+04**	**3.5200E+04**	**3.5200E+04**	**3.5200E+04**	**3.5200E+04**	**3.5200E+04**	**3.5200E+04**	**3.5200E+04**	**3.5200E+04**

Table 3. A_2 value calculation for 35.2 kg HEU for fast absorption uranium at 70 years

Isotope	Mass at 70 years (g)	Specific activity (TBq/g)	Activity (TBq)	A_2 (TBq)	f(i) (TBq/TBq)	f(i) / A_2 (1/TBq)
Pb-210	2.5274E-07	2.8000E+00	7.0767E-07	5.0000E-02	2.1890E-06	4.3781E-05
Pb-212	1.1475E-08	5.1000E+04	5.8523E-04	2.0000E-01	1.8103E-03	9.0514E-03
Bi-210	1.5558E-10	4.6000E+03	7.1567E-07	6.0000E-01	2.2138E-06	3.6896E-06
Bi-212	1.0884E-09	5.4000E+05	5.8774E-04	6.0000E-01	1.8180E-03	3.0301E-03
Po-210	4.2944E-09	1.7000E+02	7.3005E-07	2.0000E-02	2.2583E-06	1.1291E-04
Rn-222	2.7526E-10	5.7000E+03	1.5690E-06	4.0000E-03	4.8533E-06	1.2133E-03
Ra-223	7.2463E-10	1.9000E+03	1.3768E-06	7.0000E-03	4.2588E-06	6.0841E-04
Ra-224	1.0011E-07	5.9000E+03	5.9065E-04	2.0000E-02	1.8271E-03	9.1353E-02
Ra-225	3.1891E-07	1.5000E+03	4.7837E-04	4.0000E-03	1.4797E-03	3.6993E-01
Ra-226	4.2803E-05	3.7000E-02	1.5837E-06	3.0000E-03	4.8989E-06	1.6330E-03
Ra-228	1.0166E-11	1.0000E+01	1.0166E-10	2.0000E-02	3.1446E-10	1.5723E-08
Ac-225	2.1542E-07	2.1000E+03	4.5238E-04	6.0000E-03	1.3994E-03	2.3323E-01
Ac-227	5.1207E-07	2.7000E+00	1.3826E-06	9.0000E-05	4.2768E-06	4.7519E-02
Ac-228	1.2404E-15	8.4000E+04	1.0419E-10	5.0000E-01	3.2230E-10	6.4460E-10
Th-227	1.1923E-09	1.1000E+03	1.3115E-06	5.0000E-03	4.0569E-06	8.1139E-04
Th-228	1.9430E-05	3.0000E+01	5.8290E-04	1.0000E-03	1.8031E-03	1.8031E+00
Th-229	6.3149E-02	7.9000E-03	4.9888E-04	5.0000E-04	1.5432E-03	3.0863E+00
Th-230	1.3658E-01	7.6000E-04	1.0380E-04	1.0000E-03	3.2109E-04	3.2109E-01
Th-231	7.8646E-08	2.0000E+04	1.5729E-03	2.0000E-02	4.8655E-03	2.4328E-01
Th-232	2.8723E-02	4.0000E-09	1.1489E-10	1.0000E+75	3.5539E-10	3.5539E-85
Pa-231	1.3082E-03	1.7000E-03	2.2239E-06	4.0000E-04	6.8793E-06	1.7198E-02
U-232	7.0259E-04	8.3000E-01	5.8315E-04	1.0000E-02	1.8039E-03	1.8039E-01
U-233	2.1114E+02	3.6000E-04	7.6010E-02	9.0000E-02	2.3512E-01	2.6125E+00
U-234	7.0386E+02	2.3000E-04	1.6189E-01	9.0000E-02	5.0077E-01	5.5641E+00
U-235	1.9323E+04	8.0000E-08	1.5458E-03	1.0000E+75	4.7817E-03	4.7817E-78
U-236	1.4080E+04	2.4000E-06	3.3792E-02	1.0000E+75	1.0453E-01	1.0453E-76
Transuranic	1.4080E+00	1.5000E-02	2.1120E-02	9.0000E-05	6.5330E-02	7.2588E+02
Np-237	8.8000E+02	2.6000E-05	2.2880E-02	2.0000E-03	7.0775E-02	3.5387E+01
Total Mass =	3.5200E+04	∑ Act. =	3.2328E-01		∑ f(i) / A_2 =	7.7585E+02

$$A_2\,(\text{mixture}) = \frac{1}{\sum f(i) / A_2} = \frac{1}{7.7586 \times 10^{2}\ (1/\text{TBq})} = 1.2889 \times 10^{-3}\ \text{TBq}$$

Table 4. A_2 value calculation for 35.2 kg HEU for medium absorption uranium at 70 years

Isotope	Mass at 70 years (g)	Specific activity (TBq/g)	Activity (TBq)	A_2 (TBq)	f(i) (TBq/TBq)	f(i) / A_2 (1/TBq)
Pb-210	2.5274E-07	2.8000E+00	7.0767E-07	5.0000E-02	2.1890E-06	4.3781E-05
Pb-212	1.1475E-08	5.1000E+04	5.8523E-04	2.0000E-01	1.8103E-03	9.0514E-03
Bi-210	1.5558E-10	4.6000E+03	7.1567E-07	6.0000E-01	2.2138E-06	3.6896E-06
Bi-212	1.0884E-09	5.4000E+05	5.8774E-04	6.0000E-01	1.8180E-03	3.0301E-03
Po-210	4.2944E-09	1.7000E+02	7.3005E-07	2.0000E-02	2.2583E-06	1.1291E-04
Rn-222	2.7526E-10	5.7000E+03	1.5690E-06	4.0000E-03	4.8533E-06	1.2133E-03
Ra-223	7.2463E-10	1.9000E+03	1.3768E-06	7.0000E-03	4.2588E-06	6.0841E-04
Ra-224	1.0011E-07	5.9000E+03	5.9065E-04	2.0000E-02	1.8271E-03	9.1353E-02
Ra-225	3.1891E-07	1.5000E+03	4.7837E-04	4.0000E-03	1.4797E-03	3.6993E-01
Ra-226	4.2803E-05	3.7000E-02	1.5837E-06	3.0000E-03	4.8989E-06	1.6330E-03
Ra-228	1.0166E-11	1.0000E+01	1.0166E-10	2.0000E-02	3.1446E-10	1.5723E-08
Ac-225	2.1542E-07	2.1000E+03	4.5238E-04	6.0000E-03	1.3994E-03	2.3323E-01
Ac-227	5.1207E-07	2.7000E+00	1.3826E-06	9.0000E-05	4.2768E-06	4.7519E-02
Ac-228	1.2404E-15	8.4000E+04	1.0419E-10	5.0000E-01	3.2230E-10	6.4460E-10
Th-227	1.1923E-09	1.1000E+03	1.3115E-06	5.0000E-03	4.0569E-06	8.1139E-04
Th-228	1.9430E-05	3.0000E+01	5.8290E-04	1.0000E-03	1.8031E-03	1.8031E+00
Th-229	6.3149E-02	7.9000E-03	4.9888E-04	5.0000E-04	1.5432E-03	3.0863E+00
Th-230	1.3658E-01	7.6000E-04	1.0380E-04	1.0000E-03	3.2109E-04	3.2109E-01
Th-231	7.8646E-08	2.0000E+04	1.5729E-03	2.0000E-02	4.8655E-03	2.4328E-01
Th-232	2.8723E-02	4.0000E-09	1.1489E-10	1.0000E+75	3.5539E-10	3.5539E-85
Pa-231	1.3082E-03	1.7000E-03	2.2239E-06	4.0000E-04	6.8793E-06	1.7198E-02
U-232	7.0259E-04	8.3000E-01	5.8315E-04	7.0000E-03	1.8039E-03	2.5769E-01
U-233	2.1114E+02	3.6000E-04	7.6010E-02	2.0000E-02	2.3512E-01	1.1756E+01
U-234	7.0386E+02	2.3000E-04	1.6189E-01	2.0000E-02	5.0077E-01	2.5038E+01
U-235	1.9323E+04	8.0000E-08	1.5458E-03	1.0000E+75	4.7817E-03	4.7817E-78
U-236	1.4080E+04	2.4000E-06	3.3792E-02	2.0000E-02	1.0453E-01	5.2264E+00
Transuranic	1.4080E+00	1.5000E-02	2.1120E-02	9.0000E-05	6.5330E-02	7.2589E+02
Np-237	8.8000E+02	2.6000E-05	2.2880E-02	2.0000E-03	7.0775E-02	3.5387E+01
Total Mass =	3.5200E+04	$\sum$ Act. =	3.2328E-01		$\sum$ f(i) / A_2 =	8.0978E+02

$$A_2\,(\text{mixture}) = \frac{1}{\sum f(i) / A_2} = \frac{1}{8.0978 \times 10^{2}\ (1/\text{TBq})} = 1.2349 \times 10^{-3}\ \text{TBq}$$

Table 5. A_2 value calculation for 35.2 kg HEU for slow absorption uranium at 70 years

Isotope	Mass at 70 years (g)	Specific activity (TBq/g)	Activity (TBq)	A_2 (TBq)	f(i) (TBq/TBq)	f(i) / A_2 (1/TBq)
Pb-210	2.5274E-07	2.8000E+00	7.0767E-07	5.0000E-02	2.1890E-06	4.3781E-05
Pb-212	1.1475E-08	5.1000E+04	5.8523E-04	2.0000E-01	1.8103E-03	9.0514E-03
Bi-210	1.5558E-10	4.6000E+03	7.1567E-07	6.0000E-01	2.2138E-06	3.6896E-06
Bi-212	1.0884E-09	5.4000E+05	5.8774E-04	6.0000E-01	1.8180E-03	3.0301E-03
Po-210	4.2944E-09	1.7000E+02	7.3005E-07	2.0000E-02	2.2583E-06	1.1291E-04
Rn-222	2.7526E-10	5.7000E+03	1.5690E-06	4.0000E-03	4.8533E-06	1.2133E-03
Ra-223	7.2463E-10	1.9000E+03	1.3768E-06	7.0000E-03	4.2588E-06	6.0841E-04
Ra-224	1.0011E-07	5.9000E+03	5.9065E-04	2.0000E-02	1.8271E-03	9.1353E-02
Ra-225	3.1891E-07	1.5000E+03	4.7837E-04	4.0000E-03	1.4797E-03	3.6993E-01
Ra-226	4.2803E-05	3.7000E-02	1.5837E-06	3.0000E-03	4.8989E-06	1.6330E-03
Ra-228	1.0166E-11	1.0000E+01	1.0166E-10	2.0000E-02	3.1446E-10	1.5723E-08
Ac-225	2.1542E-07	2.1000E+03	4.5238E-04	6.0000E-03	1.3994E-03	2.3323E-01
Ac-227	5.1207E-07	2.7000E+00	1.3826E-06	9.0000E-05	4.2768E-06	4.7519E-02
Ac-228	1.2404E-15	8.4000E+04	1.0419E-10	5.0000E-01	3.2230E-10	6.4460E-10
Th-227	1.1923E-09	1.1000E+03	1.3115E-06	5.0000E-03	4.0569E-06	8.1139E-04
Th-228	1.9430E-05	3.0000E+01	5.8290E-04	1.0000E-03	1.8031E-03	1.8031E+00
Th-229	6.3149E-02	7.9000E-03	4.9888E-04	5.0000E-04	1.5432E-03	3.0863E+00
Th-230	1.3658E-01	7.6000E-04	1.0380E-04	1.0000E-03	3.2109E-04	3.2109E-01
Th-231	7.8646E-08	2.0000E+04	1.5729E-03	2.0000E-02	4.8655E-03	2.4328E-01
Th-232	2.8723E-02	4.0000E-09	1.1489E-10	1.0000E+75	3.5539E-10	3.5539E-85
Pa-231	1.3082E-03	1.7000E-03	2.2239E-06	4.0000E-04	6.8793E-06	1.7198E-02
U-232	7.0259E-04	8.3000E-01	5.8315E-04	1.0000E-03	1.8039E-03	1.8039E+00
U-233	2.1114E+02	3.6000E-04	7.6010E-02	6.0000E-03	2.3512E-01	3.9187E+01
U-234	7.0386E+02	2.3000E-04	1.6189E-01	6.0000E-03	5.0077E-01	8.3461E+01
U-235	1.9323E+04	8.0000E-08	1.5458E-03	1.0000E+75	4.7817E-03	4.7817E-78
U-236	1.4080E+04	2.4000E-06	3.3792E-02	6.0000E-03	1.0453E-01	1.7421E+01
Transuranic	1.4080E+00	1.5000E-02	2.1120E-02	9.0000E-05	6.5330E-02	7.2589E+02
Np-237	8.8000E+02	2.6000E-05	2.2880E-02	2.0000E-03	7.0775E-02	3.5387E+01
Total Mass =	3.5200E+04	$\sum$ Act. =	3.2328E-01		$\sum$ f(i) / A_2 =	9.0938E+02

$$A_2\,(\text{mixture}) = \frac{1}{\sum f(i) / A_2} = \frac{1}{9.0937 \times 10^{2}\ (1/\text{TBq})} = 1.0997 \times 10^{-3}\ \text{TBq}$$

Table 6. Activity, A_2 value, and activity-to-A_2 ratio for the enriched uranium contents

Year	Activity (TBq)	A_2-Mixture (TBq)			Activity/A_2		
		Fast	Medium	Slow	Fast	Medium	Slow
0	3.1846E-01	1.2796E-03	1.2255E-03	1.0885E-03	248.87	259.86	292.57
5	3.2380E-01	1.2950E-03	1.2405E-03	1.1026E-03	250.04	261.03	293.68
10	3.2427E-01	1.2960E-03	1.2415E-03	1.1037E-03	250.21	261.20	293.81
20	3.2410E-01	1.2948E-03	1.2404E-03	1.1031E-03	250.31	261.29	293.82
30	3.2386E-01	1.2934E-03	1.2391E-03	1.1022E-03	250.40	261.38	293.82
40	3.2366E-01	1.2921E-03	1.2379E-03	1.1015E-03	250.49	261.47	293.84
50	3.2350E-01	1.2909E-03	1.2368E-03	1.1008E-03	250.59	261.56	293.87
60	3.2338E-01	1.2899E-03	1.2358E-03	1.1002E-03	250.71	261.68	293.93
70	3.2328E-01[a]	1.2889E-03	1.2349E-03	1.0997E-03[a]	250.82	261.79	293.99[a]

[a] The A_2, total activity, and activity-to-A_2 ratio values used in Appendix 4.6.2.

Appendix 4.6.2

CALCULATION OF THE ES-3100 CONTAINMENT VESSEL'S REGULATORY REFERENCE AIR LEAKAGE RATES

Prepared by: Monty L. Goins
BWXT Y-12
November 2006

Reviewed by: G. A. Byington
BWXT Y-12
November 2006

Appendix 4.6.2

CALCULATION OF THE ES-3100 CONTAINMENT VESSEL'S REGULATORY REFERENCE AIR LEAKAGE RATES

Introduction

The ES-3100 leak-testing requirements of the containment boundary are based on the smallest maximum allowable leakage rate generated from the maximum uranium content defined in Table 4.3. Section 5 of ANSI N14.5-1997 defines the maximum allowable leakage rate based on the maximum allowable release rate. These leakage rates, L_N and L_A, are the maximum allowable O-ring seal leakage rates for Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC). The worst-case maximum allowable leakage rates are used to calculate an equivalent leakage hole diameter following ANSI N14.5-1997, Appendix B, for each condition of transport. This leakage hole diameter is used to calculate a reference air and a helium leakage rate for leak testing. A bounding mass for the highly enriched uranium (HEU) content of 35.2 kg is used in this calculation to certify the ES-3100 package for shipment. The maximum allowable leakage rates are calculated using this maximum content mass in a much more dispersive form (oxide powder) at the highest calculated pressures and temperatures. This appendix shows the procedure used to calculate the leak criteria for the uranium constituents in the "slow lung absorption" group. Table 1 shows the results of using this procedure for fast, medium, and slow absorption uranium constituents as a function of decay time.

HEU Content

Calculate R_N and R_A: Table 4.1

The maximum allowable release rate is based on using A_2.

$A_2 = 1.0997 \times 10^{-3}$ TBq, (2.9720×10^{-2} Ci). Table 6 (Appendix 4.6.1)

The containment requirements for NCT and HAC are:

$R_N = A_2 \times 10^{-6}$ TBq/h $= A_2 \times 2.78 \times 10^{-10}$ TBq/s, ANSI N14.5-1997 (Eq. 1)
$= 1.0997 \times 10^{-3} \times 2.78 \times 10^{-10}$ TBq/s,
$R_N = 3.0570 \times 10^{-13}$ TBq/s, (8.2623×10^{-12} Ci/s).

$R_A = A_2$ (TBq/week),
$= A_2 \times 1.65 \times 10^{-6}$ (TBq/s), ANSI N14.5-1997 (Eq. 2)
$= 1.0997 \times 10^{-3} \times 1.65 \times 10^{-6}$ TBq/s,
$= 1.8144 \times 10^{-9}$ TBq/s, (4.9038×10^{-8} Ci/s). or limited to 10 A_2/week of ^{85}Kr

Following ANSI N14.5-1997, the medium aerosol activity must be calculated to determine the leakage rates.

m = total nuclide mass in the package available for release (g),
TotA = total activity in the package available for release (TBq),
TSA = total specific activity in the package available for release (TBq/g).

For HEU content:

TSA	=	TotA / m,	
TSA	=	3.2328×10^{-1} (TBq) / 35,200 (g),	Table 6 (Appendix 4.6.1)
TSA	=	9.1842×10^{-6} TBq/g.	

ρ_P	=	9×10^{-6} g/cm^3.	The maximum density of powder aerosols in the fill gas

For any packaging arrangement:

C_N	=	activity per unit volume of medium that could escape from the containment system (TBq/cm^3).
C_N	=	TSA $\times \rho_P$,
	=	9.1842×10^{-6} (TBq/g) $\times 9 \times 10^{-6}$ (g/cm^3),
C_N	=	8.2658×10^{-11} TBq/cm^3.

Using Curren's maximum aerosol density, $C_A = C_N$:

C_A	=	activity per unit volume of exiting gas (TBq/cm^3),	HAC
C_A	=	8.2658×10^{-11} TBq/cm^3.	

Section 6.1 of ANSI N14.5-1997 calculates L_N with (Eq. 3) and L_A with (Eq. 4). L_N and L_A are the maximum allowable leakage rates for the containment vessel fill gas aerosol during NCT and HAC, respectively.

L_N	=	maximum allowable leakage rate for the medium for NCT (TBq/cm^3),	
L_N	=	R_N / C_N,	ANSI N14.5-1997 (Eq. 3)
L_N	=	3.0570×10^{-13} (TBq/s) / 8.2658×10^{-11} (TBq/cm^3),	
L_N	=	3.6984×10^{-3} cm^3/s.	

L_A	=	maximum allowable leakage rate for the medium for HAC (TBq/cm^3),
L_A	=	R_A / C_A,
L_A	=	1.8144×10^{-9} (TBq/s) / 8.2658×10^{-11} (TBq/cm^3),
L_A	=	2.1951×10^{1} cm^3/s.

L_N and L_A correspond to the upstream volumetric leakage rate (L_u) at the upstream pressure (P_u) in the ANSI N14.5-1997 formulas for use later in this appendix. The reference air leakage rates $L_{R,N}$ and $L_{R,A}$ for NCT and HAC, based on the L_N and L_A, are then calculated using maximum temperatures and pressure combinations from Table 3.16 and Table 5 in Appendix 3.6.5.

Determination of the Leakage Test Procedure Requirements for the HEU Content

This calculation will examine the most conservative effects of a fully loaded containment vessel with an HEU mass of 35.2 kg. The smallest allowable leakage values are shown in Tables 4.5 and 4.7. The A_2 value and the maximum content activity-to-A_2 value ratio for this mixture were calculated for several different decay times (Table 6, Appendix 4.6.1). As calculated in Appendix 4.6.1, the A_2 value and the maximum content activity-to-A_2 ratio used to qualify this package occur at about 70 years of decay and are 1.0997×10^{-3} TBq (2.9720×10^{-2} Ci) and 293.99, respectively. These values are used to determine the leakage test procedural requirements when packaging any convenience cans/contents arrangements in the ES-3100 package. The convenience cans are sealed inside the containment vessel in an environmentally controlled area. The ES-3100 package has been analyzed thermally in Sect. 3; it was evaluated at a maximum NCT gas temperature of 87.81°C (190.06°F) [100°F with solar insolation] and a maximum adjusted HAC gas temperature of 123.85°C (254.93°F).

The following analysis determines the maximum allowable O-ring seal air reference leakage rate for both NCT and HAC. The ANSI N14.5-1997 recommended method using a straight circular tube to model the leakage path is applied. Using this "standard" leakage hole model permits the calculation of equivalent reference leakage rates from which leak-test requirements can be established. Viscosity data for air and helium used in the following analyses were obtained from curve fitting routines at specific temperatures based on viscosity data for air (Handbook of Chemistry and Physics, 55th ed.) and helium (NBS Technical Note 631).

L_N and L_A correspond to the upstream volumetric leakage rate (L_u) at the upstream pressure (P_u).

$$L_N = 3.6984 \times 10^{-3}\ cm^3/s,$$
$$L_A = 2.1951 \times 10^{1}\ cm^3/s.$$

Find the maximum pressure and temperature in the containment vessel:

Converting the temperature to degrees Kelvin:

$$T = 273.15 + T(°C),$$
$$T = 273.15 + 5 / 9\ (°F - 32)\ (K).$$

$$T_N = 273.15 + 5 / 9\ (190.06°F - 32)\ (K), \quad \text{(Sect. 3.4.1, for T = 190.06°F)}$$
$$T_N = 360.961\ K. \quad \text{NCT}$$

$$T_A = 273.15 + 5 / 9\ (254.93°F - 32)\ (K), \quad \text{(Sect. 3.5.3, for T = 254.93°F)}$$
$$T_A = 397.000\ K. \quad \text{HAC}$$

Converting the pressures from psia to atmospheres:

$$P_N = P\ (psia) / 14.696\ (psia/atm), \quad \text{where P is the pressure in Sect. 3.4.2}$$
$$P_N = 20.004 (psia) / 14.696\ (psia/atm), \quad \text{NCT}$$
$$P_N = 1.3612\ atm.$$

$$P_A = P\ (psia) / 14.696\ (psia/atm), \quad \text{where P is the pressure in Sect. 3.5.3}$$
$$P_A = 73.625\ (psia) / 14.696\ (psia/atm), \quad \text{HAC}$$
$$P_A = 5.0099\ atm.$$

NCT Leakage Hole Diameter for the HEU Content

The following calculations determine the leakage hole diameter that generates the maximum allowable leakage rate during NCT. To keep these calculations conservative, the maximum values for temperature and pressure were used as steady-state conditions for NCT.

Input data for NCT with air fill gas:

L_N	=	3.6984×10^{-3} cm^3/s,	Maximum upstream leakage
P_u	=	1.3612 atm,	Upstream pressure = 20.004 psia
P_d	=	0.2382 atm,	Downstream pressure = 3.5 psia, per 10 CFR 71.71(3)
a	=	0.3531 cm,	Leak path length, 0.139-in. O-ring section diameter
T	=	360.96 K,	Fill gas temperature = 190.06°F
μ	=	0.02141 cP,	Viscosity at temperature
M	=	29 g/g-mole.	Molecular weight of fill gas

The average pressure is:

$$P_a = (P_u + P_d)/2\,,$$
$$= (1.3612 + 0.2382) / 2\,,$$
$$P_a = 0.7997 \text{ atm.} \qquad \text{Average pressure during NCT}$$

According to ANSI N14.5-1997, the flow leakage hole diameter is unknown. Therefore, the mass-like leakage flow rate must be calculated to calculate the average leakage flow rate.

Q is the mass-like leakage for flow using the upstream leakage, L_u, and pressure, P_u:

$$Q = P_u L_u\,, \qquad \text{(Eq. B1)}$$
$$L_u = L_N. \qquad \text{NCT leakage}$$

$$Q = (1.3612)(\text{atm})\ (3.6984 \times 10^{-3})(\text{cm}^3/\text{s}),$$
$$Q = 5.0343 \times 10^{-3} \text{ atm-cm}^3/\text{s}. \qquad \text{NCT mass-like leakage rate}$$

$$Q = P_a L_a\,, \qquad \text{(Eq. B1)}$$
$$L_a = Q / P_a = 5.0353 \times 10^{-3} (\text{atm-cm}^3/\text{s}) / (0.7997)(\text{atm}),$$
$$L_a = 6.2954 \times 10^{-3} \text{ cm}^3/\text{s}. \qquad \text{NCT average leakage rate}$$

Solve equations B2–B4 from ANSI N14.5-1997:

$$L_a = (F_c + F_m)\ (P_u - P_d) \text{ cm}^3/\text{s}, \qquad \text{(Eq. B2)}$$
$$L_a = (F_c + F_m)\ (1.3612 - 0.2382),$$
$$L_a = (1.1230)\ (F_c + F_m) \text{ cm}^3/\text{s}.$$

$$F_c = (2.49 \times 10^6)\ D^4 / (a\ \mu)\ (\text{cm}^3/\text{atm-s}), \qquad \text{(Eq. B3)}$$
$$F_c = (2.49 \times 10^6)\ D^4 / (\ (0.3531)\ (0.02141)\ ),$$
$$F_c = (3.2943 \times 10^8)\ D^4 \text{ cm}^3/\text{atm-s}.$$

$$F_m = (3.81 \times 10^3)\ D^3\ (T / M)^{.5} / (a\ P_a)\ (\text{cm}^3/\text{atm-s}), \qquad \text{(Eq. B4)}$$
$$F_m = (3.81 \times 10^3)\ D^3\ (360.96 / 29)^{.5} / (\ (0.3531)\ (0.7997)\ ),$$
$$F_m = (4.7610 \times 10^4)\ D^3 \text{ cm}^3/\text{atm-s}.$$

From the mass-like leakage calculation:

$$L_a = 6.2954 \times 10^{-3} \text{ cm}^3/\text{s}. \qquad \text{NCT average leakage rate}$$

Find the leakage hole diameter that sets:

$$L_2 = L_a.$$

Using the equations:

$$L_2 = (1.1230)(F_c + F_m)\ cm^3/s,$$
$$F_c = (3.2943 \times 10^8)\ D^4\ cm^3/atm\text{-}s,$$
$$F_m = (4.7610 \times 10^4)\ D^3\ cm^3/atm\text{-}s.$$

To get a better guess on a new D use:

$$D = D_2\,(L_a / L_2)^{0.252}.$$

Now a guess must be made for D_2 to solve Eq. B2 for NCT:

D_2 = 0.001 cm, and solve for $L_a = 6.2954 \times 10^{-3}$ cm^3/s. NCT average leakage rate

Diameter	F_c	F_m	L_2	L_a / L_2
1.00000E-03	3.29434E-04	4.76097E-05	4.23431E-04	1.48676E+01
1.97426E-03	5.00481E-03	3.66361E-04	6.03197E-03	1.04367E+00
1.99564E-03	5.22516E-03	3.78393E-04	6.29294E-03	1.00039E+00
1.99584E-03	5.22721E-03	3.78504E-04	6.29537E-03	1.00001E+00
1.99584E-03	5.22727E-03	3.78507E-04	6.29543E-03	9.99997E-01

The NCT leakage hole diameter for the HEU oxide content:

$D = 1.9958 \times 10^{-3}$ cm. NCT diameter

NCT Reference Air Leakage Rate for HEU Content

The leakage hole diameter found for the maximum allowable leakage rate for NCT will be used to determine the reference air leakage rate. O-ring seal leakage testing must ensure that no leakage is greater than the leakage generated by the hole diameter $D = 1.9958 \times 10^{-3}$ cm. Therefore, the NCT reference leakage flow rate ($L_{R,N}$) must be calculated to determine the allowable test leakage rate.

Input data for NCT reference air leakage rate:

D	=	1.9958×10^{-3} cm,	From NCT
a	=	0.3531 cm,	Leak path length, 0.139-in. O-ring section diameter
P_u	=	1.0 atm,	Upstream pressure
P_d	=	0.01 atm,	Downstream pressure
T	=	298 K,	Fill gas temperature, 77°F
M	=	29 g/g-mole,	Molecular weight of air
μ	=	0.0185 cP,	Viscosity of air at reference temperature

Calculate P_a:

$$
\begin{aligned}
P_a &= (P_u + P_d) / 2, \\
&= (1.0 + 0.01) / 2, \\
P_a &= 0.505 \text{ atm.} && \text{NCT average pressure}
\end{aligned}
$$

$$
\begin{aligned}
F_c &= (2.49 \times 10^6)\, D^4 / (a\, \mu)\ (\text{cm}^3/\text{atm-s}), && \text{(Eq. B3)} \\
F_c &= (2.49 \times 10^6)(1.9958 \times 10^{-3})^4 / ((0.3531)(0.0185)), \\
F_c &= (3.8122 \times 10^8)(1.9958 \times 10^{-3})^4, \\
F_c &= 6.04790 \times 10^{-3} \text{ cm}^3/\text{atm-s.}
\end{aligned}
$$

$$
\begin{aligned}
F_m &= (3.81 \times 10^3)\, D^3 (T / M)^{.5} / (a\, P_a)\ (\text{cm}^3/\text{atm-s}), && \text{(Eq. B4)} \\
F_m &= (3.81 \times 10^3)(1.9958 \times 10^{-3})^3 (298 / 29)^{.5} / ((0.3531)(0.505)), \\
F_m &= (6.8501 \times 10^4)(1.9958 \times 10^{-3})^3, \\
F_m &= 5.4459 \times 10^{-4} \text{ cm}^3/\text{atm-s.}
\end{aligned}
$$

$$
\begin{aligned}
L_u &= (F_c + F_m)(P_u - P_d)(P_a / P_u)\ (\text{cm}^3/\text{s}), && \text{(Eq. B5)} \\
L_u &= (6.0490 \times 10^{-3} + 5.4459 \times 10^{-4})(\text{cm}^3/\text{atm-s})(1.0 - 0.01)(\text{atm})(0.505 / 1.0), \\
L_u &= (6.5937 \times 10^{-3})(\text{cm}^3/\text{atm-s})(0.49995)(\text{atm}), \\
L_u &= 3.2965 \times 10^{-3} \text{ cm}^3/\text{s.}
\end{aligned}
$$

The reference air leakage rate as defined in ANSI N14.5-1997, Sect. B.3, is the upstream leakage in air.

$$
L_{RN,Air} = 3.2965 \times 10^{-3} \text{ ref-cm}^3/\text{s.} \qquad \text{For HEU oxide content}
$$

The same equations can be used to calculate an allowable leakage rate using helium for leak testing.

$$
\begin{aligned}
M &= 4 \text{ g/g-mole,} && \text{Molecular weight of helium} \\
\mu &= 0.0198 \text{ cP.} && \text{Viscosity of helium at temperature}
\end{aligned}
$$

$$
\begin{aligned}
F_c &= (2.49 \times 10^6)\, D^4 / (a\, \mu)\ (\text{cm}^3/\text{atm-s}), && \text{(Eq. B3)} \\
F_c &= (2.49 \times 10^6)(1.9958 \times 10^{-3})^4 / ((0.3531)(0.0198)), \\
F_c &= (3.5619 \times 10^8)(1.9958 \times 10^{-3})^4, \\
F_c &= 5.6518 \times 10^{-3} \text{ cm}^3/\text{atm-s.}
\end{aligned}
$$

$$
\begin{aligned}
F_m &= (3.81 \times 10^3)\, D^3 (T / M)^{.5} / (a\, P_a)\ (\text{cm}^3/\text{atm-s}), && \text{(Eq. B4)} \\
F_m &= (3.81 \times 10^3)(1.9958 \times 10^{-3})^3 (298 / 4)^{.5} / ((0.3531)(0.505)), \\
F_m &= (1.8444 \times 10^5)(1.9958 \times 10^{-3})^3, \\
F_m &= 1.4664 \times 10^{-3} \text{ cm}^3/\text{atm-s.}
\end{aligned}
$$

$$
\begin{aligned}
L_u &= (F_c + F_m)(P_u - P_d)(P_a / P_u)\ (\text{cm}^3/\text{s}), && \text{(Eq. B5)} \\
L_u &= (5.6518 \times 10^{-3} + 1.4664 \times 10^{-3})(\text{cm}^3/\text{atm-s})(1.0 - 0.01)(\text{atm})(0.505 / 1.0), \\
L_u &= (7.1182 \times 10^{-3})(\text{cm}^3/\text{atm-s})(0.49995)(\text{atm}), \\
L_u &= 3.5587 \times 10^{-3} \text{ cm}^3/\text{s.}
\end{aligned}
$$

The allowable leakage rate using helium for leak testing is:

$$
L_{RN,He} = 3.5587 \times 10^{-3} \text{ cm}^3/\text{s.} \qquad \text{NCT helium test value}
$$

HAC Leakage Hole Diameter for HEU Content

The calculation of a maximum allowable leakage rate hole diameter is based on the temperature and pressure of the fill gas aerosol for HAC, assuming the content is in an oxide powder form. Keeping this calculation conservative, the maximum values for temperature and pressure were used as steady-state conditions for a week. The maximum values were generated during the 30-min burn test for HAC.

Input data for HAC:

L_A	=	21.951 cm^3/s,	Maximum exit leakage
P_u	=	5.0099 atm,	Upstream pressure = 73.625 psia
P_d	=	1.0 atm,	Downstream pressure
T	=	397.000 K,	Fill gas temperature = 254.93 °F
μ	=	0.02297 cP,	Viscosity of air at temperature
M	=	29 g/g-mole,	Molecular weight of air
a	=	0.3531 cm.	Leak path length, 0.139-in. O-ring section diameter

$$P_a = (P_u + P_d) / 2$$
$$= (5.0099 + 1.0) / 2, \qquad \text{HAC average pressure}$$
$$P_a = 3.0049 \text{ atm.}$$

Q is the mass-like leakage for flow using the upstream leakage, L_u, and pressure, P_u:

$$Q = P_u L_u, \qquad \text{(Eq. B1)}$$
$$L_u = L_A. \qquad \text{HAC leakage}$$

$$Q = (5.0099)(\text{atm})\ (21.951)(\text{cm}^3/\text{s}),$$
$$Q = 109.973 \text{ atm-cm}^3/\text{s}. \qquad \text{HAC mass-like leakage rate}$$

$$Q = P_a L_a, \qquad \text{(Eq. B1)}$$
$$L_a = Q / P_a$$
$$= 109.973\ (\text{atm-cm}^3/\text{s}) / (3.0049)(\text{atm}),$$
$$L_a = 36.597 \text{ cm}^3/\text{s}. \qquad \text{HAC average leakage rate}$$

Solve equations B2–B4 from ANSI N14.5-1997:

$$L_a = (F_c + F_m)\ (P_u - P_d)\ (\text{cm}^3/\text{s}), \qquad \text{(Eq. B2)}$$
$$L_a = (F_c + F_m)\ (5.0099 - 1.0),$$
$$L_a = 4.0099\ (F_c + F_m) \text{ cm}^3/\text{s}.$$

$$F_c = (2.49 \times 10^6)\ D^4 / (a\ \mu)\ (\text{cm}^3/\text{atm-s}), \qquad \text{(Eq. B3)}$$
$$F_c = (2.49 \times 10^6)\ D^4 / ((0.3531)\ (0.02297)),$$
$$F_c = (3.0706 \times 10^8)\ D^4 \text{ cm}^3/\text{atm-s}.$$

$$F_m = (3.81 \times 10^3)\ D^3\ (T / M)^{.5} / (a\ P_a)\ (\text{cm}^3/\text{atm-s}), \qquad \text{(Eq. B4)}$$
$$F_m = (3.81 \times 10^3)\ D^3\ (397.00 / 29)^{.5} / ((0.3531)\ (3.0049)),$$
$$F_m = (1.3287 \times 10^4)\ D^3 \text{ cm}^3/\text{atm-s}.$$

From the mass-like leakage calculation:

L_a = 36.597 cm^3/s. HAC average leakage rate

Find the leakage hole diameter that sets:

L_2 = L_a.

Using the equations:

$$L_2 = 4.0099\,(F_c + F_m)\ cm^3/s,$$
$$F_c = (3.0706 \times 10^8)\,D^4\ cm^3/atm\text{-}s,$$
$$F_m = (1.3287 \times 10^4)\,D^3\ cm^3/atm\text{-}s.$$

To get a better guess on a new D use:

$$D = D_2\,(L_a / L_2)^{0.252}.$$

Now a guess must be made for D_2 to solve Eq. B2 for HAC:

D_2 = 0.01 (cm), and solve for L_a = 36.597 (cm^3/s). HAC average leakage rate

Diameter	F_c	F_m	L_2	L_a / L_2
1.0000E-02	3.0706E+00	1.3287E-02	1.2366E+01	2.9595E+00
1.3145E-02	9.1678E+00	3.0179E-02	3.6883E+01	9.9225E-01
1.3119E-02	9.0955E+00	3.0001E-02	3.6592E+01	1.0001E+00
1.3119E-02	9.0955E+00	3.0001E-02	3.6592E+01	1.0001E+00

The HAC leakage hole diameter for the HEU oxide content is:

D = 1.3119×10^{-2} cm. HAC diameter

HAC Reference Air Leakage Rate for HEU Content

The leakage hole diameter found for the maximum allowable leakage rate for HAC will be used to determine the reference air leakage rate. O-ring seal leakage testing must assure that no leakage is greater than the leakage generated by the hole diameter D = 1.3119×10^{-2} cm. Therefore, the HAC reference air leakage rate ($L_{R,\,A}$) must be calculated to determine the acceptable test leakage rate for post-HAC leakage testing.

Input data for HAC reference air leakage rate:

D = 1.3119×10^{-2} cm, From the HAC of transport
a = 0.3531 cm, Leak path length, 0.139-in. O-ring section diameter
P_u = 1.0 atm, Upstream pressure

P_d	=	0.01 atm,	Downstream pressure
T	=	298 K,	Fill gas temperature, 77°F
M	=	29 g/g-mole,	Molecular weight of air
μ	=	0.0185 cP.	Viscosity at temperature

Calculate P_a:

$P_a = (P_u + P_d) / 2$

$= 0.505$ atm. HAC average pressure

$F_c = (2.49 \times 10^6)\, D^4 / (a\, \mu)$ (cm³/atm-s), (Eq. B3)

$F_c = (2.49 \times 10^6)\,(1.3119 \times 10^{-2})^4 / ((0.3531)\,(0.0185))$,

$F_c = (3.8122 \times 10^8)\,(1.3119 \times 10^{-2})^4$,

$F_c = 1.1294 \times 10^1$ cm³/atm-s.

$F_m = (3.81 \times 10^3)\, D^3\, (T / M)^{.5} / (a\, P_a)$ (cm³/atm-s), (Eq. B4)

$F_m = (3.81 \times 10^3)\,(1.3119 \times 10^{-2})^3\,(298 / 29)^{.5} / ((0.3531)\,(0.505))$,

$F_m = (6.8501 \times 10^4)\,(13119 \times 10^{-2})^3$,

$F_m = 1.5468 \times 10^{-1}$ cm³/atm-s.

$L_u = (F_c + F_m)\,(P_u - P_d)\,(P_a / P_u)$ (cm³/s), (Eq. B5)

$L_u = (1.1294 \times 10^1 + 1.5468 \times 10^{-1})$(cm³/atm-s) (1.0 − 0.01)(atm) (0.505 / 1.0),

$L_u = (11.4485 \times 10^1)$(cm³/atm-s) (0.49995)(atm),

$L_u = 5.7237$ cm³/s.

The HAC reference air leakage rate as defined in ANSI N14.5-1997, Sect. B.3, is the upstream leakage in air.

$L_{RA,Air} = 5.7237$ ref-cm³/s. for HEU oxide content

The same equations can be used to calculate an allowable leakage rate using helium for leak testing.

M	=	4 g/g-mole,	Molecular weight of helium
μ	=	0.0198 cP.	Viscosity of helium at temperature

$F_c = (2.49 \times 10^6)\, D^4 / (a\, \mu)$ (cm³/atm-s), (Eq. B3)

$F_c = (2.49 \times 10^6)\,(1.3119 \times 10^{-2})^4 / ((0.3531)\,(0.0198))$,

$F_c = (3.5619 \times 10^8)\,(1.3119 \times 10^{-2})^4$,

$F_c = 1.0552 \times 10^1$ cm³/atm-s.

$F_m = (3.81 \times 10^3)\, D^3\, (T / M)^{0.5} / (a\, P_a)$ (cm³/atm-s), (Eq. B4)

$F_m = (3.81 \times 10^3)\,(1.3119 \times 10^{-2})^3\,(298 / 4)^{.5} / ((0.3531)\,(0.505))$,

$F_m = (1.8444 \times 10^5)\,(1.3119 \times 10^{-2})^3$,

$F_m = 4.1650 \times 10^{-1}$ cm³/atm-s.

$L_u = (F_c + F_m)\,(P_u - P_d)\,(P_a / P_u)$ (cm³/s), (Eq. B5)

$L_u = (1.0052 \times 10^1 + 4.1650 \times 10^{-1})$(cm³/atm-s) (1.0 − 0.01)(atm) (0.505 / 1.0),

$L_u = (1.0969 \times 10^1)$(cm³/atm-s) (0.49995)(atm),

$L_u = 5.4839$ cm³/s.

The allowable leakage rate using helium for leak testing for HAC is:

$L_{RA,\,He}$ = 5.4839 cm^3/s. HAC helium test value

Table 1. Regulatory leakage criteria for 35.2 kg of HEU

	Years from fabrication	NCT		HAC	
		$L_{RN\text{-}air}$ (ref-cm^3/s)	$L_{RN\text{-}He}$ (cm^3/s)	$L_{RA\text{-}air}$ (ref-cm^3/s)	$L_{RA\text{-}He}$ (cm^3/s)
Fast absorption	0	3.8917E-03	4.1797E-03	6.7586E+00	6.4689E+00
	5	3.8735E-03	4.1608E-03	6.7271E+00	6.4390E+00
	10	3.8709E-03	4.1581E-03	6.7225E+00	6.4346E+00
	20	3.8693E-03	4.1564E-03	6.7197E+00	6.4319E+00
	30	3.8680E-03	4.1550E-03	6.7174E+00	6.4297E+00
	40	3.8665E-03	4.1536E-03	6.7149E+00	6.4274E+00
	50	3.8650E-03	4.1519E-03	6.7122E+00	6.4248E+00
	60	3.8632E-03	4.1501E-03	6.7092E+00	6.4219E+00
	70	3.8614E-03	4.1482E-03	6.7061E+00	6.4189E+00
Medium absorption	0	3.7276E-03	4.0088E-03	6.4734E+00	6.1975E+00
	5	3.7110E-03	3.9915E-03	6.4445E+00	6.1701E+00
	10	3.7086E-03	3.9890E-03	6.4404E+00	6.1661E+00
	20	3.7072E-03	3.9875E-03	6.4379E+00	6.1637E+00
	30	3.7061E-03	3.9864E-03	6.4359E+00	6.1619E+00
	40	3.7048E-03	3.9851E-03	6.4337E+00	6.1598E+00
	50	3.7034E-03	3.9836E-03	6.4313E+00	6.1575E+00
	60	3.7019E-03	3.9820E-03	6.4286E+00	6.1549E+00
	70	3.7003E-03	3.9803E-03	6.4258E+00	6.1523E+00
Slow absorption	0	3.3124E-03	3.5754E-03	5.7514E+00	5.5103E+00
	5	3.2999E-03	3.5623E-03	5.7297E+00	5.4896E+00
	10	3.2985E-03	3.5609E-03	5.7273E+00	5.4873E+00
	20	3.2983E-03	3.5607E-03	5.7270E+00	5.4870E+00
	30	3.2983E-03	3.5607E-03	5.7269E+00	5.4870E+00
	40	3.2981E-03	3.5605E-03	5.7266E+00	5.4866E+00
	50	3.2977E-03	3.5600E-03	5.7259E+00	5.4860E+00
	60	3.2972E-03	3.5595E-03	5.7250E+00	5.4851E+00
	70	3.2965E-03	3.5587E-03	5.7237E+00	5.4839E+00

SECTION 4 REFERENCES

10 CFR 71, *Packaging and Transportation of Radioactive Material,* Jan. 1, 2007.

49 CFR 173, *Shippers—General Requirements for Shipments and Packagings,* Oct. 1, 2007.

ASME Boiler and Pressure Vessel Code, An American National Standard, Sect. II, *Materials*, Part C, Specifications for Welding, Rods, Electrodes, and Filler Metals, American Society of Mechanical Engineers, New York, 2001 ed. with 2002 and 2003 addenda.

ASME Boiler and Pressure Vessel Code, An American National Standard, Rules for Construction of Nuclear Facility Components, Sect. III, Div. 1, Subsection NB, American Society of Mechanical Engineers, New York, 2001 ed. with 2002 and 2003 addenda.

ASME Boiler and Pressure Vessel Code, An American National Standard, Welding and Brazing Qualifications, Sect. IX, American Society of Mechanical Engineers, New York, 2001 ed. with 2002 and 2003 addenda.

ANSI N14.5-1997, *Radioactive Materials—Leakage Tests on Packages for Shipment*, American Natl. Standards Institute, Feb. 5, 1998.

Curren, W. D. and R. D. Bond, *Leakage of Radioactive Powders from Containers*, Proceedings of the Sixth International Symposium on Packaging and Transportation of Radioactive Material: PATRAM '80, West Berlin, F.R.G., November 10–14, 1980, pp. 463–471.

Kocher, D. C., *Radioactive Decay Data Tables, A Handbook of Decay Data for Application to Radiation Dosimetry and Radiological Assessments*, DOE/TIC-11026, U.S. DOE, Office of Scientific and Technical Information, 1981.

NUREG-1609, *Standard Review Plan for Transportation Packages for Radioactive Material*, U.S. NRC, Mar. 31, 1999.

Parker O-ring Handbook, Catalog ORD 5700/US, Parker Hannifin Corp., Lexington, Ky., 2001.

R.D. McCarty Cryogenics Division, National Bureau of Standards Technical Note 631 , Thermophysical Properties of Helium-43 from 2 to 1500 K with Pressures to 100 Atmospheres, pg 49, Nov. 1972.

Regulatory Guide 7.4, *Leakage Tests on Packages for Shipment of Radioactive Materials*, U.S. NRC, June 1975.

Robert C. Weast, Ph. D., Handbook of Chemistry and Physics, 55th ed., pg. F-13, 1974.

SCALE 4.1—System Module to Calculate Fuel Depletion, Actinide Transmutation Fission Product Buildup and Decay, and Associated Radiation Source Terms, Vol. 2, Sect. F, NUREG/CR-2000, C. V. Parks, ed., Radiation Shielding Information Center, Oak Ridge Natl. Lab., Dec. 1984.

TR 96/12/20, Issue A, *SAFKEG 2863B Tests for Verification of O-ring Performance*, Los Alamos Natl. Lab., Albuquerque, N.M., December 1996.

5. SHIELDING EVALUATION

This section describes the shielding evaluation performed for the shipment of up to 36 kg of highly enriched uranium (HEU) in the ES-3100 shipping package. The objective of this evaluation is to demonstrate, for both Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC), compliance of this package with the performance requirements specified in Title 10 Code of Federal Regulations (CFR) 71 and 49 CFR 173.

5.1 DESCRIPTION OF SHIELDING DESIGN

5.1.1 Design Features

The ES-3100 package for NCT consists of stainless-steel convenience cans loaded with HEU material, spacer assemblies to support and position the convenience cans, the ES-3100 containment vessel, and an insulation-filled drum as shown in Appendix 1.4.1. None of the package materials are specifically designed for shielding gamma rays (photons), the primary contributor to external package dose rates. For HAC, it is assumed that the containment vessel and content remain intact, but all exterior packaging materials are removed. The geometry of the shielding analysis model is a conservative, cylindrical representation of the package. Two sets of contents have been investigated for the shielding analysis. These are 36 kg of HEU metal and 24 kg of HEU oxide powder. The analyses of these models have been performed such that the results and conclusions cover other proposed contents, in an equivalent or conservative manner. For the source calculation, the uranium is enriched at 92% ^{235}U and is assumed to contain 0.6% ^{233}U, 2.5% ^{237}Np, 40 ppb (parts per billion) ^{232}U, and 40 ppm (parts per million) plutonium.

5.1.2 Summary Table of Maximum Radiation Levels

The results of these shielding model evaluations, summarized in Tables 5.1 and 5.2, are the calculated dose rates at the locations indicated. The uncertainty associated with the results is one standard deviation of the mean (expressed as a percentage of the mean) of the Monte Carlo calculated results. As shown from the dose rates in the tables and the discussion in Sect. 5.3, the shielding evaluation demonstrates that the package meets all dose-rate limits for both NCT and HAC for all proposed contents.

5.2 SOURCE SPECIFICATION

The source for the calculated dose rates is from the decay and fission of the radioactive isotope contents. The isotopic content used in the shielding calculations is shown in Table 5.3. This composition was chosen to represent all proposed package contents. The primary contribution to the dose rates in Tables 5.1 and 5.2 is from decay of the ^{232}U isotope (this applies to any mix of the other uranium isotopes in Table 5.3). The photon and neutron source spectra are given in Tables 5.4 and 5.5. In Table 5.4, the group 6 photons are from the decay chain of ^{232}U. Group 18 photons were omitted from the source for the dose rate calculation due to negligible contribution from photons at these energies. Oxygen was not included in the uranium oxide source calculation, but the neutron source includes (α, n) neutrons from the UO_2 default option in the ORIGEN-S code (NUREG/CR-0200, rev. 6). The uranium metal source is assumed to be the same as that for the oxides, per unit HEU mass. Spontaneous fission neutrons are included in Table 5.5 spectrum. Induced fission neutrons and secondary photons are included in the dose rate calculations. Details of the source specifications and use in the dose rate calculations are given in the Sect. 5 appendices.

Table 5.1. Calculated external dose rates for the ES-3100 package with 36 kg of HEU metal contents (mrem/h)[a]

	Drum surface			One meter from surface[b]		
	Side	Top	Bottom	Side	Top	Bottom
			NCT			
Photon	92.264 ± 0.5%	38.746 ± 4.2%	77.410 ± 3.8%	6.441 ± 0.2%	1.419 ± 1.0%	1.450 ± 0.9%
Neutron[c]	1.767 ± 3.4%	0.144 ± 3.3%	3.690 ± 3.1%	0.062 ± 3.0%	0.028 ± 3.3%	0.084 ± 3.2%
Total	94.031 ± 0.5%	38.890 ± 4.2%	81.100 ± 3.8%	6.503 ± 0.2%[d]	1.447 ± 1.0%	1.534 ± 0.9%
Limit[e]	200	200	200	10	10	10
			HAC			
Photon	NA[f]	NA	NA	11.183 ± 0.2%	1.644 ± 0.9%	1.895 ± 0.9%
Neutron[c]	NA	NA	NA	0.029 ± 3.0%	0.007 ± 3.3%	0.018 ± 3.3%
Total	NA	NA	NA	11.212 ± 0.2%	1.651 ± 0.9%	1.913 ± 0.9%
Limit[g]	NA	NA	NA	1000	1000	1000

[a] MORSE-CGA Monte Carlo code results (ORNL-6174).
[b] Drum for NCT and containment vessel for HAC.
[c] Includes secondary photon dose rate (<1% of neutron dose rates).
[d] The radiation shielding transport index is 6.6.
[e] 10 CFR 71.47 and 49 CFR 173.441.
[f] Not applicable.
[g] 10 CFR 71.51.

Table 5.2. Calculated external dose rates for the ES-3100 package with 24 kg of HEU oxide contents (mrem/h)[a]

	Drum surface			One meter from surface[b]		
	Side	Top	Bottom	Side	Top	Bottom
				NCT		
Photon	82.201 ± 0.4%	14.315 ± 2.1%	62.208 ± 1.4%	4.975 ± 0.3%	1.249 ± 0.6%	1.519 ± 0.6%
Neutron[c]	1.035 ± 1.2%	0.084 ± 2.7%	2.181 ± 3.0%	0.038 ± 1.3%	0.016 ± 1.6%	0.047 ± 1.6%
Total	83.236 ± 0.4%	14.399 ± 2.1%	64.389 ± 1.4%	5.013 ± 0.3%[d]	1.265 ± 0.6%	1.566 ± 0.6%
Limit[e]	200	200	200	10	10	10
				HAC		
Photon	NA[f]	NA	NA	8.363 ± 0.4%	1.471 ± 1.2%	1.686 ± 1.2%
Neutron[c]	NA	NA	NA	0.027 ± 0.7%	0.010 ± 0.8%	0.018 ± 0.8%
Total	NA	NA	NA	8.390 ± 0.4%	1.481 ± 1.2%	1.704 ± 1.2%
Limit[g]	NA	NA	NA	1000	1000	1000

[a] MORSE-CGA Monte Carlo code results (ORNL-6174).
[b] Drum for NCT and containment vessel for HAC.
[c] Includes secondary photon dose rate (<1% of neutron dose rates).
[d] The radiation shielding transport index is 5.1.
[e] 10 CFR 71.47 and 49 CFR 173.441.
[f] Not applicable.
[g] 10 CFR 71.51.

5.3 DOSE RATE ANALYSIS MODELS

The photon and neutron sources from the radioactive content as described in the previous section, and listed in Tables 5.4 and 5.5, are input into the MORSE Monte Carlo radiation code to calculate the ES-3100 external package dose rates given in Tables 5.1 and 5.2. The packaging component drawings are shown in Appendix 1.4.1. A cylindrical model of this packaging and proposed HEU material content is shown in Fig. 5.1 and Table 5.6. The dose rate detector locations relative to the package model exterior are listed in Table 5.7. The materials and densities used in the calculational model are given in Table 5.8. These models are described in more detail in the input data and other information given in the Sect. 5 appendices. Kaolite is an insulation material, and Cat 277-4 is a criticality control material. Additional information on these two materials can be found in Sects. 1, 2, and 6.

Two proposed package contents have been analyzed: (1) 36 kg of HEU metal and (2) 24 kg of HEU oxide. The analyses of these models have been performed in a manner that covers other proposed contents, not analyzed, in an equivalent or conservative manner. Due to the simple source and packaging geometry, there are no radiation streaming paths from the source toward the package exterior. Each of the two models is analyzed for both photon and neutron sources, and for both NCT and HAC.

In addition to the analyses for the dose rates shown in Tables 5.1 and 5.2, many preliminary and auxiliary calculations were made for each model to ensure that all proposed content loadings in the ES-3100 vessel were covered. These extra analyses are necessary since only a content mass limit is specified for each shipment, and various geometric configurations must be investigated to determine if the maximum external package dose rates have been found. In some cases it is not possible, or practical, to find an exact maximum dose rate geometric configuration due to the statistical uncertainty of the calculation method, variations in model radius vs. height, variable densities, etc. The dose rate values shown in Tables 5.1 and 5.2 are all at or near possible maximum values for the package specifications given in Sect. 1. All package models investigated, including the preliminary and auxiliary models, are conservative.

Table 5.3. Radioisotope specification for all ES-3100 package analysis source calculations with HEU content and other nuclides per HEU unit weight

Isotope	wt %
^{232}U	0.000004
^{233}U	0.600000
^{234}U	2.000000
^{235}U	92.000000
^{236}U	1.000000
^{238}U	4.399996
^{237}Np	2.500000
Pu [a]	0.004000

[a] Nominal weapons-grade at 40 ppm plutonium by weight—see Appendix 5.5.1.

Table 5.4. Photon source for one gram of HEU for all contents [a]

Group number	Energy range (MeV)	Source (photons/s)
1	10.00–8.00	4.409×10^{-6}
2	8.00–6.50	2.552×10^{-5}
3	6.50–5.00	1.654×10^{-4}
4	5.00–4.00	5.106×10^{-4}
5	4.00–3.00	2.182×10^{-3}
6	3.00–2.50	$1.011 \times 10^{+4}$
7	2.50–2.00	1.011×10^{-1}
8	2.00–1.66	$9.761 \times 10^{+1}$
9	1.66–1.33	$1.101 \times 10^{+3}$
10	1.33–1.00	$3.426 \times 10^{+2}$
11	1.00–0.80	$1.595 \times 10^{+3}$
12	0.80–0.60	$4.620 \times 10^{+3}$
13	0.60–0.40	$2.347 \times 10^{+4}$
14	0.40–0.30	$1.977 \times 10^{+5}$
15	0.30–0.20	$4.550 \times 10^{+4}$
16	0.20–0.10	$1.150 \times 10^{+5}$
17	0.10–0.05	$3.534 \times 10^{+5}$
18 [b]	0.05–0.01	-
Total		$7.530 \times 10^{+5}$

[a] ORIGEN-S code results at 10½ years decay.
[b] Omitted in the dose rate calculations.

Table 5.5. Neutron source for one gram of HEU for all contents [a]

Group number	Energy range (MeV)	Source in uranium metal and oxide (neutrons/s)
1	$2.00 \times 10^{+1} - 6.43 \times 10^{+0}$	6.614×10^{-5}
2	$6.43 \times 10^{+0} - 3.00 \times 10^{+0}$	2.250×10^{-2}
3	$3.00 \times 10^{+0} - 1.85 \times 10^{+0}$	6.246×10^{-2}
4	$1.85 \times 10^{+0} - 1.40 \times 10^{+0}$	1.702×10^{-2}
5	$1.40 \times 10^{+0} - 9.00 \times 10^{-1}$	9.946×10^{-3}
6	$9.00 \times 10^{-1} - 4.00 \times 10^{-1}$	3.397×10^{-3}
7	$4.00 \times 10^{-1} - 1.00 \times 10^{-1}$	5.582×10^{-4}
8 [b]	$1.00 \times 10^{-1} - 1.70 \times 10^{-2}$	0
Total		1.160×10^{-1}

[a] ORIGEN-S results at 15 years decay.
[b] Source is zero for Groups 9–27.

Table 5.6. Geometric data for the shielding analysis models of the ES-3100 shipping package as shown in Fig. 5.1 for NCT. Each item is modeled as a cylinder or cylindrical shell. The content volume not identified (HEU metal photon shell model interior) is modeled as void. All materials interior to the containment vessel except the HEU material content are omitted. Each of the four content models is placed in the packaging to create four separate calculational models. All dimensions are in centimeters. For HAC, all material external to the containment vessel is omitted.

	Material	Outer radius	Base reference height[a]	Height[b] (h)	Side wall thickness	Top thickness	Bottom thickness
Content							
HEU metal shell photon model	HEU	6.35	11.195	76.2	0.6639	-	-
HEU metal neutron model	HEU	6.35	11.195	15.1003	-	-	-
Oxide photon model	UO_2	6.35	11.195	63.5	-	-	-
Oxide neutron model	UO_2	6.35	11.195	17.2864	-	-	-
Packaging (identical for all models)							
Inner vessel gap	void	6.4008	11.195	76.2	0.0508	-	-
ES-3100 containment vessel[c]	SS304[e]	6.6548	10.56	77.47	0.254	0.635	0.635
Outer vessel gap	void	7.9248	10.56	77.47	1.27	-	-
Inner Cat 277-4 liner	SS304	8.0748	10.56	77.47	0.15	-	-
Criticality control material	Cat 277-4	10.9196	10.56	77.47	2.8448	-	-
Outer Cat 277-4 liner	SS304	11.0696	10.26	77.92	0.15	0.15	0.3
Upper insulation	Kaolite[d]	23.0275	88.18	19.385	-	-	-
Base and radial insulation	Kaolite	23.0275	0.26	87.92	11.9579	-	10
Drum	SS304	23.1775	0	107.715	0.15	0.15	0.26

[a] Measured from the lower drum base.
[b] Vessel interior content (and gap) height varies with density to preserve content mass.
[c] Upper flange and lid detail omitted.
[d] Modeled as void.
[e] 304 stainless steel.

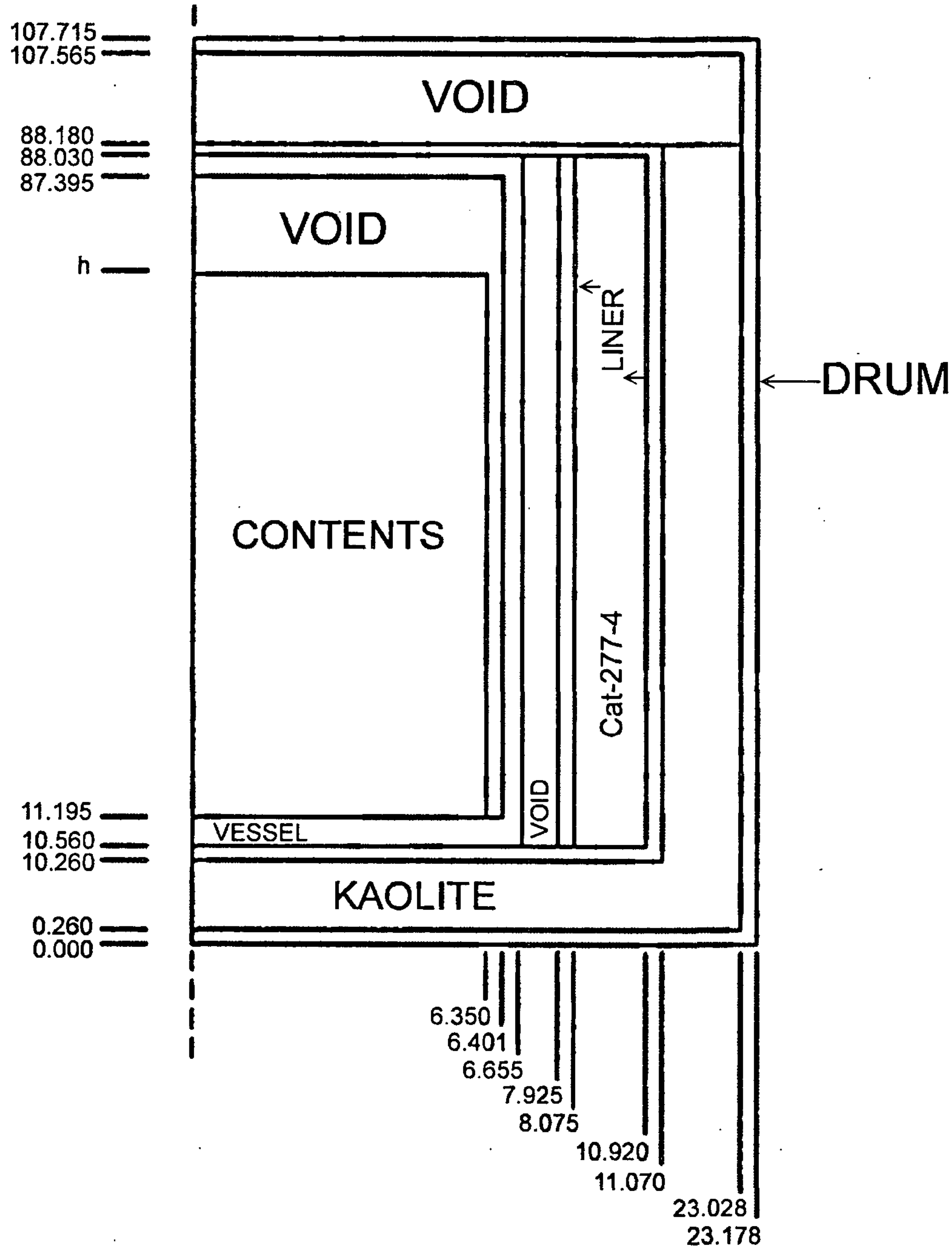


Fig. 5.1. Cylindrical calculational model of the ES-3100 shipping package for NCT. See Tables 5.6–5.8 for data on the contents, materials, and detector locations (not to scale; all dimensions are in centimeters). For HAC, all material external to the containment vessel is omitted.

Table 5.7. Detector locations relative to the drum for NCT and to the containment vessel for HAC

	Radius (cm)	H[a] (cm)
NCT detectors		
Side surface[b]	24.1775	49.2950
Top surface	0.0000	108.7150
Bottom surface	0.0000	–1.0000
Side 1 meter[b]	123.1775	49.2950
Top 1 meter	0.0000	207.7150
Bottom 1 meter	0.0000	–100.0000
HAC detectors		
Side 1 meter[b]	106.6548	49.2950
Top 1 meter	0.0000	188.0300
Bottom 1 meter	0.0000	–89.4400

[a] Height above drum base.
[b] Surface side location is at the axial mid-point of the content; values shown are for the first content in Table 5.6.

5.3.1 Packaging Model Conservative Features

Several conservative features applying to all content models are included in the packaging model. All material dimensions, thicknesses, and densities for the packaging are modeled at or less than specified or nominal values. All non-cylindrical detail has been omitted from the shielding models (see packaging drawings in Appendix 1.4.1). All minor items, such as the silicone rubber pads supporting the ES-3100 containment vessel, have been omitted. It was determined from preliminary analysis that the maximum axial external dose rates would occur relative to the package lower surface. All contents were modeled to contact the containment vessel lower surface, and the upper containment vessel geometry, containment vessel lid, and upper packaging, insulation, lids, covers, etc., were conservatively simplified and/or modeled as void (see Fig. 5.1).

All models of the vessel interior contain HEU material content only. All convenience cans, spacers, content wrappings, covers, supports, etc., are omitted. All contents are modeled as cylinders or cylindrical shells with the base resting on the lower vessel surface, with a maximum specified radius of 6.35 cm (the maximum allowable convenience can diameter is 5 in.), and with a height, inner radius (if any), and density adjusted for each specific model to preserve the content mass. In this conservative manner, it is possible to cover all specified combinations of can loadings and can and spacer placements with a minimum number of calculations. The geometric configuration of some of the cases investigated represents a situation where the maximum possible mass loading for a convenience can would be exceeded if the cans had been included in the model. However, the calculated dose rates from the models used will always exceed those from a model where the geometry is expanded so the can mass loadings are not exceeded and the internal vessel hardware is included. All the general, conservative items relative to the ES-3100 package shielding model are applied to each individual content model.

Table 5.8. Shielding model material specifications for the ES-3100 package with HEU content. Only the two principal uranium isotopes are used in the dose rate calculations. The uranium oxide density varies with the geometric model volume to preserve the 24-kg mass.

Material	Density (g/cm^3)	Constitute	Weight percent	Atomic density (atoms/barn-cm)
Uranium metal	18.82	^{235}U	95.00	4.582×10^{-2}
		^{238}U	5.00	2.381×10^{-3}
UO_2	10.960 [a]	O	11.98	4.941×10^{-2}
		^{235}U	83.62	2.349×10^{-2}
		^{238}U	4.40	1.221×10^{-3}
Stainless steel	7.92	Cr	19.00	1.743×10^{-2}
		Ni	9.50	7.721×10^{-3}
		Fe	69.50	5.936×10^{-2}
		Mn	2.00	1.736×10^{-3}
Cat 277-4	1.682	H	4.62	4.642×10^{-2}
		^{10}B	0.79	8.002×10^{-4}
		^{11}B	3.44	3.168×10^{-3}
		C	1.51	1.274×10^{-3}
		O	60.00	3.798×10^{-2}
		Mg	0.38	1.584×10^{-4}
		Al	21.16	7.944×10^{-3}
		Si	1.32	4.760×10^{-4}
		S	0.15	4.739×10^{-5}
		Na	0.13	5.728×10^{-5}
		Ca	6.18	1.562×10^{-3}
		Fe	0.32	5.804×10^{-5}
Kaolite	0.321	O	40.43	4.888×10^{-3}
		Na	1.45	1.219×10^{-4}
		Mg	7.86	6.251×10^{-4}
		Al	5.58	3.998×10^{-4}
		Si	16.12	1.110×10^{-3}
		Ca	23.40	1.129×10^{-3}
		Fe	5.16	1.786×10^{-4}

[a] Theoretical density— a density of 2.984 g/cm^3 is used in the Table 5.6 geometric model for photons.

5.3.2 Photon model for 36-kg HEU metal content

The HEU metal content in the ES-3100 may be many small, irregularly shaped pieces, each placed at some arbitrary orientation inside the convenience cans. Several conservative, regular geometry models have been devised to cover these and other proposed loadings. Some of these models violate, in a conservative manner, the volume capacity of a convenience can. The cans and spacers are omitted from the models.

The photon model for the dose rates shown in Table 5.1 is that given for the first content item in Table 5.6 (the HEU metal shell). Here, the content is a cylindrical shell the same height as the containment vessel model (76.2 cm). The inner radius of the shell, 5.6861 cm, was used to preserve the 36 kg HEU metal at 18.82 g/cm^3 for an outer radius of 6.35 cm. This case corresponds to a three-can loading configuration with a maximum of 12 kg in each 25.4-cm (10-in.) tall can.

Several 36-kg HEU metal cylindrical shell photon dose rate models were evaluated at lesser heights than the containment vessel inside height (in each case the shell wall thickness was adjusted to preserve the mass). The 1-m side photon dose rate in Table 5.1 is the calculated value for the entire analysis that gave the largest overall fraction of the regulatory limit, 65% of the 10-mrem/h limit. There was some increase in the side surface (not 1-m) photon dose at lesser shell heights. At ~50 cm content height, the maximum side surface dose rate for the shell model was calculated to be 98 mrem/h, 49% of the regulatory limit. This geometry would approximate a 12-kg loading in each of the 8.75-in. cans. The maximum calculated axial surface dose rate of 101 mrem/h was at the bottom drum surface for a shell height of 25 cm and inner radius of 4.0 cm, a gross conservative violation of an actual can loading.

The cylindrical shell representation of the HEU metal content is a convenient, conservative model for multiple possible loading configurations inside the vessel. Some other, more complicated, geometric radial models were also analyzed. The radial geometric cross-sections of these models are shown schematically in Fig. 5.2:

a. The cylindrical shell as described above.

b. A vertical flat plate across the center of the vessel. The maximum calculated photon package surface dose rate for a plate of dimensions 1.9766 cm × 12.7 cm × 76.2 cm was 63 mrem/h.

c. A cylindrical hemi-shell. The maximum calculated package surface photon dose rate was 69 mrem/h for an outer radius of 6.35 cm, an inner radius of 4.0436 cm, and a height of 50.8 cm.

d. A single solid rod placed against the side wall of the vessel. The maximum calculated package surface dose rate for a rod of radius 3.0966 cm and a height of 63.5 cm was 48 mrem/h.

e. A cylindrical segment (the shape a liquid would assume in a horizontal vessel). The maximum calculated package surface dose rate was calculated to be 71 mrem/h for a radius of 6.35 cm, an inner flat surface 11.495 cm across, and a height of 63.5 cm.

The variations in dose rates described here are due to the variations in source-detector point geometry and the self-absorption of photons in the source material. It may be possible to devise other geometric configurations that could produce slightly higher dose rates. However, it can be said with some certainty that the calculated photon dose rates for 36-kg HEU metal content defined in Table 5.3 should never exceed 110 mrem/h on the package surface, 7.6 mrem/h at one meter, and 16 mrem/h for HAC.

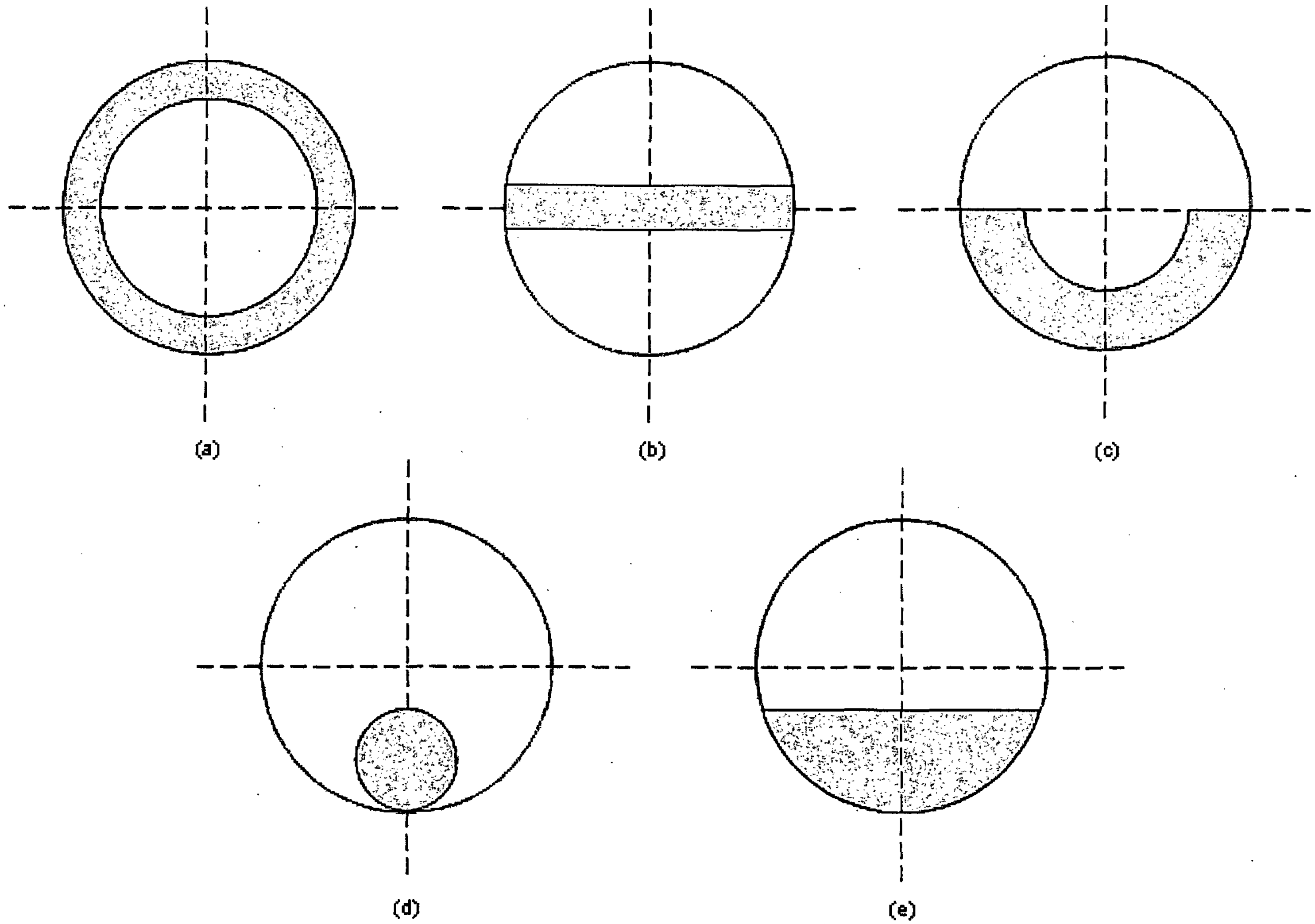


Fig. 5.2. ES-3100 HEU metal content radial (top view) geometric models. The 36-kg mass is conserved for each case. Detector locations were adjusted both vertically and radially from those given in Table 5.7 in the search for maximum external package dose rates for each case.

5.3.3 Neutron model for 36-kg HEU metal content

In the model for the neutron dose rates for the 36-kg HEU contents, it was assumed that all the uranium was a solid cylinder at the bottom of the vessel. The neutron results in Table 5.1 are from this configuration. This is a gross, conservative violation of a convenience can capacity, and no further investigation was made due to the low neutron dose rates obtained from this conservative model relative to the photon results.

5.3.4 Photon model for 24-kg HEU oxide content

A package photon dose rate calculation model was created to apply to 24 kg HEU oxide for all three common uranium oxides—UO_2, UO_3, and U_3O_8. The theoretical densities for these oxides are 10.96 g/cm^3, 7.29 g/cm^3, and 8.30 g/cm^3, respectively. In the package loading models, the convenience can locations are completely filled with oxide powder, and the actual densities, always less than theoretical values, will be the oxide mass divided by the model can volume.

The dose rate model excludes the convenience cans and spacers and it is represented by a solid cylinder of oxide resting on the vessel interior base. The cylinder radius is 6.35 cm, and the height is variable. The density is adjusted for each calculation to preserve the 24-kg oxide mass. The HEU mass is 21.126 kg for UO_2, the largest uranium mass of the three oxides for 24 kg of HEU oxide. In the dose rate calculations, the partial density of oxygen was set to zero, so that the analysis conservatively applies to all three oxides without regard to variations in the oxygen density. The photon dose rates in Table 5.2 are for a cylinder height of 63.5 cm. The 1-m dose rates are slightly higher for 76.2 cm height, and the surface values are slightly higher for a 50.8 cm height. The bottom dose rates approach the side values in Table 5.2 when the cylinder is compressed to the UO_2 theoretical density at the vessel bottom. From the various analyses, it can be stated that no calculated HEU oxide photon dose rates should exceed 90 mrem/h on the package surface, 6 mrem/h at one meter, and 12 mrem/h for HAC.

5.3.5 Neutron model for 24-kg HEU oxide content

The neutron model for the calculated values shown in Table 5.2 was for a solid cylinder of UO_2 at the containment vessel bottom. The density was 10.96 g/cm^3 and the height was 17.286 cm.

5.4 SHIELDING EVALUATION

Several computer programs were used in the shielding evaluation. These included the ORIGEN-S computer code, the CSASN analysis module, the ICE-S computer code, and the MORSE-CGA computer code. ORIGEN-S, CSASN, and ICE-S are parts of the SCALE code. MORSE-CGA is a stand-alone code. Brief descriptions of the programs are presented below.

1. ORIGEN-S is a general depletion and decay code. Given an initial isotopic distribution, materials are decayed to provide time-dependent, energy-grouped photon and neutron sources.

2. CSASN is a sequence of codes for performing resonance processing of neutron cross sections using the SCALE modules BONAMI-S and NITAWL-S.

3. ICE-S is a SCALE module used to format a cross-section library for MORSE-CGA.

4. MORSE-CGA is a general-purpose Monte Carlo code that treats multidimensional neutron, photon, and coupled problems in either a forward or an adjoint mode. Sources and detectors may be defined through user-supplied subroutines. Model definition is facilitated by combinatorial array geometry. Dose rates are calculated by the SAMBO analysis module.

Dose rates were calculated at the detector locations in Table 5.7 by converting the calculated photon and neutron fluxes using the American National Standards Institute (ANSI) conversion factors (ANSI/ANS-6.1.1). These factors for the energy groups used in the shielding evaluation are shown in Tables 5.9 and 5.10.

The results of these shielding model evaluations, summarized in Tables 5.1 and 5.2, are the calculated dose rates at the locations indicated. As shown from the dose rates in the tables and the discussion in Sect. 5.3, the shielding evaluation demonstrates that the package meets all dose-rate limits for both NCT and HAC. The shielding analyses were done in a conservative manner applicable to all proposed ES-3100 package contents listed in Sect. 1.2.3.

Table 5.9. ANSI standard photon flux-to-dose-rate conversion factors

Group number	Energy (MeV)	Factor (mrem/h)/(photons/s/cm^2)
1	10.00–8.00	8.771×10^{-3}
2	8.00–6.50	7.478×10^{-3}
3	6.50–5.00	6.374×10^{-3}
4	5.00–4.00	5.413×10^{-3}
5	4.00–3.00	4.622×10^{-3}
6	3.00–2.50	3.959×10^{-3}
7	2.50–2.00	3.468×10^{-3}
8	2.00–1.66	3.019×10^{-3}
9	1.66–1.33	2.627×10^{-3}
10	1.33–1.00	2.205×10^{-3}
11	1.00–0.80	1.832×10^{-3}
12	0.80–0.60	1.522×10^{-3}
13	0.60–0.40	1.172×10^{-3}
14	0.40–0.30	8.759×10^{-4}
15	0.30–0.20	6.306×10^{-4}
16	0.20–0.10	3.833×10^{-4}
17	0.10–0.05	2.669×10^{-4}
18	0.05–0.01	9.347×10^{-4}

Table 5.10. ANSI standard neutron flux-to-dose-rate conversion factors

Group number	Energy (MeV)	Factor (mrem/h)/(neutrons/s/cm^2)
1	$2.00 \times 10^{+1} - 6.43 \times 10^{+0}$	1.4916×10^{-1}
2	$6.43 \times 10^{+0} - 3.00 \times 10^{+0}$	1.4464×10^{-1}
3	$3.00 \times 10^{+0} - 1.85 \times 10^{+0}$	1.2701×10^{-1}
4	$1.85 \times 10^{+0} - 1.40 \times 10^{+0}$	1.2811×10^{-1}
5	$1.40 \times 10^{+0} - 9.00 \times 10^{-1}$	1.2977×10^{-1}
6	$9.00 \times 10^{-1} - 4.00 \times 10^{-1}$	1.0281×10^{-1}
7	$4.00 \times 10^{-1} - 1.00 \times 10^{-1}$	5.1183×10^{-2}
8	$1.00 \times 10^{-1} - 1.70 \times 10^{-2}$	1.2319×10^{-2}
9	$1.70 \times 10^{-2} - 3.00 \times 10^{-3}$	3.8365×10^{-3}
10	$3.00 \times 10^{-3} - 5.50 \times 10^{-4}$	3.7247×10^{-3}
11	$5.50 \times 10^{-4} - 1.00 \times 10^{-4}$	4.0150×10^{-3}
12	$1.00 \times 10^{-4} - 3.00 \times 10^{-5}$	4.2926×10^{-3}
13	$3.00 \times 10^{-5} - 1.00 \times 10^{-5}$	4.4744×10^{-3}
14	$1.00 \times 10^{-5} - 3.05 \times 10^{-6}$	4.5676×10^{-3}
15	$3.05 \times 10^{-6} - 1.77 \times 10^{-6}$	4.5581×10^{-3}
16	$1.77 \times 10^{-6} - 1.30 \times 10^{-6}$	4.5185×10^{-3}
17	$1.30 \times 10^{-6} - 1.13 \times 10^{-6}$	4.4879×10^{-3}
18	$1.13 \times 10^{-6} - 1.00 \times 10^{-6}$	4.4665×10^{-3}
19	$1.00 \times 10^{-6} - 8.00 \times 10^{-7}$	4.4345×10^{-3}
20	$8.00 \times 10^{-7} - 4.00 \times 10^{-7}$	4.3271×10^{-3}
21	$4.00 \times 10^{-7} - 3.25 \times 10^{-7}$	4.1975×10^{-3}
22	$3.25 \times 10^{-7} - 2.25 \times 10^{-7}$	4.0976×10^{-3}
23	$2.25 \times 10^{-7} - 1.00 \times 10^{-7}$	3.8390×10^{-3}
24	$1.00 \times 10^{-7} - 5.00 \times 10^{-8}$	3.6748×10^{-3}
25	$5.00 \times 10^{-8} - 3.00 \times 10^{-8}$	3.6748×10^{-3}
26	$3.00 \times 10^{-8} - 1.00 \times 10^{-8}$	3.6748×10^{-3}
27	$1.00 \times 10^{-8} - 1.00 \times 10^{-11}$	3.6748×10^{-3}

5.5 APPENDICES

Appendix	Descriptions
5.5.1	ORIGEN INPUT DATA FROM TABLE 5.3
5.5.2	CSASN AND ICE INPUT FROM TABLE 5.8
5.5.3	MORSE ROUTINES AND INPUT DATA

Appendix 5.5.1

ORIGEN INPUT DATA FROM TABLE 5.3

This file is for generation of the photon spectra in Table 5.4 and the metal and oxide neutron data in Table 5.5. The concentration for each isotope is given based on 1 g of HEU.

```
#origen
0$$ 6 13 a8 26 e
1$$ 1 t
master photon
3$$ 21 0 1 -88 6 a16 2 a33 18
4** a4 1-70 t
35$$ 0 t
56$$ 0 10 a13 13 5 3 a17 2 e
57** 0 e t
es3100 uranium metal and oxides
one gram HEU
60** 8 9 10 10.5 11 12 15 20 40 50
65$$ a7 1 a25 1 a28 1 a31 1 a49 1 e
73$$ 922320 922340 922350 922360 922380
     922330 942380 942390 932370
     942400 942410 942420 952410
74** 0.4000-07 .020000 0.92000 .010000 0.04399996
     6.00-03 8.00-09 3.7032-05 0.025
     2.60-06 2.24-07 1.60-08 1.20-07
75$$ 2 2 2 2 2 2 2 2 2 2 2 2 2
81$$ 2 0 26 1 e
82$$ 2 2 2 2 2 2 2 2 0 0
83** 10+6 8+6 6.5+6 5+6 4+6 3+6 2.5+6 2+6 1.66+6 1.33+6 1+6 .8+6
     .6+6 .4+6 .3+6 .2+6 .1+6 .05+6 .01+6 t
               8
               9
              10
              10.5
              11
              12
              15
              20
56$$ f0 t
end
```

Appendix 5.5.2

CSASN AND ICE INPUT FROM TABLE 5.8

This file is the input data for the generation and preparation (format) of the cross-section data used in the MORSE code. The data generated from this input file can be used for all MORSE media input cases. Oxygen is omitted from the oxide models, and HEU is assumed to be all U-235 in the neutron models in the MORSE data. The atomic densities for all dose rate calculations in Appendix 5.5.3 cases are given, or indicated, in Table 5.8.

```
=csasn
es3100: morse-cg cross section library
27n-18couple
multiregion
  arbmu02      10.9600  3  0  0  0   8016  11.9800
                                     92235  83.6200
                                     92238   4.4000   1 1.0000 293.0 end
  arbmsst       7.9200  4  0  0  0  26304  69.5000
                                     24304  19.0000
                                     28304   9.5000
                                     25055   2.0000   2 1.0000 293.0 end
  arbmcat       1.6820 12  0  0  0   1001   4.6200
                                      5010   0.7900
                                      5011   3.4400
                                      6012   1.5100
                                      8016  60.0000
                                     12000   0.3800
                                     13027  21.1600
                                     14000   1.3200
                                     16000   0.1500
                                     11023   0.1300
                                     20000   6.1800
                                     26304   0.3200  3 1.0000 293.0 end
  arbmkoa       0.3210  7  0  0  0   8016  40.4300
                                     11023   1.4500
                                     12000   7.8600
                                     13027   5.5800
                                     14000  16.1200
                                     20000  23.4000
                                     26304   5.1600   4 1.0000 293.0 end
end comp
spherical end
  1  1.0     noextermod
end zone
end
=ice
es31 ice : morse cross section library
-1$$ a3 2200 e
1$$ 26 26 0 10 0 0 1 1t
2$$ 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24
       25 26
3$$ 1008016 1092235 1092238
    2024304 2025055 2026304 2028304
    3001001 3005010 3005011 3006012 3008016 3012000 3013027
    3014000 3016000 3011023 3020000 3026304
    4008016 4011023 4012000 4013027 4014000 4020000 4026304
4** 26r1.0
5$$ 26r10
7$$ 3 16 60 0 0 0 e 2t
9$$ 258I1 260
10$$ 1 1452 27 3t
end
```

Appendix 5.5.3

MORSE ROUTINES AND INPUT DATA

Included here are the routines and input files used for the dose rate calculations for the results in Tables 5.1 and 5.2. For each case, there are two routines, INSCOR and SOURCE, followed by the NCT and the HAC input. The normalization is given in INSCOR, which is the appropriate Total value in Tables 5.4 or 5.5 times the content mass. The spatial and energy distributions for the photon and neutron sources from Tables 5.4 and 5.5 are in the data statements in the SOURCE routine. Induced fission and secondary photons are included in the MORSE calculation. It is assumed that the energy group distribution of induced fissions is the same as for the spectra in Table 5.5. The photon source is biased, and weight corrected, so that half of all primary photons are selected in group 6 from Table 5.4.

For each case, the input data for NCT and HAC follow the two routines. For the HAC cases, all material external to the containment vessel is set to void and the detectors (now three) are set one meter from the external vessel surface. The 24-kg oxide neutron case, similar to the 36-kg metal neutron case, is not shown.

```
      subroutine inscor
       common /pdet/   nd,     nne,     ne,      nt,      na,      nresp,
      1                nex,    nexnd,   nend,    ndnr,    ntnr,    ntne,
      2                nane,   ntndnr,  ntnend,  nanend,  locrsp,  locxd,
      3                locib,  locco,   loct,    locud,   locsd,   locqe,
      4                locqt,  locqte,  locqae,  lmax,    efirst,  egtop
       common bc(1)
       do 100 i = 1,nd
         bc(locxd + 5*nd + i) = 2.711e+10
 100   continue
       return
       end
       subroutine source( ig,u,v,w,x,y,z,
      1                   wate,med,ag,isour,itstr,ngpqt3,ddf,
      2                   isbias,nmtg )
       dimension spect(1,18)
       data xst,yst,zst /0.000,0.0,11.195/
       data cyl_ht/76.200/
c40ppb u232
       data (spect(1,k),k=1,18) /
      1  4.409e-6,2.552e-5,1.654e-4,5.106e-4, 2.182e-3,7.429e+5,
      2  1.011e-1,9.761e+1,1.101e+3,
      3  3.426e+2,1.595e+3, 4.620e+03,
      4  2.347e+04,1.977e+5,4.550e+4,1.150e+5,3.534e+5,0.000e-00/
       data icall / 1 /
       if (icall) 10,10,5
 5     icall = 0
       r02 = 5.6861**2
       r12 = 6.3500**2
        i=1
        sum=0.0
         do 92 j = 1,nmtg
 92        sum = sum + spect(i,j)
         spect(i,1) = spect(i,1) / sum
         do 93 j = 2,nmtg
 93        spect(i,j) = spect(i,j-1) + spect(i,j) / sum
 10    continue
       r1 = fltrnf( 0 )
       call azirn( sinang,cosang )
       rad = sqrt( r1 * r12 + ( 1.0 - r1 ) * r02 )
       x = rad * sinang + xst
       y = rad * cosang + yst
       z = zst + fltrnf( 0 ) * cyl_ht
       i=1
       rn = fltrnf( 0 )
       do 94 j = 1,nmtg
         if ( rn .le. spect(i,j) ) goto 95
 94    continue
       j = nmtg
 95    ig = j
c40ppb u232
       if(ig.ne.6)wate=wate*1.97315
       if(ig.eq.6)wate=wate*0.026853
       return
       end
```

```
es3100 package nct 36kg u metal 40 ppb u232
 $$ 500 850 500     1     0    17    18    18     0     0  99.    4     0
 $$  0     0     0     0  **  1.0 1.0000e-5 1.0000e+4       1.0 2.2000e+5
 **     0.0        0.0        0.0        0.0        0.0        0.0        0.0
 **
 10.000e+6 8.0000e+6 6.5000e+6 5.0000e+6 4.0000e+6 3.0000e+6 2.5000e+6
 2.0000e+6 1.6600e+6 1.3300e+6 1.0000e+6 0.8000e+6 0.6000e+6 0.4000e+6
 0.3000e+6 0.2000e+6 0.1000e+6 0.0500e+6
 b1a303a31a34
 $$   1     1     0     0     0     1    18
 $$   1     1    10     1     1     1 **   0.1     .001          .010  5.000e-01
 $$  11     1    14     1     1     1 **   1.0     .010          .100  5.000e-01
 $$  15     1    18     1     1     1 **   10.     .100          1.00  5.000e-01
 $$  -1     9r0
   0     0     0     0
   0     0                heu in es-3100 36kg cylinder model
   0     0     1     0
  rcc    1    0.000    0.0    11.1950    0.00    0.0  76.2000    5.6861
  rcc    2    0.000    0.0    11.1950    0.00    0.0  76.2000    6.3500
  rcc    3    0.000    0.0    11.1950    0.00    0.0  76.2000    6.4008
  rcc    4    0.000    0.0    10.5600    0.00    0.0  77.4700    6.6548
  rcc    5    0.000    0.0    10.5600    0.00    0.0  77.4700    7.9248
  rcc    6    0.000    0.0    10.5600    0.00    0.0  77.4700    8.0748
  rcc    7    0.000    0.0    10.5600    0.00    0.0  77.4700   10.9196
  rcc    8    0.000    0.0    10.2600    0.00    0.0  77.9200   11.0696
  rcc    9    0.000    0.0     0.2600    0.00    0.0  87.9200   23.0275
  rcc   10    0.000    0.0     0.2600    0.00    0.0 107.3050   23.0275
  rcc   11    0.000    0.0     0.0000    0.00    0.0 107.7150   23.1775
  sph   12       0.0      0.0      0.0        1000.
  sph   13       0.0      0.0      0.0        2000.
  end
  vid            1
  con            2     -1
  vid            3     -2
  sst            4     -3
  vid            5     -4
  sst            6     -5
  bor            7     -6
  sst            8     -7
  kao            9     -8
  vid           10     -9
  sst           11    -10
  vid           12    -11
  vix           13    -12
  end
   1     1     1     1     1     1     1     1     1     1     1     1     1
  13z
 1000     1 1000     2 1000     2     3     2     4 1000     2 1000     0
   0
27N18P LIBRARY (P5)
 $$ 0     0    18    18    45    60    16     4    26   26     6     3     1     3
   0     0     0     0     0     0     0    20     0     0     0
   1     2     3     4     5     6    11    12    13    14    15    16    21    22
  23    24    25    26    31    32    33    34    35    36    41    42    43    44
  45    46    51    52    53    54    55    56    61    62    63    64    65    66
  71    72    73    74    75    76    81    82    83    84    85    86    91    92
  93    94    95    96   101   102   103   104   105   106   111   112   113   114
 115   116   121   122   123   124   125   126   131   132   133   134   135   136
 141   142   143   144   145   146   151   152   153   154   155   156   161   162
 163   164   165   166   171   172   173   174   175   176   181   182   183   184
 185   186   191   192   193   194   195   196   201   202   203   204   205   206
 211   212   213   214   215   216   221   222   223   224   225   226   231   232
 233   234   235   236   241   242   243   244   245   246   251   252   253   254
```

```
 255  256
 $$ 1   12 ** 1.0000e-9
 $$ 1    2 ** 4.5820e-2
 $$ 1   -3 ** 2.3810e-3
 $$ 2    5 ** 1.7360e-3
 $$ 2    6 ** 5.9360e-2
 $$ 2    7 ** 7.7210e-3
 $$ 2   -4 ** 1.7430e-2
 $$ 3    8 ** 4.6420e-2
 $$ 3    9 ** 8.0020e-4
 $$ 3   10 ** 3.1680e-3
 $$ 3   11 ** 1.2740e-3
 $$ 3   12 ** 3.7980e-2
 $$ 3   13 ** 1.5840e-4
 $$ 3   14 ** 7.9440e-3
 $$ 3   15 ** 4.7600e-4
 $$ 3   16 ** 4.7390e-5
 $$ 3   17 ** 5.7280e-5
 $$ 3   18 ** 1.5620e-3
 $$ 3  -19 ** 5.8040e-5
 $$ 4   20 ** 4.8880e-3
 $$ 4   21 ** 1.2190e-4
 $$ 4   22 ** 6.2510e-4
 $$ 4   23 ** 3.9980e-4
 $$ 4   24 ** 1.1100e-3
 $$ 4   25 ** 1.1290e-3
 $$ 4  -26 ** 1.7860e-4
SAMBO ANALYSIS INPUT DATA
 $$   6    0    0    0    0    1    1    2
 **
   24.1775 0.0 49.295
   0.0 0.0  -1.00
   0.0 0.0  108.715
   123.1775 0.0 49.295
   0.0 0.0 -100.00
   0.0 0.0  207.7150
UNCOLLIDED AND TOTAL PHOTON DOSE RATES
ANSI STANDARD GAMMA DOSE RATES
 **
 8.7716e-3 7.4785e-3 6.3748e-3 5.4136e-3 4.6221e-3 3.9596e-3 3.4686e-3
 3.0192e-3 2.6276e-3 2.2051e-3 1.8326e-3 1.5228e-3 1.1725e-3 8.7594e-4
 6.3061e-4 3.8338e-4 2.6693e-4 9.3472e-4
```

```
es3100 package hac 36kg u metal 40 ppb u232
 $$ 500 850 500    1    0   17   18   18    0    0  99.    4    0
 $$  0    0    0    0  **  1.0 1.0000e-5 1.0000e+4       1.0 2.2000e+5
 **    0.0        0.0        0.0        0.0        0.0        0.0        0.0
 **
 10.000e+6 8.0000e+6 6.5000e+6 5.0000e+6 4.0000e+6 3.0000e+6 2.5000e+6
 2.0000e+6 1.6600e+6 1.3300e+6 1.0000e+6 0.8000e+6 0.6000e+6 0.4000e+6
 0.3000e+6 0.2000e+6 0.1000e+6 0.0500e+6
 b1a303a31a34
 $$   1    1    0    0    0    1   18
 $$   1    1   10    1    1    1 **   0.1     .001        .010  5.000e-01
 $$  11    1   14    1    1    1 **   1.0     .010        .100  5.000e-01
 $$  15    1   18    1    1    1 **   10.     .100        1.00  5.000e-01
 $$  -1     9r0
    0    0    0    0
    0    0                  heu in es-3100 36kg cylinder model
    0    0    1    0
  rcc    1    0.000    0.0    11.1950    0.00    0.0  76.2000    5.6861
  rcc    2    0.000    0.0    11.1950    0.00    0.0  76.2000    6.3500
  rcc    3    0.000    0.0    11.1950    0.00    0.0  76.2000    6.4008
  rcc    4    0.000    0.0    10.5600    0.00    0.0  77.4700    6.6548
  rcc    5    0.000    0.0    10.5600    0.00    0.0  77.4700    7.9248
  rcc    6    0.000    0.0    10.5600    0.00    0.0  77.4700    8.0748
  rcc    7    0.000    0.0    10.5600    0.00    0.0  77.4700   10.9196
  rcc    8    0.000    0.0    10.2600    0.00    0.0  77.9200   11.0696
  rcc    9    0.000    0.0     0.2600    0.00    0.0  87.9200   23.0275
  rcc   10    0.000    0.0     0.2600    0.00    0.0 107.3050   23.0275
  rcc   11    0.000    0.0     0.0000    0.00    0.0 107.7150   23.1775
  sph   12      0.0      0.0      0.0        1000.
  sph   13      0.0      0.0      0.0        2000.
  end
  vid          1
  con          2     -1
  vid          3     -2
  sst          4     -3
  vid          5     -4
  sst          6     -5
  bor          7     -6
  sst          8     -7
  kao          9     -8
  vid         10     -9
  sst         11    -10
  vid         12    -11
  vix         13    -12
  end
    1    1    1    1    1    1    1    1    1    1    1    1    1
  13z
 1000    1 1000    2 1000 1000 1000 1000 1000 1000 1000 1000     0
    0
27N18P LIBRARY (P5)
 $$ 0    0   18   18   45   60   16    4   26   26    6    3    1    3
    0    0    0    0    0    0    0   20    0    0    0
    1    2    3    4    5    6   11   12   13   14   15   16   21   22
   23   24   25   26   31   32   33   34   35   36   41   42   43   44
   45   46   51   52   53   54   55   56   61   62   63   64   65   66
   71   72   73   74   75   76   81   82   83   84   85   86   91   92
   93   94   95   96  101  102  103  104  105  106  111  112  113  114
  115  116  121  122  123  124  125  126  131  132  133  134  135  136
  141  142  143  144  145  146  151  152  153  154  155  156  161  162
  163  164  165  166  171  172  173  174  175  176  181  182  183  184
  185  186  191  192  193  194  195  196  201  202  203  204  205  206
  211  212  213  214  215  216  221  222  223  224  225  226  231  232
  233  234  235  236  241  242  243  244  245  246  251  252  253  254
```

```
 255  256
 $$ 1   12 ** 1.0000e-9
 $$ 1    2 ** 4.5820e-2
 $$ 1   -3 ** 2.3810e-3
 $$ 2    5 ** 1.7360e-3
 $$ 2    6 ** 5.9360e-2
 $$ 2    7 ** 7.7210e-3
 $$ 2   -4 ** 1.7430e-2
 $$ 3    8 ** 4.6420e-2
 $$ 3    9 ** 8.0020e-4
 $$ 3   10 ** 3.1680e-3
 $$ 3   11 ** 1.2740e-3
 $$ 3   12 ** 3.7980e-2
 $$ 3   13 ** 1.5840e-4
 $$ 3   14 ** 7.9440e-3
 $$ 3   15 ** 4.7600e-4
 $$ 3   16 ** 4.7390e-5
 $$ 3   17 ** 5.7280e-5
 $$ 3   18 ** 1.5620e-3
 $$ 3  -19 ** 5.8040e-5
 $$ 4   20 ** 4.8880e-3
 $$ 4   21 ** 1.2190e-4
 $$ 4   22 ** 6.2510e-4
 $$ 4   23 ** 3.9980e-4
 $$ 4   24 ** 1.1100e-3
 $$ 4   25 ** 1.1290e-3
 $$ 4  -26 ** 1.7860e-4
SAMBO ANALYSIS INPUT DATA
 $$    3    0    0    0    0    1    1    2
 **
   106.6548 0.0 49.295
   0.0 0.0 -89.440
   0.0 0.0 188.03
UNCOLLIDED AND TOTAL PHOTON DOSE RATES
ANSI STANDARD GAMMA DOSE RATES
 **
 8.7716e-3 7.4785e-3 6.3748e-3 5.4136e-3 4.6221e-3 3.9596e-3 3.4686e-3
 3.0192e-3 2.6276e-3 2.2051e-3 1.8326e-3 1.5228e-3 1.1725e-3 8.7594e-4
 6.3061e-4 3.8338e-4 2.6693e-4 9.3472e-4
```

```
      subroutine inscor
      common /pdet/   nd,     nne,    ne,     nt,     na,     nresp,
     1                nex,    nexnd,  nend,   ndnr,   ntnr,   ntne,
     2                nane,   ntndnr, ntnend, nanend, locrsp, locxd,
     3                locib,  locco,  loct,   locud,  locsd,  locqe,
     4                locqt,  locqte, locqae, lmax,   efirst, egtop
      common bc(1)
      do 100 i = 1,nd
        bc(locxd + 5*nd + i) = 4.1800e+03
100   continue
      return
      end
      subroutine source( ig,u,v,w,x,y,z,
     1                   wate,med,ag,isour,itstr,ngpqt3,ddf,
     2                   isbias,nmtg )
     dimension spect(1,8)
      data xst,yst,zst /0.000,0.0, 11.1950/
      data cyl_ht/15.1003/
      data (spect(1,k),k=1,8) /
     1 6.614e-5,2.250e-2,6.246e-2,1.702e-2,9.946e-3,3.397e-3,
     2 5.582e-4,0.000e+0/
      data icall / 1 /
      if (icall) 10,10,5
5     icall = 0
      r11 = 6.3500
       i=1
       sum=0.0
        do 92 j = 1,8
92        sum = sum + spect(i,j)
        spect(i,1) = spect(i,1) / sum
        do 93 j = 2,8
93        spect(i,j) = spect(i,j-1) + spect(i,j) / sum
10    continue
      r1 = fltrnf( 0 )
      call azirn( sinang,cosang )
      rad = sqrt( r1)* r11
      x = rad * sinang + xst
      y = rad * cosang + yst
      z = zst + fltrnf( 0 ) * cyl_ht
      i=1
      rn = fltrnf( 0 )
      do 94 j = 1,8
        if ( rn .le. spect(i,j) ) goto 95
94    continue
      j = 8
95    ig = j
      return
      end
```

```
es3100 package nct 36kg u metal neutrons
 $$ 200  850  200   1  27   18   27   45    0    0  60.    4    0
 $$   0    0    0    0  **    1.0 1.0000e-5 1.0000e+4       1.0 2.2000e+5
 $$     0.0       0.0       0.0       0.0       0.0       0.0       0.0
 **
 2.0000e+7 6.4300e+6 3.0000e+6 1.8500e+6 1.4000e+6 9.0000e+5 4.0000e+5
 1.0000e+5 1.7000e+4 3.0000e+3 5.5000e+2 1.0000e+2 3.0000e+1 1.0000e+1
 3.0500e+0 1.7700e+0 1.3000e+0 1.1300e+0 1.0000e+0 8.0000e-1 4.0000e-1
 3.2500e-1 2.2500e-1 9.9999e-2 5.0000e-2 3.0000e-2 1.0000e-2 1.0000e+7
 8.0000e+6 6.5000e+6 5.0000e+6 4.0000e+6 3.0000e+6 2.5000e+6 2.0000e+6
 1.6600e+6 1.3300e+6 1.0000e+6 0.8000e+6 0.6000e+6 0.4000e+6 0.3000e+6
 0.2000e+6 0.1000e+6 0.0500e+6
 b3a343b203a1
 $$   1    1    0    0    0    1   45
 $$   1    1    9    1    1    1 **   50.    .050        .500  5.000e-01
 $$  10    1   27    1    1    1 **   50.    1.00        10.0  5.000e-01
 $$  28    1   45    1    1    1 **   20.    .500        2.00  5.000e-01
 $$  -1    9r0
 $$ 0    1    0    0
 ** 1.0
 **
 6.6140e-5 2.250e-2 6.246e-2 1.702e-2 9.946e-3 3.397e-3 5.582e-4
 0.0000e-0 19z
 **   27z
 **   27z
 **   27z
 ** 27r.01
    0    0            heu in es-3100 36kg cylinder model
    0    0    1    0
  rcc    1    0.000   0.0   11.1950   0.00    0.0  15.1003   0.0001
  rcc    2    0.000   0.0   11.1950   0.00    0.0  15.1003   6.3500
  rcc    3    0.000   0.0   11.1950   0.00    0.0  76.2000   6.4008
  rcc    4    0.000   0.0   10.5600   0.00    0.0  77.4700   6.6548
  rcc    5    0.000   0.0   10.5600   0.00    0.0  77.4700   7.9248
  rcc    6    0.000   0.0   10.5600   0.00    0.0  77.4700   8.0748
  rcc    7    0.000   0.0   10.5600   0.00    0.0  77.4700  10.9196
  rcc    8    0.000   0.0   10.2600   0.00    0.0  77.9200  11.0696
  rcc    9    0.000   0.0    0.2600   0.00    0.0  87.9200  23.0275
  rcc   10    0.000   0.0    0.2600   0.00    0.0 107.3050  23.0275
  rcc   11    0.000   0.0    0.0000   0.00    0.0 107.7150  23.1775
  sph   12      0.0      0.0      0.0       1000.
  sph   13      0.0      0.0      0.0       2000.
  end
  vid           1
  con           2      -1
  vid           3      -2
  sst           4      -3
  vid           5      -4
  sst           6      -5
  bor           7      -6
  sst           8      -7
  kao           9      -8
  vid          10      -9
  sst          11     -10
  vid          12     -11
  vix          13     -12
  end
    1    1    1    1    1    1    1    1    1    1    1    1    1
  13z
 1000    1 1000    2 1000    2    3    2    4 1000    2 1000    0
    0
27N18P LIBRARY (P5)
 $$ 27   27   18   18   45   60   16   4    26   26    6    3    1    3
```

```
     0     0     0     0     0     0     0    20     0     0     0
     1     2     3     4     5     6    11    12    13    14    15    16    21    22
    23    24    25    26    31    32    33    34    35    36    41    42    43    44
    45    46    51    52    53    54    55    56    61    62    63    64    65    66
    71    72    73    74    75    76    81    82    83    84    85    86    91    92
    93    94    95    96   101   102   103   104   105   106   111   112   113   114
   115   116   121   122   123   124   125   126   131   132   133   134   135   136
   141   142   143   144   145   146   151   152   153   154   155   156   161   162
   163   164   165   166   171   172   173   174   175   176   181   182   183   184
   185   186   191   192   193   194   195   196   201   202   203   204   205   206
   211   212   213   214   215   216   221   222   223   224   225   226   231   232
   233   234   235   236   241   242   243   244   245   246   251   252   253   254
   255   256
 $$ 1   12 ** 1.0000e-9
 $$ 1    2 ** 4.8240e-2
 $$ 1   -3 ** 1.2210e-9
 $$ 2    5 ** 1.7360e-3
 $$ 2    6 ** 5.9360e-2
 $$ 2    7 ** 7.7210e-3
 $$ 2   -4 ** 1.7430e-2
 $$ 3    8 ** 4.6420e-2
 $$ 3    9 ** 8.0020e-4
 $$ 3   10 ** 3.1680e-3
 $$ 3   11 ** 1.2740e-3
 $$ 3   12 ** 3.7980e-2
 $$ 3   13 ** 1.5840e-4
 $$ 3   14 ** 7.9440e-3
 $$ 3   15 ** 4.7600e-4
 $$ 3   16 ** 4.7390e-5
 $$ 3   17 ** 5.7280e-5
 $$ 3   18 ** 1.5620e-3
 $$ 3  -19 ** 5.8040e-5
 $$ 4   20 ** 4.8880e-3
 $$ 4   21 ** 1.2190e-4
 $$ 4   22 ** 6.2510e-4
 $$ 4   23 ** 3.9980e-4
 $$ 4   24 ** 1.1100e-3
 $$ 4   25 ** 1.1290e-3
 $$ 4  -26 ** 1.7860e-4
SAMBO ANALYSIS INPUT DATA
 $$    6    0    0    0    0    3    1    2
 **
   24.1775 0.0  18.745
   0.0 0.0 -1.000
   0.0 0.0  108.715
   123.1775 0.0  18.745
   0.0 0.0 -100.00
   0.0 0.0  207.7150
uncollided and total photon dose rates
ansi standard neutron dose rates :
 **
 1.4916e-1 1.4464e-1 1.2701e-1 1.2811e-1 1.2977e-1 1.0281e-1 5.1183e-2
 1.2319e-2 3.8365e-3 3.7247e-3 4.0150e-3 4.2926e-3 4.4744e-3 4.5676e-3
 4.5581e-3 4.5185e-3 4.4879e-3 4.4665e-3 4.4345e-3 4.3271e-3 4.1975e-3
 4.0976e-3 3.8390e-3 3.6748e-3 3.6748e-3 3.6748e-3 3.6748e-3
 18r0.0
ansi standard photon dose rates :
 **
 27r0.0
 8.7716e-3 7.4785e-3 6.3748e-3 5.4136e-3 4.6221e-3 3.9596e-3 3.4686e-3
 3.0192e-3 2.6276e-3 2.2051e-3 1.8326e-3 1.5228e-3 1.1725e-3 8.7594e-4
 6.3061e-4 3.8338e-4 2.6693e-4 9.3472e-4
ansi standard total dose rates :
```

```
**
1.4916e-1 1.4464e-1 1.2701e-1 1.2811e-1 1.2977e-1 1.0281e-1 5.1183e-2
1.2319e-2 3.8365e-3 3.7247e-3 4.0150e-3 4.2926e-3 4.4744e-3 4.5676e-3
4.5581e-3 4.5185e-3 4.4879e-3 4.4665e-3 4.4345e-3 4.3271e-3 4.1975e-3
4.0976e-3 3.8390e-3 3.6748e-3 3.6748e-3 3.6748e-3 3.6748e-3
8.7716e-3 7.4785e-3 6.3748e-3 5.4136e-3 4.6221e-3 3.9596e-3 3.4686e-3
3.0192e-3 2.6276e-3 2.2051e-3 1.8326e-3 1.5228e-3 1.1725e-3 8.7594e-4
6.3061e-4 3.8338e-4 2.6693e-4 9.3472e-4



es3100 package hac 36kg u metal neutrons
 $$ 200  850  200   1   27   18   27   45    0    0  60.    4    0
 $$   0    0    0    0  **    1.0 1.0000e-5 1.0000e+4       1.0 2.2000e+5
 $$     0.0        0.0        0.0        0.0        0.0        0.0        0.0
 **
2.0000e+7 6.4300e+6 3.0000e+6 1.8500e+6 1.4000e+6 9.0000e+5 4.0000e+5
1.0000e+5 1.7000e+4 3.0000e+3 5.5000e+2 1.0000e+2 3.0000e+1 1.0000e+1
3.0500e+0 1.7700e+0 1.3000e+0 1.1300e+0 1.0000e+0 8.0000e-1 4.0000e-1
3.2500e-1 2.2500e-1 9.9999e-2 5.0000e-2 3.0000e-2 1.0000e-2 1.0000e+7
8.0000e+6 6.5000e+6 5.0000e+6 4.0000e+6 3.0000e+6 2.5000e+6 2.0000e+6
1.6600e+6 1.3300e+6 1.0000e+6 0.8000e+6 0.6000e+6 0.4000e+6 0.3000e+6
0.2000e+6 0.1000e+6 0.0500e+6
b3a343b2O3a1
 $$   1    1    0    0    0    1   45
 $$   1    1    9    1    1    1 **    50.     .050        .500  5.000e-01
 $$  10    1   27    1    1    1 **    50.     1.00        10.0  5.000e-01
 $$  28    1   45    1    1    1 **    20.     .500        2.00  5.000e-01
 $$  -1     9r0
 $$ 0    1    0    0
 ** 1.0
 **
6.6140e-5 2.107e-2 5.841e-2 1.593e-2 9.334e-3 3.219e-3 5.304e-4
0.0000e-0 19z
 **   27z
 **   27z
 **   27z
 ** 27r.01
   0    0             heu in es-3100 36kg cylinder model
   0    0    1    0
  rcc    1    0.000   0.0   11.1950   0.00    0.0  15.1003    0.0001
  rcc    2    0.000   0.0   11.1950   0.00    0.0  15.1003    6.3500
  rcc    3    0.000   0.0   11.1950   0.00    0.0  76.2000    6.4008
  rcc    4    0.000   0.0   10.5600   0.00    0.0  77.4700    6.6548
  rcc    5    0.000   0.0   10.5600   0.00    0.0  77.4700    7.9248
  rcc    6    0.000   0.0   10.5600   0.00    0.0  77.4700    8.0748
  rcc    7    0.000   0.0   10.5600   0.00    0.0  77.4700   10.9196
  rcc    8    0.000   0.0   10.2600   0.00    0.0  77.9200   11.0696
  rcc    9    0.000   0.0    0.2600   0.00    0.0  87.9200   23.0275
  rcc   10    0.000   0.0    0.2600   0.00    0.0 107.3050   23.0275
  rcc   11    0.000   0.0    0.0000   0.00    0.0 107.7150   23.1775
  sph   12      0.0      0.0      0.0        1000.
  sph   13      0.0      0.0      0.0        2000.
  end
  vid          1
  con          2     -1
  vid          3     -2
  sst          4     -3
  vid          5     -4
  sst          6     -5
  bor          7     -6
  sst          8     -7
  kao          9     -8
```

```
  vid          10     -9
  sst          11    -10
  vid          12    -11
  vix          13    -12
  end
    1    1    1    1    1    1    1    1    1    1    1    1    1
  13z
 1000    1 1000    2 1000 1000 1000 1000 1000 1000 1000 1000      0
    0
27N18P LIBRARY (P5)
 $$ 27   27   18   18   45   60   16    4   26   26    6    3    1    3
    0    0    0    0    0    0    0   20    0    0    0
    1    2    3    4    5    6   11   12   13   14   15   16   21   22
   23   24   25   26   31   32   33   34   35   36   41   42   43   44
   45   46   51   52   53   54   55   56   61   62   63   64   65   66
   71   72   73   74   75   76   81   82   83   84   85   86   91   92
   93   94   95   96  101  102  103  104  105  106  111  112  113  114
  115  116  121  122  123  124  125  126  131  132  133  134  135  136
  141  142  143  144  145  146  151  152  153  154  155  156  161  162
  163  164  165  166  171  172  173  174  175  176  181  182  183  184
  185  186  191  192  193  194  195  196  201  202  203  204  205  206
  211  212  213  214  215  216  221  222  223  224  225  226  231  232
  233  234  235  236  241  242  243  244  245  246  251  252  253  254
  255  256
 $$ 1   12 ** 1.0000e-9
 $$ 1    2 ** 4.8240e-2
 $$ 1   -3 ** 1.2210e-9
 $$ 2    5 ** 1.7360e-3
 $$ 2    6 ** 5.9360e-2
 $$ 2    7 ** 7.7210e-3
 $$ 2   -4 ** 1.7430e-2
 $$ 3    8 ** 4.6420e-2
 $$ 3    9 ** 8.0020e-4
 $$ 3   10 ** 3.1680e-3
 $$ 3   11 ** 1.2740e-3
 $$ 3   12 ** 3.7980e-2
 $$ 3   13 ** 1.5840e-4
 $$ 3   14 ** 7.9440e-3
 $$ 3   15 ** 4.7600e-4
 $$ 3   16 ** 4.7390e-5
 $$ 3   17 ** 5.7280e-5
 $$ 3   18 ** 1.5620e-3
 $$ 3  -19 ** 5.8040e-5
 $$ 4   20 ** 4.8880e-3
 $$ 4   21 ** 1.2190e-4
 $$ 4   22 ** 6.2510e-4
 $$ 4   23 ** 3.9980e-4
 $$ 4   24 ** 1.1100e-3
 $$ 4   25 ** 1.1290e-3
 $$ 4  -26 ** 1.7860e-4
SAMBO ANALYSIS INPUT DATA
 $$   3   0    0    0    0    3    1    2
 **
   106.6548 0.0  18.745
   0.0 0.0 -89.44
   0.0 0.0 188.03
uncollided and total photon dose rates
ansi standard neutron dose rates :
 **
 1.4916e-1 1.4464e-1 1.2701e-1 1.2811e-1 1.2977e-1 1.0281e-1 5.1183e-2
 1.2319e-2 3.8365e-3 3.7247e-3 4.0150e-3 4.2926e-3 4.4744e-3 4.5676e-3
 4.5581e-3 4.5185e-3 4.4879e-3 4.4665e-3 4.4345e-3 4.3271e-3 4.1975e-3
 4.0976e-3 3.8390e-3 3.6748e-3 3.6748e-3 3.6748e-3 3.6748e-3
```

```
 18r0.0
ansi standard photon dose rates :
 **
 27r0.0
 8.7716e-3 7.4785e-3 6.3748e-3 5.4136e-3 4.6221e-3 3.9596e-3 3.4686e-3
 3.0192e-3 2.6276e-3 2.2051e-3 1.8326e-3 1.5228e-3 1.1725e-3 8.7594e-4
 6.3061e-4 3.8338e-4 2.6693e-4 9.3472e-4
ansi standard total dose rates :
 **
 1.4916e-1 1.4464e-1 1.2701e-1 1.2811e-1 1.2977e-1 1.0281e-1 5.1183e-2
 1.2319e-2 3.8365e-3 3.7247e-3 4.0150e-3 4.2926e-3 4.4744e-3 4.5676e-3
 4.5581e-3 4.5185e-3 4.4879e-3 4.4665e-3 4.4345e-3 4.3271e-3 4.1975e-3
 4.0976e-3 3.8390e-3 3.6748e-3 3.6748e-3 3.6748e-3 3.6748e-3
 8.7716e-3 7.4785e-3 6.3748e-3 5.4136e-3 4.6221e-3 3.9596e-3 3.4686e-3
 3.0192e-3 2.6276e-3 2.2051e-3 1.8326e-3 1.5228e-3 1.1725e-3 8.7594e-4
 6.3061e-4 3.8338e-4 2.6693e-4 9.3472e-4
```

```
      subroutine inscor
       common /pdet/   nd,      nne,     ne,      nt,      na,      nresp,
      1                nex,     nexnd,   nend,    ndnr,    ntnr,    ntne,
      2                nane,    ntndnr,  ntnend,  nanend,  locrsp,  locxd,
      3                locib,   locco,   loct,    locud,   locsd,   locqe,
      4                locqt,   locqte,  locqae,  lmax,    efirst,  egtop
       common bc(1)
       do 100 i = 1,nd
         bc(locxd + 5*nd + i) = 2.7110e+10*24./36.*0.88024
 100   continue
       return
       end
       subroutine source( ig,u,v,w,x,y,z,
      1                   wate,med,ag,isour,itstr,ngpqt3,ddf,
      2                   isbias,nmtg )
       dimension spect(1,18)
       data xst,yst,zst /0.000,0.0,11.195/
        data cyl_ht/63.500/
 c40ppb u232
       data (spect(1,k),k=1,18) /
      1  4.409e-6,2.552e-5,1.654e-4,5.106e-4, 2.182e-3,7.429e+5,
      2  1.011e-1,9.761e+1,1.101e+3,
      3  3.426e+2,1.595e+3, 4.620e+03,
      4  2.347e+04,1.977e+5,4.550e+4,1.150e+5,3.534e+5,0.000e-00/
       data icall / 1 /
       if (icall) 10,10,5
 5     icall = 0
 c     r02 = 5.6861**2
       r02 = 0.0001**2
       r12 = 6.3500**2
        i=1
        sum=0.0
         do 92 j = 1,nmtg
 92        sum = sum + spect(i,j)
         spect(i,1) = spect(i,1) / sum
         do 93 j = 2,nmtg
 93        spect(i,j) = spect(i,j-1) + spect(i,j) / sum
 10    continue
       r1 = fltrnf( 0 )
       call azirn( sinang,cosang )
       rad = sqrt( r1 * r12 + ( 1.0 - r1 ) * r02 )
       x = rad * sinang + xst
       y = rad * cosang + yst
       z = zst + fltrnf( 0 ) * cyl_ht
       i=1
       rn = fltrnf( 0 )
       do 94 j = 1,nmtg
         if ( rn .le. spect(i,j) ) goto 95
 94    continue
       j = nmtg
 95    ig = j
 c40ppb u232
       if(ig.ne.6)wate=wate*1.97315
       if(ig.eq.6)wate=wate*0.026853
       return
       end
```

```
es3100 package photon nct 24kg u oxide 40 ppb u232
$$ 500 850 500    1    0   17   18   18    0    0  99.    4    0
$$  0    0    0    0  **  1.0 1.0000e-5 1.0000e+4       1.0 2.2000e+5
**    0.0       0.0       0.0       0.0       0.0       0.0       0.0
**
10.000e+6 8.0000e+6 6.5000e+6 5.0000e+6 4.0000e+6 3.0000e+6 2.5000e+6
2.0000e+6 1.6600e+6 1.3300e+6 1.0000e+6 0.8000e+6 0.6000e+6 0.4000e+6
0.3000e+6 0.2000e+6 0.1000e+6 0.0500e+6
b1a303a31a34
$$   1    1    0    0    0    1   18
$$   1    1   10    1    1    1 **   0.1    .001        .010  5.000e-01
$$  11    1   14    1    1    1 **   1.0    .010        .100  5.000e-01
$$  15    1   18    1    1    1 **   10.    .100        1.00  5.000e-01
$$  -1    9r0
   0    0    0    0
   0    0                 oxide in es-3100 24kg cylinder model
   0    0    1    0
  rcc    1    0.000   0.0   11.1950   0.00   0.0  63.5000   0.0001
  rcc    2    0.000   0.0   11.1950   0.00   0.0  63.5000   6.3500
  rcc    3    0.000   0.0   11.1950   0.00   0.0  76.2000   6.4008
  rcc    4    0.000   0.0   10.5600   0.00   0.0  77.4700   6.6548
  rcc    5    0.000   0.0   10.5600   0.00   0.0  77.4700   7.9248
  rcc    6    0.000   0.0   10.5600   0.00   0.0  77.4700   8.0748
  rcc    7    0.000   0.0   10.5600   0.00   0.0  77.4700  10.9196
  rcc    8    0.000   0.0   10.2600   0.00   0.0  77.9200  11.0696
  rcc    9    0.000   0.0    0.2600   0.00   0.0  87.9200  23.0275
  rcc   10    0.000   0.0    0.2600   0.00   0.0 107.3050  23.0275
  rcc   11    0.000   0.0    0.0000   0.00   0.0 107.7150  23.1775
  sph   12      0.0      0.0      0.0       1000.
  sph   13      0.0      0.0      0.0       2000.
  end
  vid          1
  con          2     -1
  vid          3     -2
  sst          4     -3
  vid          5     -4
  sst          6     -5
  bor          7     -6
  sst          8     -7
  kao          9     -8
  vid         10     -9
  sst         11    -10
  vid         12    -11
  vix         13    -12
  end
   1    1    1    1    1    1    1    1    1    1    1    1    1
  13z
1000    1 1000    2 1000    2    3    2    4 1000    2 1000    0
   0
27N18P LIBRARY (P5)
$$ 0    0   18   18   45   60   16    4   26   26    6    3    1    3
   0    0    0    0    0    0    0   20    0    0    0
   1    2    3    4    5    6   11   12   13   14   15   16   21   22
  23   24   25   26   31   32   33   34   35   36   41   42   43   44
  45   46   51   52   53   54   55   56   61   62   63   64   65   66
  71   72   73   74   75   76   81   82   83   84   85   86   91   92
  93   94   95   96  101  102  103  104  105  106  111  112  113  114
 115  116  121  122  123  124  125  126  131  132  133  134  135  136
 141  142  143  144  145  146  151  152  153  154  155  156  161  162
 163  164  165  166  171  172  173  174  175  176  181  182  183  184
 185  186  191  192  193  194  195  196  201  202  203  204  205  206
 211  212  213  214  215  216  221  222  223  224  225  226  231  232
 233  234  235  236  241  242  243  244  245  246  251  252  253  254
```

```
  255  256
 $$ 1   12 ** 1.0000e-9
 $$ 1    2 ** 6.3940e-3
 $$ 1   -3 ** 3.3220e-4
 $$ 2    5 ** 1.7360e-3
 $$ 2    6 ** 5.9360e-2
 $$ 2    7 ** 7.7210e-3
 $$ 2   -4 ** 1.7430e-2
 $$ 3    8 ** 4.6420e-2
 $$ 3    9 ** 8.0020e-4
 $$ 3   10 ** 3.1680e-3
 $$ 3   11 ** 1.2740e-3
 $$ 3   12 ** 3.7980e-2
 $$ 3   13 ** 1.5840e-4
 $$ 3   14 ** 7.9440e-3
 $$ 3   15 ** 4.7600e-4
 $$ 3   16 ** 4.7390e-5
 $$ 3   17 ** 5.7280e-5
 $$ 3   18 ** 1.5620e-3
 $$ 3  -19 ** 5.8040e-5
 $$ 4   20 ** 4.8880e-3
 $$ 4   21 ** 1.2190e-4
 $$ 4   22 ** 6.2510e-4
 $$ 4   23 ** 3.9980e-4
 $$ 4   24 ** 1.1100e-3
 $$ 4   25 ** 1.1290e-3
 $$ 4  -26 ** 1.7860e-4
SAMBO ANALYSIS INPUT DATA
 $$    6    0    0    0    0    1    1    2
 **
   24.1775 0.0 42.945
   0.0 0.0  -1.00
   0.0 0.0  108.715
   123.1775 0.0 42.945
   0.0 0.0 -100.00
   0.0 0.0  207.715
UNCOLLIDED AND TOTAL PHOTON DOSE RATES
ANSI STANDARD GAMMA DOSE RATES
 **
 8.7716e-3 7.4785e-3 6.3748e-3 5.4136e-3 4.6221e-3 3.9596e-3 3.4686e-3
 3.0192e-3 2.6276e-3 2.2051e-3 1.8326e-3 1.5228e-3 1.1725e-3 8.7594e-4
 6.3061e-4 3.8338e-4 2.6693e-4 9.3472e-4
```

```
es3100 package photon hac 24kg u oxide 40 ppb u232
 $$ 500 850 500    1    0   17   18   18    0    0  99.    4    0
 $$  0    0    0    0  **  1.0 1.0000e-5 1.0000e+4        1.0 2.2000e+5
 **    0.0        0.0        0.0        0.0        0.0        0.0        0.0
 **
 10.000e+6 8.0000e+6 6.5000e+6 5.0000e+6 4.0000e+6 3.0000e+6 2.5000e+6
 2.0000e+6 1.6600e+6 1.3300e+6 1.0000e+6 0.8000e+6 0.6000e+6 0.4000e+6
 0.3000e+6 0.2000e+6 0.1000e+6 0.0500e+6
 b1a303a31a34
 $$   1    1    0    0    0    1   18
 $$   1    1   10    1    1    1 **   0.1     .001         .010  5.000e-01
 $$  11    1   14    1    1    1 **   1.0     .010         .100  5.000e-01
 $$  15    1   18    1    1    1 **   10.     .100         1.00  5.000e-01
 $$  -1    9r0
    0    0    0    0
    0    0                    oxide in es-3100 24kg cylinder model
    0    0    1    0
  rcc    1    0.000   0.0   11.1950   0.00    0.0  63.5000   0.0001
  rcc    2    0.000   0.0   11.1950   0.00    0.0  63.5000   6.3500
  rcc    3    0.000   0.0   11.1950   0.00    0.0  76.2000   6.4008
  rcc    4    0.000   0.0   10.5600   0.00    0.0  77.4700   6.6548
  rcc    5    0.000   0.0   10.5600   0.00    0.0  77.4700   7.9248
  rcc    6    0.000   0.0   10.5600   0.00    0.0  77.4700   8.0748
  rcc    7    0.000   0.0   10.5600   0.00    0.0  77.4700  10.9196
  rcc    8    0.000   0.0   10.2600   0.00    0.0  77.9200  11.0696
  rcc    9    0.000   0.0    0.2600   0.00    0.0  87.9200  23.0275
  rcc   10    0.000   0.0    0.2600   0.00    0.0 107.3050  23.0275
  rcc   11    0.000   0.0    0.0000   0.00    0.0 107.7150  23.1775
  sph   12      0.0      0.0      0.0        1000.
  sph   13      0.0      0.0      0.0        2000.
  end
  vid            1
  con            2     -1
  vid            3     -2
  sst            4     -3
  vid            5     -4
  sst            6     -5
  bor            7     -6
  sst            8     -7
  kao            9     -8
  vid           10     -9
  sst           11    -10
  vid           12    -11
  vix           13    -12
  end
    1    1    1    1    1    1    1    1    1    1    1    1    1
  13z
 1000    1 1000    2 1000 1000 1000 1000 1000 1000 1000 1000    0
    0
27N18P LIBRARY (P5)
 $$ 0    0   18   18   45   60   16    4   26   26    6    3    1    3
    0    0    0    0    0    0    0   20    0    0    0
    1    2    3    4    5    6   11   12   13   14   15   16   21   22
   23   24   25   26   31   32   33   34   35   36   41   42   43   44
   45   46   51   52   53   54   55   56   61   62   63   64   65   66
   71   72   73   74   75   76   81   82   83   84   85   86   91   92
   93   94   95   96  101  102  103  104  105  106  111  112  113  114
  115  116  121  122  123  124  125  126  131  132  133  134  135  136
```

```
 141  142  143  144  145  146  151  152  153  154  155  156  161  162
 163  164  165  166  171  172  173  174  175  176  181  182  183  184
 185  186  191  192  193  194  195  196  201  202  203  204  205  206
 211  212  213  214  215  216  221  222  223  224  225  226  231  232
 233  234  235  236  241  242  243  244  245  246  251  252  253  254
 255  256
$$ 1   12 ** 1.0000e-9
$$ 1    2 ** 6.3940e-3
$$ 1   -3 ** 3.3220e-4
$$ 2    5 ** 1.7360e-3
$$ 2    6 ** 5.9360e-2
$$ 2    7 ** 7.7210e-3
$$ 2   -4 ** 1.7430e-2
$$ 3    8 ** 4.6420e-2
$$ 3    9 ** 8.0020e-4
$$ 3   10 ** 3.1680e-3
$$ 3   11 ** 1.2740e-3
$$ 3   12 ** 3.7980e-2
$$ 3   13 ** 1.5840e-4
$$ 3   14 ** 7.9440e-3
$$ 3   15 ** 4.7600e-4
$$ 3   16 ** 4.7390e-5
$$ 3   17 ** 5.7280e-5
$$ 3   18 ** 1.5620e-3
$$ 3  -19 ** 5.8040e-5
$$ 4   20 ** 4.8880e-3
$$ 4   21 ** 1.2190e-4
$$ 4   22 ** 6.2510e-4
$$ 4   23 ** 3.9980e-4
$$ 4   24 ** 1.1100e-3
$$ 4   25 ** 1.1290e-3
$$ 4  -26 ** 1.7860e-4
SAMBO ANALYSIS INPUT DATA
$$    3    0    0    0    0    1    1    2
**
  106.6548 0.0 42.945
   0.0 0.0 -89.440
   0.0 0.0 188.03
UNCOLLIDED AND TOTAL PHOTON DOSE RATES
ANSI STANDARD GAMMA DOSE RATES
**
8.7716e-3 7.4785e-3 6.3748e-3 5.4136e-3 4.6221e-3 3.9596e-3 3.4686e-3
3.0192e-3 2.6276e-3 2.2051e-3 1.8326e-3 1.5228e-3 1.1725e-3 8.7594e-4
6.3061e-4 3.8338e-4 2.6693e-4 9.3472e-4
```

SECTION 5 REFERENCES

10 CFR 71, *Packaging and Transportation of Radioactive Material*, Jan. 1, 2007.

49 CFR 173, *Shippers—General Requirements for Shipments and Packagings*, Oct. 1, 2007.

ANSI/ANS-6.1.1, *Neutron and Gamma Ray Flux-to-Dose-Rate Factors,* American Natl. Standards Institute, American Nuclear Society, La Grange, Ill., 1977.

MORSE-CGA, *Monte Carlo Radiation Transport Code with Array Geometry Capability,* ORNL-6174, M. B. Emmett, Oak Ridge Natl. Lab., April 1985.

SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, C. V. Parks, ed., NUREG/CR-0200, rev. 6, ORNL/NUREG/CSD-2/R6, May 2000.

6. CRITICALITY EVALUATION

This section describes the criticality safety evaluation of the Y-12 National Security Complex (Y-12) Model ES-3100 package with highly enriched uranium (HEU) metal or alloys of aluminum or molybednum (specific information for alloys are addressed in Sects. 6.2.1, 6.2.4, 6.3.2, 6.4, 6.4.1, and 6.6.1.), or HEU oxide, or highly enriched uranyl nitrate crystals (UNX). HEU metal may be solid shapes (cylinders, bars, buttons, slugs, unirradiated TRIGA fuel) or broken metal pieces of unspecified geometric shapes. Distinguished from TRIGA contents, the research reactor related contents covered under this criticality evaluation are fuel elements or fuel components, broken metal, U-Al alloy, or oxides. Physical testing of Type-B fissile material packages for surface transportation conducted in accordance with the physical testing requirements of 10 CFR 71 is limited to the ES-3100 packaging and non-fissile dummy contents. Consequently, analytic methods are used to demonstrate compliance of the ES-3100 package with the applicable performance requirements in 10 CFR 71. The specific requirements investigated for compliance in this evaluation are contained in 10 CFR 71.55, "General Requirements of all Fissile Material Packages," and 10 CFR 71.59, "Standards for Arrays of Fissile Material Packages." Physical testing of Type-B fissile material packages transported by air is not conducted for the ES-3100. Analytic methods are also used to demonstrate compliance of the ES-3100 package with the requirements of 10 CFR 71.55(f). This is accomplished by demonstrating compliance of the package with the more stringent requirements of the International Atomic Energy Agency (IAEA) for situations where the conditions of the fissile material package following the tests cannot be demonstrated. (TS-G-1.1, Sect. 680.2)

6.1 DESCRIPTION OF THE CRITICALITY DESIGN

6.1.1 Design Features

The principal design feature of interest in the criticality evaluation of the Model ES-3100 package is the containment/outer drum system (Drawing No. M2E801580A031, Appendix 1.4.8). The ES-3100 package uses a single containment system (Drawing No. M2E801580A011, Appendix 1.4.8) to contain the HEU contents. The containment is a high-integrity, watertight, post-load leak-testable, stainless-steel vessel (Fig. 1.2). The outer drum system (Drawing M2E801580A001, Appendix 1.4.8) is a recessed, double-compartment body with a removable top for insertion and removal of the containment vessel. The body weldment liner outer cavity and the top plug weldment contain Kaolite 1600™ (Kaolite), a thermal insulation material that protects the containment vessel from thermal absorption, shock and impact. The body weldment liner inner cavity contains a neutron poison, "277-4," which serves as a strong neutron absorber. Other sections of this report (Sects. 2, 3, and 4) demonstrate the integrity of the ES-3100 package [i.e., that this single containment vessel remains intact and watertight under the Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC)].

Three 4.25-in.-diam × 10.0-in.-tall or six 4.25-in.-diam × 4.88-in.-tall convenience cans separated by can pads may be placed inside the 31-in.-tall cavity of the ES-3100 containment vessel. (Drawing M2E801580A035, Appendix 1.4.8) Other can arrangements fit inside the containment vessel, such as three 4.25-in.-diam × 8.75-in.-tall or five 4.25-in.-diam × 4.88-in.-tall convenience cans. Both of these can arrangements include can pads and 277-4 canned spacers with overall dimensions of 4.25-in. diam × 1.82-in. height. (Drawing M2E801580A026, Appendix 1.4.8) Nickel alloy cans are ~3-in. diam × 4.75-in.-tall. Three Teflon or three polyethylene bottles fit inside the containment vessel; however, 277-4 canned spacers are not used with these configurations.

A modified convenience can may be used to package the TRIGA and research reactor related contents that do not fit inside these containers. A 17.5 in. or 30 in.-tall convenience can is constructed from two 4.25-in.-diam × 8.75-in.-tall or three 10-in. cans brazed together. The research reactor fuel elements and fuel components that cannot be packaged in these convenience cans due to their overall dimensions will be bundled together and protected on each end with an open-ended convenience can having an outer diameter ≤5 in.

Credit is not taken in this criticality analysis for fissile material spacing provided by the presence of the convenience cans or bottles inside the containment vessel. These containers are not manufactured to the *ASME Boiler and Pressure Vessel Code*, Sect. III, Subsection NG, or better.

6.1.2 Summary of the Criticality Evaluation

Testing conducted in accordance with the physical testing requirements of 10 CFR 71 demonstrated that water leakage into the containment is not a credible event under the NCT and HAC. However, credit for the high-integrity, watertight containment is not taken in this criticality evaluation, as agreed upon during discussions held at public meetings at the U.S. Nuclear Regulatory Commission. (Docket 71-9315)

10 CFR 71.55(b) requires the evaluation of water leakage into the containment vessel or leakage of liquid contents out of the containment vessel and of other conditions that produce maximum reactivity in the single package. For solid uranium contents, water leakage conditions are simulated by flooding all regions outside and inside of the containment vessel, including the sealed convenience cans. For liquid uranium contents, water leakage conditions are simulated by flooding all regions outside the containment vessel except the containment vessel well. Uranyl nitrate (UN) solution resides inside both the containment vessel well and the containment vessel, including the sealed convenience cans. For this evaluation, a flooded containment vessel under full water reflection is also analyzed. Under such leakage conditions, the calculated neutron multiplication factor ($k_{eff} + 2\sigma$) for the ES-3100 package with HEU contents is lower than the upper subcritical limit (USL) for a subcritical system. Water inleakage into the containment vessel or liquid content leakage out of the containment vessel will not produce a criticality in the containment vessel of a disassembled package or an assembled single package. Therefore, the Model ES-3100 shipping package complies with all of the requirements of 10 CFR 71.55(b, d).

Credit for the high-integrity, watertight containment is not taken either in the single package analysis [10 CFR 71.55(d, e)] or in the array analysis [10 CFR 71.59(a)(1)] of undamaged packages. In the evaluation of undamaged packages under 10 CFR 71.59(a)(1) and the evaluation of damaged packages under 10 CFR 71.59(a)(2), the containment vessel is flooded with water, providing moderation to such an extent as to cause maximum reactivity of the content consistent with the chemical and physical form of the material present. Solid HEU, <u>not solution HEU</u>, is being shipped in the ES-3100. Consequently, in the evaluation of damaged packages under 10 CFR 71.59(a)(2), the leakage out of the containment vessel of content moderated to such an extent as to cause maximum reactivity consistent with the physical and chemical form of material is not considered credible HAC, based on results for tests specified in 10 CFR 71.73. Because credit for the high-integrity, watertight containment is not fully taken in this criticality evaluation, the fissile material mass loading limits are very conservative.

Containment vessel flooding is assumed in the criticality calculations performed for the derivation of fissile material loading limits. Even though both the NCT tests under 10 CFR 71.71 and the HAC tests under 10 CFR 71.73 demonstrate that containment is not breached, simulation of this condition in the criticality calculations produces package configurations that are more reactive than the actual package configurations. The 7.1–10.1 kg quantities of evaluation water required in both the NCT and HAC criticality calculations bound reasonable amounts of hydrogenous material and inherent moisture of fissile material (primarily HEU oxides) present inside the containment vessel. Thus, an administrative criticality control is not needed to restrict the amount of hydrogenous material normally present inside the containment vessel and other sources of moisture present in the fissile content. Nevertheless, anticipated amounts of hydrogenous materials are not expected to exceed 2,400 g. These materials include: can pads, polyethylene bags, vinyl tape; and polyethylene or Teflon bottles if used; and moisture in oxide contents.

Under NCT and HAC, three parameters that affect criticality and may vary during transport of the Model ES-3100 package are the number of packages transported, the amount of water present in the package, and the volatile (organic material) contents of the package. The number of packages transported is limited by the criticality safety index (CSI) established by this criticality evaluation. Both the amount of water present in the package and the volatile (organic material) contents of the package are parameters for the following reasons. First, volatile materials can be driven off at the high temperatures of HAC. Second, the inherent water content of the Kaolite in NCT and the water absorption by the Kaolite in HAC are unknowns. These parameters determine a variety of competing effects that govern the fission process, including, but not limited to, mass, moderation, absorption, and reflection. To ensure that all effects are adequately included in this criticality evaluation, the range of these parameters is evaluated.

It is possible for accidents to be significantly more severe in the air transport mode than in the surface transport mode. Thus, the performance requirements for packages designed to be transported by air are more stringent. These requirements address separate aspects of the accident assessment and apply only to the criticality evaluation of an individual package under isolation. (TS-G-1.1) For a fissile material package designed to be transported by air, 10 CFR 71.55(f) requires that the criticality evaluation demonstrate that the package be subcritical assuming reflection by 20 cm (7.9 in.) of water but no water inleakage when subjected to the sequential application of the HAC free drop and crush tests of 10 CFR 71.73(c)(1 and 2) and the modified puncture and thermal tests of 10 CFR 71.55(f)(1)(iii and iv).

The ES-3100 package was not subjected to physical testing of Type-B fissile material packages transported by air. Instead, analytic methods are used to demonstrate compliance of the ES-3100 package with the requirements of 10 CFR 71.55(f). This is accomplished by demonstrating compliance of the package with the more stringent requirements imposed by the IAEA for situations where the conditions of the fissile material package following the tests cannot be demonstrated. (TS-G-1.1, Sect. 680.2) Section 680.2 of TS-G-1.1 states that worst-case assumptions regarding the geometric arrangement of the package and contents should be made in the criticality evaluation taking into account all moderating and structural components of the packaging. The assumptions should be in conformity with the potential worst-case effects of the mechanical and thermal tests, and all package orientations should be considered for the analysis. Subcriticality must be demonstrated after due consideration of such aspects as the efficiency of the moderator, loss of neutron absorbers, rearrangement of packaging components and contents, geometry changes, and temperature effects. Given that the requirements of 10 CFR 71.55(f) mirror the IAEA requirements of TS-R-1, Sect. 680, for Type B(U) fissile material packages, this approach is considered an acceptable one.

The following criticality safety evaluation shows that the Model ES-3100 package with designated content satisfies the requirements of 10 CFR 71.55 and 71.59 for surface transport, and of 10 CFR 71.55(f) for air transport. Designated contents are solid HEU metal shapes (cylinders, bars, buttons, billets, slugs, unirradiated TRIGA fuel); HEU broken metal contents of unspecified geometric shapes; HEU products or skull oxides; or UNX crystals. HEU skull oxides are distinguished from product oxides (UO_2, U_3O_8, or UO_3) as being a reside of graphite and oxidized uranium (U_3O_8) recovered in the casting process. Tables 6.1a–6.1e pertain to surface-only modes of transportation. These tables summarize the results of the evaluation for solid HEU metal shapes (Tables 6.1a and 6.1b), for HEU broken metal (Table 6.1c), for HEU product or skull oxide (Table 6.1d), and for UNX crystals or un-irradiated TRIGA fuel elements (Table 6.1e). In Table 6.1c, the fissile or the uranium masses for broken metal listed in the content headings indicate the evaluation limits of the criticality calculations. As determined by the NCT and HAC array analyses, mass loading limits may be further reduced to coincide with fissile material loadings at or below the subcritical safety limit. These reduced loading limits are identified in the "CSI" rows in bold type. Table 6.2a pertains to surface-only transport mode while Table 6.2b pertains to air-transport. The loading limit for packages under mixed-mode transportation is taken as the most restrictive limit for either mode, surface or air transport.

Research reactor fuel elements or fuel components are composed of U-Al, U_3O_8-Al, UO_2, or UO_2-Mg. Related materials are broken uranium metal, U-Al alloy, or oxides of U_3O_8, U_3O_8-Al, or UO_2-Mg. With the exception of the oxide and compounds, research reactor related contents are packaged for air transport under limits specified in Table 6.2b for solid HEU metal of unspecified geometric shapes characterized as broken metal. The oxide and compounds are packaged for air transport under limits specified in Table 6.2b for unirradiated TRIGA fuel elements. Per package limits for these items are a maximum of 716 g ^{235}U at enrichments not exceeding 20 wt% and a maximum of 408 g ^{235}U for content enrichments greater than 20 wt%.

The criticality evaluation demonstrates that the ES-3100 packaging with the HEU content satisfies the requirements for single packages and for arrays of fissile material packages when the packages are load-limited as specified in Tables 6.2a and 6.2b (reproduced in Table 1.3) under the conditions identified in Sect. 6.2.4.

6.1.3 Criticality Safety Index

A CSI is assigned to the Model ES-3100 package on the basis of an adequate margin of subcriticality for the single package and arrays of packages for both NCT and HAC. Values for the CSI given in Tables 6.2a and 6.2b (Table 1.3) are based on the uranium and the ^{235}U mass of the content and on the presence of 277-4 canned spacers as defined in Sect. 6.2.2.

The CSI is determined from bounding calculations using KENO V.a models of the containment vessel, the single package, and arrays of packages. (SCALE, Vol. 2, Sect. F11) It is a dimensionless number used to limit the number of packages in a conveyance for nuclear criticality safety control. The CSI is the larger of the CSI values for NCT and for HAC. For NCT, the CSI is equal to 50 divided by the allowable number of packages "N" that can be shipped, where the allowable number of packages is one-fifth of the maximum array size that is calculated to be subcritical. For HAC, the CSI is equal to 50 divided by the allowable number of packages "N" that can be shipped, where the allowable number of packages is one-half of the maximum array size that is calculated to be subcritical. The CSI is a calculated number rounded up to the nearest first decimal.

The array sizes examined in this evaluation are infinite, 13×13×6, 9×9×4, 7×7×3, 5×5×2, explicit triangular pitch (ETP) 27×3 for NCT, ETP 16×3 for HAC, and the degenerate single unit. For NCT, the "N" and corresponding CSI values for arrays determined to be adequately subcritical are as follows: N = ∞, CSI = 0; N = 202, CSI = 0.3; N = 64, CSI = 0.8; N = 29, CSI = 1.8; N = 10, CSI = 5.0; and N = 16, CSI = 3.2. For HAC, the "N" and corresponding CSI values for arrays determined to be adequately subcritical are as follows: N = ∞, CSI = 0; N = 507, CSI = 0.1; N =162, CSI = 0.4; N = 73, CSI = 0.7; N = 25, CSI = 2.0; and N = 24, CSI = 2.1. Absent an exact correspondence of CSI values for the NCT and HAC, the following arrays results were selected for rounded CSI values as indicated in Tables 6.1a–6.1d:

- infinite arrays evaluated for both NCT and HAC [designated as "(1,2)"], where N(1,2) = ∞ for a CSI = 0;
- 13×13×6 evaluated for NCT and 9×9×4 evaluated for HAC where N(1,2) = (202,162) for a CSI = 0.4;
- 9×9×4 evaluated for NCT and 7×7×3 evaluated for HAC where N(1,2) = (64,73) for a CSI = 0.8;
- 7×7×3 evaluated for NCT and 5×5×2 evaluated for HAC where N(1,2) = (29,25) for a CSI = 2.0; and
- 27×3 evaluated for NCT and 16×3 evaluated for HAC where N(1,2) = (16,24) for a CSI = 3.2.

6.2 PACKAGE CONTENTS

The **package content** is defined as the HEU fissile material, bottles, convenience cans, canned spacers, can pads, and the associated packing materials (plastic bags, pads, tape, etc.) inside the ES-3100 containment vessel.

6.2.1 Fissile Material Contents

The per-package HEU mass loadings considered in the criticality evaluation range from 1000 to 36,000 g for uranium metal and from 1000 to 24,000 g for uranium oxide and UNX crystals. The HEU mass may include nonradioactive contaminants and trace elements or materials in the HEU.

The bounding types of HEU content evaluated in this criticality analysis are 4.25-in.- and 3.24-in.-diam cylinders; 2.29-in.-square bars; 1.5-in.-diam × 2-in.-tall slugs; cubes ranging from 0.25 to 1 in. on a side; broken metal pieces of unspecified geometric shapes; skull oxide; uranium oxide; UNX crystals; and unirradiated TRIGA reactor fuel elements.

The term "broken metal pieces" is used to describe an HEU content without restrictions on shape or size other than a minimum size limit (spontaneous ignition), a maximum mass limit (criticality control), a minimum enrichment (the lower limit for HEU at 19 wt % ^{235}U in uranium), and the capacity limits of the convenience cans. The content geometry envelope encompasses regular, uniform shapes and sizes as well as irregular shapes and sizes.

The density of HEU metal ranges from 18.811 to 19.003 g/cm^3 for HEU metal, corresponding to enrichments ranging from 100 to 19 wt % ^{235}U. Theoretical (crystalline) densities for HEU oxide are 10.96 g/cm^3, 8.30 g/cm^3, and 7.29 g/cm^3 for UO_2, U_3O_8, and UO_3, respectively. However, bulk densities for product oxide are typically on the order of 6.54 g/cm^3; therefore, only "less-than-theoretical" mass loadings would actually be achieved. Skull oxides are a mixture of U_3O_8 and graphite, having densities on the order of 2.44 g/cm^3 for poured material and 2.78 g/cm^3 for tapped material. Combined water saturation and crystallization of the HEU oxide is not expected in the HAC because UO_2 and UO_3 are non-hygroscopic and U_3O_8 is only mildly hygroscopic. The density of UNX crystals varies depending on the degree of hydration. The most reactive form of $UO_2(NO_3)\bullet xH_2O$ is with 6 molecules of hydration, having a density of 2.79 g/cm^3. UNX crystals are highly soluble in nitric acid and mildly soluble in water. Dissolution of UNX crystals in water is assumed in this criticality evaluation. The content geometry envelope encompasses both regular, uniform clumps and densities, and irregular clumps and densities.

Table 6.1a. Summary of criticality evaluation for solid HEU metal cylinders and bars

Conditions	cylinders ($d \le 3.24$ in.) no can spacers	cylinders ($d \le 3.24$ in.) with can spacers	square bars (l,w ≤ 2.29 in.) no can spacers	cylinders ($3.24 < d \le 4.25$ in.) no can spacers	cylinders ($3.24 < d \le 4.25$ in.) with can spacers
General requirements for each fissile package (§71.55)					
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, . . . so that under the following conditions, maximum reactivity of the fissile material would be attained:" (Paragraph "b")	$k_{eff} + 2\sigma \le 0.9228$ cvcrcyt11_21_1	$k_{eff} + 2\sigma \le 0.8866$ cvcrcyt11_36_2	$k_{eff} + 2\sigma \le 0.8787$ cvcrsqt11_36_1	$k_{eff} + 2\sigma \le 0.9215$ cvcrcyct11_17_1	$k_{eff} + 2\sigma \le 0.9202$ cvcrcyct11_32_2
(1) the most reactive credible configuration consistent with the chemical and physical form of the material,	3 stacked cylinders ($d = 3.24$ in.) 1 cylinder per convenience can, no can spacers, 21,000g ^{235}U	3 stacked cylinders ($d = 3.24$ in.) 1 cylinder per convenience can, with can spacers, 36,000g ^{235}U	3 stacked bars (l,w = 2.29 in.) 1 bar per convenience can, no can spacers, 36,000g ^{235}U	3 stacked cylinders ($d = 4.25$ in.) 1 cylinder per convenience can, no can spacers, 17,000g ^{235}U	3 stacked cylinders ($d = 4.25$ in.) 1 cylinder per convenience can, with can spacers, 32,000g ^{235}U
(2) moderation by water to the most reactive credible extent,	flooding of the containment vessel	same	same	same	same
(3) close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may be provided by the surrounding material of the packaging.	30.48 cm H_2O surrounding the containment vessel	same	same	same	same

Table 6.1a. Summary of criticality evaluation for solid HEU metal cylinders and bars

Conditions	cylinders (d ≤ 3.24 in.) no can spacers	cylinders (d ≤ 3.24 in.) with can spacers	square bars (l,w ≤ 2.29 in.) no can spacers	cylinders (3.24 < d ≤ 4.25 in.) no can spacers	cylinders (3.24 < d ≤ 4.25 in.) with can spacers
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited under the tests specified in §71.71 (Normal Conditions of Transport) . . ." (Paragraph "d")					
(1) the contents would be subcritical,	$k_{eff} + 2\sigma \leq 0.9035$ ncsrcyt11_21_1_15	$k_{eff} + 2\sigma \leq 0.8731$ ncsrcyt11_36_2_15	$k_{eff} + 2\sigma \leq 0.8652$ ncsrsqt11_36_1_15	$k_{eff} + 2\sigma \leq 0.8941$ ncsrcyct11_17_1_15	$k_{eff} + 2\sigma \leq 0.8946$ ncsrcyct11_32_2_15
(2) the geometric form of the package contents would not be substantially altered,	3 stacked cylinders (d = 3.24 in.) 1 cylinder per convenience can, no can spacers, 21,000g ^{235}U	3 stacked cylinders (d = 3.24 in.) 1 cylinder per convenience can, with can spacers, 36,000g ^{235}U	3 stacked bars (l,w = 2.29 in.) 1 bar per convenience can, no can spacers, 36,000g ^{235}U	3 stacked cylinders (d = 4.25 in.) 1 cylinder per convenience can, no can spacers, 17,000g ^{235}U	3 stacked cylinders (d = 4.25 in.) 1 cylinder per convenience can, with can spacers, 32,000g ^{235}U
(3) there would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under §71.59(a)(1), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material,	moderation is present to such an extent as to cause maximum reactivity	same	same	same	same
(4) there will be no substantial reduction in the effectiveness of the packaging	30.48 cm H_2O surrounding the drum (d=18.37 in., h=43.5 in.)	same	same	same	same

Table 6.1a. Summary of criticality evaluation for solid HEU metal cylinders and bars

Conditions	cylinders ($d \leq 3.24$ in.) no can spacers	cylinders ($d \leq 3.24$ in.) with can spacers	square bars ($l,w \leq 2.29$ in.) no can spacers	cylinders ($3.24 < d \leq 4.25$ in.) no can spacers	cylinders ($3.24 < d \leq 4.25$ in.) with can spacers
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in §71.73 (Hypothetical Accident Conditions) the package would be subcritical. For this determination, it must be assumed that:" (Paragraph "e")	$k_{eff} + 2\sigma \leq 0.9044$ hcsrcyt12_21_1_15	$k_{eff} + 2\sigma \leq 0.8734$ hcsrcyt12_36_2_15	$k_{eff} + 2\sigma \leq 0.8642$ hcsrsqt12_36_1_15	$k_{eff} + 2\sigma \leq 0.8979$ hcsrcyct12_17_1_15	$k_{eff} + 2\sigma \leq 0.8977$ hcsrcyct12_32_2_15
(1) the fissile material is in the most reactive credible configuration consistent with the chemical and physical form of the contents,	3 stacked cylinders ($d = 3.24$ in.) 1 cylinder per convenience can, no can spacers, 21,000g ^{235}U	3 stacked cylinders ($d = 3.24$ in.) 1 cylinder per convenience can, with can spacers, 36,000g ^{235}U	3 stacked bars ($l,w = 2.29$ in.) 1 bar per convenience can, no can spacers, 36,000g ^{235}U	3 stacked cylinders ($d = 4.25$ in.) 1 cylinder per convenience can, no can spacers, 17,000g ^{235}U	3 stacked cylinders ($d = 4.25$ in.) 1 cylinder per convenience can, with can spacers, 32,000g ^{235}U
(2) water moderation occurs to the most reactive credible extent consistent with the chemical and physical form of content,	flooding of the package	same	same	same	same
(3) there is full reflection by water on all sides, as close as is consistent with the damage condition of the package.	30.48 cm H_2O surrounding the reduced diameter drum (d=17.20 in., h=43.5 in.)	same	same	same	same

Table 6.1a. Summary of criticality evaluation for solid HEU metal cylinders and bars

Conditions	cylinders (d ≤ 3.24 in.) no can spacers	cylinders (d ≤ 3.24 in.) with can spacers	square bars (l,w ≤ 2.29 in.) no can spacers	cylinders (3.24 < d ≤ 4.25 in.) no can spacers	cylinders (3.24 < d ≤ 4.25 in.) with can spacers
Standards for arrays of fissile material packages (§71.59)					
"... the designer of a fissile material package shall derive a number "N" based on all the following conditions being satisfied, assuming packages are stacked together in any arrangement and with close reflection on all sides of the stack by water:" (Paragraph "a")					
Transport index based on nuclear criticality control, CSI = 0.0	**load-limited to 18,000g ^{235}U no can spacers**	**load-limited to 30,000g ^{235}U can spacers**	**load-limited to 30,000g ^{235}U no can spacers**	**load-limited to 15,000g ^{235}U no can spacers**	**load-limited to 25,000g ^{235}U can spacers**
(1) five times "N" undamaged packages with nothing between the packages would be subcritical,	$k_{eff} + 2\sigma \leq 0.9237$ nciacyt11_18_1_3	$k_{eff} + 2\sigma \leq 0.9195$ nciacyt11_30_2_3	$k_{eff} + 2\sigma \leq 0.9219$ nciasqt11_30_1_3	$k_{eff} + 2\sigma \leq 0.8992$ nciacyct11_15_1_3	$k_{eff} + 2\sigma \leq 0.9147$ nciacyct11_25_2_3
(2) two times "N" damaged packages, if each package were subject to the tests specified in §71.73 (Hypothetical Accident Conditions) would be subcritical with optimum interspersed hydrogenous moderation,	$k_{eff} + 2\sigma \leq 0.9230$ hciacyt12_18_1_3	$k_{eff} + 2\sigma \leq 0.9220$ hciacyt12_30_2_3	$k_{eff} + 2\sigma \leq 0.9221$ hciasqt12_30_1_3	$k_{eff} + 2\sigma \leq 0.8986$ hciacyct12_15_1_3	$k_{eff} + 2\sigma \leq 0.9156$ hciacyct12_25_2_3
(3) the value of "N" not <0.5.	$N(1,2) = \infty$	same	same	same	same
Transport index based on nuclear criticality control, CSI = 0.4	**load-limited to 18,000g ^{235}U no can spacers**	**load-limited to 30,000g ^{235}U can spacers**	**load-limited to 30,000g ^{235}U can spacers**	**load-limited to 15,000g ^{235}U no can spacers**	**load-limited to 25,000g ^{235}U can spacers**
(1) five times "N" undamaged packages	bounded by CSI=0	same	same	same	same
(2) two times "N" damaged packages, ...	bounded by CSI=0	same	same	same	same
(3) the value of "N"	N(1,2) = 202/162	same	same	same	same

Table 6.1b. Summary of criticality evaluation for solid HEU metal slugs

Conditions	slugs, no can spacers enr. ≤ 100% ≤ 18,287g ^{235}U	slugs with can spacers 80% < enr. ≤ 100% < 36,573g ^{235}U	slugs with can spacers enr. ≤ 80% < 36,573g ^{235}U
General requirements for each fissile package (§71.55)			
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, . . . so that under the following conditions, maximum reactivity of the fissile material would be attained:" (Paragraph "b")	$k_{eff} + 2\sigma \le 0.9089$ cvcr5est11_1_1	$k_{eff} + 2\sigma \le 0.9041$ cvcr6e4st11_2	
(1) the most reactive credible configuration consistent with the chemical and physical form of the material,	cluster of 5 slugs (d = 1.5625 in., h = 2.0625 in.) per convenience can, no can spacers, 18,286g ^{235}U	cluster of 10 slugs stacked 2 high per convenience can, non-uniform content stacking, can spacers, 36,573g ^{235}U	
(2) moderation by water to the most reactive credible extent,	flooding of the containment vessel	same	
(3) close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may be provided by the surrounding material of the packaging.	30.48 cm H_2O surrounding the containment vessel	same	
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited under the tests specified in §71.71 (Normal Conditions of Transport) . . ." (Paragraph "d")			
(1) the contents would be subcritical,	$k_{eff} + 2\sigma \le 0.8704$ ncsr5est11_1_1_15	$k_{eff} + 2\sigma \le 0.8682$ ncsr5est11_2_2_15	
(2) the geometric form of the package contents would not be substantially altered,	cluster of 5 slugs (d = 1.5625 in., h = 2.0625 in.) per convenience can, no can spacers, 18,286g ^{235}U	cluster of 10 slugs stacked 2 high per convenience can, non-uniform content stacking, can spacers, 36,573g ^{235}U	

Table 6.1b. Summary of criticality evaluation for solid HEU metal slugs

Conditions	slugs, no can spacers enr. ≤ 100% ≤ 18,287g ^{235}U	slugs with can spacers 80% < enr. ≤ 100% < 36,573g ^{235}U	slugs with can spacers enr. ≤ 80% < 36,573g ^{235}U
(3) there would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under §71.59(a)(1), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material,	moderation is present to such an extent as to cause maximum reactivity	same	
(4) there will be no substantial reduction in the effectiveness of the packaging	30.48 cm H_2O surrounding the drum (d=18.37 in., h=43.5 in.)	same	
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in §71.73 (Hypothetical Accident Conditions) the package would be subcritical. For this determination, it must be assumed that:" (Paragraph "e")	$k_{eff} + 2\sigma \leq 0.8727$ hcsr5est12_1_1_15	$k_{eff} + 2\sigma \leq 0.8713$ hcsr5est12_2_2_15	
(1) the fissile material is in the most reactive credible configuration consistent with the chemical and physical form of the contents,	cluster of 5 slugs (d = 1.5625 in., h = 2.0625 in.) per convenience can, no can spacers, 18,286g ^{235}U	cluster of 10 slugs stacked 2 high per convenience can, non-uniform content stacking, 36,573g ^{235}U	
(2) water moderation occurs to the most reactive credible extent consistent with the chemical and physical form of content,	flooding of the package	same	
(3) there is full reflection by water on all sides, as close as is consistent with the damage condition of the package.	30.48 cm H_2O surrounding the reduced diameter drum (d=17.20 in., h=43.5 in.)	same	

Table 6.1b. Summary of criticality evaluation for solid HEU metal slugs

Conditions	slugs, no can spacers enr. ≤ 100% ≤ 18,287g ^{235}U	slugs with can spacers 80% < enr. ≤ 100% < 36,573g ^{235}U	slugs with can spacers enr. ≤ 80% < 36,573g ^{235}U
Standards for arrays of fissile material packages (§71.59)			
"... the designer of a fissile material package shall derive a number "N" based on all the following conditions being satisfied, assuming packages are stacked together in any arrangement and with close reflection on all sides of the stack by water: " (Paragraph "a")			
Transport index based on nuclear criticality control, CSI = 0.0	**load-limited to 18,287g ^{235}U no can spacers**	**load-limited to 25,601g ^{235}U can spacers**	**load-limited to 29,333g ^{235}U can spacers**
(1) five times "N" undamaged packages with nothing between the packages would be subcritical,	$k_{eff} + 2\sigma \leq 0.9225$ ncia5est11_1_1_8_3	$k_{eff} + 2\sigma \leq 0.9145$ ncia70st11_2_8_3	$k_{eff} + 2\sigma \leq 0.9093$ ncia5est11_2_2_5_3
(2) two times "N" damaged packages, if each package were subject to the tests specified in §71.73 (Hypothetical Accident Conditions) would be subcritical with optimum interspersed hydrogenous moderation,	$k_{eff} + 2\sigma \leq 0.9238$ hcia5est12_1_1_8_3	$k_{eff} + 2\sigma \leq 0.9133$ hcia70st12_2_8_3	$k_{eff} + 2\sigma \leq 0.9076$ hcia5est12_2_2_5_3
(3) the value of "N" not <0.5.	N(1,2) = ∞	same	same
Transport index based on nuclear criticality control, CSI = 0.4	**load-limited to 18,287g ^{235}U no can spacers**	**load-limited to 25,601g ^{235}U can spacers**	**load-limited to 34,766g ^{235}U can spacers**
(1) five times "N" undamaged packages	bounded by CSI=0	same	$k_{eff} + 2\sigma \leq 0.9208$ ncf15est11_2_2_7_3
(2) two times "N" damaged packages, ...	bounded by CSI=0	same	$k_{eff} + 2\sigma \leq 0.9053$ hcf25est12_2_2_7_3
(3) the value of "N"	N(1,2) = 202/162	same	same

Table 6.1c. Summary of criticality evaluation for solid HEU metal of unspecified geometric shapes characterized as broken metal

Conditions	95% < enr. ≤ 100% ≤ 25,894g ^{235}U	90% < enr. ≤95% ≤ 27,252g ^{235}U	80% < enr. ≤90% ≤ 28,334g ^{235}U	70% < enr. ≤80% ≤ 28,184g ^{235}U	60% < enr. ≤70% ≤ 24,693g ^{235}U	enr. ≤60% ≤ 35,320g Uranium
General requirements for each fissile package (§71.55)						
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, . . . so that under the following conditions, maximum reactivity of the fissile material would be attained:" (Paragraph "b")	$k_{eff} + 2\sigma \leq 0.9181$ cvr3lha_26_1_8_15	$k_{eff} + 2\sigma \leq 0.9206$ cvr3lha_29_1_7_15	$k_{eff} + 2\sigma \leq 0.9224$ cvr3lha_32_1_6_15	$k_{eff} + 2\sigma \leq 0.8927$ cvr3lha_36_1_5_15	$k_{eff} + 2\sigma \leq 0.8620$ cvr3lha_36_1_4_15	$k_{eff} + 2\sigma \leq 0.8289$ cvr3lha_36_1_3_15
(1) the most reactive credible configuration consistent with the chemical and physical form of the material,	A mixture of HEU metal and water is homogenized over the internal volume of an assumed content lattice, one per can location. The square footprint of the content lattice is circumscribed by the inner wall of the containment vessel. The amount of water in the flooded containment vessel is calculated on the basis that the convenience can steel is replaced with water. Can pads and spacers not used. See Appendix 6.9.3, Sect. 6.9.3.1, for justification of the content model.					
(2) moderation by water to the most reactive credible extent,	flooding of the containment vessel					
(3) close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may be provided by the surrounding material of the packaging.	30.48 cm H_2O surrounding the containment vessel					

Table 6.1c. Summary of criticality evaluation for solid HEU metal of unspecified geometric shapes characterized as broken metal

Conditions	95% < enr. ≤ 100% ≤ 25,894g ^{235}U	90% < enr. ≤95% ≤ 27,252g ^{235}U	80% < enr. ≤90% ≤ 28,334g ^{235}U	70% < enr. ≤80% ≤ 28,184g ^{235}U	60% < enr. ≤70% ≤ 24,693g ^{235}U	enr. ≤60% ≤ 35,320g Uranium
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited under the tests specified in §71.71 (Normal Conditions of Transport) . . ." (Paragraph "d")						
(1) the contents would be subcritical,	$k_{eff} + 2\sigma \leq 0.8908$ (ncsrbmt11_36_1_15)					
(2) the geometric form of the package contents would not be substantially altered,	A mixture of HEU metal and water is homogenized over the internal volume of the containment vessel. The amount of water in the flooded containment vessel is calculated on the basis that the convenience can steel is replaced with water. Can pads and spacers not used.					
(3) there would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under §71.59(a)(1), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material,	moderation is present to such an extent as to cause maximum reactivity					
(4) there will be no substantial reduction in the effectiveness of the packaging	30.48 cm H_2O surrounding the drum (d = 18.37 in., h = 43.5 in.)					

Table 6.1c. Summary of criticality evaluation for solid HEU metal of unspecified geometric shapes characterized as broken metal

Conditions	95% < enr. ≤ 100% ≤ 25,894g ^{235}U	90% < enr. ≤95% ≤ 27,252g ^{235}U	80% < enr. ≤90% ≤ 28,334g ^{235}U	70% < enr. ≤80% ≤ 28,184g ^{235}U	60% < enr. ≤70% ≤ 24,693g ^{235}U	enr. ≤60% ≤ 35,320g Uranium
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in §71.73 (Hypothetical Accident Conditions) the package would be subcritical. For this determination, it must be assumed that:" (Paragraph "e")	$k_{eff} + 2\sigma \le 0.8912$ (hcsrbmt12_36_1_15)					
(1) the fissile material is in the most reactive credible configuration consistent with the chemical and physical form of the contents,	A mixture of HEU metal and water is homogenized over the internal volume of the containment vessel. The amount of water in the flooded containment vessel is calculated on the basis that the convenience can steel is replaced with water. Can pads and spacers not used.					
(2) water moderation occurs to the most reactive credible extent consistent with the chemical and physical form of content,	flooding of the package					
(3) there is full reflection by water on all sides, as close as is consistent with the damage condition of the package.	30.48 cm H_2O surrounding the reduced diameter drum (d = 17.20 in., h = 43.5 in.)					

Table 6.1c. Summary of criticality evaluation for solid HEU metal of unspecified geometric shapes characterized as broken metal

Conditions	95% < enr. ≤ 100% ≤ 25,894g ^{235}U	90% < enr. ≤95% ≤ 27,252g ^{235}U	80% < enr. ≤90% ≤ 28,334g ^{235}U	70% < enr. ≤80% ≤ 28,184g ^{235}U	60% < enr. ≤70% ≤ 24,693g ^{235}U	enr. ≤60% ≤ 35,320g Uranium
Standards for arrays of fissile material packages (§71.59)						
"... the designer of a fissile material package shall derive a number "N" based on all the following conditions being satisfied, assuming packages are stacked together in any arrangement and with close reflection on all sides of the stack by water: " (Paragraph "a")						
Transport index based on nuclear criticality control, CSI = 0.0	**can spacers are required**	**can spacers are required**	**can spacers are required**	**load-limited to 2,967g ^{235}U no can spacers**	**load-limited to 3,249g ^{235}U no can spacers**	**load-limited to 5,577g Uranium no can spacers**
(1) five times "N" undamaged packages with nothing between the packages would be subcritical,	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \leq 0.9213$ nciabmt11_4_1_5_3	$k_{eff} + 2\sigma \leq 0.9207$ nciabmt11_5_1_4_3	$k_{eff} + 2\sigma \leq 0.9152$ nciabmt11_6_1_3_3
(2) two times "N" damaged packages, if each package were subject to the tests specified in §71.73 (Hypothetical Accident Conditions) would be subcritical with optimum interspersed hydrogenous moderation,	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \leq 0.9248$ hciabmt12_4_1_5_3	$k_{eff} + 2\sigma \leq 0.9235$ hciabmt12_5_1_4_3	$k_{eff} + 2\sigma \leq 0.9127$ hciabmt12_6_1_3_3
(3) the value of "N" cannot be <0.5.	Not applicable	Not applicable	Not applicable	N(1,2) = ∞	same	same

Table 6.1c. Summary of criticality evaluation for solid HEU metal of unspecified geometric shapes characterized as broken metal

Conditions	95% < enr. ≤ 100% ≤ 25,894g ^{235}U	90% < enr. ≤95% ≤ 27,252g ^{235}U	80% < enr. ≤90% ≤ 28,334g ^{235}U	70% < enr. ≤80% ≤ 28,184g ^{235}U	60% < enr. ≤70% ≤ 24,693g ^{235}U	enr. ≤60% ≤ 35,320g Uranium
Transport index based on nuclear criticality control, CSI = 0.0	**load-limited to 2,774g ^{235}U can spacers**	**load-limited to 3,516g ^{235}U can spacers**	**load-limited to 3,333g ^{235}U can spacers**	**load-limited to 4,450g ^{235}U can spacers**	**load-limited to 5,198g ^{235}U can spacers**	**load-limited to 11,154g Uranium can spacers**
(1) five times "N" undamaged packages ...	$k_{eff} + 2\sigma \leq 0.9041$ nciabmt11_3_2_8_3	$k_{eff} + 2\sigma \leq 0.9199$ nciabmt11_4_2_7_3	$k_{eff} + 2\sigma \leq 0.9084$ nciabmt11_4_2_6_3	$k_{eff} + 2\sigma \leq 0.9224$ nciabmt11_6_2_5_3	$k_{eff} + 2\sigma \leq 0.9218$ nciabmt11_8_2_4_3	$k_{eff} + 2\sigma \leq 0.9181$ nciabmt11_12_2_3_3
(2) two times "N" damaged packages, ...	$k_{eff} + 2\sigma \leq 0.9050$ hciabmt12_3_2_8_3	$k_{eff} + 2\sigma \leq 0.9237$ hciabmt12_4_2_7_3	$k_{eff} + 2\sigma \leq 0.9091$ hciabmt12_4_2_6_3	$k_{eff} + 2\sigma \leq 0.9230$ hciabmt12_6_2_5_3	$k_{eff} + 2\sigma \leq 0.9209$ hciabmt12_8_2_4_3	$k_{eff} + 2\sigma \leq 0.9205$ hciabmt12_12_2_3_3
(3) the value of "N" ...	N(1,2) = ∞	same	same	same	same	same
Transport index based on nuclear criticality control, CSI = 0.4	**can spacers are required**	**can spacers are required**	**can spacers are required**	**load-limited to 5,192g ^{235}U no can spacers**	**load-limited to 5,848g ^{235}U no can spacers**	**load-limited to 14,872g Uranium no can spacers**
(1) five times "N" undamaged packages	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \leq 0.9235$ ncf1bmt11_7_1_5_3	$k_{eff} + 2\sigma \leq 0.9216$ ncf1bmt11_9_1_4_3	$k_{eff} + 2\sigma \leq 0.9225$ ncf1bmt11_15_1_3_3
(2) two times "N" damaged packages, ...	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \leq 0.9064$ hcf2bmt12_8_1_5_3	$k_{eff} + 2\sigma \leq 0.9118$ hcf2bmt12_12_1_4_3	$k_{eff} + 2\sigma \leq 0.9072$ hcf2bmt12_18_1_3_3
(3) the value of "N" ...	Not applicable	Not applicable	Not applicable	N(1,2) = 202/162	same	same
Transport index based on nuclear criticality control, CSI = 0.4	**load-limited to 5,549g ^{235}U can spacers**	**load-limited to 6,154g ^{235}U can spacers**	**load-limited to 7,500g ^{235}U can spacers**	**load-limited to 8,900g ^{235}U can spacers**	**load-limited to 12,996g ^{235}U can spacers**	**load-limited to 28,813g Uranium can spacers**
(1) five times "N" undamaged packages	$k_{eff} + 2\sigma \leq 0.9241$ ncf1bmt11_6_2_8_3	$k_{eff} + 2\sigma \leq 0.9220$ ncf1bmt11_7_2_7_3	$k_{eff} + 2\sigma \leq 0.9203$ ncf1bmt11_9_2_6_3	$k_{eff} + 2\sigma \leq 0.9187$ ncf1bmt11_12_2_5_3	$k_{eff} + 2\sigma \leq 0.9245$ ncf1bmt11_19_2_4_3	$k_{eff} + 2\sigma \leq 0.9235$ ncf1bmt11_29_2_3_3
(2) two times "N" damaged packages, ...	$k_{eff} + 2\sigma \leq 0.9244$ hcf2bmt12_8_2_8_3	$k_{eff} + 2\sigma \leq 0.9232$ hcf2bmt12_11_2_7_3	$k_{eff} + 2\sigma \leq 0.9192$ hcf2bmt12_12_2_6_3	$k_{eff} + 2\sigma \leq 0.8964$ hcf2bmt12_12_2_5_3	$k_{eff} + 2\sigma \leq 0.9005$ hcf2bmt12_19_2_4_3	$k_{eff} + 2\sigma \leq 0.9112$ hcf2bmt12_36_2_3_3
(3) the value of "N"	N(1,2) = 202/162	same	same	same	same	same

Table 6.1c. Summary of criticality evaluation for solid HEU metal of unspecified geometric shapes characterized as broken metal

Conditions	95% < enr. ≤ 100% ≤ 25,894g ^{235}U	90% < enr. ≤95% ≤ 27,252g ^{235}U	80% < enr. ≤90% ≤ 28,334g ^{235}U	70% < enr. ≤80% ≤ 28,184g ^{235}U	60% < enr. ≤70% ≤ 24,693g ^{235}U	enr. ≤60% ≤ 35,320g Uranium
Transport index based on nuclear criticality control, CSI = 0.8	**can spacers are required**	**can spacers are required**	**can spacers are required**	**load-limited to 8,900g ^{235}U no can spacers**	**load-limited to 13,646g ^{235}U no can spacers**	**load-limited to 28,814g Uranium no can spacers**
(1) five times "N" undamaged packages	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \leq 0.9216$ ncf2bmt11_12_1_5_3	$k_{eff} + 2\sigma \leq 0.9234$ ncf2bmt11_20_1_4_3	$k_{eff} + 2\sigma \leq 0.9205$ ncf2bmt11_29_1_3_3
(2) two times "N" damaged packages, ...	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \leq 0.9012$ hcf3bmt12_12_1_5_3	$k_{eff} + 2\sigma \leq 0.9065$ hcf3bmt12_20_1_4_3	$k_{eff} + 2\sigma \leq 0.9022$ hcf3bmt12_29_1_3_3
(3) the value of "N"	Not applicable	Not applicable	Not applicable	same	same	same
Transport index based on nuclear criticality control, CSI = 0.8	**load-limited to 9,248g ^{235}U can spacers**	**load-limited to 10,549g ^{235}U can spacers**	**load-limited to 12,500g ^{235}U can spacers**	**load-limited to 16,317g ^{235}U can spacers**	**load-limited to 20,793g ^{235}U can spacers**	**35,320g Uranium can spacers**
(1) five times "N" undamaged packages	$k_{eff} + 2\sigma \leq 0.9233$ ncf2bmt11_10_2_8_3	$k_{eff} + 2\sigma \leq 0.9221$ ncf2bmt11_12_2_7_3	$k_{eff} + 2\sigma \leq 0.9220$ ncf2bmt11_14_2_6_3	$k_{eff} + 2\sigma \leq 0.9193$ ncf2bmt11_21_2_5_3	$k_{eff} + 2\sigma \leq 0.9233$ ncf2bmt11_30_2_4_3	$k_{eff} + 2\sigma \leq 0.9064$ ncf2bmt11_36_2_3_3
(2) two times "N" damaged packages, ...	$k_{eff} + 2\sigma \leq 0.9090$ hcf3bmt12_10_2_8_3	$k_{eff} + 2\sigma \leq 0.9005$ hcf3bmt12_12_2_7_3	$k_{eff} + 2\sigma \leq 0.9042$ hcf3bmt12_14_2_6_3	$k_{eff} + 2\sigma \leq 0.9022$ hcf3bmt12_21_2_5_3	$k_{eff} + 2\sigma \leq 0.9019$ hcf3bmt12_30_2_4_3	$k_{eff} + 2\sigma \leq 0.8850$ hcf3bmt12_36_2_3_3
(3) the value of "N"	N(1,2) = 64/73	same	same	same	same	same
Transport index based on nuclear criticality control, CSI = 2.0 *	**can spacers are required**	**can spacers are required**	**can spacers are required**	**load-limited to 17,059g ^{235}U no can spacers**	**load-limited to 21,444g ^{235}U no can spacers**	**35,320g Uranium no can spacers**
(1) five times "N" undamaged packages	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \leq 0.9223$ ncf3bmt11_22_1_5_3	$k_{eff} + 2\sigma \leq 0.9216$ ncf3bmt11_31_1_4_3	$k_{eff} + 2\sigma \leq 0.9045$ ncf3bmt11_36_1_3_3
(2) two times "N" damaged packages, ...	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \leq 0.8852$ hcf4bmt12_22_1_5_3	$k_{eff} + 2\sigma \leq 0.8858$ hcf4bmt12_31_1_4_3	$k_{eff} + 2\sigma \leq 0.8705$ hcf4bmt12_36_1_3_3
(3) the value of "N"	Not applicable	Not applicable	Not applicable	N(1,2) = 29/25	same	same

Table 6.1c. Summary of criticality evaluation for solid HEU metal of unspecified geometric shapes characterized as broken metal

Conditions	95% < enr. ≤ 100% ≤ 25,894g ^{235}U	90% < enr. ≤95% ≤ 27,252g ^{235}U	80% < enr. ≤90% ≤ 28,334g ^{235}U	70% < enr. ≤80% ≤ 28,184g ^{235}U	60% < enr. ≤70% ≤ 24,693g ^{235}U	enr. ≤60% ≤ 35,320g Uranium
Transport index based on nuclear criticality control, CSI = 2.0 *	**load-limited to 13,872g ^{235}U can spacers**	**load-limited to 18,461g ^{235}U can spacers**	**load-limited to 20,000g ^{235}U can spacers**	**load-limited to 25,218g ^{235}U can spacers**	**24,692g ^{235}U can spacers**	**35,320g Uranium can spacers**
(1) five times "N" undamaged packages	$k_{eff} + 2\sigma \le 0.9204$ ncf3bmt11_14_2_8_3	$k_{eff} + 2\sigma \le 0.9239$ ncf3bmt11_20_2_7_3	$k_{eff} + 2\sigma \le 0.9213$ ncf3bmt11_23_2_6_3	$k_{eff} + 2\sigma \le 0.9216$ ncf3bmt11_32_2_5_3	$k_{eff} + 2\sigma \le 0.9035$ ncf3bmt11_36_2_4_3	$k_{eff} + 2\sigma \le 0.8745$ ncf3bmt11_36_2_3_3
(2) two times "N" damaged packages, ...	$k_{eff} + 2\sigma \le 0.8877$ hcf4bmt12_14_2_8_3	$k_{eff} + 2\sigma \le 0.8914$ hcf4bmt12_20_2_7_3	$k_{eff} + 2\sigma \le 0.8851$ hcf4bmt12_23_2_6_3	$k_{eff} + 2\sigma \le 0.8884$ hcf4bmt12_32_2_5_3	$k_{eff} + 2\sigma \le 0.8684$ hcf4bmt12_36_2_4_3	$k_{eff} + 2\sigma \le 0.8412$ hcf4bmt12_36_2_3_3
(3) the value of "N"	N(1,2) = 29/25	same	same	same	same	same
Transport index based on nuclear criticality control, CSI = 3.2 **	**can spacers are required**	**can spacers are required**	**can spacers are required**	**load-limited to 27,443g ^{235}U no can spacers**	**24,692g ^{235}U no can spacers**	**35,320g Uranium no can spacers**
(1) five times "N" undamaged packages	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \le 0.9226$ ncf5bmt11_35_1_5_3	$k_{eff} + 2\sigma \le 0.9030$ ncf5bmt11_36_1_4_3	$k_{eff} + 2\sigma \le 0.8743$ ncf5bmt11_36_1_3_3
(2) two times "N" damaged packages, ...	Not applicable	Not applicable	Not applicable	$k_{eff} + 2\sigma \le 0.9052$ hcf5bmt12_35_1_5_3	$k_{eff} + 2\sigma \le 0.8819$ hcf5bmt12_36_1_4_3	$k_{eff} + 2\sigma \le 0.8542$ hcf5bmt12_36_1_3_3
(3) the value of "N"	Not applicable	Not applicable	Not applicable	N(1,2) = 16/24	same	same
Transport index based on nuclear criticality control, CSI = 3.2 **	**load-limited to 24,969g ^{235}U can spacers**	**load-limited to 26,373g ^{235}U can spacers**	**28,334g ^{235}U can spacers**	**28,184g ^{235}U can spacers**	**24,692g ^{235}U can spacers**	**35,320g Uranium can spacers**
(1) five times "N" undamaged packages	$k_{eff} + 2\sigma \le 0.9249$ ncf5bmt11_25_2_8_3	$k_{eff} + 2\sigma \le 0.9196$ ncf5bmt11_28_2_7_3	$k_{eff} + 2\sigma \le 0.9246$ ncf5bmt11_35_2_6_3	$k_{eff} + 2\sigma \le 0.8974$ ncf5bmt11_36_2_5_3	$k_{eff} + 2\sigma \le 0.8736$ ncf5bmt11_36_2_4_3	$k_{eff} + 2\sigma \le 0.8445$ ncf5bmt11_36_2_3_3
(2) two times "N" damaged packages, ...	$k_{eff} + 2\sigma \le 0.9089$ hcf5bmt12_25_2_8_3	$k_{eff} + 2\sigma \le 0.8981$ hcf5bmt12_28_2_7_3	$k_{eff} + 2\sigma \le 0.9032$ hcf5bmt12_35_2_6_3	$k_{eff} + 2\sigma \le 0.8804$ hcf5bmt12_36_2_5_3	$k_{eff} + 2\sigma \le 0.8532$ hcf5bmt12_36_2_4_3	$k_{eff} + 2\sigma \le 0.8265$ hcf5bmt12_36_2_3_3
(3) the value of "N"	N(1,2) = 16/24	same	same	same	same	same

* Neutron multiplication factors and "N" are shown for an NCT array size of 7×7×3 and an HAC array size of 5×5×2; both arrays are nearly cubic arrangements. An HAC array size is not available for CSI=1.8; therefore, CSI for NCT at 1.8 is rounded up to 2.0 as shown.

** Neutron multiplication factors and "N" are shown for an NCT array size of 27×3 and an HAC array size of 16×3; where packages in the planar dimension are in explicit triangular-pitch arrangement and stacked three high in the vertical direction. An NCT array size of 81 packages gives a CSI value of 3.12 (50/16), while an HAC array size of 48 packages gives a CSI value of 2.08 (50/24). The specified CSI value of 3.2 is the larger of the two numbers rounded up to the nearest tenth decimal place as shown.

Table 6.1d. Summary of criticality evaluation for HEU product and skull oxide

Conditions	HEU product oxide ≤ 21,125g ^{235}U	HEU skull oxide ≤ 15,673g ^{235}U and ≤ 921g C
General requirements for each fissile package (§71.55)		
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, . . . so that under the following conditions, maximum reactivity of the fissile material would be attained:" (Paragraph "b")	$k_{eff} + 2\sigma \leq 0.8728$ (cvcroxt11_1_24_1)	$k_{eff} + 2\sigma \leq 0.8497$ (cvcrsk3cc_9_15_17)
(1) the most reactive credible configuration consistent with the chemical and physical form of the material,	HEU oxide at the bulk density is assumed dispersed in the containment vessel, where the oxide fills the containment vessel to a height determined by the oxide mass. The oxide is saturated with water. Water fills the void region above the oxide content, but this amount of water is reduced by an amount equivalent to the volume occupied by can spacers. The can pads, spacer can, and convenience can steel are replaced by water. See Appendix 6.9.3, Sect. 6.9.3.1, for justification of the content model.	same
(2) moderation by water to the most reactive credible extent,	flooding of the containment vessel	same
(3) close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may be provided by the surrounding material of the packaging.	30.48 cm H_2O surrounding the containment vessel	same

Table 6.1d. Summary of criticality evaluation for HEU product and skull oxide

Conditions	HEU product oxide ≤ 21,125g ^{235}U	HEU skull oxide ≤ 15,673g ^{235}U and ≤ 921g C
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited under the tests specified in §71.71 (Normal Conditions of Transport) . . ." (Paragraph "d")		
(1) the contents would be subcritical,	$k_{eff} + 2\sigma \leq 0.7835$ (ncsroxt11_1_24_1_15)	$k_{eff} + 2\sigma \leq 0.7410$ (ncsrsk_9_15)
(2) the geometric form of the package contents would not be substantially altered,	HEU oxide at the bulk density is assumed dispersed in the containment vessel, where the oxide fills the containment vessel to a height determined by the oxide mass. The oxide is saturated with water. Water fills the void region above the oxide content, but this amount of water is reduced by an amount equivalent to the volume occupied by can spacers. The can pads, spacer can, and convenience can steel are replaced by water.	same
(3) there would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under §71.59(a)(1), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material,	moderation is present to such an extent as to cause maximum reactivity	same
(4) there will be no substantial reduction in the effectiveness of the packaging	30.48 cm H_2O surrounding the drum (d = 18.37 in., h = 43.5 in.)	same

Table 6.1d. Summary of criticality evaluation for HEU product and skull oxide

Conditions	HEU product oxide ≤ 21,125g ^{235}U	HEU skull oxide ≤ 15,673g ^{235}U and ≤ 921g C
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in §71.73 (Hypothetical Accident Conditions) the package would be subcritical. For this determination, it must be assumed that:" (Paragraph "e")	$k_{eff} + 2\sigma \leq 0.7867$ (hcsroxt12_1_24_1_15)	$k_{eff} + 2\sigma \leq 0.7410$ (hcsrsk_9_15)
(1) the fissile material is in the most reactive credible configuration consistent with the chemical and physical form of the contents,	HEU oxide at the bulk density is assumed dispersed in the containment vessel, where the oxide fills the containment vessel to a height determined by the oxide mass. The oxide is saturated with water. Water fills the void region above the oxide content, but this amount of water is reduced by an amount equivalent to the volume occupied by can spacers. The can pads, spacer can, and convenience can steel are replaced by water.	same
(2) water moderation occurs to the most reactive credible extent consistent with the chemical and physical form of content,	flooding of the package	same
(3) there is full reflection by water on all sides, as close as is consistent with the damage condition of the package.	30.48 cm H_2O surrounding the reduced diameter drum, (d = 17.20 in., h = 43.5 in.)	same

Table 6.1d. Summary of criticality evaluation for HEU product and skull oxide

Conditions	HEU product oxide ≤ 21,125g ^{235}U	HEU skull oxide ≤ 15,673g ^{235}U and ≤ 921g C
Standards for arrays of fissile material packages (§71.59)		
" . . . the designer of a fissile material package shall derive a number "N" based on all the following conditions being satisfied, assuming packages are stacked together in any arrangement and with close reflection on all sides of the stack by water: " (Paragraph "a")		
Transport index based on nuclear criticality control, CSI = 0.0	**load-limited to 21,125g ^{235}U no can spacers**	**load-limited to 15,673g ^{235}U no can spacers**
(1) five times "N" undamaged packages with nothing between the packages would be subcritical,	$k_{eff} + 2\sigma \leq 0.8908$ nciaoxt11_1_24_1_3	$k_{eff} + 2\sigma \leq 0.7562$ nciask_9_15
(2) two times "N" damaged packages, if each package were subject to the tests specified in §71.73 (Hypothetical Accident Conditions) would be subcritical with optimum interspersed hydrogenous moderation,	$k_{eff} + 2\sigma \leq 0.8877$ hciaoxt12_1_24_1_3	$k_{eff} + 2\sigma \leq 0.7599$ hciask_9_15
(3) the value of "N" cannot be <0.5.	N(1,2) = ∞	same
Transport index based on nuclear criticality control, CSI = 0.0	**load-limited to 21,125g ^{235}U can spacers**	**load-limited to 15,673g ^{235}U can spacers**
(1) five times "N" undamaged packages	$k_{eff} + 2\sigma \leq 0.8904$ nciaoxt11_1_24_2_3	bounded by no can spacers
(2) two times "N" damaged packages, ...	$k_{eff} + 2\sigma \leq 0.8883$ hciaoxt12_1_24_2_3	bounded by no can spacers
(3) the value of "N"	N(1,2) = ∞	same

Table 6.1e. Summary of criticality evaluation for UNX crystals and unirradiated TRIGA reactor fuel elements

Conditions	UNX crystals ≤ 11,303g ^{235}U	unirradiated TRIGA fuel elements ≤ 921g ^{235}U
General requirements for each fissile package (§71.55)		
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, . . . so that under the following conditions, maximum reactivity of the fissile material would be attained:" (Paragraph "b")	$k_{eff} + 2\sigma \leq 0.8572$ (cvcrunhct11_9_1)	$k_{eff} + 2\sigma \leq 0.5274$ (cvcrtriga_1_15)
(1) the most reactive credible configuration consistent with the chemical and physical form of the material,	UNX crystals are homogenized with water over the internal volume of the containment vessel consistent with the high solubility properties of UNX. Maximum reactivity occurs at 415 g U/l solution concentration. Amount of water calculated for the flooded containment vessel is reduced by an amount equivalent to the volume occupied by can spacers. The can pads, spacer can, and convenience can steel are replaced by water.	Three 5-in tall sectioned pieces of $UZrH_x$ from a cylindrical TRIGA fuel element per convenience can. Sectioned pieces arranged in triangular pitch. No can spacers. Convenience can steel replaced by water.
(2) moderation by water to the most reactive credible extent,	flooding of the containment vessel	same
(3) close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may be provided by the surrounding material of the packaging.	30.48 cm H_2O surrounding the containment vessel	same

Table 6.1e. Summary of criticality evaluation for UNX crystals and unirradiated TRIGA reactor fuel elements

Conditions	UNX crystals ≤ 11,303g ^{235}U	unirradiated TRIGA fuel elements ≤ 921g ^{235}U
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited under the tests specified in §71.71 (Normal Conditions of Transport) . . ." (Paragraph "d")		
(1) the contents would be subcritical,	$k_{eff} + 2\sigma \leq 0.7448$ (ncsrunhct11_9_1_15)	$k_{eff} + 2\sigma \leq 0.4950$ (ncsrtriga_1_15_15)
(2) the geometric form of the package contents would not be substantially altered,	UNX crystals are homogenized with water over the internal volume of the containment vessel consistent with the high solubility properties of UNX. Maximum reactivity occurs at 415 g U/l solution concentration. Amount of water calculated for the flooded containment vessel is reduced by an amount equivalent to the volume occupied by can spacers. The can pads, spacer can, and convenience can steel are replaced by water.	Three 5-in tall sectioned pieces of $UZrH_x$ from a cylindrical TRIGA fuel element per convenience can. Sectioned pieces arranged in triangular pitch. No can spacers. Convenience can steel replaced by water.
(3) there would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under §71.59(a)(1), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material,	moderation is present to such an extent as to cause maximum reactivity	same
(4) there will be no substantial reduction in the effectiveness of the packaging	30.48 cm H_2O surrounding the drum (d = 18.37 in., h = 43.5 in.)	same

Table 6.1e. Summary of criticality evaluation for UNX crystals and unirradiated TRIGA reactor fuel elements

Conditions	UNX crystals ≤ 11,303g ^{235}U	unirradiated TRIGA fuel elements ≤ 921g ^{235}U
"A package used for shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in §71.73 (Hypothetical Accident Conditions) the package would be subcritical. For this determination, it must be assumed that:" (Paragraph "e")	$k_{eff} + 2\sigma \leq 0.8074$ (icsrunhct12_24_1_15)	$k_{eff} + 2\sigma \leq 0.4945$ (hcsrtriga_1_15_15)
(1) the fissile material is in the most reactive credible configuration consistent with the chemical and physical form of the contents,	Consistent with the high solubility properties of UNX, crystals and water are homogenized over the internal volume of the containment vessel and void space of the containment vessel well. Amount of water calculated for the flooded containment vessel is reduced by an amount equivalent to the volume occupied by can spacers. The can pads, spacer can, and convenience can steel are replaced by water.	Three 5-in tall sectioned pieces of $UZrH_x$ from a cylindrical TRIGA fuel element per convenience can. Sectioned pieces arranged in triangular pitch. No can spacers. Convenience can steel replaced by water.
(2) water moderation occurs to the most reactive credible extent consistent with the chemical and physical form of content,	flooding of the package	same
(3) there is full reflection by water on all sides, as close as is consistent with the damage condition of the package.	30.48 cm H_2O surrounding the reduced diameter drum, (d = 17.20 in., h = 43.5 in.)	same

Table 6.1e. Summary of criticality evaluation for UNX crystals and unirradiated TRIGA reactor fuel elements

Conditions	UNX crystals ≤ 11,303g ^{235}U	unirradiated TRIGA fuel elements ≤ 921g ^{235}U
Standards for arrays of fissile material packages (§71.59)		
" . . . the designer of a fissile material package shall derive a number "N" based on all the following conditions being satisfied, assuming packages are stacked together in any arrangement and with close reflection on all sides of the stack by water: " (Paragraph "a")		
Transport index based on nuclear criticality control, CSI = 0.0	**load-limited to 3,768g ^{235}U no can spacers**	**no can spacers**
(1) five times "N" undamaged packages with nothing between the packages would be subcritical,	$k_{eff} + 2\sigma \leq 0.9191$ nciaunhct11_8_8_1_3	$k_{eff} + 2\sigma \leq 0.5254$ nciatriga_1_15_3
(2) two times "N" damaged packages, if each package were subject to the tests specified in §71.73 (Hypothetical Accident Conditions) would be subcritical with optimum interspersed hydrogenous moderation,	$k_{eff} + 2\sigma \leq 0.9216$ hciaunhct12_8_8_1_3	$k_{eff} + 2\sigma \leq 0.5261$ hciatriga_1_15_3
(3) the value of "N" cannot be <0.5.	$N(1,2) = \infty$	same
Transport index based on nuclear criticality control, CSI = 0.0	**load-limited to 11,303g ^{235}U can spacers**	**can spacers**
(1) five times "N" undamaged packages	$k_{eff} + 2\sigma \leq 0.8803$ nciaunhct11_8_24_2_3	$k_{eff} + 2\sigma \leq 0.4421$ nciatriga_2_15_3
(2) two times "N" damaged packages, ...	$k_{eff} + 2\sigma \leq 0.8811$ hciaunhct12_8_24_2_3	$k_{eff} + 2\sigma \leq 0.4427$ hciatriga_2_15_3
(3) the value of "N"	$N(1,2) = \infty$	same

Table 6.1e. Summary of criticality evaluation for UNX crystals and unirradiated TRIGA reactor fuel elements

Conditions	**UNX crystals ≤ 11,303g ^{235}U**	**unirradiated TRIGA fuel elements ≤ 921g ^{235}U**
Transport index based on nuclear criticality control, CSI = 0.4	**load-limited to 11,303g ^{235}U no can spacers**	**no can spacers**
(1) five times "N" undamaged packages	$k_{eff} + 2\sigma \le 0.8703$ ncf1unhct11_8_24_1_3	bounded by CSI=0
(2) two times "N" damaged packages, ...	$k_{eff} + 2\sigma \le 0.8321$ hcf2unhct12_8_24_1_3	bounded by CSI=0
(3) the value of "N"	N(1,2) = 202/162	same
Transport index based on nuclear criticality control, CSI = 0.4	**load-limited to 11,303g ^{235}U can spacers**	**can spacers**
(1) five times "N" undamaged packages	$k_{eff} + 2\sigma \le 0.8332$ ncf1unhct11_8_24_2_3	bounded by CSI=0
(2) two times "N" damaged packages, ...	$k_{eff} + 2\sigma \le 0.7972$ hcf2unhct12_8_24_2_3	bounded by CSI=0
(3) the value of "N"	N(1,2) = 202/162	same

Table 6.2a. HEU fissile material mass loading limits for surface-only modes of transportation

Solid HEU metal of specified geometric shapes						
Transport index based on nuclear criticality control	**cylinders (3.24 in. ≤ d)**	**cylinders (3.24 in. < d ≤ 4.25 in.)**	**bars**	**slugs enr. ≤ "N"%**	**slugs 80% < enr. ≤ 100 %**	**slugs enr. ≤ 80%**
No can spacers						
CSI = 0.0	18,000g ^{235}U	15,000g ^{235}U	18,000g ^{235}U	18,286g ^{235}U [a]	-	-
With can spacers						
CSI = 0.0	30,000g ^{235}U	25,000g ^{235}U	30,000g ^{235}U	-	25,601g ^{235}U	29,333g ^{235}U
CSI = 0.4	-	-	-	34,766g ^{235}U [b]	-	-
Solid HEU metal of unspecified geometric shapes characterized as broken metal						
Transport index based on nuclear criticality control	**95% < enr. ≤ 100%**	**90% < enr. ≤95%**	**80% < enr. ≤90%**	**70% < enr. ≤80%**	**60% < enr. ≤70%**	**enr. ≤60%**
No can spacers						
CSI = 0.0	can spacers required	can spacers required	can spacers required	2,967g ^{235}U	3,249g ^{235}U	5,576g Uranium
CSI = 0.4	can spacers required	can spacers required	can spacers required	5,192g ^{235}U	5,848g ^{235}U	14,872g Uranium
CSI = 0.8	can spacers required	can spacers required	can spacers required	8,900 ^{235}U	13,646g ^{235}U	28,814g Uranium
CSI = 2.0	can spacers required	can spacers required	can spacers required	17,059g ^{235}U	21,444g ^{235}U	35,320g Uranium
CSI = 3.2	can spacers required	can spacers required	can spacers required	27,692g ^{235}U	24,692g ^{235}U	35,320g Uranium
With can spacers						
CSI = 0.0	2,774g ^{235}U	3,516g ^{235}U	3,333g ^{235}U	4,450g ^{235}U	5,198g ^{235}U	11,154g Uranium
CSI = 0.4	5,549g ^{235}U	6,154g ^{235}U	7,500g ^{235}U	8,900g ^{235}U	12,996g ^{235}U	28,813g Uranium
CSI = 0.8	9,248g ^{235}U	10,549g ^{235}U	12,500g ^{235}U	16,317g ^{235}U	20,793g ^{235}U	35,320g Uranium
CSI = 2.0	13,872g ^{235}U	18,461g ^{235}U	20,000g ^{235}U	25,218g ^{235}U	24,692g ^{235}U	35,320g Uranium
CSI = 3.2	24,969g ^{235}U	26,373g ^{235}U	28,334g ^{235}U	28,184g ^{235}U	24,692g ^{235}U	35,320g Uranium

Table 6.2a. HEU fissile material mass loading limits for surface-only modes of transportation

HEU oxide and UNX crystals					
Transport index based on nuclear criticality control	**HEU product oxide, no can spacers**	**HEU skull oxide, no can spacers**	**UNX crystals, no can spacers**	**unirradiated TRIGA fuel elements, no can spacers [c]**	
				20% enrichment	70% enrichment
CSI = 0.0	21,124g ^{235}U	15,675g ^{235}U and 921g C	3,768g ^{235}U	921g ^{235}U	408g ^{235}U
CSI = 0.4	-	-	11,303g ^{235}U	-	-

[a] When can spacers **are not** used, "N" = 100 (wt % ^{235}U), and the mass limit = 18,286g ^{235}U.

[b] When can spacers **are** used, "N" = 95 (wt % ^{235}U), and the mass limit = 34,766g ^{235}U. For enrichments >95 wt % ^{235}U, the lower mass limit of 25,601g ^{235}U applies.

[c] For ground transport, TRIGA reactor fuel element content will be limited to 3 fuel sections ("meats") per loaded convenience can and up to 3 loaded cans per package. The TRIGA fuel content may also be configured as clad fuel rods, each rod derived from a single TRIGA fuel element. A ~15 inch long rod consists of the 3 fuel pellets and an exterior sheath of clad, where protruding clad at each end has been crimped in. Clad fuel rods will be packed into convenience cans, with a maximum of three fuel rods per loaded convenience can and one loaded can per containment vessel.

Table 6.2b. HEU fissile material mass loading limits for air transport mode of transportation

Solid HEU metal of specified geometric shapes					
One per convenience can	**cylinders (d = 3.24 in.)**	**cylinders (3.24 < d ≤ 4.25 in.)**	**bars**	**slugs**	
	500g ^{235}U	500g ^{235}U	500g ^{235}U	500g ^{235}U	
Solid HEU metal of unspecified geometric shapes characterized as broken metal [a]					
	enr. ≤ 20%		**20% < enr. ≤ 100%**		
	700g ^{235}U		500g ^{235}U		
HEU oxide, UNX crystals, unirradiated TRIGA fuel elements [b]					
	HEU product oxide	**HEU skull oxide**	**UNX crystals**	**unirradiated TRIGA [c]**	
				20% enrichment	**70% enrichment**
	not allowed	**not allowed**	**not allowed**	716g ^{235}U	408g ^{235}U

[a] Research reactor related contents of uranium metal or U-Al alloy are packaged for air transport under these specified limits.

[b] Research reactor related contents of U_3O_8, U_3O_8-Al, UO_2, or UO_2-Mg are packaged for air transport under limits specified for unirradiated TRIGA fuel elements. Per package limits for these items are a maximum of 716 g ^{235}U at enrichments not exceeding 20 wt% and a maximum of 408 g ^{235}U for content enrichments greater than 20 wt%.

[c] For air transport, TRIGA reactor fuel element content will be limited to fuel sections or clad fuel rods as described for surface-only modes of transportation in footnote "c" of Table 6.2a and the fissile mass limit specified herein, whichever is more limiting.

The approximate 40 stock items of TRIGA fuel are cataloged as one of the four basic types: a standard element, an instrumented element, a fuel follower control rod, or a cluster assembly. The active region of TRIGA element consists of three 5-in long sections "fuel meats" of uranium zirconium hydride ($UZrH_x$). The "x" in $UZrH_x$ equals 1.6 in all cases except for two stock items where x = equals 1.0 and the fissile content is < 40 g ^{235}U. The clad thickness is ~0.02 inches for a TRIGA fuel element with stainless steel cladding and ~0.03 inches for an element with aluminum cladding.

Solid form TRIGA fuel is either 20 or 70 wt % enriched in ^{235}U and has specific dimensional characteristics for its designed function. For the 20 wt % enriched TRIGA elements, the active fuel diameters are 1.44 in., 1.41 in., 1.40 in., 1.37 in., 1.34 in., or 1.31 in. The uranium weight fractions are 45 wt %, 30 wt %, 20 wt %, 12 wt %, and 8.5 wt %. The TRIGA element with a maximum fissile content of 307 g ^{235}U in 1,560 g U is 45 wt% U in $UZrH_x$ and has a computed density of ~8.6 g/cm^3 (Appendix 6.9.3.1). For the 70 wt % enriched TRIGA fuel, the active fuel diameter is 1.44 inches in the standard element and instrumented element, and 1.31 inches in the fuel follower control rod. Both the standard element and instrumented elements contain ~136 g ^{235}U in 194 g U while the fuel follower control rod contains ~113 g ^{235}U in 162 g U. The 70 wt % enriched TRIGA fuel is 8.5 wt% U in $UZrH_x$ and has a computed density of ~5.7 g/cm^3 (Appendix 6.9.3.1).

In preparation for shipment in the ES-3100, the unirradiated TRIGA fuel elements may be disassembled and the fuel sections removed from the thin-wall cladding. The TRIGA fuel may also be configured as clad fuel rods. Each clad fuel rod will be derived from a single TRIGA fuel element by removal of the stainless steel or aluminum clad beyond the plenum adjacent to the axial ends of the active fuel section. Each ~15 inch long rod consists of the 3 fuel pellets and an exterior sheath of clad, where the protruding clad at each end has been crimped in.

The 0.02 in thick sheath of stainless steel clad adds ~179 g to the mass of the active fuel for the standard element or instrumented element with 1.48 in. overall diameter, and ~163 g to active fuel mass for the fuel follower control rod with 1.35 in. overall diameter. Allowance for 1/2 in. of residual stainless steel crimped on each end of the clad fuel rod adds ~11 to 12 g stainless steel to these amounts. Likewise, the 0.03 in thick sheath of aluminum clad adds ~90 g to the mass of the active fuel for the 1.47 in. diameter standard element or instrumented element. Allowance for 1/2 in. of residual aluminum crimped on each end of the clad fuel rod adds ~6 g aluminum.

Research reactor fuel elements or fuel components are composed of U-Al, U_3O_8-Al, UO_2, or UO_2-Mg. Related materials include broken uranium metal, U-Al alloy, or oxides of U_3O_8, U_3O_8-Al, or UO_2-Mg. These items contain small quantities of fissile uranium typically ranging from 4.2 g ^{235}U to 152 g ^{235}U. The maximum enrichment is approximately 93%. Components are of various solid shapes (rods, square bars, plates, or tubes).

Skull oxides, uranium alloys of aluminum or molybdenum, and unirradiated TRIGA reactor fuel elements are evaluated where composition data is for material in the as-manufactured condition. A maximum enrichment of 100 wt % is used for HEU metal, oxide and UNX crystals in the criticality calculations strictly for the purpose of maximizing reactivity, even though HEU enrichment ranges from 19 to 97.7 wt % ^{235}U. Although mass loading limits for oxide and crystals are based on 100% enrichment, the actual enrichment is expected to be less than the stated maximum, with the remainder of the uranium being primarily ^{238}U. The HEU mass may also include nonradioactive contaminants and trace elements or materials in the HEU.

No intact weapon part or component will be shipped in this package. Weapon parts or components that have been reduced to "broken metal pieces" or processed into HEU oxide and meet the additional content requirements identified in Sect. 6.2.4 may be shipped in this package.

6.2.2 Convenience Cans, Teflon and Polyethylene Bottles, and 277-4 Canned Spacers

HEU fissile material to be shipped in the ES-3100 package will be placed in stainless steel, tin-plated carbon steel, or nickel alloy convenience cans or polyethylene or Teflon bottles. Can lids may be welded, press fit, slip lid, or crimp seal types; bottle lids are screw cap type. Convenience cans and bottles are used to hold the HEU for shipment in the ES-3100 package and to assure that the inside of the containment vessel does not become contaminated with HEU under NCT. The HEU metal or oxide content may be wrapped or bagged in polyethylene, and the convenience cans may also be wrapped in polyethylene to further reduce the possibility of radioactive contamination of the packaging. Nylon straps may be used for handling the nickel alloy cans. Masses of the convenience cans and packing materials are in addition to the fissile material mass.

Three 4.25-in.-diam × 10.0-in.-tall or six 4.25-in.-diam × 4.88-in.-tall convenience cans, separated by can pads, may be placed inside the 31-in.-tall cavity of the ES-3100 containment vessel (Fig. 1.4). The convenience cans may have press-fit or crimp-seal lids. The size of solid content placed inside a convenience can is physically limited by both the can opening and the usable height of the convenience can. Other can arrangements fit inside the containment vessel, such as three 4.25-in.-diam × 8.75-in.-tall convenience cans or five 4.25-in.-diam × 4.88-in.-tall convenience cans. Both of these can arrangements include can pads and 277-4 canned spacers. Nickel alloy cans to be used exclusively for HEU oxide content are ~3-in. diam × 4.75-in. tall. Three Teflon bottles (4.69-in. diam × 9.4-in. tall) or three polyethylene bottles (4.94-in. diam × 8.76-in. tall) fit inside the containment vessel; however, 277-4 canned spacers are not used with these configurations.

A modified convenience can may be used to package the TRIGA and research reactor fuel element and fuel components that do not fit inside these containers. A 17.5 in. or 30 in.-tall convenience can is constructed from two 4.25-in.-diam × 8.75-in.-tall or three 10-in. cans brazed together. The research reactor fuel elements and fuel components that cannot be packaged in these convenience cans due to their overall dimensions will be bundled together and protected on each end with an open-ended convenience can having an outer diameter ≤5 in.

The can pad and 277-4 canned spacer have overall dimensions of 4.25-in. diam × 1.82-in. height. (Drawing M2E801580A026, Appendix 1.4.8) A stainless-steel spacer can with a lid has a usable cavity with dimensions of 4.13-in. diam × 1.37-in. height, which nominally allows for ~510 g of neutron poison material.

6.2.3 Packing Materials

Polyethylene bags may be used for contamination control. Realistically, the number of polyethylene bags used to bag content and wrap a convenience can would be six bags at most. Given that each bag weighs ~28 g, this results in a maximum of ~510 g of polyethylene per ES-3100 package. Possible additional sources of hydrogenous material inside the containment vessel include can pads, vinyl tape, and polyethylene (~85 g) or Teflon (~303 g) bottles.

For the criticality calculations, the sources of hydrogen contained in packing materials are not distinguished from other sources of hydrogen inherent in the fissile material content. Therefore, packing material mass is defined as all sources of hydrogenous packing materials inside the containment vessel plus the actual moisture in the content as constituent (the bonded hydrogen in UNX crystals or impurities in the oxide.) Given content moisture is not to exceed 6 wt %, the actual amount of hydrogenous material inside the containment vessel is not expected to exceed 2,400 g.

Normally, the H/X ratio inside the containment vessel is specified as an administrative control used to restrict both the amount of hydrogenous packing material normally used inside the containment vessel and other sources of moisture present in the fissile content. The total amount of hydrogen inside the containment vessel is used in the determination of package H/X ratio (i.e., the ratio of the number of hydrogen atoms "H" to the number of fissile atoms "X," where the hydrogen atoms are those of the content, including absorbed moisture and of the packing material both inside and outside of the convenience cans.) However, containment vessel flooding is assumed in calculations performed for the derivation of fissile material loading limits. Even though both the NCT tests under 10 CFR 71.71 and the HAC tests under 10 CFR 71.73 demonstrate that containment is not breached, this simulated condition produces more reactive package configurations in the criticality calculations than the actual package configurations. The 7.1–10.1 kg quantities of evaluation water required in both the NCT and HAC criticality calculations bound reasonable amounts of hydrogenous material and inherent moisture of fissile material (primarily HEU oxides) present inside the containment vessel. Thus, an administrative criticality control is not needed to restrict either the amount of hydrogenous material normally present inside the containment vessel or other sources of moisture present in the fissile content.

6.2.4 Package Content Loading Restrictions

Loading restrictions based upon the results of the criticality safety calculations presented in Sects. 6.4 and 6.5 are as follows:

(1) HEU fissile material to be shipped in the ES-3100 package shall be placed in stainless steel, tin-plated carbon steel or nickel-plated carbon steel convenience cans or polyethylene or Teflon bottles. The can lids may be welded, press fit, slip lid, or crimp seal types; bottle lids are screw cap type.

(2) The content shall not exceed the "per package" fissile material mass loading limits specified in Tables 6.2a and 6.2b, and 277-4 canned spacers shall be used as indicated for criticality control. Research reactor related contents of uranium metal or U-Al alloy may packaged for air transport under limits specified in Table 6.2b for solid HEU metal of unspecified geometric shapes characterized as broken metal. Research reactor related contents of uranium oxide and oxide compounds (U_3O_8, U_3O_8-Al, UO_2, or UO_2-Mg) may be packaged for air transport under limits specified in Table 6.2b for unirradiated TRIGA fuel elements. Per package limits for these items are a maximum of 716 g ^{235}U for content at enrichments not exceeding 20 wt% and a maximum of 408 g ^{235}U for content enrichments greater than 20 wt%.

(3) Where 277-4 canned spacers are required, not greater than one-third of the permissible package content for that category indicated in Tables 6.2a or 6.2b shall be loaded in any vacancy between or adjacent to canned spacers inside the containment vessel. The content mass loading may be further restricted based on structural, mechanical, and practical considerations (see Sects. 1 and 2).

(4) As shown in Fig. 1.4, the ES-3100 package may carry up to six loaded convenience cans. In situations where the plan for loading the containment vessel calls for the use of empty convenience cans to fill the containment vessel, the heavier cans shall be loaded into the bottom of the upright shipping container, and the empty cans shall be placed above them.

(5) The presence of uranium isotopes is limited on a weight-percent basis as follows: $^{232}U \leq 40$ ppb U, $^{234}U \leq 2.0$ wt % U, $^{235}U \leq 100.0$ wt % U, and $^{236}U \leq 40.0$ wt % U.

(6) With the exception of slug content, unirradiated TRIGA reactor fuel element content, and research reactor related content, solid HEU metal or alloy content of specified geometric shapes shall be one item per loaded convenience can. HEU bulk metal or alloy content not covered by the specified geometric shapes (cylinder, square bar, billet, slug, or unirradiated TRIGA fuel element contents) will be in the HEU broken metal category, and so limited.

(7) The package content is defined as the HEU fissile material, bottles, the convenience cans, the can spacers, and the associated packing materials (plastic bags, pads, tape, etc.) inside the ES-3100 containment vessel.

(8) The CSI is determined on the basis of the uranium enrichment and total ^{235}U mass in the package and the fissile material shape or form.

6.3 GENERAL CONSIDERATIONS

The ES-3100 packaging configuration is shown on Drawing M2E801580A031 (Appendix 1.4.8). KENO V.a modeling of this configuration with the maximum allowable contents (Sect. 6.2) for a variety of array sizes and array conditions yields bounding calculations that determine the package's CSI (Tables 6.1a–6.1e) Can spacers are used as indicated in Table 6.2a and 6.2b for the purpose of reducing neutronic interaction between the contents of the package, aiding in maintaining the $k_{eff} + 2\sigma$ below the USL for the ES-3100 package. Key input listings are provided in Appendix 6.9.7.

The HEU content of a package is in one of the following forms: metal of a specified geometric shape, metal of an unspecified shape characterized as broken metal, uranium or skull oxide, UNX crystals, or unirradiated TRIGA reactor fuel elements. The bounding types of HEU content evaluated in this criticality analysis are: 4.25-in.- and 3.24-in.-diam cylinders; 2.29-in.-square bars; 1.5-in.-diam × 2-in.-tall slugs; cubes ranging from 0.25 to 1 in. on a side; broken metal pieces of unspecified geometric shapes; product and skull oxide; UNH crystals; and unirradiated TRIGA reactor fuel elements. Uranyl nitrate hexahydrate (UNH) has a chemical formula of $UO_2(NO_3)_2 + 6\ H_2O$. This most reactive form is used as the bounding composition for UNX crystals in the criticality evaluation. TRIGA fuel elements are 1.44-in.-diam × 15-in.-tall cylinders of $UZrH_x$ containing <310 g ^{235}U. Fuel elements are sectioned into three equal length pieces. Calculations demonstrate that these contents are bounded by 3.24-in.-diam HEU metal cylinders. Evaluation of the 3.93-in.-diam × 9.5-in.-tall U-Al cylinders is covered under the evaluation of the 4.25-in.-diam HEU metal cylinders. The evaluation of the U-Mo content is covered under the assessment of HEU contents at 95 wt % enrichment.

HEU metal shapes are distributed in an optimum arrangement in the flooded containment vessel. For the single package and the array calculations, the HEU broken metal is modeled as a homogeneous mixture of uranium metal and water filling the interior of a flooded containment vessel. This representation bounds the heterogeneous configuration of metal pieces interspersed with hydrogenous packing material inside of wrapped convenience cans. (Appendix 6.9.3., Sect. 6.9.3.1) Water soluble

UNH crystalline content is modeled as a homogeneous mixture of UN and water filling the interior of the containment vessel.

No credit is taken for the fissile material spacing, neutron absorption, or free volume reduction provided by the presence of can pads, spacer can steel, and convenience cans inside the containment vessel. Water is substituted for polyethylene bagging which may be in use as packing material for both the content placed inside the convenience can and the cans themselves. Pads of steel turnings rolled up into a disk-like shape may also be present in the ES-3100 package, in use as cushioning and for reducing the free volume inside the convenience cans. This steel packing material acts as a neutron absorber and is excluded from the calculation model.

Criticality calculations are performed for the containment vessel under full water reflection whereby the water content inside the containment vessel is varied from dry to fully flooded conditions. These calculations demonstrate that the fully flooded condition is most reactive. The containment vessel is flooded in the single-unit calculation model and the infinite and finite array calculation models for both the NCT and HAC evaluations.

The KENO V.a models discussed in the following sections are the single-unit calculation model (Sect. 6.3.1.1), the infinite and finite array calculation models (Sect. 6.3.1.2), the HAC calculation models (Sect. 6.3.1.3), and the air transport models (Sect. 6.3.1.4). The single-unit calculation model is evaluated with a vacuum boundary condition and with full water reflection. The finite array calculation model is evaluated in arrays consisting of packages stacked in 13×13×6, 9×9×4, 7×7×3, 5×5×2, ETP 27×3 and ETP 16×3 arrangements with full water reflection at the array boundary. The ETP arrangements are used specifically for defining smaller array configurations with a high CSI value.

The geometry of the ES-3100 package is depicted in Drawing No. M2E801580A001 (Appendix 1.4.8). Calculation models of this geometry for evaluating NCT and HAC must be constructed for the single-unit, infinite array and finite arrays within the constraints and capabilities of KENO V.a. As shown in the drawing, the ES-3100 geometry is complex. Given KENO V.a's constraints and capabilities, two methods may be used to evaluate these complex geometries: simplify the geometries with conservative approximations or construct accurate geometries from simple components. Both methods yield valid results; however, the latter method is chosen for this analysis in order to maximize accuracy and to eliminate unnecessary conservatism.

6.3.1 Model Configuration

A detailed ES-3100 geometry model is accurately constructed using many simple geometric shapes. The selection of these components is governed by two of KENO V.a's geometry constraints: geometry regions must be composed of uniform and homogeneous materials and exterior regions must completely enclose interior regions. It is apparent from Drawing No. M2E801580A001 (Appendix 1.4.8) that these constraints could not be simultaneously applied to the entire ES-3100 package. However, these constraints could be applied to vertical segments of the package. Segments (i.e., simple components) are defined by starting at the bottom of the package and defining geometry regions radially outward until the drum surface is reached. A vertical segment is extended upward to the point where the KENO V.a constraints are violated. This vertical position is the termination point of a segment (i.e., the interface with the adjacent segment above it). The vertical segments are constructed accordingly, ignoring the minor variations in the ES-3100 geometry (i.e., radii of curvature, beveled edges, nuts and bolts). The KENO V.a geometry model for the ES-3100 is then assembled from the vertical segments. The resulting calculation model geometry includes the HEU content, 277-4 canned spacers, the containment vessel, the stainless-steel liner, the inner-liner cavity filled with 277-4, the outer-liner cavity filled with Kaolite, the Kaolite top plug and steel shell, the silicon rubber spacers, and the stainless-steel drum.

6.3.1.1 Single-unit packaging calculation model

The single-unit packaging calculation model is comprised of the geometry model, material compositions, and boundary conditions. Figures 6.1–6.5 depict section views of the geometry model used to evaluate a single ES-3100 package. Excluding minor variations in the ES-3100 geometry (i.e., radii of curvature, beveled edges, nuts and bolts), the single-unit packaging calculation model is an accurate representation of the ES-3100 geometry.

Figure 6.1 depicts a vertical section view of an ES-3100 package with the content of three loaded convenience cans and 277-4 can spacers inside the containment vessel. The fissile material contents shown are HEU cylinders, but convenience cans are not modeled. As illustrated in Fig. 6.1, credit is not taken for fissile material spacing provided by convenience cans inside the containment vessel. Appendix 6.9.1 provides wire-frame schematic and isometric diagrams depicting other fissile material contents considered in this criticality calculation.

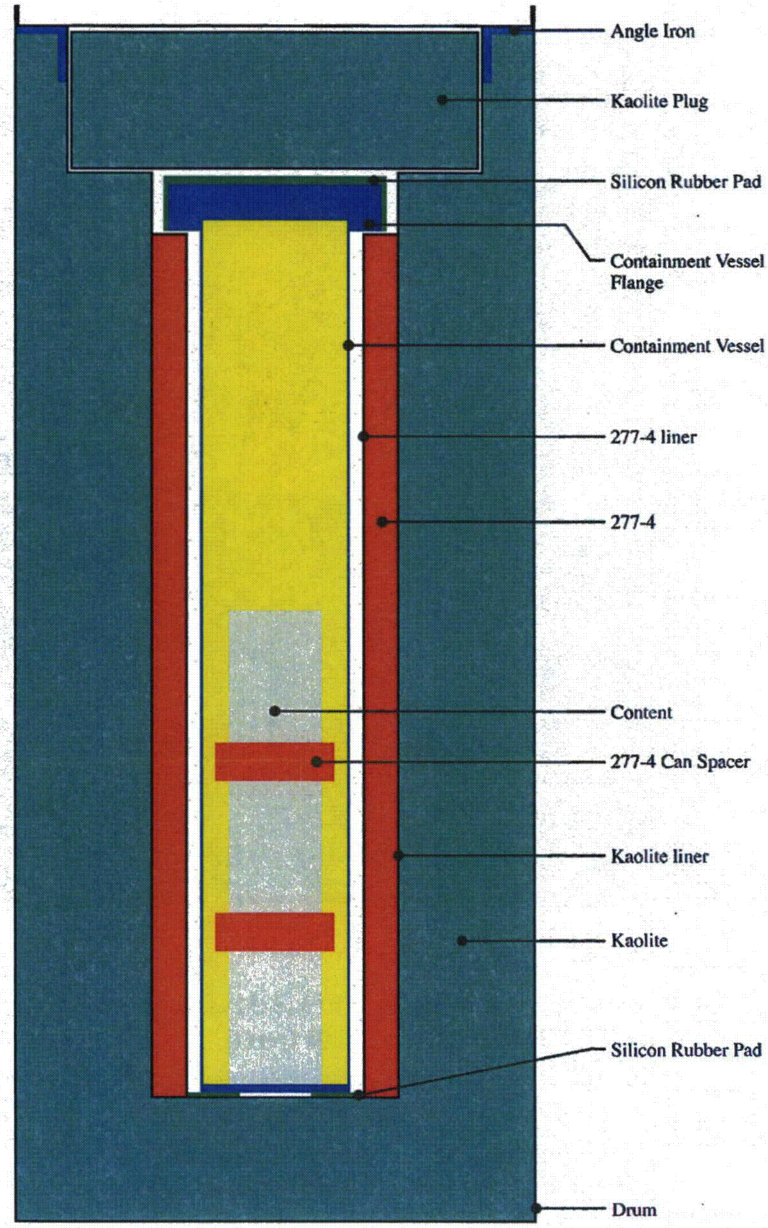
Figures 6.2–6.5 depict vertical section views at various elevations in the package. The dimensions and the material specifications of any element of the ES-3100 single-unit packaging model may be obtained directly from the KENO V.a input listings in Appendix 6.9.7. The vertical segments (KENO V.a geometry unit numbers) are denoted in parenthesis to the right of the dimensions. The dimensions are given in units of centimeters. Material specification data for the single-unit packaging calculation model are provided in Sect. 6.3.2.

The single-unit calculation model is evaluated as both a bare system (i.e., with a vacuum boundary condition) and as a reflected system with 30.48 cm (1 ft) of water surrounding the package for effectively infinite water reflection.

6.3.1.2 Infinite and finite array packaging calculation models

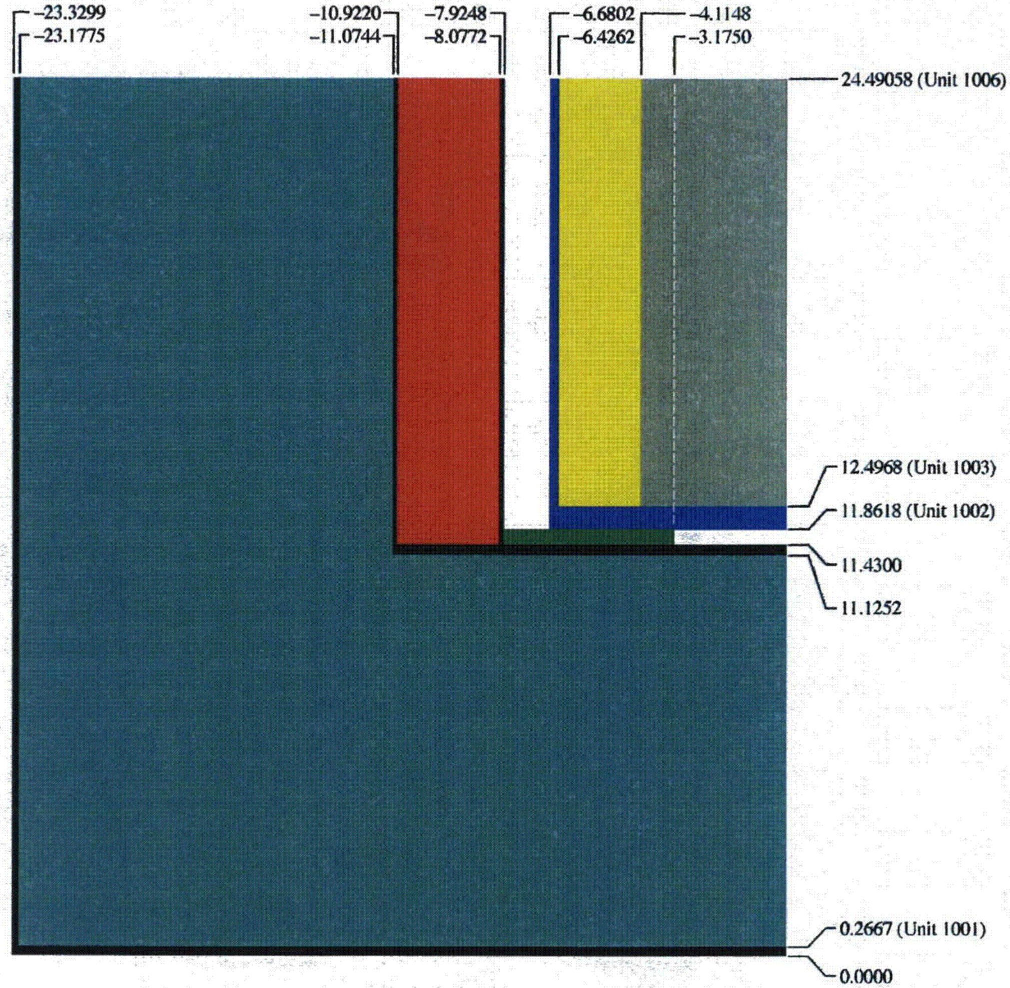
Array packaging calculation models like the single-unit packaging calculation model are comprised of geometry, material compositions, and boundary conditions. Figures 6.6–6.10 depict section views of the geometry model used to evaluate an array of ES-3100 packages. The array geometry model incorporates a 7.0% reduction in the inside diameter of the ES-3100 drum. This reduction of the drum's inside diameter produces an array density equivalent to drums in a tightly packed, triangular pitch configuration. (O'Dell and Schlesser 1991) Since all array calculations in this evaluation use a square pitch package configuration, using the 7.0% reduction in diameter of uniform-shaped packages avoids the use of a nonconservative lattice arrangement in the array analysis.

As seen in Drawings M2E801580A002 and M2E801580A003 (Appendix 1.4.8), a 7.0% reduction of the inside diameter of the ES-3100 drum affects the masses of the drum, the modified 2 × 2 × ¼ in. angle (angle iron), and the Kaolite refractory material. Also, a 7.0% reduction in diameter affects the mass of interstitial water between the ES-3100 packages of a tightly packed array. To maintain the correct mass of these materials in the array packaging calculation model, the drum's outside diameter, the density of the steel in the angle iron, the density of Kaolite, and the density of interstitial water are modified. Excluding modifications required to simulate a triangular pitch and minor variations in the ES-3100 geometry (i.e., radii of curvature, beveled edges, nuts and bolts), the array geometry model is an accurate model of the ES-3100 geometry.



YGG 06-0182

Fig. 6.1. R/Z section view of ES-3100 single-unit packaging model.



YGG 06-0183

Fig. 6.2. R/Z section view at bottom of ES-3100 single-unit packaging showing KENO V.a geometry units 1001–1003, and 1006 (partial). Dimensions are in centimeters.

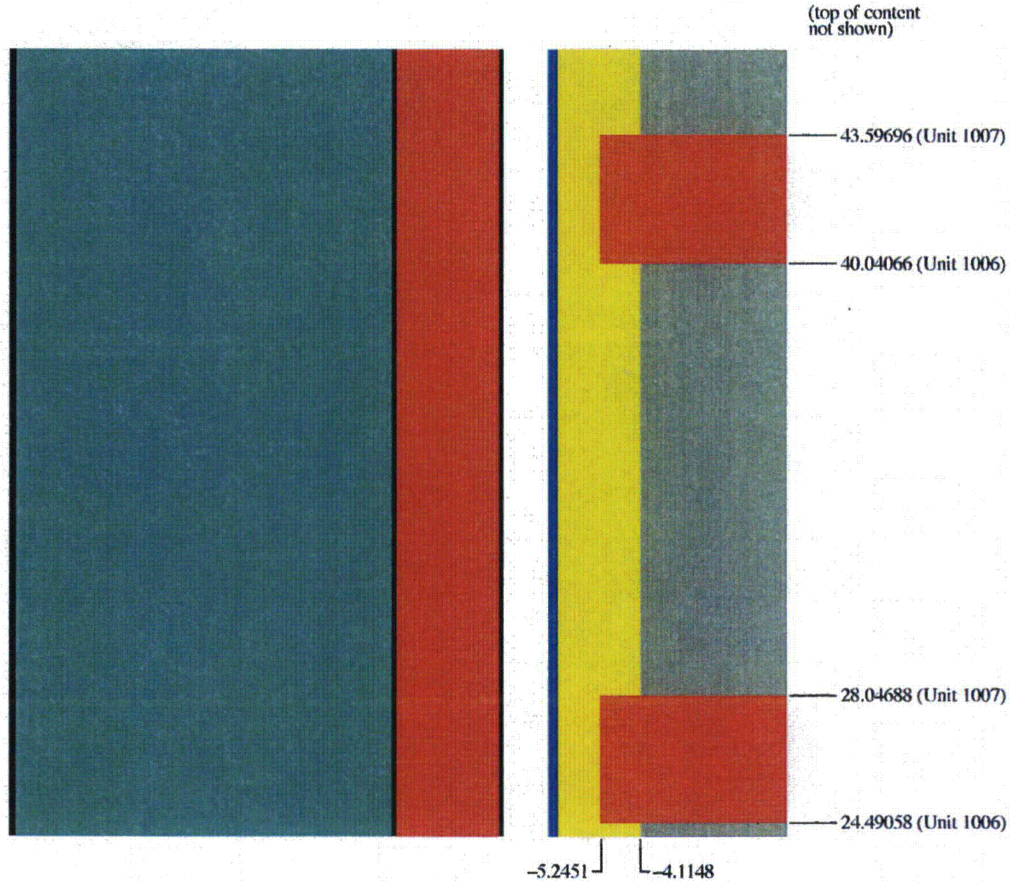


Fig. 6.3. R/Z section view at center of the ES-3100 single-unit packaging showing KENO V.a geometry units 1006 (partial), 1007, and 1008 (partial). Dimensions are in centimeters.

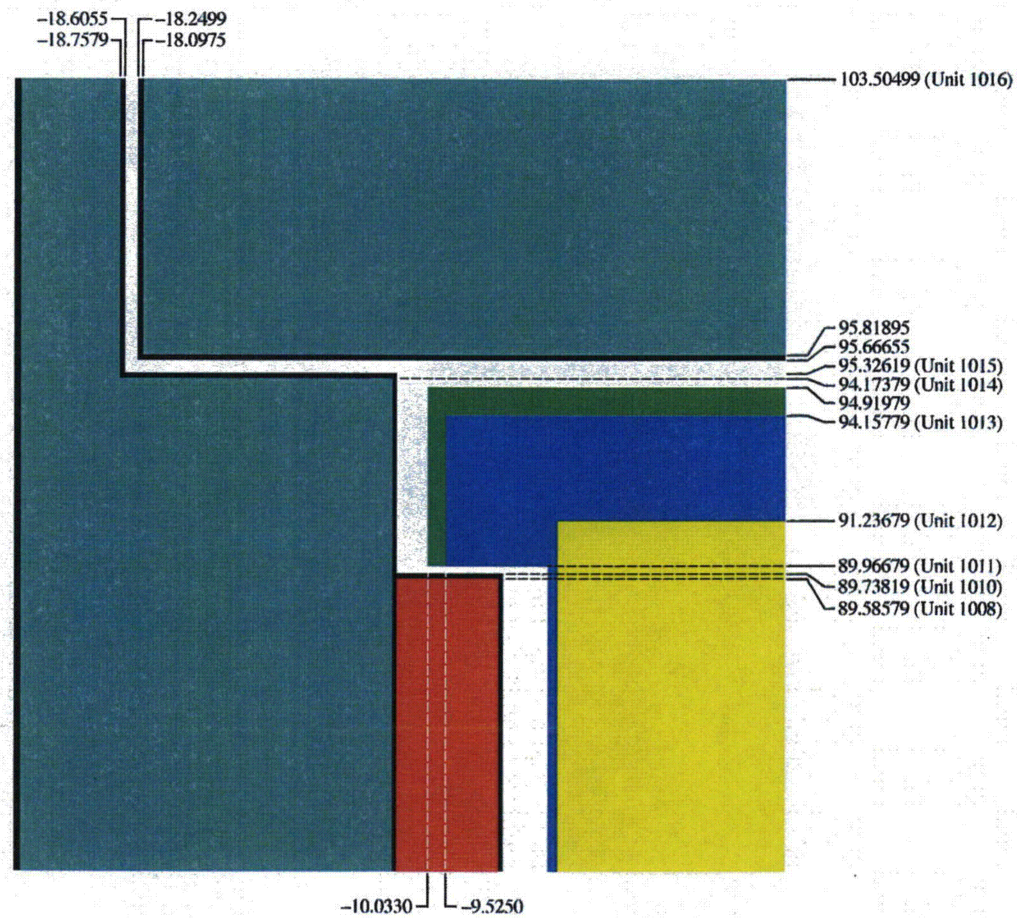
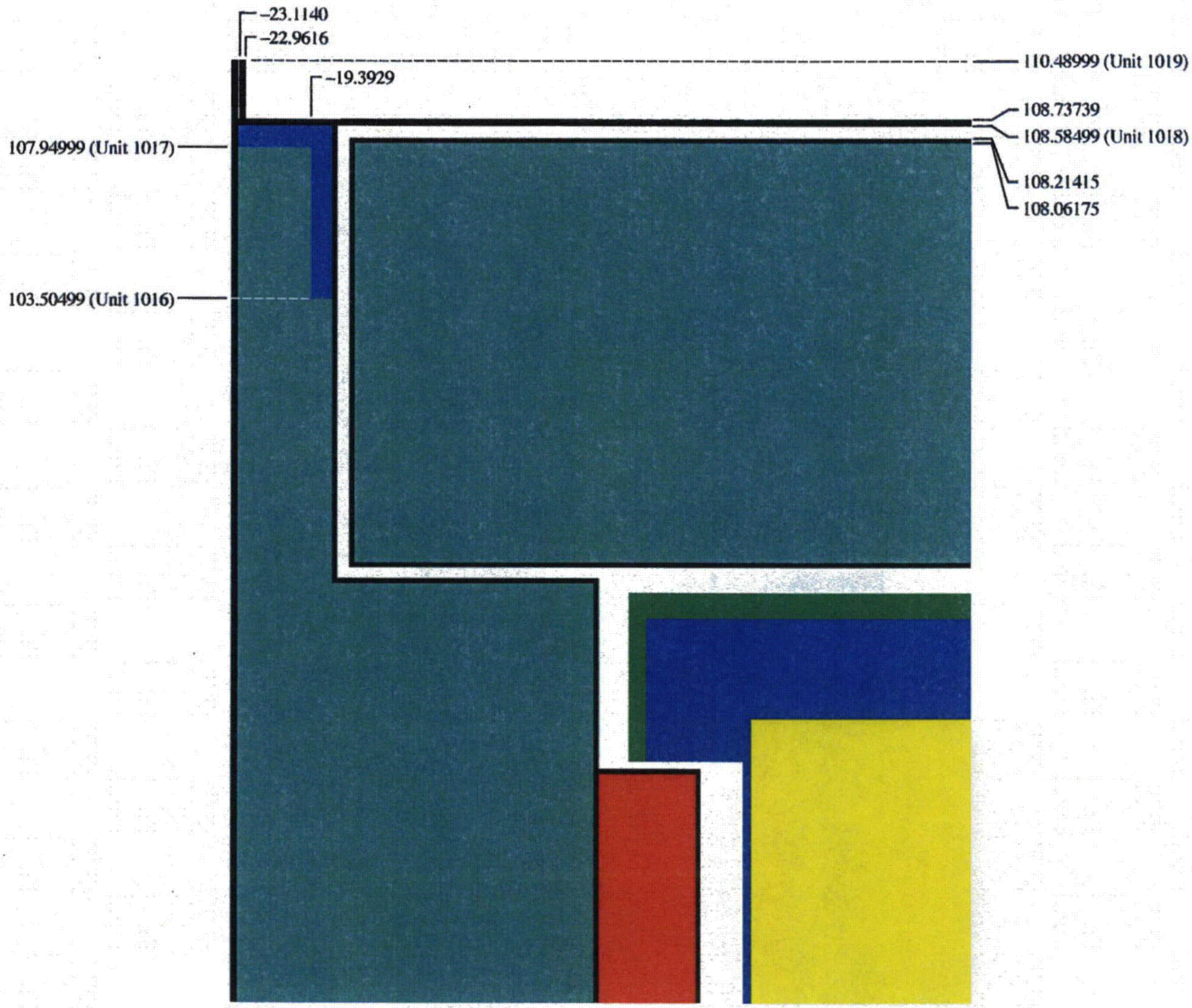
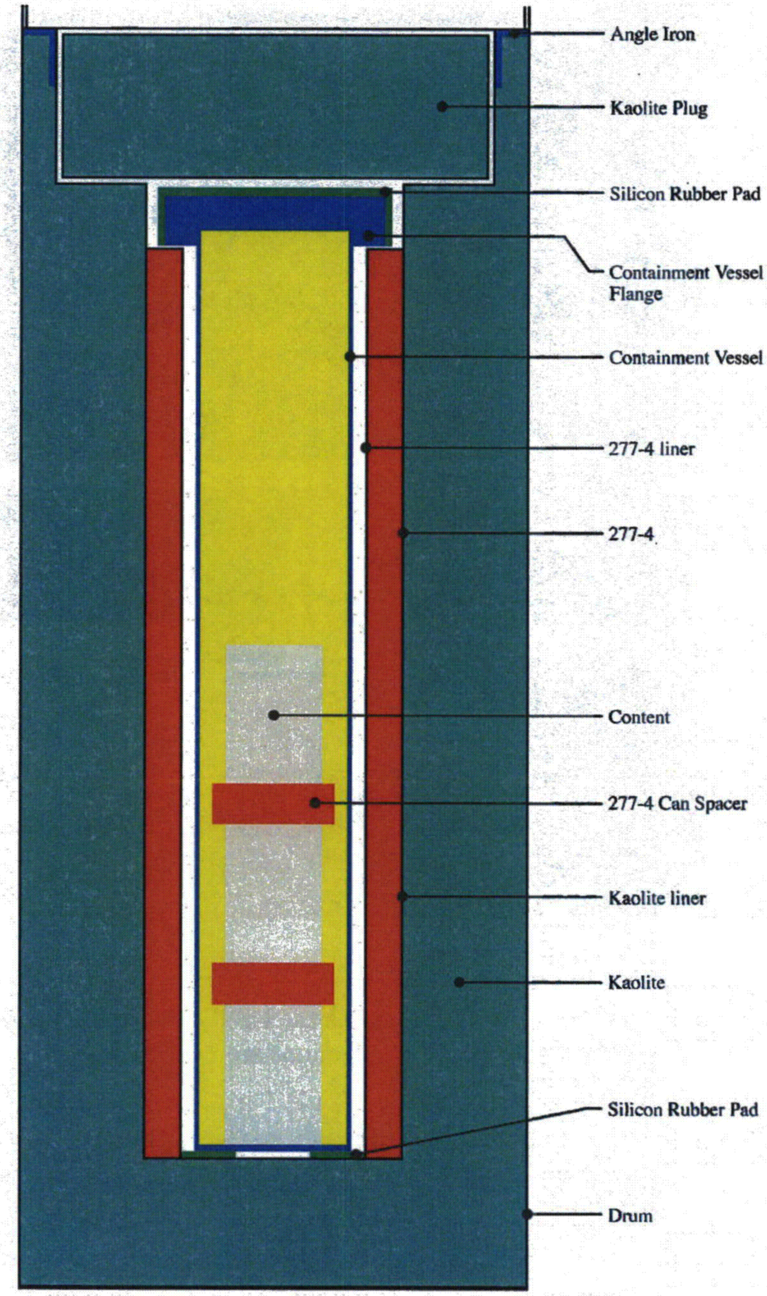


Fig. 6.4. R/Z section view of near top of the ES-3100 single-unit packaging showing KENO V.a geometry units 1008 (partial) and 1010–1016. Dimensions are in centimeters.



YGG 06-0186

Fig. 6.5. R/Z section view at the top of the ES-3100 single-unit packaging showing KENO V.a geometry units 1016–1019. Dimensions are in centimeters.



YGG 06-0187

Fig. 6.6. R/Z section view of ES-3100 array packaging model with a 7% reduction in the drum's inner diameter.

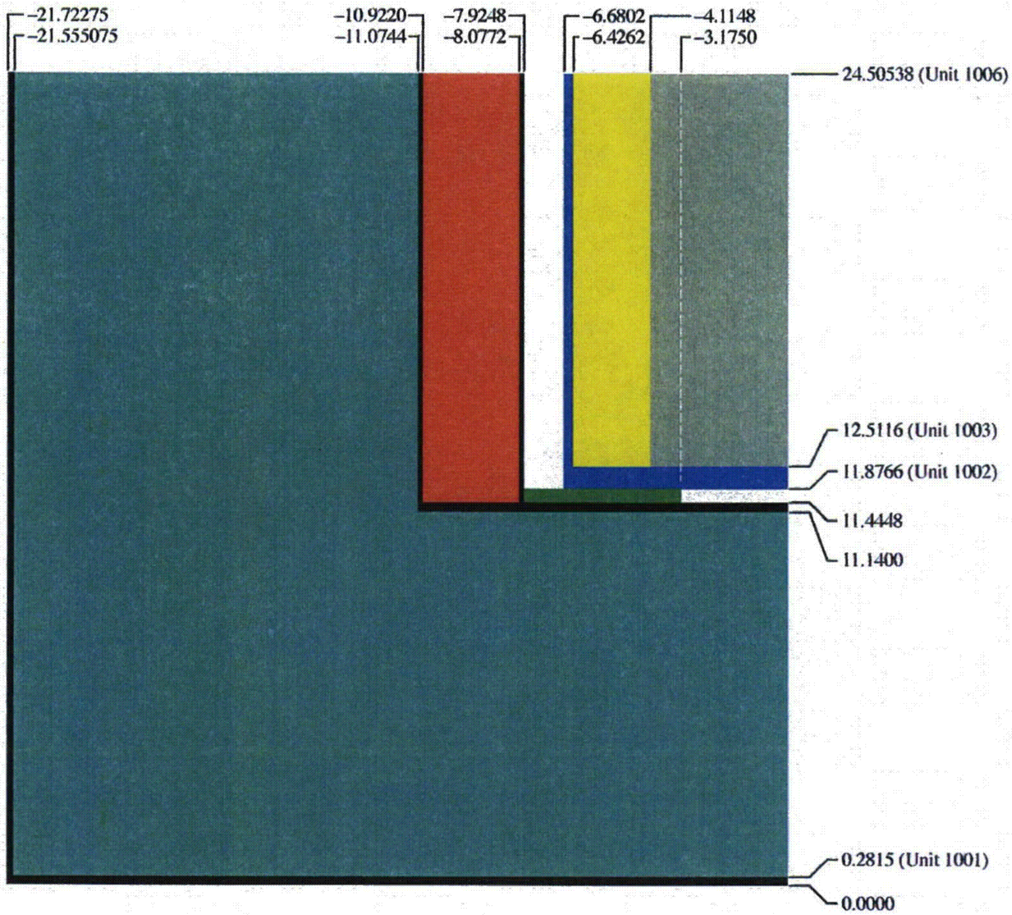


Fig. 6.7. R/Z section view at the bottom of the ES-3100 array packaging showing KENO V.a geometry units 1001–1003, and 1006 (partial). Dimensions are in centimeters.

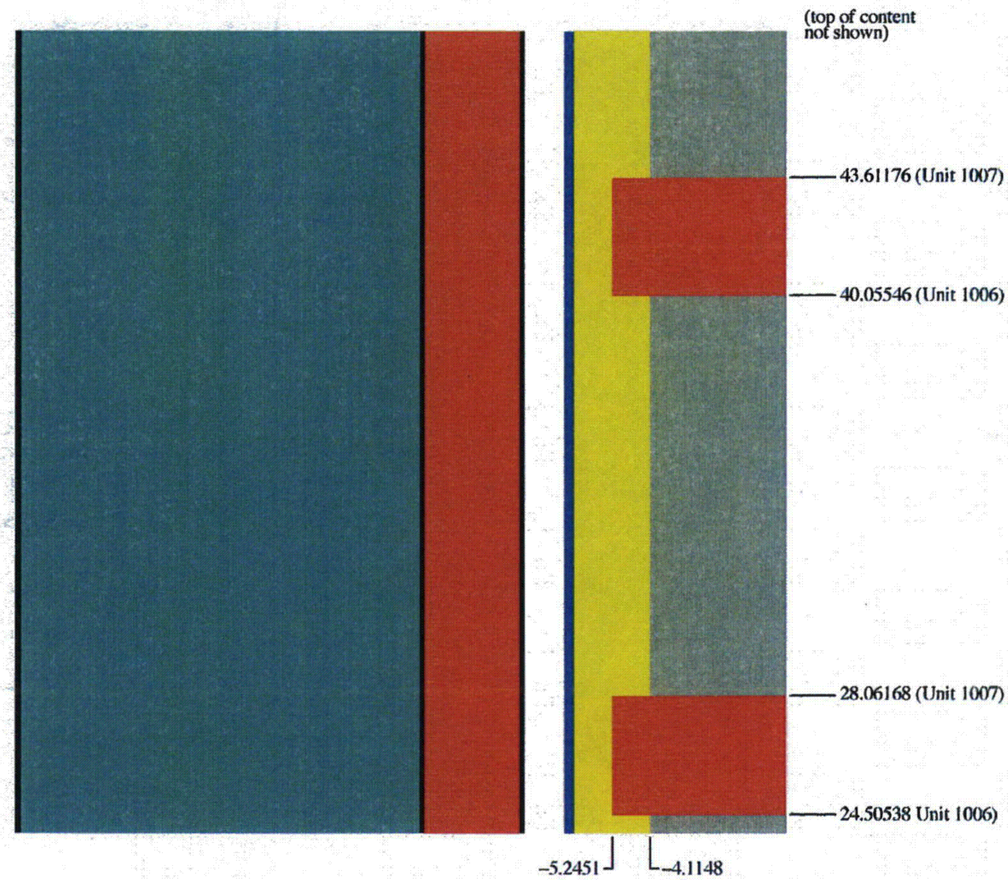
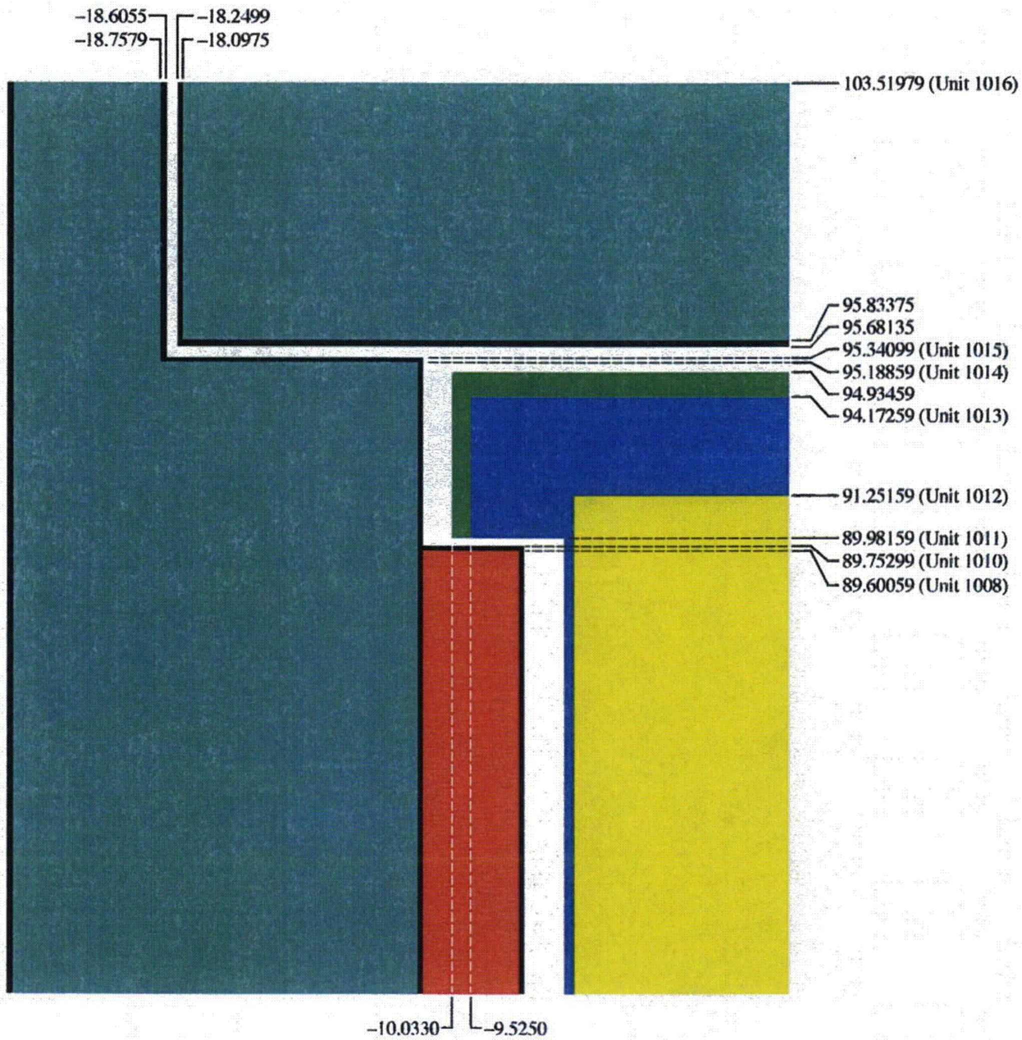


Fig. 6.8. R/Z section view at the center of the ES-3100 array packaging showing KENO V.a geometry units 1006 (partial), 1007, and 1008 (partial). Dimensions are in centimeters.



YGG 06-0190

Fig. 6.9. R/Z section view of near top of the ES-3100 array packaging showing KENO V.a geometry units 1008 (partial) and 1010–1016. Dimensions are in centimeters.

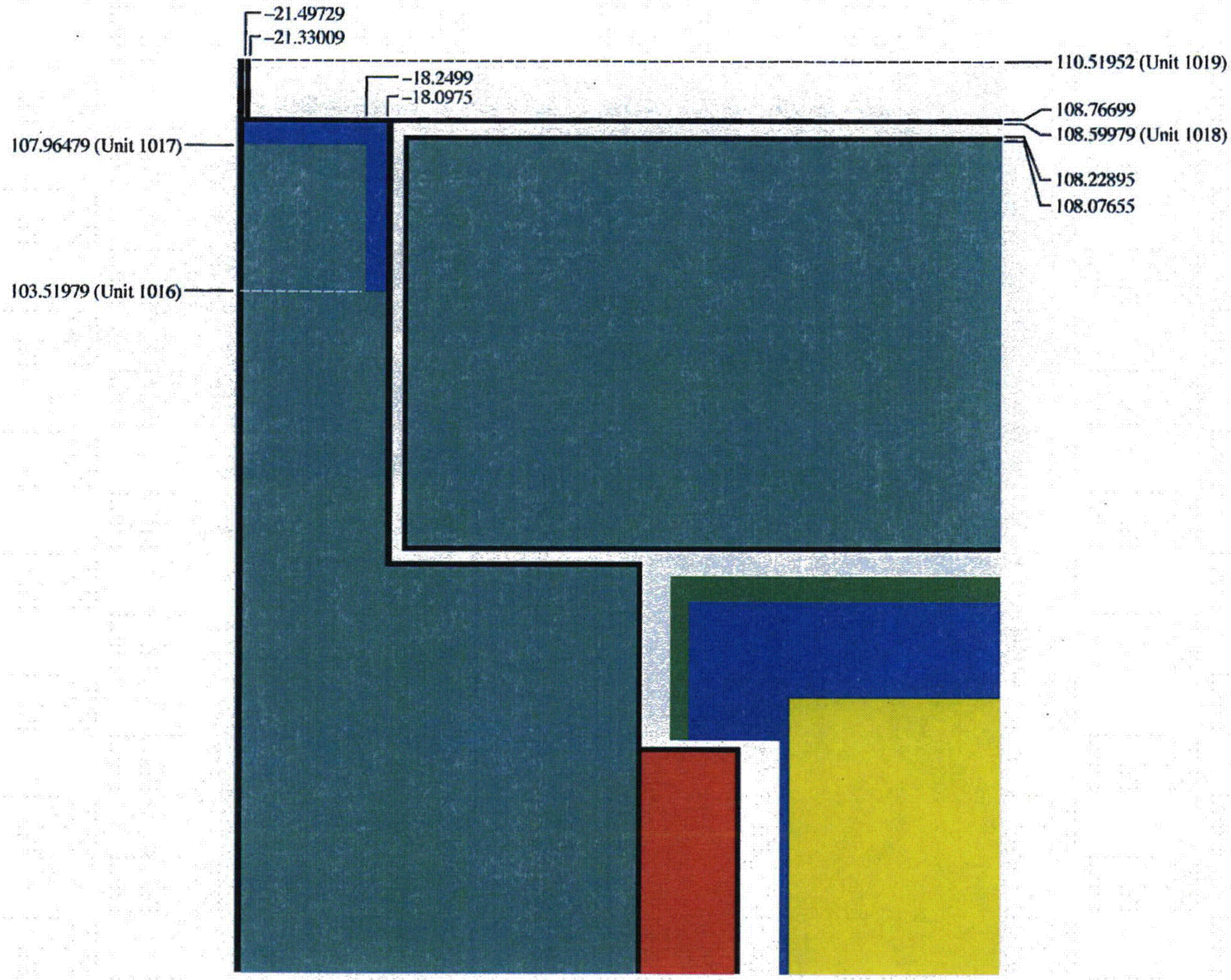


Fig. 6.10. R/Z section view at the top of the ES-3100 array packaging showing KENO V.a geometry units 1016–1019. Dimensions are in centimeters.

In the array geometry model, the drum outside diameter is 43.444550 cm (17.104154 in.) (21.722275-cm radius used as the input value), and the drum inside diameter is 43.11015 cm (16.972500 in.) (21.555075-cm radius). This inside diameter corresponds to 93% of the inside diameter of the ES-3100 drum, and the drum outer diameter is modified to maintain drum mass. The angle iron outside diameter is identical to the drum inside diameter. To maintain mass, the angle iron steel, the liner outer-cavity Kaolite, and interstitial water densities are modified by factors of 1.25705, 1.22252, and 1.16235, respectively. Excluding the modifications to drum and angle iron dimensions, the dimensions of any element of the ES-3100 array geometry model may be obtained directly from Appendix 6.9.7. Material specification data from the array calculation models are provided in Sect. 6.3.2.

O'Dell and Schlesser equate modeling triangular arrays with square pitch arrays, provided that the outer dimension of the array is reduced by a factor of 0.9306 and that the mass of materials is maintained. The actual reduction factors (ratio of the reduced radii to the single unit radii) for the Kaolite of the body weldment and for the angle iron is slightly <0.9306. Given that the array packages are closer than required by O'Dell and Schlesser, the array models are conservative with respect to their reduced radii.

The finite array calculation model is evaluated with arrays of packages stacked in 13×13×6, 9×9×4, 7×7×3, and 5×5×2 arrangements where the arrays are surrounded with 30 cm (~1 ft) of water as a boundary condition representative of full water reflection. These arrays are nearly cubic in shape for optimum reactivity of the array, thus eliminating the need for placing limitations on array configurations in terms of stack height, width, and depth.

The KENO V.a users manual (SCALE, Vol. 2, Sect. F11) describes the input statements for an ETP model of cylinders in a close-pack array. This technique is adapted for modeling the ES-3100 in small compact array configurations. The ETP geometry model is much more complicated than the reduced-diameter approximation model described previously, but it yields smaller package configurations with correspondingly higher CSI values than the 7×7×3 and 5×5×2 package arrangements. Use of the ETP geometry specification allows for constructing compact package arrangements of 27×3 (27 packages in the horizontal plane stacked 3 packages high) and 16×3 packages.

6.3.1.3 Calculation models for damaged packages

Sections 2, 3, and 4 of this report address the overall integrity of the Model ES-3100 package in tests for NCT and HAC. (ORNL/NTRC-013) From a dimensional viewpoint, the only physical changes of interest for criticality safety occurred as a result of the 9-m (30-ft) drop test, the crush test, and the 1-m (40-in.) puncture test. There was significant crushing of the drum rim at the point of impact from the 9-m (30-ft) drop test and an indentation on the drum side from the 1-m (40-in.) puncture test. Even though significant crushing of the drum mid-section and bottom occurs, the effective center-to-center spacing of the contents actually increases under HAC. Selective rearrangement of alternating packages would be required to achieve a more compact array; however, this event is not credible.

The deformation of the outer diameters of the drum as predicted by finite element analysis is presented in Table 6.3. These diameters measured along the 90–270° and 0–180° axes are specified at five node points located along the vertical axis of an upright package. These node points are designated in a downward direction from the top of the drum as "UR," "MUR," "MR," "MLR," and "LR." Equivalent circular diameters for representing the deformed drum in the calculation models for the HAC study (Appendix 6.9.2) are based on the assumption that the drum cross section at each deformation point is ellipsoidal.

The package water content in the void spaces external to the containment vessel and the interstitial spaces between drums is varied in the calculation model while maintaining the Kaolite in the dry condition. At a low moisture content where neutronic interaction between packages of an array is greatest, there is no statistically significant difference in the calculated neutron multiplication factor for package models based on the "MR," "MLR," and "LR" node points. At high moisture content where a statistically significant difference in the calculated neutron multiplication factor occurs, the packages of an array are nearly isolated. The slight increase in k_{eff} at the smaller diameters is not numerically significant. A calculation model based on the "MLR" node point is used to represent HAC.

As concluded in Sect. 6.9.2.3, the criticality analysis of an array of HAC packages based on the "MLR" package model (diameter = 17.20 in.) in rectangular pitch bounds the analysis of damaged packages (diameter = 17.26 in.) in close-pack array configurations. Considering both the irregular shape of the deformed drums and the fact that overall (maximum) dimensions rather than a mean or minimum dimension for a damaged package would establish array spacing—this assumption is a reasonable one, and the model is conservative.

The neutron poison, Kaolite refractory material, and stainless-steel components of the ES-3100 package were not significantly damaged during thermal testing. The principal material change of consequence that occurred during the thermal test was the loss of volatile material (Sect. 2.7.4). No loss of volatile material other than steam from the Kaolite was experienced during prototype testing of the Model ES-3100 package.

Table 6.3. Deformation of 18.37-in.-diam ES-3100 drum projected by finite element analysis Case "3100 RUN1HL Lower Bound Kaolite May 2004"

Deformation point	FEA node	Diameter at 90° (in.)	Diameter at 180° (in.)	Equivalent circular diameter (in.)
UR	098194	20.02	15.60	17.67
MUR	100238	20.74	15.07	17.68
MR	101589	20.74	14.18	17.15
MLR	103012	22.00	13.44	17.20
LR	105786	20.92	12.92	16.44

Physical damage is significant in terms of criticality safety when the amount of volatile hydrogenous material available for interstitial moderation is reduced. Conversely, the package could be saturated with water during water immersion conditions. Both possibilities affect only the amount of interstitial moderation. The representation of the changes to material composition from temperature extremes and water intrusion is addressed in Sect. 6.3.2.

In the case of broken metal content, the ETP geometry specification using the "MLR" package model (diameter = 17.20 in.) produces a 16×3 package configuration having a correspondingly higher CSI value than a 5×5×2 package arrangement. This allows for a greater fissile payload.

6.3.1.4 Calculation models for catastrophically-damaged packages (air-transport)

It is possible for accidents to be substantially more severe in the air transport mode than in the surface transport mode. Thus, the performance requirements for packages designed to be transported by air are more stringent. Regulatory Guide 7.9 states that the criticality evaluation should evaluate a single package under the expanded accident conditions specific in 10 CFR 71.55(f).

For a fissile material package designed to be transported by air, 10 CFR 71.55(f) requires the criticality evaluation demonstrate that the package be subcritical, assuming no water inleakage and reflection by 20 cm (7.9 in.) of water, when subjected to the sequential application of the HAC free drop and crush tests of 10 CFR 71.73(c)(1 and 2) and the modified puncture and thermal tests of 10 CFR 71.55(f)(1)(iii and iv).

Physical testing of Type-B fissile material packages transported by air was not conducted for the ES-3100. In lieu of testing, criticality calculations are performed using calculation models that conform to the basic requirements of 10 CFR 71.55(f) regarding both the reflection of the package by 20 cm of water and the absence of water inleakage. In addition, the air-transport calculation models incorporate features exhibiting the worst-case assumptions made regarding the geometric arrangement of the packaging and package contents. No credit is taken for the geometric form of the package content. Spherical geometry is used to represent the ultimate configuration of the ES-3100 package which undergoes catastrophic destruction.

The criticality analysis for air transport does not credit the ES-3100 for maintaining the geometry configuration of the packaged content. Fissile material is configured spherical in shape for optimization of neutron multiplication. Also, the criticality analysis does not credit the ES-3100 packaging for structural integrity of the containment and the confinement. The reactivity "enhancing" materials of the packaging and of the accident environment may be interspersed with the fissile material content. However, selective removal of the neutron absorbing material of the package (i.e., Kaolite and stainless steel) accompanied by retention of the moderating constituents of the neutron absorbing material (i.e., bound water in the Kaolite) is not a credible condition. Moreover, a calculation model where the fissile material is homogenized with moderating material, is only an appropriate representation when the fissile material content being shipped consists of numerous solid pieces uranium ("broken metal"), oxide, or crystals. Homogenization of fissile material with moderating material is not performed with exclusive preference given to specific constituents of the moderating material.

A set of six calculation models are constructed in order to determine the worst case or most reactive configuration with consideration given to the efficiency of moderators, loss of neutron absorbers, rearrangement of packaging components and contents, geometry changes, and temperature effects. The fissile material component is dry in each case.

Model 1 (Fig. 6.11) represents the fissile material content configured into a spherical core with a 20.0-cm thick water reflector. All ES-3100 packaging material (Kaolite, stainless steel, 277-4 neutron poison, etc.) is excluded from Model 1. The fissile material content considered in the calculation model consists of a specific material form: solid HEU metal (7 – 10 kg ^{235}U); TRIGA fuel (10 kg $UZrH_x$ having 921 g ^{235}U); or broken HEU metal ($\leq$ 7 kg ^{235}U). The presence of hydrogenous packing material inside the containment vessel is also addressed in this calculation model. 513 g of polyethylene is used to represent a maximum amount of hydrogenous packing material in the ES-3100. For 7 kg ^{235}U HEU homogenized with 513 g of polyethylene, the radius of the core is 6.0547 cm. The parametric variation of both the ^{235}U mass and the enrichment in Model 1 is used to establish a reference point for the evaluation neutron reflection and absorption effects on k_{eff} provided by the stainless steel and Kaolite packaging material, addressed separately in Models 2 and 3.

Model 2 (Fig. 6.12) represents a spherical core of Model 1 blanketed by a variable-thickness shell of stainless steel, and a 20.0 cm thick water reflector external to the shell. The shell at maximum thickness corresponds to the 66,133 g stainless steel of the containment vessel and the liner and drum assembly. For 7 kg ^{235}U HEU metal, the radius of the core is 6.0547 cm while the outer radius of the stainless steel shell is 13.0264 cm.

Model 3 (Fig. 6.13) is configured the same as Model 2 with the exception that the variable-thickness shell is composed of the Kaolite instead of stainless steel. At maximum shell thickness, the 128,034 g of Kaolite corresponds to the mass of water-saturated Kaolite inside the drum body weldment and the drum top plug. The 76,819 g of saturation water corresponds to the amount of water in the Kaolite cast-slurry prior to baking. For 7 kg ^{235}U HEU metal, the radius of the core is 6.0547 cm while the outer radius of the Kaolite shell is 31.0814 cm. The parametric variation of shell thickness in Models 2 and 3 is performed for the purpose of evaluating the effect on k_{eff} of neutron reflection and absorption provided by each packaging material.

Models 1–3 are applicable for establishing loading limits for air transport packages having a single piece or several pieces of solid HEU metal where the integrity of the content can be established based on package tests or dynamic impact simulations per Type-C test criteria. Otherwise, the criticality evaluation of additional accident models is required for those situations where: (1) data is not available or is inadequate for discounting fragmentation of the fissile content in the air transport accident, or (2) the fissile material is in a physical form (crystals, oxide, or multiple pieces of solids) having the potential to blend with the other material of the package in the air transport accident.

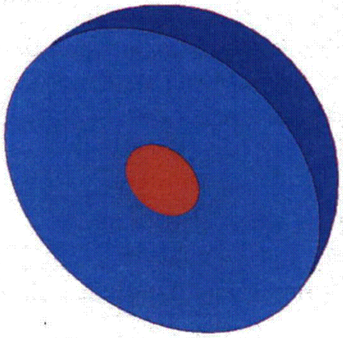


Fig. 6.11. Model 1. 7 kg ^{235}U (red) reflected by 20 cm of water (blue).

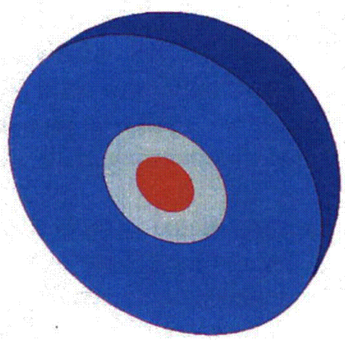


Fig. 6.12. Model 2. 7 kg ^{235}U (red) blanketed by a stainless-steel shell (gray), reflected by 20 cm of water (blue).

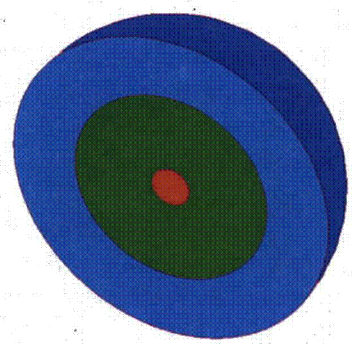


Fig. 6.13. Model 3. 7 kg ^{235}U (red) blanketed by a Kaolite shell (green), reflected by 20 cm of water (blue).

Model 4 (Fig. 6.14) depicts a spherical core of broken metal homogenized with water-saturated Kaolite and hydrogenous packing materials inside the containment vessel represented by 513 g of polyethylene. A 20-cm thick water reflector blankets the spherical core. All other ES-3100 packaging material (stainless steel, 277-4 neutron poison, etc.) is excluded from Model 4. For 25 kg of HEU (5 kg of ^{235}U), the radius of the homogenized core ranges from 31.4499 cm with dry Kaolite to 39.1608 cm with water saturated Kaolite. Model 4 also applies to TRIGA fuel content.

Model 5 (Fig. 6.15) is an extension of Model 3 applied to the case of TRIGA fuel and broken metal contents. Excess water from saturated Kaolite is homogenized with the fissile material in the core. For 25 kg of HEU where 5 kg of ^{235}U is homogenized with 513 g of polyethylene and 7,461 g of excess water from water saturated Kaolite, the radius of the spherical core is 13.0681 cm. The exterior Kaolite shell containing 69,357.9 g of water and the 51,214.9 g of vermiculite-cement constituents, has an outer radius of 31.7599 cm. At the extreme condition where the entire 74,614.0 g of excess water excess from the water saturated Kaolite is homogenized with the core, the radius of the spherical core is 26.3485 cm. The exterior shell of dry Kaolite, containing 2204.7 g of bound water and 51,214.9 g of vermiculite cement constituents, has an outer radius of 36.3668 cm. Although an unlimited amount of moderator could be imparted from the environment, optimum moderation is addressed simply by considering the full range of potential moisture from the Kaolite interspersed with pieces of fissile material.

Model 6 (Fig. 6.16) is configured similarly to Model 4 with the exception that only a partial amount of broken metal resides in the homogenized core. The remainder of the fissile material is modeled as a shell external to the spherical core and internal to the 20.0 cm thick water reflector. The amount of fissile material in the homogenized core is varied in 500 g increments from the full amount of Model 5 minus 500 g to a minimum amount of 500 g HEU. In the case of 25 kg of broken metal where 4.5 kg of ^{235}U is homogenized with 513 g of polyethylene and water-saturated Kaolite, the radius of the core is 39.1539 cm while the outer radius of the exterior 0.5 kg shell of ^{235}U is 39.1609 cm. In the case where 0.5 kg of ^{235}U is homogenized with 513 g of polyethylene and water-saturated Kaolite, the radius of core is 39.0992 cm while the outer radius of the exterior 4.5 kg shell of ^{235}U is 39.1609 cm. Model 6 addresses potential redistribution of fissile material consisting of multiple pieces, specifically the broken metal content. The parametric variation of shell thickness in Model 6 is performed for the purpose of evaluating the effect on k_{eff} of moderation efficiency of the core and enhanced neutron multiplication of the shell.

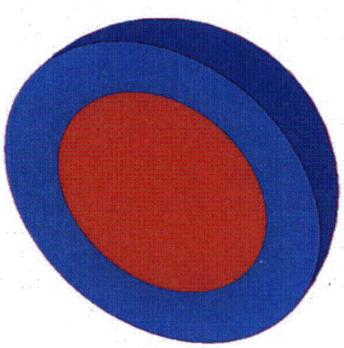


Fig. 6.14. Model 4. Homogenized sphere of 5 kg ^{235}U, polyethylene, and water saturated Kaolite (red), reflected by 20 cm of water (blue).

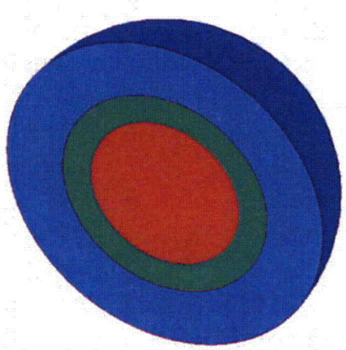


Fig. 6.15. Model 5. Homogenized sphere of 5 kg ^{235}U and excess moisture from Kaolite (red), blanketed by Kaolite shell (lt. green), reflected by 20 cm of water (blue).

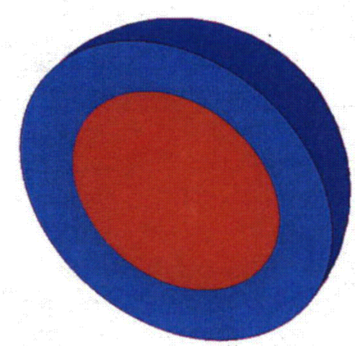


Fig. 6.16. Model 6. Homogenized sphere of 4.5 kg ^{235}U, polyethylene, and water saturated Kaolite (red); blanketed by shell of 0.5 kg ^{235}U (not visible), reflected by 20 cm of water (blue).

Models 1–6 are applicable to air transport of the ES-3100 package given that the physical integrity of the fissile content following the air transport accident can not be established, or that the physical form of fissile material content may be multiple solid pieces or oxide.

6.3.2 Material Properties

Criticality calculation models include the definition of material compositions in the geometry model for both NCT and HAC. The materials which affect nuclear criticality safety and may be present in the ES-3100 package during these conditions are fissile material content (HEU metal, aluminum and molybdenum alloys of uranium, HEU product or skull oxide, UNX crystals, or $UZrH_x$), stainless steel, 277-4 neutron poison, Kaolite, silicon rubber, Teflon or polyethylene, and various amounts of water. Other materials are present in the package (e.g., tin-coated steel convenience cans, nylon bagging, rubber O-rings, and trace impurities). However, these other materials are not present in amounts that will significantly affect the reactivity of the package. Detailed discussion of the package content and packaging materials is contained in Appendix 6.9.3. Specific topics are summarized in this section as required to support the discussion of the criticality calculations presented in Sects. 6.4–6.6.

277-4 neutron poison is designed to maximize the hydrogen content necessary for thermalizing fast neutrons for capture in the boron constituent. 277-4 is a formulation of Thermo Electron Corporation's Cat 277-0, a boron carbide additive, and water. This mixture is cast and dried. "Loss On Drying" (LOD) tests are used to measure the amount of water in the as manufactured 277-4 casting. The as-manufactured neutron poison material at 100 lb/ft^3 and 31.8% LOD has a hydrogen concentration of 3.56 wt % and a natural boron concentration of 4.359 wt %. (DAC-PKG-801624-A001, Table 5)

The testing of 277-4 reveals that the material will dehydrate at elevated temperatures. Test specimens were dried at 250°F for 168 hours to reach the NCT state, and weight measurements were taken. These specimens where subsequently heated to 320°F for 4 hours to reach the HAC state, and weight measurements were again taken. The compositions of 277-4 at NCT and HAC states were derived by adjustment of the formulation specification for measured losses taking into account the statistical variations in the data. Conservation of mass for non-volatiles was observed in the derivation of material specifications based upon testing. Given that hydrogen presence is key to the effectiveness of the neutron poison, conservative material specifications were derived for minimum hydrogen content and minimum material density.

The NCT material specification for 277-4 at minimum density (95.39 lb/ft^3) and minimum hydrogen (25.15% LOD) is given in Table 11 of DAC-PKG-801624-A001. The average density of 277-4 in the ES-3100 package used for NCT calculations in KENO V.a is 1.52797 g/cm^3; the boron constituent is 7.31015e-2 g/cm^3; the residual base is 1.07070 g/cm^3; and the water component is 3.84169e-01 g/cm^3 (Appendix 6.9.3, Table 6.9.3.3-2). The HAC material specification at minimum density (95.32 lb/ft^3) and minimum hydrogen (25.09% LOD) is given in Table 12 of DAC-PKG-801624-A001. The average density of 277-4 used for HAC calculations in KENO V.a is 1.52680 g/cm^3; the boron constituent is 7.31015e-2 g/cm^3; the residual base is 1.07071 g/cm^3; and the water component is 3.82995e-1 g/cm^3 (Appendix 6.9.3, Table 6.9.3.3-3).

A set of 20 canned spacer assemblies as-manufactured has an average weight of 591.55 g. On average, the weight of an empty can with a lid is 86.05 g, and the 277-4 is 505.5 g. On the basis of volumetric measurements, the neutron poison is ~10.075 oz or 297.95 cm^3. The resultant 277-4 density is 1.6966 g/cm^3. The corresponding minimum mass is 473.84 g (given that the 277-4 material inside the spacer can is ≥4.07-in. diam and 1.31-in. thickness). Based on model dimensions of 4.13-in. diam × 1.37-in. height and use of composition data from Table 11 of DAC-PKG-801624-A001 (as-manufactured material at minimum density and hydrogen), the mass of neutron poison in the calculation model of the canned spacer is conservatively modeled as 457.15 g.

The density of 277-4 in the liner inner cavity is further reduced by a factor of 0.966893 because the actual volume of neutron poison is 12,831.4 cm^3 in the package, but 13,270.8 cm^3 is used in the calculation model. Also, guidance in Sect. 6.5.3.2 of NUREG-1609 requires that 75% of elemental boron be used in the material specification. Consequently, the minimum weight of 277-4 in the liner inner cavity of the calculation models is 19.37 kg (42.71 lb) for NCT and 19.36 kg (42.67 lb) for HAC, determined from the input density and the volume of the 277-4 region calculated by KENO V.a.

Kaolite is a lightweight, low thermal conductivity, refractory material that is used in the ES-3100 package between the drum and the inner liner. Kaolite is formed by mixing a dry constituent powder with water, pouring the mixture into a casting form, and curing and baking the cast. For NCT, the average density of Kaolite in the Model ES-3100 package is 0.34438 g/cm^3, and the residual water is 0.01493 g/cm^3 (Appendix 6.9.3, Sect. 6.9.3.4). The nominal weights of Kaolite in the liner outer cavity and in the top plug as determined from the input density and the region volumes calculated by KENO V.a are 49.04 kg (108.1 lb) and 4.37 kg (9.63 lb), respectively.

Kaolite is a nonvolatile refractory material, not significantly damaged by thermal excursions associated with HAC. It is assumed that upon immersion, a cured and dried casting will absorb a quantity of water equal to that required during preparation. This assumption is valid because the casting is fully saturated with water prior to curing and baking, and the casting does not change volume significantly during baking. The maximum water content of Kaolite refractory material is the water content of manufacture (1.5 g water per gram dry Kaolite) equivalent to a density of 0.51655 g/cm^3. For the single package, water saturation in the Kaolite region maximizes the effect of self-reflection in the surrounding package. This condition is assumed in the NCT single package calculations. For an array of packages, neutronic interaction between packages is maximized when the minimum amount of water is present in the Kaolite, which occurs in the as-manufactured condition. A volume fraction of 1.0 in the material specification corresponds to the water-saturated condition, while a volume fraction of 0.0289 corresponds to the dry condition.

The fissile material content packed into the containment vessel is ordinarily dry. Given that a flooded containment vessel is assumed for both NCT and HAC, the concurrent condition where the water-saturated Kaolite is baked dry beyond the as-manufactured condition while the containment vessel remains flooded is considered not credible. Therefore, the as-manufactured condition is assumed for the Kaolite water content in the HAC array analysis.

All other material compositions for HAC are identical to the material compositions for NCT. For the HAC calculation model, the densities for the Kaolite in the body weldment liner inner cavity, the stainless steel of the angle iron, and the drum steel are adjusted to correspond with changes in dimensions. However, the atomic densities do not change in the HAC calculation model because mass is conserved.

The material composition used in the calculation model for the containment vessel, the drum liner, and the drum is Type 304 stainless steel at a density of 7.94 g/cm^3. The density and composition of the 304 stainless steel for the drum is taken from the Standard Composition Library of the Standardized Computer Analysis for Licensing Evaluation (SCALE) code system. The nominal weights of these components, as determined by input density and volume calculated by KENO V.a, are 14.9 kg (32.85 lb) for the containment vessel, 15.67 kg (34.55 lb) for the drum liner, and 25.44 kg (56.1 lb) for the drum.

The amount of hydrogenous packaging material used for packing the content in the containment vessel is an unknown variable. Polyethylene with a density of 0.92 g/cm^3 and a molecular formula of CH_2 has the greatest hydrogen density of potential packing materials. Although polyethylene bags are generally used in packing for contamination control, the exact mass of the polyethylene bags used is unknown. Multiple bags may be used to package the contents and to enclose the convenience cans, or bags may be omitted altogether. Realistically, the number of polyethylene bags used to bag the contents and wrap a convenience can would be six bags at most. Given that each bag weighs ~28 g, this results in a maximum of ~510 g of polyethylene per ES-3100 package. Possibly additional sources of hydrogenous material inside the containment vessel include can pads, vinyl tape, and polyethylene (~85 g) or Teflon (~303 g) bottles.

For the criticality calculations, the sources of hydrogen contained in packing materials are not distinguished from other sources of hydrogen inherent in the fissile material content. Therefore, packing material mass is defined as all sources of hydrogenous packing materials inside the containment vessel plus the actual moisture in the content as constituent (the bonded hydrogen in UNX crystals or impurities in the oxide.) Given content moisture is not to exceed 6 wt %, the amount of hydrogenous material inside the containment vessel is not expected to exceed 2,400 g.

Containment vessel flooding is assumed in calculations performed for the derivation of fissile material loading limits. Even though both the NCT tests under 10 CFR 71.71 and the HAC tests under 10 CFR 71.73 demonstrate that containment is not breached, this simulated condition produces package configurations in the criticality calculations that are more reactive than the actual package configurations. The 7.1–10.1 kg quantities of evaluation water required in both the NCT and HAC criticality calculations bound reasonable amounts of hydrogenous material and inherent moisture of fissile material (primarily HEU oxides) present inside the containment vessel (2,400 g).

HAC alter only the amount of volatile hydrogenous material contained in the package external to the containment vessel (Sect. 2.7.3). High temperatures may reduce the hydrogenous material content of the Kaolite. Conversely, water intrusion may increase the hydrogenous material content in the package. Thus, the package's range of possible volatile hydrogenous material content varies from none, caused by high temperatures, to the maximum possible value caused by water intrusion.

Table 6.4 lists the material, density, atomic or isotopic constituent, and atomic or isotopic weight percent as basic data for materials used in the single-unit packaging and array calculation models in this criticality safety evaluation for NCT and HAC, respectively. Appendix 6.9.3 provides the rationale, justification, or both for using the basic data for computing atomic densities listed in Table 6.4.

6.3.3 Computer Codes and Cross-Section Libraries

The Criticality Safety Analysis Sequences within the SCALE modular code system provide automated, problem-dependent, cross-section processing followed by calculation of the neutron multiplication factor (k_{eff}) for the system being modeled. (SCALE, Vol. 1, Sect. C4) Initiated by "=csas25" appearing on the first line in the user input, the CSAS module runs the CSAS25 control sequence. The cross-section processing functional modules BONAMI (SCALE, Vol. 2, Sect. F1) and NITAWL-II (SCALE, Vol. 2, Sect. F2) are activated, providing resonance-corrected cross sections to the multigroup Monte Carlo functional module, KENO V.a (SCALE, Vol. 2, Sect. F11). Using the processed cross sections, KENO V.a calculates the k_{eff} of three-dimensional system models. The geometric modeling capabilities available in KENO V.a, coupled with the automated cross-section processing, allow complex, three-dimensional systems to be easily analyzed.

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm³)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
Fissile material contents							
Uranium (solid metal)	1	18.81109	1.0	^{235}U	235.0441	1.00	4.81970E-02
				^{238}U	238.0510	0.00	0.0000E+00
Uranium (solid metal)	1	18.82298	1.0	^{235}U	235.0441	0.95	4.58156E-02
				^{238}U	238.0510	0.05	2.38089E-03
Uranium (solid metal)	1	18.83488	1.0	^{235}U	235.0441	0.90	4.34317E-02
				^{238}U	238.0510	0.10	4.76479E-03
Uranium (solid metal)	1	18.85873	1.0	^{235}U	235.0441	0.80	3.86548E-02
				^{238}U	238.0510	0.20	9.54164E-03
Uranium (solid metal)	1	18.88264	1.0	^{235}U	235.0441	0.70	3.38659E-02
				^{238}U	238.0510	0.30	1.43306E-02
Uranium (solid metal)	1	18.90661	1.0	^{235}U	235.0441	0.60	2.90647E-02
				^{238}U	238.0510	0.40	1.91317E-02
Uranium (solid metal)	1	18.95474	1.0	^{235}U	235.0441	0.40	1.94258E-02
				^{238}U	238.0510	0.60	2.87707E-02
Uranium (solid metal)	1	19.00554	1.0	^{235}U	235.0441	0.19	9.25199E-03
				^{238}U	238.0510	0.81	3.89445E-02
Uranium oxide (UO_2)	1	6.94256	1.0	H	1.0077	0.6490	2.69208E-02
				O	15.9904	16.4340	4.29699E-02
				^{235}U	235.0441	82.9170	1.47490E-02
Uranium oxide (U_3O_8)	1	6.75167	1.0	H	1.0077	0.3510	1.41552E-02
				O	15.9904	17.6620	4.49101E-02
				^{235}U	235.0441	81.9870	1.41826E-02
Uranium oxide (UO_3)	1	6.64270	1.0	H	1.0077	0.1730	6.86795E-03
				O	15.9904	18.0650	4.51925E-02
				^{235}U	235.0441	81.7620	1.39155E-02
Skull oxide (UO_3+graphite)	1	3.34850	1.0	H	1.0077	1.9000	3.80225E-02
				C	12.0001	11.9173	2.00261E-02
				O	15.9904	25.9923	3.27786E-02
				^{235}U	235.0441	56.0976	4.81282E-03
				^{238}U	238.0510	4.0929	3.46713E-04
UNH crystals (24000.0 g) [$UO_2(NO_3)_2+6H_2O$]	1	2.50804	1.0	H	1.0077	2.9770	4.46274E-02
				N	14.0033	5.2570	5.66985E-03
				O	15.9904	47.6480	4.50055E-02
				^{235}U	235.0441	44.1180	2.83495E-03
UNH crystals (9000.0 g) [$UO_2(NO_3)_2+6H_2O$]	1	1.56439	1.0	H	1.0077	6.2525	5.84562E-02
				N	14.0033	3.1604	2.12620E-03
				O	15.9904	64.0630	3.77431E-02
				^{235}U	235.0441	26.5240	1.06311E-03
UNH crystals (8000.0 g)	1	1.50149	1.0	H	1.0077	6.6172	5.93782E-02
				N	14.0033	2.9270	1.89000E-03

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm³)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
$[UO_2(NO_3)_2+6H_2O]$				O	15.9904	65.8907	3.72591E-02
				^{235}U	235.0441	24.5651	9.45006E-04
UNH crystals	1	1.06111	1.0	H	1.0077	10.3811	6.58313E-02
(1000.1 g)				N	14.0033	0.5178	2.36275E-04
$[UO_2(NO_3)_2+6H_2O]$				O	15.9904	84.7556	3.38697E-02
				^{235}U	235.0441	4.3455	9.45006E-04
~20% enriched UZrH	1	8.65974	1.0	H	1.0078	0.9554	4.94386E-02
(3466.7 g)				Zr	91.2196	54.0447	3.08972E-02
				^{235}U	235.0441	8.8558	1.96487E-03
				^{238}U	238.0510	36.1441	7.91816E-03
~70% enriched UZrH	1	5.70132	1.0	H	1.0078	1.5894	5.41480E-02
(2282.4 g)				Zr	91.2196	89.9107	3.38410E-02
				^{235}U	235.0441	5.9588	8.67175E-04
				^{238}U	238.0510	2.5413	3.69825E-04
~20% enr. U_3O_8-Al	1	3.52015	1.0	O	15.9904	5.2477	6.95687E-03
(12254.7 g)				Al	26.9818	65.5398	5.14922E-02
				^{235}U	235.0441	5.8425	5.26934E-04
				^{238}U	238.0510	23.3700	2.08111E-03
~70% enr. U_3O_8-Al	1	3.52095	1.0	O	15.9904	5.2800	7.00127E-03
(1995.7 g)				Al	26.9818	65.5141	5.14837E-02
				^{235}U	235.0441	20.4441	1.84427E-03
				^{238}U	238.0510	8.7618	7.80418E-04
~93% enr. U_3O_8-Al	1	3.52132	1.0	O	15.9904	5.2950	7.02187E-03
(1505.0 g)				Al	26.9818	65.5022	5.14798E-02
				^{235}U	235.0441	27.2170	2.45552E-03
				^{238}U	238.0510	1.9858	1.76895E-04
~9.9% enr. UO_2-Mg	1	3.95984	1.0	O	15.9904	6.4032	9.54899E-03
in Al clad				Mg	24.3048	7.6175	7.47379E-03
(15231.1 g)				Al	26.9818	38.3916	3.39305E-02
				^{235}U	235.0441	4.7062	4.77464E-04
				^{238}U	238.0510	42.8815	4.29560E-03
20% enr. UO_2-Mg	1	3.96017	1.0	O	15.9904	6.4110	9.56133E-03
in Al clad				Mg	24.3048	7.6169	7.47379E-03
(7523.6 g)				Al	26.9818	38.3885	3.39305E-02
				^{235}U	235.0441	9.5167	9.65604E-04
				^{238}U	238.0510	38.0670	3.81363E-03
~93% enr. UO_2-Mg	1	3.96254	1.0	O	15.9904	6.4670	6.34806E-03
in Al clad				Mg	24.3048	7.6123	4.91617E-03
(1472.1 g)				Al	26.9818	38.3655	2.23190E-02
				^{235}U	235.0441	44.3215	2.95985E-03
				^{238}U	238.0510	3.2338	2.13227E-04

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm³)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
Single-unit calculation models (NCT)							
277-4	2	1.50970	1.0	H	1.0077	2.84745	2.56909E-02
canned spacer				^{10}B	10.0130	0.65907	5.98418E-04
Table 11 data: [b]				^{11}B	11.0096	2.97264	2.45475E-03
as-manufactured,				C	12.0001	1.36094	1.03107E-03
minimum density and				N	14.0033	0.01000	6.49252E-06
hydrogen				O	15.9904	53.39650	3.03592E-02
				Na	22.9895	0.08326	3.29274E-05
				Mg	24.3048	0.23957	8.96150E-05
				Al	26.9818	27.69370	9.33137E-03
				Si	28.0853	1.74979	5.66427E-04
				S	32.0634	0.21865	6.19983E-05
				Ca	40.0803	8.39267	1.90373E-03
				Fe	55.8447	0.37575	6.11714E-05
Water-flooded CV	3	0.9982	1.0	H	1.0077	11.1909	6.67536E-02
				O	15.9904	88.8091	3.33856E-02
SS304	8	7.9400	1.0	C	12.0001	0.0800	3.18772E-04
containment				Si	28.0853	1.0000	1.70252E-03
vessel body				P	30.9741	0.0450	6.94680E-05
16.60 lb				Cr	51.9957	19.0000	1.74726E-02
but 15.74 lb used				Mn	54.9379	2.0000	1.74071E-03
				Fe	55.8447	68.3750	5.85446E-02
				Ni	58.6872	9.5000	7.74020E-03
SS304	9	7.9400	0.97267	C	12.0001	0.0800	3.10060E-04
containment vessel				Si	28.0853	1.0000	1.65599E-03
lower flange				P	30.9741	0.0450	6.75695E-05
used 3.36 lb				Cr	51.9957	19.0000	1.69951E-02
				Mn	54.9379	2.0000	1.69314E-03
				Fe	55.8447	68.3750	5.69445E-02
				Ni	58.6872	9.5000	7.52866E-03
SS304	10	7.9400	0.94348	C	12.0001	0.0800	3.00755E-04
containment vessel				Si	28.0853	1.0000	1.60629E-03
upper flange				P	30.9741	0.0450	6.55417E-05
used 13.75 lb				Cr	51.9957	19.0000	1.64850E-02
				Mn	54.9379	2.0000	1.64233E-03
				Fe	55.8447	68.3750	5.52356E-02
				Ni	58.6872	9.5000	7.30272E-03
277-4	11	1.45971	1.0	H	1.0077	2.84746	2.48404E-02
filling CV				^{10}B	10.0130	0.65907	5.78606E-04
inner-liner cavity				^{11}B	11.0096	2.97265	2.37348E-03

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm^3)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
Table 11 data: [b]				C	12.0001	1.36094	9.96935E-04
as-manufactured,				N	14.0033	0.01000	6.27756E-06
minimum density and				O	15.9904	53.39660	2.93541E-02
hydrogen				Na	22.9895	0.08326	3.18372E-05
				Mg	24.3048	0.23957	8.66479E-05
				Al	26.9818	27.69360	9.02241E-03
				Si	28.0853	1.74979	5.47673E-04
				S	32.0634	0.21865	5.99456E-05
				Ca	40.0803	8.39266	1.84069E-03
				Fe	55.8447	0.37575	5.91461E-05
Kaolite	12	(body weldment)					
Al_2O_3		0.34864	0.096	Al	26.9818	52.9390	3.95461E-04
				O	15.9904	47.0610	—
SiO_2		0.34864	0.367	Si	28.0853	46.7570	1.28281E-03
				O	15.9904	53.2430	—
Fe_2O_3		0.34864	0.067	Fe	55.8447	69.9540	1.76211E-04
				O	15.9904	30.0460	—
TiO_2		0.34864	0.012	Ti	47.8793	59.9530	3.15486E-05
				O	15.9904	40.0470	—
CaO		0.34864	0.307	Ca	40.0803	71.4810	1.14955E-03
				O	15.9904	28.5180	—
MgO		0.34864	0.131	Mg	24.3048	60.3170	6.82560E-04
				O	15.9904	39.6830	—
Na_2O		0.34864	0.020	Na	22.9895	74.1960	1.35522E-04
				O	15.9904	25.8040	—
Total O		—	—	—	15.9904	—	5.39065E-03
Water	12	0.52294	1.0	H	1.0077	11.1909	3.49711E-02
				O	15.9904	88.8091	1.74856E-02
Kaolite	13	(top plug)					
Al_2O_3		0.33241	0.096	Al	26.9818	52.9390	3.77052E-04
				O	15.9904	47.0610	—
SiO_2		0.33241	0.367	Si	28.0853	46.7570	1.22309E-03
				O	15.9904	53.2430	—
Fe_2O_3		0.33241	0.067	Fe	55.8447	69.9540	1.68008E-04
				O	15.9904	30.0460	—
TiO_2		0.33241	0.012	Ti	47.8793	59.9530	3.00799E-05
				O	15.9904	40.0470	—
CaO		0.33241	0.307	Ca	40.0803	71.4810	1.09604E-03
				O	15.9904	28.5180	—
MgO		0.33241	0.131	Mg	24.3048	60.3170	6.50785E-04
				O	15.9904	39.6830	—

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm^3)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
Na_2O		0.33241	0.020	Na	22.9895	74.1960	1.29213E-04
				O	15.9904	25.8040	—
Total O		—	—	—	15.9904	—	5.13975E-03
Water	13	0.49860	1.0	H	1.0077	11.1909	3.33433E-02
				O	15.9904	88.8091	1.66717E-02
Silicon rubber pads	14	1.21791	1.0	H	1.0077	8.1562	5.93660E-02
				C	12.0001	32.3774	1.97887E-02
				O	15.9904	21.5782	9.89729E-03
				Si	28.0853	37.8882	9.89432E-03
Water—CV well	15	0.9982	variable	H	1.0077	11.1909	variable
				O	15.9904	88.8091	variable
SS304 liner	16	7.94	1.0	C	12.0001	0.0800	3.18772E-04
				Si	28.0853	1.0000	1.70252E-03
				P	30.9741	0.0450	6.94680E-05
				Cr	51.9957	19.0000	1.74726E-02
				Mn	54.9379	2.0000	1.74071E-03
				Fe	55.8447	68.3750	5.85446E-02
				Ni	58.6872	9.5000	7.74020E-03
SS304 plug cover used 9.907 lb	17	7.94	1.06388	C	12.0001	0.0800	3.39135E-04
				Si	28.0853	1.0000	1.81128E-03
				P	30.9741	0.0450	7.39056E-05
				Cr	51.9957	19.0000	1.85887E-02
				Mn	54.9379	2.0000	1.85191E-03
				Fe	55.8447	68.3750	6.22844E-02
				Ni	58.6872	9.5000	8.23464E-03
SS304 angle iron	18	7.94	1.0	C	12.0001	0.0800	3.18772E-04
				Si	28.0853	1.0000	1.70252E-03
				P	30.9741	0.0450	6.94680E-05
				Cr	51.9957	19.0000	1.74726E-02
				Mn	54.9379	2.0000	1.74071E-03
				Fe	55.8447	68.3750	5.85446E-02
				Ni	58.6872	9.5000	7.74020E-03
SS304 steel drum	19	7.94	1.0	C	12.0001	0.0800	3.18772E-04
				Si	28.0853	1.0000	1.70252E-03
				P	30.9741	0.0450	6.94680E-05
				Cr	51.9957	19.0000	1.74726E-02
				Mn	54.9379	2.0000	1.74071E-03
				Fe	55.8447	68.3750	5.85446E-02
				Ni	58.6872	9.5000	7.74020E-03
Water—interstitial	20	0.9982	variable	H	1.0077	11.1909	variable
				O	15.9904	88.8091	variable

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm^3)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
Water—reflector	21	0.9982	1.0	H	1.0077	11.1909	6.67536E-02
				O	15.9904	88.8091	3.33856E-02
Array calculation models (NCT)							
Kaolite	12	(body weldment)					
Al_2O_3		0.42622	0.096	Al	26.9818	52.9390	4.83460E-04
				O	15.9904	47.0610	—
SiO_2		0.42622	0.367	Si	28.0853	46.7570	1.56821E-03
				O	15.9904	53.2430	—
Fe_2O_3		0.42622	0.067	Fe	55.8447	69.9540	2.15422E-04
				O	15.9904	30.0460	—
TiO_2		0.42622	0.012	Ti	47.8793	59.9530	3.85688E-05
				O	15.9904	40.0470	—
CaO		0.42622	0.307	Ca	40.0803	71.4810	1.40535E-03
				O	15.9904	28.5180	—
MgO		0.42622	0.131	Mg	24.3048	60.3170	8.34444E-04
				O	15.9904	39.6830	—
Na_2O		0.42622	0.020	Na	22.9895	74.1960	1.65678E-04
				O	15.9904	25.8040	—
Total O		—	—	—	15.9904	—	6.58477E-03
Water	12	0.63931	0.287	H	1.0077	11.1909	1.22702E-03
				O	15.9904	88.8091	6.13510E-04
Kaolite	13	(top plug)					
Al_2O_3		0.33241	0.096	Al	26.9818	52.9390	3.77052E-04
				O	15.9904	47.0610	—
SiO_2		0.33241	0.367	Si	28.0853	46.7570	1.22309E-03
				O	15.9904	53.2430	—
Fe_2O_3		0.33241	0.067	Fe	55.8447	69.9540	1.68008E-04
				O	15.9904	30.0460	—
TiO_2		0.33241	0.012	Ti	47.8793	59.9530	3.00799E-05
				O	15.9904	40.0470	—
CaO		0.33241	0.307	Ca	40.0803	71.4810	1.09604E-03
				O	15.9904	28.5180	—
MgO		0.33241	0.131	Mg	24.3048	60.3170	6.50785E-04
				O	15.9904	39.6830	—
Na_2O		0.33241	0.020	Na	22.9895	74.1960	1.29213E-04
				O	15.9904	25.8040	—
Total O		—	—	—	15.9904	—	5.13548E-03
Water	13	0.49860	0.0287	H	1.0077	11.1909	9.56954E-04
				O	15.9904	88.8091	4.78477E-04
SS304 angle iron	18	7.94	1.25705	C	12.0001	0.0800	4.00712E-04
				Si	28.0853	1.0000	2.14015E-03

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm³)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
				P	30.9741	0.0450	8.73248E-05
				Cr	51.9957	19.0000	2.19639E-02
				Mn	54.9379	2.0000	2.18816E-03
				Fe	55.8447	68.3750	7.35934E-02
				Ni	58.6872	9.5000	9.72982E-03
SS304 steel drum	19	7.94	0.99981	C	12.0001	0.0800	3.18711E-04
				Si	28.0853	1.0000	1.70220E-03
				P	30.9741	0.0450	6.94548E-05
				Cr	51.9957	19.0000	1.74693E-02
				Mn	54.9379	2.0000	1.74038E-03
				Fe	55.8447	68.3750	5.85334E-02
				Ni	58.6872	9.5000	7.73873E-03
Water—reflector	21	1.16235	1.0	H	1.0077	11.1909	7.75911E-06
				O	15.9904	88.8091	3.88058E-06
Single-unit calculation models (HAC) same as NCT except as follows:							
277-4	2	1.50853	1.0	H	1.0077	2.84094	2.56124E-02
canned spacer				^{10}B	10.0130	0.65958	5.98418E-04
Table 12 data: [b]				^{11}B	11.0096	2.97494	2.45475E-03
as-manufactured,				C	12.0001	1.36193	1.03103E-03
minimum density and				N	14.0033	0.01001	6.49258E-06
hydrogen				O	15.9904	53.36920	3.03202E-02
				Na	22.9895	0.08333	3.29277E-05
				Mg	24.3048	0.23976	8.96158E-05
				Al	26.9818	27.71520	9.33140E-03
				Si	28.0853	1.75109	5.66409E-04
				S	32.0634	0.21882	6.19989E-05
				Ca	40.0803	8.39916	1.90373E-03
				Fe	55.8447	0.37604	6.11720E-05
277-4	11	1.45858	1.0	H	1.0077	2.84097	2.47645E-02
filling CV				^{10}B	10.0130	0.65959	5.78606E-04
inner-liner cavity				^{11}B	11.0096	2.97496	2.37348E-03
Table 12 data: [b]				C	12.0001	1.36193	9.96883E-04
as-manufactured,				N	14.0033	0.01001	6.27756E-06
minimum density and				O	15.9904	53.36930	2.93162E-02
hydrogen				Na	22.9895	0.08333	3.18372E-05
				Mg	24.3048	0.23976	8.66479E-05
				Al	26.9818	27.71510	9.02236E-03
				Si	28.0853	1.75109	5.47651E-04
				S	32.0634	0.21882	5.99456E-05
				Ca	40.0803	8.39913	1.84068E-03
				Fe	55.8447	0.37604	5.91461E-05

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm^3)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
Kaolite	12	(body weldment)					
Al_2O_3		0.41898	0.096	Al	26.9818	52.9390	4.75248E-04
				O	15.9904	47.0610	—
SiO_2		0.41898	0.367	Si	28.0853	46.7570	1.54162E-03
				O	15.9904	53.2430	—
Fe_2O_3		0.41898	0.067	Fe	55.8447	69.9540	2.11763E-04
				O	15.9904	30.0460	—
TiO_2		0.41898	0.012	Ti	47.8793	59.9530	3.79136E-05
				O	15.9904	40.0470	—
CaO		0.41898	0.307	Ca	40.0803	71.4810	1.38148E-03
				O	15.9904	28.5180	—
MgO		0.41898	0.131	Mg	24.3048	60.3170	8.20270E-04
				O	15.9904	39.6830	—
Na_2O		0.41898	0.020	Na	22.9895	74.1960	1.62864E-04
				O	15.9904	25.8040	—
Total O		—	—	—	15.9904	—	6.47830E-03
Water	12	0.62843	1.0	H	1.0077	11.1909	4.20256E-02
				O	15.9904	88.8091	2.10128E-02
SS304 angle iron	18	7.94	1.23239	C	12.0001	0.0800	3.92851E-04
				Si	28.0853	1.0000	2.09817E-03
				P	30.9741	0.0450	8.56117E-05
				Cr	51.9957	19.0000	2.15330E-02
				Mn	54.9379	2.0000	2.14524E-03
				Fe	55.8447	68.3750	7.21497E-02
				Ni	58.6872	9.5000	9.53894E-03
SS304 steel drum	19	7.94	1.08401	C	12.0001	0.0800	3.45552E-04
				Si	28.0853	1.0000	1.84555E-03
				P	30.9741	0.0450	7.53040E-05
				Cr	51.9957	19.0000	1.89405E-02
				Mn	54.9379	2.0000	1.88695E-03
				Fe	55.8447	68.3750	6.34629E-02
				Ni	58.6872	9.5000	8.39045E-03
Array calculation models (HAC) same as NCT except as follows:							
Kaolite	12	(body weldment)					
Al_2O_3		0.42622	0.096	Al	26.9818	52.9390	4.83460E-04
				O	15.9904	47.0610	—
SiO_2		0.42622	0.367	Si	28.0853	46.7570	1.56821E-03
				O	15.9904	53.2430	—
Fe_2O_3		0.42622	0.067	Fe	55.8447	69.9540	2.15422E-04
				O	15.9904	30.0460	—
TiO_2		0.42622	0.012	Ti	47.8793	59.9530	3.85688E-05

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm^3)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
				O	15.9904	40.0470	—
CaO		0.42622	0.307	Ca	40.0803	71.4810	1.40535E-03
				O	15.9904	28.5180	—
MgO		0.42622	0.131	Mg	24.3048	60.3170	8.34444E-04
				O	15.9904	39.6830	—
Na_2O		0.42622	0.020	Na	22.9895	74.1960	1.65678E-04
				O	15.9904	25.8040	—
Total O		—	—	—	15.9904	—	6.58477E-03
Water	12	0.63931	0.287	H	1.0077	11.1909	1.22702E-03
				O	15.9904	88.8091	6.13510E-04
Kaolite	13	(top plug)					
Al_2O_3		0.33241	0.096	Al	26.9818	52.9390	3.77052E-04
				O	15.9904	47.0610	—
SiO_2		0.33241	0.367	Si	28.0853	46.7570	1.22309E-03
				O	15.9904	53.2430	—
Fe_2O_3		0.33241	0.067	Fe	55.8447	69.9540	1.68008E-04
				O	15.9904	30.0460	—
TiO_2		0.33241	0.012	Ti	47.8793	59.9530	3.00799E-05
				O	15.9904	40.0470	—
CaO		0.33241	0.307	Ca	40.0803	71.4810	1.09604E-03
				O	15.9904	28.5180	—
MgO		0.33241	0.131	Mg	24.3048	60.3170	6.50785E-04
				O	15.9904	39.6830	—
Na_2O		0.33241	0.020	Na	22.9895	74.1960	1.29213E-04
				O	15.9904	25.8040	—
Total O		—	—	—	15.9904	—	5.13548E-03
Water	13	0.49860	0.0287	H	1.0077	11.1909	9.56954E-04
				O	15.9904	88.8091	4.78477E-04
SS304 angle iron	18	7.94	1.25705	C	12.0001	0.0800	4.00712E-04
				Si	28.0853	1.0000	2.14015E-03
				P	30.9741	0.0450	8.73248E-05
				Cr	51.9957	19.0000	2.19639E-02
				Mn	54.9379	2.0000	2.18816E-03
				Fe	55.8447	68.3750	7.35934E-02
				Ni	58.6872	9.5000	9.72982E-03
SS304 steel drum	19	7.94	0.99981	C	12.0001	0.0800	3.18711E-04
				Si	28.0853	1.0000	1.70220E-03
				P	30.9741	0.0450	6.94548E-05
				Cr	51.9957	19.0000	1.74693E-02
				Mn	54.9379	2.0000	1.74038E-03

Table 6.4. Material compositions used in the ES-3100 calculation models

Material	Mix. No.	Theoretical density input parameter (g/cm³)[a]	Volume fraction	Constituent	Atomic weight	Weight fraction	Atomic density (atoms/b-cm)
				Fe	55.8447	68.3750	5.85334E-02
				Ni	58.6872	9.5000	7.73873E-03
Water—reflector	21	1.16235	1.0	H	1.0077	11.1909	7.75911E-06
				O	15.9904	88.8091	3.88058E-06

[a] The theoretical density input parameter is specified for each material such that the required (or desired) mass of material is represented in the KENO V.a calculation model. The mass of material is computed by KENO V.a as the product of the theoretical density input parameter times the volume fraction times the volume of the geometry region in which the material resides in the calculation model.
[b] DAC-PKG-801624-A001.

The CSAS25 control sequence and the 238-group ENDF/B-V cross-section library in SCALE are used for all calculations. The control sequence, functional modules, and cross-section library are summarized in the following paragraphs.

The CSAS25 control sequence reads user-specified input data, which include the required cross-section library, specifications for mixtures, information for resonance processing of nuclides (size, geometry, and temperature), and a detailed geometry model for KENO V.a. Physical and neutronic information not specified but required by the functional modules (such as theoretical density, molecular weights, and average resonance region background cross sections) is supplied by the Standard Composition Library or calculated by the Materials Information Processor. The Standard Composition Library (SCALE, Vol. 3, Sect. M8) consists of a standard composition directory and table, an isotopic distribution directory and table, and a nuclide information table. The Materials Information Processor (SCALE, Vol. 3, Sect. M7) checks the input data pertaining to cross-section preparation and prepares binary input files for the applicable functional modules BONAMI and NITAWL-II.

The standardized automated procedures process SCALE cross sections using the Bondarenko method (via BONAMI) and the Nordheim integral method (via NITAWL-II) to provide a resonance-corrected cross-section library based on the physical characteristics of the problem being analyzed.

BONAMI performs resonance shielding through the application of the Bondarenko shielding factor method. BONAMI reads the master format library and applies the Bondarenko correction to all nuclides that have Bondarenko data. BONAMI produces a Bondarenko-corrected master format library which is read by NITAWL-II.

NITAWL-II applies the Nordheim Integral Treatment to perform neutron cross-section processing in the resonance energy range for nuclides that have ENDF/B resonance parameter data. This technique involves the numerical integration of ENDF/B resonance parameters using a calculated flux distribution, which is based on the calculated collision density across each resonance and subsequent weighing of the reaction cross section to the desired broad group structure. In the CSAS sequence, NITAWL-II assembles the group-to-group transfer arrays from the elastic and inelastic scattering components and performs other tasks to produce a problem-dependent, working cross-section library which can be used by KENO.

KENO V.a, a multigroup Monte Carlo computer code, is used to determine k_{eff} for multidimensional systems. The basic geometrical bodies allowed in KENO V.a for defining models are cuboids, spheres, cylinders, hemispheres, and hemicylinders. KENO V.a has the following major characteristics:

- enhanced geometry package that allows arrays to be defined and positioned throughout the model;
- P_n scattering treatment;
- extended use of differential albedo reflection;
- printer plots for checking the input model;
- energy-dependent data supergrouping;
- restart capability; and
- origin specifications for cuboids, spheres, cylinders, hemicylinders, and hemispheres.

The 238-group ENDF/B-V master cross-section library in SCALE is activated in the CSAS25 control sequence by specifying 238GROUPNDF5 (238GR) as the cross-section library name. The 238-GROUP ENDF/B-V library is a general-purpose criticality analysis library and the most complete library available in SCALE. The library contains data for all nuclides (more than 300) available in ENDF/B-V processed by the AMPX-77 systems. It also contains data for ENDF/B-VI evaluations of ^{14}N, ^{15}N, ^{16}O, ^{154}Eu, and ^{155}Eu. The library has 148 fast groups and 90 thermal groups (below 3 eV). Most resonance nuclides in the 238 group have resonance data (to be processed by NITAWL-II in the resolved resonance range) and Bondarenko factors (to be processed by BONAMI) for the unresolved range. The 238-group library contains resolved resonance data for s-wave, p-wave, and d-wave resonances **R** = 0, **R** = 1, and **R** = 2, respectively. These data can have a significant effect on results for under-moderated, intermediate-energy problems. Resonance structures in several light-to-intermediate mass "nonresonance" ENDF nuclides (i.e., ^{7}Li, ^{19}F, ^{27}Al, ^{28}Si) are accounted for using Bondarenko shielding factors. These structures can also be important in intermediate energy problems.

All nuclides in the 238-group library use the same weighting spectrum, consisting of:

- Maxwellian spectrum (peak at 300 K) from 10^{-5} to 0.125 eV,
- a 1/E spectrum from 0.125 eV to 67.4 keV,
- a fission spectrum (effective temperature at 1.273 MeV) from 67.4 keV to 10 MeV, and
- a 1/E spectrum from 10 to 20 MeV.

The k_{eff} values for each KENO V.a case are based on 500,000 neutron histories produced by running for 215 generations with 2,500 neutrons per generation and truncating the first 15 generations of data. The convergence of the KENO V.a calculation is related to trends in the average calculated k_{eff}. In general, the KENO V.a output table of k_{eff} by Generation Skipped is reviewed for trends. If no trends are observed, the calculation is accepted, and the reported value of k_{eff} is the one with the most neutron histories. Usually there is no statistically significant difference between this result and the one with the smallest standard deviation.

No manual cross-section adjustment is performed for the criticality safety evaluation. Cross-section processing is performed automatically in the CSAS25 code sequence. By modeling the contents as precisely as possible, the information for correct cross-section processing is introduced into the evaluation.

The CSAS control module, the associated functional modules, cross sections, and databases used in this evaluation reside in the verified configuration control area designated **/vcc/scale4.4a** on a

Hewlett Packard Series 9000 J Class workstation at the Y-12 Safety Analysis Engineering organization in Oak Ridge, Tennessee. The detailed input and computer output for the criticality safety evaluation of the ES-3100 shipping container with HEU reside in a configuration control area **/archive/ylf717_Rv2** on the workstation. Input listings of the key cases indicated in Tables 6.1a–6.1e are provided in Appendix 6.9.7.

6.3.4 Demonstration of Maximum Reactivity

10 CFR 71.55(b) requires the evaluation of water leakage into the containment vessel, leakage of liquid contents out of the containment vessel, and other conditions which produce maximum reactivity in the single package. For solid uranium contents, water leakage conditions are simulated by flooding all regions outside and inside of the containment vessel, including the sealed convenience cans. For liquid uranium contents, water leakage conditions are simulated by flooding all regions outside the containment vessel except the containment vessel well. Uranyl nitrate solution resides inside both the containment vessel, including the sealed convenience cans, and the well external to the containment vessel. According to the 10 CFR 71.55(b) requirement, a flooded containment vessel under full water reflection is also evaluated.

Credit for the high-integrity, watertight containment is not taken either in the single package analysis [10 CFR 71.55 (d, e)] or in the array analysis [10 CFR 71.59 (a)(1)] of undamaged packages. In the evaluation of undamaged packages under 10 CFR 71.59 (a)(1) and the evaluation of damaged packages under 10 CFR 71.59 (a)(2), the containment vessel is flooded with water, providing moderation to such an extent as to cause maximum reactivity of the content consistent with the chemical and physical form of the material present. Solid HEU, <u>not solution HEU</u>, is being shipped in the ES-3100. Consequently, in the evaluation of damaged packages under 10 CFR 71.59 (a)(2), the leakage out of the containment vessel of content moderated to such an extent as to cause maximum reactivity consistent with the physical and chemical form of the material is not considered credible HAC, based on results for tests specified in 10 CFR 71.73.

Contents are generally dry, and only small quantities of hydrogenous packing materials are used. However, credit for the high-integrity, watertight containment is not taken in this criticality evaluation. (Meeting, Docket 71-9315) Fissile material loading are restricted such that if a containment vessel were flooded, it would be adequately subcritical under full water reflection. But instead of loading limits being established in accordance with conditions of confinement and containment as demonstrated under the NCT and HAC tests, these single-unit fissile material loading limits are severely reduced on the basis of the array calculations where all of the containment vessels of the packages in an array are assumed to flooded but each packaging is dry. This has the effect of maximizing both the neutron source and neutronic interaction between packages. Because credit for the high-integrity, watertight containment is not fully taken in this criticality evaluation, and the fissile material mass loading limits developed as a result are very conservative.

Section 6.3.1 provides dimensional data for content and package models, and Sect. 6.3.2 provides material composition data used in the calculation models. Appendix 6.9.3 provides justification for the composition data used in the evaluation. These sections and this appendix describe how the packaging dimensions and materials are optimized to produce a conservative model by reducing neutron absorbing materials and maximizing the reactivity of the fissile material content through the inclusion of water in the containment vessel.

As described in Sect. 6.3.1.1, a geometry region is defined by dimensions and the type of material contained therein. The amount of material within a region is quantified in terms of the density, material volume fraction, and the volume of the region (Sect. 6.3.2). The term "water fraction" means the fraction of the maximum specific gravity of water possible for any geometry region in the ES-3100 package. Therefore, the water content is the fraction of the maximum specific gravity of water possible in a material in the ES-3100 package when flooded.

The maximum value for the void regions (spaces external to the containment vessel) is 1.0 for both single-unit and array geometries. For the geometry regions containing Kaolite, the maximum values are 0.51655 for the single unit and 0.63931 for the array. For the geometry regions containing 277-4, the maximum value is 0.6942 for both the single-unit and array geometries. The same single-unit values are used in the array calculation models because the 277-4–bearing regions do not require adjustment for the close-pack approximation (Sect. 6.3.1.2).

In general, boundary condition specifications are not required in KENO V.a, so that calculation models analyzed using KENO V.a require no special boundary conditions. However, in this evaluation, the infinite array cases use a single package with a reduced radius modeled with spectral reflection on all faces of a surrounding cuboid. The energies and angular dependence of the neutrons are treated such that an infinite system with no neutron leakage is simulated.

For simplicity, the NCT and HAC sets of calculations were performed for selected ES-3100 array sizes by varying the water content of the ES-3100 package external to the containment vessel from zero to its maximum value. This technique bounds all NCT and HAC; however, only the relevant calculation results are used in determination of the CSI for criticality purposes.

Although four decimal places are shown for k_{eff} values in result tables, the actual accuracy of the code, for a particular calculation, may be on the order of ± 0.02–0.03 based on the spread in results for benchmark calculations. (Y/DD-896/R1, Y/DD-972/R1) Also, the standard deviation of the mean for a particular calculation is on the order of 0.001 for benchmark cases and somewhat higher for the package calculations. Therefore, numerical values are considered physically meaningful or significant to the third decimal place. The primary reason for reporting four (or five) decimal places for a calculation result is to confirm that the result reported actually originated from a given output file. A value of 0.925 is the USL used for this safety analysis report (Sect. 6.8.3).

6.4 SINGLE PACKAGE EVALUATION

The HEU content of a package is in one of the following forms: metal of a specified geometric shape; metal of an unspecified shape characterized as broken metal; uranium oxide; or UNH crystals. The bounding types of HEU content evaluated in this criticality analysis are 3.24-in. and 4.25-in.-diam cylinders; 2.29-in.-square bars; 1.5-in.-diam × 2-in.-tall slugs; cubes ranging from 0.25 to 1 in. on a side; broken metal pieces of unspecified geometric shapes; product oxide; skull oxide; UNH crystals; and unirradiated TRIGA reactor fuel elements.

Dimensions for bounding calculation models of a TRIGA fuel section are 1.44 inches in diameter × 5 in. tall. The 20% enrichment fuel contains more fissile mass than the 70% enrichment fuel on account of a higher uranium weight fraction and associated material density. A 20 wt % enriched TRIGA fuel element with 45 wt % U contains ≤307 g ^{235}U; whereas, a 70 wt % enriched TRIGA fuel with 8.5 wt % U contains ≤136 g ^{235}U. The physical parameters which characterize the TRIGA fuel are enrichment, uranium weight fraction, and the fuel diameter. (Appendix 6.9.3.1). These parameters are addressed in this section demonstrating that the calculation models are bounding in the criticality evaluation.

Uranium-aluminum (U-Al) alloy averages 0.94 kg of uranium, 0.60 kg of ^{235}U, and 4.07 kg of aluminum. Uranium-molybdenum alloy varies from 1.5 wt % to 10 wt % molybdenum and the enrichment varies from 92.65 to 93.21 wt % ^{235}U. Where scattering media (aluminum or molybdenum) is treated as multiplying media (^{235}U), the alloy is conservatively assessed. The evaluation of the 3.93-in.-diam × 9.5-in.-tall U-Al cylinders is covered under the evaluation the 4.25-in.-diam HEU metal cylinders. Evaluation of the U-Mo content is covered under the assessment of HEU contents at 95 wt% enrichment.

6.4.1 Solid HEU Metal of Specified Geometric Shapes

For bare and reflected single packages with HEU metal content, the neutron multiplication factor increases as a function of the ^{235}U mass and the moisture fraction of the package external to the containment vessel (MOIFR). For example, consider the ES-3100 package loaded with three convenience cans for a total of 36,000 g ^{235}U. Each press-fit lid type can contains a single 3.24-in.-diam, 12,000 g cylinder of ^{235}U. The $k_{eff} + 2\sigma$ values in the bare package increase from 0.911 to 0.955 with increasing moisture fraction of the package external to the containment vessel (MOIFR), Cases **ncsbcyt11_36_1_1** through **ncsbcyt11_36_1_15** (Appendix 6.9.6, Table 6.9.6-2). The $k_{eff} + 2\sigma$ values increase from 0.918 to 0.955 with increasing MOIFR in the water-reflected package, Cases **ncsrcyt11_36_1_1** through **ncsrcyt11_36_1_15** (Appendix 6.9.6, Table 6.9.6-2). The addition of water to the package reduces the neutron leakage fraction (NLF), thereby increasing k_{eff}. Water reflection external to a flooded package is inconsequential to package reactivity as a comparison of results for Cases **ncsbcyt11_36_1_15** and **ncsrcyt11_36_1_15** indicates.

In this series of calculations using the ES-3100 package model with NCT geometry (Cases **ncsrcyt11_36_1_1** through **ncsrcyt11_36_1_15**), the MOIFR is varied uniformly over the package model with the exception of the neutron poison of the body weldment liner inner cavity and the flooded containment vessel. The single-unit case with a MOIFR = 1.0 pertains specifically to the flooded drum under conditions specified in 10 CFR 71.55(b). This pseudo-HAC condition is more reactive than either the true NCT where both the containment vessel and Kaolite are dry [10 CFR 71.55(d)] or this evaluation for NCT where the containment vessel is flooded and the Kaolite is dry (in the as-manufactured condition, MOIFR ~ 0.0289). At MOIFR = 1.0, external water reflection of the package is inconsequential to package reactivity. Moisture in the Kaolite and in the recesses of the package acts as a close reflector that decreases neutron leakage away from the package.

The single-unit case with a MOIFR = 1e-20 pertains specifically to a package under pseudo-HAC where both the Kaolite and the recesses of the package do not contain **any** water or bound hydrogen. This configuration is less reactive than the cases for Kaolite in the as-manufactured condition or water-saturated Kaolite because water in the Kaolite will provide some neutron moderation and reflection of neutrons back into the content.

Single package reactivity changes slightly between the bare and reflected conditions, while NLF changes considerably over the range of water fractions. This behavior illustrates the dependence of package reactivity on internal conditions of the package. Bare packages with low MOIFR values manifest low reactivity (k_{eff} = 0.911) and high neutron leakage (NLF = 0.47). These parameters indicate that fast neutrons scattered in the packaging do not slow down significantly but escape the package. Consequently, neutron interaction between these packages when they are configured into an array is high. Bare packages with MOIFR values >1e-2 manifest increasing k_{eff} values (from 0.911 to 0.955) and reduced NLF values (from 0.47 to 0.18). The increase in k_{eff} occurs due to water present in the regions of the package external to the content which reflects neutrons back into the HEU content that would otherwise escape the package.

The ES-3100 package is not as efficient a reflector as full water reflection provided to the flooded containment vessel. For the containment vessel loaded with 3.24-in.-diam cylinders, the $k_{eff} + 2\sigma$ values increase monotonically from 0.907 to 0.974 with increasing moisture fraction inside the containment vessel (MOCFR) [Cases **cvcrcyt11_36_1_1** through **cvcrcyt11_36_1_15** (Appendix 6.9.6, Table 6.9.6-1, and Appendix 6.9.1, Fig. 6.9.1-1)]. The flooded containment vessel under full water reflection is a more reactive configuration than the containment vessel inside of flooded ES-3100 packaging, Case **ncsrcyt11_36_1_15**. For this reason, calculation results for the flooded containment rather than a flooded package are reported in Tables 6.1a–6.1e for 10 CFR 71.55(b) and are taken into consideration in the determination of HEU fissile mass loading limits.

Examination of the results for Cases **cvcrcyt11_36_1** through **cvcrcyt11_18_1** indicates that an adequately subcritical loading is achieved when the mass loading is limited to 21,000 g ^{235}U (Case **cvcrcyt11_21_1**). Case **cvcrcyt11_36_2** reveals that the 277-4 canned spacers are adequate for mass loading >8,000 g ^{235}U but ≤36,000 g ^{235}U.

Case **hcsrcyt12_36_1_15** (Appendix 6.9.6, Table 6.9.6-3) represents the HAC model of the damaged ES-3100 package, where the outer dimensions of the package are reduced accordingly and the entire package is flooded with the exception of the neutron poison of the body weldment liner inner cavity. The containment vessel well is flooded with water, and the Kaolite contains maximum water content. This single-unit case with a MOIFR = 1.0 pertains specifically to the flooded drum under conditions specified in 10 CFR 71.55(e). The $k_{eff} + 2\sigma = 0.955$ for this HAC. The changes in both the outer dimensions of the package and the compositions of the Kaolite and 277-4 due to HAC result in an ~0.002 change in the neutron multiplication factor.

Repeated for 4.25-in.-diam cylinders (Appendix 6.9.6, Tables 6.9.6-6 and 6.9.6-7); 2.29-in. -square bars (Tables 6.9.6-4 and 6.9.6-5); and 1.5-in.-diam × 2-in.-tall slugs (Tables 6.9.6-8 and 6.9.6-9), this type of analysis for the 3.24-in.-diam cylinders demonstrates that single packages with restricted fissile material (^{235}U) loading remain subcritical over the entire range of water content or MOIFR. HEU bulk metal or alloy content not covered by the specified geometric shapes (cylinder, square bar, or slug contents) will be in the HEU broken metal category, and so limited.

Each TRIGA fuel element for shipment in the ES-3100 is to be disassembled. The active fuel consisting of three 5-in.-tall cylinders of $UZrH_{1.6}$ is to be packed into a metal-type convenience can. (Section 6.2.1) The use of three loaded convenience cans in the ES-3100 is assumed in the criticality calculation models for ground transport. However, credit is only taken for vertical spacing provided by canned spacers when present. For configurations where canned spacers are not used, the TRIGA content is assumed stacked end-to-end. The three cylinders of the active fuel are modeled in triangular-pitch configuration within the radial boundary of the convenience can(s). The pitch used in the criticality calculation models for the NCT and HAC ground transport is 4.9809 cm (1.96 in.)

As shown in Table 6.9.6-19b, a set of parametric cases were run where the triangular pitch for the fuel content is varied from 1.44 in. (fuel sections touching) to 2.413 in. (fuel sections at maximum separation and touching the inner boundary of the convenience can). As shown in Fig. 6.16a, the calculated $k_{eff} + 2\sigma$ values increase slightly from 0.48 to an asymptotic value of ~ 0.53 at a pitch of 2.121 in. or a foci of 3.1106 cm. (Table 6.9.6-19b) The foci of 2.8757 cm. for the TRIGA fuel sections used in the criticality calculation models, corresponds to a 1.96 in. pitch. The calculated $k_{eff} + 2\sigma$ value is

0.527 for Case **cvcrtriga_1_15** (Table 6.9.6-19a) indicating that the configuration is in the range of optimum moderation. Moreover, the calculated $k_{eff} + 2\sigma$ value is substantially below the USL. (The foci is the distance perpendicular from the CV's vertical axis to the center of each fuel section. The value is obtained from the computer input listing on the comment line for geometry Unit 1006 which reads: " -2.875700 minus-y location lower cylinder. ")

Parametric cases were also run as a function of the moisture fraction ("mocfr") in the CV. As shown in Fig. 6.16b, the $k_{eff} + 2\sigma$ values decrease with decreasing water content in the CV; and content spacing has less of an effect on k_{eff} at lower at the lower fractions.

The first of the three physical parameters which characterize the TRIGA fuel is enrichment. Table 6.9.6-19a presents results for 20 % enriched and 70 % enriched TRIGA fuel where the diameter of the active fuel is 1.44 in. The calculated $k_{eff} + 2\sigma$ values for TRIGA fuel content at the lower enrichment is greater than for the fuel with the higher enrichment due to the greater quantity of fissile isotope present. The 20% enriched TRIGA fuel is considered the bounding content.

The second physical parameter which characterizes the TRIGA fuel is the uranium weight fraction. For a TRIGA fuel enriched to 20 wt % ^{235}U in U, the uranium weight fractions for the $UZrH_{1.6}$ fuel are: 45 wt %; 30 wt %; 20 wt %; 12 wt %; and 8.5 wt % U. A set of parametric cases were run where the uranium weight fraction is varied from the minimum value of 8.5 wt % to the maximum value of 45 wt %. Additionally, moderation in the containment vessel is varied from the dry to the flooded condition. As shown in Fig. 6.16c, the $k_{eff} + 2\sigma$ values decrease with both decreasing uranium weight fraction and decreasing water content in the CV. Details are provided in Table 6.9.6-19c.

Third physical parameter which characterizes the TRIGA fuel is active fuel diameter. The active fuel diameters are: 1.44 in.; 1.41 in.; 1.40 in.; 1.37 in.; 1.34in.; and 1.31 in. The smaller the diameter, the less the amount of ^{235}U is present resulting in lower $k_{eff} + 2\sigma$ values, as demonstrated later in this section.

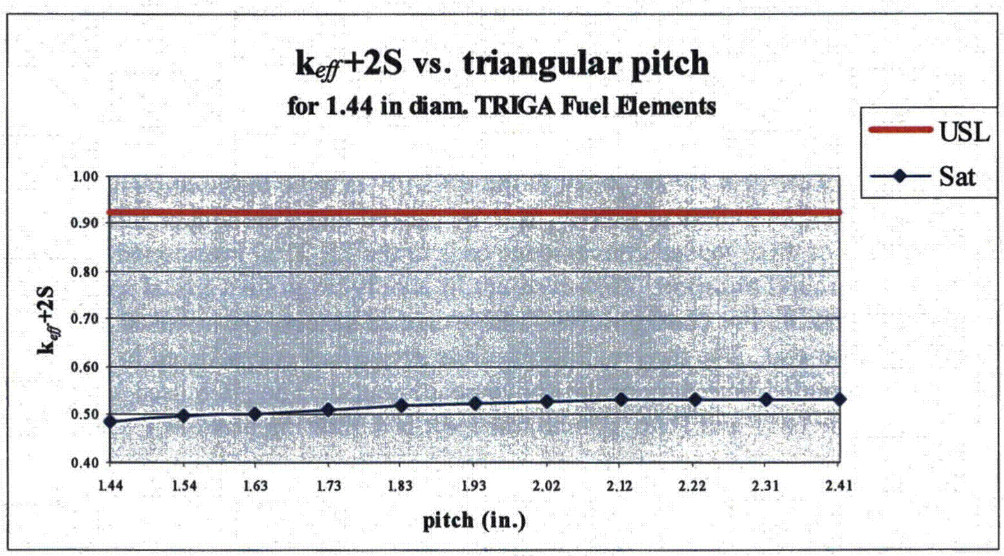


Fig. 6.16a. $k_{eff} + 2\sigma$ versus pitch for triangular arrangement of content in CV.

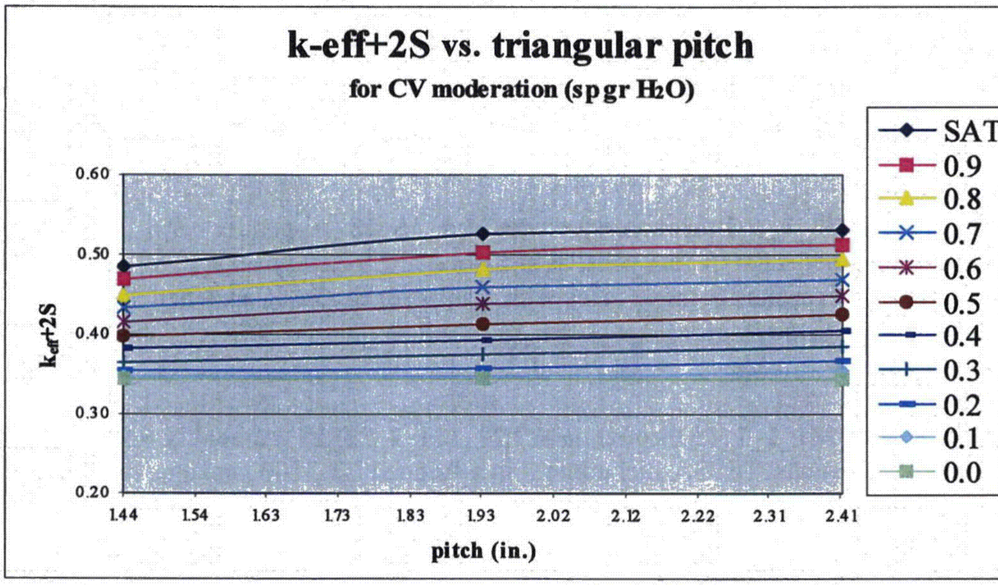


Fig. 6.16b. $k_{eff} + 2\sigma$ versus triangular pitch over range of CV moderation.

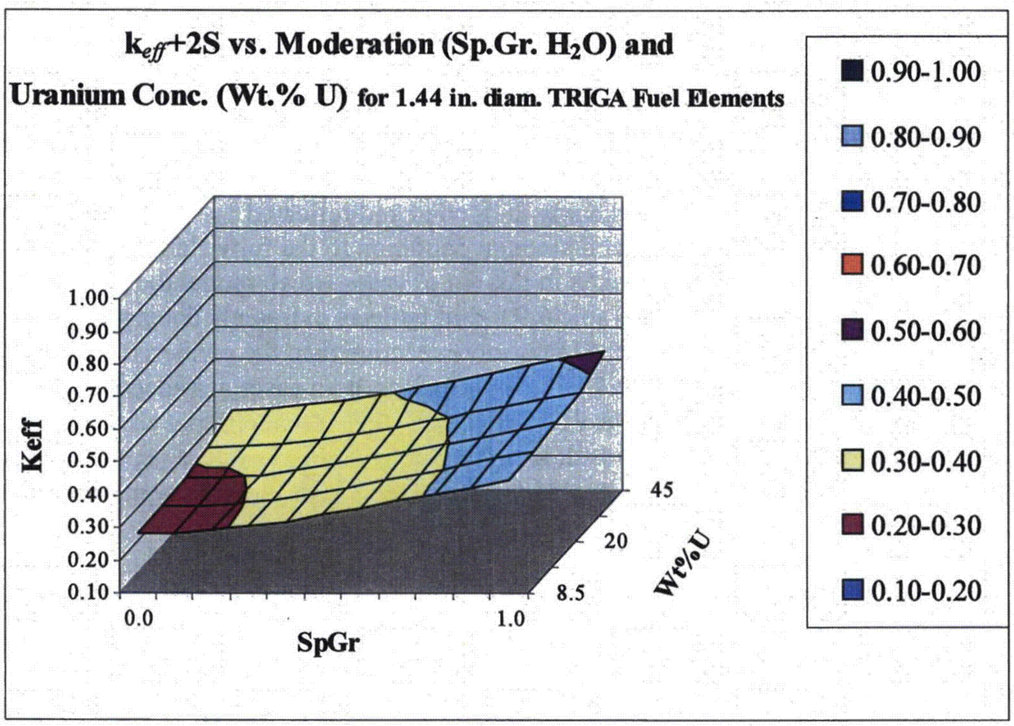


Fig. 6.16c. $k_{eff} + 2\sigma$ as a function of uranium weight fraction and moderation.

Cases **cvcrtriga_1_1** through **cvcrtriga_1_15** (Appendix 6.9.6, Table 6.9.6-19a) represent the TRIGA fuel content in a flooded containment vessel, reflected by 30.48 cm of water. The flooded containment vessel under full water reflection is a more reactive configuration than the containment vessel inside of flooded ES-3100 packaging, Case **ncsrtriga_1_15_15** (Appendix 6.9.6, Table 6.9.6-20a).

Further examination of the results in Table 6.9.6-20a indicates the content loading is adequately subcritical loading and that 277-4 canned spacers are not required for criticality control. Comparison of results for the NCT Case **ncsrtriga_1_15_15** and the HAC **hcsrtriga_1_15_15**, both with $k_{eff} + 2\sigma = 0.495$, reveal that changes to the packaging external to the containment vessel due to the HAC do not result in an appreciable change in the neutron multiplication factor for the single package.

Cases **ncsrtriga70_1_1_1** through **ncsrtriga70_1_15_15** (Appendix 6.9.6, Table 6.9.6-20b) represent the 70 % enriched TRIGA fuel content in a flooded ES-3100 package, reflected by 30.48 cm of water. Comparison of results for these cases with results for the 20 % enriched TRIGA fuel (Cases **ncsrtriga_1_1_1** through **ncsrtriga_1_15_15**) confirm that the 20 % enriched TRIGA fuel is the bounding content.

Cases **ncsrT70_131_1_1_15** through **ncsrT70_131_1_15_15** (Appendix 6.9.6, Table 6.9.6-20c) represent the 1.31 in. diameter TRIGA fuel content in a flooded ES-3100 package, reflected by 30.48 cm of water. Comparison of results for these cases with results for the larger 1.44 in. diameter TRIGA fuel content (Cases **ncsrtriga70_1_1_1** through ncsrtriga70_1_15_15) confirm that the 1.44 in. diameter TRIGA fuel is the bounding content.

10 CFR 71.55(d)(2) requires the geometric form of a package's content not be substantially altered under the NCT. Also, 10 CFR 7155(e)(1) requires that the package be adequately subcritical under HAC with the package contents in the most reactive credible configuration. However, conclusions about damage to the fuel content can not be extrapolated from test data because a mock (test weight) content rather than actual TRIGA content is evaluated in the NCT and HAC tests of 10 CFR 71.71 and 10 CFR 71.73. Consequently, one way for addressing these requirements is to model the content in an extremely damaged condition and make a determination of subcriticality through a series of criticality calculations. Cases **ncsrt55d2_ 1_1_15** through **ncsrt55d2_1_15_15** (Appendix 6.9.6, Table 6.9.6-20d) represent TRIGA fuel content homogenized with variable density water over the free volume of the containment vessel, where the ES-3100 packaging is flooded and reflected by 30.48 cm of water. The variable density water ranges from the dry containment condition to the fully flooded condition. Credit for physical integrity of the content is not taken in this set of cases which model the substantially altered content. The calculation results in Table 6.9.6-20d indicate extremely damaged content (Case **ncsrt55d2_1_15_15** with $k_{eff} + 2\sigma = 0.611$) is more reactive than the unaltered configuration (Case **ncsrtriga_1_15_15** with $k_{eff} + 2\sigma = 0.403$). Nevertheless, both cases are adequately below the USL of 0.925 and the requirement of 10 CFR 71.55(d)(2) is satisfied. Given that changes external to the containment vessel due to the HAC do not result in an appreciable change in the neutron multiplication for the single package, similar results are expected for the cases demonstrating compliance with 10 CFR 7155(e)(1).

The shipping configuration for disassembled TRIGA fuel addressed in this subsection is not the only permissible shipping configuration for TRIGA fuel in the ES-3100. TRIGA fuel may also be configured as clad fuel rods (Appendix 6.9.3.1). Each 15 in. long rod is derived from a single TRIGA fuel element. The clad fuel rod consists of the 3 fuel pellets and the exterior sheath of stainless steel or aluminum clad. Clad fuel rods are packed into stainless steel or tin-plated carbon steel convenience cans with a maximum of three fuel rods per loaded convenience can. This shipping configuration requires that only one convenience can is loaded with clad fuel rods.

Except for a 0.02 in. thick sheath of stainless steel clad added to the exterior surface of the $UZrH_{1.6}$, a calculation model for the clad fuel rod configuration is essentially the same as the NCT shipping configuration model for disassembled TRIGA fuel. A 0.02 in thickness of stainless steel is insignificant for an external reflection. As illustrated in Fig. 6.21 (Sect. 6.7.2), stainless steel up to several cm in thickness acts as a neutron absorber. Several inches in thickness are required for neutron multiplication to increase from neutron reflection by the stainless steel. The NCT shipping configuration model for disassembled TRIGA fuel is bounding. The same applies for TRIGA fuel with aluminum clad.

A clad fuel rod with 1.44 in. diameter fuel pellets contains 2,282.4 g $UZrH_x$ (Appendix 6.9.3.1) and ~179 to 191 g of stainless steel. Stainless steel tends to act as a neutron absorber; moreover, its presence as clad in the TRIGA fuel content replaces water moderator otherwise present in the geometry configuration of TRIGA fuel meats. When stainless steel is homogenized with the $UZrH_{1.6}$ as in the calculation model for package content in the extremely damaged condition [10 CFR 71.55(d)(2) and 10 CFR 7155(e)(1)], the stainless steel acts more effectively as a neutron absorber. However, the amount of stainless steel added and water displaced is not expected to have a statistically significant affect on neutron multiplication. Thus, the HAC shipping configuration model for disassembled TRIGA fuel (bare fuel meats) is bounding.

A clad fuel rod with 1.41 in. diameter fuel pellets contains ~2,188 g $UZrH_x$ and ~90 to 96 g of aluminum. While aluminum tends to act as a neutron scatter, its presence in the TRIGA fuel content replaces water moderator otherwise present in the geometry configuration of TRIGA fuel meats. The amount of aluminum added and water displaced is not sufficient to have a statistically significant affect on neutron multiplication. Thus, the HAC shipping configuration model for disassembled TRIGA fuel is also bounding for aluminum clad TRIGA fuel content.

The TRIGA content is to be transported by air; consequently, additional discussion is included in Sect. 6.7.

The 1.5-in.-diam × 2-in.-tall slugs may be packed up to ten items per press-fit lid type convenience can and up to twelve items per crimp-lid type convenience can. With nominal dimensions, each slug weighs ~1,090 g. With +1/16 in. tolerance on both the diameter and height, each slug in the calculation model weighs ~1291 g. As described in Appendix 6.9.1, different arrangements of slugs in the convenience cans are possible. A configuration of slugs in a flooded reflected containment vessel must be shown to be adequately subcritical and so limited either by the use of 277-4 canned spacers, by limitation of fissile mass, or by both.

For this evaluation, the slugs are modeled as 100 wt % ^{235}U but the convenience cans are not modeled. The calculation results presented in Table 6.9.6-8 (Appendix 6.9.6) for a flooded containment vessel indicate that only five slugs per convenience can (18,287 g ^{235}U per package) may be loaded without the use of 277-4 canned spacers (Cases **cvcr5st11_1_1** and **cvcr5est11_1_1**). Further evaluation of the single package and array configurations are required. Also, results highlighted in red indicate that 277-4 canned spacers are required for the slug content.

The sensitivity of k_{eff} to the space between slugs arranged in a pentagonal ring is evident in the calculation results for Cases **cvcr5st11_1_2** and **cvcr5est11_1_2,** which model content for three convenience cans, each location with two pentagonal rings (10 slugs) per can. These two configurations represent degrees of separation between adjacent neighbors in the pentagonal rings of 0.0 cm and 1.0 cm, respectively. The most reactive configuration occurs when the slugs are spaced 1.0 cm apart from direct contact with adjacent neighbors in the pentagonal rings. Because positioning devices are not used in the convenience cans to control spacing and prevent an optimal arrangement of contents from occurring, the latter calculation model for slug arrangement "**5est**" is the basis for additional calculations where spacers may not be required.

The calculation results presented in Table 6.9.1-8 for a flooded containment vessel indicate that up to 10 slugs per convenience can (36,573 g ^{235}U per package) *might* be loaded when 277-4 canned spacers are used. Conceivable arrangements are considered in Cases **cvcr5st11_2_2, cvcr5est11_2_2, cvcr6e4st11_2**, and **cvcr73st11_2**, (Table 6.9.6-8). Suitability is contingent on the results of single package and array calculations.

Cases **ncsr5est11_2_1_1** through **ncsr5est11_2_1_15** (Appendix 6.9.6, Table 6.9.6-9) model a reflected package with one pentagonal ring of slugs per content location with 277-4 canned spacers between content locations and HEU content at 100 wt % ^{235}U. The $k_{eff} + 2\sigma$ values range from a low value of 0.687 to 0.737. These results show that the most reactive configuration is the flooded condition with MOIFR=1.0.

As shown by the cylindrical content calculation models, the ES-3100 package is not as an efficient reflector as full water reflection provided to the flooded containment vessel. This is also true for content loadings of one pentagonal ring of slugs per content location without 277-4 canned spacers between locations and HEU content at 100 wt % ^{235}U. For Cases **cvcr5est11_1_1** and **ncsr5est11_1_1_15**, the $k_{eff} + 2\sigma$ values are 0.909 and 0.870, respectively. Likewise, the $k_{eff} + 2\sigma$ values are 0.903 and 0.868 for content loadings of two pentagonal rings of slugs per content location with 277-4 canned spacers between locations, Cases **cvcr5est11_2_2** and **ncsr5est11_2_2_15**, respectively.

For the flooded containment vessel under full water reflection and loaded with the slugs in a pentagonal arrangement and 277-4 canned spacers between content locations, $k_{eff} + 2\sigma = 0.904$, Case **cvcr6e4st11_2** [10 CFR 71.55(b).] This value is below the USL value of 0.925. For packages with the required 1.4-in. 277-4 canned spacers, the calculated $k_{eff} + 2\sigma$ value is 0.868 for the water-reflected package, Case **ncsr5est11_2_2_15** [10 CFR 71.55(d).] Case **hcsr5est11_02_2_15** represents the HAC model of the damaged ES-3100 package, where the outer dimensions of the package are reduced accordingly and the entire package is flooded except the neutron poison of the body weldment liner inner cavity. This single-unit case with a MOIFR = 1.0 pertains specifically to the flooded drum under conditions specified in 10 CFR 71.55(e). The $k_{eff} + 2\sigma = 0.871$ for this HAC. The changes in both the outer dimensions of the package and the compositions of the Kaolite and 277-4 due to HAC result in an ~0.003 change in the neutron multiplication factor.

6.4.2 HEU Solid Metal of Unspecified Geometric Shapes or HEU Broken Metal

Like packages with HEU metal, the neutron multiplication factor for reflected single packages with HEU broken metal increases as a function of the ^{235}U mass and the MOIFR. For example, consider the ES-3100 package loaded with three convenience cans for a total of 35,142 g ^{235}U. The $k_{eff} + 2\sigma$ values range from 0.814 to 0.891 with increasing MOIFR in the water-reflected package [Cases **ncsrbmt11_36_1_1** through **ncsrbmt11_36_1_15** (Appendix 6.9.6, Table 6.9.6-11)]. The addition of water to the package reduces the NLF, thereby increasing k_{eff}.

For the containment vessel loaded with the broken metal content but without 277-4 canned spacers between content locations, the $k_{eff} + 2\sigma$ values increase from 0.751 to 0.949 as the water content in the containment vessel increases [Cases **cvr3lha_36_1_8_1** through **cvr3lha_36_1_8_15** (Appendix 6.9.6, Table 6.9.6-10, and Appendix 6.9.3, Fig. 6.9.3.1-5)]. Cases **cvr3lha_36_1_8_15** through **cvr3lha_36_1_1_15** model the flooded containment vessel with 35 kg of broken HEU metal where the enrichment ranges from 100 to 19 wt % ^{235}U. The result for Case **cvr3lha_36_1_6_15** indicates that for an enrichment of 90 wt % ^{235}U, the mass loading cannot exceed 31,482 g in the absence of canned spacers. However, the application of this limit to higher enrichment material is non-conservative, as illustrated by the result for Case **cvr3lha_26_1_8_15**. (The results for cases highlighted in yellow indicate potential limits.) As the enrichment increases, the mass loading limit decreases (from 28,334 g to 25,894 g in this example) due to less ^{238}U for neutron absorption.

For the flooded containment vessel under full water reflection and loaded with the broken metal content and 277-4 canned spacers between content locations, the $k_{eff} + 2\sigma = 0.876$, Case **cvr3lha_36_2_8_15** [10 CFR 71.55(b)]. For packages loaded with the broken metal content and 277-4 canned spacers between content locations, the calculated $k_{eff} + 2\sigma$ value is 0.872, Case **ncsrbmt11_36_2_15** [10 CFR 71.55(d)]. Case **hcsrbmt12_36_2_15** (Appendix 6.9.6, Table 6.9.6-11) represents the HAC model of the damaged ES-3100 package, where the outer dimensions of the package are reduced accordingly and the entire package is flooded except the neutron poison of the body weldment liner inner cavity. This single-unit case with a MOIFR = 1.0 pertains specifically to the flooded drum under conditions specified in 10 CFR 71.55(e). The $k_{eff} + 2\sigma = 0.874$ for this HAC condition. The changes in both the outer dimensions of the package and the compositions of the Kaolite and 277-4 due to HAC result in an ~0.002 change in the neutron multiplication factor.

6.4.3 HEU Oxide

HEU product oxide content is non-hygroscopic or mildly hygroscopic with a bulk density over 6 times greater than water. HEU skull oxide is a less dense form of U_3O_8 intermixed with graphite. Of interest are those compositions with carbon/fissile uranium ($C/^{235}U$) ratios up to 2.3×10^5 μg $C/g^{235}U$. The quantity of fissile uranium is extremely low where even higher $C/^{235}U$ ratios are present.

HEU product oxide content is packed in polyethylene bottles; HEU skull oxide content is packed in metal convenience cans. While both the size and shape of the polyethylene bottles prevents the use of 277-4 canned spacers with product oxide loadings, the 10-in. height of the metal convenience cans eliminates the use of 277-4 canned spacers for skull oxide content packaged in these size cans. However, the convenience cans are not modeled in this criticality evaluation and this analysis is performed from the standpoint that metal-type convenience cans may be used for both oxide types.

HEU oxide content is not considered a "rigid" content like solid or broken HEU metal. Thus a different modeling approach is used to avoid having to address uncertainties regarding the location of spacers in the containment vessel. For the calculation models of HEU oxides, the 277-4 canned spacers are not actually modeled. Instead, the evaluation water in the void region of the containment vessel is homogenized over the volume of the containment vessel that is not occupied by oxide content. (The void region of the containment vessel is defined as the containment vessel volume minus the oxide content volume minus the canned spacer volume.) Because of the differences in densities and compositions, product oxides are addressed separately from skull oxides.

Like packages with HEU metal or broken metal, the neutron multiplication factor for reflected single packages with HEU oxide increases as a function of increasing MOIFR and decreases with decreasing ^{235}U mass. For example, consider the ES-3100 package loaded with three convenience cans for a total of 24,000 g UO_2 (21,125 g ^{235}U) but lacking 277-4 canned spacers between content locations. The $k_{eff} + 2\sigma$ values range from 0.700 to 0.784 with increasing MOIFR in the reflected package [Cases **ncsroxt11_1_24_1_1** through **ncsroxt11_1_24_1_15** (Appendix 6.9.6, Table 6.9.6-13)]. The addition of water to the package reduces the NLF, thereby increasing k_{eff}. Results for Cases **ncsroxt11_2_24_1_15** and **ncsroxt11_3_24_1_15** indicate that both the U_3O_8 content ($k_{eff} + 2\sigma = 0.678$) and the UO_3 content ($k_{eff} + 2\sigma = 0.619$) are bounded by the UO_2 content ($k_{eff} + 2\sigma = 0.784$).

For the containment vessel loaded with the HEU product oxide but lacking 277-4 canned spacers between content locations, the $k_{eff} + 2\sigma$ values decrease from 0.873 to 0.846 [Cases **cvcroxt11_1_24_1** through **cvcroxt11_1_20_1** (Appendix 6.9.6, Table 6.9.6-12, and Appendix 6.9.1, Fig. 6.9.1-5)]. Canned spacers are not required for criticality control. For the containment vessel loaded with the HEU product oxide and 277-4 canned spacers between content locations, the $k_{eff} + 2\sigma$ values decrease from 0.874 to 0.848 [Cases **cvcroxt11_1_24_2** through **cvcroxt11_1_20_2** (Appendix 6.9.6, Table 6.9.6-12)]. Comparison of results for a package without spacers (Case **cvcroxt11_1_*nn*_1**) versus results for one with spacers (Case **cvcroxt11_1_*nn*_2**) reveals that the amount of water removed by the spacers has an insignificant effect on the neutron multiplication factor.

For the flooded containment vessel under full water reflection containment, the $k_{eff} + 2\sigma$ value is 0.873, Case **cvcroxt11_1_24_1**. [10 CFR 71.55(b)] The $k_{eff} + 2\sigma$ value is 0.784 for water-reflected package, Case **ncsroxt11_1_24_1_15**. [10 CFR 71.55(d)] Case **hcsroxt12_1_24_1_15** (Appendix 6.9.6, Table 6.9.6-13) represents the HAC model of the damaged ES-3100 package, where the outer dimensions of the package are reduced accordingly and the entire package is flooded except the neutron poison of the body weldment liner inner cavity. This single-unit case with a MOIFR = 1.0 pertains specifically to the flooded drum under conditions specified in 10 CFR 71.55(e). The $k_{eff} + 2\sigma = 0.787$ for this HAC condition. The changes in both the outer dimensions of the package and the compositions of the Kaolite and 277-4 due to HAC result in an ~0.002 change in the neutron multiplication factor.

Cases **cvcrsk3cc_1_15_17** through **cvcrsk3cc_10_15_17** (Appendix 6.9.6, Table 6.9.6-17) evaluate the 10 skull oxide compositions identified Table 6.9.3.1-3b, where the containment vessel is loaded with 3 convenience cans. The fissile enrichment is ~70 wt % ^{235}U for contents "**sk3cc_1**" through "**sk3cc_5**" and ~38 wt % ^{235}U for contents "**sk3cc_6**" through "**sk3cc_8**." The observed maximums of 7.1 kg skull oxide and ~93.2 wt % ^{235}U enrichment present in 292 convenience cans are the basis for the hypothetical contents "**sk3cc_9**" and "**sk3cc_10**." Content "**sk3cc_10**" differs from "**sk3cc_9**" in that 921 g of graphite in the skull oxide is replaced with 921 g ^{235}U in the form of U_3O_8. Content "**sk3cc_10**" is more representative of high enrichment skull oxides where < 85 g of graphite is present. For each of the 10 content models, 513 g of the skull oxide content is modeled as polyethylene, representing the potential use of hydrogenous packing material. The U_3O_8 content ranges from 8 to 21 kg, while the ^{235}U content ranges from 3.7 to 16.4 kg. The graphite content ranges from 0 to 921 g C, and the C/^{235}U ratio ranges from 0 to 223,432 µg C/g ^{235}U. For Cases **cvcrsk3cc_1_15_17** through **cvcrsk3cc_10_15_17** at both saturation moisture and full graphite content, the $k_{eff} + 2\sigma$ values range from 0.759 to 0.855 in the flooded, water-reflected containment vessel. The results for Cases **cvcrsk3cc_9_15_17** and **cvcrsk3cc_10_15_17** reveal that $k_{eff} + 2\sigma$ increases slightly when the 921 g of carbon in the skull oxide is replaced with 921 g ^{235}U.

Cases **cvcrsk3cc_4_1_1** through **cvcrsk3cc_4_*m*_*g*** evaluate the effects of moisture and skull oxide graphite content on reactivity. The case designator "***m***" ranges from 1 to 15, signifying the variation in moisture; "***g***" ranges from 1 to 17, signifying the variation in graphite. Examination of results indicates that moisture content has a predominate effect on the neutron multiplication factor, while variation in graphite content has a minor effect. The peak $k_{eff} + 2\sigma$ value occurs at saturated moisture and full graphite content.

Cases **ncsrsk_1_15** through **ncsrsk_10_15** (Appendix 6.9.6, Table 6.9.6-18a) model the 10 skull oxide compositions in a flooded containment vessel in a water reflected package. The $k_{eff} + 2\sigma$ values range from 0.641 to 0.745. Likewise, Cases **hcsrsk_1_15** through **hcsrsk_10_15** (Appendix 6.9.6, Table 6.9.6-18b) evaluate infinite arrays damage packages. The $k_{eff} + 2\sigma$ values ranging from 0.658 to 0.767 are not statistically significantly different from the NCT case results.

Cases **ncsrsk_9_1** through **ncsrsk_9_15** evaluate the bounding skull oxide content containing ~19.9 kg U_3O_8 and 921 g graphite, where the saturation moisture is varied from 0 to 3801 g water. Likewise, Cases **ncsrsk_10_1** through **ncsrsk_10_15** evaluate the skull oxide content containing ~20.8 kg U_3O_8 where the 921 g of graphite in the skull oxide is replaced with 921 g ^{235}U in the form of U_3O_8. The $k_{eff} + 2\sigma$ values range from 0.403 to 0.741 in the former set of cases and range from 0.404 to 0.746 in the latter set of cases. The $k_{eff} + 2\sigma$ values are well below the USL value of 0.925, indicating that canned spacers are not required for criticality control. Moreover, these results indicate that moisture content has a predominate effect on the neutron multiplication factor, while the graphite content has an insignificant effect.

6.4.4 UNH Crystals

Although UNH crystal content is to be packed in Teflon bottles, this criticality evaluation does not preclude use of metal-type convenience cans that the 277-4 canned spacers are designed to accommodate. Therefore, 277-4 canned spacers are considered in the evaluation of UNH crystals. UNH crystal content is not considered a "rigid" content like solid or broken HEU metal. Therefore, the same modeling technique is used for HEU oxide content and UNH crystal content. The 277-4 canned spacers are not actually modeled.

Unlike the other HEU contents, UNH crystals are soluble in water. The most reactive credible configuration for a solution occurs for a concentration at ~450 g ^{235}U/l. Fixed by the inner dimensions of the containment vessel, the near optimum solution concentration occurs at a content mass loading of 9000 g UNH crystals. This condition assumes the mixing of UNH crystals with water of the flooded containment vessel, producing a solution with a concentration on the order of 415 g ^{235}U/l.

For a flooded containment vessel loaded with the UNH crystals but without 277-4 canned spacers between content locations, the $k_{eff} + 2\sigma$ values change from 0.823 to 0.609 [Cases **cvcrunhct11_24_1** through **cvcrunhct11_1_1** (Appendix 6.9.6, Table 6.9.6-14, and Appendix 6.9.1, Fig. 6.9.1-6)]. In the range of the optimal concentration (~9000 g UNH), the $k_{eff} + 2\sigma = 0.857$ for Case **cvcrunhct11_9_1**. [10 CFR 71.55(b)] Where dilution of UNH crystals is confined to the containment vessel, the reported $k_{eff} + 2\sigma$ value is 0.745 for a water-reflected package (Case **ncsrunhct11_9_1_15**). [10 CFR 71.55(d)]

Similar to packages loaded with HEU metal, broken metal, or oxide, the neutron multiplication factor for a reflected single package with UNH crystals increases as a function of MOIFR. Consider the ES-3100 package loaded with three convenience cans for a total of 24,000 g UNH crystals (11,303 g ^{235}U). The concentration is 1106 g ^{235}U/l in the flooded containment vessel. For Cases **ncsrunhct11_24_1_1** through **ncsrunhct11_24_1_15** (Appendix 6.9.6, Table 6.9.6-15), the $k_{eff} + 2\sigma$ values range from 0.605 to 0.702 with increasing MOIFR in the water-reflected package. However, the results for Cases **ncsrunhct11_24_1_15** through **ncsrunhct11_1_1_15** demonstrate that the optimal concentration is not reached until the mass loading is reduced to ~9000 g UNH or 415 g ^{235}U/l. For Case **ncsrunhct11_9_1_15**, the $k_{eff} + 2\sigma = 0.745$.

Unlike solid HEU metal and oxide content, which are confined to the containment vessel and only water leakage into the containment vessel need be considered, the evaluation of UNH crystal content for compliance with 10 CFR 71.55(b) also requires that the leakage of liquid HEU contents out of the containment be addressed. This is because UNH crystals are soluble in water and would dissolve in the water flooding the containment vessel and the containment vessel well. For a content loading of 24,000 g UNH, the uranium concentration drops from 1106 g ^{235}U/l to 689 g ^{235}U/l. Optimum moderation at solution concentrations in the range of 415 g ^{235}U/l to 429 g ^{235}U/l occurs for fissile mass loadings of ~15,000 g UNH.

Cases **hcsrunhct12_9_1_15** (Appendix 6.9.6, Table 6.9.6-15) and **icsrunhct12_15_1_15** (Appendix 6.9.6, Table 6.9.6-16) represent the HAC model of the damaged ES-3100 package where the outer dimensions of the package are reduced accordingly. For Case **hcsrunhct12_9_1_15**, the dilution of the UNH crystals is confined to the containment vessel, and maximum reactivity occurs at ~9000 g UNH. For Case **icsrunhct12_15_1_15**, UNH crystals are dissolved in the water flooding the containment vessel and the containment vessel well, and maximum reactivity occurs at ~15,000 g UNH. The respective solution concentrations are 415 g ^{235}U/l and 429 g ^{235}U/l. These single-unit cases with a MOIFR = 1.0 pertain specifically to the flooded drum under conditions specified in 10 CFR 71.55(e), where the $k_{eff} + 2\sigma$ values are 0.748 and 0.807 for this HAC condition. The changes in both the outer dimensions of the package and the compositions of the Kaolite and 277-4 due to HAC do not result in a significant change in the neutron multiplication factor.

6.5 EVALUATION OF PACKAGE ARRAYS UNDER NORMAL CONDITIONS OF TRANSPORT

For the NCT array evaluation of ES-3100 packages, the package content is confined within the containment vessel, consistent with the result of the tests specified in §71.71 (Normal Conditions of Transport). The array sizes examined in this evaluation are infinite, 13×13×6, 9×9×4, 7×7×3, 5×5×2, ETP 27×3, and the degenerate single unit. The "N" and corresponding CSI values for arrays determined to be adequately subcritical are as follows: N = ∞, CSI = 0; N = 202, CSI = 0.3; N = 64, CSI = 0.8; N = 29, CSI = 1.7; ~~N = 10, CSI = 5.0;~~ and N = 16, CSI = 3.2. All arrays, except the infinite array, are reflected with 30 cm (1 ft) of water. These arrays are nearly cubic in shape for optimum array reactivity, thus eliminating the need for placing criticality controls on package arrangements in terms of stack height, width, and depth of an array. The array configurations and the range of water contents (Table 6.4) evaluated bound all possible packaging arrangements and moderation conditions for NCT.

6.5.1 Solid HEU Metal of Specified Geometric Shapes

For infinite and finite arrays of packages with HEU metal, the neutron multiplication factor increases as a function of the ^{235}U mass and decreases as a function of MOIFR. For example, consider the ES-3100 package loaded with three convenience cans for a total of 36,000 g ^{235}U where each can contains a 3.24-in.-diam cylinder. For package content without 277-4 canned spacers, the $k_{eff} + 2\sigma$ values range from 1.027 to 0.963 with increasing MOIFR in the package [Cases **nciacyt11_36_1_1** through **nciacyt11_36_1_15** (Appendix 6.9.6, Table 6.9.6-3)]. For package content with 277-4 canned spacers, the $k_{eff} + 2\sigma$ values range from 0.957 to 0.880 with increasing MOIFR in the package [Cases **nciacyt11_36_2_1** through **nciacyt11_36_2_15** (Appendix 6.9.6, Table 6.9.6-3)].

The effect of increasing the water content of the array is straightforward. As interspersed water is added to the packages of an array, two reactivity effects occur in series. The first effect is the tendency for reactivity to remain constant due to controlled neutron interaction between the packages of the array. For an infinite array where neutrons cannot escape from the system, neutrons are scattered about the array. In the MOIFR range of 1e-20 to 1e-02, both the interspersed moderator inside the packages of the dry array and the interstitial moderator between the package drums of the array are not sufficient for neutron thermalization and absorption to occur in the adjacent packaging materials. However, hydrogen in the 277-4 provides moderation, and neutrons are absorbed in the interspersed boron of this neutron poison. This results in a subcritical system with near constant neutron multiplication factors over the range of MOIFR. The second effect is the tendency for reactivity to decrease due to internal moderation in packages of the array. The introduction of water above ~0.01 MOIFR shows the effect of isolating the individual array units from each other. The neutron multiplication factor approaches k_{eff} for the single, water-reflected unit at a full-content water fraction (MOIFR = 1.0).

The array case with a water fraction of MOIFR = 1e-04 pertains specifically to packages under NCT where the Kaolite and recesses of the package external to the containment vessel do not contain any residual moisture. This NCT case is more reactive than all other NCT cases where more moisture is present in the Kaolite and recesses of the package. Interspersed water between the containment vessels in the array will reduce neutronic interaction between the flooded contents because neutrons are absorbed in the hydrogen of the water. As more water is added, the packages of the array become isolated, and array reactivity ($k_{eff} + 2\sigma = 0.880$, Case **nciacyt11_36_2_15**) approaches the reactivity of the single unit ($k_{eff} + 2\sigma = 0.873$, Case **ncsrcyt11_36_2_15**).

Repeated for 4.25-in.-diam cylinders (Appendix 6.9.6, Table 6.9.6-7); 2.29-in. -square bars (Table 6.9.6-5); and 1.5-in.-diam × 2-in.-tall slugs (Table 6.9.6-9), this type of analysis for the 3.24-in.-diam cylinders demonstrates that arrays of packages with restricted fissile material (^{235}U) loading remain subcritical over the entire range of water content or MOIFR. HEU bulk metal or alloy content not covered by the specified geometric shapes (cylinder, square bar, or slug contents) will be in the HEU broken metal category, and so limited.

Cases **nciatriga_1_1_3** through **nciatriga_1_15_3** (Appendix 6.9.6, Table 6.9.6-20a) represent infinite arrays of packages containing the bounding TRIGA fuel content (20 % enrichment with 45 wt % U containing 307 g ^{235}U), without 277-4 canned spacers. The MOIFR is set at 1.0e-04 such that neutronic interaction between packages is maximized. For these cases, the $k_{eff} + 2\sigma$ values increase from 0.218 to 0.525 as MOIFR increases. The $k_{eff} + 2\sigma = 0.525$ for Case **nciatriga_1_15_3** is substantially below the USL value of 0.925, indicating that canned spacers are not required for criticality control.

As stated in Sect. 6.4.1, 10 CFR 71.55(d)(2) requires the geometric form of a package's content not be substantially altered under the NCT. Similarly, 10 CFR 71.55(e)(1) requires that the package be adequately subcritical under HAC with the package contents in the most reactive credible configuration. Even though visible signs of damage to the metal convenience can have not been observed resulting from the regulatory tests, conclusions about damage to the TRIGA fuel content are not extrapolated from test data. The regulatory requirements are addressed by modeling the TRIGA fuel content in an extremely damaged condition. A series of criticality calculations is performed for making a determination of subcriticality. Cases **nciat55d2_1_1_3** through **ncia55d2_1_15_3** (Appendix 6.9.6, Table 6.9.6-20d) represent TRIGA fuel content homogenized with variable density water over the free volume of the containment vessel. The packaging is flooded in each ES-3100 package of the infinite array. The variable density water ranges from the dry containment condition to the fully flooded condition. Credit for physical integrity of the content is not taken in this set of cases which model the substantially altered content. The calculation results in Table 6.9.6-20d indicate extremely damaged content (Case **nciat55d2_1_15_3** with $k_{eff} + 2\sigma = 0.716$) is more reactive than the unaltered configuration (Case **nciatriga_1_15_3** with $k_{eff} + 2\sigma = 0.442$). Nevertheless, both cases are adequately below the USL of 0.925 and the requirement of 10 CFR 71.55(d)(2) is satisfied. Given that changes external to the containment vessel due to the HAC do not result in an appreciable change in neutron multiplication for an array of packages, similar results are expected for the cases demonstrating compliance with 10 CFR 71.55(e)(1).

For TRIGA fuel content as clad fuel rods, the amount of clad added (stainless steel as a neutron absorber or aluminum as a neutron scatter) and corresponding amount of water moderator displaced by the clad is not expected to have a statistically significant affect on the calculated k_{eff}. Thus, the NCT shipping configuration model for disassembled TRIGA fuel (bare fuel meats) bounds shipping configuration model for TRIGA fuel configured as clad fuel rods (Appendix 6.9.3.1).

The array results for three slug configurations presented in Table 6.9.6-9 (Appendix 6.9.6) are for 5 or 10 slugs spaced apart in a pentagonal ring (**ncia5est11**) and for 7 slugs formed by a hexagonal ring of slugs with one slug in the center of the ring (**ncia70st11**). These cases are used to establish the mass loading limitations, which in turn limit the number of slugs in the package to less than the number required to assemble a critical configuration.

Cases **ncia5est11_1_1_8_3** through **ncia5est11_1_1_1_3** (Appendix 6.9.6, Table 6.9.6-9) represent infinite arrays of packages containing 18,287 g U without 277-4 canned spacers. For these cases, the $k_{eff} + 2\sigma$ values increase from 0.550 to 0.923 as the enrichment is increased from 19.0 wt % to 100.0 wt % ^{235}U. The $k_{eff} + 2\sigma = 0.923$ for Case **ncia5est11_1_1_8_3** is below the USL of 0.925.

Cases **ncia5est11_2_1_8_3** through **ncia5est11_2_1_1_3** (Appendix 6.9.6, Table 6.9.6-9) represent infinite arrays of packages containing 36,573 g ^{235}U without 277-4 canned spacers. For these cases, the $k_{eff} + 2\sigma$ values increase from 0.452 to 0.806 as the enrichment is increased from 19.0 wt % to 100.0 wt % ^{235}U. The $k_{eff} + 2\sigma = 0.806$ value for Case **ncia5est11_1_2_8_3** does not exceed the USL of 0.925.

At 60 wt % ^{235}U, the $k_{eff} + 2\sigma = 0.905$ value for Case **ncia5est11_1_2_3_3** is below the USL. However, a restriction placed upon mass that requires that values must be $\leq$18,287 g ^{235}U still applies to satisfy the subcriticality requirement for the reflected containment vessel.

Cases **ncia70st11_1_8_3** through **ncia70st11_1_1_3** (Appendix 6.9.6, Table 6.9.6-9) represent infinite arrays of packages containing 25,601 g U without 277-4 canned spacers. For these cases, the $k_{eff} + 2\sigma$ values increase from 0.530 to 1.025 as the enrichment is increased from 19.0 wt % to 100.0 wt % ^{235}U. The $k_{eff} + 2\sigma = 1.025$ for Case **ncia70st11_1_8_3** exceeds the USL of 0.925.

At 70 wt % ^{235}U, the $k_{eff} + 2\sigma = 0.884$ value for Case **ncia70st11_1_4_3** is below the USL. The 17,989 g ^{235}U falls within the restriction placed upon mass that requires that values must be $\leq$18,287 g ^{235}U to satisfy the subcriticality requirement for the reflected containment vessel.

Cases **ncia5est11_2_2_8_3** through **ncia5est11_2_2_1_3** (Appendix 6.9.6, Table 6.9.6-9) represent infinite arrays of packages containing 36,573 g ^{235}U with 277-4 canned spacers. For these cases, the $k_{eff} + 2\sigma$ values increase from 0.582 to 0.983 as the enrichment is increased from 19.0 wt % to 100.0 wt % ^{235}U. At 80 wt % ^{235}U, the $k_{eff} + 2\sigma = 0.909$ value for Case **ncia5est11_2_2_5_3** is just below the USL. Therefore, a restriction on mass and enrichment for slug content is that for $\leq$80 wt % ^{235}U, the mass of ^{235}U in the package must not exceed 29,333 g as a prerequisite for the shipment of the package slug content and with 277-4 canned spacers under a CSI = 0.0.

Cases **ncia70st11_2_8_3** through **ncia70st11_2_1_3** (Appendix 6.9.6, Table 6.9.6-9) represent infinite arrays of packages containing 25,601 g ^{235}U with 277-4 canned spacers. The $k_{eff} + 2\sigma$ values increase from 0.473 to 0.914 as the enrichment is increased from 19.0 wt % to 100.0 wt % ^{235}U. Therefore, the restriction placed on mass and enrichment for slug content is that for between 80 and 100 wt % ^{235}U, the mass of ^{235}U in the package must not exceed 25,601 g as a prerequisite for the shipment of the package with slug content and 277-4 canned spacers under a CSI = 0.0.

Cases **ncf15est11_2_2_8_3** through **ncf15est11_2_2_1_3** (Appendix 6.9.6, Table 6.9.6-9) represent a 13 × 13 × 6 array of packages for which the corresponding rounded CSI = 0.4. The $k_{eff} + 2\sigma = 0.941$ for Case **ncf15est11_2_2_8_3** at 100 wt % ^{235}U exceeds the USL of 0.925. Case **ncf15est11_2_2_7_3** at 95 wt % ^{235}U with $k_{eff} + 2\sigma = 0.921$ is below the USL of 0.925 to permit increasing the limit on enrichment for mass loadings of ≤34.7 kg uranium metal. These CSI determinations are contingent upon satisfactory results under the HAC evaluation (Sect. 6.6.1).

6.5.2 HEU Solid Metal of Unspecified Geometric Shapes or HEU Broken Metal

Like packages with HEU metal, the neutron multiplication factor for arrays of packages with HEU broken metal decreases as a function of MOIFR and increases as a function of the ^{235}U mass. For example, consider the ES-3100 package loaded with three convenience cans for a total of 35,142 g ^{235}U and no canned spacers between content locations. The $k_{eff} + 2\sigma$ values range from 1.138 to 0.913 with increasing MOIFR [Cases **nciabmt11_36_1_8_1** through **nciabmt11_36_1_8_15** (Appendix 6.9.6, Table 6.9.6-11)]. The introduction of water above ~0.01 MOIFR shows the effect of isolating the individual array units from each other. Array reactivity ($k_{eff} + 2\sigma = 0.913$) approaches the reactivity of the water-saturated, water-reflected single package Case **ncsrbmt11_36_1_15** ($k_{eff} + 2\sigma = 0.891$).

In the series of calculations using the ES-3100 package model with NCT geometry (Cases **nciabmt11_1_*n*_*m*_3** through **nciabmt11_*36*_*n*_*m*_3**), the enrichment of the content is varied from 19 wt % to 100 wt % ^{235}U. These array cases with a water fraction of MOIFR = 1e-04 pertain specifically to NCT packages where both the neutron poison of the body weldment liner inner cavity and the Kaolite are dry (in the as-manufactured condition) and both the recesses of the package external to the containment vessel and the interstitial space between the drums of the array do not contain any residual moisture. As stated before, this NCT case is more reactive than all other NCT cases where more moisture is present in the Kaolite and recesses of the package. Increased interspersed water between the containment vessels in the array will reduce neutronic interaction between the flooded contents to a point where the packages of the array become isolated.

Ranges of enrichment are specified in Table 6.1b (10 CFR 71.59) for identifying fissile mass loading limits for HEU broken metal. Consider specifically enrichments >95 wt % ^{235}U. The containment vessel calculations (Case **cvr3lha_36_1_8_15** versus Case **cvr3lha_36_2_8_15**) indicate that 277-4 canned spacers are required in this enrichment range, where the maximum evaluated fissile mass loading of 35,142 g ^{235}U is possible. However, the fissile mass loading must be limited to 2,774 g ^{235}U (Case **nciabmt11_3_2_8_3**) in order for the $k_{eff} + 2\sigma$ value (= 0.904) to be below the USL of 0.925. This fissile mass limit is conservative when applied to enrichments only slightly greater than 95 wt % ^{235}U. A reduction in the enrichment within the range of 80 to 95 wt % ^{235}U (Cases **nciabmt11_4_2_7_3** and **nciabmt11_4_2_6_3**) does not result in a sufficient reduction in the $k_{eff} + 2\sigma$ from neutron absorption in ^{238}U to allow for increased mass loadings. Therefore, the uranium mass limit remains at ~2774 g, while the fissile mass loading limit decreases with the reduction in enrichment as illustrated in Table 6.1b. As stated previously, these fissile mass loading limits for a CSI = 0 are contingent upon the infinite array of damaged packages also being adequately subcritical for HAC (Sect. 6.6.2).

This evaluation technique for determination of mass loading limits for enrichment intervals is repeated over the range of HEU enrichments identified in Table 6.1b. At HEU enrichment <60 wt % ^{235}U, the evaluated package mass loading limit of 35 kg uranium is achieved, so further delineation of fissile mass loading limits is not required.

6.5.3 HEU Oxide

Like packages with HEU metal or broken metal, the neutron multiplication factor for an array of packages with product oxide decreases as a function of decreasing ^{235}U mass [Cases **nciaoxt11_1_24_1_3** through **nciaoxt11_1_16_1_3** (Appendix 6.9.6, Table 6.9.6-13)]. The $k_{eff} + 2\sigma$ values are below the USL of 0.925.

Cases **nciask_1_15** through **nciask_10_15** (Appendix 6.9.6, Table 6.9.6-18) evaluate infinite arrays of packages having the 10 skull oxide compositions described in Sect. 6.4.3. The MOIFR is set at 1.0e-04 such that neutronic interaction between packages is maximized. The $k_{eff} + 2\sigma$ values, ranging from 0.656 to 0.764, are substantially below the below the USL value of 0.925, indicating that canned spacers are not required for criticality control.

The CSI = 0.0 for an infinite array of packages having product oxide content with a maximum of 21,125 g ^{235}U or having skull oxide content with a maximum of 16,399 g ^{235}U and 921 g graphite. The suitability of this determination is contingent upon satisfactory results under the HAC evaluation (Sect. 6.6.3.)

6.5.4 UNH Crystals

Unlike HEU metal, broken metal or oxide content, UNH crystal is soluble in water (Sect. 6.4.4). Near optimum solution concentration occurs at content mass loadings of ~9000 g UNH crystals or 415 g $^{235}U/l$. Like packages with HEU metal, broken metal, or oxide, the neutron multiplication factor for an array of packages with UNH crystals decreases as a function of increasing MOIFR. Cases **nciaunhct11_8_24_1_1** through **nciaunhct11_8_24_1_15** (Appendix 6.9.6, Table 6.9.6-15) reveal the effect of isolating the individual array units from each other with the introduction of water above ~0.01 MOIFR. Array reactivity ($k_{eff} + 2\sigma = 0.721$) approaches the reactivity of the water-saturated, water-reflected unit single package ($k_{eff} + 2\sigma = 0.702$).

The difference between Cases **nciaunhct11_8_*nn*_1_3** and **nciaunhct11_8_*nn*_2_3** is the absence of 277-4 canned spacers in the former case and the presence of 277-4 canned spacers in the latter case. The 277-4 canned spacers are not actually modeled in the NCT package calculation models for UNH crystals content. Instead, the amount of solution content distributed over the entire containment vessel is reduced proportionally by the free volume of the containment vessel not occupied by canned spacers.

In a series of NCT calculations (Cases **nciaunhct11_8_*nn*_1_3**), the $k_{eff} + 2\sigma$ values are above the USL of 0.925 for mass loadings between 9,000 and 21,000 g UNH crystals. Although mass loadings above 21,000–24,000 are adequately subcritical, credit is not taken for water tightness that is demonstrated by the NCT (and HAC) tests. Therefore, mass loading are restricted to <8 kg UNH crystals. The CSI = 0.0 [10 CFR 71.59(a)(1) and 10 CFR 71.59(b)] for ES-3100 packages having a maximum of 3789g ^{235}U. The suitability of this determination depends upon satisfactory results under the HAC evaluation of Sect. 6.6.4. [10 CFR 71.59(a)(2) and 10 CFR 71.59(b)]

The $k_{eff} + 2\sigma$ values are expected to range from subcritical to critical for array configurations where UNH crystals are dissolved in the water flooding the containment vessel and the containment vessel well. It requires intrusion of water into the containment vessel, dissolving of UNH crystals in the influent, and leakage of solution out of the containment vessels in each package of an array. This condition is an accident condition, not credible for NCT.

For Cases **ncf1unhct12_8_*nn*_1_3**, the $k_{eff} + 2\sigma$ values are below the USL of 0.925. Given that the results for NCT (Sect. 6.5.4) and HAC are adequately subcritical, packages may be shipped under a CSI = 0.4 for packages with a maximum of UNH crystals. The CSI = 0.4 [10 CFR 71.59(a)(1) and 10 CFR 71.59(b)] for ES-3100 packages having a maximum of 24,000 g UNH crystal. Likewise, the suitability of this determination depends upon satisfactory results under the HAC evaluation of Sect. 6.6.4.

6.6 EVALUATION OF PACKAGE ARRAYS UNDER HYPOTHETICAL ACCIDENT CONDITIONS

Except for UNH crystals, the package content is confined within the containment vessel for the HAC array evaluation of ES-3100 packages, consistent with the result of the tests specified in 10 CFR 71.73 (Hypothetical Accident Conditions). The array sizes examined in this evaluation are infinite, 13×13×6, 9×9×4, 7×7×3, 5×5×2, ETP 16×3, and the degenerate single unit. The "N" and corresponding CSI values for arrays determined to be adequately subcritical are as follows: N = ∞, CSI = 0; ~~N = 507, CSI = 0.1;~~ N =162, CSI = 0.4; N = 73, CSI = 0.7; N = 25, CSI = 2.0; and N = 24, CSI = 2.1. All arrays, except the infinite array, are reflected with 30 cm (1 ft) of water. These array are nearly cubic in shape for optimum reactivity of the array, thus eliminating the need for placing criticality controls on package arrangements in terms of stack height, width, and depth of an array. The array configurations and the range of water contents (Table 6.4) evaluated bound all possible packaging arrangements and moderation conditions for HAC.

For the single damaged package, and for infinite and finite arrays of damaged packages with HEU metal, HEU oxide, or UNH crystal content, the neutron multiplication factor changes as a function of the ^{235}U mass, MOIFR, or applicable solution concentration in the same manner as in an array of undamaged packages.

6.6.1 Solid HEU Metal of Specified Geometric Shapes

For infinite and finite arrays of damaged packages with HEU metal, the neutron multiplication factor increases as a function of the ^{235}U mass and decreases as a function of MOIFR. For example, consider the ES-3100 package loaded with three convenience cans that contain a 3.24-in.-diam cylinder with 12,000 g ^{235}U for a total of 36,000 g ^{235}U. For package content without 277-4 canned spacers, the $k_{eff} + 2\sigma$ values range from 1.027 to 0.967 with increasing MOIFR in the damaged packages [Cases **hciacyct12_36_1_1** through **hciacyt12_36_1_15** (Appendix 6.9.6, Table 6.9.6-3)]. For package content with 277-4 canned spacers, the $k_{eff} + 2\sigma$ values range from 0.956 to 0.884 with increasing MOIFR in the damaged package [Cases **hciacyt12_36_2_1** through **hciacyt12_36_2_15** (Appendix 6.9.6, Table 6.9.6-3)].

The introduction of water above ~0.01 MOIFR shows the effect of isolating the individual array units from each other. The neutron multiplication factor approaches k_{eff} for the single, water-reflected unit at a full content-water fraction (MOIFR = 1.0). Comparison of these results with the corresponding NCT cases (Sect. 6.5.1.) indicates no significant differences. This result is as expected given that the neutron multiplication in an infinite array is independent of pitch between fissile contents but is dependent on changes in mass and moderation in the array. The changes in both the outer dimensions and the compositions of the Kaolite and 277-4 of the ES-3100 package due to HAC result in changes to the neutron multiplication factor on the order of ~0.001 to 0.002.

Repeated for 4.25-in.-diam cylinders (Appendix 6.9.6, Table 6.9.6-7); 2.29-in.-square bars (Table 6.9.6-5); and 1.5-in.-diam × 2-in.-tall slugs (Table 6.9.6-9), this type of analysis for the 3.24-in.-diam cylinders demonstrates that arrays of damaged packages with restricted fissile material (^{235}U) loading remain subcritical over the entire range of water content or MOIFR. HEU bulk metal or alloy content not covered by the specified geometric shapes (cylinder, square bar, or slug contents) will be in the HEU broken metal category, and so limited.

Cases **hciatriga_1_1_3** through **hciatriga_1_15_3** (Appendix 6.9.6, Table 6.9.6-20) represent infinite arrays of damaged packages containing the bounding TRIGA fuel content (20 % enrichment with 45 wt % U containing 307 g ^{235}U), without 277-4 canned spacers. For these cases, the $k_{eff} + 2\sigma$ values increase from 0.218 to 0.526 as MOIFR increases. The $k_{eff} + 2\sigma = 0.526$ for Case **hciatriga_1_15_3** is substantially below the USL of 0.925, indicating that canned spacers are not required for criticality control. Given that the results for NCT (Sect. 6.5.2) and HAC are adequately subcritical, packages with TRIGA fuel content ≤921 g ^{235}U may be shipped under a CSI = 0.

As stated in Sect. 6.5.1, calculation results presented in Table 6.9.6-20d for an infinite array of ES-3100 packages (NCT packaging with extremely damaged TRIGA fuel content) is substantially below the USL of 0.925. Similar results are expected for an infinite array of ES-3100 packages (HAC packaging with extremely damaged TRIGA fuel content) given that changes external to the containment vessel due to the HAC do not result in an appreciable change in the neutron multiplication for the an array of packages. Therefore, the 10 CFR 71.55(e)(1) requirement that the package be adequately subcritical under HAC with the package contents in the most reactive credible configuration is satisfied.

Consider the ~1090 g, 1.5-in.-diam x 2-in.-tall slugs that may be packed ten per convenience can. Cases **hcf25est11_2_2_8_3** through **hcf25est11_2_2_1_3** (Appendix 6.9.6, Table 6.9.6-9) represent a 9×9×4 array of packages for which the corresponding rounded CSI = 0.4. Case **hcia5est12_2_2_8_3** at 100 wt % with $k_{eff} + 2\sigma = 0.923$ is below the USL. Therefore, the restriction upon enrichment under the NCT evaluation, which requires values be ≤80 wt % as prerequisite for shipment of the package under a CSI = 0, need not change. Likewise, the fissile mass loading limit must be reduced to 18,287 g ^{235}U for eliminating the restriction on enrichment.

Case **hcf25est12_2_2_8_3** at 100 wt % ^{235}U (Appendix 6.9.6, Table 6.9.6-9) represents a 9 × 9 × 4 array of packages for which the corresponding rounded CSI = 0.4. Although Case **ncf25est12_2_2_7_3** at 95 wt % ^{235}U with $k_{eff} + 2\sigma = 0.905$ is below the USL of 0.925, the fissile mass loading limit is conservatively set by the NCT result, which requires that the enrichment not exceed 80 wt % ^{235}U for 36 kg uranium mass loadings. The $k_{eff} + 2\sigma$ decreases sufficiently when the array size is reduced to permit increasing the limit on enrichment to ≤95 wt % ^{235}U for mass loadings of 34.7 kg uranium metal. These CSI determinations are contingent upon satisfactory results under the HAC evaluation (Sect. 6.6.1).

The changes in both the outer dimensions of the package and the compositions of the Kaolite and 277-4 due to HAC result in an ~0.001 change in the neutron multiplication factor.

6.6.2 HEU Solid Metal of Unspecified Geometric Shapes or HEU Broken Metal

Consider the ES-3100 package loaded with three convenience cans for a total of 35,142 g ^{235}U with no canned spacers between content locations. The $k_{eff} + 2\sigma$ values range from 1.14 to 0.939 with increasing MOIFR, Cases **hciabmt12_36_1_8_1** through **hciabmt12_36_1_8_15** (Appendix 6.9.6, Table 6.9.6-11). The introduction of water above ~0.01 MOIFR shows the effect of isolating the individual array units from each other. Array reactivity ($k_{eff} + 2\sigma = 0.939$) approaches the reactivity of the water-saturated, water-reflected single package ($k_{eff} + 2\sigma = 0.891$, Case **hcsrbmt12_36_1_15**, Table 6.9.6-11).

Cases **hciabmt12_1_*nn_mm*_3** through **hciabmt12_36_*nn_mm*_3** model the ES-3100 with the reduced-diameter HAC package model where the enrichment of the content is varied from 60 wt % to 100 wt % ^{235}U. These array cases with MOIFR = 1e-04 pertain specifically to HAC packages where the neutron poison of the body weldment liner inner cavity is at 90% moisture content, but the Kaolite is dry (in the as-manufactured condition), and neither recess of the package external to the containment vessel and the interstitial space between the drums of the array contains any residual moisture. As stated before, this HAC case is more reactive than all other HAC cases where more moisture is present in the Kaolite and recesses of the package. Increasing the interspersed water between the containment vessels in the array will reduce neutronic interaction between the flooded contents to a point where the packages of the array become isolated.

Ranges of enrichment are specified in Table 6.1b (10 CFR 71.59) for identifying fissile mass loading limits for HEU broken metal. Consider specifically enrichments >95 wt % ^{235}U. The containment vessel calculations (Case **cvr3lha_36_1_8_15** versus Case **cvr3lha_36_2_8_15**, Appendix 6.9.6, Table 6.9.6-10) indicate that 277-4 canned spacers are required in this enrichment range, where the maximum evaluated mass loading of 35,142 g ^{235}U is possible. However, the fissile mass loading must be limited to 2774 g ^{235}U (Case **hciabmt12_3_2_8_3**) in order for the $k_{eff} + 2\sigma$ value (= 0.905) to be below the USL of 0.925. This fissile mass limit is conservative when applied to enrichments only slightly greater than 95 wt % ^{235}U. Given that the results for NCT (Sect. 6.5.2) and HAC are adequately subcritical, packages ≤2774 g ^{235}U and enrichment >95 wt % ^{235}U may be shipped under a CSI = 0.

This evaluation technique for determination of mass loading limits for enrichment intervals is repeated over the range of HEU enrichments. At HEU enrichment <60 wt % ^{235}U, the package mass loading limit is achieved, so no further delineation is required.

6.6.3 HEU Oxide

Like packages with HEU metal or broken metal, the neutron multiplication factor for an array of packages with HEU oxide decreases as a function of decreasing ^{235}U mass [Cases **hciaoxt12_1_24_1_3** through **hciaoxt12_1_16_1_3** (Appendix 6.9.6, Table 6.9.6-13)]. The $k_{eff} + 2\sigma$ values are below the USL of 0.925. Given that the results for NCT (Sect. 6.5.3) and HAC are adequately subcritical, packages may be shipped under a CSI = 0.0 for an infinite array of packages having product oxide with a maximum of 21,125 g ^{235}U.

Cases **hciask_1_15** through **hciask_10_15** (Appendix 6.9.6, Table 6.9.6-18b) evaluate infinite arrays of damaged packages having the 10 skull oxide compositions described in Sect. 6.4.3. The differences between the HAC case results and the NCT case results are not statistically significant. The $k_{eff} + 2\sigma$ values, ranging from 0.658 to 0.767, are substantially below the USL value of 0.925. Given that the results for NCT (Sect. 6.5.3) and HAC are adequately subcritical, packages may be shipped under a CSI = 0.0 for an infinite array of packages having skull oxide content with a maximum of 16,399 g ^{235}U and 921 g graphite.

6.6.4 UNH Crystals

Unlike HEU metal, broken metal or oxide content, UNH crystal is soluble in water, as mentioned in Sect. 6.4.4. Near optimum solution concentration occurs at content mass loadings of 9000 g UNH crystals or 415 g ^{235}U/l. Like packages with HEU metal, broken metal, or oxide, the neutron multiplication factor for an array of damaged packages with UNH crystals decreases as a function of increasing MOIFR. Cases **hciaunhct12_8_24_1_1** through **hciaunhct12_8_24_1_15** (Appendix 6.9.6, Table 6.9.6-15) reveal the effect of isolating the individual array units from each other with the introduction of water above ~0.01 MOIFR. Array reactivity ($k_{eff} + 2\sigma = 0.729$) approaches the reactivity of the water-saturated, water-reflected unit single package ($k_{eff} + 2\sigma = 0.705$).

Like the NCT evaluation, the difference in the HAC package calculation model represented by Cases **hciaunhct12_8_*nn*_1_3** and **hciaunhct12_8_*nn*_2_3** is the absence of 277-4 canned spacers in the former case and the presence of 277-4 canned spacers in the latter case. For the HAC package calculation models with UNH crystal content, the 277-4 canned spacers are not actually modeled. Instead, the amount of solution content distributed over the entire containment vessel is reduced proportionally by the free volume of the containment vessel not occupied by canned spacers.

In a series of HAC calculations, Cases **hciaunhct11_8_*nn*_1_3**, the calculated $k_{eff} + 2\sigma$ values are below the USL of 0.925 for loading with <9000 g UNH crystals. Given that the results for NCT (Sect. 6.5.4) and HAC are adequately subcritical, the CSI = 0.0 [10 CFR 71.59(a)(2) and 10 CFR 71.59(b)] for packages having a maximum of 3789g ^{235}U crystal.

The $k_{eff} + 2\sigma$ values are expected to range from subcritical to critical for array configurations where UNH crystals are dissolved in the water flooding the containment vessel and the containment vessel well. It requires intrusion of water into the containment vessel, dissolving of UNH crystals in the influent, and leakage of solution out of the containment vessels in each package of an array. The leakage out of the containment vessel of content moderated "to such an extent as to cause maximum reactivity consistent with the physical and chemical form of material" is not considered credible HAC based on results for tests specified in 10 CFR 71.73.

In a series of HAC calculations, Cases **hcf2unhct121_8_*nn*_1_3**, the calculated $k_{eff} + 2\sigma$ values are below the USL of 0.925 for loading with <9000 g UNH crystals. Given that the results for NCT (Sect. 6.5.4) and HAC are adequately subcritical, packages may be shipped under a CSI = 0.4 for packages with a maximum of 24,000 g UNH crystals.

6.7 FISSILE MATERIAL PACKAGES FOR AIR TRANSPORT

A series of calculations are performed for determining the most reactive configuration of the content and surrounding packaging material in an ES-3100 package that undergoes catastrophic destruction. Fissile content mass loadings that remain under the USL for these catastrophic events are identified. Subcriticality is demonstrated after due consideration of such aspects as the efficiency of moderator, loss of neutron absorbers, rearrangement of packaging components and contents, geometry changes and temperature effects. The seven calculation models used in this evaluation are described in Sect 6.3.1.4. Key model dimensions and parameters are tabulated in Tables 6.9.6-21 through 6.9.6-23, Appendix 6.9.6.

6.7.1 Results for Solid HEU, One Piece per Convenience Can

Cases **atdmr_10_8** through **atdmr_7_1** (Table 6.9.6-21, Appendix 6.9.6) pertain to Model 1 (Fig. 6.11, Section 6.3.1.4). HEU solid or broken metal of three convenience cans is homogenized with 513 g of polyethylene, representing the potential use of hydrogenous packing materials in the ES-3100. The fissile core is configured into a spherical shape with an exterior 20 cm water reflector. As shown in

Fig. 6.17 and Table 6.9.6-21, the $k_{eff} + 2\sigma$ values decrease as a function of both enrichment and ^{235}U mass. The $k_{eff} + 2\sigma$ values are substantially below the USL of 0.925. The $k_{eff} + 2\sigma$ value for 7 kg of HEU at 100 % enrichment is 0.792.

Cases **atdmsr_7_8_11** through **atdmsr_7_1_1** (Table 6.9.6-21, Appendix 6.9.6) pertain to Model 2 (Fig. 6.12, Section 6.3.1.4). The fissile core of Model 1 is blanketed with a variable thickness stainless steel shell and an exterior 20 cm water reflector. As shown in Fig 6.18 and Table 6.9.6-21, the maximum $k_{eff} + 2\sigma$ values occur at zero stainless steel thickness. The $k_{eff} + 2\sigma$ value for 7 kg of HEU at 100% enrichment and zero stainless steel thickness is 0.793. The $k_{eff} + 2\sigma$ values decrease to minimum values for steel thicknesses of 0.99 cm at 100 % enrichment and thicknesses of 1.47 cm at 19 % enrichment while the NLF decreases from ~ 0.07 to 0.06. The NLF continues to decrease to a value of 0.03 as stainless steel is added up to the 66,133 g amount of the containment vessel and the liner and drum assembly. At maximum stainless steel thickness, the $k_{eff} + 2\sigma$ values increase to within 0.02 of the values for zero stainless steel thickness. The stainless steel of the ES-3100 packaging is not as effective a reflector as the 20 cm water surrounding the core. The overall effect of adding the stainless steel of the ES-3100 packaging to the configuration is the reduction in the neutron multiplication factor for the system. The $k_{eff} + 2\sigma$ values are substantially below the USL of 0.925.

Cases **atdmkr_7_8_11** through **atdmkr_7_1_1** (Table 6.9.6-21, Appendix 6.9.6) pertain to Model 3 (Fig. 6.13, Section 6.3.1.4). The fissile core of Model 1 is blanketed with a variable thickness Kaolite shell and an exterior 20 cm water reflector. As shown in Fig. 6.19 and Table 6.9.6-21, the maximum $k_{eff} + 2\sigma$ values occur at a Kaolite shell thickness of zero. For 7 kg of HEU at 100 % enrichment, the $k_{eff} + 2\sigma$ value is 0.793. $K_{eff} + 2\sigma$ values decrease to asymptotic values at shell radii of ~14.74 cm (Kaolite thicknesses of ~ 8.66 cm) while the NLF decreases from ~ 0.07 to 0.03. The NLF continues to decrease to a value of 0.006 as water-saturated Kaolite is added up to the 128,034 g pre-bake amount inside the body weldment outer liner and the drum top plug. The asymptotic values for $k_{eff} + 2\sigma$ are ~ 0.043 below maximum $k_{eff} + 2\sigma$ values which occur at zero Kaolite thickness.

Cases **atdmdkr_7_8_11** through **atdmdkr_7_8_1** (Table 6.9.6-21, Appendix 6.9.6) pertain to Model 3 (Fig. 6.13, Section 6.3.1.4), where dry Kaolite is being evaluated consistent with test results of 10 CFR 71.55(d). Only 7 kg of HEU at 100 % enrichment is evaluated in these cases. As shown in Fig. 6.20 and Table 6.9.6-21, the $k_{eff} + 2\sigma$ value decreases to a minimum value at a shell radius of 31.08 cm (dry Kaolite thickness of 25.03 cm). The NLF decreases gradually from ~ 0.07 to 0.05 as NCT Kaolite is added up to the amount contained inside the body weldment outer liner and the drum top plug. Neutron leakage from a configuration with the NCT Kaolite of the ES-3100 packaging is much greater than leakage from one with water-saturated Kaolite. Over the range of moisture content, the Kaolite of the ES-3100 packaging is not as effective a reflector as the 20 cm water surrounding the core. The overall effect of adding Kaolite to the system is the reduction in neutron multiplication in the system. The $k_{eff} + 2\sigma$ values are substantially below the USL of 0.925.

A uranium alloy is conservatively assessed when scattering media (aluminum or molybdenum) is treated as multiplying media (^{235}U) in the calculation models. Thus, limits established for HEU metal apply to the alloy.

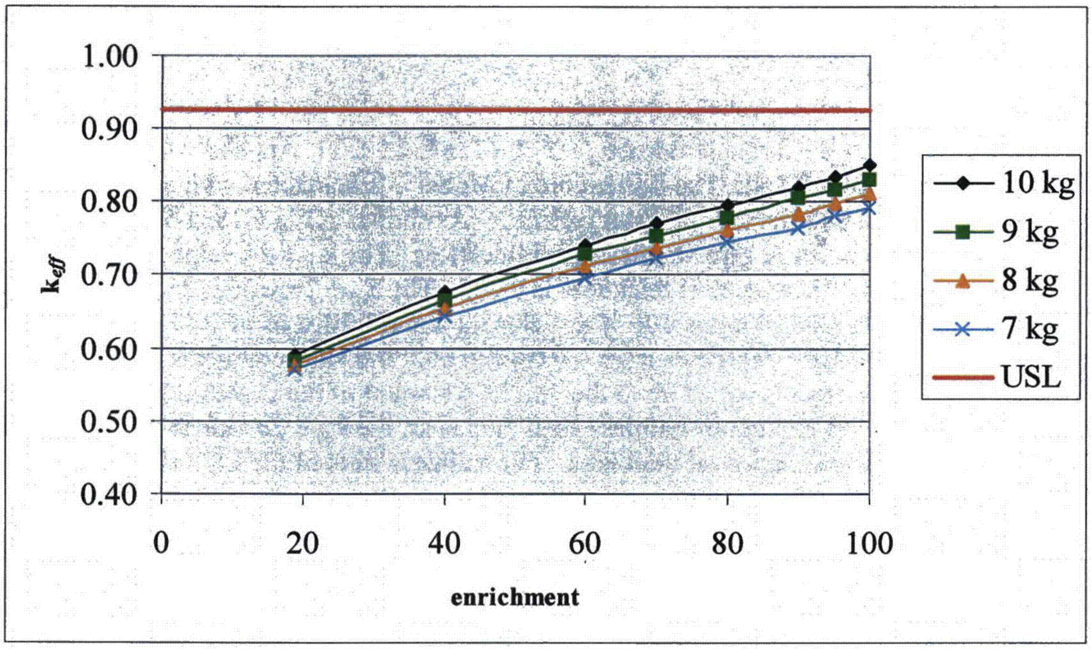


Fig. 6.17. K_{eff} vs. enrichment (wt % ^{235}U) for HEU ranging from 7 to 10 kg.

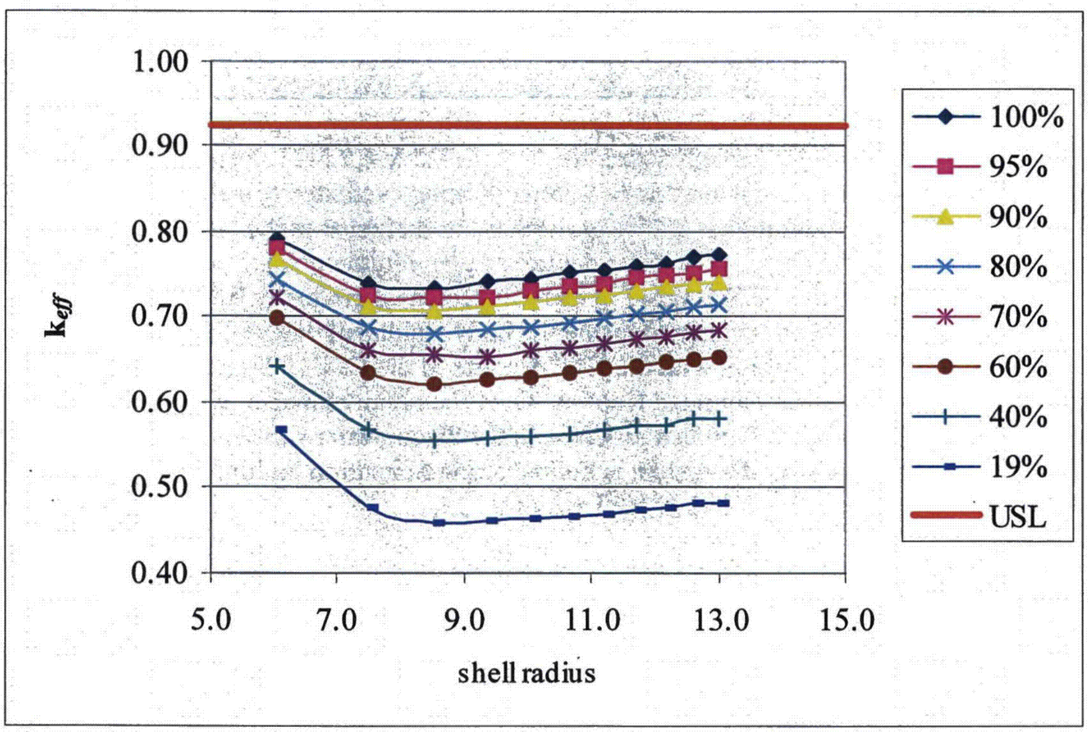


Fig. 6.18. K_{eff} vs. stainless steel shell radius (cm) for 7 kg HEU with enrichments ranging from 19 to 100 wt % ^{235}U.

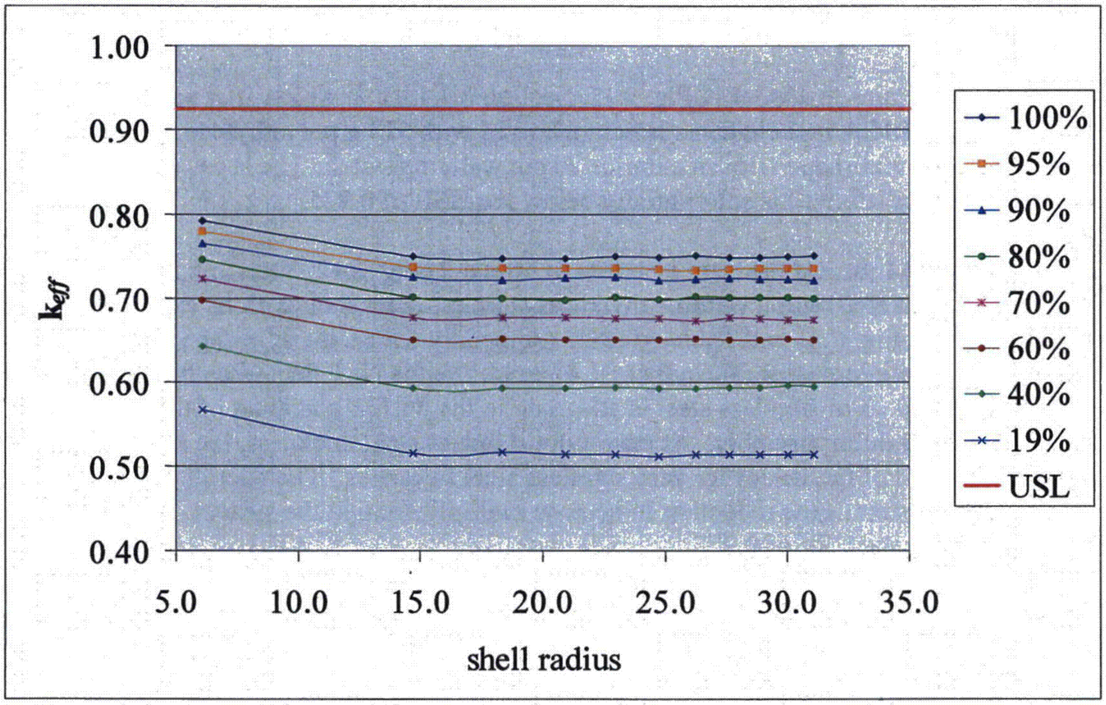


Fig. 6.19. K_{eff} vs. Kaolite shell radius (cm) for 7 kg HEU with enrichments ranging from 19 to 100 wt % ^{235}U.

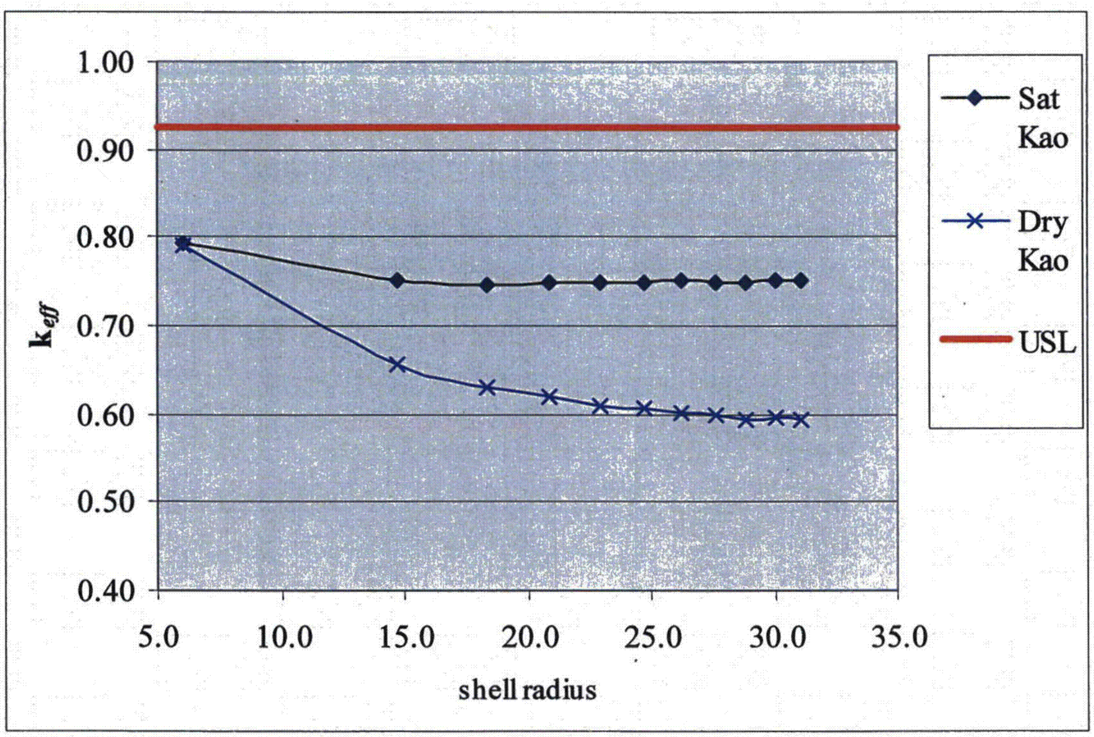


Fig. 6.20. K_{eff} vs. shell radius (cm) for dry and water-saturated Kaolite for 7 kg HEU at 100% enrichment.

6.7.2 Results for TRIGA Fuel Elements, Three Pieces per Convenience Can

Case **atdzr** pertains to Model 1 (Fig. 6.11, Section 6.3.1.4), in which 10.4 kg of $UZrH_x$ (921 g ^{235}U) of three TRIGA fuel elements is homogenized with 513 g polyethylene. The fissile core is configured into a spherical shape with an exterior 20 cm water reflector. The $k_{eff} + 2\sigma$ value of 0.679 (Table 6.9.6-22, Appendix 6.9.6) is substantially below the USL of 0.925.

Cases **atdzsr_11** through **atdzsr_1** pertain to Model 2 (Fig. 6.12, Section 6.3.1.4). The fissile core of Model 1 is blanketed with a variable thickness stainless steel shell with an exterior 20 cm water reflector. As shown in Fig. 6.21 and Table 6.9.6-22 (Appendix 6.9.6), the $k_{eff} + 2\sigma$ value decreases to minimum value for a stainless steel thickness of 1.87 cm while the NLF decreases from ~ 0.06 to 0.05. The NLF decreases to 0.03 as stainless steel is added up to the 66,133 g amount of the containment vessel and the liner and drum assembly. At maximum stainless steel thickness, the $k_{eff} + 2\sigma$ values increase to within 0.058 of the values for zero stainless steel thickness. The overall effect of adding stainless steel to the system is the reduction in neutron multiplication in the system. The $k_{eff} + 2\sigma$ values are substantially below the USL of 0.925.

Cases **atdzkr_11** through **atdzkr_1** pertain to Model 3 (Fig. 6.13, Section 6.3.1.4). The fissile core of Model 1 is blanketed with a variable thickness Kaolite shell with an exterior 20 cm water reflector. As shown in Fig. 6.21 and Table 6.9.6-22 (Appendix 6.9.6), the $k_{eff} + 2\sigma$ values decrease to an asymptotic value at a shell radius of 18.54 cm (Kaolite thickness of 11.08 cm.) The NLF decreases from ~ 0.06 to 0.02. The NLF continues to decrease to a value of 0.006 as water-saturated Kaolite is added up to the 128,034 g pre-bake amount contained inside the body weldment outer liner and the drum top plug. The asymptotic value for $k_{eff} + 2\sigma$ is ~ 0.048 below the maximum $k_{eff} + 2\sigma$ value at zero Kaolite thickness. The overall effect of adding water-saturated Kaolite to the system is the reduction in neutron multiplication in the system. The $k_{eff} + 2\sigma$ values are substantially below the USL of 0.925.

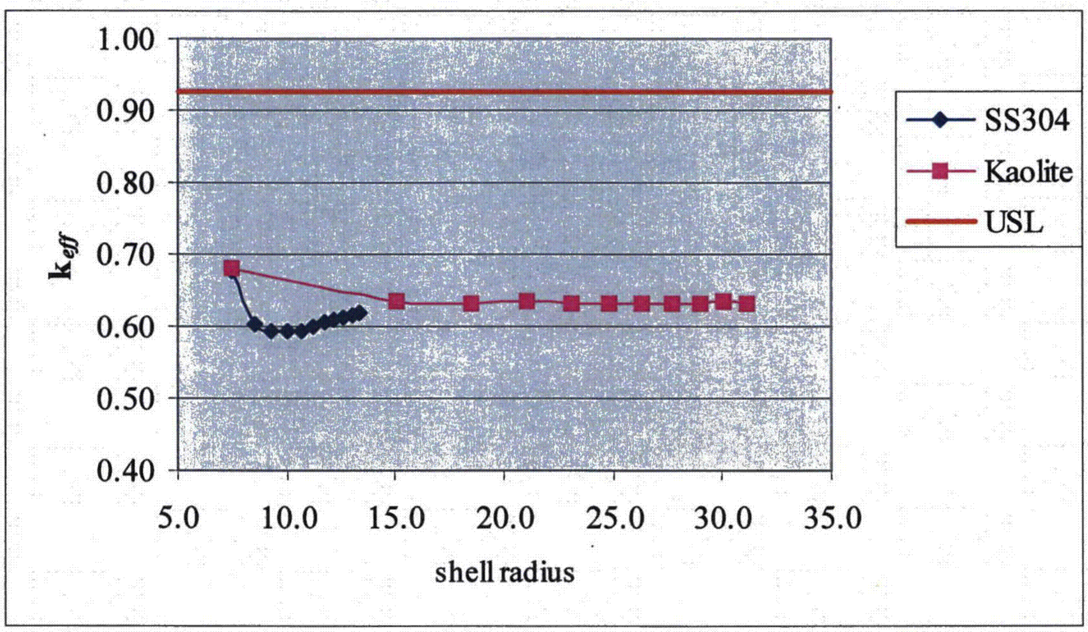


Fig. 6.21. K_{eff} vs shell radius (cm) for 10.4 kg core of $UZrH_x$ with stainless steel or Kaolite shell.

Cases **athzpkr_11** through **athzpkr_1** pertain to Model 4 (Fig. 6.14, Section 6.3.1.4). The homogenized core of Model 4 consists of $UZrH_x$, 500 g of polyethylene, and Kaolite. For this set of parametric cases, the Kaolite water content ranges from the water-saturated (maximum radius) to the dry condition (minimum radius) [Table 6.9.6-22, Appendix 6.9.6]. As shown in Fig. 6.22 and Table 6.9.6-22, the $k_{eff} + 2\sigma$ value remains constant at ~0.468 until the moisture content drops to 60% at a core radius of 37.17 cm. The $k_{eff} + 2\sigma$ value decreases to a minimum value of 0.306 at the dry Kaolite condition. The homogenization of the Kaolite shell and the fissile material core from Model 3 produces two competing effects in Model 4:

- addition of Kaolite expands the fissile material core, reducing neutron multiplication (evident in the comparison of results for Cases **athzphr_11** and **atdzkr_11**), and
- the addition of the water in the Kaolite increases neutron multiplication.

Consequently, the scenario of water penetration of fissile material core requires evaluation.

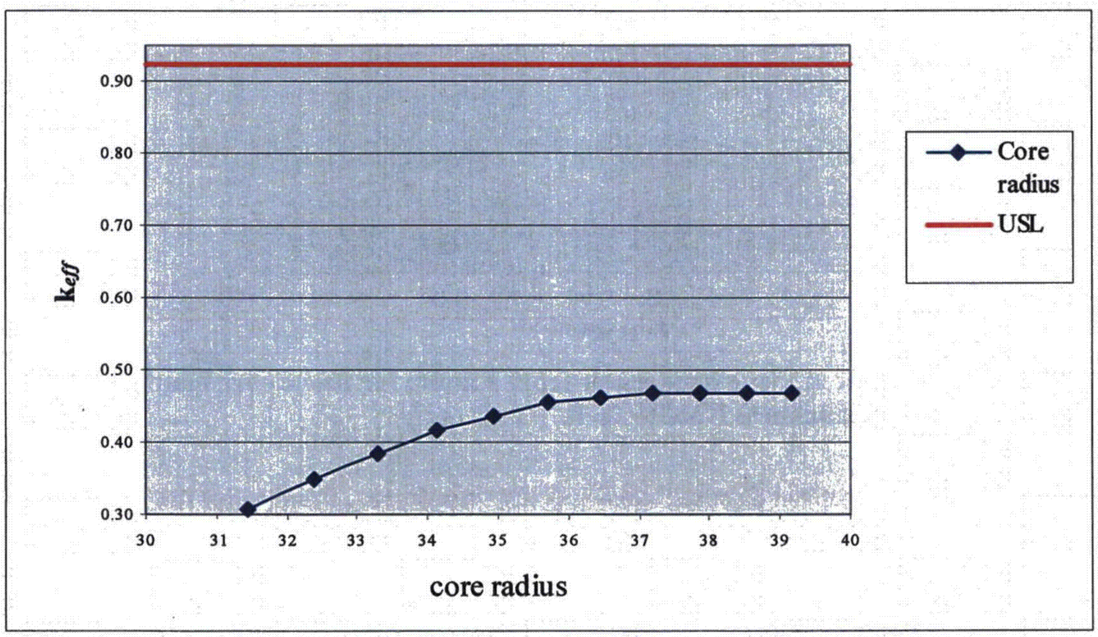


Fig. 6.22. K_{eff} vs. core radius (cm) for homogenized core of $UZrH_x$, 500 g polyethylene, and Kaolite where the Kaolite water content ranges from the dry to the water-saturated condition.

Cases **athzpwskr_9_11** through **athzpwskr_9_1** pertain to Model 5 (Fig. 6.15, Section 6.3.1.4). Model 5 is a variation of Model 3, which represents a fissile material core blanketed with water-saturated Kaolite and reflected by 20 cm of water. In Model 5, excess water from the Kaolite shell is assumed to penetrate the core. As shown in Fig. 6.23 and Table 6.9.6-22a (Appendix 6.9.6), the $k_{eff} + 2\sigma$ value increases to a maximum of 0.962 with the addition of 14,922 g of excess water into the fissile material core. As more water is added to the core, the core becomes overmoderated and the $k_{eff} + 2\sigma$ value decreases below the subcritical limit.

In the absence of Part 71.55(f) Test data or dynamic impact simulations per Type-C test criteria, the assumption of water penetration of the fissile material core is valid under the severe air transport evaluation criteria. Models 5 is applicable to air transport of the ES-3100 package given that the geometric form of the TRIGA fuel content following the air transport accident can not be established.

Additional case results are presented in Table 6.9.6-22a (Cases **athzpwskr_8_11** through **athzpwskr_6_1**) where the number of fuel segments is reduced from 9 items in a package to 6 items where the content remains subcritical over the entire range of moderation. As shown in Fig. 6.23 and Table 6.9.6-22a (Appendix 6.9.6), the $k_{eff} + 2\sigma$ value increases to a maximum of 0.911with the addition of 14,922 g of excess water into the fissile material core for a package loading of 716 g ^{235}U. As more water is added to the core, the core becomes overmoderated and the $k_{eff} + 2\sigma$ value decreases to 0.601.

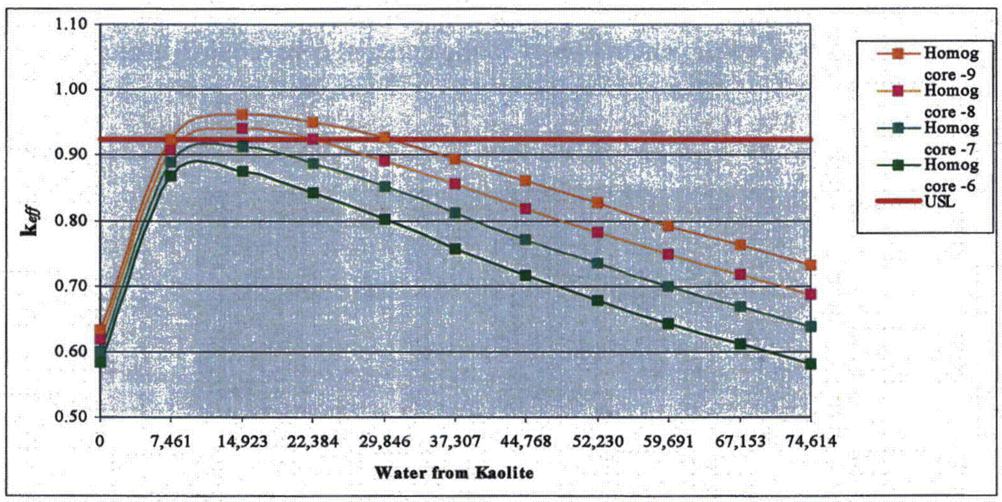


Fig. 6.23. k_{eff} vs. excess water from Kaolite for fissile core blanketed with a variable thickness Kaolite shell.

Case results are presented in Table 6.9.6-22c (Cases **athopwskr20_1** through **athopwskr20_11**, Cases **athopwskr70_1** through **athopwskr70_11**, and Cases **athopwskr93_1** through **athopwskr93_11**) for U_3O_8-Al at the TRIGA content limits of 716 g ^{235}U for 20% enrichment and 408 g ^{235}U for 70% and 93.2 % enrichments. As shown in Fig. 6.23b and Table 6.9.6-22c (Appendix 6.9.6), the $k_{eff} + 2\sigma$ values for U_3O_8-Al are exceeded by values for TRIGA content; therefore, the TRIGA loading limits are bounding for the research reactor related contents of U_3O_8 and U_3O_8-Al oxide content.

Case results are presented in Table 6.9.6-22d (Cases **athupwskr09_1** through **athupwskr09_11**, Cases **athupwskr20_1** through **athupwskr20_11**, and Cases **athupwskr93_1** through **athupwskr93_11**) for UO_2-Mg in Al clad at content limits of 716 g ^{235}U for 9.89% and 20% and 408 g ^{235}U for 93.2 % enrichment. Cases **athupwskr93_1** through **athupwskr93_11** for UO_2 are at content limits of 408 g ^{235}U for 93.2 % enrichment. As shown in Fig. 6.23c and Table 6.9.6-22d (Appendix 6.9.6), the $k_{eff} + 2\sigma$ values for research reactor contents of UO_2-Mg or UO_2 exceed values for TRIGA content. These values are well below the USL; therefore, the TRIGA loading limits may be applied to the research reactor related content of UO_2 and UO_2-Mg oxide content.

Model 6 (Fig. 6.16, Section 6.3.1.4) pertains to the case where fissile material from the homogenized core of Model 4 forms a shell external to the core. Given that the total amount of fissile material is limited to 716 g ^{235}U, the quantity is not sufficient to form a critical configuration. As shown in Section 6.7.3, this is apparent even if kilogram quantities of additional fissile material were located in the shell adjacent to the core.

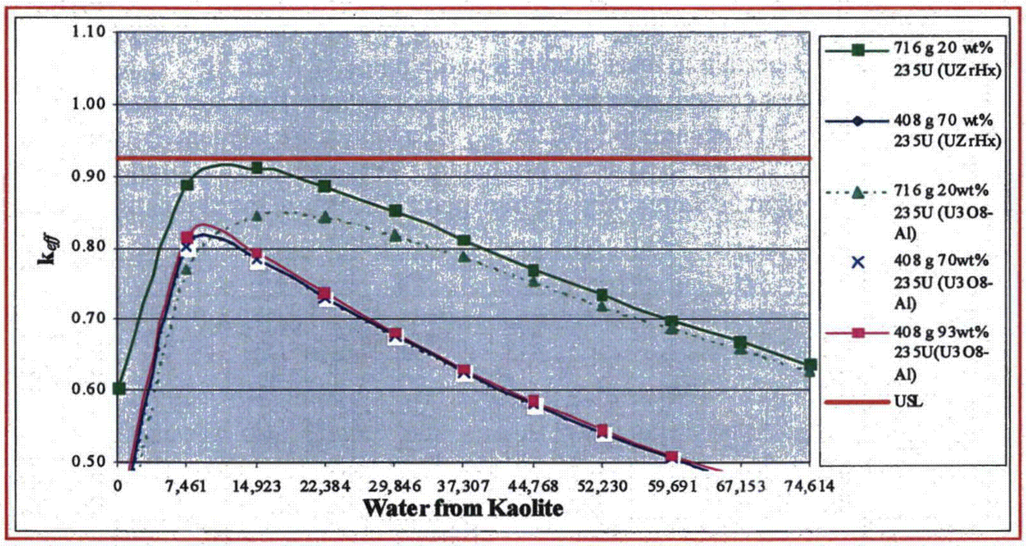


Fig. 6.23b. k_{eff} vs. excess water from Kaolite for $UZrH_x$ and U_3O_8-Al contents.

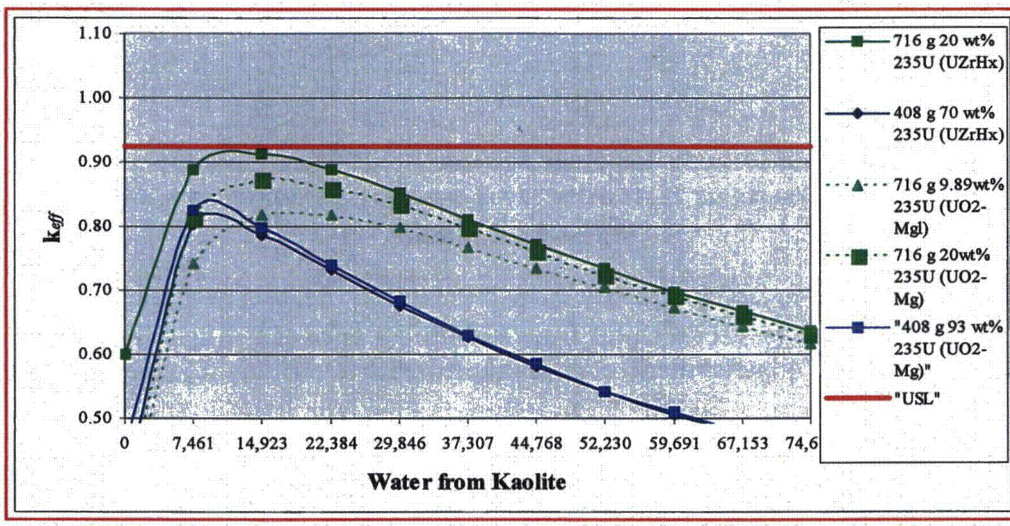


Fig. 6.23c. k_{eff} vs. Excess water from Kaolite for $UZrH_x$, UO_2-Mg, and UO_2 contents.

6.7.3 Results for HEU Broken Metal, More Than One Piece per Convenience Can

The results for Models 1–3 are presented in Sect. 6.7.1. Cases **athmpkr_12_8_11** through **athmpkr_8_8_1** (Table 6.9.6-23, Appendix 6.9.6) pertain to Model 4 (Fig. 6.14, Section 6.3.1.4) for broken metal loadings in the range of 4–7 kg ^{235}U for uranium at 100% enrichment and Kaolite water content from dry to water saturation values. Cases **athmpkr_12_1_11** through **athmpkr_10_1_1** pertain to loadings in the range of 5–7 kg ^{235}U for uranium at 20% enrichment and Kaolite water content from dry to water saturation values. The $k_{eff} + 2\sigma$ values are adequately below the USL of 0.925 for mass loadings up to 4,000 g ^{235}U at 100% enrichment and up to 6,000 g ^{235}U at 20% enrichment.

Cases **athmpwskr_12_8_11** through **athmpwskr_2_8_1** (Table 6.9.6-23, Appendix 6.9.6) pertain to Model 5 (Fig. 6.15, Section 6.3.1.4) for mass loadings in the range of 1–7 kg ^{235}U for uranium at 100% enrichment and for core water content over the range of excess water from the Kaolite shell. As shown in Fig. 6.24 and Table 6.9.6-23 (Appendix 6.9.6), the $k_{eff} + 2\sigma$ values are significantly above the USL.

Cases **athmpwskr_12_1_11** through **athmpwskr_2_1_1** and Cases **athm2pwskr_5_1_11** through **athm2pwskr_2_1_1** pertain to mass loadings in the range of 7–0.6 kg ^{235}U for uranium at 20% enrichment and core water content over the range of excess water from the Kaolite shell. As shown in Fig. 6.25 and Table 6.9.6-23 (Appendix 6.9.6), the $k_{eff} + 2\sigma$ values are adequately below the USL of 0.925 for mass loadings up to 700 g ^{235}U at 20% enrichment. This signifies the fissile mass loading limit for HEU broken metal content to be shipped by air transport. Therefore, the evaluation of Model 6 is focused on this limit.

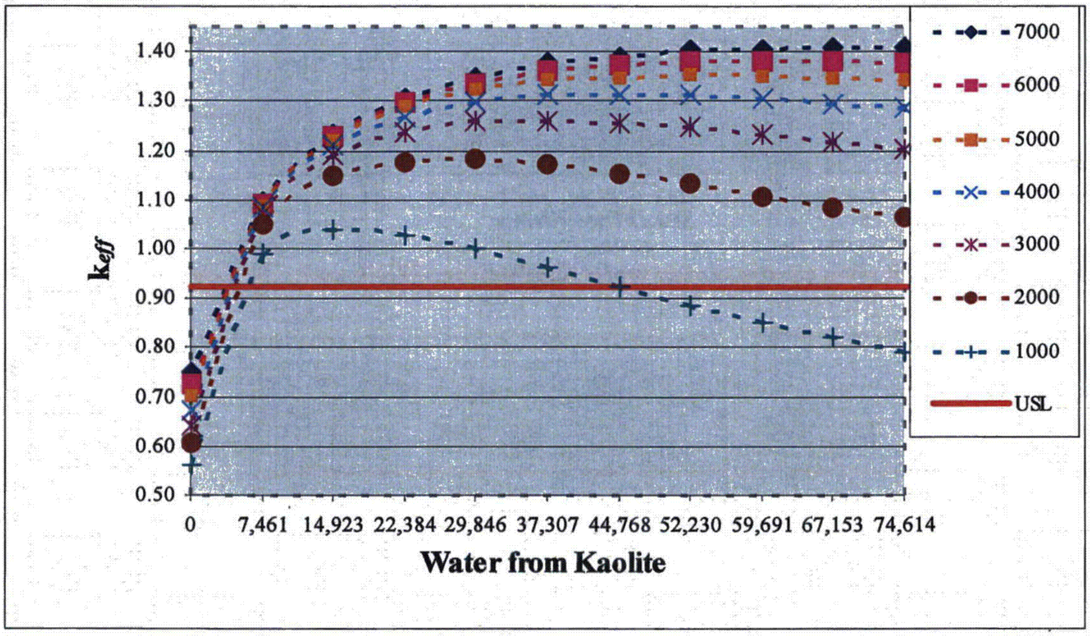


Fig. 6.24. K_{eff} vs. excess water from Kaolite for core of 1–7 kg enriched HEU broken metal at 100% enrichment, core blanketed with a variable thickness Kaolite shell.

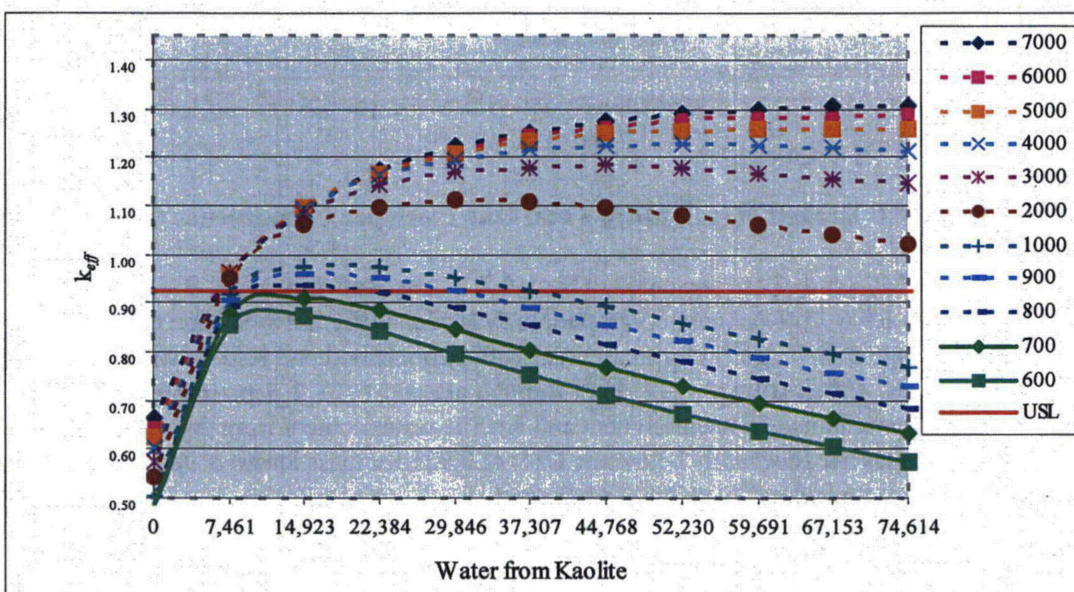


Fig. 6.25. K_{eff} vs. excess water from Kaolite for core 20% enriched HEU broken metal (0.6–7 kg ^{235}U), core blanketed with a variable thickness Kaolite shell.

Cases **athmpkmr_6_1_1_11** through **athmpwskr_1_6_1_1** (Table 6.9.6-23, Appendix 6.9.6) pertain to Model 6 (Fig. 6.16, Section 6.3.1.4), where fissile material from the homogenized core of Model 4 forms a shell external to the core. Cases **athmpkmr_6_1_1_11** through **athmpkmr_6_1_1_1** represent 3 kg ^{235}U in the core and 0.5 kg ^{235}U in the shell (17.5 kg total HEU at 20% enrichment) with the water content of the Kaolite ranging from the water-saturated to the dry condition. In the subsequent series of cases, the $k_{eff} + 2\sigma$ values decrease as HEU is moved from the core to the shell in 2.5 kg increments. Although the total HEU of 17.5 kg greatly exceeds the 3.5 kg limit identified in the evaluation of Model 5, all $k_{eff} + 2\sigma$ values are adequately below the USL of 0.925.

As stated previously, uranium alloy is conservatively assessed when the aluminum or molybdenum constituents are modeled as ^{235}U. Thus, limits established for HEU metal apply to the uranium alloys of aluminum or molybednum.

6.7.4 Conclusions

Given that the results for catastrophic damage are adequately subcritical, ES-3100 packages may be shipped via air transport with:

- Solid or broken HEU metal or U-Al alloy with up to 700 g ^{235}U,
- 3 fuel sections ("meats") of $UZrH_x$ per loaded convenience can and up to 3 loaded cans per package where the ^{235}U does not exceed 716 g at 20% enrichment or 408 g ^{235}U at 70%,
- ~15 inch long clad fuel rods, each rod derived from a single TRIGA fuel element, and where per package ^{235}U does not exceed 716 g at 20% enrichment or 408 g ^{235}U at 70%, or
- oxides of U_3O_8, U_3O_8-Al, UO_2, and UO_2-Mg where the per package ^{235}U content does not exceed 716 g at enrichments less than or equal to 20% or 408 g ^{235}U at enrichments greater than 20%.

6.8 BENCHMARK EXPERIMENTS

6.8.1 Applicability of Benchmark Experiments

The criticality validation is specific to uranium, plutonium, and uranium-233 systems encompassing a substantial subset of the database used to prepare the Organization for Economic Cooperation and Development (OECD) Handbook, Volumes I–VI. The benchmark specifications are intended for use by criticality safety engineers to validate the application of criticality calculation techniques such as SCALE 4.4a. Example calculations presented in the handbook do not constitute a validation of the codes or cross-section data sets by themselves, but the Handbook information can be and has been used to validate SCALE 4.4a by competent nuclear criticality safety persons.

The data from the benchmark experiments involving uranium represent a sufficiently wide range of enrichments and physical and chemical forms to cover many existing or presently planned activities for Y-12. These include enriched uranium with ^{235}U only and natural and depleted uranium, as well as highly enriched uranium, intermediate enriched uranium, and low enriched uranium. Data analyzed from critical experiments in this validation include systems having fast, intermediate, and thermal neutron energy spectra, and they include materials in various physical and chemical forms such as uranium metals, solutions, and oxide compounds. With the benchmark experiments that are directly applicable to uranium systems, there is a high level of confidence that the calculated results presented in this evaluation are sufficiently accurate to establish the safety of the package under both NCT and HAC. This conclusion is based on the validation of the code and cross-section library described in Sect. 6.3.3.

6.8.2 Details of Benchmark Calculations

The validation of CSAS25 control module of SCALE 4.4a with the 238-group ENDF/B-V cross-section library is documented in Y/DD-896/R1 and Y/DD-972/R1. Y/DD-896/R1 addresses the establishment of bias, bias trends, and uncertainty associated with the use of SCALE 4.4a for performance of criticality calculations. This evaluation is directed at uranium systems consisting of fissile and fissionable material in metallic, solution, and other physical forms, as well as plutonium and ^{233}U systems, as described in the OECD Handbook. [NEA/NSC/DOC(95)03] The focus is on comparison of k_{eff} with the associated experimental results for establishment of bias, bias trends, and uncertainty as a final step. Compiled data for 1217 critical experiments are used as the basis for the calculation models. The calculated results from SCALE 4.4a using the 238-group ENDF/B-V cross-section library have been compared with reported results for the benchmark experiments. Comparison of results demonstrates that SCALE 4.4a run on the SAE HP J-5600 unclassified workstation (CMODB) produces the same results within the statistical uncertainty of the Monte Carlo calculations as reported by the OECD for the experiments.

Y/DD-972/R1 addresses determining USL and for incorporating uncertainty and margin into this USL. Y/DD-972/R1 establishes subcritical limits determined through an evaluation of statistical parameters of calculation results for critical experiments. The correlating parameters (i.e., mass, enrichment, geometry, absorption, moderation, reflection) and values for applying additional margin to the subcritical limits are application dependent. The determination of correlating parameters and additional margin is an integral part of the process analysis for a particular application. For the critical experiment results, no correlation between calculation results and neutron energy causing fission was found. As such, this document does not specify "final" USL values as has been done in the past.

6.8.3 Bias Determination

The USL is based on the non-parametric statistics-based lower tolerance limit (LTL) for greater than 0.99/99% where there is a probability of greater than 0.99 that 99% of the population is greater than a specified result, reduced by additional margin. From Table 1 of Y/DD-972R1, the LTL combining bias and bias uncertainty is 0.975 for uranium systems, including HEU metal, indicating a bias value of 0.025. Ordinarily the USL would be 0.955 where an additional margin of subcriticality of 0.02 is subtracted from the LTL of 0.975. However, guidance provided by NUREG/CR-5661 requires that the bias value of 0.025 be subtracted from 0.95 for determination of the USL, giving a value of 0.925.