RECENT EXPERIENCE IN PLANNING, PACKAGING AND PREPARING NONCOMMERCIAL SPENT FUEL FOR SHIPMENT IN THE UNITED STATES

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SUMMARY

The U.S. Department of Energy - Headquarters (DOE-HQ) has issued a Record of Decision (ROD) which identified the plan to be followed in managing spent nuclear fuel (SNF) belonging to the Department. As a result, the aluminum-clad fuels stored at Oak Ridge National Laboratory (ORNL) in Oak Ridge, Tennessee, were directed to be shipped to the Savannah River Site (SRS) near Aiken, South Carolina. The BMI-1 cask was chosen to make the shipments of SNF from dry storage that had to be placed in canisters. However, the Certificate of Compliance (COC) for the BMI-1 cask limited the fissile material loading to 800 g of unirradiated fissile material for the cask configuration chosen. Because about half of the canisters were already filled and sealed with more fissile material than was permitted by the COC, approval to make these shipments in the BMI-1 was requested from the Nuclear Regulatory Commission (NRC). A safety analysis showed that the shipments could be made safely under the conditions identified. The waiver was granted in September 1997 and the three shipments were successfully completed in January 1998.

INTRODUCTION

DOE is committed to the safe, efficient, and cost-effective transport of radioactive material. However, some of the unique contents that must be shipped by DOE often create challenges not typically seen in the commercial sector. This paper reviews one such experience that highlights the need for careful planning in a shipping campaign and illustrates the types of problems that can arise when shipping unique materials in existing packages. The material to be shipped were pieces of aluminum-clad SNF irradiated at ORNL, resulting primarily from the Reduced Enrichment Research and Test Reactor (RERTR) Program carried out in the early 1980s. Following completion of the irradiation, some of the fuel was cut up to study its physical behavior, and some was analyzed to determine its postirradiation fissile content. After completion of the test program, the fuel was stored in below-grade storage positions at ORNL.

In 1996, DOE issued a ROD (Fed. Regis: **60**, 28680), which selected regional storage locations for SNF by cladding type as the option for long-term retention of DOE-owned SNF before its disposal in a repository. This ROD resulted in plans for aluminum-clad SNF pieces stores at ORNL to be shipped to the DOE-designated storage location

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at SRS. This paper will review the efforts involved in the planning, contents preparation, and additional safety analysis related to shipment of this material to SRS.

INITIAL PLANNING

The aluminum-clad fuel destined for SRS had been stored in large stainless steel containers at ORNL. However, SRS would accept fuel only in containers fabricated from aluminum. The fuel had to be inspected and repackaged in a hot cell and then placed in aluminum canisters that had a cavity of at least 95 cm in length because some of the fuel, in the form of intact plates, was that length. The internal diameter of the canister was limited to approximately 11 cm by handling constraints at SRS. In addition, to ease the handling and accountability issues, small aluminum cans were used to accept whatever small pieces existed. The total quantity of fuel available for shipment was sufficient to fill up to 12 canisters.

In evaluating the facilities available at ORNL to transfer the fuel from storage to a cask, it became apparent that the fuel, once inspected and repackaged, could be transferred on-site to a specific hot cell for the final loading operation. The shipping cask could be placed into the hot cell and a bottom-loading transfer cask containing one canister, could be placed on top of the hot cell and mated to a hole in the cell roof plug. The canister would then be lowered into the cell and into a basket designed to accept it. The basket, once filled with several canisters, would then be lifted and placed into the cask cavity. This procedure placed another restriction on the cask: it had to be physically small enough to fit into the hot cell.

One shipping cask was identified whose cavity could hold four such canisters (thus reducing the number of shipments to SRS to three) was already certified to ship irradiated plutonium and uranium in solid form when placed in a sealed inner can assembly (ICA). In addition, the cask was small enough to allow it to be moved into a hot cell and be loaded with the canisters. This was the BMI-1 cask. Another advantage of this cask was that SRS had experience in unloading this cask when receiving irradiated fuel from university research reactors.

PACKAGING IN HOT CELL

It became apparent that in order to transfer the fuel into and out of the cask, the fuel would also need to have certain handling characteristics that could be provided by canisters designed for that purpose. The canisters would have to fit into the sealed ICA as required by the COC for the BMI-1 cask to ship this type of fuel.

The fuel pieces destined for the SRS had been stored in containers at ORNL for more than 20 years, and the quality of the records that identified their contents was unknown. It was imperative that the cans be transferred to a hot cell, opened, their contents inspected, and then loaded into the appropriate canisters for shipping. Following packaging, the fuel was then to be transferred to a second hot cell in which the canisters could be loaded directly into the BMI-1 cask for off-site shipment.

Existing small aluminum cans with a capacity of about 400 cm^3 and a screwtop were used as inner containers for the smallest fuel pieces. These cans, when filled, could efficiently contain small pieces and were easy to handle individually in the hot cell; also, several could be stacked in the aluminum canister that could then be transferred to a second hot cell and loaded into the BMI-1 cask. The canisters also supplied a second line of containment for the fuel pieces, some of which had been cut to examine fuel.

The aluminum canisters are 12 cm in diameter, 99 cm long, and were designed to be closed with a freeze plug (Fig. 1). The freeze-plug closure was attractive because the canister was not to be introduced into the cell, but remain clean, outside, with just the bottom of the canister butted up to an opening in the cell wall. In this condition, just the inside of the canister was exposed to the cell atmosphere. The freeze plug was introduced into the cell and cooled with liquid nitrogen. Once the canister was loaded, the plug was removed from the coolant and pushed into the bottom of the canister, where it expanded and sealed. Prototype tests of this design confirmed that the canisters were sealed and would hold more weight than could be loaded.

Oak Ridge staff designed a basket that could support four aluminum canisters when they were placed inside the ICA. This inner basket weldment is constructed of type 304 stainless steel; the basic structure consists of a cylinder that has a 31.5-cm outside diameter, a 30.5-cm inner diameter, and is 103 cm long. The basket has 12.7-mm-thick upper and lower plates that support four canisters. These plates are welded to the outer shell and have a 5-cm-diam center steel rod and eight 2.2-cm rods around the periphery that connect the two plates together. The top plate has four 12.4-cm holes drilled through it, and the bottom plate has matching holes countersunk about halfway through the plate to support the canisters. Attached to the bottom of the basket is a 27.3-cm long centering device that keeps the basket located in the center of the ICA. The center steel rod is equipped with an eyebolt to facilitate handling and removal of the basket from the ICA. Figure 2 shows the basket containing four canisters.

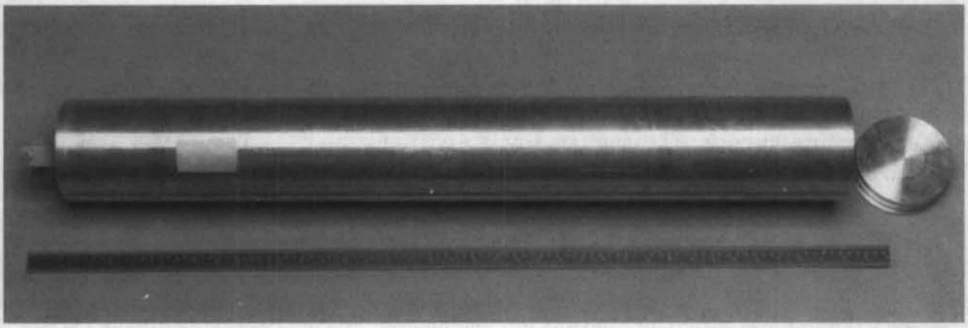

Fig. 1. Typical aluminum canister with a freeze plug.

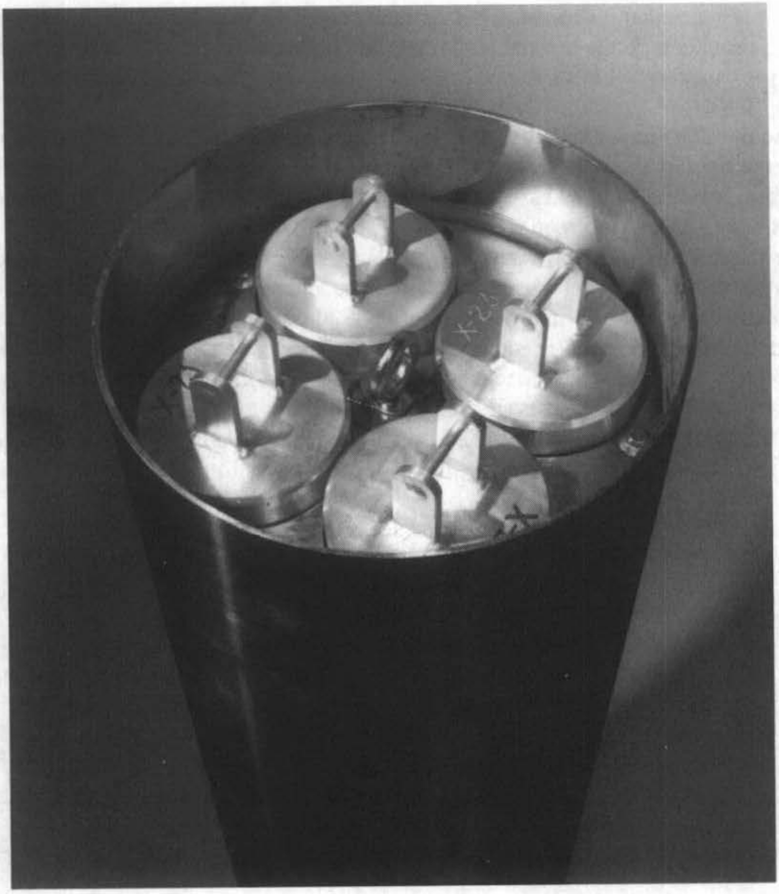

Fig. 2. Stainless steel basket containing four canisters.

THE BMI-1 COC

During the process of preparing the fuel for shipment, the COC was revised by NRC to require that the mass of fissile material allowed in the cask be based on preirradiation values. Earlier versions of the BMI-1 COC had not specified this condition, but simply limited the fissile quantities to 800 g of ^{2335}U. Early in the analysis of the planned shipments, the fissile content of the material planned for each canister was examined in detail. Most of the fuel plates came from the RERTR Program. Because of the nature and purpose of this program, the irradiation of the fuel had been carefully managed and documented, and samples of the fuel were analyzed to verify the fuel burnup and remaining fissile content.

Six of the canisters had been loaded so that each shipment would contain less than 800 g of ^{235}U based on their postirradiation fissile content. Although the remainder of this paper focuses on preirradiation contents, the fissile loadings for each shipment, as given in Table 1, provide both the pre and postirradiation numbers. Note that shipment 1 contained only three canisters, whereas shipments 2 and 3 contained four canisters each.

Because the COC now limits the fissile loadings to less than 800 g ^{235}U based preirradiation values and the fact that ORNL had already loaded and sealed several canisters containing significant quantities of ^{235}U based on postirradiation data, a choice had to be made. The choices were either (1) to request of the NRC a waiver to permit greater than 800 g of ^{235}U in each shipment based on an analysis that showed the increased fissile loadings would be safe, or (2) to open the previously sealed canisters and repackage the fuel to bring down the fissile loadings per canister in order to meet the 800 g ^{235}U preirradiation limit per shipment. The latter course would have required hot-cell technicians to receive higher radiation doses unnecessarily because of the repackaging operations, and the number of shipments would have increased from three to seven, resulting in exposing the public to over twice the dose during the entire shipping campaign, and also significantly increasing the possibility of nonradiological accidents. Therefore, the path chosen was to petition the NRC through

Table 1. Canister and cask fissile loadings

Canister ID number	Pre-irradiation		Post-irradiation			Decay heat (W)
	U(g)	^{235}U(g)	U(g)	^{235}U(g)	Pu(g)	
Shipment 1						
X-13	1485	327	1302	144	5	≤6
X-14	2915	583	2471	139	11	≤10
X-15	4246	848	3604	206	15	≤15
Total	8646	1758	7377	489	31	≤31
Shipment 2						
X-16	3934	905	3331	320	18	≤14
X-17	1082	249	916	83	5	≤4
X-20	285	265	126	106	0	≤1
X-21	752	232	524	61	5	≤3
Total	6053	1651	4897	552	16	≤22
Shipment 3						
X-12	2248	601	1849	203	13	≤8
X-19	4408	882	3781	255	16	≤15
X-22	505	192	451	138	1	≤3
X-23	125	117	125	117	0	≤1
Total	7286	1792	6206	713	30	≤27

DOE to permit the higher fissile loadings; which, in turn, initiated a criticality and structural evaluation of the canisters and basket to prove that such shipment would be safe.

DETAILS OF THE ANALYSIS

The analysis (Safety Analysis 1996) focused on two primary concerns: (a) would the canisters remain sealed and in their proper positions under the hypothetical accident conditions and (b) would the shipments remain critically safe with the higher than 800 g ^{235}U fissile loading even when flooded and reflected by water. Thus, the ICA, the basket and the sealed canisters were evaluated structurally using the DYNA3D code (Hallquist et al., 1995) to confirm that the canisters would not release their fissile contents. Once that was established and the most reactive geometry from a criticality point of view could be determined, the criticality analysis was initiated.

THE STRUCTURAL ANALYSIS

Three-dimensional, finite-element (FE) models of the ICA and contents were developed. The FE model was used as input to a numerical simulation of the impact problem in which the model had a one-to-one correspondence in geometry and inertia when compared with the prototype specimen. The analysis modeled the impact of the ICA and its contents when this assembly contacted the unyielding surface in four orientations: (1) vertically onto its base, (2) vertically onto its top, (3) horizontally onto its side, and (4) at an oblique (CG-over-corner) angle. No credit was taken for energy absorption through the walls of the cask or at the interface of the cask cavity and internal can. Because the model geometry for the top and bottom drops had two planes of symmetry, only one ¼ model of the ICA and contents had to be developed. For the case of a side and an oblique drop, it was necessary to develop two other separate models.

The results indicated that in each of the drop scenarios, the calculated maximum stresses in the canister was always below the allowable stress for the maximum temperature considered.

THE CRITICALITY SAFETY ANALYSIS

The criticality safety analysis was performed to demonstrate that the three separate loadings of the BMI-1 cask would be adequately subcritical with the specified preirradiated fissile mass loadings. The analysis was performed assuming that the only available information was the elemental form of the SNF (U_3O_8, UAl_x, etc.) and the mass of ^{235}U and uranium in each canister (from which average enrichments were determined). With this information a process was followed to determine the combination of particle size, fissile-to-moderator ratio, and the SNF position within the canister that provides the highest neutron multiplication factor (k_{eff}) for the package under all transport conditions specified by the regulations. All the computational analyses were performed with the criticality safety analysis sequences (CSAS's) within the SCALE code system (SCALE 1995). Final k_{eff} values were calculated with the KENO V.a Monte Carlo code within SCALE.

Because the SNF pieces varied in size and shape, the first step was to assume that the SNF consisted of spherical particles in a close-packed, triangular-pitched array. Then for each fresh fuel enrichment, the particle size and pitch were varied to determine the size and pitch that provided the highest reactivity. This information was used to generate (using CSAS/XSDRNPM analyses) cell-weighted cross sections representative of these (one for each enrichment) optimum fuel–moderator mixtures.

Within the physical limitations of the known mass and calculated particle size and pitch, various arrangements of the uranium–water mixtures within the canisters were studied using a detailed KENO V.a model of the BMI-1 cask loaded with canisters. The various arrangements were studied to obtain maximum neutron interaction between canisters and provide the highest k_{eff} value for a single package. The highest single package k_{eff} value obtained (for all three shipment loadings) was 0.8075 ± 0.0023. To meet the regulatory conditions for the analysis of arrays of packages, an infinite array of packages was analyzed using the single package model that provided the maximum k_{eff} value. The reflection provided by the thick shielding of the BMI-1 cask prevents neutron interaction between packages and so, as expected, the maximum k_{eff} value for an infinite array of packages was statistically the same as the result for the single package.

The adequacy of the k_{eff} value determined by calculations was compared to an upper subcritical limit (USL). The value of the USL was determined by analyzing 66 critical experiments chosen for their representation of the physics involved in the actual package model. Based on a statistical assessment of the calculational bias and uncertainty from analyzing these 66 critical experiments together with an additional 5% margin of subcriticality, the lowest USL value over the full range of experiment characterization parameters was 0.9161. Thus, the conclusion of the criticality safety analysis was that the largest k_{eff} value of any package loading under consideration was far below this demonstrated limit of safety.

NRC APPROVAL

On September 12, 1997, based on the high fissile loading limits and analysis previously discussed, NRC approved the request by DOE to make three shipments of irradiated research reactor fuel plates in the BMI-1 cask from ORNL to SRS. As a result, SRS scheduled dates to receive the loaded BMI-1 cask at their Receiving Basin for Off-Site Fuels (RBOF) facility.

Shipments

The three shipments were made to SRS site during the three months from November 1997 through January 1998. No technical problems were encountered during the transfer of the spent fuel.

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