Safety Evaluation Report for the

Safety Analysis Report for Packaging

Y-12 National Security Complex

Model ES-3100 Package with Bulk HEU Contents

SP-PKG-801940-A001, Revision 0

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Background

The U.S. Department of Energy (DOE) Packaging Certification Program (PCP) and Nuclear Regulatory Commission (NRC) have established an agreement concerning the use of NRC Certificate of Compliance (CoC) and DOE CoCs for DOE shipments. DOE shipments to the greatest extent possible will be made under DOE CoCs and NRC CoC shall mainly be issued when an NRC license holder will be making shipments. Under this agreement on October 5, 2009 the DOE CoC USA/9315/B(U)F-96 (DOE), Revision 0 for the Model No. ES-3100 was issued based on Revision 9 of the NRC CoC USA/9315/B(U)F-96. The NRC CoC was based on the Safety Analysis Report, Y-12 National Security Complex, Model ES-3100 Package with Bulk HEU Contents, Y/LF-717, Revision 3, May 2009, and the BWXT Y-12, L.L.C, Y-12 National Security Complex Packaging Engineering, Quality Assurance Program Plan, QAP-Y-91-273860-1, Rev. 11, 10/26/2006. Revision 0 was issued because the overwhelming majority of the shipments made using the ES-3100 packaging are DOE Shipments made by DOE contractors and therefore can be made under a DOE CoC and do not need to be made under the NRC CoC. There is a cost saving in using DOE CoC for shipments and there is a cost saving in not paying for additional reviews by NRC. The Revision 1 DOE CoC for the 9315, issued under this action with added new contents, is based on a Safety Evaluation Report for Packaging that meets the DOE requirements for issuing DOE CoCs.

SUMMARY

By a letter dated September 30, 2009, B&W Y-12 LLC submitted to the U.S. Department of Energy (DOE) Packaging Certification Program (PCP), Office of Packaging and Transportation (EM-45), an application requesting that the DOE issue a Certificate of Compliance (CoC) for the ES-3100 package with new bulk HEU contents such as the TRIGA fuel; the SNAP fuel; research reactor fuel components; items such as clad U-Zr, U-Al, U_3O_8-Al, UO_2-Mg and UO_2; and uranium compounds. Along with the application request, a Safety Analysis Report for Packaging (SARP), identified as SP-PKG-801940-A001, Rev. A, dated September 25, 2009 was submitted with the intent to provide documentation that (a) the ES-3100 package design satisfies the Department of Transportation (DOT) and Nuclear Regulatory Commission (NRC) relevant regulatory safety requirements as specified in Title 49 of the Code of Federal Regulations (CFR) Part 173 and Title 10 of the CFR Part 71, respectively; and (b) the SARP was prepared in accordance with DOE Order 460.1B and in the format specified in NRC Regulatory Guides (RGs) 7.9 and 7.10.

The DOE PCP staff reviewed the Rev. A SARP and generated forty-six (46) Q1 questions on the nine chapters in the SARP. The DOE PCP staff also generated three (3) additional questions during the confirmatory review. The applicant responded to all the questions and provided revisions to the SARP. The DOE PCP staff also conducted independent confirmatory evaluation of the SARP. On the basis of the statements and representations in Revision 0 of the SARP and the DOE PCP staff's confirmatory evaluation, as summarized in this Safety Evaluation Report (SER), DOE PCP finds that the design and performance of the ES-3100 package is acceptable and will provide reasonable assurance that the regulatory requirements of 49 CFR Part 173, 10 CFR Part 71 and DOE Order 460.1B have been met.

In addition to adding the new bulk HEU contents, Revision 1differs from Revision 0 of the DOE CoC (and the NRC CoC it was based on) and the SARP used for justification of Revision 1 for the DOE CoC also differs from the SAR used to justify the NRC CoC. The DOE CoC and the supporting SARP replaced the neutron absorber can material from carbon steel to stainless steel; confirmed and established the technical basis for flow forming as an acceptable fabrication method for the Containment Vessel; and provided additional clarification of the role of efflorescence on uranyl nitrate crystals on maximum normal operating pressure and max internal pressure.

Reference

Safety Analysis Report for Packaging, Y-12 National Security Complex, Model ES-3100 Package with Bulk HEU Contents, prepared by B&W Y-12 LLC, Oak Ridge, TN, SP-PKG-801940-A001, Rev. 0, August 12, 2010.

1. GENERAL INFORMATION AND DRAWINGS

1.1 Packaging Description

The ES-3100 packaging, which is depicted in Figure 1.1 below, is a cylindrical container that is approximately 43.5 in. (110 cm) in overall height, including the cover and lid and approximate 19 in. (49 cm) in overall diameter.

The packaging is composed of an outer drum assembly and an inner containment vessel (CV). The main functions of the packaging are to provide containment, shielding, and nuclear criticality safety. Table 2.7 of the SARP provides detailed material specifications for the packaging components.

Outer Drum Assembly

The outer drum assembly consists of (a) a reinforced stainless steel, standard military specification 30-gallon drum with an increased length; (b) a cylindrical layer of castable refractory material (Kaolite 1600™), which is comprised of concrete and vermiculite, and which acts as both an impact-absorbing and thermal-insulating material; (c) a cylindrical layer of castable refractory (277-4 special dry mix) for neutron attenuation; (d) an inner steel liner; and (e) a removable top plug that also has a layer of the castable refractory material (Kaolite 1600™) for impact absorption and thermal insulation.

The 30-gallon drum is manufactured from 16-gauge Type 304 or 304L stainless steel. The fabrication is accomplished according to requirements specified in NUREG/CR-3854, and is in accordance with the dimensional requirements of MIL-D-6054F as modified according to Drawing M2E801580A004. The inner liner is also manufactured from Type 304 or 304L stainless steel.

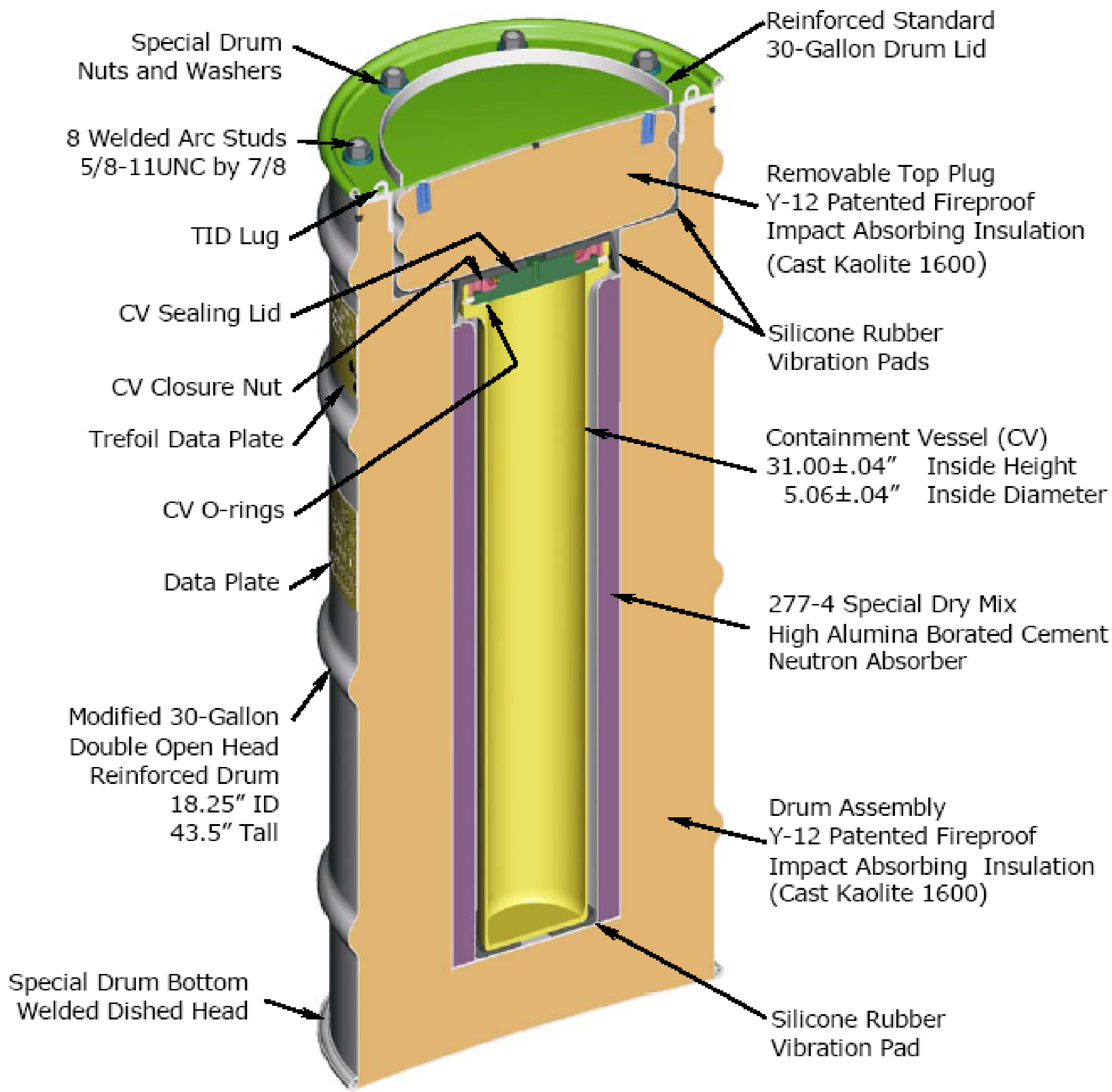

Figure 1.1. Schematic of the ES-3100 shipping package.

Containment Vessel (CV)

The CV is placed inside the outer drum assembly, surrounded by the neutron-attenuating and the impact-absorbing and thermal-insulating materials. It is approximately 32 in. (82 cm) in overall height and 5 in. (13 cm) in overall diameter, and is constructed of Type 304L stainless steel. The lid assembly consists of a sealing lid; a closure nut; an external retaining ring that holds both the assembly and closure nut together; and double ethylene-propylene elastomer O-rings. The double O-rings in the top flange of the CV permit leak testing of the CV. The containment boundary consists of the 0.1 in. (0.254 cm) thick CV body, the CV sealing lid assembly, and the inner ethylene-propylene elastomer O-ring.

Drawings

The drawings that pertain to the ES-3100 package are listed in Table 1.0.

Table 1.0. List of Drawings Pertaining to the ES-3100 Package

Drawing No.	Revision	Title
M2E801580A001	C	Drum Assembly
M2E801580A002	B	Body Weldment
M2E801580A003	B	Inner Liner Weldment (2 sheets)
M2E801580A004	B	Double Open Head Reinforced Drum
M2E801580A005	D	Misc. Details
M2E801580A006	B	Drum Lid Weldment
M2E801580A007	B	18.25" Diameter Drum Lid
M2E801580A008	B	Top Plug Weldment
M2E801580A009	C	Pad Details
M2E801580A010	E	Data Plate Details
M2E801580A011	D	Containment Vessel Assembly
M2E801580A012	C	Containment Vessel Body Assembly (2 sheets)
M2E801580A013	C	Containment Vessel O-ring Details
M2E801580A014	B	Containment Vessel Lid Assembly
M2E801580A015	C	Containment Vessel Sealing Lid
M2E801580A016	B	Containment Vessel Closure Nut
M2E801580A023	C	Containment Vessel Leak Test Assemblies
M2E801580A024	B	Containment Vessel Vibration Absorbing Silicone 4.25" Can Pad
M2E801580A031	E	Main Assembly
M2E801580A037	D	Consolidated Assembly Drawing (3 sheets)
M2E801580A043	0	Heavy Can Spacer Assembly (SST)
T2E801827A008	A	Leak Check Flange Assembly

1.2 Contents

The contents to be shipped in the ES-3100 package consist of bulk HEU in the form of oxide (UO_2, UO_3, U_3O_8, U_3O_8-Al, UO_2-Mg, and UO_2-Zr); uranium metal and alloy in the form of solid geometric shapes or broken pieces; uranium compounds; uranyl nitrate crystals (UNX); and research reactor fuel elements or fuel components.

The maximum content decay heat load shall not exceed 5.0 watts.

Uranium metal and alloy pieces must have a surface-area-to-mass ratio of not greater than 1.00 cm^2/g or must not pass freely through a 3/8-inch (0.0095 m) mesh sieve, or equivalent size-grading system. The uranium metal must also have had no more than a limited contact with water and been subsequently dried. Particles and small shapes that do not pass this size restriction, as well as powders, foils, turnings, and wires, are not permitted, unless they are in a sealed container under an inert cover gas. Uranium material or alloy which has been stored in water or is visibly wet at the time of packaging is not authorized to be shipped in this package.

The radioactive materials are placed in convenience cans or bottles. Typical loading of the bulk HEU into the packaging, using convenience cans is depicted in Figure 1.2 of the SARP. Typical shipping configurations inside the CV are depicted in Figure 1.4 of the SARP.

The maximum mass of off-gassing packing materials in the containment vessel (e.g., polyethylene containers or bagging, silicone rubber pads, nylon bags, etc.) shall not exceed 500 grams. Off-gassing packing materials may be any type of hydrogenous material, except in the case of shipping uranium in the form of broken metal, in which case the hydrogenous material must have a hydrogen atom density less than or equal to that of water. With the use of Teflon bottles as convenience containers, an additional 990 grams of off-gassing material is authorized in the containment vessel. If closed convenience cans with an outer diameter greater than 4.25 inches are used, the containment vessel cannot contain any materials that off-gas.

The maximum concentrations of uranium isotopes that are permitted in the ES-3100 package content are listed in Table 1.1.

Table 1.1. Uranium Concentration Limits

Uranium Isotope	Limit
^{232}U	0.040 µg/gU
^{233}U	0.006 g/gU
^{234}U	0.02 g/gU
^{235}U	1.00 g/gU
^{236}U	0.40 g/gU
^{238}U	1.00 g/gU

The bounding uranium isotopic concentration in oxide is listed in Table 1.2.

Table 1.2. Bounding Uranium Isotopic Concentration in Oxide

Isotope	Bounding Limit
^{232}U	40 ppb
^{233}U	200 ppm
^{234}U	2.0 wt %
^{235}U	100.0 wt %
^{236}U	40.0 wt %

In addition to the uranium isotopes shown in Table 1.1, transuranic isotopes (with the exception of ^{237}Np) may be present in the HEU metal and alloy contents, and are allowed to be transported according to the limits shown below.

Type and Form of Contents / Mode of Transport	Concentration Limits		
	^{237}Np (in All Forms)	**Transuranics Other Than ^{237}Np (As Metal or Alloy)**	**Transuranics Other Than ^{237}Np (as Other Than Metal or Alloy)**
Ground	0.0250 g/gU	800.0 µg/gU	40.0 µg/gU
Air	0.0250 g/gU	40.0 µg/gU	40.0 µg/gU

<u>Weights and Contents Descriptions</u>

The maximum gross shipping weight of the ES-3100 package, with any contents, is 420 lb (190.5 kg).

The weight of the radioactive contents in the ES-3100 package is limited to 77.6 lb (35.2 kg). The maximum weight of all contents, including the radioactive contents, the convenience cans or bottles, can spacers, polyethylene bagging and other packing materials, is limited to 90 lb (40.82 kg).

<u>Radioactive/Fissile Constituents</u>

The maximum number of A_2s is 4752.8 (at 70 yrs) and the maximum activity is 0.72554 TBq (at 10 yrs).

The following loading limits are imposed based on the mode of transport:

a) For ground transport, fissile material loading limits are presented in Tables 1.3 and 1.3a.
b) For air transport, HEU in the form of metal/alloy and research reactor fuel components/items, the fissile material mass loading limits are presented in Table 1.3b.
c) The loading limit for mixed-mode transport is taken as the most restrictive limit for either ground or air mode of transportation (Table 1.3 or 1.3b).

Table 1.3 –Authorized Content and Fissile Mass Loading Limits for Ground Transportation [a, b, c]

Content description		Enrichment	CSI	No Spacers, ^{235}U (kg)	Basis for limit	277-4 can Spacers, [d] ^{235}U (kg)	Basis for limit
Solid HEU metal or alloy (specified geometric shape) [e]	Cylinder A	≤ 100%	0.0	15.000	Crit.	25.000	Crit.
	Cylinder B	≤ 100%	0.0	18.000	Crit.	30.000	Crit.
	Square bars	≤ 100%	0.0	30.000	Crit.	35.200 [f]	Struct.
	Slugs	≤ 95%	0.0	17.374	Crit.	-	-
	Slugs	≤ 80%	0.0	-	-	29.318	Crit.
	Slugs	> 80%, ≤ 95%	0.0	-	-	24.324	Crit.
	Slugs	>80%, ≤ 95%	0.4	-	-	34.749	Crit.
Broken HEU metal or alloy [g]		>95%, ≤ 100%	0.0	Spacers req'd [d]		2.774	Crit.

			0.4	Spacers req'd		5.549	Crit.
			0.8	Spacers req'd		9.248	Crit.
			2.0	Spacers req'd		13.872	Crit.
			3.2	Spacers req'd		24.969	Crit.
		>90%, ≤ 95%	0.0	Spacers req'd		3.516	Crit.
			0.4	Spacers req'd		6.154	Crit.
			0.8	Spacers req'd		10.549	Crit.
			2.0	Spacers req'd		18.461	Crit.
			3.2	Spacers req'd		26.373	Crit.
		>80%, ≤ 90%	0.0	Spacers req'd		3.333	Crit.
			0.4	Spacers req'd		7.500	Crit.
			0.8	Spacers req'd		12.500	Crit.
			2.0	Spacers req'd		20.000	Crit.
			3.2	Spacers req'd		28.334	Crit.
		>70%, ≤ 80%	0.0	2.967	Crit.	4.450	Crit.
			0.4	5.192	Crit.	8.900	Crit.
			0.8	8.900	Crit.	16.317	Crit.
			2.0	17.059	Crit.	25.218	Crit.
			3.2	27.692	Crit.	28.184	Crit.
		>60%, ≤ 70%	0.0	3.249	Crit.	5.198	Crit.
			0.4	5.848	Crit.	12.996	Crit.
			0.8	13.646	Crit.	20.793	Crit.
			2.0	21.444	Crit.	24.692	Crit.
			3.2	24.692	Crit.	24.692	Crit.
		≤ 60%	0.0	5.576 kg U	Crit.	11.154 kg U	Crit.
			0.4	14.872 kg U	Crit.	28.813 kg U	Crit.
			0.8	28.814 kg U	Crit.	35.20 kg U [f]	Struct.
			2.0	35.20 kg U [f]	Struct.	35.20 kg U [f]	Struct.
			3.2	35.20 kg U [f]	Struct.	35.20 kg U [f]	Struct.
HEU oxide [h, j] (UO_2, UO_3, U_3O_8, U_3O_8-Al, UO_2-Mg, UO_2-Zr)		≤ 100%	0.0	15.13 kg oxide 9.682 kg ^{235}U 921 g carbon	Crit. H_2 gen.	Spacers not req'd	
Research reactor fuel components or items[k]	$UZrH_x$ (TRIGA)	≤ 20%	0.0	0.921[i]	Crit.	Spacers not req'd	
		> 20%, < 93%	0.0	0.408[i]	Crit.	Spacers not req'd	
	$UZrH_x$ (SNAP)	≥ 93%	0.0	0.857[i]	Crit.	Spacers not req'd	
	U-Zr	≤ 100%	0.0	See limit for broken metal or alloy	Crit.	Spacers not req'd	
	U-Al	≤ 100%	0.0	7.333 kg U-Al 525 g U 473 g ^{235}U	Crit.	Spacers not req'd	
	U_3O_8-Al	≤ 100%	0.0	See limit for HEU oxide	Crit.	Spacers not req'd	
	UO_2	≤ 100%	0.0	21.937 kg UO_2	Crit.	Spacers not	

				19.308 kg ^{235}U		req'd	
	UO_2-Mg	≤ 100%	0.0	See limit for HEU oxide	Crit.	Spacers not req'd	
Uranium compounds	UF_4	≤ 100%	0.0	3 kg UF_4 2.267 kg ^{235}U	Crit.	Spacers not req'd	
	UO_2F_2	≤ 100%	0.0	3 kg UO_2F_2 2.067 kg ^{235}U	Crit.	Spacers not req'd	
	UC	≤ 100%	0.0	2 kg UC 1.815 kg ^{235}U	Crit.	Spacers not req'd	
	UN	≤ 100%	0.0	2 kg UN 1.888 kg ^{235}U	Crit.	Spacers not req'd	
	TRISO	≤ 100%	0.0	2 kg TRISO 1.815 kg ^{235}U	Crit.	Spacers not req'd	

a With the exception of the UNX crystals (Sec. 1.2.2.2), which are loaded for shipment in crystalline solid form, 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. M2E801580A043 (Appendix 1.3.7)

e Geometries of solid shapes are as follows:
- Cylinder A is a 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.
- Square bars are no larger than 2.29 in. × 2.29 in. (cross section): maximum of 1 bar per can.
- Slugs are a maximum of 1.5 in. diameter × 2.0 in. tall: a maximum of 10 per convenience can where the actual number permitted is restricted by the stated loading limit.

f Maximum planned content weight is 35.20 kg. Maximum analyzed for criticality safety is 35.32 kg.

g Mass limits for alloys (uranium with aluminum, molybdenum, zirconium, stainless steel, titanium, tungsten, niobium, silicon, or vanadium) must assume that non-uranium portion is ^{235}U.

h Seal time must be 12 months or less. Seal time is the length of time after the ES-3100 containment vessel is sealed that the shipment must be complete.

i Evaluation limit based on specific fuel type as opposed to a maximum calculated limit for $UZrH_2$.

j Allowable HEU bulk oxide densities are 2.0-6.54 g/cm^3.

k For SNAP $UZrH_x$, $x \leq 2$. For TRIGA $UZrH_x$, $x \leq 1.6$.

Table 1.3a – Loading Limits for Uranyl Nitrate Crystals for Ground Transport

Product [a, b]	Seal time [c] (months)	CSI	Loading limit [d, e] (kg UNX)	U content [f] (wt%)
UNX $0 < X \leq 3$	2	0.4	11.90	$52 < U \leq 61$
	4	0.4	6.70	$52 < U \leq 61$
UNX $X>3$	2	0.4	9.17	$46 < U \leq 52$
	4	0.0	4.75	$46 < U \leq 52$

a UNX is uranyl nitrate hydrate $[UO_2(NO_3)_2 * XH_2O]$ where $0 < X \leq 6$.

b Must be shipped in Teflon bottles.

c Seal Time – length of time after the ES-3100 containment vessel is sealed that the shipment must be complete. Seal times listed in this table are much lower than the calculated values shown in Table 3.5.4.1 in Appendix 3.5.4 and have been reduced for additional conservatism.

d Total mass of UNX crystals. Spacers are not required for this content type.

e Loading limits for uranyl nitrate crystals are based on hydrogen generation calculations presented in Appendix 3.5.4.

f Enrichment up to 100%.

Table 1.3b - Loading Limits for HEU metal or alloy and Research Reactor Fuel Components or Items for Air Transport [a, b, c]

Content description	Enrichment	CSI	^{235}U (kg)
HEU metal or alloy [d]	≤ 100%	0.0	7.00
Research reactor fuel components or items ($UZrH_x$, U-Zr, U-Al, U_3O_8-Al, UO_2, UO_2-Mg)[e]	≤ 20%	0.0	0.921
	> 20%	0.0	0.408

a All limits are expressed in kg ^{235}U unless otherwise indicated.

b Mass loadings cannot be rounded up.

c The loading limit for mixed-mode transport is taken as the most restrictive limit for either ground or air mode of transportation (Table 1.3 or 1.3b).

d Mass limits for alloys (uranium with aluminum, molybdenum, zirconium, stainless steel, titanium, tungsten, niobium, silicon, or vanadium) must assume that non-uranium portion is ^{235}U.

e For SNAP $UZrH_x$, $x \leq 2$. For TRIGA $UZrH_x$, $x \leq 1.6$.

1.3 Criticality Safety Index

On the basis of the results of the criticality safety analysis presented in Chapter 6 of the SARP, the DOE PCP staff has confirmed using the procedure in 10 CFR 71.59(b), that the Criticality Safety Index (CSI) is zero (0) for the newly added contents, i.e., for the TRIGA fuel; the SNAP reactor fuel; research reactor fuel components; items such as clad U-Zr, U-Al, U_3O_8-Al, UO_2-Mg and UO_2; and uranium compounds. The U-Zr was not analyzed in the SARP. It can be shipped in accordance with the limits and CSI for uranium alloy with the non-uranium components included as uranium.

For the contents already approved by NRC in Rev. 9 of CoC 9315, the CSI varies from 0 to 3.2, depending upon the radioactive contents of the package.

1.4 Radiation Level and Transport Index

The DOE PCP staff has confirmed the maximum radiation transport index (TI) to be 5.8, which is lower than 10, the TI limit in 10 CFR 71.47(a) for non-exclusive use shipment. The actual TI of the ES-3100 package will be determined by measurement prior to shipment.

1.5 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, DOE PCP finds that the general information (and drawings) presented in Chapter 1 of the SARP is acceptable. Evaluations of design and performance of the package for safety and regulatory compliance with respect to structural, thermal, containment, shielding, criticality safety, operating procedures, acceptance tests and maintenance, and quality assurance are given in the remaining sections of this SER.

2. STRUCTURAL

2.1 Discussion

The DOE PCP staff reviewed the structural design and performance of the ES-3100 package described in Chapter 2 of the SARP. The DOE PCP staff also performed finite-element analysis to independently confirm the structural performance of the package during the Hypothetical Accident Conditions (HAC) specified in 10 CFR 71.73. The review and analyses were focused on (a) one of the fabrication methods for the containment vessel (CV), i.e., flow forming, (b) the conditions of the certification tests, (c) the mechanical properties of materials, and (d) the verification of the technical basis of the structural design and performance described in the SARP.

2.2 Fabrication of the CV (Flow Forming)

The CV of the ES-3100 package is constructed of Type 304L stainless steel (SS). It may be fabricated by welding a stainless-steel cylindrical pipe to a machined flat-head bottom at one end and a machined top flange at the other end, or by flow forming to create the complete body (flat bottom, cylindrical body, and flange), followed by solution annealing and subsequent rapid cooling to room temperature. During flow forming, a large amount of cold work is introduced into the work piece resulting in significantly higher strength and lower ductility. The low ductility in the flow formed CV can be ameliorated by stress relief at 1600°F for 30 minutes, or solution anneal at 1900°F for 30 minutes (M. Fonte, "*Package Plutonium Metal and Oxide in Flowformed Canisters,*" Intl. J. Packaging, Transport, Storage & Security of Radioactive Material, Vol. 21, No. 2, 2010). Both heat treatment processes would recrystalize the elongated grains in the CV and remove residual stresses introduced by the cold work.

However, the martensite created by flow forming may not all transform back to austenite by heat treatment; for example, annealing of Type 304L SS (89% initial strain-induced martensite) at 1,832°F for 90 s transformed ≈92% of martensite back to austenite (Ravi Kumar, B., et al. "*Role of Strain-Induced Martensite on Microstructural Evolution during Annealing of Metastable Austenitic Stainless Steel*," J of Mater. Sci. 45, 2010, pp. 911–918). Lamada Research estimated that the volume of retained austenite is 97.6 ± 2.0 % in the flow-formed and annealed Type 304L SS container. A conservative estimate of martensite content, assuming absence of ferrite, is 4.4 % [Lamada Research, Inc. "*X-Ray Diffraction Volume Percent Retained Austenite Analysis of Three 304L Stainless Steel Coupons*," Report No. 1485-15663, May 12, 2010, prepared for Dynamic Flowform Corp.] It is also possible that not all of the residual stresses and strains have been relieved during solution annealing, especially at interfaces between the remaining martensite and austenitic matrix that can be sensitized at lower temperatures and in shorter times than required for the austenitic grain boundary. (See Briant, C. L., and A.M. Ritter, "*The Effect of Deformation-Induced Martensite on the Sensitization of Austenitic Stainless Steels*," Metall. Trans. A, Vol. 11A, 1980, pp. 1009–1017.) The main concern is the susceptibility of the heat-treated, highly cold-worked Type 304L SS CV to corrosion and stress corrosion cracking.

The flow-formed container should be rapidly cooled (for example, by water quenching) from the heat treatment temperature to room temperature to avoid sensitization, which is a standard procedure for Type 304 SS. Because of its lower carbon content (0.03% max.), a Type 304L SS flow-formed CV should be more resistant to intergranular corrosion and stress corrosion cracking than one made of Type 304 SS. In the absence of deformation, the temperature at which transformation of austenite to martensite occurs is significantly below room temperature. Therefore, no additional martensite, except the residual martensite present at the end of annealing process, is likely to form in the heat treated and rapidly cooled flow-formed containers. It should be noted that the presence of martensite itself is not harmful, unless tensile stresses and a corrosive environment are present for a sufficient length of time to cause corrosion and stress corrosion cracking. As discussed in Section 2.2.2 of the SARP, the CV will not be exposed to the corrosive environment of halide gases; therefore, no corrosion or stress corrosion cracking of the CV (flow-formed or welded) is expected during packaging operation. It should also be noted that penetrant testing of the surface of the flow formed and annealed CV for the ES-3100 package did not detect any cracking, which could have been introduced by the flow forming process.

The 30-minute heat treatment temperature of 1,900 ºF appears more desirable than 1,600 ºF because the higher solution annealing temperature resulted in a homogeneous microstructure, and more uniform mechanical properties exceeding the minimum mechanical properties (i.e., tensile strength, yield strength, and elongation) for annealed Type 304L SS provided in the ASME Boiler and Pressure Vessel Code (BPVC), Section II, Part A, Ferrous Material Specifications, 2007 Edition (May, C. G., "*Rationale for Flowforming of Containment Vessels of Model 9975, 9977, and 9978 Radioactive Material Shipping Packages,*" Attachment to SRNL-L1300-2010-00026, February 25, 2010). For example, the annealed flow-formed container has elongation in the range of 58-65%, compared to the minimum of 30% as required by the ASME BPVC. A properly heat-treated, flow-formed CV, therefore, will meet the ASME BPVC mechanical-property requirements and is more efficient to fabricate than a welded CV. It was similarly concluded that flow forming is an adequate and efficient method for mass production of small radioactive materials containment vessels, provided that the fabrication process is controlled with quality assurance (G. Mok et al., "*On the Application of Flow Forming to the Fabrication of Type B Radioactive Material Package Containment Vessels*," Lawrence Livermore National Laboratory report, LLNL-TR-442131, Rev. 1, May 25, 2010.)

2.3 Structural Evaluation

The SARP documented the structural evaluation of the package performed by a combination of physical testing and finite-element analysis using the computer code LS-DYNA. Five full-scale ES-3100 prototype units (TU-1 to TU-5) were tested in accordance with a test plan specified in Tables 2.18 and 2.19 of the SARP to demonstrate that the design of the package meets the requirements under NCT and HAC conditions prescribed in 10 CFR 71.71 and 71.73, respectively. Following the certification tests, several design changes were made to the packaging: (1) replacing the neutron poison BoroBond4 with Cat 277-4; (2) changing the mid-liner design; and (3) increasing (by 0.06 in.) the thickness of the silicone rubber pad on the drum assembly bottom. These changes were evaluated by finite-element analyses and the effects were found to be negligible. All analytical results are documented in References DAC-EA-801699-A001 and DAC-EA-801699-A002.

For the confirmatory structural analyses, the DOE PCP staff used the general-purpose finite-element code ABAQUS. The DOE PCP staff constructed models for the prototype packages and compared the analysis results with the data from the certification tests and the analysis results reported in the SARP. The calculated maximum effective plastic strains of the CV components are nearly identical to those reported in the SARP; the maximum equivalent plastic strains (called PEEQ in ABAQUS) of the confinement drum components show discrepancies; however, because the calculated PEEQs are highly localized and they are sensitive to the geometrical details and the mesh of the of finite-element models constructed for ABAQUS and LS-DYNA. For example, the diameter of the stud weld in the LS-DYNA model is 0.5488 in., whereas the ABAQUS model used a weld diameter of 0.82 in. at the weld base according to the dimension shown in the drawings provided in the appendix to Chapter 1 of the SARP. To be conservative, all of the confirmatory analyses were performed at -40^{o}F. The confirmatory analyses used 1.2 in./in. as the failure strain for Type 304 and 304L stainless steels (see *High-Temperature Property Data: Ferrous Alloys*, ASM International, 1988, p. 9.28.)

The DOE PCP staff also performed additional confirmatory analyses to evaluate the following:

1) Effects of friction coefficient during slapdown drop
2) Effects of reduced test temperature on structural performance
3) Effects of broken Kaolite (impact limiter material) on structural performance

Effects of friction coefficient during slapdown drop

In the SARP analysis of the slapdown drop of the package at a shallow angle (12^o), the friction coefficient (μ) was assumed to be zero because, according to the SLAPDOWN computer code, the lid-end velocity of the package was the highest for $\mu = 0$. However, the kinetic energy associated with the rotational motion of the package can be higher for a finite μ and produce higher lid-end velocity. (V.L. Bergmann and D.J. Ammerman, "*An Analysis of Parameters Affecting Slapdown of Transportation Packages*," Waste Management 1991 Symposium, February 27, 1991, Tucson, AZ). Therefore, the DOE PCP staff investigated the effects of a range of friction coefficients, i.e., μ = 0.0, 0.1, 0.2, 0.3 and 0.4, on the structural performance of the package during the bottom-to-lid-end slapdown drops. The analysis results show the highest calculated maximum plastic strain (0.05538 in./in.) in the CV body for $\mu = 0.2$, which is only slightly higher than those calculated for other values of μ, whereas the CV lid and the closure nut show no plastic deformation. The calculated maximum plastic strain in the drum body varies between 0.4646 and 1.194 in./in. (the highest value is also obtained for $\mu = 0.2$), which is still lower than the failure strain of 1.2 in./in. and is highly localized at the top curl edge under the drum lid where the package impacts the unyielding surface. The differences in the plastic strains calculated for other drum assembly components are small, and the maximum plastic strains in the drum lid and the drum weld studs are significantly lower than the failure strain. Therefore, the DOE PCP staff concluded that the variation in the coefficient of friction would not challenge the integrity of the CV, or the confinement of the CV during the shallow-angle slapdown drop.

Effects of reduced test temperature on structural performance

Among the five prototype units tested, only TU-2 was chilled to -40°F for the side drop tests under both NCT and HAC conditions. Other prototype units were tested at ambient temperature. The DOE PCP staff performed finite-element analyses of the package under three HAC test sequences at -40^{o}F: (1) a slapdown drop followed by a side crush, (2) a CG-over-lid-corner (CGOC) drop followed by a corner crush, and (3) a top-down drop followed by a top crush that represented the worst combinations of sequential drop and crush tests to inflict the maximum damage to the package. The calculated maximum plastic strains of the packaging components under the -40^{o}F test conditions are nearly identical to those obtained from the tests at ambient temperature. For example, the CV components showed slightly higher calculated plastic deformation following drops at -40^{o}F than at ambient temperature because the impact-limiting material (Kaolite 1600) is more rigid at lower temperature. On the other hand, according to the analysis results, Kaolite appeared to have dissipated more of the impact energy due to the plastic deformation at -40^{o}F than at ambient temperature. In all three HAC sequential tests at -40^{o}F, the calculated plastic strains are $\leq$ 0.063 in./in. for the CV body, $\leq$ 0.006 in./in. for the CV lid, and 0.0 for the CV closure nut. In addition, the calculated maximum plastic strains in the drum lid and the drum studs are well below the failure strain of 1.2 in./in. For the CGOC and the top-down drops, the calculated maximum plastic strain in the drum body is 0.3326 in./in., which is less than the failure strain of 1.2 in./in. For the slapdown drop and the side-crush test, two small regions (total area $\approx$ 0.21 in.^{2}) at the top curl edge of the drum body under the lid show plastic strains of 1.222 in./in. slightly exceeding the failure strain; these highly localized deformations, however, are not likely to result in any opening of the drum surface causing loss of confinement.

On the basis of the analysis results of the three HAC sequential tests, the DOE PCP staff concluded that the reduced temperature of -40^{o}F does not adversely affect the structural performance of the ES-3100 package.

Effects of broken Kaolite on structural performance

After vibration testing, radiography of the prototype package TU-4 showed that the lower half of the impact limiter was broken into small pieces. TU-4 was reassembled and subjected to the HAC sequential tests. After vibration and impact testing, many three-dimensional curving cracks were seen around the impact areas, and the inner liner was visibly deformed. The DOE PCP staff performed finite-element analyses to evaluate the effects of the broken Kaolite on the structural performance of the package under two HAC sequential tests: (1) a slapdown drop followed by side crush, and (2) a top-down drop followed by top crush. In the finite-element analysis of these tests, the upper half of the Kaolite was modeled as a monolithic piece and the lower half was modeled with small, stacked pieces without any void between them. The small pieces of Kaolite are in good contact with each other before the impact, and can slide relative to one another with an assumed friction coefficient of 0.1.

Comparison of the analysis results for packages with the Kaolite either intact or broken into small pieces shows that the broken Kaolite in the lower half does not affect the structural performance of the package; the calculated maximum plastic strains of the major packaging components are only slightly different. The calculated plastic strains for the CV components are small, while the CV closure nut shows no plastic deformation. The calculated maximum plastic strains for the drum assembly are well below the failure strain, and the drum should maintain its integrity and provide confinement for the CV.

The SARP reported that in the TU-3 certification test, one drum stud was sheared off during the corner crush test following the CGOC drop. However, neither the LS-DYNA analysis in the SARP nor the DOE PCP staff analysis using ABAQUS predicted such a failure. Since the drum assembly is closed with eight weld studs and closure nuts, the loss of one stud should not result in the loss of confinement.

2.4 Conclusion

On the basis of the statements and the representations in the SARP and the DOE PCP staff's confirmatory evaluation, DOE PCP finds that the structural design and performance of the ES-3100 package presented in Chapter 2 of the SARP is acceptable, and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71 have been met.

3. THERMAL

3.1 Discussion

The DOE PCP staff reviewed the thermal design and performance of the ES-3100 package described in Chapter 3 of the SARP. The DOE PCP staff also performed confirmatory analyses to independently confirm the thermal performance of the package during NCT and HAC. The review and analysis were focused on the material properties, the temperature limits, the maximum temperatures and the maximum thermal stresses of the packaging components, and the maximum normal operating pressure (MNOP) during NCT and the maximum pressure in the CV during HAC.

3.2 Thermal Evaluation

The bulk HEU in various forms to be shipped in the ES-3100 package has a maximum decay heat load of 5W. The thermal-related design features of the ES-3100 package, e.g., thermal properties, maximum temperature limits, maximum temperatures and pressures, and thermal stresses, are described in the SARP. Analyses and tests are used in the SARP to evaluate the packaging component temperatures under NCT and HAC. The DOE PCP staff also performed confirmatory analysis to evaluate the thermal design and performance of the package. The thermal properties of the materials for the fabrication of packaging components and air in the cavities are provided in the SARP, including the references. The listed thermal properties in the SARP are in good agreement with the values found in the published technical reports, standards, test reports, and handbooks.

3.2.1 Thermal Evaluation under NCT

The SARP evaluated the NCT thermal performance by analysis using MSC.Patran/THERMAL and ANSYS. Details of the models and the results are provided in Reference DAC-PKG-801699-A001. Two-D axisymmetric models were used in the SARP owing to symmetry considerations. The NCT transient analyses simulated a five-day period at an ambient temperature of 100^0F with cyclic insolation loading of 12 hours on/12 hours off. The 5W decay heat load was applied as a uniform heat flux to the inner surface of the CV.

Maximum component temperatures under NCT

Table 3.1 shows the calculated maximum temperatures for the packaging components in the SARP, which are well below the corresponding allowable temperature limits. As calculated in the SARP and confirmed by the DOE PCP staff, the maximum surface temperature of the ES-3100 package is 104^oF in still air at 100^oF and in the shade without solar insolation. Therefore, no accessible surface of the package would have a temperature exceeding 122°F in a nonexclusive use shipment per 10 CFR 71.43(g).

Table 3.1. Maximum Temperatures (^{o}F) under NCT

Components	SARP (Calculated)	Allowable
Drum body	230.64	1600
Drum liner	212.96	662
Drum lid	245.94	1600
Drum top plug	238.81	1600
Kaolite 1600	238.81	1600
Neutron poison (CAT 277-4)	197.32	302
Silicone rubber pads	204.85	450
Containment Vessel (body, lid and nut)	197.48	800
Ethylene propylene O-ring	197.36	302

Maximum normal operating pressure (MNOP)

Appendix 3.5.1 of the SARP contains detailed calculations for the MNOP. The calculated maximum temperature in the CV under NCT is 197.46 ^oF (91.92 ^oC). The bounding containment vessel arrangement (CVA) for the MNOP calculations is CVA7, for which the free volume is 320.32 in^3. The corresponding calculated MNOP is 31.218 psia. The contribution of water vapor to the MNOP is 0.568 psia due to the thermal expansion of the initially sealed water vapor at 25^{o}C. However, the Teflon FEP bottles are used as the convenience container for 9.17 kg of uranyl nitrate hexhydrate (UNH), which readily loses its water of hydration at an elevated temperature of 197.46 ^oF. The total amount of water in 9.17 kg UNH is (9170 g/502 $g.mole^{-1}$)*6 = 109.6 moles. At 197.46 ^oF, only 2.86×10^{-4} moles of water is required to reach the water vapor saturation in a free volume of 320.32 in^3. Since sufficient water is available from the UNH, the partial pressure of the water vapor at 197.46 ^oF is thus conservatively estimated to be the saturated vapor pressure of water of 10.9 psia (Frank P. Incropera, David P. Dewitt, Theodore L. Bergman, and Adrienne S. Lavine. *Fundamentals of Heat and Mass Transfer*, 6th edition, 2007, John Wiley & Sons, page 949.). The MNOP of the gas mixtures in the CV at 197.46 ^oF is calculated by the DOE PCP staff to be (20.305-0.568+ 10.9) = 30.64 psia, which is lower than the calculated pressure of 31.218 psia in the SARP and the design pressure of 116.2 psia for the CV.

Maximum thermal stresses

Since most of the packaging components are unrestrained, thermal stresses due to differences in thermal expansion are insignificant and they will have no effect on the ability of the packaging to maintain containment, shielding integrity, and criticality safety.

3.2.2 Thermal Evaluation under HAC

In the SARP, the five prototypic units subjected to the HAC structural tests were subsequently thermally tested in a furnace to evaluate the thermal design of the package. Analyses conducted with MSC.Patran/Thermal and ANSYS in the SARP were used to determine the effects of the internal decay heat, application of insolation during cool down, and thermal capacitance difference between the mock-up payloads and proposed shipping contents. Analytical adjustments of temperatures were made to obtain the peak temperatures in the CV (see Table 3.3 below), which are in turn used to calculate the maximum internal pressure.

Thermal analyses in the SARP were performed on a model representing an undeformed ES-3100 package. The DOE PCP staff performed confirmatory analysis using a model representing a deformed package following the dynamic slapdown drop and the sequential crush. The diameter of the drum body at the lid-end is reduced from 18.25 to 16.1 in. based on the DOE PCP staff's structural analyses. For simplification, a 2-D axisymmetric model was used in the thermal analysis, assuming a reduced radius of 8.05 in. for the drum. The reduction in the thickness of each component layer (drum shell, Kaolite insulation, inner liner, air gap, silicone rubber and neutron absorber) is linearly proportional to the radial distance of each component from the center of the package.

Maximum component temperatures under HAC

Table 3.2 shows the maximum temperatures of the packaging components calculated in the SARP and by DOE PCP staff, and the allowable temperature limits under HAC. The DOE PCP staff's calculated maximum temperatures of the components are significantly higher than the SARP values owing to the reduction in drum diameter; however, they are still well below the corresponding allowable temperature limits of components.

Table 3.2. Calculated Maximum Temperatures (°F) under HAC

Components	SARP	DOE PCP staff	Allowable
Drum body	1471.39	1472.96	1600
Drum liner	366.11	462.85	662
Drum lid	1453.52	1464.80	1600
Drum top plug	1340.05	1356.31	1600
Kaolite 1600	1471.39	1472.96	1600
Neutron poison (CAT 277-4)	254.07	277.28	302
Containment Vessel (body, lid and nut)	238.06	397.59	800
Ethylene propylene O-ring*	237.89	256.45	302

*Peak temperature is reached 4.14 hours after the 0.5-hour fire test.

Maximum internal pressure under HAC

The calculated maximum internal pressure of the CV under HAC at 284.64 °F is 112.458 psia, as shown in Appendix 3.5.2 of the SARP. The saturated vapor pressure of water is 53.33 psia at 284.64 °F (Frank P. Incropera, David P. Dewitt, Theodore L. Bergman, and Adrienne S. Lavine. *Fundamentals of Heat and Mass Transfer*, 6th edition, 2007, John Wiley & Sons, page 949.). Also at 284.64 °F, only 1.24×10^{-3} moles of water is required to reach the water vapor saturation in a free volume of 320.32 in^3, which is much less than the amount of available water from UNH (109.6 moles). For conservatism, the water vapor pressure is assumed to be saturated at 53.33 psia. The maximum internal pressure of the gas mixtures in the CV is calculated by the DOE PCP staff to be (44.362 -0.643 + 53.33) = 97.05 psia, which is lower than the design pressure (116.2 psia) of the CV. Each CV assembly is also hydrostatically pressure-tested at 164.7 psia (Section 3.7 of Equipment Specification JS-YMN3-801580-A001, Rev. F, ES-3100 Containment Vessel, Appendix 1.3.3 of the SARP).

The volume-averaged adjusted peak temperature of 284.64 °F, on the basis of which the maximum internal pressure of the CV is calculated, is conservative relative to the peak temperatures calculated by the DOE PCP staff, as shown in Table 3.3 below.

Table 3.3. Adjusted Peak Temperatures of the CV after HAC (°F)

Locations in the CV	Peak temperatures during test	Analytical temperature adjustments	Adjusted* peak temperatures	Peak temperatures calculated by DOE PCP staff
CV lid, top, center	261	71.99	332.99	256.61
CV flange at interface, inner	241	72.14	313.14	256.45
CV shell, mid-height, inner	199	84.06	283.06	235.18
CV bottom, center, inner	210	73.23	283.23	251.08

*The volume-averaged adjusted peak temperature is 284.64 °F.

3.3 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, DOE PCP finds that the thermal design and performance of the ES-3100 package presented in Chapter 3 of the SARP is acceptable, and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71 have been met.

4. CONTAINMENT

4.1 Discussion

The DOE PCP staff reviewed the containment design and performance of the ES-3100 package described in Chapter 4 of the SARP. The basis for the acceptance leakage rate of the CV is 1x 10^{-7} ref-cm^3/s (air) for fabrication, periodic, and maintenance. Preshipment leakage-rate testing, shall be performed to the acceptance criterion of either (1) a leakage rate of not more than the reference air leakage rate, L_R (ANSI NI4.5-1997, Subclause 6.2), or (2) no detected leakage when tested to a sensitivity of at least 10^{-3} ref-cm^3/s. (ANSI NI4.5-1997, Subclause 7.6.4). The DOE PCP staff finds the methodology to demonstrate compliance and the acceptance basis to be acceptable.

Uranyl nitrate crystals (UNX) may, through efflorescence, lose some of its water of hydration in the form of water vapor inside the CV, thus adding pressure to the MNOP during NCT and internal pressure during HAC. This contribution has been estimated in the DOE PCP staff's confirmatory analysis for NCT and HAC and is discussed in Section 3 of this SER. The DOE PCP staff concludes that even with this contribution from the efflorescence of UNX the upper bound to the MNOP (30.64 psia) under NCT is below the design pressure rating of the CV of 116.2 psia. Under HAC the maximum pressure including the effect of efflorescence is 97.05 psia, which is below 164.7 psia, the hydrostatic pressure tested for each CV assembly.

4.2 Description of the Containment Boundary

The containment boundary of the ES-3100 package consists of the vessel's body, lid assembly, and the inner O-ring, which is considered part of the boundary. The outer O-ring is provided to allow a post-assembly verification leak check. The CV O-rings 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 302°F (-40 to 150°C) [*Parker O-ring Handbook*, Figure 2-24]. The O-rings are certified to ASTM D 2000 as M3BA712A14B13F17. The continuous service temperature rating of the ethylene-propylene elastomer has been verified by testing. This testing was accomplished by subjecting an ES-3100 full-scale test unit (Test Unit-2), after being chilled to < -40°C to an NCT drop test and the HAC test sequence. The CV was leak tested after the drops and found to be leaktight.

It should be noted that the "*weld filler metal,*" shown as 4 of "*Part or Identifying Number*" of "*CONTAINMENT VESSEL SEALING LID*" in drawing M2E801580A015 (see Chapter 1, page 1-159 of the SARP), is considered as a component part of the sealing lid. It is used to seal around the round bar (plug) inserted into the hole for leak testing of the O-rings. This hole is outside the containment boundary, and is therefore not important to safety. However, the weld is important to operation, because its sealing capability is necessary for the preshipment leakage test. The weld is inspected visually and using dye penetrant (Paragraph 3.4, JS-YMN3-801580-A001, Appendix 1.3.3 of the SARP). The requirements of Subsection NB-5110 of the ASME BPVC apply. The DOE PCP staff finds the description and explanation of the "weld filler metal" part and its function to be acceptable.

4.3 Containment under NCT

Following fabrication, the CV undergoes hydrostatic pressure testing to 150 psig (1034 kPa). Each vessel is then leak tested with helium to 2 x 10^{-7} cm^3/s. The first sentence of paragraph 4 of Section 4.2 of the SARP states that "*The design, fabrication, maintenance and periodic leakage rate limit is 1 × 10^{-7} ref-cm^3/s air.*" The maintenance and periodic leakage tests are discussed in Chapter 8, Section 8.2.2 of the SARP, in a 13-step procedure. In step 5a, section 8.2.2 of the SARP, the accompanying note states: "*Y-12 Product*

Specification Procedure Y51-01-B2-R-140, He Leak Testing in B2 (Appendix 8.3.1), is provided as an example leak-test procedure".

These leak tests are for the full containment boundary, in which all containment boundary components, either in series or in parallel, are tested together as a total containment boundary system. In the leak test procedure described in Appendix 8.3.1, pages 8-24 to 8-28 of the SARP, the leak-test duration must be sufficient to ensure that a single containment boundary component, having a low leakage rate, cannot result in an erroneous conclusion that the full containment boundary is leaktight.

The SARP clarified that in accordance with ASME BPVC, Sec. V, Subsection A, Article 10, Appendix IX, "Helium Mass Spectrometer Test-Hood Technique," a system response time must be established to determine the helium leak testing duration. The fabricators of this package used the ASME BPVC approach and determined that the system response time for their setup of this CV was ≈15 s. To ensure that the leak detector steady-state (equilibrium) response is reached, the helium leak-test duration was established at 2 min (8 times the system response time). This test ensures the CV's integrity (walls, welds, inner O-ring seal) as delivered for use in accordance with paragraph 6.3.2 of ANSI NI4.5-1997. Y-12 Product Specification Procedure Y51-01-B2-R-140 *He Leak Testing in B2* (Appendix 8.3.1) has been revised to include a 2-min hold time. The DOE PCP staff finds the specification of the leak-test duration and its inclusion in the revised Y-12 Product Specification Procedure to be acceptable.

4.4 Containment under HAC

Containment vessel design/compliance-verification leakage tests and their criteria are discussed in Chapter 4. Further details are found in the report ORNL/NTRC-013. In ORNL/NTRC-013, it was reported that significant oscillations in the leak rates have been observed in two of the five test units. The amplitudes of these leak rate oscillations increase as the tests proceed throughout the duration of the tests. For one of the test unit (TU-2) the amplitude exceeds 2×10^{-7} cc/sec (He) early in the test and reaches 1.4×10^{-6} cc/sec (He) at the end of the test, which violated the ANSI-N14.5 (1997) leak tight criterion. Although the root cause of these oscillatory leak rates has not been determined, the seal around the machined port in the lid used for helium leak checking of the test units may be the culprit. A system response time of ≈15 seconds has been established by the fabricator for the ES-3100 leak test setup (see Section 4.3 above). Using a factor of eight, the fabricator sets their helium testing duration at two minutes. Since the helium leakage rate documented in ORNL/NTRC-013 was determined using a similar configuration and it remained below the ANSI N14.5-1997 leak tight value for approximately 10 minutes, it was determined that the test units were leaktight. Therefore the SARP concluded that the observed oscillatory behavior in the leakrates of some of the test units has no impact on the conclusion drawn about the containment vessel meeting the leaktight criterion of ANSI N14.5-1997. The SARP states that the above anomaly has not been observed during any fabrication, periodic, or maintenance leak testing of the ES-3100 CV. The DOE PCP staff finds the clarification of the observed oscillatory leak rates, and their lack of impact on the CV's ability to meet the leaktight criterion, to be reasonable and acceptable.

4.5 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, DOE PCP finds that the containment design and performance of the ES-3100 package presented in Chapter 4 of the SARP is acceptable, and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71 have been met.

5. SHIELDING

5.1 Discussion

The DOE PCP staff reviewed the shielding design and performance of the ES-3100 package described in Chapter 5 of the SARP. The dose rates calculated for 36 kg of HEU metal bounds those for the shipping configurations of all authorized contents described in Tables 1.3, 1.3a, and 1.3b of the SARP. The ES-3100 package is to be shipped under non-exclusive use. The Monte Carlo N-Particle Transport Code 5 (MCNP 5) was used for shielding evaluations in the SARP; the DOE PCP staff used MCNP 5, version 5.1.4, for the confirmatory evaluation.

5.2 Shielding Design

The ES-3100 package does not contain material specifically for shielding, although the stainless steel of the drum and the CV, the Kaolite material, and the 277-4 material provide some radiation attenuation. Restricting the amount of source material is the means for keeping the radiation dose rates below the regulatory limits.

5.3 Source Specification

For the contents of the ES-3100 package, the source of photons is the combination of decay of the uranium isotopes, fission, and decay of fission products. Direct decay of the ^{232}U isotope is the major contributor to photon source terms. The decay of ^{232}U leads to ^{208}Tl (thorium series), which produces high-energy gammas (≈2.6 MeV). The concentration of ^{232}U is limited to 40 ppb (parts per billion) of total uranium (see Table 5.3 and Table 1.1 in the SARP). Table 5.4 of the SARP shows the ORIGEN-S calculated photons per second per gram of HEU for all contents that are decayed for 10.5 years. For the contents of the ES-3100 package, the source of neutrons is the combination of alpha-neutron (α, n) reaction, spontaneous fission, and neutron-induced fission. Table 5.5 of the SARP shows the ORIGEN-S-calculated neutrons per second per gram of HEU for all contents that are decayed for 15 years.

Weapon-grade plutonium is used to represent all transuranic trace elements except for neptunium in the calculations of the neutron and photon source terms. The concentration of the weapon-grade plutonium used in the calculations is 800 ppm (parts per million) of total uranium. The concentration of neptunium used in the calculation is 2.5% of total uranium. The neutrons from neutron-induced fissions and the secondary photons are not included in the ORIGEN-S source terms but are included in the neutron transport calculations. The DOE PCP staff used ORIGEN-ARP 5.1.01 (Oak Ridge Isotope Generator - Automatic Rapid Processing 5.1.01) in the confirmatory evaluations.

5.4 Shielding Model

Multiple shipping configurations are listed in the SARP. In the shielding models, all materials interior to the CV except for the HEU content are omitted. The HEU content may have any geometry in the CV. The bounding cases for the shielding calculations are determined by comparing radiation dose rates calculated from models with different HEU-content geometries (see Figure 5.2 of the SARP).

In NCT, the surface of the drum is considered to be the package surface where dose rates are calculated. In HAC, the SARP assumes total loss of all package components outside the CV, with the CV and contents remaining intact. Thus, the surface of the CV is considered as the package surface in the HAC dose rate calculations. The DOE PCP staff has confirmed the dimension of the shielding model and verified the assumption used in the SARP for the dose-rate calculations.

5.5 Shielding Evaluation

The MCNP 5 was used for shielding evaluations in the SARP and by the DOE PCP staff for the confirmatory evaluation. The cross-section library used in the evaluations was based on ENDF-VI (Evaluated Nuclear Data Formats - VI). ANSI/ANS-6.1.1-1977 Neutron and Gamma-Ray Flux-to-Dose-Rate Factors was used to calculate personnel doses. Comparisons of the calculated dose rates are shown in Table 5.1 below.

Table 5.1. Maximum Dose Rates Calculated for the ES-3100 Package under NCT and HAC

	Maximum Dose Location	**SARP (mSv/h)**	**DOE PCP Staff (mSv/h)**	**10 CFR 71 Limits (mSv/h)**
NCT	Bottom surface of the package	0.864*	0.968	2
NCT	1 m from the side surface of the package	0.0572	0.0574	0.1
HAC	1 m from the side surface of the CV	0.123	0.123	10

* The dose rate was calculated at 1 cm from the package external surface using a spherical detector.

The maximum dose rates calculated in the SARP and by the DOE PCP staff are all significantly below the regulatory limits for non-exclusive use shipment, as shown in Table 5.1. The calculated transport index (TI) is 5.8. During actual operations with the ES-3100 package, the TI will be determined by measurement prior to shipment.

5.6 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, DOE PCP finds that the shielding design and performance presented in Chapter 5 of the SARP is acceptable, and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71 have been met.

6. CRITICALITY

6.1 Discussion

The DOE PCP staff reviewed the criticality safety design of the ES-3100 package described in Chapter 6 of the SARP. The DOE PCP staff also performed Monte Carlo analyses to independently confirm the criticality safety for a single package, as well as for an array of packages under the most reactive conditions during NCT and HAC. The ES-3100 package may be used to transport various contents in Tables 1.3, 1.3a, and 1.3b with Criticality Safety Indexes (CSIs) ranging from 0.0 to 3.2 under non-exclusive use shipment.

6.2 Package Description

The ES-3100 package design includes a stainless steel CV inside a 30-gallon outer drum (see Figure 1.1). The payload is placed in convenience cans or bottles or otherwise protected to prevent contamination of the interior surface of the CV. The package includes two features intended for criticality control: neutron absorber that surrounds the CV and can spacers placed between convenience cans, both filled with alumina borated cement. The drawings included in the SARP provide the dimensions of the relevant packaging components. Chapter 2 of the SARP provides material specifications for the packaging components.

Contents

The contents of the ES-3100 package include various forms of uranium metal, uranium alloys, uranium oxides, UNH and unirradiated TRIGA fuel elements (see Tables 1.3, 1.3a and 1.3b for loading limits). Some of these contents were previously approved by NRC; the new contents added are:

a) $UZrH_2$ (TRIGA fuel, SNAP reactor fuel),
b) research reactor fuel components or items such as clad U-Zr, U-Al, U_3O_8-Al, UO_2-Mg and UO_2, and
c) uranium compounds such as uranium tetrafluoride, uranyl fluoride, uranium carbide, uranium nitride and TRISO.

The ^{235}U enrichment limit varies with uranium form and payload mass and can range up to 100% for some payload configurations. Neutron absorber can spacers are required to meet criticality safety requirements for some payload configurations.

Each of the new content categories is evaluated with respect to criticality safety in Chapter 6 of the SARP. Descriptions of the ES-3100 package design features include identification of packaging materials, densities and compositions of packaging materials, and the fissile/fissionable material forms, masses and isotopic compositions of the payloads. The DOE PCP staff confirmed that the criticality-related information in the SARP is complete and representative of the actual materials specified for the ES-3100 package. The DOE PCP staff also confirmed that the models used in the criticality calculations are consistent with the drawings and the detailed package description given in the SARP.

6.3 Criticality Models

The KENO V.a code was used in the SARP for criticality analyses. The payload and the neutronically significant components of the ES-3100 package were included in the KENO V.a models. Separate models were developed for single-package, NCT, HAC and air transport analyses. Two single-package models, one consisting of a full ES-3100 package and the other just the CV, were used to calculate the neutron multiplication factors for the new contents under fully flooded and reflected conditions.

The NCT and HAC array calculations were based on detailed models of the ES-3100 package and payloads. Both triangular arrays and square arrays were modeled in the KENO V.a calculations. To simulate a triangular pitch in the array calculations, the outer radius of the package was reduced by 7% in the square lattice KENO V.a models and the compositions of the outer regions of the package were then adjusted to conserve mass.

Each of the air transport models had a central sphere consisting of ^{235}U, or ^{235}U homogenized with packaging components, surrounded by an external 20-cm water reflector shell. In some cases there was an intermediate shell consisting of ^{235}U, or packaging material between the central sphere and the outer water shell. No explicit analyses were performed for air transport of the proposed new contents. Only the research reactor fuels are proposed for air transport and those fuels can be packaged under limits for solid uranium metal of unspecified shapes, or under limits for unirradiated TRIGA fuel.

The array size was assumed infinite in the array calculations for all of the new contents. The SARP criticality analysis did not take credit for watertight containment either in the single-package analyses or in the array analyses. Water was modeled as the moderator and reflector for single-package and array calculations. The SARP determined the configurations of maximum reactivity with respect to moisture content within the CV and moisture contents of the neutron absorber and impact-absorbing insulation.

The Standard Composition Library and the 238GROUPNDF5 nuclear data library in the SCALE code package were used for all KENO V.a calculations in the SARP and in the confirmatory analyses. Section 6.8 of the SARP and Section 6.9.8.7 of Appendix 6.9.8 of the SARP summarize the determination of the minimum k_{safe} value. The lowest k_{safe} value determined from the validation for the proposed new contents is 0.924. Therefore, any configuration of ES-3100 packages with $k_{eff} + 2\sigma < k_{safe}$ is deemed subcritical. All calculations incorporated sufficient neutron histories to ensure statistical uncertainty (σ) less than 0.002 and adequate convergence. The DOE PCP staff concurs that the benchmark experiments and corresponding bias value are applicable and conservative as applied to the ES-3100 package.

6.4 Summary of SARP Criticality Analysis and DOE PCP staff's Confirmatory Evaluation

Evaluation of a single package under NCT and HAC

The analyses in Section 6.4 of the SARP show that maximum reactivity occurs for a fully flooded reflected package, so the single-package analysis is based on a fully flooded, reflected package. Chapter 6 of the SARP analyzed both a fully flooded, reflected CV and a fully flooded, reflected package.

Table 6.1 shows the maximum $k_{eff} + 2\sigma$ reactivity results listed in Chapter 6 of the SARP and the DOE PCP staff's confirmatory analyses for the new contents in the single-package configuration. All single-package configurations resulted in acceptable $k_{eff} + 2\sigma$ values that are below the k_{safe} limit of 0.924. Therefore, the ES-3100 single package with the proposed new contents and loading limits listed in Tables 1.3, 1.3a and 1.3b of the SARP is subcritical and satisfies the requirements of 10 CFR 71.55(b) related to a flooded single package.

Evaluation of undamaged package arrays (NCT)

The NCT undamaged package array model for each of the proposed new contents consisted of an infinite array of packages. The analyses in Chapter 6 of the SARP show that maximum reactivity occurs in an array of ES-3100 packages when the CV is flooded and the packaging is dry, referring to a configuration in which (a) the neutron poison of the body weldment liner inner cavity and the impact-absorbing insulation are dry, (b) recesses of the package external to the CV do not contain any residual moisture, and (c) the interstitial space between packages in the array does not contain any residual moisture. All of the NCT array configurations are based on a flooded CV and dry packaging to maximize the k_{eff} of the array.

Table 6.1 shows the maximum $k_{eff} + 2\sigma$ reactivity results listed in the SARP and the DOE PCP staff's confirmatory analyses for the new contents under NCT. All NCT arrays resulted in acceptable $k_{eff} + 2\sigma$ values that are below the k_{safe} limit of 0.924. Therefore, the ES-3100 package with the proposed new contents and loading limits listed in Tables 1.3, 1.3a and 1.3b of the SARP satisfies the requirements of 10 CFR 71.55(d) and 10 CFR 71.59(a)(1).

Evaluation of damaged package arrays (HAC)

Sections 6.4, 6.5 and 6.6 of the SARP show that the difference between the calculated $k_{eff} + 2\sigma$ values for NCT arrays and corresponding HAC arrays is not significant. For that reason HAC calculations were not performed in the SARP for some of the proposed new contents such as uranium tetrafluoride, uranyl fluoride, uranium carbide, and uranium nitride. The DOE PCP staff concurs that the HAC results should not differ significantly from the NCT results for these contents; as indicated in the similar pattern in Table 6.1 between NCT and HAC arrays for the U/Al fuel plates, reactor components-oxide and $UZrH_2$ fuel.

The HAC damaged package array model for the new contents consisted of an infinite array of packages, each with a flooded containment vessel and dry packaging to maximize the k_{eff} of the array. Table 6.1 shows the maximum $k_{eff} + 2\sigma$ reactivity results listed in the SARP and the DOE PCP staff's confirmatory analyses for the new contents under HAC. All HAC arrays resulted in acceptable $k_{eff} + 2\sigma$ values that are below the k_{safe} limit of 0.924. Therefore, the ES-3100 package with the proposed new contents and loading limits listed in Tables 1.3, 1.3a and 1.3b of the SARP satisfies the requirements of 10 CFR 71.55(e) and the HAC-related requirements of 10 CFR 71.59(a)(2).

6.5 Criticality Safety Index (CSI) for Nuclear Criticality Control

Based on the NCT/HAC infinite array analyses of the new contents, a minimum CSI of 0.0 was determined and reported in Chapter 1 of the SARP. The DOE PCP staff concurs that this CSI value is appropriate for the ES-3100 package with the new contents and loading limits listed in Tables 1.3, 1.3a and 1.3b of the SARP.

6.6 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, DOE PCP finds that the nuclear criticality safety design in Chapter 6 of the SARP is acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71 have been met.

Table 6.1. Summary of Criticality Analyses for the ES-3100 Package

Case	Content	SARP Case	Maximum $k_{eff} + 2\sigma$[a]	
			SARP	DOE PCP staff
Single Package – Flooded CV Reflected by Water				
S1	U/Al fuel plates	5cvcrualfpt11_3_1_15	0.38362	0.38215
S2	Uranium oxide fuel pins	5cvsroxt11_1_4_1_15	0.80106	0.79838
S3	$UZrH_2$ (SNAP) fuel	5ncsrs9355d2_1_15_15	0.67441	0.67635
S4	Uranium tetrafluoride	5cvcrpuf4t11_11_15	0.88952	0.88730
S5	Uranyl fluoride	5cvcrpuofhct11_14_15	0.90660	0.90160
S6	Uranium carbide (UC)	5cvcrpuct11_1_10	0.91490	0.91170
S7	Uranium nitride (UN)	5cvcrpunt11_10	0.90939	0.90895
NCT Array - Infinite				
N1	U/Al fuel plates	5nciaualfpt11_3-1_3	0.38199	0.38020
N2	Reactor components-oxide	5nciaoxt11_1_4_1_3	0.88453	0.88611
N3	$UZrH_2$ fuel	5nciasnap_93_1_15_3	0.61929	0.61910
N4	Uranium tetrafluoride	5nciapuf4t11_3_3	0.90281	0.90416
N5	Uranyl fluoride	5nciapuofhct11_3_3	0.89782	0.90050
N6	Uranium carbide (UC)	5nciadpuct11_1-2_15_3	0.89454	0.89453
N7	Uranium nitride (UN)	5nciadpunt11_2_15_3	0.89160	0.89229
HAC Array - Infinite				
H1	U/Al fuel plates	5hciaualfpt12_3_1_3	0.38024	0.38222
H2	Reactor components-oxide	5hciaoxt12_1_4_1_3	0.87745	0.87771
H3	$UZrH_2$ fuel	5hciasnap_93_1_15_3	0.62020	0.61826
H4[b]	Uranium tetrafluoride	----------	----------	----------
H5[b]	Uranyl fluoride	----------	----------	----------
H6[b]	Uranium carbide (UC)	----------	----------	----------
H7[b]	Uranium nitride (UN)	----------	----------	----------

a) Upper subcritical limit (USL) k_{safe} value is 0.924.
b) No HAC calculation was performed because HAC configuration is essentially identical to NCT configuration.

7. PACKAGE OPERATIONS

7.1 Discussion

The DOE PCP staff reviewed the requirements for general operating procedures in loading, unloading, shipping, and receiving ES-3100 pacakges; preparation of empty ES-3100 packages for transport; and other operations as described in Chapter 7 of the SARP. These requirements for general operating procedures shall be implemented to ensure the package is used in accordance with the CoC for the ES-3100 package. In addition, packaging-specific requirements are reviewed to ensure that the package operations are in accordance with the CoC and Chapter 7 of the SARP. Each user of an ES-3100 packaging shall register with the DOE Assistant Secretary for Environmental Management (EM) prior to first use of the packaging. Quality Assurance (QA) shall participate in package operations.

7.2 Package Loading

Section 7.1 of the SARP describes the package loading requirements for the ES-3100 package. Before each shipment, the user must have site-specific procedures that comply with the requirements of the CoC, Chapter 7 of the SARP, and the requirements of 10 CFR71.5 and 71.87. Before any packaging operations are begun, the payloads to be shipped must be fully characterized with respect to the chemical and physical forms, the specific requirements of Section 1.2.2 of the SARP, and the 7 steps for content preparation described in Section 7.1.1.1of the SARP.

Section 7.1.1.2 of the SARP describes the 25 steps related to packaging preparation. The user shall develop detailed operating procedures to implement these steps as a minimum.

Section 7.1.2 of the SARP describes the loading of contents into the CV. The user shall develop operating procedures that contain as a minimum the following steps; operating personnel shall ensure that the CV has been emptied of radioactive material, the O-rings and grooves on the CV are protected during loading, and the HEU content and packing materials are prepared and loaded in accordance with Section 1.2.2 of the SARP.

Section 7.1.2.1 of the SARP describes the assembly and leak testing of the CV. Steps 1-6 address the assembly process. Step 7 addresses the leak-testing process. The pre-shipment leak test meets the requirements of ANSI N14.5-1997. The leakage test shall demonstrate that there is no leakage between the O-rings at a sensitivity of $1x10^{-3}$ ref-cc/s per second of air.

Section 7.1.2.2 of the SARP describes the loading of the CV into the drum and closure. This section describes 21 steps that must be done as a minimum. The CV is loaded into the drum, the drum lid is installed, and the drum-lid hex nuts are torqued to 30 ± 5 ft-lbs. This torqueing shall be done by hand and an impact wrench shall not be used. The TIDs are attached, the radiation levels measured, and the appropriate labeling completed.

Section 7.1.3 of the SARP describes the preparation for transport, addressing package transfer or handling (Section 7.1.3.1), decontamination (Section 7.1.3.2), requirements prior to shipment (Section7.1.3.3), and securing to the approved conveyance (Section 7.1.3.4).

7.3 Package Unloading

Section 7.2.1 of the SARP describes the 9 steps involved in the receipt of the package from the carrier and 5 types of incident notifications. The user shall develop detailed operating procedures to implement these steps as a minimum.

Section 7.2.2 of the SARP describes the 17 steps involved in the removal of the contents from the package, and disassembly of the CV. The user shall develop detailed operating procedures to implement these steps as a minimum.

7.4 Preparation of Empty Package for Transport

Section 7.3 of the SARP describes the 31 steps involved in the preparation of an empty package for transport. The user shall develop detailed operating procedures to implement these steps as a minimum. The package will be prepared and shipped in accordance with 49 CFR 173.

7.5 Other Operations

Section 7.4 of the SARP addresses other operations. There are no special controls unique to this package.

7.6 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, DOE PCP finds that the operating procedure requirements presented in Chapter 7 of the SARP are acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71 have been met.

8. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Discussion

The DOE PCP staff reviewed the acceptance tests and maintenance program described in Chapter 8 of the SARP. The packaging acceptance testing and maintenance operations are consistent with maintaining occupational radiation exposures as low as reasonably achievable (ALARA). The fabrication requirements for the ES-3100 package components are listed on the design drawings (Appendix 1.3.7) and the following appendices to the SARP: (a) the CV in Appendix 1.3.3, (b) the drum assembly in Appendix 1.3.2, (c) the casting of the Kaolite 1600 in Appendix 1.3.4, and (d) the 277-4 neutron absorber in Appendix 1.3.5. The CV is built to the ASME BPVC Section III, Division 1.

8.2 Acceptance Tests

Section 8.1 of the SARP describes acceptance tests. Before first use of the packaging, the owner shall determine that the packaging has been fabricated in accordance with the approved design including the preliminary determinations in 10 CFR 71.85, the quality requirements of 49 CFR 173.474 and 10 CFR 71 Subpart H, and the conditions of the CoC. The required inspections, tests, and measurements shall be in conformance with 10 CFR 71.85(a) and Tables 8.1—8.3 of the SARP. Table 8.1 describes the acceptance tests for the drum assembly, Table 8.2 the CV assembly, and Table 8.3 the packing materials. All welds and weld-repaired surfaces shall be visually examined by a qualified weld examiner for indications of inclusions, cracks or porosity using approved written weld-examination procedures.

Section 8.1.3 of the SARP describes structural and pressure tests. The CV assembly is hydrostatically tested at 150 ± 5 psig. The CV MNOP is 15.94 psig, and the CV design pressure is 101.5 psig. 10 CFR 71.85(b) requires hydrostatic testing at 1.5 times operating pressure, and the ASME BPVC requires hydrostatic testing at 1.25 times design pressure. Therefore, doing the hydrostatic test of the CV at 150 psig satisfies both the regulatory and ASME code testing requirements. Two sample drums in each lot fabricated are to be pressure tested to verify the integrity of the welded seams. The drums are pressurized with air to 10 psig, the air supply is closed, the initial pressure is recorded, and all joints are covered with a bubble-supporting film. After five minutes the pressure is recorded, and the seams checked for evidence of bubble leakage. Any evidence of leakage, either pressure loss or bubbles, is cause for rejection.

Section 8.1.4 of the SARP describes leakage tests. Following hydrostatic testing, a fabrication leakage test of the containment boundary is performed with the inner O-ring installed in accordance with ANSI N14.5-1997, Sub-clause 7.3. The CV leakage rate testing shall be performed using certified equipment and written procedures. An integrated air leakage rate exceeding $1x10^{-7}$ ref-cm^3/s of air is cause for rejection. In addition, a leakage rate test is performed at initial fabrication on the fully assembled CV with both O-rings installed. This test demonstrates the functionality of the CV leak-test port and the sealing capability of the outer O-ring. The acceptance criterion is that the CV shall not have an air leakage rate greater than$1x10^{-4}$ ref-cm^3/s. Doing this test at fabrication, to this more stringent criterion of an air leakage rate of $1x10^{-4}$ ref-cm^3/s, provides increased assurance that the CV will pass the actual pre-shipment leakage test when tested to an air leakage rate of $1x10^{-3}$ ref-cm^3/s.

Section 8.1.5 of the SARP describes component and material tests. The CV O-rings are visually inspected for defects, and each O-ring is packaged separately and adequately identified to provide traceability and have an identified expiration date. The identifications shall be adequate to trace the O-rings to their raw-material master batch. The mechanical properties of hardness and elongation shall be determined for each lot of the O-ring material. The acceptance criterion for hardness is a SHORE A of 70 ± 5 durometer; for elongation, the acceptance criterion is 100% minimum.

8.3 Maintenance Program

Section 8.2 of the SARP describes the ES-3100 package maintenance program. This maintenance program ensures that the packaging continues to meet the design requirements and the conditions of approval in the CoC. The periodic maintenance shall be performed on a 12-month basis.

Section 8.2.2 of the SARP describes leakage tests. This section contains a series of steps to measure performance in the ANSI 14.5-1997 maintenance leakage-rate test (Sub-clause 7.4), or the periodic leakage-rate test (Sub-clause 7.5), to demonstrate that the containment boundary is "leaktight" when using a leak-check flange assembly as shown on Drawing T2E801827A008. An integrated air leakage rate exceeding 1×10^{-7} ref-cm^3/s of air is cause for rejection.

Section 8.2.3 of the SARP describes component and material tests. The inner and outer O-rings are replaced during periodic maintenance of the packaging. Certified O-rings are used for replacement, and visually inspected for defects prior to use. Replacement O-rings are stored in sealed containers and have an expiration date marked on the package. The Kaolite insulation and the Cat 277-4 neutron absorber material are encased in stainless steel. No damage or deterioration is expected; however, the drum parts are visually inspected. Additionally, the drum assembly and top plug are weighed prior to first use and periodic maintenance to evaluate any density changes. Drum-assembly weight changes of greater than 9 lb or top-plug weight changes of greater than 3 lb are cause for rejection and evaluation for rework.

Section 8.2.5 of the SARP describes miscellaneous tests. The CV is removed from the drum assembly for periodic maintenance inspections and the interior and exterior CV surfaces shall be examined for signs of moisture, corrosion or physical damage. Any CV exhibiting these conditions shall be tagged and separated until the cause is determined and corrected. All threaded parts are examined and evaluated. The threads are cleaned and any small nicks or burrs are removed. If installed, the O-rings are removed, and the CV flange grooves and the CV sealing lid are cleaned and inspected.

The ES-3100 packaging is stored indoors and corrosion is not expected. However, during periodic maintenance inspections, all accessible surfaces shall be visually inspected for corrosion, moisture or damage. Any drum exhibiting these conditions shall be tagged and separated until the cause is determined and corrected. Worn or faded packaging markings are touched up as necessary, and the data plate and trefoil plate are examined for legibility and secure attachment. The drum-lid fasteners, both studs and nuts, are inspected for damage. The threads are cleaned and any small nicks or burrs are removed. The drum closure nuts may be replaced with certified nuts as part of routine maintenance. The silicone rubber pads are inspected during periodic maintenance to verify that there are no signs of moisture, and that there are no gouges, cuts, tears or non-design voids in the pads.

8.4 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, DOE PCP finds that the acceptance tests and maintenance program requirements presented in Chapter 8 of the SARP are acceptable and will provide reasonable assurance that the regulatory requirements of 49 CFR 173, Subpart I and 10 CFR Part 71 have been met.

9. QUALITY ASSURANCE REQUIREMENT

9.1 Discussion

The DOE PCP staff reviewed the requirements for a QA Program described in Chapter 9 of the SARP. These QA requirements provide sufficient control over all items and quality-affecting activities that are important to safety as applied to the design, fabrication, assembly, inspection, testing, operation, maintenance, modification, and repair of the ES-3100 packaging. The QA requirements are based on a graded approach, as described in 10 CFR 71.105.

9.2 QA Program

The QA Chapter of the SARP, along with the *Y-12 National Security Complex Packaging Engineering Quality Assurance Program Plan* (QAP-Y-91-273860-1) provides QA requirements and implementing procedures that demonstrate compliance with each of the 18 QA requirements in 10 CFR 71, Subpart H. Appendix B of QAP-Y-91-273860-1 provides a crosswalk matrix that documents the conformance of the Y-12 packaging QA program to the 18 QA requirements of 10 CFR 71, Subpart H. The crosswalk matrix also provides requirements for software QA and integrated safety management.

Graded Approach

The graded approach in the QA Chapter of the SARP includes an important-to-safety Q-list for each significant item and activity; each item is graded on the basis of its design function relative to the safety and performance requirements for the complete packaging. Table 9.2 of the SARP contains the quality categories for each component, based on 10 CFR 71.105 and NRC Regulatory Guide 7.10, Appendix A. The Q-list establishes three QA categories with associated definitions for each. The QA level of each important-to-safety item is based on specific criteria. The QA requirements ensure that the packaging components are designed, fabricated, tested, and operated in accordance with the drawings identified in the SARP. In addition, the QA Chapter requires the user to invoke the same level of QA requirements for the use, maintenance, and repair of the packaging components, as is required for the procurement, fabrication, and acceptance testing of the original packaging components.

Section 9.3.2 of the SARP contains definitions for each QA category for important-to-safety items and activities and non-safety-related items:

1. Category A – Components are those whose failure or malfunction will directly result in an unacceptable condition of containment, shielding, or nuclear criticality.
2. Category B – Components are those whose failure or malfunction will indirectly result in an unacceptable condition of containment, shielding, or nuclear criticality (if the primary event occurred in conjunction with a secondary event, another failure, or an environmental event).
3. Category C – Components are those whose failure or malfunction does not result in an unacceptable condition of containment, shielding, or nuclear criticality regardless of other failures in this category.

Level of QA Effort

After determining the applicable QA category, the appropriate level of QA effort for design, procurement, fabrication, testing, operations, maintenance, modification, and repair activities is determined from the eighteen (18) QA elements identified in 10 CFR Part 71, Subpart H. Table 9.1 of the SARP includes specific QA requirements (Level of QA Effort) from Subpart H of 10 CFR 71 relative to packaging activities based on specific category. The eighteen (18) requirements identified in the SARP are as follows: organization; quality assurance program; design control; procurement document control; instructions, procedures, and drawings; document control; control of purchased material, equipment, and services; identification and control of material, parts, and components; control of special processes; inspection control; test control; control of measuring and test equipment; handling, shipping, and storage control; inspection, test, and operating status; control of nonconforming materials, parts, or components; corrective action; QA records; and audits. Each of the eighteen (18) requirements has assigned QA requirements on the basis of Quality Category A, B, or C.

Independent Verification

The QA Chapter of the SARP includes independent verification of fabrication and operational activities considered to be critical in satisfying the regulatory requirements as identified in 10 CFR 71, Subpart H. Section 9.3.10 of the SARP requires independent verification of critical activities, including inspection criteria for acceptance of the fabricated ES-3100 packaging components, assembly operations, and package loading. Specific inspection criteria are contained in drawings and Chapters 7 and 8.

Records

Table 9.3 of the SARP specifies which documents are considered to be lifetime records, e.g., the SARP, design drawings, audit reports, and nonconformance reports (and resolutions). The record retention program specifies that the design authority must retain records for three (3) years beyond the date when the package was last used in a particular activity that is documented by the prescribed records.

9.3 Conclusion

On the basis of the statements and representations in the SARP and the DOE PCP staff's confirmatory evaluation, DOE PCP finds that the Quality Assurance Program and requirements in Chapter 9 of the SARP are acceptable and will provide reasonable assurance that the regulatory requirements of 10 CFR Part 71 have been met.