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TMI Fuel Characteristics for Disposal Criticality Analysis

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ABSTRACT

This report documents the reported contents of the Three Mile Island Unit 2 (TMI-2) canisters. proposed packaging, and degradation scenarios expected in the repository.

Most fuels within the U.S. Department of Energy spent nuclear fuel inventory deal with highly enriched uranium, that in most cases require some form of neutronic poisoning inside the fuel canister. The TMI-2 fuel represents a departure from these fuel forms due to its lower enrichment (2.96% max.) values and the disrupted nature of the fuel itself.

Criticality analysis of these fuel canisters has been performed over the years to reflect conditions expected during transit from the reactor to the Idaho National Engineering and Environmental Laboratory, water pool storage,[1] and transport/dry-pack storage at Idaho Nuclear Technology and Engineering Center.[2,3] None of these prior analyses reflect the potential disposal conditions for this fuel inside a postclosure repository.

SUMMARY

Three Mile Island Unit 2 (TMI-2) fuel canisters represent a significant MTHM quantity of fuel intended for repository disposal. Within this grouping, there are three different types of canisters. There are three types of TMI canisters under consideration for disposal in the national repository.

This document provides details of the parameters of the various TMI canister types needed to perform a criticality analysis. Furthermore, the document also stipulates the conditions anticipated for degradation failures within the repository that must be addressed in the criticality analysis prior to acceptance of this spent nuclear fuel in the repository.

The knockout canister with a full assembly’s worth of fuel pellets has been selected as the bounding case for criticality analysis. The knockout canister design offers fewer constraints for fuel pellet distribution and obtaining optimal modernization due to the greater free volume for moderator. Any such modeling should investigate optimized fuel distribution and moderator ratios toward investigating maximum reactivity.

FOREWORD

The work done in this characterization effort and the planned criticality analysis complements previous criticality analyses done for other U.S. Department of Energy (DOE) fuel types. These studies are intended to demonstrate an ability to qualify the DOE spent nuclear fuel (SNF) canisters from a criticality safety standpoint for the licensed duration of the national repository.

The approach taken in previous analyses employed a combination of five defense high-level waste glass logs (located peripherally) and a single DOE SNF canister in the center position in a waste package. This combination of canisters makes up what is referred to as a codisposal waste package. The type of fuel determines the length of the SNF canisters (either 10 or 15 ft). The length of the high-level waste glass canisters (either 10 or 15 ft) will be dependent on the glass production facility.

Prior criticality analyses have been directed at qualifying single SNF canisters for conceived (probable) configurations inside a codisposal package. These configurations range from intact fuels to totally degraded SNF canister contents (either fuel or basket or both), mixing degraded high-level waste into the SNF canister, and distribution of degraded fissile material outside the SNF canister but within the waste package.

The TMI canisters present low-enriched fuels in an already degraded condition due to the damage incurred during the TMI-2 reactor accident. The material in any given canister cannot be verified in terms of actual condition or position. This analysis will focus on identifying an acceptable fissile loading supposition. A subsequent analysis will then use known and postulated fuel canister parameters to identify a most reactive configuration within the expected repository environment.

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ACRONYMS

BOL	beginning-of-life
DOE	U.S. Department of Energy
EOL	end-of-life
GPU	General Public Utilities (TMI operator)
HLW	high-level waste (glass)
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
MCNP	Monte Carlo N-Particle
NRC	Nuclear Regulatory Commission
ORIGEN	Oak Ridge Isotope Generation
PWR	pressurized water reactor
SNF	spent nuclear fuel
SNM	special nuclear material
TMI	Three Mile Island
TMI-2	Three Mile Island Unit 2

NOMENCLATURE

Active length	portion of a fuel rod or pin that contains fissionable material
Borosilicate glass	generic description of solidified waste form from radioactive liquid waste
Burnup	a measure of the amount of fissile material consumed before the fuel element is removed from the reactor
Codisposal	the combination of both defense high-level waste and U.S. Department of Energy spent nuclear fuel in a single waste package
Filter canister	separated fine particulate from the water stream once it has passed through the knockout canister
Fuel canister	also known as a defueling canister by its “D” designation; contains damaged Three Mile Island fuel assemblies
Goethite	a ferric oxide hydroxy [FeO(OH)] that forms upon iron decomposition inside a breached waste package
Knockout canister	used to disengage particulate that was hydro-vacuumed from the reactor vessel
LiCon™	“light” concrete; grout mix found inside “D” type Three Mile Island canisters
MTHM	metric tonnes heavy metal
Postbreach clay	high-level waste degraded glass plus materials from degraded spent nuclear fuel canister contents
Prebreach clay	high-level waste degraded glass without fuel degradation products
SNF canister	a standard canister design that will employ a variety of internals to accommodate the variety of DOE spent nuclear fuels
Waste package	the engineered barrier surrounding either commercial spent nuclear fuel, or a combination of DOE spent nuclear fuel and defense high-level waste

TMI Fuel Characteristics for Disposal Criticality Analysis

1. INTRODUCTION

The Three Mile Island Unit 2 (TMI-2) canisters (344) were generated as a result of recovery and cleanup of a reactor core after the TMI-2 accident. The physical dimensions of the TMI-2 canisters are such that not more than a single, intact commercial 15 × 15 Babcock & Wilcox pressurized water reactor (PWR) fuel assembly could be installed in any one package. Indeed, there were 177 assemblies in the initial reactor core load and a total of 344 TMI-2 core fuel canisters.[4]

Of the original 344 TMI-2 canisters shipped to the Idaho National Engineering and Environmental Laboratory (INEEL), 341 have been dried and shipped to a Nuclear Regulatory Commission (NRC)-licensed facility at the Idaho Nuclear Technology and Engineering Center (INTEC). Of the three remaining canisters, two were consolidated into one canister.[5] The other two canisters remain in other storage pending identification of a disposal path.

There is no documented information indicating that even one, intact PWR assembly was ever installed in any TMI-2 canister. The highest reported physical mass of debris inside any canister (842.18 kg) (see Reference 4) represents 123% of the specified weight of an intact PWR assembly (see Appendix B). The highest reported fissile loading in any TMI-2 canister (10.06 kg) is only 73.3% of a beginning-of-life (BOL) fissile load for a PWR assembly with maximum enrichment (2.96%). The maximum BOL U-235 in any TMI-2 assembly of 13.72 kg will provide the basis for any single package criticality analysis conducted with TMI-2 canisters.

2. REACTOR INFORMATION

2.1 Three Mile Island Unit 2

The TMI-2 nuclear reactor was a Babcock & Wilcox PWR design. The core for this reactor consisted of 177 15 × 15 Babcock & Wilcox PWR assemblies. As a basis of operation for a new reactor, the PWR assemblies were constructed with decreased uranium enrichments comparable to what might be expected in a reactor with decreased reactivity as it approaches a typical refueling operation. These starting enrichments were 2.96% (max), 2.64%, and 1.98% (see Reference 4). Low power testing leading up to the limited operating time suggests approximately 10% of the useful life of the assemblies had been used at the time of the accident. Operational problems and subsequent attempts to recover operations after the reactor scram led to a partial draining loss of cooling water. This series of events led in turn to a partial meltdown of the fuel assemblies and damage to the reactor internals because of the latent heat in the reactor core.

2.1.1 Accident Recovery

After the reactor core melt down, initial recovery operations focused on stabilizing the core in a safe configuration. Follow up investigation included a determination of the condition of the core and what operations would be required to remove the radioactive portion of the resultant debris.

2.1.2 Core Removal

For core debris pieces that could be handled or grappled with underwater, remote tools were used to physically place the debris into a set of canisters with a “D” designation. Much of the core melting occurred in the center portion of the core, although damage appears to have intruded into even the peripheral assemblies. Later stages of core recovery included hydro-vacuuming operations of debris and this material was collected in the knockout canisters before flow was directed to filter canisters. A more detailed description of these canisters is presented in Section 3 of this report.

The tare weight and filled weight of each canister were recorded (see Reference 4). The materials loaded into the knockout canisters could not be characterized because the pieces entrained in the water stream were not suitable for sizing or accounting other than total added weight to each knockout canister. The filter canisters (those with an “F” designator) are the least heavily loaded of any of the canisters[6] and have never factored into any criticality calculations (see Reference 4).

3. FUEL INFORMATION

3.1 PWR Fuel Assembly

The PWR assemblies associated with the TMI-2 core represented a startup core in a new reactor, so maximum enrichment values are somewhat less than those found in a typical refueling situation. The description of the standard PWR assembly is of importance in understanding the quantity of materials associated with an intact assembly. This information provides the basis of subsequent assumptions needed to support calculations that detail fissile material distributions for criticality modeling.

3.1.1 Fuel Assembly Structure

Each TMI assembly consisted of a 15 × 15-pin square array for a total of 225 fuel pin locations. During manufacturing of the assembly, only an average of 208 of these 225 pins is filled. Therefore, the total quantity of heavy metal installed in a typical assembly actually occupies a slightly smaller volume than that occupied by the displaced volume of 225 pins. Location of the “empty” pins should be represented with an assumption of their uniform distribution throughout the assembly (see Appendix B for referenced information).

The typical fuel assembly used in the TMI-2 was a Babcock & Wilcox 15 × 15-rod array inside a 216.92-mm (8.54-in.) square envelope (see Figure 1 for a depiction of a typical PWR assembly). Although this array produces a total count of 225 rods per assembly, the manufacturer specification (Appendix B) indicates that typically only 204 of the rods are filled with uranium oxide pellets. While the total assembly is 4,206.24 mm (165.6 in.) long, and the individual rod lengths are 3,942.08 mm (155.2 in.), the active length of each rod is listed as 3,657.6 mm (144.0 in.). The selection of the specifications for a typical Babcock & Wilcox PWR assembly is based on the desire to model the system with more pellets. This higher pellet count could be considered to be more representative of the debris containing damaged fuel pellets. The important consideration in this decision was ensuring the total fissile mass was in agreement between the Babcock & Wilcox specification and the TMI-2 fuel specification. How this material ends up being distributed in terms of pellet quantities, porosities, pitch spacing , etc., all provide for an added degree of conservatism in the model construct.

3.1.2 Material Properties

Each fuel assembly is constructed of various metallic components, and in the case of the fuel matrix, an oxide of uranium. The details of these components are identified in the following sections of this report. All these items play an integral role in construction of a criticality analysis model for any intact assembly. For the damaged assemblies and debris layers inside any of the TMI-2 canisters, their omission generally promotes added conservatisms to the calculations for the remaining fuel matrix within each canister.

3.1.2.1 Cladding. TMI fuel cladding (Appendix B) was composed of Zircaloy-4 (Appendix A, Table A-7), but its presence in the TMI canister would be of more value for moderator exclusion when accounted for as a weight percent distributed throughout the package debris rather than as intact tubes in an assembly. For design purposes, and sake of reference, the cladding is 10.92 mm (0.430 in.) in diameter and had a thickness of 0.67 mm (0.0265 in.). These dimensions result in a fuel-clad gap of 0.107 mm (0.0042 in.). The fuel pins are installed in a square array with a pitch dimension of 14.43 mm (0.568 in.) (see Appendix B for referenced information).

The environment inside the reactor core during the thermal degradation created a complex chemical system. The reported formation of a hydrogen bubble in the reactor headspace immediately after

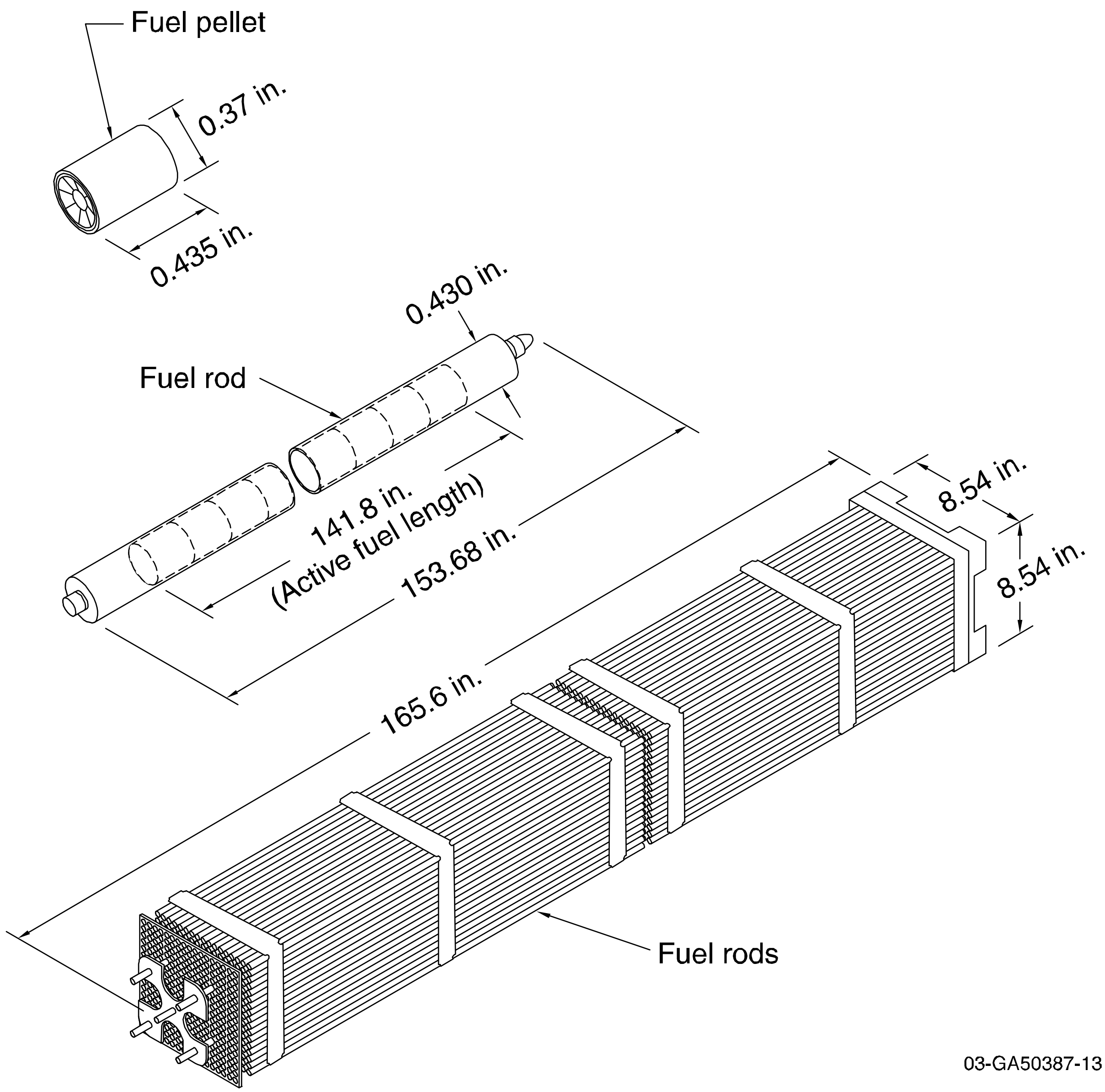

Figure 1. Typical pressurized water reactor fuel assembly.

Note: TMI-2 fuel assembly dimensions (see Appendix D for detail) vary slightly from the dimensions shown in this figure.

the accident suggests significant hydrolysis of water into its basic components of hydrogen and oxygen. A small portion of the metallic zirconium is postulated to have oxidized to zirconium dioxide (ZrO_2) (see Reference 6). Such oxide would likely be distributed throughout the canisters, but the quantities in any given canister are highly variable. Neglecting its presence in any canister would provide a conservative basis for the criticality calculations.

3.1.2.2 Fuel matrix. The fuel matrix is composed of cylindrical pellets that have dished ends and chamfered edges. The pellets are composed of uranium dioxide (UO_2) with an O/U ratio of 2-2.02:1. The theoretical density of uranium dioxide is reported to be 10.97 g/cc.[7] Reported production density that is 95% of theoretical; the production specification identifies an open porosity of <1%. (see Appendix B for referenced information).

Each pellet has a specified diameter of 9.362 mm (0.3686 in.) and a length of 11.05 mm (0.435 in.). Within each pin, the active length can best be described as a stacked column of uranium oxide pellets. Dividing the active length of 3,601.7 mm (141.8 in.) by the pellet length of 11.05 mm (0.435 in.) results in a calculated 325.98 pellets in a stack. This analysis assumes there are 325 pellets per pin. The fractional pellet count will be accounted for as gaps created by accumulated gaps between pellets in the stack. The total heavy metal mass will be modeled as individual, cylindrical pellets. The total number of pellets to be modeled in a canister would be 208 × 325 = 67,600. Each pin is listed as weighing 3.17 kg (7.00 lb) and contains a fuel matrix of 2.53 kg (5.58 lb) within each pin. With the exception of stainless steel springs (0.042 lb) inside the end of each pin, the balance of weight (7.00 – 5.58 = 1.42 lb) can be attributed to cladding material of Zircaloy-4. A total assembly weight equals 687.1 kg (1,515 lb) (see Appendix B for reference specification).

For purposes of modeling within the fuel canister ('D' designator), the available volume within each fuel canister is limited to the displaced volume described by the dimensional envelope of a single intact, PWR assembly (see Reference 1). A conservative approach to criticality modeling would neglect the metallic components of the assembly (due to moderator displacement) and assume a redistribution of the equivalent number of assembly pellets within the fuel canister.

The fuel matrix mass loaded into a single, intact PWR fuel assembly should provide the basis for calculating any criticality scenario associated with the TMI-2 canisters. The original design for the TMI-2 fuel canister provided the space or volume inside a canister to the equivalent of a single, intact PWR assembly. There is no information to confirm reactor recovery operations retrieved any completely intact assemblies from the core. Therefore, it should only be necessary to demonstrate (through data submittal prior to shipment) that none of the TMI-2 canisters contains more than either the total uranium or total fissile uranium (BOL) present in an intact assembly.

The specified quantities of total uranium and U-235 at 2.96% enrichment in a single 15 × 15 PWR assembly are 463.63 kg (1,022.12 lb) and 13.72 kg (30.25 lb) respectively (Appendix B). While reported enrichments of 2.64% and 1.98% exist for some of the assemblies, 2.96% enrichment is considered the bounding case and will be used for all criticality modeling. Any ingrowth of Pu-239 is considered minimal in relation to the burnup experienced by the various assemblies. The concentration of Pu-239 at the ~10% burnup levels is considered to be bounded by the assumption that a full, intact assembly of maximum enrichment is present in any TMI canister, e.g., TMI canister U-235 fissile (10.06 kg max) + 0.900 kg Pu-239 < 13.72 kg U-235 in PWR assembly.

The GPU reported uranium quantities (see Reference 4) that are in general agreement with the BOL uranium loadings that can be calculated from the Babcock & Wilcox fuel specifications.[8–10] With 177 assemblies in a reactor core, this yields a total uranium mass of 82,047.73 kg (180,915.24 lb) at BOL. This quantity compares within 1.0% of the 82,945 kg uranium mass reported (see Reference 4) in earlier studies related to fission product inventories estimated in the TMI-2 core at the time of the accident. While the 82,945 kg total uranium exceeds the Babcock & Wilcox specification, it does provide confirmation of the degree of accountability for heavy metals loaded into the TMI canisters. The total uranium of an intact fuel assembly represents a greater uranium loading than that shown in Table 1; this offers further support that using the fissile loading of an intact assembly represents a conservative basis for configuration modeling.

3.1.2.3 Other Materials. Other materials used in construction of the commercial PWR fuel assemblies include stainless steels (304 or 304L or 316), Zircaloy-4, and Inconel 718. Quantity and mass of specific items for this fuel assembly hardware are detailed in Appendix B, Table B-1.

The use of any of these materials in the reactor core is selected for their serviceability in a reactor environment and neutronic neutrality. Within a fuel canister, they would have little effect on the system neutronics other than the displacement of moderator, whether intact or degraded. For the knockout canister, the assumption for system modeling ignored the presence of such materials. Rather it was assumed the expected materials within the knockout canister would be the loose fuel pellets. These pellets were transported to the canister during the hydraulic cleanout phase of the reactor vessel.

Table 1. GPU reported heavy metal inventory.

	Total for the Core[a] (kg)	Calculated Canister Average[b] (kg)
U (total)	81,437.393	238.82
U-235	1,801.131	5.28
Pu (total)	158.202	0.464
Pu-239–241	147.784	0.433
U + Pu (total) heavy metal	81,595.595	239.28

a. See Reference 4.

b. Based on a TMI-2 canister count of 341 in dry storage.

3.2 TMI Defueling Canisters

The TMI defueling operations employed three types of canisters to facilitate removal and storage of core materials that were damaged during the thermal excursion. These canisters were standardized relative to external dimensions (length, diameter) and features (connections, lifting points). However, the internal design features differed between canisters because of the functions they were expected to perform.

The TMI-2 reactor had a core that consisted of 177 assemblies. The number of TMI-2 canisters created as a result of debris removal from the degraded reactor core generated a canister count of 344. These canisters contain the fuel from the assemblies, but also assembly end-fittings, support posts, in-core guide tubes, grid supports, nozzles, and the remnants of the core structure. The canisters generated as a result of reactor core cleanup consist of three types: fuel (debris), filter, and knockout canisters. While differing in use and application, the external dimensions of all TMI-2 canisters were the same.

Defueling operations of the TMI-2 core resulted in the filling of 343 canisters of all types (see Figure 2), each with their own particular designation and numerical code. Subsequent consolidation of two of these canisters (see Reference 5) at the INEEL resulted in shipment of 341 canisters from the wet

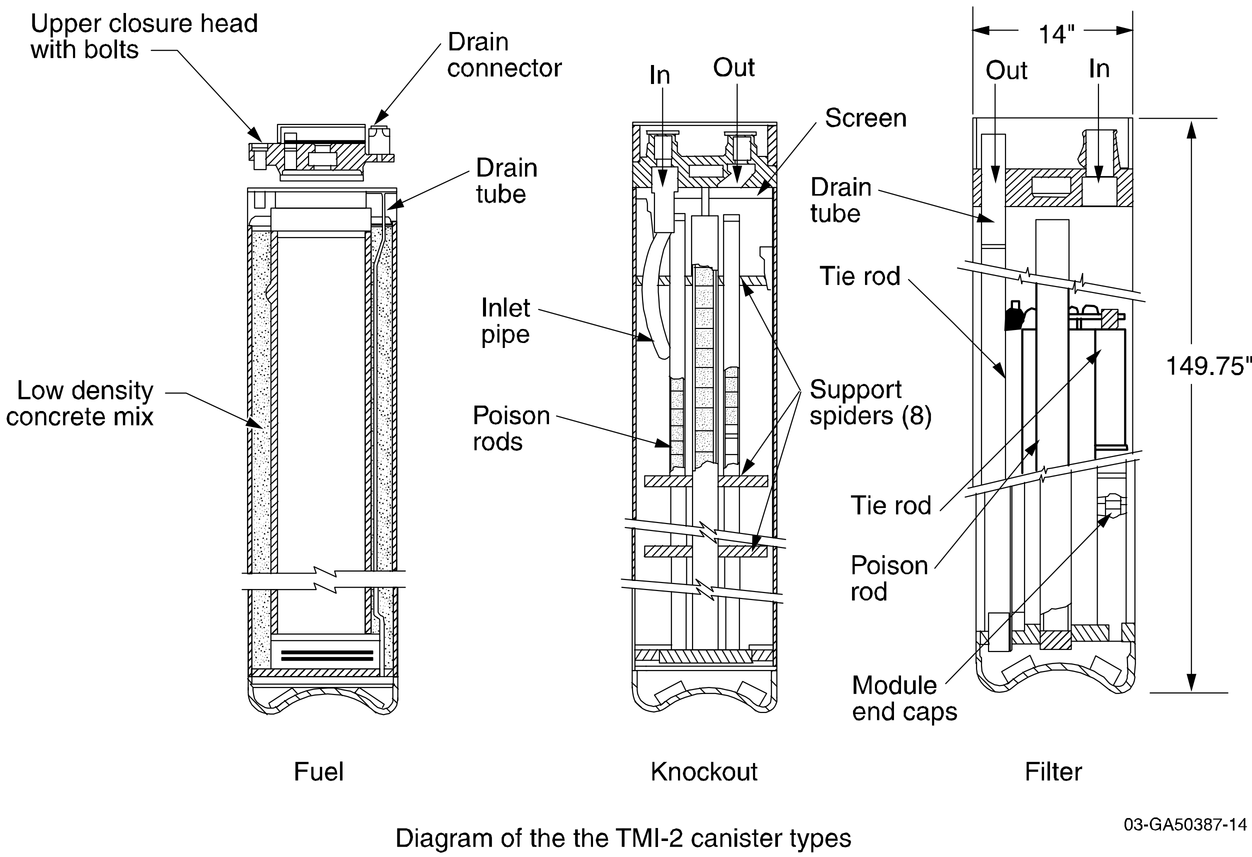

Figure 2. Schematic cross section of TMI-2 canister types.

storage pools at Test Area North to the INTEC dry storage facility. Two TMI canisters remain in storage at Test Area North.

The basic structure of the TMI canister centers on a 14-in. Schedule 10 pipe (355.6-mm outer diameter and 6.35-mm wall thickness). The bottom of the canister is a reversed dish head with a 9.525-mm (0.375-in.) thickness. The top of the canister is a 101.6-mm (4-in.) thick metal plate with penetrations and hardware connections suitable for hydraulic loading and dewatering. All structural canister materials used 304L stainless steel for canister construction (see Reference Drawings 1–9).

3.2.1 Fuel Canister Construction Details

The D-type canisters have a sleeve structure located in the center of the canister that has internal dimensions of 231.78 mm (9.125 in.) square and a useable length of 3,465.51 mm (136-7/16 in.) that will accommodate a full-size TMI PWR element (see Reference 3). The chord sections between the internal sleeve and the inside of the TMI canister are filled with LiCon™ (see Reference 2).

The overall height of the fuel canister is 3,803.65 mm (149-3/4 in.). The outer diameter of the canister is 355.6 mm (14 in.) and a wall thickness of 6.35 mm (0.25 in.) as determined by the use of 14-in. Schedule 10 pipe for the outer shell (see Reference 1).

3.2.2 Fuel Canister

The fuel canisters with the "D" prefix comprise 268 of the 341 total canisters currently in dry storage. These canisters comprise the bulk of core materials that were of a size that enabled grappling, could be picked up, or remotely handled. Such debris could be represented by nearly intact fuel assemblies all the way down to shards, metal remnants, end fixtures, pieces of the core support structure, etc.

Materials of construction employ a variety of the 300 series stainless steels, and those in turn are predominated by the use of 304L type stainless. The internal sleeve is a seal-welded sandwich of 304L stainless steel that completely encompasses the Boral™ (boron aluminide) layers (see Reference 10).

3.2.2.1 Materials Loading. The heavy metals content of each fuel canister is detailed in Appendix C. The total uranium values range from a maximum of 402.7 $\pm$ 42.7 kg in the D-330 canister down to 0.0 kg for the D-139 and D-160 canisters.

3.2.3 Knockout Canister

The "K" prefix relates to the knockout canisters that were used in association with wet-vacuuming operations used to remove loose debris that did not lend itself to physical grappling. The quantity of knockout canisters totals 12, one of which reportedly contains the highest total uranium loading of any TMI-2 canister (Appendix C, Table C-1). Figure 3 depicts the details of the knockout canister.

3.2.3.1 Knockout Canister Construction Details. The canisters with the "K" prefix have large internal diameters over which fuel matrix material is not constrained as it is in the D-type canisters. The internal assembly is designed to support five internal tubes with boron carbide (B_4C) poisoning. The larger center tube consists of a 73.03-mm (2.875-in.) diameter tube with a 7.92-mm (0.312-in.) wall thickness and a length of 3,371.85 mm (132.75 in.). The large center "A" tube has an internal tube diameter of 53.98 mm (2.125 in.) with a 1.60-mm (0.063-in.) wall thickness and is filled with boron carbide (B_4C) pellets. The four outer "B" rods are centered approximately 63.50 mm (2.50 in.) in the X-Y plane from the vessel centerline. These peripheral tubes have a 33.35-mm (1.313-in.) outer diameter with a 6.35-mm (0.25-in.) wall thickness with a minimum length of 3,327.4 mm (131 in.). The seven intermediate support plates are held in place by the poison rods, and the plates are spaced approximately 406.4 mm (16 in.) apart with a 12.7-mm (0.5-in.) plate thickness. The single, bottom support plate has a 340.52-mm (13-13/32-in.) diameter and a 31.75-mm (1.25-in.) thickness (see Reference 1 and Reference Drawings 1–9).

Once again, most of the materials of construction employ 300 series stainless steels, and all of them are expected to perform similarly given the same corrosion conditions.

3.2.3.2 Materials Loading. The heavy metals content of each knockout canister is detailed in Appendix C. The total uranium values range from a maximum of 441.9 $\pm$ 99.9 kg in the K-506 canister down to 0.0 kg for the K-523 canister.

3.2.4 Filter Canister

Subsequent and downstream collection of the vacuumed debris stream then passed through a filter canister with an "F" designator. These filter canisters represent the least heavily loaded for either total uranium or U-235 per package and, in several cases, are reported to have no uranium contained within. There were 61 such filter canisters transferred to the dry storage inventory at INTEC.

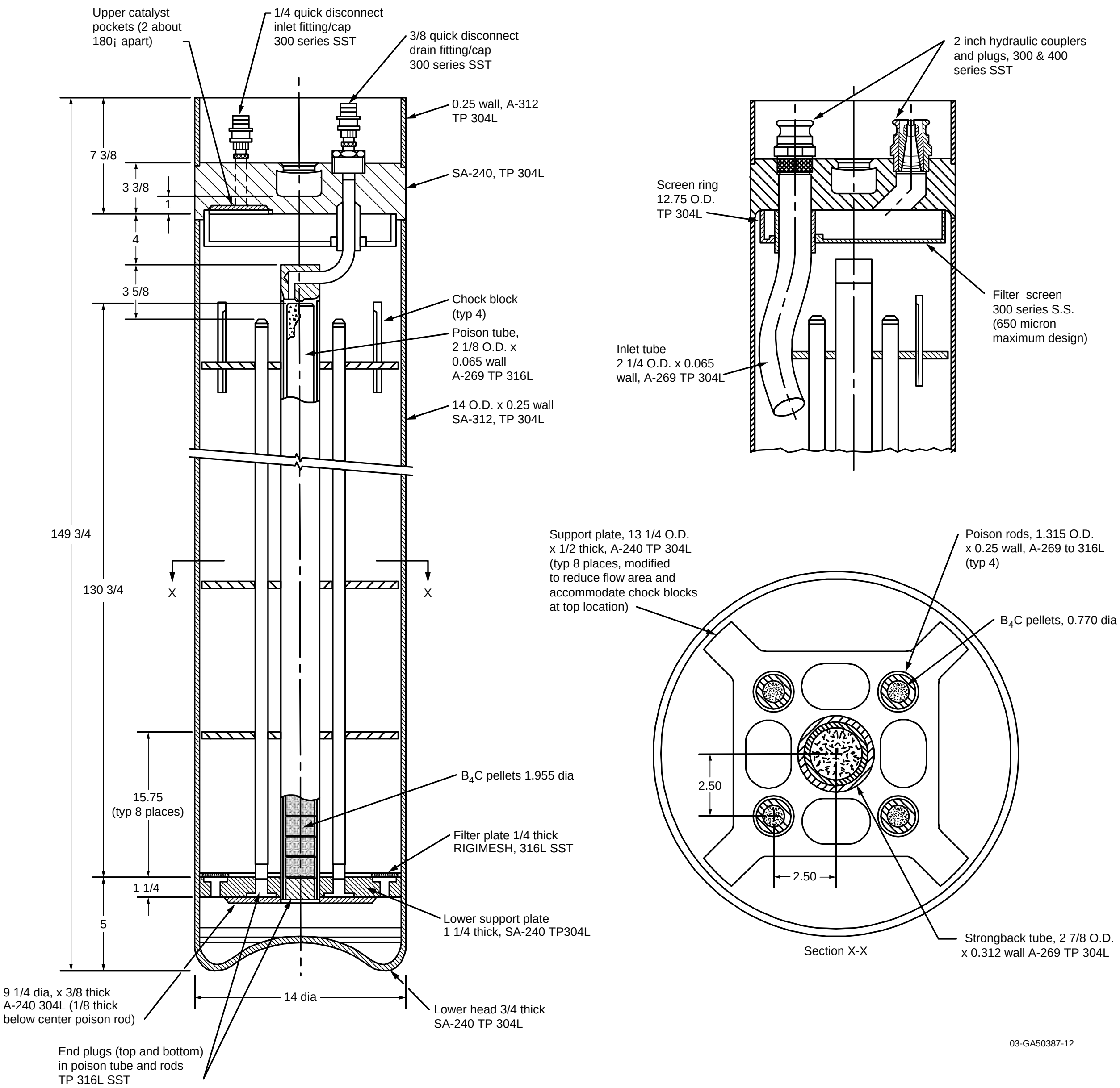

Figure 3. Composite drawing of knockout canister details.

3.2.4.1 Filter Canister Construction Details. The outer dimensions (length and diameter) of the filter canister are maintained consistent to those associated with the fuel and knockout canisters. The external materials of construction also use 304L stainless steel (see Reference 1).

Internals of the filter canister were designed to separate entrained particles and particulate fines from the hydraulic stream used to hydro-vacuum the TMI-2 core. Regardless of the canister internal details, the free volume inside the filter canister remains essentially the same as the knockout canister if the internals of the knockout canister are ignored. The total uranium load inside any filter canister (maximum) is only 46% of a PWR assembly and only 48.8% of the total uranium reported for the K-506 knockout canister.

3.2.4.2 Filter Canister Materials Loading. The heavy metals content of each filter canister is detailed in Appendix C. The total uranium values range from a maximum of 215.60 ± 48.7 kg in the F-471 canister down to 0.0 kg for various filter canisters.

3.3 Core Debris

As a result of the debris removal from the TMI-2 core, part of that work provided estimates of the various quantities of special nuclear material (SNM) in each canister. General Public Utilities (GPU) provided their own set of fuel assembly characteristics that show minor dimensional variations from numbers quoted in a Babcock & Wilcox fuel specification. Both sets of values are provided in Appendix B for comparison purposes.

A set of postirradiation calculations attempted to allocate an amount of SNM to each canister based on variables such as core position (determines enrichment within an assembly), percentage of fuel assembly installed in a canister, quantity of debris (inches of depth), etc. The following details of how these values were determined and the assumptions made to account for canister weights and provide the distributions of heavy metals are covered in the following discussions.

There was a basic set of data associated with both tare and shipping weights of the TMI-2 canisters. To facilitate the following analysis, the canister contents were divided into three classes, as follows:

- Recognized fuel assemblies, which were known lengths of more or less intact fuel assemblies. Most of these were identifiable so that their initial enrichment (2.96%, 2.64%, or 1.98%) was known.
- Structural materials, which was any piece of material with a weight estimate and which contained no SNM except perhaps as surface contamination. Structural material included such items as fuel assembly end fittings (with no active fuel attached) and spiders.
- Debris, which was any other core material in the canister. Debris included parts of fuel pins, parts of structural material, pieces of resolidified molten material, chips from drilling or cutting operations, and other material.

Reported materials loading of individual canisters came from the Fuel Canister Loading data sheets supplied by GPU for the D-type canisters (see Reference 4). These fuel canister loading data sheets were the source of the estimated lengths of more or less intact fuel assemblies (2.96%, 2.64%, and 1.98% and unknown enrichments). The fuel assembly lengths were obtained by subtracting the canister fill depths before and after loading the fuel assembly. The counts of structural components were based on notations for the canister loading sheets.

The total amount of uranium in the TMI-2 core was estimated to be 82,945 kg (see Reference 4). An ORIGEN calculation gave predictions of the amounts of each type of SNM in the TMI-2 core.[11] These values were scaled by a factor of 1.017 in the Lassahn report to make the total uranium inventory agree with that given by Lassahn (see Reference 4). The values for total amounts of each type of SNM in the TMI-2 reactor were used with the known values of the total fuel assembly lengths for each of the three enrichment values. This was done to obtain estimates of the amount of each type of SNM per length of fuel assembly for each enrichment. These values from both Lassahn and from Schnitzler were used only to estimate mass of SNM per length of fuel assembly. They were not used to estimate the amount of any type of material shipped from TMI-2 to the INEEL. The total core inventory values from both Akers and from Schnitzler and Briggs are expected to be a little different from the amount shipped.

The weights of the fuel assembly structural components are presented in Table 2 (see Reference 4).

A review of the data used in past analyses (criticality concerns at time of canister loading, shipment of multiple canisters both across the country and between INEEL facility locations, and multicanister dry storage in an NRC-approved facility) relies on GPU-supplied uranium loadings (per canister). From shipping data (see Reference 4), the five canisters with the highest reported debris weights are shown in Table 3. These data report both a total debris and uranium mass at the time of packaging.

The reported masses have published “plus/minus” limits applied to each of the canisters, although all past criticality analyses (see References 1, 2, and 3) have used the stated weight without any plus or minus bias. However, the maximum fissile uranium reported in any package (including the uncertainty values for U-235 specific to any specific canister) is less than that shown in Table 4 for an intact assembly at 2.96% enrichment. The adoption of the specified new, intact assembly (see Table 4 for total uranium and U-235 values) should be considered the most conservative values for criticality modeling (see Reference 4).

The PWR assembly specification (see Appendix A) for the typical 15 × 15 PWR assembly stipulates that there is 463.63 kg (1,022.12 lb) of uranium per fuel assembly. This value of uranium mass in a typical intact assembly equates to a U-235 mass of 13.72 kg (30.25 lb) at 2.96% enrichment. Neither the total uranium mass nor the U-235 mass reported or any TMI-2 canister exceeds those masses in an intact, BOL PWR assembly (see Table 3). Any total uranium mass value accounts for the U-238 used in criticality calculation in terms of increased parasitic capture. None of the reported total U-235 for

Table 2. Weights for other fuel assembly components.

Description	Weight (lb)
Fuel assembly top end fitting	22.6 ± 4.4
Fuel assembly bottom end fitting	22.6 ± 4.4
Fuel assembly fitting	22.6 ± 5.7
Spider	6.5 ± 0.6
Support post	17.4 ± 2.2
In-core guide tube	6.6 ± 0.7

Table 3. TMI maximum debris mass[a] (by canister ID number).

Canister ID Number	Five Highest Debris Masses (kg)	Corresponding Uranium Mass (kg)
K506	842.18 ± 168.3	441.90 ± 99.90
D330	767.35 ± 9.07	402.70 ± 42.70
D331	761.00 ± 9.07	399.30 ± 42.30
D361	760.09 ± 9.07	398.80 ± 42.30
D332	575.37 ± 9.07	397.40 ± 42.10

a. Data extracted from Appendix C.

Table 4. TMI maximum uranium (both total uranium and U-235) and plutonium mass.

Canister ID Number	Nine Highest Uranium Masses (kg)[a]	Interim Storage Uranium Mass (kg)[b]	Nine Highest U-235 Masses (kg)[a]	Interim Storage U-235 Mass (kg)[b]	Total Pu (kg)	Corresponding Debris Mass (kg)
K506	441.9 ± 99.9[c]	441.90	9.42 ± 2.13[d]	9.42	0.900	842.18 ± 168.3
D330	402.7 ± 42.7[c]	402.70	8.58 ± 0.92	8.58	0.820	767.35 ± 9.1
D283	400.5 ± 103[c]	400.50	7.57 ± 1.95	7.57	0.898	0 ± 0
D331	399.3 ± 42.3[c]	399.30	8.51 ± 0.91	8.51	0.813	761.00 ± 9.1
D361	398.8 ± 42.3[c]	398.80	8.50 ± 0.91	8.50	0.812	760.09 ± 9.1
D119	376.5 ± 97.3	376.50	10.06 ± 2.6[d]	10.06	0.539	0 ± 0
D260	373.8 ± 96.4	373.80	9.66 ± 2.42[d]	9.66	0.571	115.64 ± 102.9
D299	353.9 ± 104.0	353.90	9.37 ± 2.60[d]	9.37	0.516	0 ± 0
D193	351.9 ± 93.0	351.90	9.41 ± 2.49[d]	9.41	0.504	0 ± 0
New PWR assembly[e]	463.63	—	13.72	—	—	—

a. Reported masses at time of shipment from TMI to the INEEL.

b. Masses reported for TMI-2 canisters transferred to interim, dry storage; these values provide the basis for criticality safety analysis for up to 12 TMI-2 canisters (per position) in NRC-licensed dry storage.

c. Five highest uranium masses.

d. Five highest U-235 masses.

e. Comparison masses associated with an intact, beginning-of-life fuel loading.

any of the most highly loaded canisters exceeds the maximum U-235 value based on an assembly with BOL 2.96% enrichment (463.55 * 0.0296 = 13.72 kg U-235). The D119 canister with its reported U-235 loading of 10.06 ±2.49 kg U-235 still does not exceed the specification basis; fissile load represents only 92.3% of a new assembly with 2.96% enrichment. However, the total U-235 reported for K506 still falls below the 13.72 kg U-235 for a new assembly because of lower enrichment. Therefore, use of an intact 15 × 15 Babcock & Wilcox PWR assembly with 2.96% enriched BOL uranium mass values will exceed both total uranium and U-235 content used in prior criticality analyses for any of the most highly loaded TMI-2 fuel canisters. Representation of the BOL PWR assembly uranium masses within the partitioned volume of a TMI "D" canister would represent the highest fissile loading within a canister.

The internal volume within the fuel canister is a square cross section of 228.60-mm (9.0-in.). The internal length of the cavity is 3,465.51 mm (136-7/16 in.). These values result in a total volume of 181,100.48 cm^3 (11,051.4 $in.^3$) in which to distribute the 463.63 kg (1,022.12 lb) of heavy metal within the canister. The displaced volume created by the fuel pellets, based on 208 filled rods with 325 pellets per rod equals 51,420 cm^3 (3,137.28 $in.^3$) or 27.6% of the filled volume. This equates to 72.4% void volume within the canister if cladding and structural materials, and canister internals are neglected or ignored. Addition of inert materials will only further decrease the available volume for moderator. The nature of the core debris as it was loaded into the individual canisters suggests that just the fuel matrix pellets without cladding would constitute the bounding maximum reactivity case for a canister fuel load. Furthermore, a comparison of uranium masses reported in the TMI canisters versus those of an intact PWR assembly buttresses the argument of a bounding case analysis using a single PWR assembly.

While Boral™ was installed as an integral portion of the box liner for the fuel canister (see Reference 10), no credit will be taken for its presence. None of the other degraded, inert materials that may be present in the canister (zirconium or its oxides, stainless steel or its components) will be used to provide any credit for either poisoning effects or moderator displacement within the fuel compartment.

The fuel canister was specifically designed to hold either complete or partial fuel assemblies. The chord regions between the box liner and the canister inner wall were filled with the low density concrete mixture consisting of 60% Alcoa CA-25C refractory cement, 11% glass microspheres, and 29% water by weight. This mixture was intended to create a solid filler with an approximate density of 1.0 g/cm^3. Atom number density values used in the criticality analysis for this grout mix are shown in Table A-6.[12]

In a nonbreeder reactor, historical experience indicates that there will not be as much of an increase in plutonium concentrations as there is a decrease in U-235.[13] Fissile plutonium quantities can be offset by corresponding decreases in U-235 mass; the plutonium would be modeled as a fissile species with its own atom-density representative of the particular fissile isotope concentration. There may be some standard ratio of Pu-239 and Pu-241 that can be apportioned given a combined accounting of the two isotopes produced in a commercial reactor.

The disrupted fuel started in the new assemblies as UO_2. Damage to the core as a result of the accident led to exposure of the fuel matrix to both high temperatures and potentially oxidizing conditions. Formation of an oxide species other than the UO_2 in the fuel matrix may be of scientific interest. However, presence of UO_3 or U_3O_8 (or other oxide forms of uranium) should have no significant consequence to the overall criticality calculation if the uranium atom-density is modeled correctly based on the uranium oxide density in the fuel pellet. The zirconium cladding may have undergone similar oxidation. But its presence in creating a model within the TMI-2 canister would be neutral in terms of contributing to criticality analysis and can, therefore, be ignored in the Monte Carlo N-Particle (MCNP) models. Any net effect for oxide forms of either material (zirconium or uranium) would be in their expansion, decreased atom-density, and moderator displacement.

There are predominant differences in modeling assumptions because of the different conditions expected in storage and transportation versus those expected in repository disposal. Some of those differences include:

- Waste package void flooding with full-density water (repository conditions) rather than partial fill with full-density in monitored environment (storage conditions)
- Model development to analyze for maximum reactivity in a single canister (repository conditions) rather than maximum reactivity based on maximum fuel loading in a multicanister array (storage conditions)
- Analysis of packages without poison due to prescribed solubility of boron in any package (repository conditions) versus calculations based on 75% retention of boron in a package in a controlled environment (storage conditions for finite time period.).

3.3.1 Core Burnup

Reported value for the average core burnup may vary slightly from source to source. One reported value with supporting documentation provided an average 3,175 MWD/MTU.[14] This burnup represents ~10% of the typical burnup experienced by a standard PWR assembly. As such, the average heat generation expected of any TMI canister will be ~20 watts (see Reference 10). The following core maps (Figures 4 through 11) showing enrichment and calculated burnup by position (see Reference 11).

					1 A-6 2.96	2 A-6 2.96	3 A-8 2.96	4 A-9 2.96	5 A-10 2.96					
			6 B-4 2.96	7 B-5 2.96	8 B-6 2.96	9 B-7 2.64	10 B-8 2.96	11 B-9 2.64	12 B-10 2.96	13 B-11 2.96	14 B-12 2.96			
		15 C-3 2.96	16 C-3 2.96	17 C-3 1.98	18 C-3 2.64	19 C-3 1.98	20 C-3 2.64	21 C-3 1.98	22 C-3 2.64	23 C-3 1.98	24 C-3 2.96	25 C-3 2.96		
	26 D-2 2.96	27 D-3 2.96	28 D-4 1.98	29 D-5 2.64	30 D-6 1.98	31 D-7 2.64	32 D-8 1.98	33 D-9 2.64	34 D-10 1.98	35 D-11 2.64	36 D-12 1.98	37 D-13 2.96	38 D-14 2.96	
	39 E-2 2.96	40 E-3 1.98	41 E-4 2.64	42 E-5 1.98	43 E-6 2.64	44 E-7 1.98	45 E-8 2.64	46 E-9 1.98	47 E-10 2.64	48 E-2 1.98	49 E-2 2.64	50 E-2 1.98	51 E-14 2.96	
52 F-1 2.96	53 F-2 2.96	54 F-3 2.64	55 F-4 1.98	56 F-5 2.64	57 F-6 1.98	58 F-7 2.64	59 F-8 1.98	60 F-9 2.64	61 F-10 1.98	62 F-11 2.64	63 F-12 1.98	64 F-13 2.64	65 F-14 2.96	66 F-15 2.96
67 G-1 2.96	68 G-2 2.64	69 G-3 1.98	70 G-4 2.64	71 G-5 1.98	72 G-6 2.64	73 G-7 1.98	74 G-8 2.64	75 G-9 1.98	76 G-10 2.64	77 G-11 1.98	78 G-12 2.64	79 G-13 1.98	80 G-14 2.64	81 G-15 2.96
82 H-1 2.96	83 H-2 2.96	84 H-3 2.64	85 H-4 1.98	86 H-5 2.64	87 H-6 1.98	88 H-7 2.64	89 H-8 1.98	90 H-9 2.64	91 H-10 1.98	92 H-11 2.64	93 H-12 1.98	94 H-13 2.64	95 H-14 2.96	96 H-15 2.96
97 K-1 2.96	98 K-2 2.64	99 K-3 1.98	100 K-4 2.64	101 K-5 1.98	102 K-6 2.64	103 K-7 1.98	104 K-8 2.64	105 K-9 1.98	106 K-10 2.64	107 K-11 1.98	108 K-12 2.64	109 K-13 1.98	110 K-14 2.64	111 K-15 2.96
112 L-1 2.96	113 L-2 2.96	114 L-3 2.64	115 L-4 1.98	116 L-5 2.64	117 L-6 1.98	118 L-7 2.64	119 L-8 1.98	120 L-9 2.64	121 L-10 1.98	122 L-11 2.64	123 L-12 1.98	124 L-13 2.64	125 L-14 2.96	126 L-15 2.96
	127 M-2 2.96	128 M-3 1.98	129 M-4 2.64	130 M-5 1.98	131 M-6 2.64	132 M-7 1.98	133 M-8 2.64	134 M-9 1.98	135 M-10 2.64	136 M-11 1.98	137 M-12 2.64	138 M-13 1.98	139 M-14 2.96	
	140 N-2 2.96	141 N-3 2.96	142 N-4 1.98	143 N-5 2.64	144 N-6 1.98	145 N-7 2.64	146 N-8 1.98	147 N-9 2.64	148 N-10 1.98	149 N-11 2.64	150 N-12 1.98	151 N-13 2.96	152 N-14 2.96	
		153 O-3 2.96	154 O-4 2.96	155 O-5 1.98	156 O-6 2.64	157 O-7 1.98	158 O-8 2.64	159 O-9 1.98	160 O-10 2.64	161 O-11 1.98	162 O-12 2.96	163 O-13 2.96		
			164 P-4 2.96	165 P-5 2.96	166 P-6 2.96	167 P-7 2.64	168 P-8 2.96	169 P-9 2.64	170 P-10 2.96	171 P-11 2.96	172 P-12 2.96		Element # Grid Location Enrichment	
					173 R-6 2.96	174 R-7 2.96	175 R-8 2.96	176 R-9 2.96	177 R-10 2.96					

Figure 4. TMI-2 element location and enrichment map.

					1 9 966	2 9 1031	3 9 1334	4 9 1032	5 9 967					
			6 9 912	7 9 1456	8 9 1724	9 5 1650	10 9 1832	11 5 1650	12 9 1724	13 9 1457	14 9 912			
		15 9 1003	16 9 1416	17 1 1589	18 5 1848	19 1 1872	20 5 1974	21 1 1872	22 5 1914	23 1 1590	24 9 1417	25 9 1003		
	26 9 912	27 9 1417	28 1 1580	29 5 1804	30 1 1781	31 5 2072	32 1 1708	33 5 2072	34 1 1781	35 5 1804	36 1 1581	37 9 1417	38 9 912	
	39 9 1457	40 1 1597	41 5 1908	42 1 1438	43 5 1885	44 1 2020	45 5 2153	46 1 1931	47 5 1885	48 1 1439	49 5 1909	50 1 1597	51 9 1457	
52 9 967	53 9 1727	54 5 1850	55 1 1736	56 5 2025	57 1 2042	58 5 2141	59 1 1802	60 5 2141	61 1 2042	62 5 2026	63 1 1737	64 5 1872	65 9 1727	66 9 967
67 9 1032	68 5 1699	69 1 1868	70 5 2060	71 1 2076	72 5 2184	73 1 2053	74 5 2105	75 1 2053	76 5 2184	77 1 1939	78 5 2060	79 1 1868	80 5 1699	81 9 1032
82 9 1334	83 9 1832	84 5 1974	85 1 1708	86 5 2152	87 1 1801	88 5 2104	89 1 2057	90 5 2104	91 1 1801	92 5 2152	93 1 1707	94 5 1973	95 9 1832	96 9 1334
97 9 1032	98 5 1699	99 1 1868	100 5 2059	101 1 2002	102 5 2183	103 1 2053	104 5 2104	105 1 2052	106 5 2183	107 1 1977	108 5 2059	109 1 1867	110 9 1698	111 9 1031
112 9 967	113 9 1726	114 5 1789	115 1 1735	116 5 2025	117 1 2040	118 5 2139	119 1 1800	120 5 2139	121 1 2040	122 5 2024	123 1 1734	124 5 1862	125 9 1725	126 9 966
	127 9 1456	128 1 1596	129 5 1907	130 1 1437	131 5 1883	132 1 1992	133 5 2151	134 1 1970	135 5 1882	136 1 1436	137 5 1906	138 9 1595	139 9 1455	
	140 9 911	141 9 1416	142 1 1579	143 5 1802	144 1 1779	145 5 2069	146 1 1705	147 5 2069	148 1 1779	149 5 1801	150 1 1578	151 9 1415	152 9 911	
		153 9 1002	154 9 1415	155 1 1587	156 5 1787	157 1 1869	158 5 1971	159 1 1869	160 5 1752	161 1 1587	162 9 1414	163 9 1002		
			164 9 911	165 9 1455	166 9 1722	167 5 1648	168 9 1829	169 5 1647	170 9 1721	171 9 1454	172 9 910		Element # Fuel Group MWd/MT	
					173 9 965	174 9 1029	175 9 1332	176 9 1029	177 9 964					

Figure 5. TMI-2 burnup for axial Level 1 of 7.

					1 10 2186	2 10 2260	3 10 3050	4 10 2260	5 10 2186					
			6 9 1807	7 10 2975	8 11 3854	9 6 3649	10 12 4507	11 6 3649	12 11 3854	13 10 2975	14 9 1807			
		15 9 1973	16 10 2741	17 3 3219	18 7 3928	19 3 3801	20 7 4101	21 3 3801	22 6 3864	23 3 3220	24 10 2742	25 9 1973		
	26 9 1807	27 10 2741	28 2 3158	29 6 3806	30 3 3964	31 7 4060	32 3 3347	33 7 4060	34 3 3964	35 6 3807	36 2 3158	37 10 2742	38 9 1807	
	39 10 2975	40 3 3251	41 6 3879	42 3 3644	43 7 4121	44 3 3929	45 7 4161	46 3 3925	47 7 4121	48 3 3644	49 6 3880	50 3 3251	51 10 2975	
52 10 2186	53 11 3917	54 6 3845	55 3 3902	56 7 4036	57 3 4021	58 7 4212	59 3 3845	60 7 4212	61 3 4021	62 7 4036	63 3 3902	64 6 3883	65 11 3917	66 10 2186
67 10 2260	68 6 3587	69 3 3838	70 7 4078	71 3 3915	72 7 4237	73 4 4089	74 7 4284	75 4 4089	76 7 4237	77 3 3888	78 7 4077	79 3 3838	80 6 3587	81 10 2260
82 10 3050	83 12 4705	84 7 4100	85 3 3346	86 7 4160	87 3 3844	88 7 4283	89 7 4799	90 7 4283	91 3 3844	92 7 4160	93 3 3346	94 7 4100	95 12 4507	96 10 3050
97 10 2260	98 6 3587	99 3 3838	100 7 4077	101 3 3911	102 7 4236	103 4 4088	104 7 4283	105 4 4088	106 7 4236	107 3 3952	108 7 4076	109 3 3837	110 6 3586	111 10 2259
112 10 2186	113 11 3916	114 6 3792	115 3 3901	116 7 4034	117 3 4019	118 7 4210	119 3 3843	120 7 4209	121 3 4019	122 7 4033	123 3 3899	124 6 3775	125 11 3915	126 10 2186
	127 10 2974	128 3 3249	129 6 3878	130 3 3642	131 7 4118	132 3 3943	133 7 4157	134 3 3986	135 7 4117	136 3 3641	137 6 3877	138 3 3249	139 10 2973	
	140 9 1806	141 10 2740	142 2 3155	143 6 3803	144 3 3960	145 7 4056	146 3 3342	147 7 4055	148 3 3960	149 6 3802	150 2 3154	151 10 2738	152 9 1805	
		153 9 1971	154 10 2739	155 3 3216	156 6 3805	157 3 3797	158 7 4096	159 3 3796	160 6 3756	161 3 3216	162 10 2738	163 9 1971		
			164 9 1805	165 10 2972	166 11 3850	167 6 3645	168 12 4503	169 6 3645	170 11 3850	171 10 2971	172 9 1804		Element # Fuel Group MWd/MT	
					173 10 2183	174 10 2256	175 10 3046	176 10 2256	177 10 2183					

Figure 6. TMI-2 burnup for axial Level 2 of 7.

					1 10 2632	2 10 3053	3 11 3738	4 10 3054	5 10 2632					
			6 10 2107	7 11 3401	8 12 4603	9 7 4387	10 12 5551	11 7 4387	12 12 4603	13 11 3402	14 10 2108			
		15 10 2220	16 10 3142	17 3 3630	18 7 4342	19 4 4370	20 7 4832	21 4 4371	22 7 4303	23 3 3630	24 10 3143	25 10 2220		
	26 10 2108	27 10 3142	28 3 3560	29 7 4188	30 3 3862	31 7 4480	32 3 3920	33 7 4480	34 3 3862	35 7 4189	36 3 3560	37 10 3143	38 10 2108	
	39 11 3402	40 3 3604	41 7 4127	42 4 4197	43 7 4589	44 4 4421	45 7 4774	46 4 4485	47 7 4589	48 4 4198	49 4 4128	50 3 3604	51 11 3402	
52 10 2632	53 12 4666	54 7 4301	55 3 3794	56 7 4470	57 4 4559	58 7 4933	59 4 4675	60 7 4933	61 4 4559	62 7 4470	63 3 3794	64 7 4267	65 12 4667	66 10 2633
67 10 3053	68 7 4276	69 4 4395	70 7 4494	71 4 4393	72 7 4921	73 4 4836	74 8 5236	75 4 4836	76 7 4921	77 4 4439	78 7 4494	79 4 4395	80 7 4276	81 10 3053
82 11 3738	83 122 5550	84 7 4831	85 3 3919	86 7 4773	87 4 4674	88 8 5235	89 8 6009	90 8 5235	91 4 4674	92 7 4773	93 3 3919	94 7 4831	95 12 5550	96 11 3737
97 10 3053	98 7 4276	99 4 4394	100 7 4494	101 4 4418	102 7 4920	103 4 4835	104 8 5234	105 4 4834	106 7 4920	107 4 4487	108 7 4493	109 4 4394	110 7 4275	111 10 3053
112 10 2632	113 12 4666	114 7 4212	115 3 3792	116 7 4468	117 4 4557	118 7 4931	119 4 4673	120 7 4931	121 4 4557	122 7 4468	123 3 3792	124 7 4245	125 12 4665	126 10 2632
	127 11 3401	128 3 3602	129 7 4126	130 4 4195	131 7 4586	132 4 4452	133 7 4771	134 4 4473	135 7 4585	136 4 4195	137 7 4125	138 3 3602	139 11 3400	
	140 10 2107	141 10 3141	142 3 3558	143 7 4186	144 3 3859	145 7 4476	146 3 3916	147 7 4476	148 3 3858	149 7 4185	150 3 3557	151 10 3140	152 10 2106	
		153 10 2218	154 10 3141	155 3 3627	156 7 4279	157 4 4367	158 7 4828	159 4 4367	160 7 4237	161 3 3626	162 10 3139	163 10 2218		
			164 10 2106	165 11 3399	166 12 4599	167 7 4383	168 12 5547	169 7 4383	170 12 4599	171 11 3398	172 10 2105		Element # Fuel Group MWd/MT	
					173 10 2630	174 10 3050	175 11 3734	176 10 3050	177 10 2629					

Figure 7. TMI-2 burnup for axial Level 3 of 7.

					1 10 2617	2 11 3380	3 11 3794	4 11 3381	5 10 2618					
			6 10 2101	7 11 3347	8 12 4567	9 7 4384	10 12 5572	11 7 4385	12 12 4568	13 11 3348	14 10 2102			
		15 10 2170	16 10 3122	17 3 3478	18 7 4016	19 4 4264	20 7 4794	21 4 4264	22 7 4063	23 3 3479	24 10 3124	25 10 2171		
	26 10 2102	27 10 3123	28 3 3413	29 6 3818	30 2 2925	31 7 4224	32 3 3790	33 7 4225	34 2 2925	35 6 3819	36 3 3413	37 10 3124	38 10 2102	
	39 11 3347	40 3 3401	41 6 3712	42 2 3805	43 7 4207	44 4 4301	45 7 4723	46 4 4350	47 7 4208	48 3 3806	49 6 3713	50 3 3401	51 11 3348	
52 10 2618	53 12 4598	54 7 4035	55 2 2867	56 7 4195	57 4 4435	58 7 4936	59 4 4748	60 7 4936	61 4 4436	62 7 4195	63 2 2867	64 7 3984	65 12 4598	66 10 2618
67 11 3380	68 7 4324	69 4 4265	70 7 4208	71 4 4294	72 7 4952	73 4 4904	74 8 5421	75 4 4904	76 7 4952	77 4 4314	78 7 4208	79 4 4265	80 7 4324	81 11 3381
82 11 3794	83 12 5572	84 7 4793	85 3 3789	86 7 4722	87 4 4747	88 8 5421	89 8 6213	90 8 5421	91 4 4747	92 7 4722	93 3 3789	94 7 4793	95 12 5572	96 11 3794
97 11 3380	98 7 4324	99 4 4265	100 7 4208	101 4 4315	102 7 4951	103 4 4904	104 8 5421	105 4 4904	106 7 4951	107 4 4320	108 7 4207	109 4 4265	110 7 4324	111 11 3380
112 10 2618	113 12 4597	114 7 3927	115 2 2866	116 7 4193	117 4 4434	118 7 4934	119 4 4746	120 7 4934	121 4 4433	122 7 4193	123 2 2865	124 7 4046	125 12 4597	126 10 2618
	127 11 3347	128 3 3400	129 6 3711	130 3 3804	131 7 4205	132 4 4295	133 7 4720	134 4 4286	135 7 2.64	136 3 3804	137 6 3710	138 3 3400	139 11 3346	
	140 10 2101	141 10 3122	142 3 3411	143 6 3816	144 2 2922	145 7 4222	146 3 3787	147 7 4221	148 2 2922	149 6 3815	150 3 3411	151 10 3121	152 10 2101	
		153 10 2170	154 10 3121	155 3 3476	156 7 4010	157 4 4261	158 7 4790	159 4 4261	160 7 3982	161 3 3476	162 10 3120	163 10 2169		
			164 10 2100	165 11 3345	166 12 4565	167 7 4382	168 12 5569	169 7 4381	170 12 4565	171 11 3344	172 10 2100		Element # Fuel Group MWd/MT	
					173 10 2616	174 11 3378	175 11 3791	176 11 3378	177 10 2615					

Figure 8. TMI-2 burnup for axial Level 4 of 7.

					1 10 2522	2 11 3321	3 11 3717	4 11 3321	5 10 2523					
			6 9 2020	7 11 3295	8 12 4455	9 7 4264	10 12 5500	11 7 4264	12 12 4456	13 11 3296	14 9 2021			
		15 10 2131	16 10 3040	17 3 3361	18 7 3906	19 4 4106	20 7 4611	21 4 4106	22 7 3296	23 3 3362	24 10 3041	25 10 2131		
	26 9 2020	27 10 3040	28 3 3265	29 6 3645	30 2 2855	31 7 4067	32 3 3401	33 7 4067	34 2 2855	35 6 3646	36 3 3266	37 10 3041	38 9 2021	
	39 11 3295	40 3 3310	41 6 3612	42 3 3635	43 7 4049	44 4 4167	45 7 4529	46 4 4186	47 7 4050	48 3 3636	49 6 3612	50 3 3311	51 11 3295	
52 10 2523	53 12 4479	54 6 3890	55 2 2829	56 7 4024	57 4 4263	58 7 4714	59 4 4572	60 7 4714	61 4 4263	62 7 4025	63 2 2830	64 7 3920	65 12 4479	66 10 2523
67 11 3321	68 7 4252	69 4 4133	70 7 4036	71 4 4147	72 7 4804	73 4 4739	74 8 5228	75 4 4739	76 7 4805	77 4 4145	78 7 4036	79 4 4133	80 7 4252	81 11 3321
82 11 3717	83 12 5500	84 7 4611	85 3 3400	86 7 4528	87 4 4572	88 8 5228	89 8 6117	90 8 5228	91 4 4572	92 7 4528	93 3 3400	94 7 4611	95 12 5500	96 11 3717
97 11 3321	98 7 4252	99 4 4132	100 7 4036	101 4 4185	102 7 4804	103 4 4738	104 8 5227	105 4 4738	106 7 4804	107 4 4121	108 7 4035	109 4 4132	110 7 4252	111 11 3321
112 10 2523	113 12 4478	114 6 3839	115 2 2828	116 7 4023	117 4 4261	118 7 4713	119 4 4571	120 7 4713	121 4 4261	122 7 4023	123 2 2828	124 7 3917	125 12 4478	126 10 2523
	127 11 3295	128 3 3309	129 6 3611	130 3 3634	131 7 4047	132 4 4124	133 7 4527	134 4 4157	135 7 4047	136 3 3634	137 6 3610	138 3 3309	139 11 3294	
	140 9 2020	141 10 3039	142 3 3264	143 6 3644	144 2 2853	145 7 4064	146 3 3398	147 7 4064	148 2 2852	149 6 3643	150 3 3264	151 10 3038	152 9 2020	
		153 10 2130	154 10 3039	155 3 3359	156 6 3872	157 4 4103	158 7 4609	159 4 4103	160 6 3864	161 3 3359	162 10 3038	163 10 2130		
			164 9 2019	165 11 3293	166 12 4453	167 7 4262	168 12 5498	169 7 4261	170 12 4453	171 11 3293	172 9 2019		Element # Fuel Group MWd/MT	
					173 10 2521	174 11 3319	175 11 3715	176 11 3319	177 10 2521					

Figure 9. TMI-2 burnup for axial Level 5 of 7.

					1 10 2290	2 10 2970	3 11 3410	4 10 2971	5 10 2290					
			6 9 1845	7 10 3075	8 11 4163	9 7 3966	10 12 5163	11 7 3966	12 11 4163	13 10 3075	14 9 1845			
		15 9 1996	16 10 2837	17 3 3194	18 6 3856	19 3 3840	20 7 4261	21 3 3840	22 6 3811	23 3 3194	24 10 2838	25 9 1997		
	26 9 1845	27 10 2837	28 2 3064	29 29 3615	30 3 3419	31 7 3939	32 2 3061	33 7 3940	34 3 3419	35 6 3615	36 2 3065	37 10 2838	38 9 1845	
	39 10 3075	40 3 3192	41 6 3625	42 3 3646	43 7 4014	44 3 3891	45 7 4147	46 3 3909	47 7 4014	48 3 3646	49 6 3625	50 3 3192	51 10 3075	
52 10 2290	53 11 4192	54 6 3790	55 3 3416	56 7 3921	57 3 3950	58 7 4274	59 4 4148	60 7 4274	61 3 3950	62 7 3921	63 3 3416	64 6 3854	65 11 4192	66 10 2290
67 10 2970	68 7 3949	69 3 3907	70 7 3925	71 3 3878	72 7 4384	73 4 4270	74 7 4678	75 4 4270	76 7 4385	77 3 3875	78 7 3925	79 3 3907	80 7 3949	81 10 2970
82 11 3409	83 12 5163	84 7 4260	85 2 3060	86 7 4146	87 4 4147	88 7 4678	89 8 5571	90 7 4678	91 4 4147	92 7 4146	93 2 3060	94 7 4260	95 12 5163	96 11 3409
97 10 2970	98 7 3949	99 3 3907	100 7 3925	101 3 3904	102 7 4384	103 4 4270	104 7 4677	105 4 4270	106 7 4384	107 3 3865	108 7 3924	109 3 3907	110 7 3949	111 10 2970
112 10 2290	113 11 4191	114 6 3796	115 3 3416	116 7 3920	117 3 3949	118 7 4273	119 4 4147	120 7 4273	121 3 3949	122 7 3919	123 3 3415	124 6 3784	125 11 4191	126 10 2290
	127 10 3075	128 3 3191	129 6 3624	130 3 3645	131 7 4012	132 3 3870	133 7 4145	134 3 3932	135 7 4012	136 3 3645	137 6 3623	138 3 3191	139 10 3074	
	140 9 1844	141 10 2837	142 2 3064	143 6 3613	144 3 3417	145 7 3937	146 2 3058	147 7 3937	148 3 3417	149 6 3612	150 2 3063	151 10 2836	152 9 1844	
		153 9 1996	154 10 2836	155 3 3192	156 6 3788	157 3 3838	158 7 4258	159 3 3838	160 6 3976	161 3 3192	162 10 2836	163 9 1995		
			164 9 1844	165 10 3074	166 11 4161	167 7 3964	168 12 5161	169 7 3964	170 11 4161	171 10 3073	172 9 1843		Element # Fuel Group MWd/MT	
					173 10 2288	174 10 2969	175 11 3408	176 10 2969	177 10 2288					

Figure 10. TMI-2 burnup for axial Level 6 of 7.

					1 9 1416	2 9 1984	3 10 2134	4 9 1984	5 9 1416					
			6 9 1205	7 9 1936	8 10 2681	9 5 2582	10 11 3263	11 5 2582	12 10 2682	13 9 1936	14 9 1205			
		15 9 1279	16 9 1895	17 1 2125	18 5 2591	19 2 2548	20 6 2811	21 2 2548	22 5 2612	23 1 2125	24 9 1895	25 9 1279		
	26 9 1205	27 9 1895	28 1 2039	29 5 2514	30 2 2582	31 5 2726	32 1 2240	33 5 2726	34 2 2582	35 5 2515	36 1 2039	37 9 1895	38 9 1205	
	39 9 1936	40 1 2107	41 5 2454	42 2 2489	43 5 2741	44 2 2547	45 5 2735	46 2 2569	47 5 2741	48 2 2489	49 5 2454	50 1 2108	51 9 1936	
52 9 1416	53 10 2683	54 5 2593	55 2 2569	56 5 2740	57 2 2586	58 6 2819	59 2 2689	60 6 2819	61 2 2586	62 5 2740	63 2 2569	64 5 2548	65 10 2683	66 9 1416
67 9 1984	68 5 2552	69 2 2594	70 5 2728	71 2 2606	72 6 2847	73 2 2735	74 6 3042	75 2 2734	76 6 2847	77 2 2578	78 5 2728	79 2 2593	80 5 2552	81 9 1984
82 10 2134	83 11 3262	84 6 2811	85 1 2240	86 5 2735	87 2 2688	88 6 3042	89 6 3464	90 6 3042	91 2 2688	92 5 2735	93 1 2240	94 6 2811	95 11 3262	96 10 2134
97 9 1984	98 8 2552	99 2 2593	100 5 2728	101 2 2553	102 6 2847	103 2 2734	104 6 3041	105 2 2734	106 6 2846	107 2 2603	108 5 2728	109 2 2593	110 5 2552	111 9 1984
112 9 1416	113 10 2683	114 5 2565	115 2 2568	116 5 2739	117 2 2585	118 6 2819	119 2 2688	120 6 2819	121 2 2585	122 5 2739	123 2 2568	124 5 2583	125 10 2683	126 9 1416
	127 9 1936	128 1 2107	129 5 2453	130 2 2488	131 5 2740	132 2 2580	133 5 2734	134 2 2567	135 5 2740	136 2 2488	137 5 2453	138 1 2107	139 9 1936	
	140 9 1204	141 9 1894	142 1 2038	143 5 2514	144 2 2581	145 5 2725	146 1 2240	147 5 2725	148 2 2581	149 5 2513	150 1 2039	151 9 1895	152 9 1204	
		153 9 1278	154 9 1894	155 1 2124	156 5 2589	157 2 2547	158 6 2810	159 2 2547	160 5 2593	161 1 2124	162 9 1894	163 9 1278		
			164 9 1204	165 9 1935	166 10 2680	167 5 2581	168 11 3261	169 5 2581	170 10 2680	171 9 1935	172 9 1204		Element # Fuel Group MWd/MT	
					173 9 1416	174 9 1983	175 10 2133	176 9 1983	177 9 1415					

Figure 11. TMI-2 burnup for axial Level 7 of 7.

3.3.2 Radionuclide Inventory

Radionuclide inventories for this commercial fuel inventory are significantly less than for standard, fully burned commercial fuels due to the limited time the fuel was in service before the reactor experienced a partial meltdown.

The information presented in Table D-1 represents two radionuclide values associated with the D-153 canister. Values for other canisters can be scaled from these values by the ratio of the heavy metal masses between canisters. These reported values are from template values calculated and assigned to certain TMI-2 canisters. The nominal burnup values are reported based on the calculated heavy metal mass destroyed. Bounding values are merely assumed to be twice the nominal values, whether as burnup, radionuclide curie inventory, or heat output.

3.3.2.1 ***Heat Output.*** On a per canister basis, the nominal thermal power for the D-153 canister is shown in Table D-1 as 2.0 watts per canister (nominal) and 3.99 watts (bounding). Scaleup using the same burnup percentage applied to an otherwise intact PWR assembly would use a multiplier of

$$(463.63 \text{ kg U} * \frac{2.0 \text{ W}}{19.08 \text{ kg U}} = 48.60 \text{ W}.$$

This canister is considered to be representative of the 341 canisters in storage.

3.3.2.2 Shielding Source Term. Gamma source information can be extracted from Table D-1 as a bounding case value. The same multiplier shown above can be used to adjust individual radionuclide curie values.

4. DOE STANDARD FUEL CANISTER INFORMATION

A standard canister design for U.S. Department of Energy (DOE) spent nuclear fuel (SNF) has been adopted to facilitate uniformity in terms of: handling at the repository, packaging the canister within the waste package, analyses (thermal, shielding, criticality), and qualification of a minimal number of designs. Details of this canister design can be found in a design specification published by the National Spent Nuclear Fuel Program.[15]

4.1 SNF Canister Characteristics

The standard SNF canister design is intended to provide fuel containment with a standard size for predictability for handling and loading operations within a waste package. SNF canister internals are varied internally with baskets to facilitate loading operations, and predictability of fuel positions for intact criticality analysis.

4.1.1 Dimensions

Overall length of the standard canister to be used for the three fuel types studied in this analysis is 4,569 mm (15 ft nominal). Available internal length shrinks to approximately 4,114.8 mm (13.5 ft) to accommodate the internal impact plate, the head closure, and the standoff ring (for impact absorption). Figure 12 depicts the DOE SNF standard canister.

4.1.2 Mass

Each component in a standard canister has an associated weight, dependent on the diameter and length. A summary of these weights follows in Table 5.

While the total weight of the heavy metals in each canister is important to the criticality analysis, it is the total weight of each TMI-2 canister that determines the combined allowable weight for each standard canister.

4.1.3 Materials of Construction

SNF canister materials are standardized around 300 series stainless steel with the exception of both the carbon steel impact plate and TMI-2 canister guide tube inside the standard canister. Figure 13 is a proposed layout of the sleeve inside the SNF canister used to position the TMI-2 canisters.

4.1.3.1 316L Stainless Steel. The stainless steel outer shell will use 316L stainless steel material; composition is as shown in Table A-1.

4.1.3.2 Carbon Steel (Internal Impact Plates). The curved, carbon steel impact plates fit inside each end of the canister, and vary from 15.24 mm (0.60 in.) to 50.8 mm (2.0 in.) in thickness. The chemical composition of these plates is shown in Table A-4.

4.1.4 Basket Internals

Conceptual basket internals for the canister would be intended as nothing more than a centering device for the TMI canister installation. Such a basket could take the form of a pipe sleeve, some form of positioning brackets, or be nothing more than expanded metal mesh. In each case, some form of standoff from the inner wall of the standard canister is designed to maintain a centered position of the sleeve. A sleeve constructed of pipe material (16-in. Schedule 10 carbon steel) would be the approach creating the maximum reflection. Such a configuration would necessarily create a 19.05-mm (3/4-in.) gap between the TMI canister and the sleeve. There would also be a corresponding 15.88-mm (5/8-in.) gap between the sleeve and the inside of the standard canister.

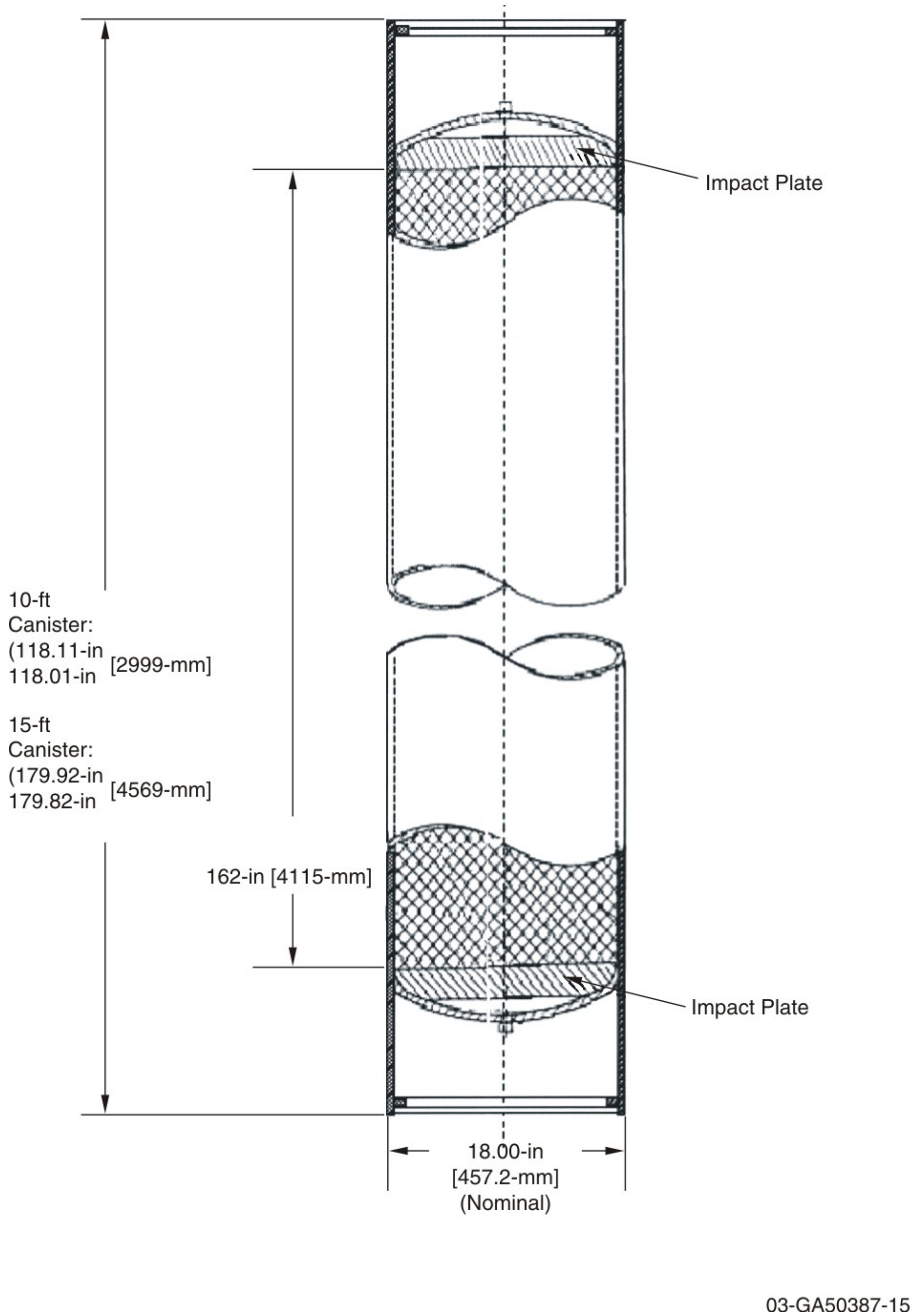

Figure 12. SNF standard canister.

Table 5. Empty spent nuclear fuel canister weights.

Length	Canister Body	Impact Plate	Sleeve	Total Canister Weight	Maximum Total Allowable Weight
10 ft	728 lb (330.1 kg)	182 lb (82.5 kg)	281 lb (127.4 kg)	1,191 lb (540.1 kg)	5,005 lb (2,270 kg)
15 ft	1,073 lb (486.6 kg)	182 lb (82.5 kg)	455 lb (206.3 kg)	1,710 lb (775.5 kg)	6,000 lb (2,721 kg)

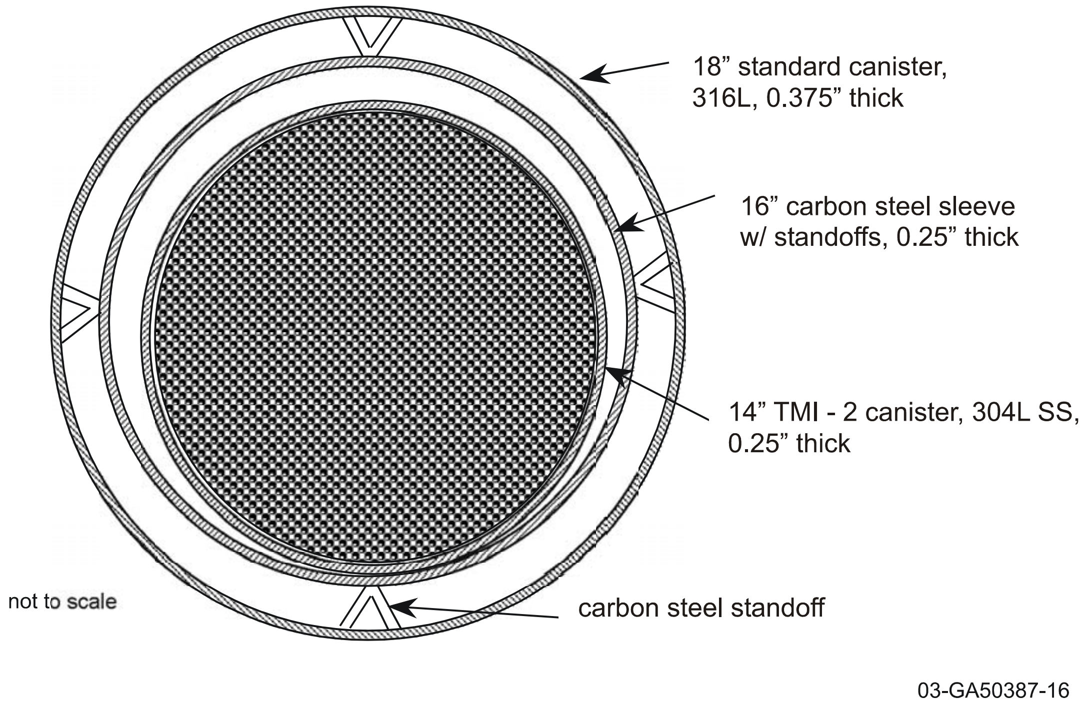

Figure 13. Conceptual cross-section sketch of TMI canister in standard SNF canister. (This drawing is not to scale.)

5. WASTE PACKAGE DESIGN

A standard waste package design to contain the DOE SNF canister has been adopted to facilitate uniformity in terms of: handling at the repository, packaging the canister for codisposal, analyses (thermal, shielding, criticality), and qualification of a minimal number of designs.

5.1 Codisposal Concept

The codisposal waste package is designed to accommodate five high-level waste (HLW) canisters surrounding a single SNF canister in the center position. Dimensions vary only in the length of the package needed to accommodate either the 10 or 15-ft HLW/DOE SNF canisters. Figure 14 depicts the cross-section view of the waste package and its internals.[16]

5.1.1 Waste Package Shell

The outside diameter of the waste package is 2,030 mm (79.92 in.), and the inside cavity length is 4,618 mm (181.8 in.), which is designed to accommodate Hanford 15-ft HLW glass canisters. The lids of the inner barrier are 105 mm (4.134 in.) thick; those of the outer barrier are 25 mm (0.984 in.) thick. There is a 30-mm (1.181-in.) gap between the inner and outer barrier upper lids. Each end of the waste package has a 225-mm-long (8.858-in.) skirt.

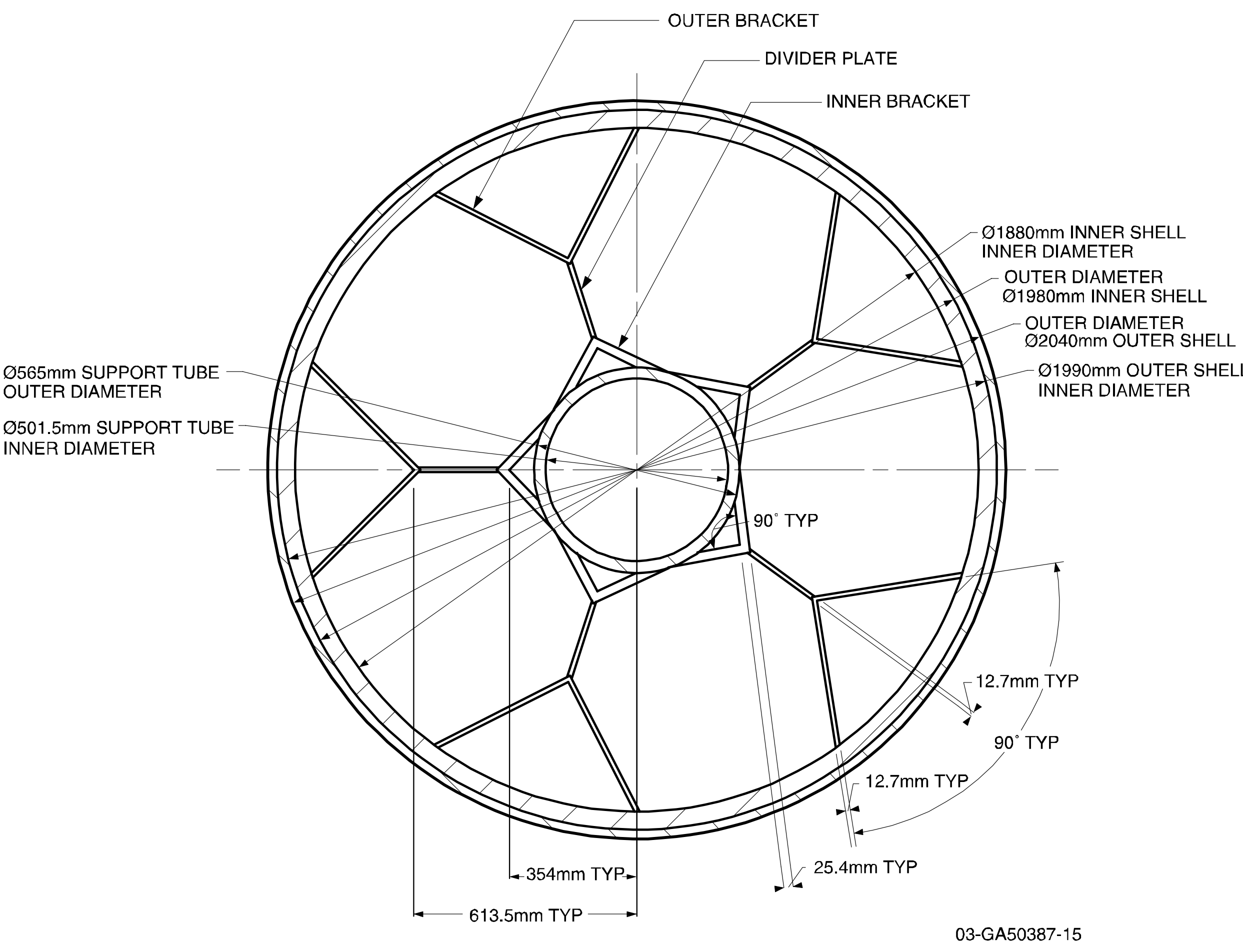

Figure 14. Cross-sectional view of typical codisposal waste package (see Reference 16).

The DOE SNF canister is placed in a 31.75-mm (1.250-in.)-thick support tube with a nominal outer diameter of 565 mm (22.244 in.). The support tube is connected to the inside wall of the waste package by a web-like structure of basket plates to support five long HLW glass canisters. The support tube and the plates are 4,607 mm (181.378 in.) long (see Reference 16).

5.1.2 Materials of Construction

The barrier materials of the waste package are typical of those used for commercial SNF waste packages. The inner barrier is composed of 50-mm (1.969-in.)-thick Type 316 NG stainless steel. The outer barrier comprises 25 mm (0.984 in.) of high-nickel alloy ASTM B 575 (Alloy 22). The SNF canister support tube and basket plates are constructed of carbon steel (ASTM A 516 Grade 70). A summary of pertinent dimensions and material specifications is provided in Table 6.

Table 6. Codisposal waste package dimensions and material specifications (see Reference 16).

Component	Material	Parameter	Dimension (mm)
Outer barrier shell	ASTM B 575 (Alloy 22)	Thickness	25
		Outer diameter	2,030
Inner barrier shell	SS 316 NG	Thickness	50
		Inner length	4,618
Extended outer shell lid	ASTM B 575 (Alloy 22)	Thickness	25
Outer shell flat closure lid	ASTM B 575 (Alloy 22)	Thickness	10
Inner shell lid	SS 316 NG	Thickness	105
Closure lid to extended outer lid gap	Air	Thickness	30
Inner shell lid to closure lid gap	Air	Thickness	30
Support tube	ASTM A 516 Grade 70	Outer diameter	565
		Inner diameter	501.5
		Length	4,607
Inner bracket	ASTM A 516 Grade 70	Thickness	25.4
		Length	4,607
Outer bracket	ASTM A 516 Garde 70	Thickness	12.7
		Length	4,607

5.2 Borosilicate Glass

5.2.1 HLW Glass Canister

There is no 15-ft-long Savannah River Site HLW glass canister. Therefore, the expected Hanford 15-ft HLW glass canister is used as found in the Fort St. Vrain waste package analysis (see Reference 16). The Hanford 15-ft HLW glass canister is a 4,572-mm (180-in.)-long stainless steel Type 304L canister with an outer diameter of 610 mm (24 in.).[17] The wall thickness is 10.5 mm (0.4,134 in.). These parameters are the same as the Savannah River Site canister, except that it is longer. The maximum loaded canister weight is 4,200 kg, and the fill volume is 87%.

5.2.2 HLW Glass Composition

The borosilicate glass intended for the Hanford canister has yet to be precisely specified in terms of composition, waste loading, physical properties, etc. However, given similar characteristics of the waste produced in the fuel dissolution and neutralization before tank farm storage, the resulting glass composition should be similar to that produced at Savannah River; see Appendix A for both chemical and radioisotope composition. While these compositions may not reflect the actual glass compositions ultimately produced by Hanford, they provide a basis for criticality evaluations for prebreach conditions.

6. DEGRADATION SCENARIOS

There are several engineered components within each waste package and also within the SNF canisters. Each of the components is expected to fail either sequentially or in conjunction with other components. Some components provide a barrier against moderator introduction to the package. Other components provide some degree of moderator exclusion (both intact and degraded).

Design of any of the TMI canisters and criticality analyses done at the time of loading and any prior transports took credit for canister internals and installed poisons (boron carbide). For purposes of comparison to past analyses, internal structures will be left in the initial model for baseline comparison. This planned analysis will examine various conditions that might be expected during degradation. Any further analysis will include removal or repositioning of materials within each style of canister in order to identify the most reactive configurations. Such removal of materials in the analysis reflects the more conservative approach toward disallowing credit of installed components for which condition or presence cannot be verified. However, failure scenarios do take credit for corrosion products in any degradation cases where solubilities of the degradation products suggest their retention, e.g., goethite formed from iron corrosion.

6.1 Waste Package Degradation

The waste package has been designed for durability and for preventing the introduction of water into the waste package. Until water actually enters the waste package, criticality remains a nonissue for any of the DOE fuels. However, water intrusion into any waste package effectively adds moderation around and potentially into the DOE SNF canister.

Each component within any waste package has a finite longevity upon the introduction of water. Because of nesting of these components and barriers within one another, the failures leading to the introduction of water immediately adjacent to the fuel matrix tend to be sequential. However, only a top breach of the waste package can realistically promote failure of other waste package components via a constant immersion of those components. Under these conditions of immersion in accumulated water, expected failures of components within the waste package are expected to occur from the bottom up. Such sequential failures may preclude certain configurations from forming. Scenario descriptions associated with any criticality analysis take no credit for any such limitation. The emphasis of any criticality analysis will be to model the most reactive configuration with respect to internal moderation and reflective boundaries.

6.1.1 Outer Shell

The outer shell of the waste package uses an Alloy 22 material as a primary barrier against a water-induced breach of the waste package itself. Water dripping onto the waste package at other than very specific widths along the top surfaces should not promote a waste package breach. Specific locations that might eventually hold water could promote collection/concentration of electrolytes in the ground water for a top-breach of the waste package.

6.1.2 Inner Shell

The inner shell is constructed of 316NG material. Its application as part of the barrier against water intrusion into the waste package potentially works against promotion of criticality safety, because it provides a barrier that promotes water retention if breached from the top.

6.1.3 Waste Package Canister Lattice

The internal lattice is constructed of A516 Grade 70 carbon steel. As a material less resistant to corrosion than the other stainless steel components, it offers a preferential failure mode through an expected galvanic corrosion coupling within a breached waste package. Long-term corrosion of this material is expected to produce either iron oxides or goethite [FeO(OH)]. In the horizontal orientation, there is an expectation of some settling of the individual canisters, one on another, that will close the designed tolerances between components. This closure of distance between individual, intact canisters is discounted in any model because there are more reactive configurations to evaluate.

6.1.4 HLW Canister (304L)

HLW packages higher in the waste package may experience intermittent drips, but general corrosion of the submerged HLW canisters is expected to occur from a rising water level within a top-breached waste package.

6.1.4.1 Borosilicate Glass. Prior analyses have, upon degradation of the borosilicate glass, used a clay-like material in place of the glass. While certain materials may leach from the glass and other chemical compounds may form, formation of the clayey material permits an assumed redistribution of these chemicals throughout the waste package. The clay forms a viso-elastic material with flow properties that either fills in interstitial spaces or allows heavier materials to settle through the clay.

The clay can exhibit several properties conducive to promoting decreasing system reactivity, depending on the conditions of the clay (degree of water saturation) and its location (as a reflector surrounding the fissile material or as moderating material when mixed with the fissile material).

6.1.5 SNF Canister (316L)

The SNF canister will be surrounded by the carbon steel centering sleeve within the waste package. Goethite expected to form around the SNF canister may delay the breach of the SNF canister. Expectation of the SNF canister breach would be from underneath because of early immersion and galvanic corrosion coupling with the centering sleeve. The ability to fully immerse the TMI canister would also be from below. Eventually, degraded components of the SNF canister shell would convert to goethite.

6.1.5.1 SNF Canister Sleeve. The internal SNF canister sleeve would use carbon steel. Whether a top or bottom breach of the SNF canister occurred, goethite formation is expected from the sleeve based on a corrosion breach/water ingress into the SNF canister and prior to actual failure of the TMI canister. On a conceptual basis, the centering sleeve would consist of a 16-in. Schedule 10 pipe (0.250-in. wall thickness) with standoffs inside the SNF canister. There are no implied geometry controls created by the sleeve; its installation provides a centering function and minimizes rattle room inside the SNF canister. Upon degradation, the sleeve would convert to a goethite layer between two 300 series stainless steel layers of the SNF canister and the TMI-2 canister.

6.1.5.2 TMI Knockout Canister (304L SS). The knockout canister contains a distinctive set of internals that were designed to: (1) facilitate disengagement and retention of core debris material that was hydro-vacuumed from the core and (2) maintain a degree of neutron poisoning with boron carbide installed.

Information for the K-506 canister fissile loading, and by neglecting the knockout canister intervals, provides the largest free volume for water fill, highest fissile mass, and opportunity for optimum moderation.

6.1.5.2.1 Installed Structure (300 series SST)—The internal structure of the knockout canister may provide a number of features associated with enforcement of criticality safety at the time of loading and for interim storage. On the relatively short-term basis, up to and including recent transport and current storage in a dry storage facility at INTEC, credit for the presence of the internal basket structure might be taken.

On a longer-term basis that ultimately includes a breach of the TMI canister, credit for the structure would be disallowed. Given the unknown condition of the TMI core debris, a simplifying assumption would distribute the fissile mass as fuel pellets uniformly throughout the debris mixture inside the knockout canister. With a projected degradation of the knockout canister internals, there is the possibility of a radial redistribution of fuel pellets.

6.1.5.2.2 Installed Poisons—Boron carbide was installed specifically in both the fuel canisters and the knockout canisters in various shapes and forms. While perhaps practical to take credit for such poisons during the short-term storage of materials and known integrity of the TMI canister, the long-term disposal environment precludes any ability to take credit for the presence of any boron compound in the system due to boron solubility.

Within each knockout canister, there is a tube and plate support structure that provides displacement of both core debris with included fuel pellets and moderator in the case of a breached TMI canister. Because of the postclosure conditions mandated for any DOE SNF canister, credit for engineered structures inside any canister must be discounted.

6.1.5.2.3 Core Debris—Vacuumed TMI core fuel consists of a mixture of fuel pellets and other inert debris (metal and metal oxides) from damaged fuel assemblies. There is significant uncertainty relative to the mix or portions of inerts to fuel mass, other than a reported total debris mass, a total uranium mass, and a U-235 mass. Furthermore, any fissile material must assume optimum distribution within a canister.

6.1.5.3 TMI Fuel Canister. The design of the fuel canister provides the greatest constraint of fuel pellet movements. The box construction for each assembly and the associated LiCon™ prevents pellet movement in any radial or outward expansion. Pellet pitch spacing for a fuel assembly within the confines of the fuel canister sleeve would be less than that found in a knockout canister. This suboptimal spacing has been shown (see Reference 20) to be less reactive. Materials in the concrete (calcium and silicon) promote water displacement from a reflective position. Furthermore, upon degradation and formation of a clay-like material that could mix in with the pellets, further water displacement is expected. Pellets within damaged fuel assemblies are not expected to expand or redistribute as with the knockout canisters because even the damaged fuel rods and the longevity of the zirconium cladding provide a degree of constraint against pellet movement. However, the knockout canister (fissile loading, free volume) bounds the fuel canister in any case.

6.1.5.4 TMI Filter Canister. While the filter canisters also have a large free volume similar to the knockout canister after degradation of the internals, the listed fissile loading is dwarfed and, therefore, bounded by any of the knockout canisters.

6.2 Degradation of Waste Package Internals

Debris in the fuel canister consists of fuel core materials ranging in shape and mass from nearly intact fuel assemblies (fuel canisters) to loose materials subject to hydro-vacuuming (knockout canisters). The materials contained within each waste package vary as to the degree the integrity of fuel pellet containment is still provided by the zirconium tubing. Historical experience reports that systems with fissile enrichments <5% are best modeled as heterogeneous rather than homogeneous systems.[18,19]

The most significant issue related to criticality modeling for these canisters revolves around assumptions made relative to fuel pellet orientation within each canister. In the case of the fuel canisters, nearly intact fuel assemblies provide some degree of containment by virtue of fuel pins that maintain fuel pellet orientation. The knockout canister provides a unique set of circumstances, where individual pellets were loaded and mixed into the same canister with other debris from the damaged reactor core. Neither the orientation of the pellets or their relative spacing in the debris mix inside the knockout canisters is well defined.

The total mass and heavy metal masses (uranium and plutonium) are reported for these canisters, yet even those reported values have stated uncertainties and ranges of values. Without some degree of verification (radiography, gamma scan, etc.), certain assumptions are needed to support the material distribution with a given TMI canister when establishing the parameters used in any criticality model. Previous criticality analysis[20] has shown that the most reactive condition to be examined would rely on a uniform distribution of fuel particles throughout the internal volume of the knockout canister.

6.2.1 Water Ingress to Waste Package

Water ingress into the waste package through a top breach is capable of creating both wetted surfaces as well as a ponding or bathtub accumulation of water in the bottom of the package. For the wetted surface condition, galvanic corrosion couples and localized corrosion can promote breaching of the HLW or SNF canister walls, but would not contribute significantly to quantitative water additions to the canister internals.

6.2.2 Waste Package Spacer Degradation

The carbon steel spacer (or centering sleeve) used to facilitate loading of the waste package is the component inside the waste package most susceptible to general corrosion and loss of form. Corrosion byproducts for this degradation may form iron oxides or goethite. As a relatively insoluble material, the goethite would provide a superficial layer around the outside of the SNF canister as it expands during corrosion. The corrosion behavior of the carbon steel sleeve is that it will degrade faster than the stainless steel of the TMI-2 canister once exposed to water.

6.2.2.1 Stainless Steel Degradation. The HLW canister degradation is the second component expected to undergo corrosion. For the horizontal waste package orientation, the bottom HLW canisters would be the first to be submerged. The uppermost canisters, including the SNF canister, inside the waste package are expected to fail over a longer period of time. Generally, a delayed failure of these upper canisters would allow them to settle in an intact condition into the previously degraded components. The molybdenum and chromium form molybdic and chromic acids out of the 300 series stainless steel degradation. The nickel forms nickel oxide (NiO), and the iron decomposes to goethite [FeO(OH)]. From any standpoint that promotes the most reactive configuration, analysis should examine the uniform distribution of these degradation materials.[21]

6.2.2.2 Clay Formation. Clay formation is a natural outgrowth of the HLW degradation. Boron, sodium, and potassium are the major components with high solubility and are, therefore, expected to transport from the canister/waste package; this would leave the calcium, aluminum, iron, and silicon as the major constituents of the clay.

6.3 SNF Canister Degradation

Installation of an otherwise intact TMI-2 canister inside a standardized SNF canister merely provides an already qualified size canister for installation in the waste package.

6.3.1 SNF Canister Degradation

The TMI-2 canisters installed in each standard SNF canister represent one additional barrier to water intrusion and access to any fuel material. The stainless steel construction of the SNF canister and its positioning in the center of the waste package are expected to delay breaching due to delayed exposure to water. Such a delay is predicated on an ability to accumulate water inside the waste package but with no direct water contact with the waste package until the water level inside the breached waste package approaches package centerline. Furthermore, SNF canister breach is expected at the bottom of the SNF canister where first water contact and galvanic corrosion coupling occurs between carbon steel and stainless steel due to the orientation of the canisters.

6.3.2 Carbon Steel Sleeve Degradation

Centering sleeve degradation (inside the standard SNF canister) would follow the same degradation path and result in the same components expected from the carbon steel degradation associated with the waste package basket structure.

The center sleeve is an open-ended tube that acts as a positioning or centering device to minimize the rattle room of any TMI canisters. When flooded, it will be subject to corrosion on all surfaces. The carbon steel sleeve material upon degradation forms goethite that under unconfined conditions provides merely decreased reflection.

6.3.3 TMI Canister Degradation

The stainless steel TMI canister may already have experienced some localized, external corrosion while in wet storage for the approximately 15 years prior to transfer to dry storage. The early storage environment may lead to an otherwise premature failure once exposed to another water immersion occurring inside a failed waste package. Such a premature failure would likely promote preferential introduction of water rather than degradation products from HLW canister degradation.

Any breach of the TMI-2 canister would be subsequent to the degradation of the sleeve and standoff brackets or spacers. In a horizontal package orientation, such degradation would allow the TMI-2 canister to settle to the inside surface of the canister. This would result in a reflective layer of both the goethite formed from the sleeve and the 316 stainless steel layer of the standard SNF canister.

6.3.3.1 Fuel Canister Internals. The fuel canister, as designated with a “D” prefix on the canister callout, consists of stainless steel sleeve to create a void space within the canister that is capable of accommodating a single PWR assembly. The interstitial space between the box and the internal side-walls of the canister were filled with a cementatious material with a trade name of LiCon™. The LiCon™ composition and quantity is expected to have a negligible effect on the chemistry relative to the heavy metal and debris inside the internal sleeve. If the LiCon™ were to promote any solubility of the TMI heavy metal debris, its contribution would only serve to make the system less reactive by further homogenizing the debris.

The square liner inside the fuel canister provides a definable boundary within which to distribute a designated number of fuel pellets. The liner, the LiCon™, and the TMI canister shell compose a set of reflective boundaries for purposes of modeling in the criticality analysis.

The quantity of damaged assembly / debris materials is constrained by the space available inside the sleeve. Neither the physical mass nor the reported heavy metal mass in any fuel canister exceeds the specified weight of an intact PWR fuel assembly for the TMI reactor (reference Appendix C). The constraints provided by the surrounding materials prevent redistribution of any fuel pellets to a more optimal pitch or the addition of significant moderation toward a more optimally moderated system. Intact, the LiCon™ and liner prevent expansion of any pellet array, and their degradation would promote collapse.

6.3.3.2 Knockout Canister Internal Structures. The design of the knockout canister provides the largest free volume coupled with one of the highest reported debris/heavy metal loadings (441.9 kg U, 9.42 kg U-235, 842.18 kg debris; see Table 4). The structural materials inserted into the knockout canisters are constructed of a combination of 304 and 316 series stainless steels. Boron carbide in a pellet form was installed inside each of the five poisoned rods (see Figure 3) used to maintain support plate spacing.

Over time, neither the structural integrity of the rods/support plates nor the continued presence of the boron carbide can be ensured. Furthermore, the nature or condition of any inert debris inside the canister cannot be given any credit for purposes of moderator displacement because its location and quantity cannot be verified. The most conservative approach would be to assume knockout canister without any internals or poisons and a full complement of fuel pellets (67,600) found in a single, intact assembly. Such an assumption results in a fissile mass that is approximately 45% in excess of the maximum reported knockout (K506) canister loading.

Typical criticality analyses conducted with highly enriched systems have relied on a homogeneous distribution of fissile materials inside their respective SNF canisters. For the case of low-enriched (<5% U-235) systems, typical models rely on discrete or heterogeneous distribution of the fissile material (see References 18 and 19). Within a knockout canister, such an analysis will rely on a distribution of 67,600 fuel pellets within the free volume (between the lower support plate and the upper closure lid). Furthermore, the fuel pellets are assumed to be uniformly distributed throughout the debris at the time of loading. The follow on assumption is that they will continue to be uniformly distributed throughout the canister in spite of the disappearance of any debris, structural material, or boron carbide.

6.3.3.3 Filter Canister Internals. The structural internals associated with the filter canisters differ from either the debris or knockout canisters. However, the physical free volume is considered comparable to that of the knockout canister. The major distinction between the knockout and filter canisters is the reported fissile loads. The most highly loaded of the filter canisters (F471 with 4.59 kg U-235) represents approximately 33% of a complete fuel assembly's worth of pellets. Furthermore, the nature of the solids collected in the knockout canister provides a more homogeneous mixture (fines and small particulate) that would be less conducive to promotion of a system with maximum reactivity. To the extent that both the filter and knockout canisters have similar free volumes, both the nature and quantity of the solids in the knockout would provide the bounding case analysis for any degradation scenario.

6.4 Order-of-Failure Scenarios

There will necessarily be an ordered sequence of events that could ultimately lead to water ingress into the TMI-2 canisters. This order is prescribed by the nature of the other barriers and may be influenced by the time-dependent corrosion of these various barriers. Both sequential and concurrent events contribute to eventual moderator ingress to the SNF canister.

6.4.1 Waste Package Failures

Failure or breach of a waste package may occur for a number of events, but only a drip from above onto the centerline of the waste package provides a credible means of accumulating a significant amount of water in a waste package. The degree of degradation will be a function of the ability to collect and retain water. Ultimately, the most reactive configuration conceivable needs to account for the failure mechanisms of the various barriers. The analysis must necessarily allow for retention of materials associated either with reflection or moderator in their respective, optimal location within a waste package. Various conditions will be analyzed to identify the most reactive system configuration.

While various failure scenarios can be depicted for waste package failure and a subsequent release, only a top-drip on the waste package centerline provides a mechanism to breach and promote water accumulation inside the package. Any other breach mechanism does not promote water ingress and retention inside the waste package.

6.4.2 HLW Canister Failure

For any water accumulation inside the waste package, there will be various degrees of corrosion based on the rate of water introduction. There is an expected, generalized corrosion of the carbon steel insert that separates the SNF and HLW canisters. Generalized corrosion of the lower HLW canister shell will occur concurrent with the carbon steel insert. Degradation of the lower HLW canisters is expected to occur incrementally as water level rises inside the breached waste package.

6.4.3 SNF Canister Failure

The SNF canister corrosion should be delayed due to its elevated position within the waste package. The position of any corrosion induced breach of the SNF canister now becomes important because it impacts the potential for formation of a more or less reactive SNF canister configuration. Ultimately, sequential failures of the SNF canister and any TMI-2 canister result in a combination of clay in either a reflective layer (outside the TMI-2 canister) or moderating location (inside the TMI-2 canister). Water may also be found in either a reflective or moderating position.

Water internal to the SNF canister (located approximately mid-point of the horizontal axis of the waste package) will promote corrosion on the inner surface of the SNF canister, inside and outside surfaces of the centering sleeve, and the outer surface of the TMI canister.

There is a predictable sequence of breaching to the various components, starting with the top breach of the waste package. Subsequent breaches of barriers within the waste package will experience bottom-to-top, generalized corrosion as water accumulates inside the waste package. The rate and sequential corrosion of various components is very dependent on the rate of accumulation for water entering the breached waste package. The bottom-to-top corrosion would tend to promote filling canisters within the waste package from bottom to top. Such flow direction is expected to be somewhat limiting in terms of flowing clay-like material from any HLW degradation into the SNF canister. Furthermore, at least in the TMI canister disposal scheme, there is not only the required breach of the SNF canister, but also the TMI canister internal to the SNF canister to get water to moderate the fuel debris. Any breach of each additional barrier creates only a more convoluted path for any material transport other than water, whether entering or exiting the TMI canister.

6.4.4 TMI Canister Failure

The TMI canister will be resting on the inside curvature of the centering sleeve. The centering sleeve would be positioned inside the SNF canister with some form of standoff (assuming they had not corroded). Water is expected to be the sole moderator of influence (see Reference 20) (rather than hydrated clayey material). Any clayey material will have little if any fissile material, because any heavy metal presence is derived from trace quantities of slightly enriched natural uranium (1.15% maximum expected in Hanford glass) in the waste stream used for glass production.

The fuel debris inside the TMI canister is assumed to be uniformly distributed throughout the inside of the canister as cylindrical pellets the size of the original pellets in the PWR assembly. Criticality analysis will use the uniform distribution of fuel pellets as a starting point with varying pitch and spacing of the pellets to identify "most reactive" configurations. Intrusion of water into the void space inside the TMI canister would provide a fully flooded environment. Neither the zirconium metal or oxide from the cladding, nor the uranium oxide pellets are considered to be soluble; so, transport or movement of these materials outside the confines of the TMI canister in solution form can be ignored. Furthermore, the heterogeneous nature of low-enriched systems (<5%) is generally acknowledged to be more reactive than an equivalent mass of fissile material in a homogeneous system.

Given the combination of barriers and boundaries created by nesting one canister inside another canister within a waste package, there is a multitude of combinations and permutations of degraded conditions to examine. The key to any such analysis would be to identify the most reactive system configuration.

The most seriously degraded mode anticipated for the TMI canister should examine a water-filled TMI canister surrounded by prebreach clay from HLW decomposition. Experience in a criticality scoping analysis (see Reference 18) with TMI-2 canisters revealed two significant findings. Addition of a nonmoderator, such as clay to the fuel pellet zone, displaces the moderator and, thereby, reduces system reactivity. Conversely, addition of water to a reflective layer, e.g. clay around the fuel pellet zone, appears to decrease the system reactivity, although not as dramatically as the prior condition.

Additional calculations could include comparisons on the basis of:

- Added plutonium (EOL) values
- Spherical rather than cylindrical pellets
- Moderator displacement based on inert content assumptions.

A comparative loading of both hypothetical and reported values for fissile materials and heavy metals is presented in Table 7.

Table 7. Comparative fissile loadings for canisters and assemblies.

Total U/assembly, kg:	463.63
Total U/core, kg:	82,039.5
Pellets/assembly	67,600

Fuel Assembly	Number of Assemblies	Enrichment	Fissile per Assembly (kg)	Total Fissile, (kg)	Total U (kg)
Type I	57	1.98	9.1799	523.253	26,426.91
Type II	60	2.64	12.2398	734.390	27,817.80
Type III	60	2.96	13.7234	823.407	27,817.80
Reactor totals				2,081.050	82,062.51

Quantities for modeling:

	Type	% Enrichment	U-235 (kg)	Pu-239 (kg)	Basis
	High enrich + EOL Pu	—	13.72	1.0	(Calculated)
	Criticality model	3.00	13.90	—	(Bounding assumption)
Intact assembly	High enrich	2.96	13.72	—	(Specification)
Canister	D119	2.67	10.06	0.506	(Reported)
Canister	D193	2.67	9.41	0.473	(Reported)
Canister	D299	2.65	9.37	0.485	(Reported)
Intact assembly	Med. enrich	2.64	12.24	—	(Specification)
Canister	D260	2.58	9.66	0.537	(Reported)
Whole core	Avg. enrich	2.54	11.77	—	(Calculated)
Center core	Avg. enrich	2.32	10.75	—	(Calculated)
Canister	K506	2.13	9.42	0.845	(Reported)
Intact assembly	Low enrich	1.98	9.18	—	(Specification)

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Appendix A

Materials and Compositions

Appendix A

Materials and Compositions

316L Stainless Steel—The stainless steel inner shell of the waste package will use 316L stainless steel material; composition is as shown in Table A-1.[A-1]

Table A-1. Chemical composition of stainless steel Type 316L.

Element	Composition (wt%)	Value Used (wt%)	Number Density [atom/(cm·b)]
Carbon (C)	0.03 (max.)	0.03	1.2003E-04
Manganese (Mn)	2.00 (max.)	2.00	1.7494E-03
Phosphorus (P)	0.045(max.)	0.045	6.9817E-05
Sulfur (S)	0.03 (max.)	0.03	4.5091E-05
Silicon (Si)	1.00 (max.)	1.0	1.7110E-03
Chromium (Cr)	16.00–18.00	17.00	1.5712E-02
Nickel (Ni)	10.00–14.00	12.00	9.8251E-03
Molybdenum (Mo)	2.00–3.00	2.50	1.2522E-03
Nitrogen (N)	0.10 (max.)	0.10	3.4318E-04
Iron (Fe)	Balance	65.295	5.6187E-02

Source: ASTM A 276-91a, p.2 (UNS S31603)

Density = 7.98 g/cm^3

304L stainless steel—The material used in the construction of the typical HLW glass canister (see Reference A-1).

Table A-2. Chemical composition of stainless steel Type 304L.

Element	Composition (wt%)	Value Used (wt%)	Number Density [atom/(cm·b)]
Carbon (C)	0.03 (max.)	0.03	1.1943E-04
Manganese (Mn)	2.00 (max.)	2.00	1.7407E-03
Phosphorus (P)	0.045 (max.)	0.045	6.9467E-05
Sulfur (S)	0.03 (max.)	0.03	4.4865E-05
Silicon (Si)	0.75 (max.)	0.75	1.2769E-03
Chromium (Cr)	18.00–20.00	19.00	1.7472E-02
Nickel (Ni)	8.00–12.00	10.00	8.1465E-03
Nitrogen (N)	0.1	0.10	3.4146E-04
Iron (Fe)	Balance	68.045	5.8260E-02

Source: ASTM A 240/a 240M-97a, p.2. (UNS S30403)

Density = 7.94 g/cm^3

Alloy-22—The outer shell of the waste package will use Alloy-22 material composition as shown in Table A-3 (see Reference A-1).

Table A-3. Chemical composition and density of Alloy-22.

Element	Composition (wt%)	Value Used	Number Density [atom/(cm·b)]
C	0.015 (max)	0.015	6.5356E-05
Mn	0.50 (max)	0.5	4.7627E-04
Si	0.08 (max)	0.08	1.4906E-04
Cr	20–2.5	21.25	2.1387E-02
Mo	12.5–4.5	13.5	7.3637E-03
Co	2.50 (max)	2.5	2.2199E-03
W	2.5–.5	3.0	8.5397E-04
V	0.35 (max)	0.35	3.5955E-04
Fe	2.0–.0	4.0	3.7483E-03
P	0.02 (max)	0.02	3.3791E-05
S	0.02 (max)	0.02	3.2736E-05
Ni	Balance	54.765	4.8829E-02

Density = 8.69 g/cm^3

ASTM B 575 (UNS N06022)

Carbon Steel—The curved, carbon steel impact plates fit inside each end of the SNF canister (see Reference A-1).

Table A-4. Chemical composition of ASTM A 516 Grade 70 carbon steel.

Element	Composition (wt%)	Value Used (wt%)	Number Density [atom/(cm·b)]
Carbon (C)	0.30 (max.)	0.30	1.1808E-03
Manganese (Mn)	0.85–1.20	1.025	8.8199E-04
Phosphorous (P)	0.035 (max.)	0.035	5.3418E-05
Sulfur (s)	0.035 (max.)	0.035	5.1750E-05
Silicon (Si)	0.15–0.40	0.275	4.6287E-04
Iron (Fe)	Balance	98.33	8.3236E-02

Source: ASTM A 516/A 516M-90, Table 1 (UNS K02700)

Density = 7.85 g/cm^3

Table A-5. Composition and density of Savannah River Site high-level waste glass.

Element/Isotope	Composition[a] (wt%)	Number Density [atom/(cm·b)]	Element/ Isotope	Composition[a] (wt%)	Number Density [atom/(cm·b)]
O	44.77	4.8038E-02	Ag	0.0503	7.9241E-06
U-234	0.0003	2.4048E-08	Ni	0.7349	2.1489E-04
U-235	0.0044	3.1773E-07	Pb	0.0610	5.0495E-06
U-236	0.0010	7.5726E-08	Si	21.8880	1.3375E-02
U-238	1.8666	1.3458E-04	Th	0.1856	1.3727E-05
Pu-238	0.0052	3.7360E-07	Ti	0.5968	2.1397E-04
Pu-239	0.0124	8.9111E-07	B-10	0.5918	1.0143E-03
Pu-240	0.0023	1.6282E-07	B-11	2.6189	4.0826E-03
Pu-241	0.0010	6.8960E-08	Li-6	0.0960	2.7378E-04
Pu-242	0.0002	1.3591E-08	Li-7	1.3804	3.3767E-03
Cs-133	0.0409	5.2878E-06	F	0.0319	2.8774E-05
Cs-135	0.0052	6.5664E-07	Cu	0.1526	4.1225E-05
Ba-137	0.1127	1.4022E-05	Fe	7.3907	2.2714E-03
Al	2.3964	1.5243E-03	K	2.9887	1.3119E-03
S	0.1295	6.9489E-05	Mg	0.8248	5.8238E-04
Ca	0.6619	2.8344E-04	Mn	1.5577	4.8662E-04
P	0.0141	7.7901E-06	Na	8.6284	6.4414E-03
Cr	0.0826	2.7253E-05	Cl	0.1159	5.6112E-05

Density[b] at 25°C = 2.85 g/cm^3

a. See Reference A-2.

b. See Reference A-3.

The LiCon material identified in Table A-6 was used to fill the cord-section space between the canister shell and the square sleeve internal to the TMI-2 fuel canister.[A-4]

Table A-6. Low density concrete (LiCon™) composition.

Element	Number Density (atoms/b-cm)
Oxygen (O)	1.1855E-02
Sodium (Na)	1.4380E-04
Aluminum (Al)	2.8385E-03
Silicon (Si)	8.7196E-04
Magnesium (Mg)	3.5859E-05
Calcium (Ca)	1.3101E-03
Iron (Fe)	1.3576E-05

Table A-7 provides the compositions associated with the cladding material used in the construction of the TMI-2 PWR assembly.[A-5]

Table A-7. Composition of Zircaloy-4.

Element	Composition (wt%)
Tin (Sn)	1.20–1.70
Iron (Fe)	0.18–0.24
Chromium (Cr)	0.07–0.13
Nickel (Ni)	—
Niobium (Columbium)	—
Oxygen (O)	—
Iron + chromium + nickel	—
Iron + chromium	0.28–0.37
	Max. Impurities (wt%)
Aluminum (Al)	0.0075
Boron (B)	0.00005
Cadmium (Cd)	0.00005
Carbon (C)	0.027
Chromium (Cr)	—
Cobalt (Co)	0.0020
Copper (Cu)	0.0050
Hafnium (Hf)	0.010
Hydrogen (H)	0.0025
Iron (Fe)	—
Magnesium (Mg)	0.0020
Manganese (Mn)	0.0050
Molybdenum (Mo)	0.0050
Nickel (Ni)	0.0070
Nitrogen (N)	0.0080
Phosphorus (P)	—
Silicon (Si)	0.0120
Tin (Sn)	—
Tungsten (W)	0.010
Titanium (Ti)	0.0050
Uranium (U)	0.00035

Table A-8. Curie content per 15-ft Hanford glass log.[A-6]

Isotope	Half-life (yrs)[a]	Curies/15-ft Canister: 21 Months After Fuel Discharge[b]	1 Year[c]	5 Year[c]	10 Year[c]	20 Year[c]
Fe-55	2.73E-00	2.24E+02	1.74E+02	6.29E+01	1.77E+01	1.40E-00
Co-60	5.27E-00	6.81E-00	5.97E-00	3.53E-00	1.83E-00	4.91E-01
Ni-59	7.60E+04	2.16E-01	2.16E-01	2.16E-01	2.16E-01	2.16E-01
Ni-63	1.00E+02	2.49E+01	2.48E+01	2.41E+01	2.33E+01	2.17E+01
Sr-90	2.91E+01	6.64E+04	6.48E+04	5.89E+04	5.23E+04	4.12E+04
Y-90	7.32E-03	6.64E+04	6.48E+04	5.89E+04	5.23E+04	4.12E+04
Zr-93	1.50E+06	2.05E-00	2.05E-00	2.05E-00	2.05E-00	2.05E-00
Zr-95	1.75E-01	4.38E-00	8.43E-02	1.15E-08	3.03E-17	2.10E-34
Nb-95	9.58E-02	9.01E-00	6.50E-03	1.77E-15	3.47E-31	1.33E-62
Tc-99	2.13E+05	1.49E+01	1.48E+01	1.48E+01	1.48E+01	1.48E+01
Ru-106	1.02E-00	7.93E+03	4.02E+03	2.65E+02	8.87E-00	9.93E-03
Rh-106	2.49E-04	7.93E+03	0.00E+01	0.00E+01	0.00E+01	0.00E+01
Ag-110m	6.84E-01	2.53E-00	9.17E-01	1.60E-02	1.01E-04	4.04E-09
Cd-113m	1.41E+01	2.32E+01	2.21E+01	1.81E+01	1.42E+01	8.68E-00
Sn-119m	8.03E-01	8.61E-00	3.63E-00	1.15E-01	1.53E-03	2.73E-07
Sn-121m	5.50E+01	1.68E-01	1.66E-01	1.58E-01	1.48E-01	1.31E-01
Sn-123	3.54E-01	4.59E-00	6.48E-01	2.57E-04	1.44E-08	4.53E-17
Sb-125	2.76E-00	2.80E+03	2.17E+03	7.96E+02	2.26E+02	1.83E+01
Sn-126	1.00E+05	7.31E-01	7.31E-01	7.31E-01	7.31E-01	7.30E-01
Sb-126m	3.40E-02	7.31E-01	1.01E-09	3.68E-45	1.86E-89	4.72E-178
Te-125m	1.59E-01	6.81E+02	8.70E-00	2.31E-07	7.81E-17	8.94E-36
Te-127m	2.99E-01	4.76E-00	4.68E-01	4.35E-05	3.98E-10	3.32E-20
Cs-134	2.07E-00	1.91E+03	1.36E+03	3.56E+02	6.64E+01	2.32E-00
Cs-135	2.30E+06	3.99E-01	3.99E-01	3.99E-01	3.99E-01	3.99E-01
Cs-137	3.02E+01	8.10E+04	7.92E+04	7.22E+04	6.44E+04	5.12E+04
Ba-137m	4.86E-06	7.66E+04	7.48E+04	6.82E+04	6.08E+04	4.84E+04
Ce-144	7.80E-01	4.73E+04	1.95E+04	5.56E+02	6.53E-00	9.02E-04
Pm-147	2.62E-00	6.31E+04	4.84E+04	1.68E+04	4.49E+03	3.20E+02
Sm-151	9.00E+01	1.33E+03	1.32E+03	1.28E+03	1.23E+03	1.14E+03
Eu-152	1.35E+01	4.35E-00	4.13E-00	3.37E-00	2.60E-00	1.56E-00
Eu-154	8.59E-00	5.34E+02	4.92E+02	3.56E+02	2.38E+02	1.06E+02
Eu-155	4.71E-00	6.53E+02	5.63E+02	3.13E+02	1.50E+02	3.44E+01

Table A-8. (continued).

Isotope	Half-life (yrs)[a]	Curies/15-ft Canister 21 Months After Fuel Discharge[b]	1 Year[c]	5 Year[c]	10 Year[c]	20 Year[c]
Heavy Metals						
U-235	7.04E+08	3.13E-04	3.13E-04	3.13E-04	3.13E-04	3.13E-04
U-238	4.47E+09	5.91E-03	5.91E-03	5.91E-03	5.91E-03	5.91E-03
Np-237	2.14E+06	3.16E-01	3.16E-01	3.16E-01	3.16E-01	3.16E-01
Pu-238	8.77E+01	1.22E-00	1.21E-00	1.17E-00	1.13E-00	1.04E-00
Pu-239	2.41E+04	2.24E-00	2.24E-00	2.24E-00	2.24E-00	2.24E-00
Pu-240	6.56E+03	8.61E-01	8.61E-01	8.60E-01	8.60E-01	8.59E-01
Pu-241	1.44E+01	4.10E+01	3.91E+01	3.22E+01	2.53E+01	1.56E+01
Am-241	4.33E+02	9.16E+02	9.15E+02	9.09E+02	9.02E+02	8.88E+02
Am-242	1.41E+02	6.51E-01	6.48E-01	6.35E-01	6.20E-01	5.90E-01
Am-243	7.37E+03	1.07E-01	1.07E-01	1.07E-01	1.07E-01	1.07E-01
Cm-242	4.46E-01	7.93E-01	1.68E-01	3.35E-04	1.42E-07	2.53E-14
Cm-244	1.81E+01	1.99E+01	1.91E+01	1.64E+01	1.35E+01	9.23E-00

a. Chart of the Nuclides, 14th edition, General Electric, 1989.[A-7]

b. Scaled values based on a ratio of 13.5/8.5 for 15-ft vs. 10-ft glass canister fill.

c. Calculated values based on decayed half-life values.

d. Not all isotopes in the referenced source document were included in this table, either because of their low Ci concentrations or short decay half-lives.

References

A-1. DOE 2002, *Criticality Scoping Analysis for a Dual Canister/Waste Package Disposal Strategy*, REP/SNF/REP-080, Rev. 0, August 2002.

A-2. CRWMS M&O 1999, *DOE SRS HLW Glass Chemical Composition*, BBA000000-01717-00017 Rev 00, Las Vegas Nevada: ACC: MOL.199990215.0397.

A-3. R. B. Stout and H. R. Leider, eds., 1991, *Preliminary Waste Form Characteristics Report*, Version 1.0, Livermore, California, Lawrence Livermore National Laboratory, ACC: MOL.19940726.0118, 1991.

A-4. R. R. Jones, *Criticality Safety Evaluation for the TMI-2 Fuel Canister Storage Rack*, Report No. SE-PB-85-25, EG&G, Idaho Falls, September 1985.

A-5. ASTM 2001, B 350/B350M-01, *Standard Specification for Zirconium and Zirconium Alloy Ingots for Nuclear Application*, July 2001.

A-6. Office of Civilian Radioactive Waste Management, *Characteristics of Potential Repository Wastes, and Other Radioactive Wastes,* DOE/RW-0184, Vol. 1, Rev. 1, December 1987.

A-7. Nuclides and Isotopes: Chart of the Nuclides, 14th edition, GE Nuclear Energy, San Jose, CA, 1989.

Appendix B

Pressurized Water Reactor Fuel Specification—Physical Description Report for TMI Fuel

Appendix B

Physical Description Report

	[Ref. B-1]	[Ref. B-2]
Initial year of manufacture	1971	—
Final year of manufacture	—	—
Total number fabricated to date	(Note 1)	—
Assembly width (inches)	8.536	8.54
Assembly length (inches)	165.625	165.6
Rod pitch (inches)	0.568	0.568
Total assembly weight (lb)	1515.0	1510 to 1588
Weight of heavy metal (lb)	1022.12	(Note 2)
Weight of UO_2 (lb)	(Note 2)	1159
Metric tonnes initial heavy metal (metric tons)	0.45363	—
Enrichment range (% U-235)	2.0–4.0	—
Average pellet enrichment (% U-235)	—	2.57
Average design burnup (MWd/MTIHM)	35000	—
Maximum design burnup (MWd/MTIHM)	50200	—

Note 1: Total quantity fabricated to date (as of 1987) = 3764 assemblies.

Note 2: The heavy metal (U) reported at 1022.12 lb translates into: [[(238.0289)+32]/238.0289]*1022.12 lb U = 1159.53 lb UO_2.

Table B-1. Fuel assembly hardware parts and materials.

Part Name	Parts/ Assembly	Weight (kg)/ Assembly	Zone	Material Name	Material Fraction
Top nozzle	1	7.4800	Top	Stainless steel CF3M	1.00000
Bottom nozzle	1	8.1600	Bottom	Stainless steel CF3M	1.00000
Guide tubes	16	8.0000	In core	Zircaloy-4	1.00000
Instrument tube	1	0.6400	In core	Zircaloy-4	1.00000
Spacer-plenum	1	1.0400	Gas plenum	Inconel-718	1.00000
Spacer-bottom	1	1.3000	Bottom	Inconel-718	1.00000
Spacer-incore	6	4.9000	In core	Zircaloy-4	1.00000
Spring retainer	1	0.9100	Top	Stainless steel CF3M	1.00000
Holddown spring	1	1.800	Top	Inconel-718	1.00000
Upper end plug	2	0.0600	Top	Stainless steel 304	1.00000
Upper nut	15	0.5100	Top	Stainless steel 304L	1.00000
Lower nut	16	0.1500	Bottom	Stainless steel 304	1.00000
Grid supports	7	0.6400	In core	Zircaloy-4	1.00000

Drawing number associated with assembly: Drawing number 1155248 F Rev.0 (See Reference B-3).

Table B-2. Fuel rod description table.

	[Ref. B-1]	[Ref. B-2]
Type of Rod	Per Fuel Rod	
Fuel rod positions per assembly	225	225
Typical number of fueled rods per assembly	208	204 (Note 1)
Guide tubes per assembly	16	—
Instrument tubes per assembly	1	—
Rod diameter (inches)	0.430	0.430
Rod length (inches)	153.68	153.2
Active length (inches)	141.8	144
Weight per rod (lb)	7.0	6.87
Clad material	Zircaloy-4	Zircaloy-4
Clad thickness	0.0265	0.0265
Fuel-clad gap (inches)	0.0042	0.0035
Fill gas used	He	He
Initial gas pressure (psig)	415	450
Nitrogen content of fill gas (percent)	3.0	—
177 subassemblies/core [Ref. B-4]		
Fuel pellet material	Uranium oxide (UO_2)	Uranium oxide (UO_2)
Fuel pellet shape	Dished, chamfered	Cylindrical
Fuel pellet diameter (inches)	0.3686	0.370
Fuel pellet length (inches)	0.435	0.7 (Note 2)
Fuel pellet weight per rod (lbs)	5.58	5.57
Open porosity (percent)	<1%	—
Grain size (microns)	10–14	—
Fuel density (% theoretical)	95	92.5 $\pm$ 1.5
O/U ratio	2–2.02 : 1	—
Smear density (g/cm^3)	9.75	—
Spacer pellet length (inches)	—	0.440
Plenum spring material	Stainless steel 302	304 stainless steel
Plenum spring weight per assembly (lb)	0.42	0.04
Plenum length (inches)	11.72	—
Plenum volume (cubic inches)	1.308	—

Note 1. The 204 fuel rod count is in conflict with written text and a figure in Ref. B-2 that indicates 208 is the loaded fuel rod count; this 204 rod count from the Ref. B-2 table is thought to be in error.

Note 2. This published length represents a significant deviation from fuel pellet length in typical PWR assemblies and is also thought to be in error. Total mass for uranium in the fuel rod is in close agreement with Reference B-4, and use of fuel pellets of smaller dimensions to create an optimized configuration will negate this assumed error.

References

B-1. Office of Civilian Radioactive Waste Management, *Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation: Appendix 2A. Physical Description of LWR Fuel Assemblies*, DOE/RW-0184, Vol. 3, UC-70, -71, and -85, December 1987.

B-2. United States Department of Energy, *Agreement for Transportation, Storage and Disposal Services for TMI-2 Reactor Core*, Agreement No. DE-SC07-84ID12355, GPU Nuclear Corporation [no date].

B-3. Office of Civilian Radioactive Waste Management, *Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation: Appendix 2A. Physical Description of LWR Fuel Assemblies*, DOE/RW-0184, Vol. 3, UC-70, -71, and -85, December 1987.

B-4. B. G. Schnitzler and J. B. Briggs, *TMI-2 Isotopic Inventory Calculations*, EGG-PBS-6798, August 1985.

Appendix C

Data from Spent Fuel Database

Appendix C

Data from Spent Fuel Database

The following information (Table C-1) was captured from data in the Spent Fuel Database maintained by the National Spent Nuclear Fuel Program.[C-1] The weights for the fissile materials and total heavy metals are reported with associated uncertainties. The quantities of both fissile and heavy metals without the associated uncertainties form the basis of quantities reported to the Nuclear Regulatory Commission (NRC) in an NRC-licensed storage facility for the Three Mile Island (TMI) canisters.

There are varying, reported numbers of TMI Unit 2 (TMI-2) canisters between those shipped from TMI by GPU to the INEEL (344) and those eventually placed in storage (341) pending packaging and shipment to the repository. Between 1988 and 1995, the sample debris and metallurgical samples from some of the canisters were returned to Test Area North and placed into three canisters (D-153, D-388, and D-389). Canister D-388 was also contains samples the Halden, H. B. Robins, and Dresden reactors. In 2001, canister D-388 and D-389 were combined leaving only two canisters of sample material.[C-2] Canisters D-153 and D-388 are currently stored in a 125-B Cask in the Test Area North Hot Shop and are not candidates for repository disposal at this time.

For a simplifying assumption, the blend of Pu-239 and Pu-241 should be modeled at strictly Pu-239 because the small, fractional portion of Pu-241 can be adequately represented by the Pu-239. Furthermore, although the balance of Pu [total Pu – (Pu-239 + Pu-241)] can be assumed to be Pu-240, its presence in any package should be neglected. Any Pu-240 contribution as a parasitic neutron capture agent will disappear with time (6.56 E03 year half-life); therefore, to neglect its initial presence represents a more conservative criticality modeling approach.

Table C-1. TMI canister inventories.

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
D101	7.53 ± 2.00	291.60 ± 79.40	0.420 ± 0.123	0.447 ± 0.131
D104	7.45 ± 2.26	275.50 ± 89.90	0.361 ± 0.139	0.384 ± 0.148
D105	7.09 ± 0.76	332.90 ± 35.30	0.637 ± 0.067	0.678 ± 0.071
D106	7.25 ± 0.78	340.30 ± 36.10	0.651 ± 0.068	0.693 ± 0.073
D107	3.61 ± 1.62	161.90 ± 74.70	0.282 ± 0.137	0.310 ± 0.148
D108	7.17 ± 0.77	336.50 ± 35.70	0.644 ± 0.068	0.685 ± 0.072
D110	4.09 ± 0.44	192.00 ± 20.50	0.367 ± 0.039	0.391 ± 0.041
D111	7.16 ± 0.77	336.30 ± 15.70	0.643 ± 0.067	0.684 ± 0.072
D112	6.79 ± 0.73	318.90 ± 33.80	0.610 ± 0.064	0.649 ± 0.068
D113	4.49 ± 0.48	210.80 ± 22.50	0.403 ± 0.042	0.429 ± 0.045
D114	1.55 ± 0.17	72.60 ± 8.10	0.139 ± 0.015	0.148 ± 0.016
D116	7.39 ± 0.79	347.00 ± 36.80	0.664 ± 0.070	0.706 ± 0.074
D117	6.85 ± 0.73	321.50 ± 34.10	0.615 ± 0.065	0.654 ± 0.069
D118	0.24 ± 0.06	11.40 ± 2.80	0.022 ± 0.005	0.023 ± 0.006
D119	10.06 ± 2.60	376.50 ± 97.30	0.506 ± 0.131	0.539 ± 0.139

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
D120	0.86 ± 0.11	40.60 ± 5.10	0.078 ± 0.010	0.083 ± 0.010
D121	4.09 ± 0.44	191.80 ± 20.50	0.367 ± 0.039	0.390 ± 6.041
D122	8.94 ± 2.42	339.80 ± 96.20	0.472 ± 0.149	0.502 ± 0.158
D125	6.82 ± 0.73	320.30 ± 34.00	0.613 ± 0.064	0.652 ± 0.069
D126	6.38 ± 0.68	299.60 ± 31.80	0.573 ± 0.060	0.610 ± 0.064
D127	6.94 ± 0.74	325.50 ± 34.60	0.623 ± 0.065	0.663 ± 0.070
D128	6.96 ± 0.75	326.50 ± 34.70	0.624 ± 0.066	0.665 ± 0.070
D129	7.85 ± 1.76	353.80 ± 79.10	0.678 ± 0.152	0.734 ± 0.164
D131	6.53 ± 0.70	306.50 ± 32.70	0.586 ± 0.062	0.624 ± 0.066
D132	7.31 ± 0.78	343.20 ± 36.40	0.656 ± 0.069	0.698 ± 0.073
D136.	1.62 ± 0.68	73.80 ± 30.80	0.133 ± 0.055	0.144 ± 0.060
D137	3.64 ± 0.39	170.60 ± 18.20	0.326 ± 0.034	0.347 ± 0.037
D138	0.51 ± 0.21	24.00 ± 10.10	0.046 ± 0.019	0.049 ± 0.020
D139	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
D140	0.40 ± 0.24	18.70 ± 11.10	0.036 ± 0.021	0.038 ± 0.023
D141	2.76 ± 0.86	130.10 ± 41.20	0.230 ± 0.075	0.250 ± 0.082
D142	6.30 ± 0.68	295.60 ± 31.40	0.565 ± 0.059	0.602 ± 0.063
D143	7.52 ± 0.81	352.90 ± 37.40	0.675 ± 0.071	0.718 ± 0.075
D144	6.94 ± 0.74	325.80 ± 34.60	0.623 ± 0.065	0.663 ± 0.070
D145	7.98 ± 0.85	374.30 ± 39.70	0.716 ± 0.075	0.762 ± 0.080
D146	8.06 ± 0.86	378.40 ± 40.10	0.724 ± 0.076	0.770 ± 0.081
D147	7.84 ± 0.84	368.10 ± 39.00	0.704 ± 0.074	0.749 ± 0.079
D148	7.90 ± 0.85	370.80 ± 39.30	0.709 ± 0.074	0.755 ± 0.079
D149	7.86 ± 0.84	369.10 ± 39.10	0.706 ± 0.074	0.751 ± 0.079
D150	7.35 ± 0.79	344.80 ± 36.60	0.660 ± 0.069	0.702 ± 0.074
D151	8.23 ± 0.88	386.50 ± 41.00	0.739 ± 0.077	0.787 ± 0.083
D152	6.86 ± 0.73	322.00 ± 34.20	0.616 ± 0.065	0.655 ± 0.069
D153	0.27 ± 0.25	12.70 ± 11.90	0.024 ± 0.023	0.026 ± 0.024
D154	5.74 ± 1.58	214.60 ± 59.00	0.288 ± 0.079	0.307 ± 0.084
D155	0.24 ± 0.25	10.50 ± 10.90	0.020 ± 0.021	0.022 ± 0.023
D156	8.13 ± 0.87	381.70 ± 40.50	0.730 ± 0.077	0.777 ± 0.082
D157	6.97 ± 0.75	327.20 ± 34.70	0.626 ± 0.066	0.666 ± 0.070
D158	7.62 ± 0.82	357.70 ± 37.90	0.684 ± 0.072	0.728 ± 0.076
D159	0.20 ± 0.06	9.50 ± 2.80	0.018 ± 0.005	0.019 ± 0.006
D160	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
D161	7.45 ± 0.80	349.60 ± 37.10	0.669 ± 0.070	0.712 ± 0.075

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
D162	6.31 ± 1.79	327.10 ± 91.30	0.664 ± 0.184	0.726 ± 0.200
D163	7.29 ± 0.78	342.00 ± 36.30	0.654 ± 0.069	0.696 ± 0.073
D164	6.75 ± 0.72	317.00 ± 33.60	0.606 ± 0.064	0.645 ± 0.068
D165	7.32 ± 0.78	343.60 ± 36.50	0.657 ± 0.069	0.699 ± 0.073
D166	8.11 ± 0.87	380.50 ± 40.40	0.728 ± 0.076	0.775 ± 0.081
D167	4.93 ± 0.53	231.50 ± 24.60	0.443 ± 0.047	0.471 ± 0.050
D168	3.42 ± 0.38	160.50 ± 17.80	0.307 ± 0.034	0.327 ± 0.036
D169	7.66 ± 0.82	359.30 ± 38.10	0.687 ± 0.072	0.731 ± 0.077
D170	6.97 ± 0.75	327.00 ± 34.70	0.625 ± 0.066	0.666 ± 0.070
D171	4.88 ± 0.52	228.90 ± 24.40	0.438 ± 0.046	0.466 ± 0.049
D172	8.78 ± 2.42	332.60 ± 96.20	0.458 ± 0.149	0.488 ± 0.158
D173	7.67 ± 1.99	327.10 ± 85.60	0.522 ± 0.139	0.563 ± 0.151
D174	1.49 ± 0.17	69.70 ± 7.80	0.133 ± 0.015	0.142 ± 0.016
D175	3.15 ± 0.34	148.00 ± 15.90	0.283 ± 0.030	0.301 ± 0.032
D176	6.00 ± 0.64	281.80 ± 29.90	0.539 ± 0.057	0.574 ± 0.060
D177	7.62 ± 0.82	357.40 ± 37.90	0.684 ± 0.072	0.728 ± 0.076
D178	5.78 ± 2.02	208.70 ± 80.30	0.259 ± 0.124	0.276 ± 0.132
D179	5.61 ± 0.60	263.20 ± 28.00	0.503 ± 0.053	0.536 ± 0.056
D180	8.40 ± 0.90	394.10 ± 41.80	0.754 ± 0.079	0.802 ± 0.084
D181	7.55 ± 0.81	354.30 ± 37.60	0.678 ± 0.071	0.721 ± 0.076
D182	3.42 ± 0.37	160.40 ± 17.20	0.307 ± 0.032	0.326 ± 0.035
D183	5.32 ± 0.57	249.90 ± 26.6	0.478 ± 0.050	0.509 ± 0.054
D184	2.19 ± 0.25	102.70 ± 11.60	0.196 ± 0.022	0.209 ± 0.023
D185	6.14 ± 0.66	287.90 ± 30.60	0.551 ± 0.058	0.586 ± 0.062
D186	6.87 ± 0.74	322.50 ± 34.20	0.617 ± 0.065	0.656 ± 0.069
D187	7.52 ± 1.92	281.50 ± 71.80	0.378 ± 0.096	0.403 ± 0.103
D188	8.42 ± 0.90	395.30 ± 41.90	0.756 ± 0.079	0.805 ± 0.084
D189	7.65 ± 0.82	358.90 ± 38.10	0.686 ± 0.072	0.730 ± 0.077
D190	7.15 ± 1.94	267.40 ± 72.50	0.359 ± 0.097	0.383 ± 0.104
D191	5.64 ± 2.74	211.10 ± 103.00	0.284 ± 0.138	0.302 ± 0.147
D192	8.90 ± 2.31	343.30 ± 92.10	0.491 ± 0.142	0.522 ± 0.152
D193	9.41 ± 2.49	351.90 ± 93.00	0.473 ± 0.125	0.504 ± 0.133
D194	3.96 ± 0.43	185.60 ± 19.80	0.355 ± 0.037	0.378 ± 0.040
D195	6.93 ± 1.47	278.70 ± 58.80	0.429 ± 0.092	0.457 ± 0.098
D196	6.77 ± 0.73	317.90 ± 33.70	0.608 ± 0.064	0.647 ± 0.068
D197	6.79 ± 0.73	318.90 ± 33.80	0.610 ± 0.064	0.649 ± 0.068

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
D198	0.21 ± 0.06	10.00 ± 2.80	0.019 ± 0.005	0.020 ± 0.006
D199	6.21 ± 0.67	291.30 ± 30.90	0.557 ± 0.058	0.593 ± 0.062
D200	0.10 ± 0.06	4.50 ± 2.60	0.009 ± 0.005	0.009 ± 0.005
D201	0.20 ± 0.06	9.50 ± 2.80	0.018 ± 0.005	0.019 ± 0.006
D203	6.47 ± 1.69	349.50 ± 89.20	0.716 ± 0.181	0.781 ± 0.198
D204	4.24 ± 0.46	198.90 ± 21.20	0.381 ± 0.040	0.405 ± 0.043
D205	3.66 ± 0.40	171.70 ± 18.40	0.328 ± 0.035	0.349 ± 0.037
D206	8.60 ± 2.66	322.00 ± 106.00	0.434 ± 0.164	0.462 ± 0.174
D207	5.47 ± 0.59	256.50 ± 27.3	0.491 ± 0.052	0.522 ± 0.055
D208	2.99 ± 0.32	140.30 ± 15.10	0.268 ± 0.029	0.286 ± 0.030
D209	5.87 ± 0.63	275.60 ± 29.30	0.527 ± 0.055	0.561 ± 0.059
D210	2.62 ± 0.29	122.90 ± 13.30	0.235 ± 0.025	0.250 ± 0.027
D211	5.48 ± 0.84	254.80 ± 38.80	0.479 ± 0.072	0.513 ± 0.077
D212	2.45 ± 0.27	115.00 ± 12.50	0.220 ± 0.02	0.234 ± 0.025
D213	5.13 ± 0.55	240.80 ± 25.60	0.461 ± 0.048	0.490 ± 0.052
D214	6.40 ± 0.69	300.60 ± 31.90	0.575 ± 0.060	0.612 ± 0.064
D215	9.11 ± 2.61	347.70 ± 104.00	0.487 ± 0.161	0.518 ± 0.171
D216	8.07 ± 2.47	297.20 ± 98.20	0.386 ± 0.152	0.411 ± 0.162
D217	4.82 ± 0.52	226.20 ± 24.10	0.433 ± 0.046	0.460 ± 0.049
D218	2.01 ± 0.22	94.20 ± 10.30	0.180 ± 0.019	0.192 ± 0.021
D219	9.02 ± 2.35	369.00 ± 95.70	0.613 ± 0.158	0.663 ± 0.171
D220	3.77 ± 1.21	171.60 ± 55.40	0.309 ± 0.102	0.335 ± 0.110
D221	6.61 ± 1.74	330.20 ± 86.30	0.663 ± 0.173	0.725 ± 0.189
D222	7.18 ± 1.87	275.90 ± 74.40	0.392 ± 0.115	0.417 ± 0.123
D223	8.55 ± 2.13	331.60 ± 84.90	0.478 ± 0.131	0.509 ± 0.140
D224	5.56 ± 1.38	248.90 ± 62.10	0.477 ± 0.119	0.518 ± 0.129
D225	2.27 ± 0.25	106.70 ± 11.60	0.204 ± 0.022	0.217 ± 0.023
D226	6.46 ± 1.60	324.90 ± 79.70	0.651 ± 0.159	0.709 ± 0.173
D227	5.59 ± 1.46	337.30 ± 87.80	0.717 ± 0.187	0.788 ± 0.205
D228	7.06 ± 1.85	312.50 ± 81.90	0.600 ± 0.157	0.654 ± 0.171
D229	5.71 ± 1.50	302.10 ± 79.70	0.618 ± 0.163	0.677 ± 0.179
D230	5.49 ± 1.47	302.10 ± 79.70	0.625 ± 0.164	0.686 ± 0.180
D231	8.75 ± 2.22	327.30 ± 83.00	0.440 ± 0.112	0.468 ± 0.119
D232	9.29 ± 2.40	369.20 ± 95.70	0.577 ± 0.151	0.621 ± 0.163
D233	8.09 ± 1.82	342.90 ± 76.60	0.593 ± 0.133	0.638 ± 0.143

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
D234	8.75 ± 2.22	327.30 ± 83.00	0.440 ± 0.112	0.468 ± 0.119
D235	8.42 ± 2.31	319.10 ± 92.00	0.440 ± 0.142	0.469 ± 0.151
D236	6.12 ± 1.58	309.10 ± 78.70	0.620 ± 0.157	0.676 ± 0.171
D237	6.79 ± 1.77	305.90 ± 80.60	0.518 ± 0.138	0.562 ± 0.150
D238	6.35 ± 1.70	309.10 ± 82.30	0.615 ± 0.164	0.672 ± 0.179
D239	7.62 ± 2.22	285.50 ± 88.60	0.386 ± 0.138	0.411 ± 0.147
D240	4.72 ± 1.30	284.60 ± 78.40	0.605 ± 0.167	0.665 ± 0.183
D241	9.10 ± 2.35	435.50 ± 113.00	0.861 ± 0.222	0.941 ± 0.243
D242	6.74 ± 1.81	316.10 ± 85.40	0.620 ± 0.168	0.677 ± 0.184
D243	6.14 ± 1.67	295.00 ± 80.50	0.584 ± 0.160	0.638 ± 0.175
D244	5.81 ± 1.55	302.10 ± 79.70	0.615 ± 0.162	0.673 ± 0.177
D245	7.35 ± 1.94	284.90 ± 75.60	0.419 ± 0.114	0.449 ± 0.122
D246	6.19 ± 1.62	305.60 ± 80.50	0.611 ± 0.161	0.668 ± 0.177
D247	6.36 ± 1.02	291.90 ± 46.10	0.559 ± 0.088	0.600 ± 0.095
D248	8.82 ± 2.34	336.80 ± 93.10	0.472 ± 0.144	0.503 ± 0.153
D250	8.14 ± 2.13	312.20 ± 84.80	0.441 ± 0.131	0.469 ± 0.140
D251	7.34 ± 1.87	274.50 ± 70.00	0.369 ± 0.094	0.393 ± 0.100
D252	8.75 ± 2.22	327.30 ± 83.00	0.440 ± 0.112	0.468 ± 0.119
D253	8.75 ± 2.26	336.20 ± 90.00	0.477 ± 0.139	0.508 ± 0.148
D254	7.17 ± 1.92	309.30 ± 84.10	0.526 ± 0.146	0.570 ± 0.158
D255	5.83 ± 1.59	305.60 ± 83.00	0.623 ± 0.169	0.683 ± 0.185
D256	9.22 ± 2.33	344.80 ± 87.40	0.463 ± 0.117	0.493 ± 0.125
D257	7.96 ± 2.06	344.70 ± 89.80	0.559 ± 0.148	0.605 ± 0.160
D258	8.78 ± 2.39	333.10 ± 95.10	0.461 ± 0.147	0.491 ± 0.157
D259	8.61 ± 2.21	331.70 ± 87.90	0.472 ± 0.136	0.503 ± 0.145
D260	9.66 ± 2.42	373.80 ± 96.40	0.537 ± 0.149	0.571 ± 0.159
D261	7.15 ± 1.90	267.40 ± 71.30	0.359 ± 0.096	0.383 ± 0.102
D262	5.42 ± 1.91	245.60 ± 86.40	0.412 ± 0.150	0.449 ± 0.162
D263	9.03 ± 2.29	348.50 ± 91.10	0.498 ± 0.141	0.531 ± 0.150
D264	5.62 ± 1.48	288.00 ± 76.30	0.583 ± 0.155	0.639 ± 0.170
D265	6.58 ± 1.69	246.30 ± 63.20	0.331 ± 0.085	0.352 ± 0.090
D266	5.18 ± 1.36	312.70 ± 82.00	0.664 ± 0.174	0.731 ± 0.191
D267	8.12 ± 2.12	351.70 ± 91.50	0.571 ± 0.149	0.618 ± 0.161
D268	8.09 ± 2.06	302.60 ± 77.00	0.407 ± 0.103	0.433 ± 0.110

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
D269	7.32 ± 1.87	282.80 ± 74.40	0.405 ± 0.115	0.431 ± 0.123
D270	7.98 ± 2.09	309.50 ± 81.40	0.456 ± 0.123	0.489 ± 0.132
D271	8.33 ± 2.16	323.70 ± 85.70	0.460 ± 0.128	0.492 ± 0.137
D272	5.64 ± 1.54	302.10 ± 82.20	0.620 ± 0.169	0.680 ± 0.185
D273	7.70 ± 2.03	365.20 ± 96.90	0.720 ± 0.191	0.786 ± 0.209
D274	5.81 ± 1.53	302.10 ± 79.70	0.615 ± 0.162	0.673 ± 0.178
D275	5.63 ± 1.08	247.20 ± 44.30	0.435 ± 0.072	0.463 ± 0.077
D276	8.49 ± 2.18	371.20 ± 95.70	0.616 ± 0.162	0.665 ± 0.174
D277	6.87 ± 1.84	330.10 ± 88.70	0.654 ± 0.176	0.715 ± 0.192
D278	5.77 ± 1.50	347.90 ± 90.40	0.739 ± 0.192	0.813 ± 0.211
D279	7.02 ± 2.01	303.80 ± 89.10	0.493 ± 0.153	0.533 ± 0.164
D280	6.74 ± 1.93	350.50 ± 98.70	0.712 ± 0.199	0.780 ± 0.217
D281	5.61 ± 1.50	281.00 ± 74.70	0.565 ± 0.150	0.618 ± 0.164
D282	5.10 ± 1.62	258.50 ± 81.20	0.522 ± 0.162	0.572 ± 0.177
D283	7.57 ± 1.95	400.50 ± 103.00	0.819 ± 0.211	0.898 ± 0.232
D284	6.68 ± 1.41	329.60 ± 70.60	0.654 ± 0.141	0.709 ± 0.153
D285	8.93 ± 2.31	343.10 ± 92.10	0.486 ± 0.142	0.518 ± 0.152
D286	4.58 ± 1.46	203.40 ± 65.70	0.390 ± 0.126	0.425 ± 0.136
D287	6.61 ± 2.51	228.10 ± 99.80	0.252 ± 0.155	0.268 ± 0.165
D288	6.37 ± 1.72	312.60 t 84.60	0.624 ± 0.169	0.682 ± 0.185
D289	5.87 ± 1.57	259.80 ± 69.40	0.499 ± 0.133	0.543 ± 0.145
D290	7.99 ± 2.26	300.80 ± 89.90	0.409 ± 0.139	0.436 ± 0.148
D291	8.37 ± 2.18	323.60 ± 84.80	0.473 ± 0.126	0.507 ± 0.136
D292	5.21 ± 1.38	267.00 ± 71.40	0.541 ± 0.145	0.592 ± 0.159
D293	5.58 ± 1.49	302.10 ± 79.70	0.622 ± 0.164	0.683 ± 0.179
D294	8.03 ± 2.07	337.70 ± 88.10	0.531 ± 0.141	0.572 ± 0.153
D295	7.40 ± 1.98	316.4 ± 85.70	0.530 ± 0.147	0.573 ± 0.159
D296	6.66 ± 1.74	340.70 ± 89.80	0.689 ± 0.182	0.755 ± 0.199
D297	6.29 ± 1.88	236.00 ± 74.70	0.319 ± 0.116	0.340 ± 0.123
D298	7.59 ± 1.59	375.60 ± 78.60	0.745 ± 0.157	0.809 ± 0.170
D299	9.37 ± 2.60	353.90 ± 104.00	0.485 ± 0.160	0.516 ± 0.170
D300	6.90 ± 1.66	386.50 ± 90.80	0.799 ± 0.187	0.871 ± 0.204
D301	5.89 ± 1.55	309.10 ± 81.30	0.631 ± 0.166	0.691 ± 0.182
D302	3.85 ± 0.41	180.60 ± 19.30	0.345 ± 0.036	0.368 ± 0.039

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
D303	7.05 ± 1.83	273.50 ± 72.90	0.395 ± 0.113	0.420 ± 0.120
D304	7.44 ± 2.07	331.60 ± 93.20	0.636 ± 0.179	0.691 ± 0.193
D305	8.62 ± 2.29	329.20 ± 91.00	0.461 ± 0.141	0.491 ± 0.150
D306	8.38 ± 2.13	323.30 ± 84.80	0.462 ± 0.131	0.492 ± 0.140
D307	8.93 ± 2.27	334.30 ± 84.70	0.449 ± 0.114	0.478 ± 0.121
D308	4.89 ± 1.38	190.90 ± 54.90	0.279 ± 0.085	0.297 ± 0.091
D309	6.37 ± 0.68	299.10 ± 31.80	0.572 ± 0.060	0.609 ± 0.064
D310	3.10 ± 0.34	145.40 ± 15.60	0.278 ± 0.030	0.296 ± 0.031
D311	5.35 ± 0.57	251.10 ± 26.70	0.480 ± 0.050	0.511 ± 0.054
D312	4.14 ± 0.45	194.40 ± 20.70	0.372 ± 0.039	0.396 ± 0.042
D313	5.54 ± 0.59	259.90 ± 27.60	0.497 ± 0.052	0.529 ± 0.056
D314	3.44 ± 0.37	161.60 ± 17.30	0.309 ± 0.033	0.329 ± 0.035
D315	2.51 ± 0.27	118.00 ± 12.80	0.226 ± 0.024	0.240 ± 0.026
D316	4.51 ± 0.49	211.80 ± 22.60	0.405 ± 0.043	0.431 ± 0.045
D317	2.60 ± 0.28	122.10 ± 13.20	0.233 ± 0.025	0.248 ± 0.027
D318	6.51 ± 0.70	305.60 ± 32.40	0.584 ± 0.061	0.622 ± 0.065
D319	2.18 ± 0.26	102.10 ± 12.20	0.195 ± 0.023	0.208 ± 0.025
D320	4.44 ± 0.48	208.20 ± 22.20	0.398 ± 0.042	0.424 ± 0.045
D321	6.73 ± 0.72	316.00 ± 33.50	0.604 ± 0.063	0.643 ± 0.068
D322	4.44 ± 0.48	208.50 ± 22.20	0.399 ± 0.042	0.424 ± 0.045
D323	2.44 ± 0.30	114.60 ± 14.00	0.219 ± 0.026	0.233 ± 0.028
D324	5.33 ± 0.57	250.30 ± 26.70	0.479 ± 0.051	0.510 ± 0.054
D325	7.06 ± 0.76	331.30 ± 35.20	0.634 ± 0.066	0.674 ± 0.071
D326	3.47 ± 0.39	162.90 ± 17.90	0.312 ± 0.034	0.332 ± 0.036
D327	8.24 ± 0.88	386.70 ± 41.00	0.740 ± 0.078	0.787 ± O.083
D328	6.50 ± 0.70	304.80 ± 32.40	0.583 ± 0.061	0.621 ± 0.065
D329	8.16 ± 0.87	383.10 ± 40.60	0.733 ± 0.077	0.780 ± 0.082
D330	8.58 ± 0.92	402.70 ± 42.70	0.770 ± 0.081	0.820 ± 0.086
D331	8.51 ± 0.91	399.30 ± 42.30	0.764 ± 0.080	0.813 ± 0.085
D332	8.47 ± 0.91	397.40 ± 42.10	0.760 ± 0.080	0.809 ± 0.085
D333	7.14 ± 0.77	335.30 ± 35.60	0.641 ± 0.067	0.683 ± 0.072
D334	7.29 ± 0.78	342.20 ± 36.30	0.655 ± 0.069	0.697 ± 0.073
D335	7.61 ± 0.81	357.00 ± 37.90	0.683 ± 0.072	0.727 ± 0.076
D336	7.66 ± 0.82	359.60 ± 38.10	0.688 ± 0.072	0.732 ± 0.077

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
D337	7.88 ± 0.84	369.80 ± 39.20	0.707 ± 0.074	0.753 ± 0.079
D338	7.39 ± 0.79	346.70 ± 36.80	0.663 ± 0.070	0.706 ± 0.074
D339	7.71 ± 0.83	362.00 ± 38.40	0.692 ± 0.073	0.737 ± 0.077
D340	7.82 ± 0.84	367.20 ± 38.90	0.702 ± 0.074	0.747 ± 0.078
D341	7.86 ± 0.84	369.10 ± 39.10	0.706 ± 0.074	0.751 ± 0.079
D342	8.06 ± 0.86	378.10 ± 40.10	0.723 ± 0.076	0.770 ± 0.081
D343	7.40 ± 0.79	347.40 ± 36.90	0.665 ± 0.070	0.707 ± 0.074
D344	4.16 ± 0.45	195.20 ± 20.90	0.373 ± 0.039	0.397 ± 0.042
D346	7.92 ± 0.85	371.70 ± 39.40	0.711 ± 0.075	0.757 ± 0.079
D347	7.43 ± 0.80	348.90 ± 37.00	0.667 ± 0.070	0.710 ± 0.075
D348	7.96 ± 0.85	373.40 ± 39.60	0.714 ± 0.075	0.760 ± 0.080
D349	7.87 ± 0.84	369.30 ± 39.20	0.706 ± 0.074	0.752 ± 0.079
D350	8.45 ± 0.90	396.70 ± 42.10	0.759 ± 0.080	0.807 ± 0.085
D351	7.93 ± 0.85	372.20 ± 39.50	0.712 ± 0.075	0.758 ± 0.080
D352	7.98 ± 0.85	374.60 ± 39.70	0.716 ± 0.075	0.762 ± 0.080
D353	7.85 ± 0.84	368.60 ± 39.10	0.705 ± 0.074	0.750 ± 0.079
D354	7.74 ± 0.83	363.10 ± 38.50	0.695 ± 0.073	0.739 ± 0.078
D355	7.45 ± 0.80	349.60 ± 37.10	0.669 ± 0.070	0.712 ± 0.075
D356	7.68 ± 0.82	360.50 ± 38.20	0.690 ± 0.072	0.734 ± 0.077
D357	8.30 ± 0.89	389.30 ± 41.30	0.745 ± 0.078	0.792 ± 0.083
D358	8.19 ± 0.88	384.30 ± 40.80	0.735 ± 0.077	0.782 ± 0.082
D359	7.81 ± 0.84	366.50 ± 38.90	0.701 ± 0.074	0.746 ± 0.078
D360	7.61 ± 0.81	357.00 ± 37.90	0.683 ± 0.072	0.727 ± 0.076
D361	8.50 ± 0.91	398.80 ± 42.30	0.763 ± 0.080	0.812 ± 0.085
D362	8.24 ± 0.88	386.70 ± 41.00	0.740 ± 0.078	0.787 ± 0.083
D363	7.75 ± 0.83	363.60 ± 38.60	0.695 ± 0.073	0.740 ± 0.078
D364	7.16 ± 0.77	336.00 ± 35.70	0.643 ± 0.067	0.684 ± 0.072
D365	6.13 ± 0.66	287.50 ± 30.50	0.550 ± 0.058	0.585 ± 0.062
D366	7.63 ± 0.82	358.20 ± 38.00	0.685 ± 0.072	0.729 ± 0.077
D367	7.56 ± 0.81	354.80 ± 37.60	0.679 ± 0.071	0.722 ± 0.076
D368	7.63 ± 0.82	358.20 ± 38.00	0.685 ± 0.072	0.729 ± 0.077
D369	7.35 ± 0.79	345.10 ± 36.60	0.660 ± 0.069	0.702 ± 0.074
D370	8.15 ± 0.87	382.40 ± 40.60	0.731 ± 0.077	0.778 ± 0.082
D371	7.16 ± 0.77	336.30 ± 35.70	0.643 ± 0.067	0.684 ± 0.072

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
D373	5.24 ± 0.56	246.10 ± 26.20	0.471 ± 0.049	0.501 ± 0.053
D374	3.12 ± 0.34	146.60 ± 15.70	0.280 ± 0.030	0.298 ± 0.032
D376	8.18 ± 0.88	383.90 ± 40.70	0.734 ± 0.077	0.781 ± 0.082
D378	4.39 ± 0.47	206.10 ± 22.00	0.394 ± 0.042	0.419 ± 0.044
D379	8.02 ± 0.86	376.50 ± 39.90	0.720 ± 0.076	0.766 ± 0.080
D384	8.28 ± 0.89	388.60 ± 41.20	0.743 ± 0.078	0.791 ± 0.083
D385	7.38 ± 0.79	346.30 ± 36.70	0.662 ± 0.069	0.705 ± 0.074
D386	7.39 ± 0.79	346.70 ± 36.80	0.663 ± 0.070	0.706 ± 0.074
D390	7.58 ± 0.81	355.50 ± 37.70	0.680 ± 0.071	0.724 ± 0.076
D391	7.95 ± 0.85	372.90 ± 39.60	0.713 ± 0.075	0.759 ± 0.080
F401	0.11 ± 0.03	5.20 ± 1.20	0.010 ± 0.002	0.011 ± 0.002
F402	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F403	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F404	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F406	0.10 ± 0.02	4.50 ± 1.00	0.009 ± 0.002	0.009 ± 0.002
F407	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F408	0.04 ± 0.01	1.70 ± 0.40	0.003 ± 0.001	0.003 ± 0.001
F409	0.12 ± 0.03	5.70 ± 1.30	0.011 ± 0.002	0.012 ± 0.003
F410	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F411	0.03 ± 0.01	1.40 ± 0.30	0.003 ± 0.001	0.003 ± 0.001
F412	0.12 ± 0.03	5.70 ± 1.30	0.011 ± 0.002	0.012 ± 0.003
F413	0.53 ± 0.12	25.00 ± 5.60	0.048 ± 0.011	0.051 ± 0.011
F414	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F415	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F416	0.02 ± 0.00	0.70 ± 0.20	0.001 ± 0.000	0.001 ± 0.000
F417	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F418	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F419	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F420	0.13 ± 0.03	6.20 ± 1.40	0.012 ± 0.003	0.013 ± 0.003
F422	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F424	1.14 ± 0.26	53.50 ± 12.10	0.102 ± 0.023	0.109 ± 0.025
F425	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F426	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F427	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
F428	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F429	0.00 ± 0.00	0.00 ± 0.00	0.00 ± 0.000	0.000 ± 0.000
F430	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F431	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F432	0.23 ± 0.05	10.90 ± 2.50	0.021 ± 0.005	0.022 ± 0.005
F433	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F434	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F435	0.28 ± 0.06	13.10 ± 3.00	0.025 ± 0.006	0.027 ± 0.006
F436	0.78 ± 0.18	36.40 ± 8.20	0.070 ± 0.016	0.074 ± 0.017
F437	0.73 ± 0.17	34.30 ± 7.70	0.066 ± 0.015	0.070 ± 0.016
F438	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F443	0.05 ± 0.01	2.10 ± 0.50	0.004 ± 0.001	0.004 ± 0.001
F444	1.17 ± 0.27	55.00 ± 12.40	0.105 ± 0.024	0.112 ± 0.025
F445	0.91 ± 0.21	42.60 ± 9.60	0.081 ± 0.018	0.087 ± 0.020
F446	0.63 ± 0.14	29.50 ± 6.70	0.056 ± 0.013	0.060 ± 0.014
F447	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
F448	0.24 ± 0.06	11.40 ± 2.60	0.022 ± 0.005	0.023 ± 0.005
F449	1.28 ± 0.29	60.20 ± 13.60	0.115 ± 0.026	0.123 ± 0.028
F450	1.33 ± 0.30	62.60 ± 14.10	0.120 ± 0.027	0.127 ± 0.029
F451	1.46 ± 0.33	68.50 ± 15.50	0.131 ± 0.030	0.140 ± 0.031
F452	1.22 ± 0.28	57.40 ± 13.00	0.110 ± 0.025	0.117 ± 0.026
F453	2.42 ± 0.55	113.80 ± 25.70	0.218 ± 0.049	0.232 ± 0.052
F454	2.74 ± 0.62	128.70 ± 29.10	0.246 ± 0.056	0.262 ± 0.059
F455	0.18 ± 0.04	8.60 ± 1.90	0.016 ± 0.004	0.017 ± 0.004
F456	0.08 ± 0.02	3.60 ± 0.80	0.007 ± 0.002	0.007 ± 0.002
F457	2.31 ± 0.52	108.50 ± 24.50	0.208 ± 0.047	0.221 ± 0.050
F458	2.40 ± 0.54	112.60 ± 25.40	0.215 ± 0.049	0.229 ± 0.052
F459	3.82 ± 0.87	179.40 ± 40.60	0.343 ± 0.077	0.365 ± 0.082
F460	2.41 ± 0.55	113.30 ± 25.60	0.217 ± 0.049	0.231 ± 0.052
F461	3.25 ± 0.74	152.30 ± 34.40	0.291 ± 0.066	0.310 ± 0.070
F462	3.46 ± 0.78	162.50 ± 36.70	0.311 ± 0.070	0.331 ± 0.075
F466	2.03 ± 0.46	95.20 ± 21.50	0.182 ± 0.041	0.194 ± 0.044
F467	2.36 ± 0.53	110.70 ± 25.00	0.212 ± 0.048	0.225 ± 0.051
F468	0.53 ± 0.12	25.00 ± 5.60	0.048 ± 0.011	0.051 ± 0.011

Table C-1. (continued).

Can I.D.	U-235 (kilograms)	Total Uranium (kilograms)	Pu-239 + Pu-241 (kilograms)	Total plutonium (kilograms)
F469	2.97 ± 0.67	139.50 ± 31.50	0.267 ± 0.060	0.284 ± 0.064
F470	2.51 ± 0.57	118.00 ± 26.70	0.226 ± 0.051	0.240 ± 0.054
F471	4.59 ± 1.04	215.60 ± 48.70	0.412 ± 0.093	0.439 ± 0.099
F473	2.79 ± 0.63	130.90 ± 29.60	0.250 ± 0.056	0.266 ± 0.060
K501	5.62 ± 1.27	263.90 ± 59.70	0.505 ± 0.114	0.537 ± 0.121
K502	0.73 ± 0.17	34.30 ± 7.70	0.066 ± 0.015	0.070 ± 0.016
K504	6.34 ± 1.44	297.50 ± 67.20	0.569 ± 0.128	0.605 ± 0.137
K506	9.42 ± 2.13	441.90 ± 99.90	0.845 ± 0.191	0.900 ± 0.203
K507	2.70 ± 0.61	126.60 ± 28.60	0.242 ± 0.055	0.258 ± 0.058
K508	2.31 ± 0.52	108.50 ± 24.50	0.208 ± 0.047	0.221 ± 0.050
K513	8.10 ± 1.83	380.00 ± 85.90	0.727 ± 0.164	0.774 ± 0.174
K519	0.49 ± 0.10	22.80 ± 5.20	0.044 ± 0.010	0.047 ± 0.010
K520	1.53 ± 0.35	71.60 ± 16.20	0.137 ± 0.031	0.146 ± 0.033
K521	0.37 ± 0.08	17.40 ± 3.90	0.033 ± 0.007	0.035 ± 0.008
K523	0.00 ± 0.00	0.00 ± 0.00	0.000 ± 0.000	0.000 ± 0.000
K563	0.64 ± 0.14	30.00 ± 6.80	0.057 ± 0.013	0.061 ± 0.014
Totals	1801.12	81437.2	147.781	155.005
D130	0.61 ± 0.01	20.40 ± 0.20	0.040 ± 0.010	0.040 ± 0.010

References

C-1. G. Lassahn, *Uranium and Plutonium Content of TMI-2 Defueling Canisters*, EG&G internal technical report, September 1993.

C-2. L. R. Zirker, Engineering Design File EDF-1956, *Contents of TMI Canisters D-153/D-130, D-388 and D-389*, INEEL, March 2001.

Appendix D

TMI-2 Source Term Inventory

Appendix D

TMI-2 Source Term Inventory

The following data in Table D-1 represents the loading information for a combination of the D-153 canister and a repackaged TMI-2 (D-388) canister.

The total heavy metal content reported for canister D-153 in Appendix C represents the as-shipped condition between Three Mile Island and the INEEL. Subsequent analysis of TMI-2 debris from various canisters, and subsequent repackaging lead to an increase in the reported the heavy metal contents for canister D-153 as shown in Table D-1. The details associated with this repackaging effort are documented in an engineering design file.[D-1]

Where the material in D-153 and D-388 retains their characteristics associated solely with TMI-2 debris, there is an allowable straight line (or ratioed) extrapolation that can be applied to any other TMI-2 canister for estimating radionuclide inventories. As an example applicable to the K506 canister, the radionuclide inventory for that canister would be a heavy-metal value of $[(441.9 + 0.9)/19.08] = 23.2075$ times any radionuclide species. Such a calculation would also be applicable to estimating increased thermal power for other canisters.

Table D-1. Radionuclide inventory for the D-153 and D-388 canisters.

Fuel Radionuclide Inventory Worksheet

I. Fuel and Template Information

Fuel Name:	TMI-2 CORE DEBRIS (D-153 & 388)	[1]**Fuel decay start date:**	1979
SNF ID #:	229	**Estimates as of:**	2003
Fuel Units & Descr:	2 - DEBRIS	**Template:**	PWR (Light Water, Zirc, 0 to 5%, U)
Heavy Metal Mass:	BOL=19.08kg; EOL=19.01kg	[2]**Template Burnup(MWd):**	61.92
ROD Storage Site:	INEEL	**Template BOL Heavy Metal Mass (MT):**	0.00176911
Current Location:	INEEL, TAN CASK STORAGE PAD TAN-791	**Template Decay Time:**	20 years

Estimated Canister usage:
18"x15'
2.00

II. Estimates

	m	x_n	x_b	b	y_n	y_b
Radionuclide	**Ci/MWd From Template**	**Nominal Fuel Burnup (MWd)[2]**	**Bounding Fuel Burnup (MWd)[2]**	**Initial Activity (Ci)**	**Nominal Fuel Inventories(Ci)**	**Bounding Fuel Inventories(Ci)**
Ac-227	5.0630E-10	66.57	133.13	0.00E+00	3.37E-08	6.74E-08
Am-241	1.1489E-01	66.57	133.13	0.00E+00	7.65E+00	1.53E+01
Am-242m	3.0733E-04	66.57	133.13	0.00E+00	2.05E-02	4.09E-02
Am-243	6.2661E-04	66.57	133.13	0.00E+00	4.17E-02	8.34E-02
C-14	4.7997E-05	66.57	133.13	0.00E+00	3.20E-03	6.39E-03
Cl-36	8.0313E-07	66.57	133.13	0.00E+00	5.35E-05	1.07E-04
Cm-243	3.6127E-04	66.57	133.13	0.00E+00	2.40E-02	4.81E-02
Cm-244	8.6999E-02	66.57	133.13	0.00E+00	5.79E+00	1.16E+01
Co-60	1.8379E-02	66.57	133.13	0.00E+00	1.22E+00	2.45E+00
Cs-134	6.2548E-03	66.57	133.13	0.00E+00	4.16E-01	8.33E-01
Cs-135	1.4433E-05	66.57	133.13	0.00E+00	9.61E-04	1.92E-03
Cs-137	1.9767E+00	66.57	133.13	0.00E+00	1.32E+02	2.63E+02
Eu-154	6.7603E-02	66.57	133.13	0.00E+00	4.50E+00	9.00E+00
Eu-155	1.4373E-02	66.57	133.13	0.00E+00	9.57E-01	1.91E+00
Fe-55	2.3466E-03	66.57	133.13	0.00E+00	1.56E-01	3.12E-01
H-3	4.8143E-02	66.57	133.13	0.00E+00	3.20E+00	6.41E+00
I-129	9.8288E-07	66.57	133.13	0.00E+00	6.54E-05	1.31E-04
Kr-85	7.4386E-02	66.57	133.13	0.00E+00	4.95E+00	9.90E+00
Np-237	1.0145E-05	66.57	133.13	0.00E+00	6.75E-04	1.35E-03
Pa-231	1.0258E-09	66.57	133.13	0.00E+00	6.83E-08	1.37E-07
Pb-210	1.4163E-11	66.57	133.13	0.00E+00	9.43E-10	1.89E-09
Pm-147	1.9170E-02	66.57	133.13	0.00E+00	1.28E+00	2.55E+00
Pu-238	8.3915E-02	66.57	133.13	0.00E+00	5.59E+00	1.12E+01
Pu-239	1.1628E-02	66.57	133.13	0.00E+00	7.74E-01	1.55E+00
Pu-240	1.5050E-02	66.57	133.13	0.00E+00	1.00E+00	2.00E+00
Pu-241	1.8524E+00	66.57	133.13	0.00E+00	1.23E+02	2.47E+02
Pu-242	6.4260E-05	66.57	133.13	0.00E+00	4.28E-03	8.56E-03
Ra-226	6.0562E-11	66.57	133.13	0.00E+00	4.03E-09	8.06E-09
Ra-228	4.9919E-12	66.57	133.13	0.00E+00	3.32E-10	6.65E-10
Ru-106	1.8330E-05	66.57	133.13	0.00E+00	1.22E-03	2.44E-03
Se-79	1.2379E-05	66.57	133.13	0.00E+00	8.24E-04	1.65E-03
Sn-126	2.5210E-05	66.57	133.13	0.00E+00	1.68E-03	3.36E-03
Sr-90	1.3098E+00	66.57	133.13	0.00E+00	8.72E+01	1.74E+02
Tc-99	3.9357E-04	66.57	133.13	0.00E+00	2.62E-02	5.24E-02
Th-229	6.2968E-11	66.57	133.13	0.00E+00	4.19E-09	8.38E-09
Th-230	1.0362E-08	66.57	133.13	0.00E+00	6.90E-07	1.38E-06
Th-232	5.2891E-12	66.57	133.13	0.00E+00	3.52E-10	7.04E-10
Tl-208	1.9977E-07	66.57	133.13	0.00E+00	1.33E-05	2.66E-05
U-232	5.4490E-07	66.57	133.13	0.00E+00	3.63E-05	7.25E-05
U-233	2.3934E-08	66.57	133.13	0.00E+00	1.59E-06	3.19E-06
U-234	4.4816E-05	66.57	133.13	0.00E+00	2.98E-03	5.97E-03
U-235	-1.4492E-06	66.57	0.00	1.29E-03	1.19E-03	1.29E-03
U-236	7.5711E-06	66.57	133.13	0.00E+00	5.04E-04	1.01E-03
U-238	-2.6129E-07	66.57	0.00	6.21E-03	6.19E-03	6.21E-03
Y-90	1.3101E+00	66.57	133.13	0.00E+00	8.72E+01	1.74E+02
Other Radionuclides					1.26E+02	2.53E+02

Gamma Sources

Photon Energy Group	Total Photons/sec (bounding)
Avg. MeV	
0.0150	1.019E+13
0.0250	2.074E+12
0.0375	2.027E+12
0.0575	2.129E+12
0.0850	1.167E+12
0.1250	8.835E+11
0.2250	1.000E+12
0.3750	4.311E+11
0.5750	9.788E+12
0.8500	2.536E+11
1.2500	3.809E+11
1.7500	7.012E+09
2.2500	1.626E+06
2.7500	1.141E+06
3.5000	1.689E+05
5.0000	7.114E+04
7.0000	8.202E+03
11.0000	9.422E+02

Thermal Power

Nominal Heat Output (Watts)	Bounding Heat Output (Watts)
2.00E+00	**3.99E+00**
Total	Total

III. Template Selection Summary, Burnup Summary, and Checks

Template Selection Summary

	From SFD	Used
Reactor Moderator:	LIGHT WATER	LIGHT WATER
Fuel Cladding:	ZIRC	ZIRC
BOL HM Constituents:	U	U
BOL Enrichment %:	3.125	0 to 5

Basis for Parameter Differences:

Burnup Summary (MWd)[2]

	From SFD	Estimated
Nominal:	60.58	66.57
Bounding:	113.81	133.13

Basis for burnup used in estimate:

Nominal burnup calculated from the heavy metal mass destroyed.
Bounding burnup assumed to be twice nominal burnup.

Checks

References

D-1. L. R. Zirker, Engineering Design File EDF-1956, *Contents of TMI Canisters D-153/D-130, D-388 and D-389*, INEEL, March 2001.