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Shielding Analysis of the Model 9602 Type B Packaging Design for Disused Radiological Sources

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ABSTRACT

A new compact Type B transportation packaging, designated as Model 9602, is being designed by Argonne National Laboratory researchers for storage, transport and disposal of disused radiological sources. This paper describes the shielding analysis of the Model 9602 packaging design performed as part of the preparation of a Safety Analysis Report for Packaging, which is to be submitted to the regulatory authority as part of an application for a Certificate of Compliance for the packaging design. The shielding performance of Model 9602 under normal conditions of transport and hypothetical accidents, described in Title 10 of the U.S. Code of Federal Regulations (10 CFR 71), *Packaging and Transportation of Radioactive Material*, was evaluated by using the Oak Ridge Isotope Generation code for the source terms and the Monte Carlo N-Particle Transport Code (MCNP), version 6.2, for calculating the gamma dose rates based on the ANSI/ANS-6.1.1-1977 dose conversion factors. The 10 CFR 71 regulatory limits on dose rates at the external package surfaces (top/side/bottom) for the Model 9602 packaging design are 2 mSv/h (0.2 rem/h) and 10 mSv/h (1 rem/h), respectively, for non-exclusive and exclusive-use shipment. The results obtained from the MCNP shielding analyses showed that the calculated radiation dose rates satisfy the 10 CFR 71 requirements for transporting disused CsCl sealed capsules in Model 9602 off-site by exclusive-use shipment for direct disposal. For interim storage on-site or off-site after transport, the radiation dose rates at the external package surface of Model 9602 will decrease over time because of the continuing isotope decay. Moreover, the all-stainless-steel packaging design of Model 9602 provides excellent performance with respect to general corrosion resistance during long-term storage, thus enabling subsequent transportation, without repackaging of the disused radiological sources, to a geological repository or deep borehole for final disposal.

INTRODUCTION

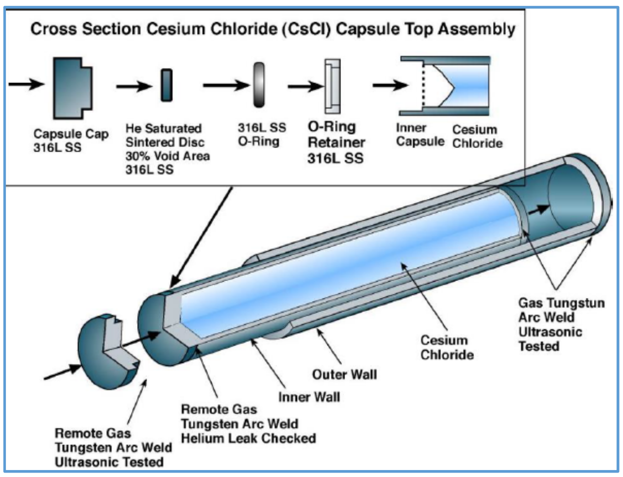
Radiological sources are used in many countries because of their beneficial uses in medical and industrial applications. Some of the radioisotopes used in the sources have relatively short half-lives—for example, 73.8 days for Ir-192 and 5.27 years for Co-60—while others have much longer half-lives: 30.17 and 28.79 years, respectively, for Cs-137 and Sr-90. These radioisotopes are all high-energy β-γ emitters, and the lack of a disposition pathway for the disused radiological sources poses a significant risk in terms of inadvertent or deliberate misuse of the material and other problems [1, 2].

The U.S. Department of Energy (DOE) has planned since the mid-1980s to dispose of all high-level waste (HLW) and spent nuclear fuel (SNF), regardless of commercial, defense, or research origin, in a common mined geologic repository. A separate mined repository was proposed in 2015 for DOE-managed SNF and HLW, as well as an option for deep borehole disposal of "small" waste forms, such as the disused CsCl capsules currently stored in the pool cells at Hanford's Waste Encapsulation Storage Facility. However, space restrictions and other limits (e.g., heat load, radioactivity, floor loading) imposed on the adjacent hot cells would permit only a limited number of CsCl sealed capsules to be brought from the pool cells into the hot cells for packaging and transfer to an on-site facility for dry storage, followed by transportation for direct disposal at a mined geological repository or a deep borehole without repackaging of the capsules.

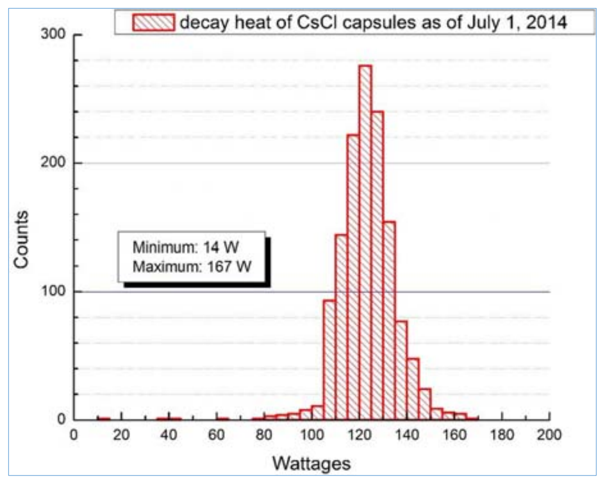
The compact Type B packaging design should also be readily applicable to other disused commercial radiological sources found in the United States and other countries. The conceptual design of this work originated from a project supported by the DOE Office of Environmental Management, which was described in "Groundwork for Universal Canister System Development." [3]

SOURCE SPECIFICATION

The CsCl sealed capsule, shown in Fig. 1(a), is constructed of 316L stainless steel, and the outer diameter and full length of the capsules are approximately 6.76 cm (2.66 in) and 52.78 cm (20.78 in), respectively. The radioactive isotopes of Cs that are of concern are Cs-135 and Cs-137. Cs-135 transforms into Ba-135 via beta emission (0.205 MeV) with a half-life of 2.3 million years, whereas Cs-137 has a half-life of 30.17 years and in 95% of the emissions transforms into Ba-137m, which has a metastable half-life of 2.55 min. In 90.1% of the emissions, the excited nuclei of Ba-137m emit 0.662-MeV gamma rays, which are the dominant radiation sources for shielding analysis of Model 9602 for packaging, transfer, storage, transportation and disposal.



(a)



(b)

Figure 1. Schematic of a CsCl capsule (a), and its population distribution according to decay heat (b).

Figure 1(b) shows the population distribution of disused CsCl sealed capsules according to decay heat, or source term, as of July 1, 2014. Over 260 CsCl capsules had decay heat ~120 W, whereas a significant number of them had decay heat above or below 120 W. In Fig. 1(b), the entire population of CsCl capsules would shift to the left over time, because their decay heat decreases with time in accordance with the half-life of 30.17 years for Cs-137.

For the radiation shielding analysis presented in this paper, the reference CsCl composition used to calculate the source term was derived from data obtained by the characterization of eight CsCl capsules by chemical dissolution and calorimetry [4]. The mass of Cs-137 was adjusted to match the decay heat value of capsule C-1512, i.e., 245 W on 4/12/1983. The total mass of CsCl in the reference capsule was 2077.91 g.

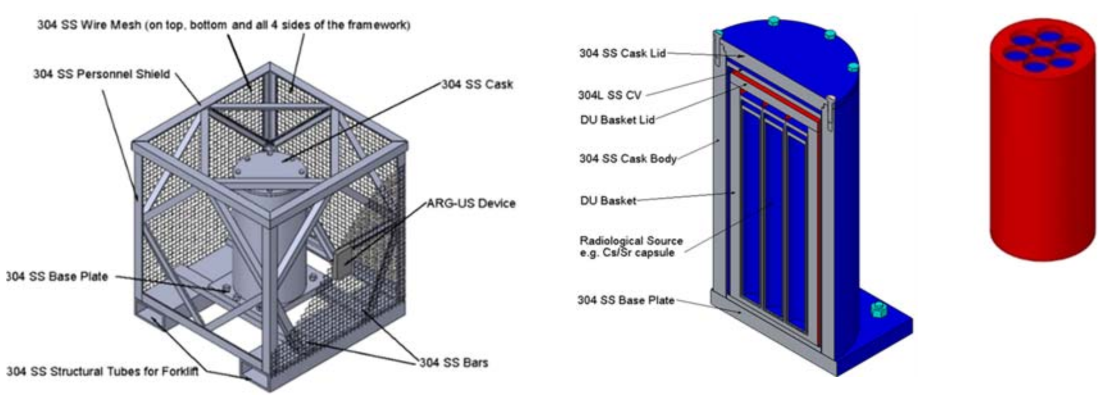
The gamma source term was calculated in 18 energy groups as shown in Table 1, using the Oak Ridge Isotope Generation (ORIGEN) module of the SCALE 6.2.3 software package [5]. The calculated decay heat of one CsCl capsule after 33 years of decay was 115 W, a value corresponding to a large number of disused CsCl sealed capsules as shown in Fig. 1(b).

Table 1. ORIGEN-calculated 18-group Photon Spectrum for One CsCl capsule after 33 Years of Decay

Energy Group	Mean Energy (MeV)	1 Capsule Source Strength (ph/s)
1	1.00E-02	8.84E+13
2	2.50E-02	1.58E+13
3	3.50E-02	6.62E+13
4	5.50E-02	1.37E+13
5	8.50E-02	6.44E+12
6	1.25E-01	4.20E+12
7	2.25E-01	2.68E+12
8	3.75E-01	4.00E+11
9	5.75E-01	8.70E+14
10	8.50E-01	9.07E+09
11	1.25E+00	1.13E+08
12	1.75E+00	0.00E+00
13	2.25E+00	0.00E+00
14	2.75E+00	0.00E+00
15	3.50E+00	0.00E+00
16	5.00E+00	0.00E+00
17	7.00E+00	0.00E+00
18	9.00E+00	0.00E+00
	Total	1.07E+15

MODEL 9602 PACKAGING DESIGN

Figure 2 shows the compact Type B packaging design for Model 9602, which consists of a personnel shield with wire mesh, a bolted-closure cask containing a containment vessel (CV), and a depleted-uranium (DU) basket holding up to seven (7) CsCl capsules, with a total heat load limit of up to



(a) (b) (c)

Figure 2. (a) Schematic of the compact Type B packaging design for Model 9602; (b) cask and CV; and (c) DU basket accommodating up to seven CsCl capsules.

1,000 W. [6] The overall dimensions of the packaging are ~0.91 m (3 ft) L × 0.91 m (3 ft) W × 1.07 m (3.5 ft) H. All packaging components, except for the DU basket, are made of 304 SS or 304L SS (CV only). Each component of the compact Type B packaging design serves one or more important-to-safety functions. For example, the 304 SS personnel shield provides protection for radiation from the cask during normal operation, whereas the space in between the personnel shield and the cask enables heat dissipation by natural convection. The 304 SS framework of the personnel shield also serves as an *excellent impact limiter in a hypothetical accident (HAC), e.g., a 9-m (30-ft) drop followed by impact* on a puncture bar, as prescribed in 10 CFR 71.73. Structural analyses conducted to date have shown that the structural performance of the packaging design meets all the requirements specified in 10 CFR 71. [7]

SHIELDING PERFORMANCE EVALUATION

The shielding performance of Model 9602 was evaluated by using the Monte Carlo N-Particle Transport Code (MCNP) version 6.2, [8] with the ENDF/b-VII.1 cross section libraries. The 304 SS personnel shield and the 304 SS structural tubes, shown in Fig. 2(a), were not included in the MCNP 6.2 model. The dimensions of the basket insert, DU liner, CV, and cask of Model 9602 are listed in Table 2. The thicknesses of the top and bottom of the DU basket liner used in the shielding calculations are 3.18 cm (1.25 in) and 2.54 cm (1.0 in), respectively.

Table 2. Dimensions of the Basket Insert, Basket Liner, CV and Cask of Model 9602 Used in the MCNP Model for Shielding Analysis

Component	Material	Inner Radius (cm/in)	Outer Radius (cm/in)	Height (cm/in)
Basket Insert	Stainless Steel	n/a	8.33/3.28	53.34/21.0
Basket Liner	Depleted Uranium	8.33/3.28	11.94/4.70	59.10/23.27
Containment Vessel	Stainless Steel	12.19/4.80	12.70/5.0	62.23/24.5
Cask	Stainless Steel	12.89/5.07	16.19/6.37	64.14/25.25

The dose rates were calculated using MCNP's F4 tallies with the ANSI/ANS-6.1.1-1977 flux-to-dose conversion factors. Only four CsCl capsules were included in the calculations, with their source terms derived for the reference capsule content after 33 years of decay. Table 3 shows the MCNP-calculated dose rates and 10 CFR 71.47 regulatory limits for non-exclusive and exclusive-use shipment under normal conditions of transport (NCT) and HAC. For non-exclusive use shipment under NCT, the calculated dose rates on all sides of the package (top, side, and bottom) exceed the regulatory limit of 2 mSv/h, but not 10 mSv/h for exclusive-use shipment under NCT.

Table 3. MCNP-calculated Dose Rates and Regulatory Limits* under NCT and HAC

Top (mSv/h)	Side (mSv/h)	Bottom (mSv/h)	Non-exclusive (mSv/h)/(rem/h)	Exclusive (mSv/h)/(rem/h)	Description
7.82 ± 0.11	8.90 ± 0.03	6.85 ± 0.10	2/0.2	10/1	Package Surface (NCT)
1.24 ± 0.05	1.46 ± 0.01	0.60 ± 0.03	2/0.2	10/1	Package Surface + 1 m (NCT)
2.31 ± 0.06	2.53 ± 0.01	0.60 ± 0.03	n/a	10/1	Cask Surface + 1 m (HAC)

*Column 4 and Column 5 show the regulatory limits for non-exclusive and exclusive-use shipment, respectively.

According to 10 CFR 71.47, an exclusive-use shipment is defined as follows: (i) The shipment is made in a closed transport vehicle; (ii) The package is secured within the vehicle so that its position remains fixed during transportation; and (iii) There are no loading or unloading operations between the beginning and end of the transportation. There are also regulatory dose rate limits for exclusive-use shipment: (1) 2 mSv/h (200 mrem/h) at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure, if used, and on the lower external surface of the vehicle; (2) 0.1 mSv/h (10 mrem/h) at any point 2 meters (80 in) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle); and (3) 0.02 mSv/h (2 mrem/h) in any normally occupied space, except that this provision does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry devices in conformance with 10 CFR 20.1502. We have not performed MCNP calculations of dose rates at different vehicle locations, because the configuration for the exclusive-use shipment is not sufficiently defined.

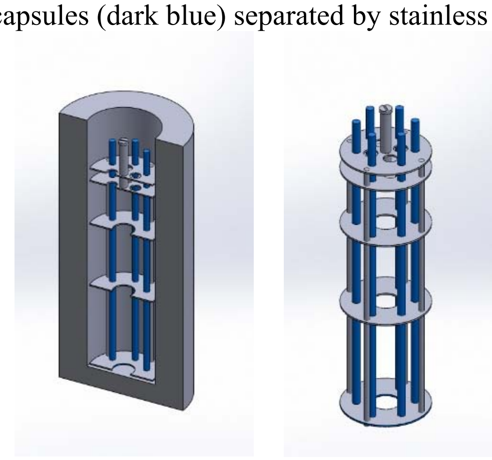
For exclusive-use shipment under HAC, the package surface is that of the cask, assuming damage and complete removal of the personnel shield after the sequence of drop, puncture and fire HACs specified in 10 CFR 71.73. The calculated dose rates on all sides of the cask surface + 1 m (top, side, and bottom) are lower than the regulatory limit of 10 mSv/h.

It should be noted that the calculated radiation levels for Model 9602 are based on a reference capsule with 115 W of decay heat. The authors of reference [9] reported that as of January 2016, the average decay heat of CsCl capsules was 118.6 W, with a standard deviation of 11.6 W; the minimum decay heat of CsCl capsules was 13 W and the maximum was 161 W. Therefore, many options are available for loading the disused CsCl sealed capsules into the DU basket to ensure that the radiation levels will stay well below the regulatory limits for exclusive-use shipment of Model 9602.

SUMMARY AND DISCUSSION

The shielding performance of the Model 9602 compact Type B packaging design for disused CsCl sealed sources has been evaluated by using the MCNP code under the NCT and HAC prescribed in 10 CFR 71. The results indicate that the calculated dose rates for the CsCl capsules meet the 10 CFR 71.47 regulatory limits for both NCT and HAC, for exclusive-use shipment. While the calculations were based on four disused CsCl radiological sources of the reference composition, the number of disused CsCl capsules can be increased by mixing capsules with lower decay heat loads from the population shown in Fig. 1(b); the heat loads themselves decrease with time in accordance with the half-life of 30.17 years for Cs-137.

It should be noted that the design and the material of the basket can be altered to accommodate other disused radiological sources, and, therefore, our findings are not limited to the shipment of disused CsCl sealed capsules in the Model 9602 packaging design. Figure 3 shows an alternate DU basket design with a central cavity (a), for disused Co-60 capsules (dark blue) separated by stainless steel disc spacers (b).



(a) (b)

Figure 3 An alternate DU basket design with a central cavity (a), for disused Co-60 capsules (dark blue) separated by stainless steel disc spacers (b).

Because the basket is an internal structure inside the CV in the Model 9602 packaging design, it is straightforward to include alternative designs for other disused radiological sources, as long as the total weight of the contents is bounded by that used in the structural performance evaluation [7] and the total decay heat load is bounded by that used in the thermal performance evaluation [6]. The Model 9602 packaging design is therefore suitable for end-of-life management of disused radiological sources for transportation, storage and disposal. The packaging design also includes options to use ARG-US remote monitoring systems to enhance safety and security during storage, transportation and disposal. [10]

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