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Extending the Utility of a Radioactive Material Package

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ABSTRACT

Once a package has been certified for the transportation of DOT Hazard Class 7 – Radioactive Material in compliance with the requirements of 10 CFR 71, it is often most economical to extend its utility through the addition of content specific configuration control features or the addition of shielding materials. The SRNL Model 9977 Package's authorization was expanded from its original single to twenty contents in this manner, and most recently, the 9977 was evaluated for a high-gamma source content. This paper discusses the need for and the proposed shielding modifications to the package for extending the utility of the package for this purpose.

Background

Savannah River National Laboratory (SRNL) developed the Model 9977 Radioactive Material (RAM) Package for the Department of Energy (DOE) National Nuclear Security Administration (NNSA) as a replacement for the Specification 6M package for which the authorization was withdrawn by the U.S. Department of Transportation (DOT) in 2006. The 9977 RAM shipping package Safety Analysis Report for Package (SARP) (S-SARP-G-00001, Revision 2) [Ref. 1] was authorized by the DOE Environmental Management (EM) Packaging Certification Program (PCP) Certificate of Compliance (CoC) Revision 0 [Ref. 2] on October 9, 2007. The SARP and CoC authorized the shipment of a single Content, 100 grams of plutonium and/or uranium "Heat Source" material in two container configurations. The initial SARP established the design of the package, its compliance with the requirements of Title 10 of the Code of Federal Regulations (CFR) Part 71 [Ref. 3], its design pressures, temperatures, shielding and criticality features, its minimum essential operating procedures, its acceptance and maintenance requirements, and its Quality Assurance program. The SARP follows the content and format requirements specified in Regulatory Guide 7.9 [Ref. 4]. The 9977 was authorized with a maximum content weight of 100 pounds and radioactive decay heat rate (DHR) of 19 watts and has a Containment Vessel maximum design pressure of 800 psi at 300°F.

A RAM Package must perform, in the broadest terms, the following three functions:

Containment

> 10 CFR 71.51(a)(1), requires that the Package Containment Vessel be designed, constructed, and prepared for shipment so that, under Normal Conditions of Transport (NCT) test conditions, "there will be no loss or dispersal of radioactive contents, as demonstrated to a sensitivity of 10^{-6} A_2 per hour." Likewise, 10 CFR 71.51(a)(2), requires that the containment vessel be designed, constructed and prepared for shipment so that during the Hypothetical Accident Conditions (HAC), no escape of radioactive materials can occur in excess of A_2 in one week.

Shielding

> The Package must be demonstrated to be in compliance with the performance requirements specified in 10 CFR 71.47, 10 CFR 71.51, 49 CFR 173.403, and 49 CFR 173.441. These regulations specify that the dose rate limits for an undamaged package are 200 mrem/h at the accessible surface of the package, 10 mrem/h at 1 meter from the accessible surface of the package, and 1000 mrem/h at 1 meter from the surface of a damaged package after a hypothetical accident.

Criticality

> The Package must be demonstrated to remain subcritical in compliance with the performance requirements of 10 CFR 71.55 and 71.59 for criticality safety.

The utility of the 9977 was expanded since it was initially certified by the inclusion of additional RAM Contents and packing configurations that support numerous DOE and NNSA sites, programs, and functions. A total of fifteen (15) unique content envelopes, four (4) generic container configurations, and eight (8) specific container configurations (which includes three (3) configurations for Shielded Containers), were authorized. There were a total of seven (7) Addenda and five (5) Letter Amendments issued. One change was made to the Package Safety Basis with a doubling of the DHR to 38 Watts and the addition of a Heat Dissipation Sleeve surrounding the Containment Vessel (CV).

Discussion

SRNL was approached in the Spring of 2014 with whether the 9977, as currently certified or in a modified configuration, could be used to ship the waste residues from the manufacturing of medical isotopes. The wastes are a high-gamma source consisting of enriched uranium with activation and fission products, and actinides, mainly ^{144}Ce, ^{90}Sr, ^{106}Ru, ^{125}Sb and ^{137}Cs. The targets were exposed to a maximum flux of 1.5×10^{14} n/cm^{2}s for a maximum of 120 hours. The waste material from each target were placed either singly or doubly into "cans" with the waste from each target containing a maximum of 175 g ^{235}U. The shipper wanted to ship two cans per 9977 in order to minimize the number of packages needed and the shipments made.

To evaluate the proposed content, the bounding case of two doubly loaded Cans was compared against the previously evaluated and certified contents and configurations. The certified 9977 Package design was evaluated for several kilograms of ^{235}U and a Package maximum gross weight of 350 lbs with a maximum 100 lbs of Contents (everything within the CV).

Containment

No changes to the CV were anticipated. Therefore, as long as the CV Contents weighed no more than 100 pounds, the baseline leaktight nature of the CV and Package would be maintained and would meet the 10 CFR 71.51(a)(1) and (2) allowed leakage limits. Dose rate measurements of the Cans confirmed that additional shielding would be needed within the 9977 in order to meet the regulatory limits. If additional shielding material were needed inside the CV and the weight of that shielding and the contents exceeded the 100 lb design limit the CV would have to be re-evaluated at that higher weight and demonstrated to still be leaktight.

Criticality

The Content was evaluated at a maximum limit of 700 grams of ^{235}U (i.e. two double loaded Cans or 2 loadings/Can × 2 Cans/9977 × 175 gram ^{235}U/loading). At this mass the 9977 Contents would be at the ^{235}U sub-critical mass limit. [Ref. 5] Therefore, all the Normal Condition of Transport and Hypothetical Accident Condition previously evaluated, including the Single Flooded Package performance requirements of 10 CFR 71.55, bounded the new Content and would be subcritical. Arrays of sub-critical masses in the 9977 have been demonstrated to remain subcritical [Ref. 6] in compliance with the requirements of 10 CFR 71.59.

Shielding

SRNL was provided by the manufacturer with a source term of the waste RAM material at the time of loading into their Cans. SRNL was also given some results of measurements of the dose rates at the surface of the Cans, which varied between 10,000 and a few hundred mrem/h. For the desired maximum content definition for the Package, it was assumed that the two waste cans were both double loaded with "fresh" waste. Preliminary dose rate calculations determined that the 9977 Package with the maximum content loading would be within the Regulatory dose limits if the Package incorporated a minimum of 1.6 inches of tungsten shielding all around the contents. To emplace this thickness of shielding within the package would require shielding material filling the available space both within and around the CV. The conceptual design for the tungsten shielding within the 9977 Package is shown in Figure 1. The estimated weight of the shielding within the CV is 260 lbs and the total configuration weight is 850 lbs. This was referred to as the "9977-Heavy" configuration.

Structural

The addition of 1.6 inches of tungsten shielding raised the total weight of the Package by approximately 500 lbs over the certified maximum package weight and the CV Content weight to more than 100 lbs. Therefore, it became necessary to structurally reevaluate the entire 9977 Package and the CV to determine if they would meet the performance requirements at the higher weights. For the 9977 certification HAC drop testing and analysis, the 30-ft drop in the Side Down drop orientation produced the highest demand for the CV stress condition. The 30-ft drop in the center-of-gravity over top corner (CGOTC) orientation was almost as demanding, with stress levels approximately 90% of the Side Down drop. In terms of overpack deformation and the loads to the overpack lid bolts, the CGOTC orientation was by far the most challenging. To analyze the relative changes for the 9977 package with the shielding structure added, the CGOTC was chosen as the bounding case for a feasibility study. The demands on the overpack during this drop are dependent upon the weight it contains, so it was expected that the overpack would deform more, experience higher metal strains, and that the lid bolt loads would increase. The stress levels on the CV would be expected to follow a similar trend; except for special issues associated with the 9977 certification simulations. To address a general loading condition, the HAC drop simulations used in the 9977 certification were based on a dense point source for the 100 lbs content (with the point source having a finite volume based on practical material densities). For the 9977 package with the shielding, the CV stresses are expected to be reduced, since the inner and external shielding are formed to the CV shape, acting as strong-backs. These two effects are expected to overcompensate for the penalty of the heavier payload.

Impact G-Loads

Deceleration levels (“G” Loads) of the Package are established from the steepest slope (rate of change) of the velocity and acceleration time histories during the drop. The additional shielding weight causes more drum deflection and a greater impact duration. In essence, the overpack design used in the certified 9977 package now responds as if it is “softer”, resulting in lower deceleration. This effect is borne out in the analytical simulation. The G Load for the 850 lbs package in the worst case orientation is 178 G, as compared with 217 G for the certified 9977 package. The deceleration of the CV is similarly reduced.

Package Lid Bolting

The time history of the bolt tensile forces during the 30 foot free-fall impact, with drum oriented CGOTC were evaluated using a finite element analysis (FEA). The FEA simulation begins with no bolt pre-load, and the drum with a 527.5 in/sec initial velocity, positioned 1.0 inch above the target. The bolt pre-load is ramped on in the first 0.0027 seconds of the simulation.

The bolt tensile force history shows that the bolt closest to the impact point experiences loss of tension, which is regained after the impact phase. A compressive load of approximately 6500 lbs is caused by the drum top ring folding over onto the bolt head. The maximum bolt force is approximately 13,000 lbs, occurring in the bolt most distant to the point of impact. Bolts 1 and #4 see the maximum sustained shear load, 9,300 lbs, which return to near zero shear after the impact. Compared to the certified 9977 configuration, the bolt tension loads for the 9977-Heavy case are about twice as severe, shear loads are about 25% more severe. Because of the larger loads, the bolt stress interaction was computed for each bolt and for each time instance, rather than the bounding non-contemporary method used in the 9977 certification analysis. The highest elliptical tension-shear interaction for the 9977-Heavy configuration is 0.9 (90%).

Outer Drum Shell

The significant parameters in terms of overpack performance during the CGOTC impact include the deflected shape (which can readily be compared to physically tested specimens), energy absorber utilization (e.g. the percent of foam crush), and the maximum strains on the drum shell.

<u>Deflected Shape and Crush Utilization</u>

Figure 3 shows the deflected shape of the package after the CGOTC drop simulation. The inertial loads from the content and CV weight in the package core region cause an outward bending of the overpack lid, which deform until contact with the impact surface occurs. In the base 9977 certification analysis the TR-19 in the lid bottom section was only slightly compressed, with approximately ¼-inch compression of the over 4 inch thickness of TR-19 material, or about 6% compression. In the analysis of the 9977-Heavy, the TR-19 is compressed about 2.5 times that of the 9977 certification case, the utilization of about 15% is still acceptable.

<u>Maximum Strains on Overpack Metal Components</u>

Figure 4 shows the total accumulated plastic strain in the metal components of the overpack for the CGOTC drop simulation. The maximum strain is 58%, localized to a small area around the bolt holes closest to the impact point and also at the lid-to-lid bottom weld. These strains exceed the 40% uniform elongation capacity, but are within the material's local strain tolerance of 100% (based on area reduction requirements during tensile tests).

Containment Vessel

Prior to the HAC drop, the initial stress level in the containment vessel is a 4 ksi general primary membrane stress which is due to the 422 psi initial pressure. For the 9977 certification, the HAC drop of the 9977 Package in the CGOTC orientation, the maximum stresses in the CV were 30.5 ksi P_m and 39.6 ksi P_l. In the modified 9977-Heavy configuration, the stress levels are reduced, with P_m less than 30 ksi, and P_l = 35.5 ksi. The reduction in the stress is due to the more distributed load bearing condition provided by the internal shield structure verses the more concentrated content load condition used in the 9977 certification basis.

Crush Analysis Results

The crush is no applicable due to package density. Since the package overall stiffness has not changed, and the package internal mass does not alter crush response, the additional effects of crush are not changed from the 9977 certification basis.

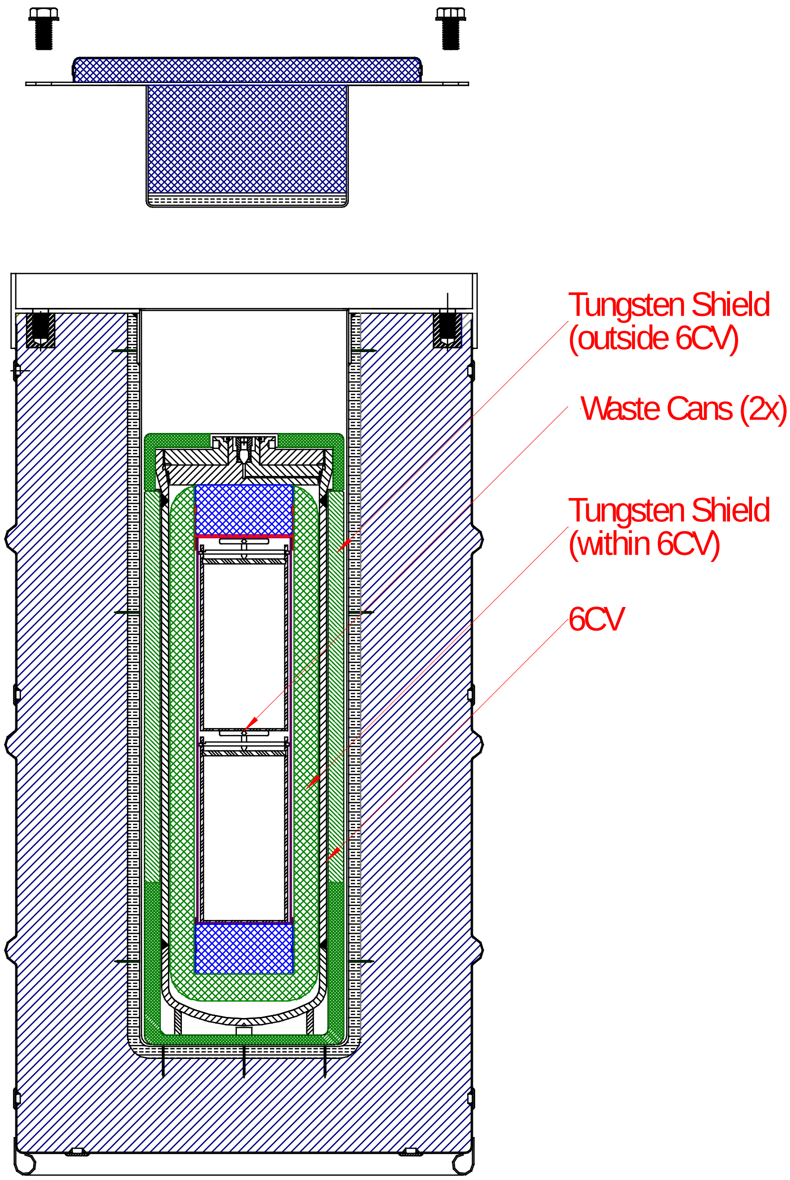

Figure 1 - The 9977 Package with Tungsten Shielding (a.k.a. the “9977-Heavy”)

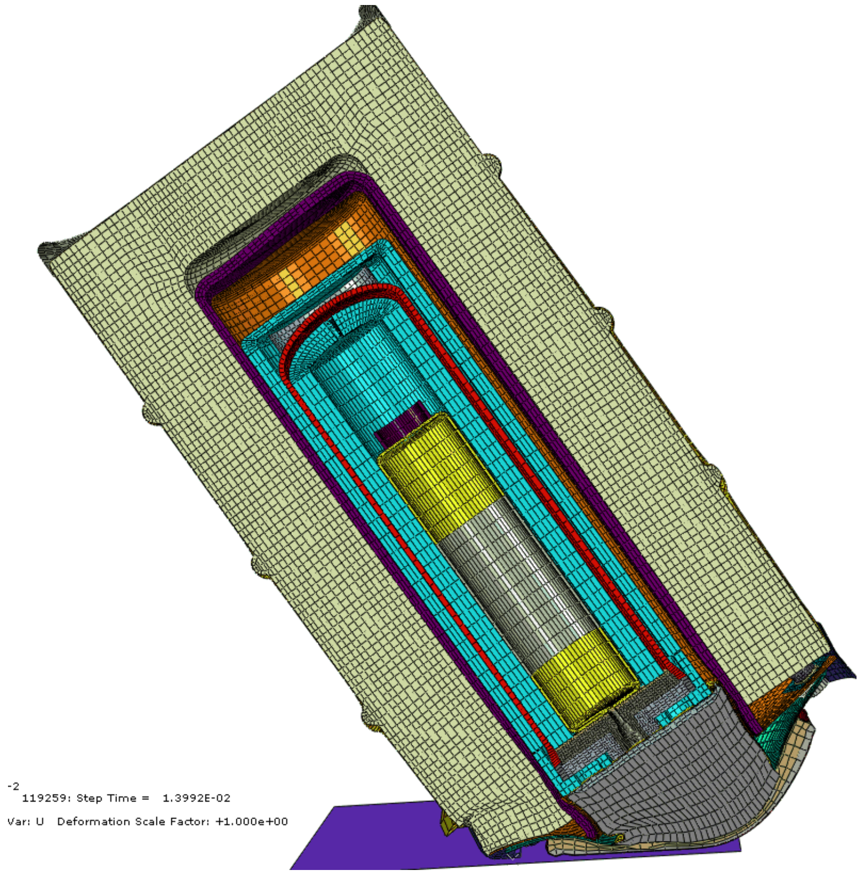

Figure 2 - Deflected Shape of the 9977-Heavy after HAC Free-Drop Simulation, CGOCT Orientation, Room Temperature Foam Condition.

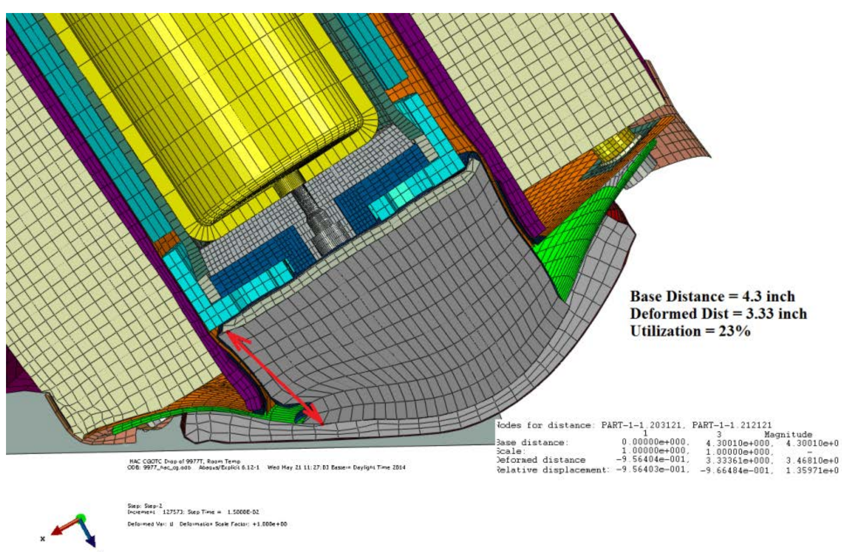

Figure 3 – “9977-Heavy” Analysis – 30 Foot CGOTC (Showing less than 23% Lid Plug crush)

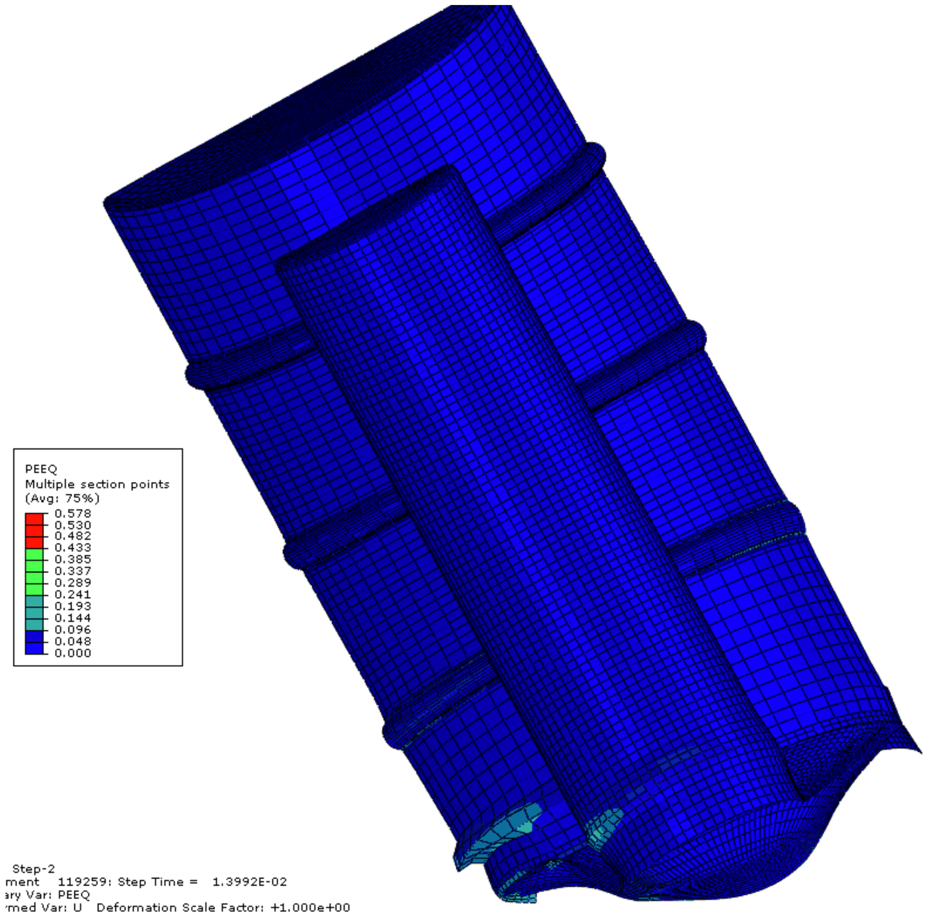

Figure 4 – "9977-Heavy" Analysis – Contour Plot of Total Accumulated Plastic Strain in Drum Shell After HAC CGOTC Free-Drop Simulation (Showing less than 23% Lid Plug crush)

Conclusion

The 9977 Package requires modification outside it certified configuration in order to ship the waste residues from medical isotope production. Its certified configuration does not provide sufficient shielding for the high gamma source material. The 9977 configuration does have sufficient internal volume such that sufficient tungsten shielding could be added to reduce the bounding waste content to be within the regulatory package dose rate limits. The weight of the additional shielding material would cause the gross package weight to exceed the maximum certified shipping weight of the 9977. Therefore, a new Certification for the "9977 Heavy" Package would be required. A preliminary structural analysis has determined that the 9977 Heavy would pass the HAC testing.

References

1 *Safety Analysis Report for Packaging Model 9977 Type B(M)F-96*, S-SARP-G-00001, Revision 2, Savannah River National Laboratory, Savannah River Site (August 2007).

2 *DOE Certificate of Compliance No. 9977 for the 9977*, 9977, Revision 0, U.S. Department of Energy (October 2007).

3 *Packaging and Transportation of Radioactive Material*, Code of Federal Regulations, Title 10, Part 71, Washington, DC (January 2015).

4 Regulatory Guide 7.9, *Standard Format and Content of Part 71 Applications for Approval Of Packages For Radioactive Material*, Revision 2 (March 2005)

5 ANSI/ANS-8.1-2014, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors," (April 2014)

6 M.D. Harris, *Nuclear Criticality Safety Evaluation: Single Parameter Subcritical Masses in a 9977 Shipping Package, Analysis for SARP*, N-NCS-A-00035, Savannah River Nuclear Solutions (January 2015)