"designated Original"

71-9315

Department of Energy

Washington, DC 20585

AUG 2 8 2007

Attn: Document Control Desk
Director, Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

This is a request for an emergency Revision to Modify the TRIGA Fuel Definition, on Certificate of Compliance No. 9315, Docket No. 71-9315. In a letter dated June 28, 2007, the NRC issued Revision 4 of USA/9315/B(U)F-96 for the ES-3100 shipping container. This revision allowed the transport of TRIGA fuel pellets by land and by air. In August, 2007, the ES-3100 was put into service to ship TRIGA fuel from South Korea for the Foreign Research Reactor (FRR) Program. In Korea, while attempting to load the ES-3100 with TRIGA fuel pellets, as authorized in the CoC, operators from the Y-12 National Security Complex encountered problems releasing fuel pellets intact from the cladding tubes. It appeared that the fuel pellets would fracture while being removed from the cladding, a situation that was not desirable at the Korean reactor facility where this operation was taking place.

The loading operation is now on hold pending a modified definition of TRIGA fuel contents in the CoC. The TRIGA fuel element can be cut to remove the portion of the element with fuel, and that section can be loaded into the ES-3100. That section of the element will be clad. The Department of Energy is hereby requesting an emergency revision of the ES-3100 CoC to accommodate TRIGA fuel with cladding. Maximum fissile loading is not changing, and fuel material form is also not changing. This request is specifically to authorize shipment of TRIGA fuel with cladding, in addition to currently authorized bare TRIGA fuel pellets.

Criticality safety of the proposed clad fuel modification has been determined to be bound by the bare fuel models currently in Section 6 of the SAR. Discussions of the safety of the clad fuel configuration have been added to SAR Section 6, as noted on the attached change pages.

Attached you will find change pages to the ES-3100 SAR (Document No. Y/LF-717, R1) and a draft mark-up of the CoC. The proposed modification of the TRIGA content is on SAR pages 1-12 and 1-13 (attached). A guide for insertion of these SAR page changes is included.

This mission is essential to the FRR program and can only be successful if the TRIGA fuel is shipped by mid-September. The shipping packages are on-site in Korea. Therefore, DOE is requesting an expedited review of this content modification and issuance of a CoC revision by September 7, 2007.

NMSS01

Ten copies of this letter with the attachment are being delivered to Kimberly J. Hardin, Project Manager, Licensing Branch, Division of Spent Fuel Storage and Transportation, Office of Nuclear Material Safety and Safeguards.

If you have any questions, please contact me at 301-903-5513.

Sincerely,

James M. Shuler

James Shuler
Manager, Packaging Certification Program
Safety Management and Operations
Office of Environmental Management

Enclosure

cc:
Kimberly J. Hardin, NRC
Joe Bozik, NNSA NA-261
Dana Willaford, DOE ORO
Jeff Arbital, BWXT Y-12
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ATTACHMENT 1

SAR PAGE CHANGES

GUIDE TO PAGE CHANGES

Y/LF-717, Rev. 1, page change #5

SAR SECTION	PAGE CHANGES
Volume 1, Front section	Replace pages i and xix
Volume 1, Section 1	Replace page 1-13
Volume 2, Front section	Replace pages i and xix
Volume 2, Section 6	Replace pages 6-29, 6-30a, 6-66c, 6-73, 6-87, 6-119, 6-119b

SAFETY ANALYSIS REPORT, Y-12 NATIONAL SECURITY COMPLEX, MODEL ES-3100 PACKAGE WITH BULK HEU CONTENTS

Prepared by the
Oak Ridge Y-12 National Security Complex
Oak Ridge, Tennessee 37831
Managed by
BWXT Y-12, L.L.C.
for the
U. S. Department of Energy
under contract DE-AC05-84OR21400

August 28, 2007

REVISION LOG

Date	SAR Revision No.	Description	Affected Pages
38407	0	Original issue	All
08/15/05	0, Page Change 1	Page changes resulting from *Responses to Request for Additional Information* #1, Y/LF-747.	title page, iv, xxiii, 1-4, 1-145, 2-2, 2-3, 2-6, 2-31, 2-32, 2-33, 2-34, 2-57, 2-59, 2-61, 2-107, 2-125, 2-131, 2-171, 2-173, 2-181, 2-183, 2-185, 2-186, 2-189, 2-367, 2-458, 2-675, 8-8, 8-9, 8-31
38753	0, Page Change 2	Page changes resulting from *Responses to Request for Additional Information #2*, Y/LF-761.	All Sections
38795	0, Page Change 3	Page changes resulting from *Responses to Request for Additional Information #3*, Y/LF-764.	1.38, 1.48, Appendix 1.4.1, 2-120, Table 6.4
38844	0, Page Change 4	Added polyethylene bottles and nickel alloy cans as convenience containers for authorized HEU contents. (CoC Revision 1)	Various pages in chapters 1, 2, 3 and 4.
08/21/06	0, Page Change 5	Revised equipment specifications for Kaolite and 277-4 neutron absorber. (CoC Revision 3)	Appendices 1.4.4 and 1.4.5.
11/15/06	1	Updated definition of pyrophoric uranium. Evaluated air transport. Revised criticality safety calculations to remove bias correct factors. Added a CSI option of 3.2. Increased mass of off-gassing material allowed in containment vessel. Increased carbon concentration in HEU contents. Increased Np-237 concentration is HEU contents. Added uranium zirconium hydride and uranium carbide as contents (TRIGA fuel). Revised equipment specifications for 277-4 neutron absorber. (CoC Revision 3)	All Sections

Date	SAR Revision No.	Description	Affected Pages
3/29/07	1, Page Change 1	Updated definition of TRIGA fuel for air transport and added TRIGA-related criticality safety cases.	title pages, viii, xi, xx, 1-12, 1-13, 1-20, 6-30, 6-54, 6-64, 6-66, 6-87, 6-119, 6-240 to 6-286, 6-385 to end
5/31/07	1, Page Change 2	Revised SAR in response to RAIs dated May 9, 2007 in reference to CoC Revision 4	title pages, xiii, xx, Section 1 and Section 6
6/30/07	1, Page Change 3	Revised SAR in response to RAIs dated May 9, 2007 in reference to CoC Revision 5	title pages, table of contents, Section 1, and Section 7
7/31/07	1, Page Change 4	Removed oxidation as an option for treating pyrphoric uranium metal	title pages, xx, 1-12, 1-201, 1-203, 1-212, 2-26,7-4
8/28/07	1, Page Change 5	Modified TRIGA fuel definition to include fuel pellets with cladding	title pages, xx, 1-13, 6-29, 6-30, 6-30a, 6-66c, 6-66d, 6-73, 6-74, 6-87, 6-119a, and 6-119b

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TRIGA fuel may be shipped a crimped fuel elements or as UZrHx fuel pellets (if disassembled), which shall be packed into convenience cans prior to shipment. Convenience cans of 4.25-inch diameter by various lengths shall be used. Fuel pellets loaded into convenience cans shall be up to 5 inches in length (full-length) and no more than three full-length pellets shall be loaded into a convenience can. Crimped fuel elements are clad fuel pellets and can be up to 15 inches in length (full-length). Cladding material is stainless steel or aluminum. Only the fuel section of the TRIGA fuel element is allowed to be shipped (Fig. 1.5), however there may be a residual of cladding up to ½ inch in length at either end of the crimped fuel element. Up to three 15-inch long crimped fuel elements shall be loaded into a single convenience can for shipping. Maximum loading of bare fuel pellets and crimped fuel elements shall be 3 fuel element equivalence per ES-3100 containment vessel. Only 70% enriched TRIGA fuel will be shipped. For SFEs and FTCs, the maximum allowable loading is 408 g ^{235}U per package, and for FFCRs, the maximum allowable loading is 336 g ^{235}U per package. No spacer cans are required.

Air Transport

Contents for air transport of the ES-3100 shall include HEU in the form of unirradiated TRIGA fuel pellets or crimped fuel elements. The characteristics of the air transport contents shall be similar to the ground transport contents, but the fissile loading per package will be as follows:

> TRIGA fuel elements and pellets- 3 fuel element equivalence per package. Fuel shall be 70% enriched and in the form of SFEs, FTCs, and FFCRs. Maximum fissile loading for SFEs and FTCs shall be 408 g ^{235}U per package, and for FFCRs, the maximum allowable load shall be 336 g ^{235}U per package.

1.2.3.1 Radioactive/fissile constituents

Fissile material mass loading limits for the contents of the ES-3100, as determined by criticality analyses, are presented in Table 1.3 (for ground transport only). For the ES-3100 package with bulk HEU content, the maximum number of A_2s is 294.00 (at 70 years) and the maximum activity is 0.3243 Tbq (at 10 years) [Table 4.4].

1.2.3.2 Chemical and physical form

The fissile material contents are in solid (HEU metal, alloy, or TRIGA fuel), crystalline (UNX) or powder (HEU oxide) form. Some moisture (up to 6 wt.%) may be present in the HEU oxide material, thereby making the oxide content clump together.

1.2.3.3 Reflectors, absorbers, and moderators

The reflectors, absorbers, and moderators present in the ES-3100 package are those associated with the materials of construction. For example, the thermal insulation acts as a neutron reflector to the contents of a single package and as a neutron moderator in an array of packages. The degree of neutron moderation is a function of the hydrogen content in the Kaolite 1600 and 277-4 materials. The stainless-steel materials of the containment vessel and the drum also act as neutron reflectors to the contents of a single package but act as neutron absorbers in an array of packages. The nuclear properties of the materials of construction and of the contents are important and have been taken into account in the criticality safety evaluation (Sect. 6). In addition to the materials of construction in the ES-3100 shipping package mentioned above, the 277-4 material has been specifically added to the ES-3100 package for the purpose of enhancing the neutron absorption characteristics for safety purposes (see Sect. 6 for additional discussion of the neutron-absorbing characteristics of this material).

Table 1.3. Authorized content [a] and fissile mass loading limits [b, c] for ground transport

Content description		Enrichment	CSI	No spacers, ^{235}U (kg)	277-4 can spacers,[d] ^{235}U (kg)
Solid HEU metal or alloy (specified geometric shapes) [e]	Cylinder A	≤ 100%	0	15	25.000
	Cylinder B	≤ 100%	0	18	30.000
	Square bars	≤ 100%	0	18	30.000
	Slugs	> 80%	0	18.286	25.601
	Slugs	≤ 80%	0	18.286	29.333
	Slugs	≤ 95%	0.4	Can spacers req'd[d]	34.766
Broken HEU metal or alloy		> 95%, ≤ 100%	0	Can spacers req'd	2.774
			0.4	Can spacers req'd	5.549
			0.8	Can spacers req'd	9.248
			2	Can spacers req'd	13.872
			3.2	Can spacers req'd	24.969
		> 90%, ≤ 95%	0.0	Can spacers req'd	3.516
			0.4	Can spacers req'd	6.154
			0.8	Can spacers req'd	10.549
			2	Can spacers req'd	18.461
			3.2	Can spacers req'd	26.373
		> 80%, ≤ 90%	0.0	Can spacers req'd	3.333
			0.4	Can spacers req'd	7.500
			0.8	Can spacers req'd	12.500
			2	Can spacers req'd	20.000
			3.2	Can spacers req'd	28.334
		> 70%, ≤ 80%	0.0	2.967	4.450
			0.4	5.192	8.900
			0.8	8.900	16.317
			2	17.059	25.218
			3.2	27.692	28.184
		> 60%, ≤ 70%	0	3.249	5.198
			0.4	5.848	12.996
			0.8	13.646	20.793
			2.0	21.444	24.692
			3.2	24.692	24.692
		≤ 60%	0.0	5.576 kgU	11.154 kgU
			0.4	14.872 kgU	28.813 kgU
			0.8	28.814 kgU	35.320 kgU
			2	35.320 kgU	35.320 kgU
			3.2	35.320 kgU	35.320 kgU
HEU oxide		> 20%, ≤ 100%	0.0	21.124 [f]	Spacer not req'd
UNX crystals [a, g]		> 20%, ≤ 100%	0	3.768	Spacer not req'd
UNX crystals [a, g]		> 20%, ≤ 100%	0.4	11.303 [f]	Spacer not req'd
TRIGA fuel		70 %	0	0.408	Spacer not req'd

[a] HEU in solution form is not permitted for shipment in the ES-3100.
[b] All limits are expressed in kg ^{235}U unless otherwise indicated.
[c] Mass loadings cannot be rounded up.
[d] 277-4 can spacers as described on Drawing No. M2E801580A026 (Appendix 1.4.8).
[e] Geometries of solid shapes are as follows:
- Cylinder A is larger than 3.24 in. diameter but no larger than 4.25 in. diameter: maximum of 1 cylinder per can.
- Cylinder B is no larger than 3.24 in. diameter: maximum of 1 cylinder per can.

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

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]**	
				enr. = 20%	enr. = 70%
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				
With/without can spacers	**cylinders (d = 3.24 in.)**	**cylinders (3.24 < d ≤ 4.25 in.)**	**bars**	**slugs**
One per convenience can	700g ^{235}U	700g ^{235}U	700g ^{235}U	700g ^{235}U

Solid HEU metal of unspecified geometric shapes characterized as broken metal			
With/without can spacers	**20% < enr. ≤ 100%**	**19% < enr. ≤20%**	**Nat.U. < enr. ≤ 19%**
	not allowed	3,500g Uranium	not allowed

HEU oxide, UNX crystals, unirradiated TRIGA fuel elements					
With/without can spacers	**HEU product oxide**	**HEU skull oxide**	**UNX crystals**	**unirradiated TRIGA [a]**	
				enr. = 20%	enr. = 70%
Three per convenience can	not allowed	not allowed	not allowed	716g ^{235}U	408g ^{235}U

[a] 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.

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.

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 stainless steel crimped on each end of the clad fuel rod adds ~6 g aluminum.

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 can be shipped in this 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 nuetron 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 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 the neutron multiplication for the an array of packages, similar results are expected for the cases demonstrating compliance with 10 CFR 7155(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 forTRIGA 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 ≤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 ≤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 ≤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.

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.

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 with up to 700 g ^{235}U, or

- 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%.

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.

The General Atomics catalog of stock items lists approximately 40 TRIGA fuel elements classified into four basic types: standard element, instrumented element, fuel-follower control rod, or cluster assembly. The TRIGA element active fuel region consists of three 5-in long sections "fuel meats" of $UZrH_x$. The H/Z atom ratio "x" in $UZrH_x$ equals 1.6 in all cases except for two stock items. For these cases, x = equals 1.0 and the fissile content is < 40 g ^{235}U. The unirradiated solid form TRIGA fuel is identified as either 20 % enriched or 70 % enriched, and has dimensions and material properties specific to its design function. Table 1.4 provides a summary description.

The fuel diameter for the 20 % enriched TRIGA elements is either 1.44 in., 1.41 in., 1.40 in., 1.37 in., 1.34 in., or 1.31 in. The uranium composition of the fuel is 45 wt %, 30 wt %, 20 wt %, 12 wt %, and 8.5 wt %. As illustrated in Table 6.9.3.1.4-a, the TRIGA element with a maximum fissile content of 307 g ^{235}U in 1,560 g U, 45 wt% U in $UZrH_x$, and a H/Zr atom ratio of 1.6 has a computed fuel density of ~8.66 g/cm^3. The calculated number density (N_i) for each element or isotope is also given. The TRIGA fuel element with a fuel diameter of 1.44 inches contains 3,466.7 g $UZrH_x$. Further evaluation of the manufacturers data reveals that fuel density is proportional to the uranium weight fraction. Calculated density values are: 8.6597 g/cm^3 for 45 wt% U in $UZrH_x$, 6.8995 g/cm^3 for 30 wt% U in $UZrH_x$, 6.2825 g/cm^3 for 20 wt% U in $UZrH_x$, 5.9328 g/cm^3 for 12 wt% U in $UZrH_x$, and 5.7895 g/cm^3 for 8.5 wt% U in $UZrH_x$.

The active fuel diameter for 70 % enriched TRIGA fuel is 1.44 inches in both the standard element and instrumented element, and 1.31 inches in the fuel follower control rod. The uranium composition of the fuel is 8.5 wt %. 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. As the calculation in Table 6.9.3.1.4-b illustrates, the 70 wt % enriched TRIGA fuel has a computed density of ~5.70 g/cm^3. The TRIGA fuel element with a fuel diameter of 1.44 inches contains 2,282.4 g $UZrH_x$ while the element with a fuel diameter of 1.31 inches contains 1,888.9 g $UZrH_x$.

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. In preparation for shipment in the ES-3100, a TRIGA fuel element is disassembled, the fuel meats removed from the thin cladding and packed into convenience cans.

Table 6.9.3.1-4a. Calculation of constituent weight-percentage values for 20 wt % enriched uranium-zirconium hydride content in KENOV.a calculation models

Avogadro No. (N_o) =	6.0221370e+23						
U(ZrH$_x$)							
atom	**wt %**	**x**	**mass**	**at. wt.**		**calc. N_i**	**N_iA_i**
			g				
Hydrogen	0.9554	1.6		1.00780	1.6125	4.9439e+22	4.9824e+22
Zirconium	54.0447	1		91.21960	91.2196	3.0897e+22	2.8184e+24
u-235	19.6795		307.0	235.04410			
u-238	80.3205			238.05099			
uranium	45.0000		1560.0	237.45318	75.9535	9.8830e+21	2.3468e+24
	100.0001				168.7856		
summations	100.0001					9.0219e+22	5.2150e+24
At. wt molecule							
Volume (cm^3)	400.3200						
Mass (g)	3466.66667						
density (g/cm^3)	8.65974						
den.=($\sum$NiAi)/No							8.6597

Table 6.9.3.1-4b. Calculation of constituent weight-percentage values for 70 wt % enriched uranium-zirconium hydride content in KENOV.a calculation models

Avogadro No. (N_o) =	6.0221370e+23						
U(ZrH$_x$)							
atom	**wt %**	**x**	**mass**	**at. wt**		**calc. N_i**	**N_iA_i**
Hydrogen	1.5894	1.6		1.00780	1.6125	5.4148e+22	5.4571e+22
Zirconium	89.9107	1		91.21960	91.2196	3.3841e+22	3.0870e+24
u-235	70.1031		136.0	235.04410			
u-238	29.8969			238.05099			
uranium	8.5000		194.0	235.93508	8.6237	1.2370e+21	2.9184e+23
	100.0001				101.4558		
summations	100.0001					8.9227e+22	3.4334e+24
At. wt molecule							
Volume (cm^3)	400.3200						
Mass (g)	2282.35294						
density (g/cm^3)	5.70132						
den.=(∑NiAi)/No							5.7013

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 extending 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 stainless steel or aluminum clad, where the protruding clad at each end has been crimped in. The fuel rods will be 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 a minimum of two convenience cans; where only one convenience can is loaded with clad fuel rods. The loaded can is 17.5 inches tall while the empty one is 8.75 inches tall. Although can spacers are not required for criticality control, can spacers or stainless steel pads may be used to take up free volume over the 31 in. internal height of the containment vessel. The maximum quantity of fissile material per package is 408 g ^{235}U.

The clad fuel rod with 1.44 inch diameter fuel pellets contains 2,282.4 g UZrH$_x$ while rod with the 1.31 inch diameter fuel pellets contains 1,888.9 g UZrH$_x$. The 0.02 in thick sheath of stainless steel clad adds ~179 g to the mass of the active fuel for the 1.48 in. diameter standard element or instrumented element, and ~163 g to active fuel mass for the 1.35 in. diameter fuel follower control rod. Allowance for 1/2 in. of residual stainless steel crimped on each end of the clad fuel rod adds ~11 - 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 stainless steel crimped on each end of the clad fuel rod adds ~6 g aluminum.

6.9.3.2 TYPE 304 STAINLESS STEEL

The metallic components of the ES-3100 package are composed of type 304 stainless steel. These include the containment vessel, the convenience cans, the drum liner, and the drum. Type 304 stainless steel with a density of 7.9400 g/cm^3 is included as a material in the SCALE Standard Composition Library.

6.9.3.3 277-4 NEUTRON ABSORBER

Catalog No. 277 dry mix is a proprietary mixture of Thermo Electron Corporation for producing a heat-resistant shielding material which combines the most effective shielding components into a single homogeneous composite. The shielding composite material is designed to maximize the hydrogen content necessary for thermalizing fast neutrons for capture in the boron constituent. Widely used in nuclear power plant applications, the heat-resistant shielding material is capable of retaining a significant portion of its shielding properties up to 230°C (450°F). The recommended operating limit is 350°F, which is well above HAC temperatures expected inside the body weldment liner inner cavity and canned spacers.

The 277-4 neutron absorber material used in the ES-3100 is a formulation of Cat 277-0 dry mix, a boron carbide additive, and water. 277-4 is produced through a quality-controlled batch process of dry blending, wet mixing, vibration casting, and timed cure. (Equipment Specification JS-YMN3-801580-A005, Appendix 1.4.5) "Loss On Drying" (LOD) tests are used to measure the amount of water in the as-manufactured 277-4 casting. The as-manufactured 277-4 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 ability of 277-4 to perform its function depends upon the masses of hydrogen and ^{10}B locked inside the high alumina borated cement cast into the body weldment liner inner cavity and the spacer cans. A calculated amount of boron carbide is added to Cat 277-0 dry mix for producing as-manufactured 277-4 material with a volumetric isotopic concentration >7.621 × 10^{20} at/cm^3 of ^{10}B. The additive is boron carbide (B_4C) with small amount of a frit-like compound and trace amounts of unaccounted elements (0.17 wt %). Boron carbide has a theoretical density from 2.45 to 2.52 g/cm^3. The boron carbide grit partical sizes used are also small after passing through mesh sizes of 200 and 40 (63 to 355 μm). 277-4 contains a large amount of hydrated alumina, also known as aluminum trihydrate [$Al(OH)_3$]. It is a nonabrasive powder with a specific gravity of 2.42. Given that both materials are of like density and similar partial size, separation and in homogeneity of 277-4 material is not expected during the controlled vibration casting process.

Table 6.9.3.3-1 provides detailed elemental composition data derived for as-manufactured neutron absorber material at 100 lb/ft^3 and 31.8% LOD. This material description allows for clear specification of reduced boron and water contents required in the evaluation of NCT and HAC. As shown in Table 6.9.3.3-1, both the water and boron components are extracted from the material specification for the neutron absorber, and the constituent weight percents are recalculated accordingly (green box). 277-4 is specified in KENO V.a as three arbitrary materials: **arbmnpmx** with a density of 1.02276 g/cm^3, **arbmnph2o** with a density of 0.509253 g/cm^3, and **arbmboron** with a density of 6.98257e-02 g/cm^3. Model densities for 277-4 inside the body weldment liner inner cavity are reduced by a factor of 0.966893 to account for a material gap at the top of the liner. This material description allows for clear specification of reduced boron and water contents required in the evaluation of NCT and HAC.

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 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 nonvolatile materials was observed in the derivation of material specifications based upon testing. Given that hydrogen must be present for the neutron absorber to be effective, conservative material specifications were derived for minimum hydrogen content and minimum material density.

Tables 6.9.3.3-2 and 6.9.3.3-3 respectively provide NCT and HAC composition data for as-manufactured 277-4 material at minimum density and hydrogen content. Respectively, Tables 6.9.3.3-4 and 6.9.3.3-5 provide NCT and HAC composition data or as-manufactured neutron absorber material at minimum density and boron content. Because the amount of ^{10}B present in the neutron absorber material is near saturation, a change in the hydrogen concentration has a major effect on the neutron multiplication factor, while a change in the amount of boron has a minor effect.

ATTACHMENT 2

CoC MARK-UP

5.(b) Contents (continued)

(4) Unirradiated TRIGA fuel pellets (sections). The fuel is composed of uranium zirconium hydride (UZrH). The uranium concentration in the fuel is a nominal 8.5 weight percent, and the maximum H to Zr ratio in the fuel is 2.0. The maximum uranium enrichment is 70 weight percent U-235. The fuel sections may be from any of three types of fuel elements: standard fuel elements, instrumented standard fuel elements, and fuel follower control rods. The U-235 mass for standard and instrumented fuel elements is a nominal 136 grams per element, and the U-235 mass for fuel follower control rods is a nominal 112 grams per element. Each fuel element contains three fuel sections, which are removed from the cladding for transport. The fuel sections are approximately 5 inches in length; the approximate diameter is 1.44 inches for the standard and instrumented fuel elements, and 1.31 inches for the fuel follower control rods. The fuel sections are packaged within stainless steel or tin-plated carbon steel convenience cans, with a maximum of three fuel sections per convenience can. Fuel sections from different fuel elements may not be mixed within a single convenience can. A maximum of three convenience cans may be loaded into a single package. No spacers are required. The maximum quantity of fissile material per package is 408 grams U-235. The CSI is 0.0.

INSERT 6

The vent holes on the outer steel drum shall be capped closed during transport and storage to preclude entry of rain water into the insulation cavity of the drum.

7. Content forms may not be mixed in a single ES-3100 containment vessel.

8. Any combination of convenience can sizes is allowed in a single package, as long as the total height of the can stack (including silicone rubber pads and spacers, if required) does not exceed the inside working height of the containment vessel (31 in). Any closure on the convenience can is allowed.

9. Empty convenience cans, spacers, silicone rubber pads, and/or stainless-steel scrubbers (i.e., stainless steel trimmings that act as dunnage) may be used to fill the void space in the containment vessel. Empty convenience cans must have a minimum 0.125 in diameter hole through the lid.

10. The contents and the convenience cans may be bagged or wrapped in polyethylene for contamination control provided the limits of Condition No. 5.(b) are met.

11. Transport by air is not authorized, except for shipment of unirradiated TRIGA fuel pellets, as described and limited in Condition No. 5(b)(4).

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

(a) The package shall be prepared for shipment and operated in accordance with the Package Operations in Section 7 of the application, as supplemented.

(b) Each package must meet the Acceptance Tests and Maintenance Program of Section 8 of the application, as supplemented.

CoC MARKUP SUGGESTION

USA/9315/B(U)F-96
ES-3100 Package

Current CoC, Section 5.(b)(4)

5.(b) Contents (continued)

(4) Unirradiated TRIGA fuel pellets (sections). The fuel is composed of uranium zirconium hydride (UZrH). The uranium concentration in the fuel is a nominal 8.5 weight percent, and the maximum H to Zr ratio in the fuel is 2.0. The maximum uranium enrichment is 70 weight percent U-235. The fuel sections may be from any of three types of fuel elements: standard fuel elements, instrumented standard fuel elements, and fuel follower control rods. The U-235 mass for standard and instrumented fuel elements is a nominal 136 grams per element, and the U-235 mass for fuel follower control rods is a nominal 112 grams per element. Each fuel element contains three fuel sections, which are removed from the cladding for transport. The fuel sections are approximately 5 inches in length; the approximate diameter is 1.44 inches for the standard and instrumented fuel elements, and 1.31 inches for the fuel follower control rods. The fuel sections are packaged within stainless steel or tin-plated carbon steel convenience cans, with a maximum of three fuel sections per convenience can. Fuel sections from different fuel elements may not be mixed within a single convenience can. A maximum of three convenience cans may be loaded into a single package. No spacers are required. The maximum quantity of fissile material per package is 408 grams U-235. The CSI is 0.0.

INSERT C

Add the following as the second paragraph of Section 5.(b)(4):

The unirradiated TRIGA fuel, as described, may also be shipped as clad fuel elements. Each fuel element will be a maximum of 15 inches-long (fuel pellet section only of the fuel element) within stainless steel cladding, plus up to ½ -inch of crimped stainless steel cladding at each end. The cladding diameters are 1.48 inches for the standard and instrumented fuel elements and 1.35 inches for the fuel follower control rods. The fuel elements are packaged in stainless steel or tin-plated carbon steel convenience cans. Maximum can loading is 3 fuel elements and maximum package loading is 3 fuel elements. No spacers are required. The maximum quantity of fissile material per package is 408 grams U-235. The CSI is 0.0.