Introduction

As part of the Isotope Production Program at Sandia National Laboratories New Mexico (SNL/NM), procedures are being finalized for the production of $^{99}\text{Mo}$ from the irradiation of $^{235}\text{U}$-coated stainless steel targets at the Technical Area (TA) V reactor and hot cell facilities. Methods have been identified and tested for the management of the non-product (waste) material as the final step in the production process. These methods were developed utilizing the waste material from a series of cold and hot tests, beginning with depleted uranium powder and culminating with a test involving an irradiated $^{235}\text{U}$ target with an initial fission product inventory of approximately 18,000 Ci at the end of the irradiation cycle.

Description of Work

The waste material from $^{99}\text{Mo}$ production includes two distinct streams; contaminated/activated hardware and equipment, and liquid residue. These materials are generated in three heavily shielded structures in the TA V Hot Cell Facility known as Steel Confinement Boxes (SCBs). The contaminated hardware consists mainly of glass bottles and resin columns, plastic syringes and tubing, needles, the stainless steel target tube, and two copper tubes used to capture the fission gases. Of these materials, only the stainless steel target tube and associated connecting hardware contain activation products from irradiation in the reactor. The remaining materials are contaminated with fission products.

After separation of the $^{99}\text{Mo}$ from the target uranium, approximately 300 ml of liquid residue contains the bulk of the remaining fission products (~20,000 Ci after a 6-hour decay for a production level target) and has a pH of about 1 due to dissolution of the target material in nitric and sulfuric acids during the $^{99}\text{Mo}$ separation process. Procedures were developed so that this high activity liquid could be stabilized and temporarily stored on site, allowing decay of the shorter-lived radionuclides, followed by eventual disposal.
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at a DOE low-level disposal site. The procedure developed for stabilization included reducing the possibility of iodine release from the material and solidifying the liquid. Requirements for the stabilization agent included the following criteria:

- The final waste form must be able to withstand the high radiation levels (3E+07 Gy lifetime dose) associated with production-level material.
- The stabilization agent must be amenable to remote mixing with the liquid residue using the manipulators in the SCBs.
- The final waste form must be acceptable for disposal at the Nevada Test Site low-level radioactive waste disposal facility, which is the DOE disposal facility to which SNL/NM is assigned.

**Results**

Studies performed at SNL/NM to support processing of the waste from ⁹⁹Mo production included the testing of several commercially available stabilization agents for the ⁹⁹Mo liquid residue: Portland cement (Type I-II, low alkali, ASTM C150-94), Aquaset II⁺, Aquaset II Hard⁺, and Aquaset II Granular⁺. The effects of a 3E+07 Gy dose on the four potential agents were identified by preparing a surrogate ⁹⁹Mo waste solution and mixing portions of this solution with the candidate agents. The resulting mixtures were then exposed to the prescribed dose in the SNL/NM Gamma Irradiation Facility and the results documented. Of the four commercial products tested, both the Aquaset II Hard⁺ and the Portland cement showed little change when exposed to the 3E+07 Gy, while the Aquaset II⁺ and Aquaset II Granular⁺ both exhibited damage. Because the Portland cement appeared to form a stronger, less porous product, it was eventually selected as the stabilization agent for the ⁹⁹Mo liquid residue using a ratio of 2.0 grams of cement/ml of liquid. In addition, because the Portland cement is a basic material, no neutralization of the acidic liquid residue is required prior to injection into the cement. Using this procedure, bench-top tests with a surrogate solution indicated that the resulting pH of the waste/cement mixture is about 6.5. After waste injection, commercially available equipment was used to mix the liquid residue and Portland cement in a stainless steel containment vessel designed by Marion McDonald of SNL/NM(1). No additional water is required. Tests with surrogate waste demonstrated that 15 minutes of mixing was sufficient to produce a
homogeneous slurry. Initial curing to a hard solid was complete in less than 24 hours. Samples of stabilized waste have passed the Resource Conservation and Recovery Act Toxicity Characteristic Leaching Procedure for metals, which is one of the requirements for disposal of the waste at the Nevada Test Site.

To limit the possibility of iodine release from the liquid residue, the waste processing also includes the addition of sodium bisulfite (NaHSO₃). This allows the waste to be more effectively stabilized due to increased solubility in aqueous solutions by reduction of the elemental iodine (I₂) to iodide ions (I⁻).

References

(1) Applicable Sandia Drawing numbers R30859 and R30860.