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Source Term Estimates for DOE Spent Nuclear Fuels

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Office of Nuclear Material and Spent Fuel

Enclosure 1

This document was developed and is controlled in accordance with NSNFP procedures. It has been reviewed and determined adequate for Beyond Category 2 consequence, TSPA, shielding, and decay heat analysis. For other uses, the information must be evaluated for adequacy if relied on to support design or decisions important to safety or waste isolation.

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Source Term Estimates for DOE Spent Nuclear Fuels

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Idaho National Engineering and Environmental Laboratory
Idaho Falls, Idaho 83415

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Brett W Carlsen 3/12/03

Brett Carlsen/Layne Pincock — Layne Pincock (Signature) — Date: 3-11-03

National Spent Nuclear Fuel Program
Document Preparer

Henry Loo — Henry Loo (Signature) — Date: 3-11-03

National Spent Nuclear Fuel Program
Technical Lead

Neal Mackay — Neal S MacKay (Signature) — Date: 3-11-03

National Spent Nuclear Fuel Program
Program Support Organization Quality Engineer

Philip Wheatley — D Hill for PDW (Signature) — Date: 12 March 2003

National Spent Nuclear Fuel Program
Program Support Organization Manager

SUMMARY

Spent nuclear fuel owned by the U.S. Department of Energy (DOE) includes hundreds of fuel types from various experimental, research, and production reactors. These fuels currently reside at several DOE sites, universities, and foreign research reactor sites. In accordance with the Record of Decision, Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs (May 1995); all DOE spent nuclear fuel will be consolidated at the Hanford Site, the Savannah River Site, and the Idaho National Engineering and Environmental Laboratory for storage until final disposition at the national repository, which is currently under development.

Safe storage, transportation, and ultimate disposal of these spent nuclear fuels will require safety analyses to support design and licensing of the necessary equipment and facilities. These safety analyses will require radionuclide inventories to represent the radioactive source term that must be accommodated during handling, storage, and disposition of these fuels.

This report provides the results and summarizes the analytical processes employed to estimate the radiological inventories associated with DOE-owned spent nuclear fuel. Based on these estimates, the heat loading and photon emission spectrum for each spent nuclear fuel are also provided. This information will facilitate analyses that support safe storage, handling, transportation, and eventual disposition of these fuels.

ACKNOWLEDGMENTS

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ACRONYMS

BOL	beginning-of-life
DOE	U.S. Department of Energy
EOL	end-of-life
FIS	Fuel Information Sheet
NSNFP	National Spent Nuclear Fuel Program
SFD	Spent Fuel Database
SNF	spent nuclear fuel

Source Term Estimates for DOE Spent Nuclear Fuels

1. PURPOSE

This report provides the results and summarizes the analytical processes employed to estimate the radiological source terms for spent nuclear fuels (SNFs) owned by the U.S. Department of Energy (DOE). Based on the source term estimates, the heat loading and photon spectrum for each SNF are also provided. The results of this analysis will serve as a single, reference document that provides isotopic information with a consistent and documented basis for all DOE-owned SNF intended for repository disposal. This information will facilitate analyses that support safe storage, handling, transportation, and eventual disposition of these fuels. The results of this report are adequate to be used for preclosure and postclosure safety analysis at Yucca Mountain.

2. BACKGROUND

In accordance with the Record of Decision for Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs,[1] DOE SNF will be consolidated at the Hanford Site, the Savannah River Site, and the Idaho National Engineering and Environmental Laboratory. Each storage site is responsible for the safe handling, storage, and final disposition of the DOE SNF in its custody. The National Spent Nuclear Fuel Program (NSNFP) consolidates SNF information from each of the sites and makes it available to support DOE planning and scoping activities as well as design and licensing efforts to enable final repository disposal of DOE SNF.

DOE is responsible for storage and final disposition of nuclear fuel that spans several decades of nuclear research and defense-related material production. To support nuclear nonproliferation objectives, DOE has also taken custody of many foreign research reactor fuels. The SNF presently in DOE custody consists of several hundred different fuel types. Much of the available historical data on these fuels may not meet current quality assurance requirements if needed to demonstrate compliance with repository license criteria. Although this historical data, such as fuel fabrication, operations, and storage records, are incomplete or questionable for some of these fuels, these fuels have been safely handled and stored for many years at DOE storage facilities.

The fuel information currently available at the DOE storage sites is often determined by the records requirements and the intended disposition path at the time the fuel was placed into storage. These requirements and disposition paths were often unique to each of the sites and evolved over time. As a result, the availability and completeness of the radionuclide inventories and associated documentation varies considerably for DOE SNF. Costly characterization of these fuels can be avoided by employing a credible means to obtain a conservative source term estimate for use in repository design, analyses, and licensing activities.

A process for creating a conservative estimate of these SNF source terms was developed by a team of experts representing each of the DOE SNF storage sites. The process relies on precalculated results that provide radionuclide inventories for typical SNFs at a range of decay times. These results are used as templates that are scaled to estimate radionuclide inventories for other similar fuels. The templates were generated using ORIGEN-based calculational techniques described in DOE/SNF/REP-055,[2] which includes discussion and references to relevant experimental data and validation studies. Additional validation studies[3,4,5,6] have been performed that further demonstrate the validity of the model and underlying codes.

To estimate an SNF source term, an appropriate template is selected to model the production of activation products and transuranics by matching the reactor moderator and fuel cladding, constituents, and beginning-of-life (BOL) enrichment. Precalculated radionuclide inventories are extracted from the appropriate template at the desired decay period and then scaled to account for differences in fuel mass and specific burnup. By modeling various combinations of reactor moderator, fuel enrichment, fuel compound, and fuel cladding; templates have been developed to reasonably model a broad range of DOE SNFs.

The template methodology enables a source term estimate to be completed for virtually any DOE SNF for decay dates up to 100 years following reactor shutdown. This process, which was introduced in DOE/SNF/REP-059[7] and further refined in this report, uses available information, conservative assumptions, and similarity principles to estimate SNF radiological inventories. Needless expense and personnel exposure associated with characterization are avoided by applying this process to estimate DOE SNF source terms.

The scope of this report includes all DOE-owned SNF destined for the repository except for Navy SNF. The Navy will be providing source term information separately. Sodium-bonded SNF that is projected to be treated is not included in this report.

3. QUALITY ASSURANCE

The radionuclide inventory estimates presented here have been developed to support preclosure and postclosure licensing and design considerations at the proposed Monitored Geologic Repository near Yucca Mountain, Nevada.

Preliminary dose calculations and scoping studies have indicated that repository performance is relatively insensitive to the form and composition of DOE SNFs. There are three reasons for this. First, DOE SNFs comprise a relatively small fraction (~3% by MTHM) of the total SNF that will be placed in the repository. Second, DOE SNFs are primarily from research, test, and production reactors that are typically low burnup fuels. Third, the DOE standard canister serves as an engineered barrier that provides additional confinement.

A recent study concluded that the DOE SNF standard canister and the canister handling equipment and facilities could be designed to preclude an accident resulting in a breach (i.e., any release of DOE SNF canister during preclosure operations).[8] The Yucca Mountain Project strategy that demonstrates that a preclosure release from a DOE standard canister is not credible is outlined in Reference 9. Even though not credible, preliminary dose calculations have shown that radiological doses from postulated accident scenarios involving a release remain well below the regulatory limit.[10] Similarly, even though analyses show that a postclosure release from DOE SNFs is not expected during the 10,000-year regulatory period, calculations again indicate that doses remain well below the regulatory limit.[11] Further, as noted in Section 8, conservative assumptions have been applied such that the source term estimates are expected to significantly overpredict the actual source terms. The risk associated with uncertainty in the source term estimates for individual DOE SNFs is expected to be much less than indicated in the abovementioned calculations for both preclosure and postclosure. Because of these relatively large margins of safety, the precision of DOE SNF information is sufficient for demonstrating compliance with repository preclosure and postclosure safety requirements.

NSNFP procedures that implement *Quality Assurance Requirements and Description* principles were applied to this activity. PSO 3.03, "Engineering Analyses," requires the validation of models used in NSNFP engineering analyses to ensure that processes, systems, and phenomena are represented to an appropriate level of detail based on the intended use of the results.[12] The estimates provided here rely on two models. First, the templates are created by modeling nuclear reactor fuel depletion using MCNP-ORIGEN2 Coupled Utility Program Code (MOCUP). Detailed discussion of these codes and associated validation is given in Reference 13. Additional studies that further validate the models and calculational techniques used to generate the templates and to demonstrate the applicability of this methodology to a wide variety of DOE SNF are included with References 3, 4, 5, 6, and 14. Second, the template methodology scales precalculated radionuclide inventories from one fuel to model other similar fuels. This methodology was developed by a team of experts representing the INEEL, the Hanford Site, the Savannah River Site, and the Yucca Mountain Project and has been formally documented and reviewed in DOE/SNF/REP-059 (Reference 7).

The templates and associated logic used to determine scaling factors and calculate the source term estimates were codified using Excel 2000. In accordance with PSO 19.01, "Software Control," the software routines and macros employed are uniquely identified and have been independently verified to produce correct results.[15] This was achieved by:

1. Including on the output sheet a comparison of the ratio of heavy metal mass estimated using the methodology to that currently residing in the NSNFP Spent Fuel Database (SFD). These ratios, which provide an indication of the reliability of the estimate, remain near unity for fuels when not using the "worst case" template. This ratio exceeds 1 (often by large amounts) when the worst

case template is used, which is to be expected based on the very conservative construction of this template.

2. Reviewing results to ensure the calculated results correctly implemented the logic described in this report (by a designated technical reviewer who sampled several of the output sheets).

3. Independently checking implementation of the logic employed for the estimates (Figure 1) by comparing results obtained from a different programmer using a different program (Microsoft Access) to independently codify the same logic.[16]

Based on the considerations outlined above, the estimates presented here are considered to be adequate to support dose calculations for postclosure analyses as well as preclosure beyond design basis events analyses. If used for analyses that support conclusions beyond these purposes, responsibility for specifying applicable standards and for determining adequacy resides with the user. This report includes references and documentation intended to facilitate any such subsequent determinations of adequacy.

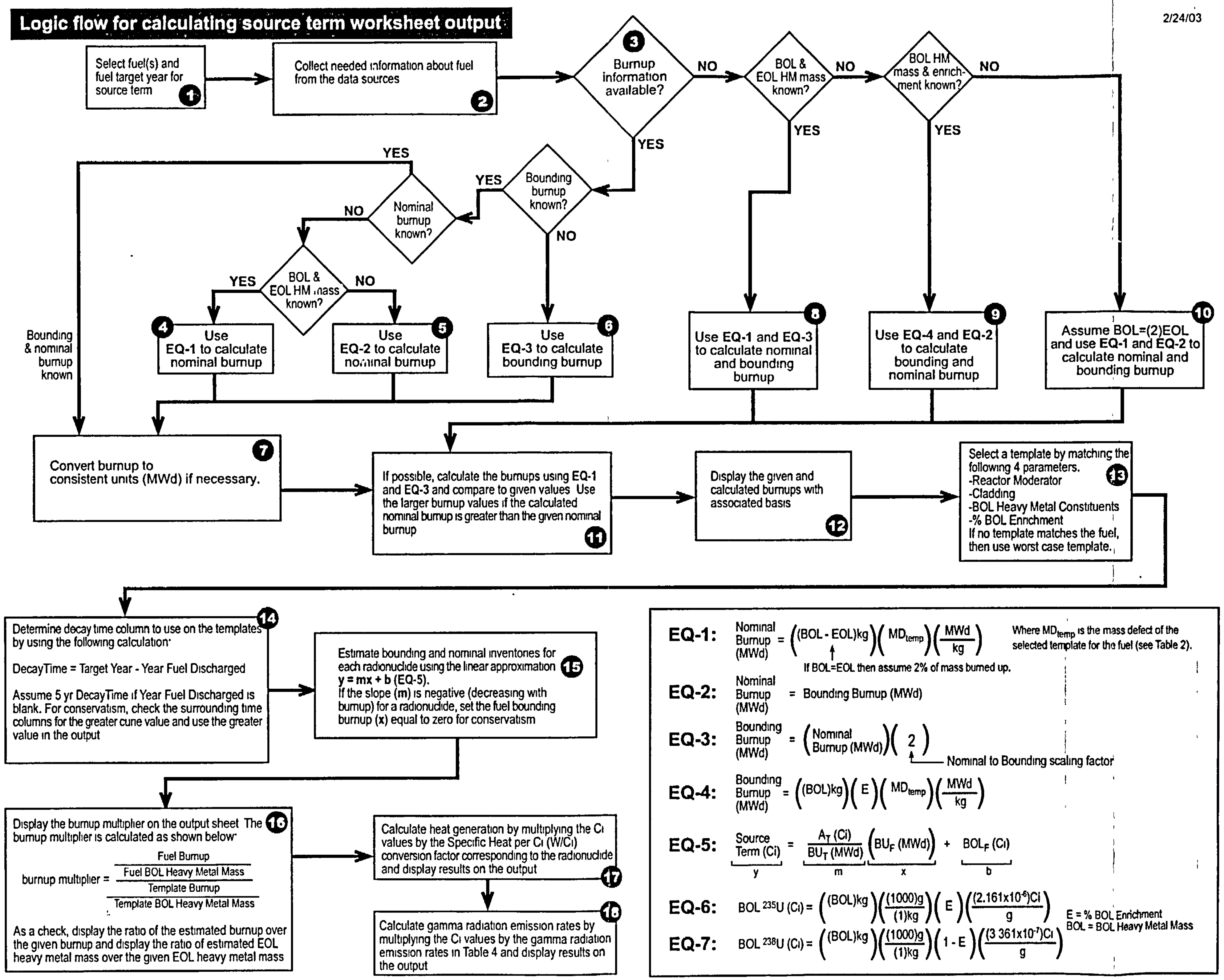

Figure 1. Logic flow for calculating source term worksheet output.

4. REQUIREMENTS AND CONSTRAINTS

In accordance with the applicable analysis plan, this report and the supporting analysis were performed in accordance with NSNFP procedures PSO 3.03, PSO 3.04, and PSO 19.01 (see References 12, 15, and 17).

5. INPUT

The input relied on to estimate the radionuclide inventories includes the precalculated templates and a Fuel Information Sheet (FIS). The FIS contains the information needed to identify an appropriate template, to select the applicable decay period, and to calculate the appropriate scaling. The templates and FISs are discussed in Sections 5.1 and 5.2 respectively.

5.1 Precalculated Templates

A template contains precalculated (i.e., ORIGEN output) radionuclide inventories at each of 10 specified decay periods, ranging from 5 to 100 years following irradiation. Templates include 145 radionuclides that typically account for over 99.9% of the total curie inventory.

The source term estimates rely on the availability and proper selection of a template that reasonably models the production and destruction of radionuclides (as a function of burnup) within the fuel being estimated. The source term is strongly dependent on the neutron energy spectrum and the fuel composition. The reactor moderator is a key factor in determining the neutron energy spectrum. Fuel composition can be reasonably well characterized by the fuel compound (i.e., uranium, uranium-thorium, uranium-plutonium), BOL enrichment, and cladding. These four parameters (i.e., reactor moderator and fuel compound, enrichment, and cladding) serve as the basis for identifying a template that reasonably models the fuel whose source term is to be estimated. Reference 7 suggests that most DOE SNFs can be accommodated by 28 templates, each representing potential combinations of these parameters. In order to help conservatively estimate source terms for fuels that do not fit well within one of the 28 suggested templates or when sufficient information is not available to determine the appropriate template, a bounding or "Worst Case" template is used.

A hypothetical template was developed with the intention of bounding the actual source term for virtually any conceivable SNF. It was produced by using ORIGEN to model a hypothetical fuel with properties (reactor and fuel parameters, and cross-section libraries) that maximize the production of actinides and activation products. To help ensure that this template would conservatively estimate source terms when linearly scaled to account for different burnups, the burnup on this template fuel was adjusted to try to maximize the curies per MWd (for key radionuclides). This template is included in Appendix A as the Hypothetical template. To further ensure conservatism, the resulting radionuclide inventories were then normalized to a per-MWd/kg basis and, for each radionuclide, were compared to the corresponding normalized value from each of the other templates, and the highest was retained. The net result (included in Appendix A as Worst Case [Template 29]), contains for each radionuclide a normalized curie content equal to the highest of all the templates including the Hypothetical template. The Hypothetical template was used in the analysis only as a step in deriving the Worst Case template.

The 1980 version of ORIGEN2 was used for the generation of all templates (see Reference 13). Newer versions of ORIGEN exist, but the 1980 version was used to be consistent with all the preceding work and validation studies. The powerful numerical solution methodology (matrix exponential method) used the ORIGEN code did not change from the 1980 to 1991 code versions. The differences in the versions lie in the updated data libraries. The differences in the libraries mainly pertain to updated half-life data for radionuclides. These differences were scrutinized, and it was determined that these differences did not have a significant effect on the result of this analysis.

Using the techniques outlined in Reference 2, 16 of the 29 templates proposed in Reference 7 have been developed and are used in this analysis. These 16 templates are sufficient to address 99.9% (by heavy metal mass) of the DOE spent fuels (95% of the SFD records). These 16 completed templates are included in Appendix A along with a crosswalk table that shows how the 29 proposed templates are represented by the 16 templates used. The Worst Case template was employed to conservatively estimate source terms for the remaining DOE SNFs. Documentation of template generation and review is contained in Reference 16.

5.2 Fuel Information Sheets

Fuel-specific information needed to select a template and to calculate the scaling factor was collected and recorded using a FIS. The reactor moderator and fuel cladding, fuel compound, and BOL enrichment are used to select an appropriate template. The fuel quantity and burnup are used to determine the proper scaling of the template results. If known, both nominal and bounding burnups are included directly in the FIS. If not known, the nominal and bounding burnups are conservatively estimated as described in Section 6. The fuel removal or reactor shutdown date is used to account for decay time. If not known, a date of fuel storage, shipment, or other date that confirms that the fuel is out of the core may be used.

An FIS was prepared for each fuel record in the NSNFP SFD.[a] The FISs were prepopulated with the available information from the SFD and provided to each of the three SNF custodial sites (Hanford, Savannah River, and the Idaho National Engineering and Environmental Laboratory) to review and make any necessary changes and to provide the basis (i.e., references or rationale) for the information included.[18] Sites were also asked to provide, when available, existing source term information for each fuel. The site input was provided with References 19, 20, and 21. The SFD was then updated to include this information, before being used to provide input for this analysis.

As noted previously, complete information is not available for many DOE SNFs. In the absence of information needed to select a template or to calculate scaling factors, assumptions that tend to err toward a more conservative result were used. These assumptions are intended to cause a conservative bias such that the resulting estimate will predict a source term with a higher dose. Table 1 suggests assumptions that are expected to provide conservative results when substituted for missing information. One or more of these parameters may also be used in lieu of known information when such a substitution allows selection of a template other than the Worst Case template. When matching a fuel record to a template, the order of importance of the four criteria is: 1-reactor moderator, 2-fuel type, 3-cladding, and 4-enrichment.

a. The Spent Fuel Database (SFD) is owned by the U.S. Department of Energy Office of Environmental Management (DOE-EM) and is maintained by the National Spent Nuclear Fuel Program (NSNFP). The SFD contains records for all DOE-owned and/or managed SNF including nuclear fuel at non-DOE-owned domestic research reactors and foreign research reactors. Information used to create records in the SFD was obtained from the sites where the SNF is in storage or use. Sources for this information came from the best available documentation and include fuel fabrication records, Appendix A data supplied by the irradiating reactor, and other technical documents. The sites where the SNF is in storage or use have reviewed the data in the SFD and have provided updated Fuel Information Sheets (FIS) where appropriate. These updated FISs have been used to update the data in the SFD. The data are checked regularly against Nuclear Materials Safeguards and Security records and/or Material Control and Accountability records. Because the SFD is used by several organizations, including the DOE sites, Headquarters, and Yucca Mountain Project personnel who need information about SNFs, the SFD records were chosen as the basis for performing the source term estimates.

Table 1. Conservative assumptions.

Unknown Parameter	Conservative Assumption	Basis
Cladding	If cladding is unknown, assume it is stainless steel.	Stainless steel is more conducive to the production of activation products than other typical cladding materials (e.g., aluminum, zirconium, graphite).
Fuel compound	If end-of-life (EOL) plutonium exceeds 1% by weight, assume a mixed oxide fuel. If thorium is present at EOL, assume a U-Th oxide fuel. Otherwise, assume a uranium fuel.	Because the majority of spent nuclear fuels (SNFs) are uranium fuels, this is assumed unless information provides evidence of other fuel compounds.
BOL enrichment	Assume the initial fissile mass equals the fissile mass depleted (i.e., 100% depletion).	Estimates the lowest possible enrichment (i.e., will underpredict the actual enrichment).
	If needed, the initial uranium inventory may be estimated as the EOL heavy metal mass plus the initial fissile mass.	These correlations assume uranium fuels. Uranium fuels comprise the majority of DOE SNFs. These correlations also provide reasonable approximations for other fuel types.
Moderator	Heavy water.	Heavy water moderation produces a soft neutron spectrum that is generally more conducive to transmutation of heavy metals.
Reactor shutdown or fuel removal date	Date for fuel shipping, storage, or any other activity that confirms the fuel was no longer in the reactor.	Use of a later date will produce a conservative result for all radionuclides of interest except Neptunium-237 and Americium-241 because, for a period, they may increase rather than decrease with decay time.

6. ANALYSIS

The analytical method employed is based on the template methodology described in Reference 7. An appropriate template is selected by matching the fuel compound, BOL enrichment, cladding material, and the reactor moderator to those of a precalculated template fuel with a specified mass and burnup. By matching these parameters, the template fuel provides a reasonable model for the generation of activation products, actinides, and fission products that can be scaled to account for burnup. The template provides radionuclide inventories for 145 radionuclides at 10 decay times ranging from 5 to 100 years.

After identifying an appropriate template, the SNF radionuclide inventory is estimated by scaling the template results to account for differences in burnup. The scaling factor accounts for the ratio of the absolute burnup (given in MWd) of the SNF to the absolute burnup of the template fuel. Absolute burnup differences result from differences in both the mass and the specific burnup (given in MWd/MTIHM) of the SNF relative to the template. It is, therefore, useful to consider the scaling factor as the product of a mass multiplier (M_M) and a burnup multiplier (M_{BU})

where

$$M_M = \frac{(\text{Mass of Fuel(kg)})}{(\text{Mass of Template(kg)})} = \text{mass multiplier}$$

$$M_{BU} = \frac{(\text{Burnup of Fuel(MWd)} / \text{Mass of Fuel(kg)})}{(\text{Burnup of Template(MWd)} / \text{Mass of Template(kg)})} = \text{burnup multiplier}$$

$$\text{scaling factor} = M_M * M_{BU} = \frac{(\text{Burnup of Fuel(MWd)})}{(\text{Burnup of Template(MWd)})}$$

Although these two component multipliers combine to produce a single scaling factor, each contributes differently to the uncertainty in the resulting estimate.

All radionuclide inventories scale linearly with the mass multiplier. Fission products also scale linearly with the burnup multiplier. However, because the buildup and depletion of actinides and activation products is not a linear function of burnup, these radionuclides are not true linear functions with respect to burnup. A small amount of error may be introduced when linearly scaling to account for differences in specific burnup. To aid in assessing the impacts of this uncertainty, the Fuel Radionuclide Inventory Worksheets (the output of this analysis) include information to show the contribution of the burnup multiplier to the overall scaling factor (in the "Checks" block at the bottom of the page under Burnup Multiplier).

Figure 1 shows the equations and associated logic used to prepare a source term estimate for each DOE SNF intended for repository disposal. The analytical approach uses available information and, in the absence of needed information, conservative assumptions in the estimate. The inputs are gathered as explained in Blocks 1 and 2. Blocks 3 through 12 show the logic for using the available information to obtain nominal and bounding burnups that will be used to scale the template results. Blocks 13 through 15 show how applicable template results are selected and scaled to obtain the source term estimate. Blocks 16 through 18 show how other output information is calculated.

The following provides more detailed information for each of the blocks shown in Figure 1. Excel 2000 was used with a number of imbedded software routines and macros in order to facilitate management of the input information, assumptions, and calculations.

Block 1: The fuels whose source term is to be estimated is selected, and the date for the desired source term estimate is a user input. The date is used in Block 14 to determine the elapsed decay time to the date of the source term prediction. For the analyses documented here, a source term estimate was performed for each DOE SNF record in the SFD (marked to go to a repository) for the years 2010 and 2030. These years correspond to the projected dates for beginning and completion of shipment of DOE SNF to the repository.

Block 2: For each SNF, available information is extracted from the SFD. This information includes the fuel name and SFD identification number (SNF ID#), reactor moderator, fuel cladding, BOL fuel enrichment, fuel compound, BOL heavy metal mass, burnup, and decay time as well as the number and type of canisters expected for this fuel. For the purposes of this document and to be consistent with the SFD, the heavy metal mass is defined as the sum of the masses of all plutonium, uranium, and thorium isotopes.

Block 3: The nominal and bounding burnup (MWd) of the SNF being estimated are used to determine the nominal and bounding burnup multipliers. If only one of the burnups (bounding or nominal) is known, it is used directly, and the other is estimated as shown in Blocks 4, 5, or 6. If neither the nominal nor the bounding burnup is available, they are estimated as shown in Blocks 8, 9, or 10.

Burnups Given in SFD—Blocks 4 Through 7

Block 4: If the bounding burnup is given but the nominal is not, and the change in heavy metal mass is known, the nominal burnup is calculated directly by assuming that the change in heavy metal mass resulted from fission using Equation 1 in Figure 1. Equation 1 multiplies the change in heavy metal mass by a mass defect factor (specific to the template that will be used for the fuel). The mass defect is defined as the template burnup divided by the change in heavy metal mass for the template and has units of MWd/kg. The values for MD_{temp} calculated from the templates and used in Equation 1 are shown below in Table 2.

Table 2. Mass defect values for each template.

Template	MD_{temp} (MWd/kg)	Template	MD_{temp} (MWd/kg)
3 (FFTF)	998.1412	12 (ATR)	947.0194
5 (FERMI)	881.8022	15 (Pathfinder)	944.6476
6 (FSV)	945.7257	21 (LWBR)	973.1629
7 (N-Reactor)	1054.9570	24 (PWR)	950.9527
8 (HFBR High E)	921.1030	26 (TRIGA Al)	954.5186
9 (HFBR Med E)	950.4648	27 (TRIGA FLIP)	950.4202
10 (HFBR Low E)	954.7123	28 (TRIGA SS)	954.6073
11 (HFBR Zr)	958.5533	29 (Worst Case)	950.3525[a]

a. A default value was used for template 29 (Worst Case) because it is a very conservative nonphysical fuel. The default value comes from the following formula: 950.3525 MWd/kg = $(1.854 \times 10^{-24}$MWd/Mev)(200Mev/atom)$(6.023 \times 10^{23}$atom/235g)(1000g/kg).

Block 5: If the bounding burnup is given but the nominal is not, and if the change in heavy metal is not known, the nominal burnup is conservatively estimated to be the same as the bounding burnup. This obtains the maximum attainable nominal burnup by presuming there was no power peaking (i.e., flat power distribution) within the reactor core. The conservatism of this assumption has a positive correlation with the actual peak to average power distribution within the reactor.

Block 6: If the nominal burnup is given but the bounding is not, the bounding burnup is conservatively assumed to be twice the nominal burnup because (1) radial power peaking factors in a typical nuclear reactor core rarely exceed a factor of two and (2) axial peaking is not a factor because the DOE SNF canister contains the full length of the fuel. The conservatism of this assumption has an inverse correlation to the peak to average power distribution within the reactor.

Block 7: The equations used in the estimate are based on absolute burnup using units of MWd. Consequently, if burnups are given per unit fuel (i.e., specific burnup), they are converted to absolute burnups by multiplying by the appropriate quantity of fuel. If specific burnups are given as MWd per MTIHM at BOL but BOL mass is not given, the BOL heavy metal mass is estimated using Equation 1 and the relationship: $\text{Burnup}_A(\text{MWd}) = \text{Burnup}_S(\text{MWd/MTIHM})(\text{BOL(kg)})(1\text{MT}/1000\text{kg})$. The two equations are solved for the two unknowns (BOL mass and Burnup_A).

No Burnups Given in SFD—Blocks 8 Through 10

Block 8: If the change in heavy metal mass is known, the nominal burnup is calculated directly by assuming that the change in heavy metal mass resulted from fission (see explanation in Block 4). The bounding burnup is then conservatively assumed to be twice the nominal burnup. The basis for this assumption is explained in Block 6. In some cases, the given end-of-life (EOL) and BOL heavy metal masses are equal, which indicates very little burnup. However, all fuels intended for repository disposal are conservatively assumed to have some burnup. Consequently, a burnup of 2% of the initial heavy metal mass is assumed in the event that the given BOL and EOL heavy metal masses are the same.

Block 9: If EOL heavy metal mass is not known but BOL heavy metal mass and enrichment are known, 100% burnup of all available fissile material is conservatively assumed. Available fissile material is estimated as the BOL heavy metal mass times the percent enrichment. The conservatism of this assumption is inversely correlated to the actual burnup of the fuel. For fertile fuels, nonconservatism could be introduced to the extent that fissile isotopes are produced during reactor operation.

Block 10: The minimum information needed to estimate burnup (using this methodology) is the EOL heavy metal mass, which is available for virtually all DOE SNFs. If burnup, loss of initial heavy metal mass, or initial fissile mass is unknown, the BOL heavy metal mass is assumed to be twice the EOL heavy metal mass. Having assumed the BOL heavy metal mass to be twice the EOL heavy metal mass, the nominal is calculated directly by assuming that the change in heavy metal mass resulted from fission (see explanation in Block 4), and the bounding burnup is assumed to be equal to the nominal burnup (i.e., using Equation 2 of Figure 1). As described below, this assumption is invoked for approximately 10% of the 565 SFD records.

Assuming BOL heavy metal mass to be twice the EOL heavy metal mass will result in overestimating burnup for all fuels where the material fissioned is less than one half of the original heavy metal mass. Consequently, this assumption could be nonconservative only for highly enriched and highly burned fuels. The percent of initial heavy metal mass depleted (fissioned) can be approximated as the product of the percent enrichment and the percent burnup. Because DOE SNFs are primarily the products of research, test, and materials production programs, the bulk of the DOE SNF is low burnup. Burnups approaching 50% are very uncommon. Fuels with greater than 50% of the initial heavy metal mass depleted are not expected.

The April 2002 NSNFP SFD data were reviewed to evaluate both the need for and the conservatism of assuming BOL heavy metal mass to be twice the EOL heavy metal mass. Of 610 records for repository-bound SNF, the BOL heavy metal mass is not known for 163 records. Of these 163 records, 56 have burnup information (i.e., nominal or bounding burnup can be obtained directly from the SFD), resulting in the need to estimate burnup for the other 107 SNF records. The assumption is made that BOL heavy metal mass is twice the EOL heavy metal mass for these 107 of 610 (17.5%) of the SNF records in the SFD.

The conservatism of this assumption was evaluated by reviewing the 447 SFD records with both BOL and EOL heavy metal mass known. Of these 447 records, estimating the BOL heavy metal mass to be twice the EOL heavy metal mass would have resulted in overestimating the heavy metal mass depleted (i.e., burnup) for 96.4% (431) of the 447 fuel records (see Figure 2).

The actual EOL heavy metal mass to BOL heavy metal mass was examined for these same 447 SNF records. As shown in Table 3, assuming this ratio to be 0.5 would overestimate the burnup by one to three orders of magnitude for over 96% (all but 16 of 447) of these SNFs. A review of the sixteen fuels, whose present data indicate that BOL heavy metal mass is more than twice the EOL heavy metal mass, reveals that these are primarily foreign research reactor fuels not yet returned to DOE custody. The burnup shown in the database for these fuels conservatively provides the maximum burnup that could be expected for these fuels. Historically, actual burnups, which are provided by the reactor sites when the fuel is returned, have been significantly less than the conservative estimates in the SFD projections.

By assuming that the burnup distribution of the fuels with unknown BOL heavy metal mass can be reasonably represented by the 447 fuel records evaluated above, one concludes that estimating BOL mass to be twice the EOL mass will produce a burnup estimate that is extremely conservative.

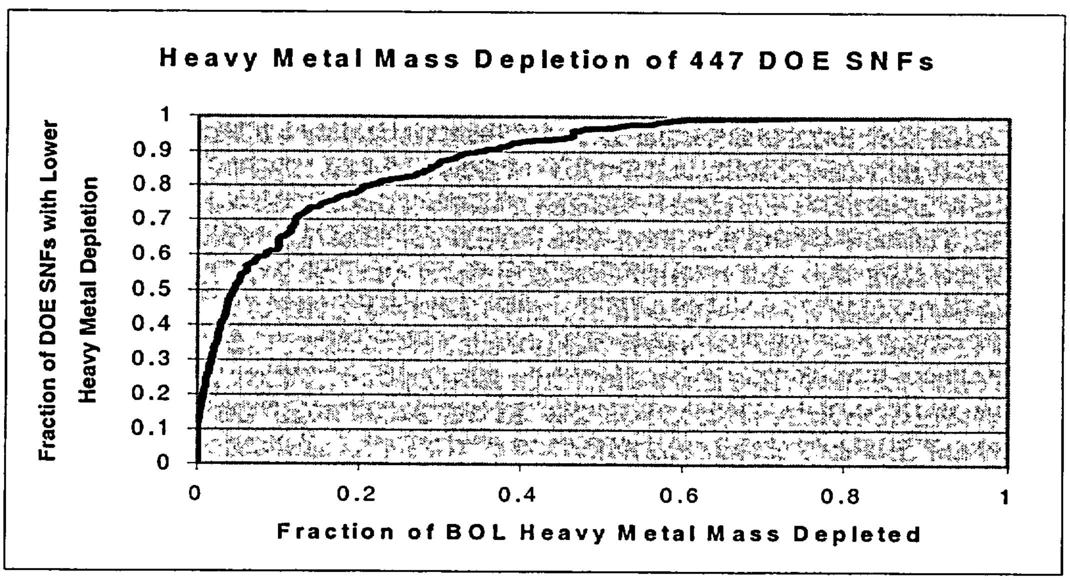

Figure 2. Heavy metal mass depletion of 447 DOE spent nuclear fuels.

Table 3. EOL over BOL heavy metal mass distribution of 447 DOE spent nuclear fuels (SNFs).

Heavy Metal Mass Ratio (EOL/BOL)	No. of SNFs in Range	Conservatism
.999 to 1.0	52	>1000
.99 to .999	48	>100
.9 to .99	173	>10
.83 to .9	65	>5
.667 to .83	62	>2
.5 to .667	31	>1
.135 to .5	16	>-6
Total SNF records evaluated	447	

Block 11: If burnup information and both BOL and EOL heavy metal mass are available directly, a consistency check is performed to ensure that the reported nominal burnup is conservative relative to that calculated based on the heavy metal depleted (i.e., Equation 1). If the calculated nominal burnup is greater than that provided, it is used in the estimate for conservatism. When this occurs, the bounding burnup estimate (i.e., twice the calculated nominal burnup) is also used if it is greater than the bounding burnup given.

Block 12: To facilitate evaluation of the estimate, the output sheet displays the bounding and the nominal burnups given along with the calculated and estimated values for the same. The bases of the calculated and estimated values are also displayed with the output.

Block 13: An appropriate template is selected based on four properties: the reactor moderator, the fuel type, BOL enrichment, and cladding. These four properties were selected because they are the primary factors for modeling production of activation products and actinides, and also because they are either available or can be conservatively estimated for DOE SNFs. When possible, a template is selected that matches all four of these parameters. If one or more of these parameters is not known, a conservative assumption (see Table 1) may be applied. If a template matching all four parameters is not available, an alternative template may be applied in accordance with the template selection guide (see Appendix A). The four template-selection parameters are displayed for both the fuel being estimated and the template fuel in Section III of the Fuel Radionuclide Inventory Worksheet. Lastly, an alternative template may be manually selected. When this occurs, the justification for the selection is recorded and displayed in the output (i.e., the Fuel Radionuclide Inventory Worksheet).

Block 14: The precalculated template results include inventories (curies) for 145 radionuclides at each of 10 decay intervals (5, 10, 15, 20, 25, 35, 50, 65, 80, and 100 years). The number of years between the desired date of the estimated source term and the date the SNF irradiation activities ended (i.e., reactor shutdown or fuel removal from the core) determines the decay time used in the estimate. The desired source term date is an input. For conservatism, the 5-year decay period is selected if no information is available to identify the fuel decay period.

When the desired decay time falls in the interval between two of the precalculated intervals, the higher of the two surrounding values is selected for each radionuclide. For example, if the desired decay period is 13 years, the inventory at both the 10 and 15-year decay periods is considered for each radionuclide, and the higher of the two inventories is selected. This provides conservatism even for radionuclides whose inventory may be building up rather than being depleted with time. The template radionuclide inventories at the selected decay time are displayed on the Fuel Radionuclide Inventory Worksheet.

Block 15: Most SNF radionuclide inventories can be estimated simply by scaling the precalculated template result by the ratio of the SNF burnup to the template fuel burnup. However, the calculations retain the general form of the linear correlation in order to properly account for radionuclides that have nonzero initial values ($b \neq 0$) and are depleted rather than produced by increasing burnup ($m<0$).

$Y_i = m_i x + b_i$ ...

where

Y_i = the estimated inventory (curies) for radionuclide$_i$

m_i = slope of the buildup ($\Delta Ci/\Delta MWd$) and is determined for each radionuclide from the precalculated template inventory

Note: When the BOL inventory is zero (i.e., $b_i = 0$), which is the case for most radionuclides of interest, the slope reduces to the precalculated template value at the desired decay period divided by the template burnup, $m_i = Ci_{i,t}/BU_t$.

x = burnup of the fuel being estimated

Note: Both a nominal and a bounding burnup are given in order to estimate nominal and bounding radionuclide inventories. For radionuclides whose inventory decreases with burnup (i.e., m is negative), the bounding burnup is set to zero.

b_i = initial inventory of radionuclide$_i$ for the fuel being estimated. If the initial inventory is not available for the fuel being estimated (or, for uranium fuels, cannot be calculated using Equations 6 and 7 of Figure 1), it is approximated by the initial inventory of the template fuel, after scaling it to account for any difference in mass.

Note: For radionuclides of interest other than Am-241, U-233, U-235, U-238, Th-232, Pu-238, Pu-239, Pu-240, Pu-241, and Pu-242; the BOL inventory, b, is set to zero. Consequently, the estimate reduces to $Y = mx$ where $m = Ci_t/BU_t$ and $x = BU_f$. This can be reformulated as $Ci_f = Ci_t*(BU_f/BU_t)$ where (BU_f/BU_t) is the factor used to scale the template radionuclide inventories to obtain an estimate of the fuel radionuclide inventories. Special consideration is taken for Am-243 in a spent fuel record, Americium Targets (SNF ID-776) at Hanford. A known initial value of 9.5712 Ci = (48 g)(0.1994 Ci/g) of Am-243 is entered on the output sheet to account for this special case.

The resulting estimate for each of the specified radionuclides is displayed on the Fuel Radionuclide Inventory Worksheet (see Section 7). The last row of the estimated radionuclide inventories includes the sum of the curies from the 104 radionuclides estimated but not individually displayed with the output. To facilitate checking the calculations, each of the above factors is displayed on the worksheet along with the basis for the burnup used and any identified issues or discrepancies.

Block 16: The absolute burnup of the fuel being estimated is the product of its specific burnup and its mass. Because the buildup and depletion of actinides and activation product is not a linear function of burnup, error is introduced when scaling to account for differences in specific burnup. Hence, to aid the analyst in assessing any resulting uncertainty in the estimate, the ratio of the specific burnup of the SNF being estimated to the template fuel (i.e., the burnup multiplier) is displayed with the output. To further aid the analyst in assessing uncertainty associated with the input data and template selection, the ratio of the estimated burnups (see explanation in Block 11) to the given burnups is displayed as is the estimated EOL heavy metal mass over the given heavy metal mass. The estimated EOL heavy metal mass is calculated by multiplying the curies of heavy metal (uranium, plutonium, and thorium) in the nominal estimate by the appropriate grams to curies conversion factors (see Table 4).

Block 17: Based on the estimated radionuclide inventories, the decay heat production is also calculated and displayed on the worksheet. The total decay heat produced is calculated by summing the decay heat from each of the 145 radionuclides. The decay heat from each of the radionuclides is calculated by multiplying the estimated curies of each radionuclide by its respective curies to watts conversion factor (see Table 5).

Block 18: Based on the estimated radionuclide inventories, the photon emission rates for each of 18 specified energy groups are also summed over each of the radionuclides (bounding inventories) and displayed on the worksheet. The conversion factors used for each of the 18 energy groups for each radionuclide are shown in Table 5. The data in Tables 4 and 5 are from the ORIGEN2 library files (see Reference 13). Because of space constraints, only the average values are shown on the output sheets, but the values shown correspond to a range (as shown in Table 5). Each bounding Ci value for each isotope is multiplied by the photon/sec/Ci value in Table 5 to get a photon/sec value. These values are then summed across all isotopes for each energy group and displayed on the output sheet.

Table 4. Specific activity of heavy metals.

	Half-Life[a] (Years)	Atomic Weight[b]	Specific Activity[c] Ci/g	Specific Activity[c] g/Ci
Am241	4.322E+02	241.0568229	3.431E+00	2.914E-01
Am243	7.380E+03	243.0613727	1.993E-01	5.018E+00
Pu236	2.851E+00	236.0460481	5.312E+02	1.882E-03
Pu237	1.248E-01	237.0484038	1.208E+04	8.278E-05
Pu238	8.774E+01	238.0495534	1.712E+01	5.843E-02
Pu239	2.406E+04	239.0521565	6.215E-02	1.609E+01
Pu240	6.537E+03	240.0538075	2.278E-01	4.390E+00
Pu241	1.440E+01	241.0568453	1.030E+02	9.709E-03
Pu242	3.869E+05	242.0587368	3.817E-03	2.620E+02
Pu244	8.261E+07	244.064198	1.773E-05	5.640E+04
Th227	5.124E-02	227.027699	3.073E+04	3.254E-05
Th228	1.913E+00	228.0287313	8.195E+02	1.220E-03
Th229	7.339E+03	229.031755	2.127E-01	4.702E+00
Th230	7.700E+04	230.0331266	2.018E-02	4.955E+01
Th231	2.911E-03	231.0362971	5.315E+05	1.881E-06

Table 4. (continued).

	Half-Life[a] (Years)	Atomic Weight[b]	Specific Activity[c]	
			Ci/g	g/Ci
Th232	1.405E+10	232.0380504	1.097E-07	9.120E+06
Th234	6.597E-02	234.043595	2.315E+04	4.319E-05
U232	7.200E+01	232.0371463	2.140E+01	4.673E-02
U233	1.585E+05	233.039628	9.678E-03	1.033E+02
U234	2.445E+05	234.0409456	6.247E-03	1.601E+02
U235	7.038E+08	235.0439231	2.161E-06	4.627E+05
U236	2.341E+07	236.0455619	6.468E-05	1.546E+04
U237	1.848E-02	237.048724	8.161E+04	1.225E-05
U238	4.468E+09	238.0507826	3.361E-07	2.975E+06

a. A. G. Croff, *ORIGEN2—A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, July 1980.

b. J. S. Coursey, D. J. Schwab, and R. A. Dragoset, Atomic Weights and Isotopic Compositions (version 2.3.1), [Online, 2001], Available: http://physics.nist.gov/Comp (January 17, 2003), National Institute of Standards and Technology, Gaithersburg, Maryland.

c. Specific Activity = Ci/g = $(3.575 \times 10^5)/[(A)(T)]$

where A = Atomic Weight, T = Half-Life in years.

Table 5. Conversion factors for curies to watts and photon emission rates.

Photon Energy Group No.			1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
Lower Photon Energy Group Limit (MeV)			0.0100	0.0200	0.0300	0.0450	0.0700	0.1000	0.1500	0.3000	0.4500	0.7000	1.0000	1.5000	2.0000	2.5000	3.0000	4.0000	6.0000	8.0000
Upper Photon Energy Group Limit (MeV)			0.0200	0.0300	0.0450	0.0700	0.1000	0.1500	0.3000	0.4500	0.7000	1.0000	1.5000	2.0000	2.5000	3.0000	4.0000	6.0000	8.0000	14.0000
Mean Photon Energy (MeV)			0.0150	0.0250	0.0375	0.0575	0.0850	0.1250	0.2250	0.3750	0.5750	0.8500	1.2500	1.7500	2.2500	2.7500	3.5000	5.0000	7.0000	11.0000
	Half-Life (years)	Heat Gen. (W/Ci)	Photons/sec/Ci																	
AC227	2.177E+01	4.842E-04	2.942E+08	2.405E+06	3.811E+04	2.882E+06	1.913E+07	7.252E+06	6.364E+06	2.738E+05	0.000E+00	1.706E+00	9.694E-01	4.847E-01	2.438E-01	1.221E-01	8.991E-02	2.679E-02	1.735E-03	1.095E-04
AG110	7.795E-07	7.183E-03	2.590E+10	5.698E+09	3.774E+09	5.587E+09	3.530E+09	2.405E+09	3.504E+09	1.713E+09	2.905E+09	3.774E+08	1.658E+08	3.415E+07	3.363E+06	4.773E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG110M	6.841E-01	1.670E-02	1.284E+09	4.477E+08	1.273E+08	1.558E+08	7.548E+07	7.659E+07	1.010E+08	1.695E+09	4.884E+10	5.883E+10	1.188E+10	4.588E+09	3.437E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AG111	2.040E-02	2.240E-03	7.733E+09	1.621E+09	9.953E+08	1.373E+09	8.214E+08	4.699E+08	9.657E+08	2.165E+09	3.996E+07	4.662E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AM241	4.322E+02	3.322E-02	8.177E+09	9.361E+08	7.659E+07	1.376E+10	9.028E+06	8.621E+06	4.810E+05	4.662E+05	2.557E+05	7.511E+04	7.067E+01	3.537E+01	1.776E+01	8.917E+00	6.549E+00	1.957E+00	1.280E-01	8.177E-03
AM242	1.827E-03	1.162E-03	2.390E+10	5.402E+08	3.589E+08	4.292E+08	1.806E+09	2.886E+09	9.694E+07	1.228E+07	3.319E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AM242M	1.520E+02	3.951E-04	1.635E+10	2.416E+03	0.000E+00	7.474E+07	1.806E+07	1.976E+07	8.214E+06	0.000E+00	0.000E+00	2.490E+01	1.166E+01	5.698E+00	3.293E+00	1.906E+00	1.706E+00	7.289E-01	8.325E-02	9.546E-03
AM243	7.380E+03	3.214E-02	8.621E+09	0.000E+00	2.405E+09	4.551E+06	2.157E+10	2.457E+08	6.623E+05	0.000E+00	5.402E+05	1.510E+02	8.288E+01	4.144E+01	2.128E+01	1.095E+01	8.399E+00	2.749E+00	2.187E-01	1.865E-02
BA136M	9.760E-09	1.210E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA137M	4.851E-06	3.926E-03	1.754E+08	0.000E+00	2.409E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.811E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA140	3.502E-02	2.790E-03	9.509E+09	7.585E+09	1.314E+09	1.084E+09	5.994E+08	4.329E+08	1.883E+09	2.856E+09	6.919E+09	4.699E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BE10	1.600E+06	1.201E-03	4.181E+09	7.955E+08	4.884E+08	6.327E+08	3.189E+08	1.684E+08	1.295E+08	1.214E+07	7.252E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BI211	4.050E-06	3.989E-02	3.774E+08	5.587E+03	3.400E+03	4.329E+03	7.881E+08	1.125E+03	1.169E+03	4.218E+09	8.325E-01	1.232E+02	6.993E+01	3.511E+01	1.761E+01	8.843E+00	6.512E+00	1.935E+00	1.254E-01	7.918E-03
BI212	1.151E-04	1.700E-02	9.620E+09	1.380E+09	1.336E+09	1.317E+09	8.880E+08	5.365E+08	8.991E+08	3.674E+08	2.745E+08	4.847E+09	3.297E+08	1.251E+09	1.077E+01	3.182E+00	2.342E+00	6.956E-01	4.514E-02	2.853E-03
C14	5.729E+03	2.933E-04	8.658E+08	1.225E+08	5.883E+07	4.625E+07	7.585E+06	5.550E+05	3.297E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113	0.000E+00	0.000E+00	1.909E+09	3.282E+08	1.865E+08	2.094E+08	8.177E+07	3.101E+07	9.546E+06	3.774E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113M	1.459E+01	1.683E-03	3.996E+09	7.585E+08	4.662E+08	5.957E+08	3.001E+08	1.587E+08	1.247E+08	1.310E+07	1.528E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD115M	1.221E-01	3.728E-03	1.336E+10	2.797E+09	1.828E+09	2.616E+09	1.573E+09	1.014E+09	1.317E+09	5.143E+08	2.708E+08	5.772E+08	2.568E+08	4.292E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	8.901E-02	1.464E-03	3.574E+09	5.698E+08	6.549E+09	4.255E+08	2.005E+08	2.076E+10	6.475E+07	4.514E+06	3.774E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	1.049E+11	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.101E+02	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
CE144	7.783E-01	6.632E-04	1.980E+09	2.738E+08	3.996E+09	2.301E+08	9.176E+08	4.292E+09	6.253E+06	9.361E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CL36	3.010E+05	1.462E-03	5.143E+09	1.010E+09	6.290E+08	8.399E+08	4.477E+08	2.520E+08	2.338E+08	3.959E+07	1.203E+07	1.184E+01	2.054E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CM242	4.468E-01	1.162E-03	6.031E+09	0.000E+00	1.413E+07	0.000E+00	0.000E+00	7.548E+05	3.848E+05	0.000E+00	9.694E+04	2.061E+04	6.068E+03	2.387E+03	1.384E+03	7.992E+02	7.178E+02	3.071E+02	3.526E+01	4.070E+00
CM243	2.850E+01	3.670E-02	3.238E+10	0.000E+00	5.291E+07	1.672E+08	6.216E+09	1.092E+10	1.191E+10	2.094E+07	0.000E+00	1.232E+02	6.993E+01	3.511E+01	1.761E+01	8.843E+00	6.512E+00	1.935E+00	1.254E-01	7.918E-03
CM244	1.811E+01	3.498E-02	5.550E+09	0.000E+00	1.099E+07	0.000E+00	0.000E+00	0.000E+00	3.186E+05	0.000E+00	0.000E+00	2.050E+05	9.546E+04	4.662E+04	2.701E+04	1.565E+04	1.410E+04	6.031E+03	6.956E+02	7.992E+01
CM245	8.499E+03	3.317E-02	3.219E+10	0.000E+00	4.292E+07	0.000E+00	6.179E+09	1.328E+10	1.887E+09	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
CM246	4.731E+03	3.273E-02	4.921E+09	0.000E+00	1.232E+07	0.000E+00	0.000E+00	0.000E+00	6.179E+07	0.000E+00	0.000E+00	3.959E+07	1.846E+07	9.028E+06	5.254E+06	3.038E+06	2.731E+06	1.169E+06	1.351E+05	1.554E+04
CM247	1.560E+07	3.196E-02	5.661E+09	0.000E+00	0.000E+00	3.411E+07	5.402E+08	1.306E+09	2.512E+09	3.115E+10	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
CO60	5.270E+00	1.542E-02	1.961E+09	3.389E+08	1.935E+08	2.183E+08	8.584E+07	3.297E+07	1.084E+07	3.041E+06	1.746E+05	2.764E+06	7.400E+10	0.000E+00	3.922E+05	1.214E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CR51	7.586E-02	2.140E-04	4.144E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.093E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS134	2.062E+00	1.018E-02	3.360E+09	6.364E+08	6.327E+08	5.069E+08	2.612E+08	1.425E+08	1.306E+08	2.176E+07	4.699E+10	3.263E+10	2.157E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS135	2.300E+06	3.337E-04	1.069E+09	1.617E+08	8.325E+07	7.659E+07	1.950E+07	3.774E+06	1.743E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS136	3.587E-02	1.364E-02	2.583E+09	3.700E+08	5.772E+09	5.624E+09	2.475E+09	1.754E+08	1.317E+10	1.591E+10	3.448E+08	3.552E+10	3.197E+10	3.737E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS137	3.000E+01	1.105E-03	3.659E+09	6.919E+08	4.218E+08	5.365E+08	2.675E+08	1.402E+08	1.114E+08	1.669E+07	3.134E+06	2.068E+05	7.881E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU152	1.360E+01	7.582E-03	3.700E+09	1.043E+08	2.346E+10	4.440E+09	5.143E+07	1.029E+10	3.363E+09	1.173E+10	1.410E+09	1.265E+10	1.846E+10	9.324E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU154	8.600E+00	8.946E-03	6.771E+09	9.990E+08	9.213E+09	2.472E+09	4.736E+08	1.506E+10	3.112E+09	4.958E+08	3.504E+09	1.761E+10	1.950E+10	5.846E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU155	4.959E+00	7.273E-04	2.671E+09	2.427E+08	7.770E+09	2.342E+09	1.169E+10	6.438E+09	2.372E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EU156	4.157E-02	1.032E-02	9.657E+09	1.758E+09	5.587E+09	2.468E+09	4.403E+09	6.327E+08	1.117E+09	4.921E+08	4.181E+09	8.510E+09	1.580E+10	3.541E+09	5.846E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Table 5. (continued).

Photon Energy Group No.			1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
Lower Photon Energy Group Limit (MeV)			0.0100	0.0200	0.0300	0.0450	0.0700	0.1000	0.1500	0.3000	0.4500	0.7000	1.0000	1.5000	2.0000	2.5000	3.0000	4.0000	6.0000	8.0000
Upper Photon Energy Group Limit (MeV)			0.0200	0.0300	0.0450	0.0700	0.1000	0.1500	0.3000	0.4500	0.7000	1.0000	1.5000	2.0000	2.5000	3.0000	4.0000	6.0000	8.0000	14.0000
Mean Photon Energy (MeV)			0.0150	0.0250	0.0375	0.0575	0.0850	0.1250	0.2250	0.3750	0.5750	0.8500	1.2500	1.7500	2.2500	2.7500	3.5000	5.0000	7.0000	11.0000
	Half-Life (years)	Heat Gen. (W/Ci)	Photons/sec/Ci																	
FE55	2.600E+00	1.375E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FE59	1.232E-01	9.324E-03	2.427E+09	4.366E+08	2.564E+08	3.067E+08	1.362E+08	6.253E+07	3.648E+07	2.394E+06	3.589E+05	6.216E+04	5.550E+03	5.994E-04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FR223	4.145E-05	2.597E-03	2.387E+10	1.728E+09	9.398E+08	1.265E+10	6.142E+09	9.583E+08	2.468E+09	4.292E+08	2.986E+07	3.078E+08	1.069E+02	2.113E-03	1.058E-03	5.328E-04	3.922E-04	1.166E-04	7.548E-06	4.773E-07
GD153	6.626E-01	8.619E-04	5.217E+09	0.000E+00	3.959E+10	8.473E+09	1.265E+10	6.438E+09	1.273E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
H3	1.235E+01	3.366E-05	1.362E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	1.570E+07	4.625E-04	1.954E+09	2.516E+10	7.363E+09	3.164E+07	4.403E+06	2.398E+05	2.683E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I131	2.201E-02	3.396E-03	3.996E+09	2.461E+09	7.548E+08	5.846E+08	1.210E+09	1.558E+08	3.060E+09	2.938E+10	3.186E+09	5.661E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	2.278E-06	4.591E-03	1.691E+10	3.848E+09	2.364E+09	3.434E+09	2.105E+09	1.391E+09	1.887E+09	8.103E+08	3.922E+08	9.842E+07	9.916E+07	4.699E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	1.356E-01	1.410E-03	6.438E+08	1.339E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	4.810E+09	0.000E+00	1.580E+09	1.365E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	4.905E-04	1.994E-03	4.144E+08	1.240E+10	1.092E+06	1.476E+06	8.103E+05	4.736E+05	4.847E+05	1.550E+10	5.106E+05	4.551E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR85	1.072E+01	1.497E-03	5.476E+09	1.073E+09	6.697E+08	8.954E+08	4.773E+08	2.708E+08	2.549E+08	4.440E+07	1.447E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA140	4.588E-03	1.676E-02	1.206E+10	2.501E+09	2.172E+09	2.346E+09	1.380E+09	1.132E+09	1.513E+09	7.659E+09	1.365E+10	1.539E+10	5.032E+06	3.226E+10	3.389E+08	1.228E+09	1.010E+07	0.000E+00	0.000E+00	0.000E+00
MN54	8.556E-01	4.979E-03	5.106E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.633E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO93	3.498E+03	9.344E-05	3.959E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB93M	1.360E+01	1.771E-04	6.993E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB94	2.030E+04	1.019E-02	3.086E+09	5.698E+08	3.419E+08	4.218E+08	1.987E+08	9.657E+07	5.994E+07	2.845E+06	1.676E-05	6.845E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB95	9.624E-02	4.795E-03	7.770E+08	1.062E+08	5.032E+07	3.885E+07	6.364E+06	4.995E+05	1.565E+02	0.000E+00	0.000E+00	3.334E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB95M	9.880E-03	1.389E-03	2.749E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.006E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND144	2.099E+15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.101E+02	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
ND147	3.028E-02	2.413E-03	6.623E+09	9.879E+08	1.817E+10	8.214E+08	1.151E+10	3.922E+08	6.697E+08	1.584E+09	5.032E+09	1.332E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NI59	8.000E+04	6.366E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NI63	9.200E+01	3.971E-04	2.046E+08	1.247E+07	2.527E+06	2.557E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NP237	2.140E+06	3.056E-02	2.905E+10	4.255E+09	0.000E+00	2.771E+08	7.474E+09	8.473E+08	4.144E+08	1.043E+06	8.880E+05	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
PA231	3.277E+04	3.013E-02	2.139E+10	3.848E+09	1.036E+08	1.595E+08	7.215E+07	2.646E+07	9.435E+08	2.720E+09	1.254E+06	1.240E+02	7.030E+01	3.534E+01	1.772E+01	8.880E+00	6.549E+00	1.954E+00	1.273E-01	8.103E-03
PA233	7.393E-02	2.270E-03	2.638E+10	2.609E+08	1.362E+08	1.280E+08	1.169E+10	2.572E+09	4.070E+08	1.476E+10	7.733E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PA234	7.643E-04	1.436E-02	6.845E+10	2.120E+09	1.369E+09	3.141E+09	2.028E+10	1.310E+10	1.243E+10	2.997E+09	1.728E+10	5.254E+10	5.698E+09	2.494E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PA234M	2.225E-06	4.943E-03	3.641E+10	7.733E+09	5.106E+09	7.437E+09	4.699E+09	3.075E+09	4.218E+09	1.869E+09	9.546E+08	3.996E+08	2.664E+08	3.534E+07	3.167E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PB210	2.230E+01	2.317E-04	9.768E+09	2.065E+06	3.737E+05	1.214E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PB211	6.864E-05	2.996E-03	1.010E+10	2.057E+09	1.328E+09	1.883E+09	1.277E+09	6.845E+08	8.658E+08	2.094E+09	1.088E+08	1.306E+09	4.218E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PB212	1.214E-03	1.904E-03	8.251E+09	3.596E+08	2.072E+08	2.394E+08	1.269E+10	2.475E+08	1.754E+10	1.018E+09	8.695E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PD107	6.496E+06	5.929E-05	6.031E+07	3.622E+05	5.217E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM145	1.770E+01	2.550E-04	2.779E+09	0.000E+00	2.801E+10	2.812E+08	6.882E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM147	2.623E+00	3.588E-04	1.199E+09	1.869E+08	9.842E+07	9.509E+07	2.712E+07	7.437E+06	5.180E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	1.470E-02	7.698E-03	3.186E+10	6.771E+09	4.514E+09	6.475E+09	3.996E+09	2.631E+09	3.600E+09	1.587E+09	9.509E+09	5.217E+09	9.694E+09	8.177E+06	5.846E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148M	1.131E-01	1.268E-02	6.401E+09	1.095E+09	4.107E+09	1.254E+09	2.372E+09	2.031E+08	6.290E+09	1.106E+10	7.807E+10	1.765E+10	6.031E+09	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PO212	9.506E-15	5.300E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.101E+02	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
PO215	5.640E-11	4.465E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.101E+02	1.732E+07	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
PO216	4.753E-09	4.093E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.101E+02	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03

Table 5. (continued).

	Half-Life (years)	Heat Gen. (W/Ci)	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
Photon Energy Group No.			1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
Lower Photon Energy Group Limit (MeV)			0.0100	0.0200	0.0300	0.0450	0.0700	0.1000	0.1500	0.3000	0.4500	0.7000	1.0000	1.5000	2.0000	2.5000	3.0000	4.0000	6.0000	8.0000
Upper Photon Energy Group Limit (MeV)			0.0200	0.0300	0.0450	0.0700	0.1000	0.1500	0.3000	0.4500	0.7000	1.0000	1.5000	2.0000	2.5000	3.0000	4.0000	6.0000	8.0000	14.0000
Mean Photon Energy (MeV)			0.0150	0.0250	0.0375	0.0575	0.0850	0.1250	0.2250	0.3750	0.5750	0.8500	1.2500	1.7500	2.2500	2.7500	3.5000	5.0000	7.0000	11.0000
			Photons/sec/Ci																	
PR143	3.714E-02	1.863E-03	6.882E+09	1.376E+09	8.732E+08	1.191E+09	6.623E+08	3.922E+08	4.181E+08	1.032E+08	1.661E+07	1.906E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144	3.286E-05	7.350E-03	2.657E+10	5.772E+09	3.848E+09	5.735E+09	3.637E+09	2.483E+09	3.626E+09	1.791E+09	1.724E+09	3.922E+08	3.115E+08	3.615E+07	2.816E+08	2.523E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144M	1.369E-05	3.421E-04	2.054E+09	9.435E+02	1.106E+10	3.001E+07	5.217E+02	3.356E+02	4.292E+02	1.617E+02	2.690E+07	2.128E+07	8.325E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU236	2.851E+00	3.481E-02	6.660E+09	0.000E+00	0.000E+00	2.113E+07	0.000E+00	3.885E+06	1.795E+05	0.000E+00	1.924E+05	2.453E+02	1.269E+02	6.290E+01	3.371E+01	1.817E+01	1.487E+01	5.513E+00	5.402E-01	5.550E-02
PU237	1.248E-01	9.606E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.025E-02	0.000E+00	0.000E+00	4.070E-03	2.316E-03	1.162E-03	5.846E-04	2.923E-04	2.150E-04	6.401E-05	4.144E-06	2.620E-07
PU238	8.774E+01	3.315E-02	5.809E+09	0.000E+00	1.672E+07	0.000E+00	3.215E+06	0.000E+00	2.560E+05	0.000E+00	0.000E+00	1.806E+04	7.363E+02	9.879E+01	5.476E+01	3.027E+01	2.575E+01	1.021E+01	1.077E+00	1.173E-01
PU239	2.406E+04	3.082E-02	2.113E+09	0.000E+00	2.309E+06	7.807E+06	7.844E+05	3.123E+06	4.440E+05	2.361E+06	8.695E+04	7.918E+03	8.806E+01	3.537E+01	1.776E+01	8.917E+00	6.549E+00	1.961E+00	1.280E-01	8.214E-03
PU240	6.537E+03	3.113E-02	5.550E+09	0.000E+00	0.000E+00	1.310E+07	0.000E+00	2.161E+06	1.328E+05	0.000E+00	6.956E+03	7.955E+03	3.608E+03	1.765E+03	1.021E+03	5.920E+02	5.291E+02	2.264E+02	2.597E+01	2.982E+00
PU241	1.440E+01	3.101E-05	2.812E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	7.585E-05	0.000E+00	0.000E+00	3.027E-05	1.721E-05	8.621E-06	4.329E-06	2.172E-06	1.595E-06	4.736E-07	3.078E-08	1.946E-09
PU242	3.869E+05	2.954E-02	4.366E+09	0.000E+00	1.517E+07	0.000E+00	0.000E+00	2.272E+06	1.425E+06	0.000E+00	0.000E+00	8.436E+05	3.922E+05	1.920E+05	1.114E+05	6.475E+04	5.809E+04	2.490E+04	2.868E+03	3.300E+02
PU244	8.261E+07	2.900E-02	3.922E+09	0.000E+00	1.058E+07	0.000E+00	0.000E+00	0.000E+00	2.838E+08	0.000E+00	0.000E+00	1.824E+08	8.510E+07	4.144E+07	2.409E+07	1.395E+07	1.254E+07	5.402E+06	6.216E+05	7.141E+04
RA223	3.130E-02	3.561E-02	1.058E+10	0.000E+00	3.119E+04	0.000E+00	1.898E+10	1.854E+09	7.844E+09	3.138E+09	9.324E+07	1.084E+06	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
RA224	1.002E-02	3.431E-02	1.695E+08	0.000E+00	0.000E+00	0.000E+00	1.580E+08	0.000E+00	1.550E+09	1.617E+06	2.930E+06	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
RA226	1.600E+03	2.887E-02	3.482E+08	0.000E+00	0.000E+00	0.000E+00	2.287E+08	0.000E+00	1.006E+09	2.794E+05	2.357E+05	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
RA228	6.700E+00	7.706E-05	4.292E+07	5.513E+05	6.512E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB87	4.696E+10	8.358E-04	1.580E+09	2.616E+08	1.447E+08	1.539E+08	5.365E+07	1.728E+07	3.271E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH103M	1.067E-04	2.302E-04	4.033E+08	2.338E+09	2.745E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	9.475E-07	9.592E-03	3.097E+10	6.808E+09	4.551E+09	6.808E+09	4.366E+09	3.012E+09	4.477E+09	2.383E+09	1.243E+10	7.437E+08	9.472E+08	1.765E+08	5.439E+07	7.955E+06	1.040E+06	0.000E+00	0.000E+00	0.000E+00
RN219	1.255E-07	4.149E-02	4.144E+08	0.000E+00	0.000E+00	0.000E+00	6.179E+08	4.995E+07	4.440E+09	2.657E+09	2.409E+07	9.435E+05	1.876E+05	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
RN220	1.762E-06	3.797E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.101E+02	0.000E+00	2.475E+07	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
RU103	1.075E-01	3.345E-03	1.439E+09	5.069E+08	1.310E+08	2.708E+08	5.365E+07	2.616E+07	1.450E+08	1.476E+08	3.134E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	1.008E+00	5.947E-05	1.476E+08	2.035E+06	2.819E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB124	1.648E-01	1.328E-02	8.362E+09	1.843E+09	1.136E+09	1.573E+09	9.287E+08	5.920E+08	7.696E+08	7.030E+08	4.144E+10	5.957E+09	3.452E+09	1.769E+10	2.035E+09	1.487E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	2.770E+00	3.127E-03	2.453E+09	1.565E+10	4.292E+09	2.083E+08	9.324E+07	1.621E+08	2.250E+09	1.336E+10	1.717E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB126	3.394E-02	1.848E-02	5.957E+09	1.665E+09	8.362E+08	1.036E+09	5.846E+08	4.847E+08	4.403E+09	3.604E+10	1.121E+11	2.816E+10	1.606E+09	8.177E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB126M	3.612E-05	1.274E-02	1.132E+10	2.801E+09	1.621E+09	2.227E+09	1.351E+09	8.806E+08	1.166E+09	3.552E+10	7.437E+10	5.661E+08	8.695E+08	4.551E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE79	6.496E+04	2.489E-04	1.014E+09	1.499E+08	7.474E+07	6.290E+07	1.180E+07	1.040E+06	2.831E+02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	9.310E-01	5.527E-04	4.995E+09	0.000E+00	5.550E+10	5.032E+09	0.000E+00	5.365E+05	0.000E+00	0.000E+00	8.880E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM147	1.070E+11	1.369E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.101E+02	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
SM151	8.999E+01	1.172E-04	2.812E+08	2.875E+07	4.958E+06	9.583E+05	8.621E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	6.708E-01	5.168E-04	1.521E+09	1.621E+10	0.000E+00	2.535E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN121M	4.997E+01	2.004E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN123	3.536E-01	3.124E-03	1.166E+10	2.416E+09	1.569E+09	2.227E+09	1.317E+09	8.399E+08	1.051E+09	3.848E+08	1.391E+08	1.976E+07	2.050E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN125	2.639E-02	6.627E-03	1.795E+10	3.848E+09	2.546E+09	3.737E+09	2.313E+09	1.547E+09	2.220E+09	1.524E+09	9.620E+08	3.630E+09	5.032E+09	9.620E+07	7.881E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN126	1.000E+05	1.247E-03	3.600E+09	1.465E+10	4.218E+08	4.218E+09	1.758E+10	4.070E+07	1.528E+07	6.660E+04	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR89	1.383E-01	3.457E-03	1.302E+10	2.720E+09	1.772E+09	2.535E+09	1.517E+09	9.731E+08	1.251E+09	4.773E+08	1.869E+08	3.774E+07	2.427E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR90	2.912E+01	1.161E-03	8.473E+09	1.617E+09	9.916E+08	1.284E+09	6.512E+08	3.463E+08	2.738E+08	2.701E+07	1.369E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TB160	1.980E-01	8.145E-03	6.993E+09	8.547E+08	5.291E+08	7.030E+09	5.365E+09	1.991E+08	1.650E+10	9.361E+08	2.805E+08	2.668E+10	1.117E+10	7.326E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Table 5. (continued).

		Photon Energy Group No.	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
		Lower Photon Energy Group Limit (MeV)	0.0100	0.0200	0.0300	0.0450	0.0700	0.1000	0.1500	0.3000	0.4500	0.7000	1.0000	1.5000	2.0000	2.5000	3.0000	4.0000	6.0000	8.0000
		Upper Photon Energy Group Limit (MeV)	0.0200	0.0300	0.0450	0.0700	0.1000	0.1500	0.3000	0.4500	0.7000	1.0000	1.5000	2.0000	2.5000	3.0000	4.0000	6.0000	8.0000	14.0000
		Mean Photon Energy (MeV)	0.0150	0.0250	0.0375	0.0575	0.0850	0.1250	0.2250	0.3750	0.5750	0.8500	1.2500	1.7500	2.2500	2.7500	3.5000	5.0000	7.0000	11.0000
	Half-Life (years)	Heat Gen. (W/Ci)	Photons/sec/Ci																	
TC99	2.130E+05	5.014E-04	3.423E+09	5.772E+08	3.226E+08	3.515E+08	1.288E+08	4.477E+07	1.073E+07	1.943E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE123M	3.277E-01	1.457E-03	1.221E+09	1.632E+10	2.782E+09	0.000E+00	3.341E+07	0.000E+00	2.198E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE125M	1.588E-01	8.405E-04	2.105E+09	3.811E+10	8.843E+09	0.000E+00	0.000E+00	9.139E+07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE127	1.067E-03	1.350E-03	4.847E+09	9.361E+08	5.809E+08	7.770E+08	4.033E+08	2.246E+08	2.331E+08	4.884E+08	1.565E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE127M	2.984E-01	5.380E-04	1.018E+09	1.203E+10	2.050E+09	5.180E+06	3.341E+07	1.613E+05	1.550E+05	2.919E+04	1.587E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129	1.323E-04	3.573E-03	1.240E+10	8.806E+09	1.547E+09	2.194E+09	1.299E+09	8.251E+08	1.569E+09	3.959E+08	2.805E+09	1.232E+08	2.235E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	9.199E-02	1.754E-03	2.220E+09	9.768E+09	1.820E+09	2.971E+08	1.776E+08	1.610E+08	1.469E+08	5.587E+07	1.447E+09	2.964E+08	1.358E+07	1.117E+01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TH227	5.124E-02	3.651E-02	1.883E+10	1.069E+08	1.839E+08	3.004E+09	3.511E+09	4.366E+08	1.114E+10	2.124E+09	1.846E+06	6.253E+06	1.517E+04	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
TH228	1.913E+00	3.269E-02	4.292E+09	0.000E+00	0.000E+00	0.000E+00	4.366E+08	4.366E+07	1.284E+08	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
TH229	7.339E+03	3.060E-02	4.033E+10	1.502E+07	3.774E+09	2.194E+08	2.198E+10	2.749E+09	3.996E+09	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
TH230	7.700E+04	2.831E-02	3.885E+09	0.000E+00	0.000E+00	1.658E+08	0.000E+00	1.917E+07	7.548E+06	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.769E+01	8.880E+00	6.512E+00	1.943E+00	1.258E-01	7.955E-03
TH231	2.911E-03	5.610E-04	1.802E+09	5.883E+09	1.695E+08	3.471E+08	3.411E+09	2.375E+08	8.288E+07	1.092E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TH232	1.405E+10	2.420E-02	3.012E+09	0.000E+00	0.000E+00	5.698E+07	0.000E+00	1.602E+07	3.101E+02	0.000E+00	0.000E+00	1.236E+02	7.030E+01	3.522E+01	1.765E+01	8.843E+00	6.512E+00	1.939E+00	1.258E-01	7.955E-03
TH234	6.597E-02	4.054E-04	6.401E+09	2.294E+08	1.088E+08	1.698E+09	2.320E+09	9.102E+07	9.472E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	7.966E-06	9.034E-03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL207	9.069E-06	2.937E-03	1.092E+10	2.253E+09	1.458E+09	2.065E+09	1.214E+09	7.696E+08	9.509E+08	3.363E+08	1.154E+08	1.088E+08	7.067E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL208	5.837E-06	2.353E-02	1.358E+10	2.594E+09	1.687E+09	2.409E+09	3.811E+09	9.250E+08	4.773E+09	4.551E+08	3.959E+10	5.291E+09	1.576E+08	7.585E+05	0.000E+00	3.511E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00
U232	7.200E+01	3.210E-02	6.031E+09	0.000E+00	0.000E+00	7.807E+07	0.000E+00	2.864E+07	1.691E+06	1.099E+06	6.438E+03	1.236E+02	7.030E+01	3.526E+01	1.769E+01	8.880E+00	6.512E+00	1.946E+00	1.262E-01	7.992E-03
U233	1.585E+05	2.907E-02	3.030E+09	7.733E+06	3.700E+07	7.326E+06	1.565E+07	1.291E+07	1.550E+07	5.920E+06	3.637E+04	1.236E+02	7.030E+01	3.526E+01	1.769E+01	8.880E+00	6.512E+00	1.946E+00	1.265E-01	8.029E-03
U234	2.445E+05	2.880E-02	5.032E+09	0.000E+00	0.000E+00	4.033E+07	0.000E+00	1.432E+07	3.123E+02	0.000E+00	1.684E+04	1.092E+03	7.104E+01	3.556E+01	1.787E+01	8.991E+00	6.623E+00	1.987E+00	1.310E-01	8.547E-03
U235	7.038E+08	2.619E-02	2.213E+10	0.000E+00	3.378E+07	6.586E+06	2.886E+09	5.698E+09	2.017E+10	5.032E+07	2.475E+06	3.374E+05	2.542E+02	1.251E+02	6.993E+01	3.922E+01	3.371E+01	1.362E+01	1.469E+00	1.624E-01
U236	2.341E+07	2.708E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	6.327E+06	5.920E+02	0.000E+00	0.000E+00	3.060E+02	1.550E+02	7.659E+01	4.181E+01	2.283E+01	1.906E+01	7.326E+00	7.474E-01	7.918E-02
U237	1.848E-02	1.893E-03	3.090E+10	1.147E+09	1.628E+08	1.450E+10	6.734E+09	1.173E+10	8.843E+09	5.217E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
U238	4.468E+09	2.536E-02	4.218E+09	0.000E+00	0.000E+00	2.231E+07	0.000E+00	0.000E+00	1.210E+05	0.000E+00	0.000E+00	7.770E+04	3.615E+04	1.769E+04	1.025E+04	5.957E+03	5.328E+03	2.290E+03	2.638E+02	3.034E+01
XE131M	3.258E-02	9.620E-04	1.273E+09	1.939E+10	3.386E+09	0.000E+00	0.000E+00	0.000E+00	5.291E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE133	1.436E-02	1.071E-03	3.034E+09	3.608E+08	1.480E+10	2.375E+08	1.325E+10	3.885E+07	3.056E+07	2.412E+06	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y90	7.301E-03	5.543E-03	4.181E+10	8.954E+09	5.920E+09	8.695E+09	5.439E+09	3.637E+09	5.106E+09	2.353E+09	1.251E+09	3.885E+08	1.284E+08	1.003E+07	1.103E+03	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y91	1.602E-01	3.592E-03	1.343E+10	2.816E+09	1.839E+09	2.631E+09	1.580E+09	1.018E+09	1.317E+09	5.143E+08	2.068E+08	3.774E+07	1.106E+08	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN65	6.677E-01	3.505E-03	1.188E+10	3.737E+07	2.442E+07	3.489E+07	2.083E+07	1.332E+07	1.680E+07	7.104E+06	9.620E+08	1.258E+06	1.676E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR93	1.530E+06	1.162E-04	2.442E+08	1.573E+07	3.056E+06	2.338E+05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR95	1.752E-01	5.066E-03	2.412E+09	4.292E+08	2.520E+08	2.986E+08	1.291E+08	5.661E+07	2.731E+07	1.106E+06	1.269E+05	3.215E+10	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

7. RESULTS

The results of the estimate are presented on a Fuel Radionuclide Inventory Worksheet. Appendixes C and D provide Fuel Radionuclide Inventory Worksheets for the years 2010 and 2030 for each DOE SNF record in the SFD. These dates, respectively, represent the estimated timeframes for packaging and shipment of fuels to the repository and for completion of emplacement of fuels in the repository. The results include a nominal and bounding source term estimate along with the associated heat generation rates and photon emission spectra. Appendix B provides an index by site and fuel name that supplies the SNF ID number, Total System Performance Assessment category, design basis event category, and page number for the 2010 and 2030 source term estimates.

To facilitate checking and to aid the analyst in determining the uncertainty associated with the estimate, the worksheet displays all input used within the estimate, including any assumptions that were necessary in order to compensate for lack of information. Each Fuel Radionuclide Worksheet contains three sections. Each of these sections provides information that can be used to assess the potential uncertainty associated with the estimate. Section I includes header information that identifies the fuel being estimated and the template used in the estimate as well as the size and estimated number of DOE SNF canisters for the fuel.

Section II shows for each of the 45 radionuclides of interest, the factors used in the linear estimate (Y = mx+b) where m represents the change in curies relative to the change in burnup; x is the burnup; b is the initial curie content; and Y is the resulting estimate. These 45 radionuclides include all radionuclides identified as important for Total System Performance Assessment and preclosure safety analysis (see Table 6). Although 145 radionuclides are calculated and included in the totals, the Fuel Radionuclide Worksheet displays only the 45 shown in Table 6.

Section III includes subsections for Template Selection Summary, Burnup Summary, and Checks. The Template Selection Summary subsection provides the information relied on to select an appropriate template (i.e., the reactor moderator, the fuel cladding, the fuel compound, and the fuel enrichment). A table is provided that identifies these parameters as given for the fuel along with those of the template selected and the basis for any differences.

The Burnup Summary subsection provides a table that identifies both the given and estimated burnup along with the basis (i.e., from the SFD, estimated or calculated). For conservatism, the larger of the given and estimated is used for the estimates given in Section II. The basis may include one or more of the following messages.

Burnup (bounding or nominal) taken from SFD and converted to MWd using BOL = ... This message indicates that in Block 7 of Figure 1 the BOL heavy metal mass was estimated in order to convert a burnup given in units of MWd per MTIHM BOL to units of MWd. The estimate calculates BOL heavy metal mass as described in Block 7 of Section 6.

Nominal burnup calculated from the heavy metal mass destroyed. This message indicates that the nominal burnup was calculated by converting the fission energy for the heavy metal atoms fissioned to MWd. In other words, the nominal burnup was calculated using Equation 1 of Figure 1 (Block 4 or 8 of Figure 1).

Nominal burnup set equal to bounding burnup. This message indicates that information is not available to support an estimation of the nominal burnup, but the bounding burnup was either provided or estimated. In this case, the nominal burnup is conservatively assumed to be the same as the bounding burnup. In other words, the nominal burnup was conservatively estimated using Equation 2 of Figure 1 (Block 5 or 9 of Figure 1).

Table 6. List of radionuclides shown on output.

Radionuclides	Total System Performance Assessment (TSPA)[a] Dose Contribution		Preclosure Safety Analysis (PSA)[b]
	Up to 1×10^4 yrs	$> 1 \times 10^4$ to 10^8 yrs	
AC227	0.95	0.95	X
AM241	0.95		X
AM242M			X
AM243	0.95	0.95	X
C14	0.95	0.95	
CL36	0.99	0.95	
CM243			X
CM244	0.99		X
CO60			X
CS134			X
CS135	0.95	0.95	
CS137	0.95		X
EU154			X
EU155			X
FE55			X
H3			X
I129	0.95	0.95	X
KR85			X
NP237	0.95	0.95	X
PA231	0.95	0.95	X
PB210	0.99	0.95	
PM147			X
PU238	0.95		X
PU239	0.95	0.95	X
PU240	0.95	0.95	X
PU241	0.99		X
PU242	0.99	0.95	X
RA226	0.95 (EPA)	0.95 (EPA)	
RA228	EPA	EPA	
RU106			X
SE79		0.95	
SN126	0.99	0.95	
SR90	0.95		X
TC99	0.95	0.95	
TH229	0.95	0.95	X
TH230		0.95	
TH232	0.99	0.95	X
TI208*			
U232	0.95		X
U233	0.95	0.95	X
U234	0.95	0.95	X
U235	0.99	0.99	
U236	0.99	0.95	X
U238	0.95	0.95	X
Y90			X

0 95 = For postclosure analysis, the doses from the radionuclide total to 0 95 fraction
0 99 = For postclosure analysis, the additional doses from the radionuclide totals to 0 99 fraction
EPA = Additional isotopes required by EPA 10CFR197 30 and 10CFR63.331 for ground water protection standard
* = Thallium-208 is shown on the list because it is the dominant contributor to Group 14 of the photon emission spectra out to 100 years.
a. Office of Civilian Radioactive Waste Management, "Radionuclide Screening," ANL-WIS-MD-000006 Rev. 01, August 2002 Tables 10 and 11. MOL.20020923 0177.
b. Office of Civilian Radioactive Waste Management, "Significant Radionuclides Determination," CAL-WHS-SE-000002 Rev. 00, July 2001 Tables 5. MOL.20010905 0143

Bounding burnup assumed to be twice nominal burnup. This message indicates that information is not available to support an estimation of the bounding burnup but the nominal burnup was either provided or estimated. In this case, the bounding burnup was conservatively assumed to be twice the nominal burnup. In other words, the bounding burnup was estimated using Equation 3 of Figure 1 (Block 6 or 8 of Figure 1).

Bounding burnup estimated using BOL heavy metal and enrichment. This message indicates that the bounding burnup was conservatively estimated by assuming 100% depletion of the initial fissile inventory. This allows burnup estimates to proceed in the event that only BOL information is available. In other words, the bounding burnup was conservatively estimated using Equation 4 of Figure 1 (Block 9 of Figure 1).

Nominal burnup assumed 2% of BOL Heavy Metal mass. This message indicates that the BOL and EOL heavy metal masses were equal, and therefore, nominal burnup was conservatively estimated by assuming 2% burnup of the initial heavy metal mass (see Block 9 of Section 6).

Nominal (or Bounding) burnup taken directly from SFD (converted to MWd). This message indicates that the burnup was given (from SFD) in MWd/MTIHM and was converted to MWd using BOL heavy metal mass and was used for scaling the template (see Block 7 of Section 6).

Bounding burnup estimated by assuming BOL heavy metal mass was twice EOL. This message indicates that the nominal burnup was calculated using Equation 1 of Figure 1 after assuming that half the original heavy metal mass has fissioned. The estimate assumes BOL heavy metal mass was twice the EOL heavy metal mass, which is equivalent to assuming that the heavy metal destroyed (enrichment times burnup) is equal to the heavy metal remaining (i.e., one half of the original heavy metal 0.5). This is a conservative assumption for virtually all DOE SNF (see discussion in Section 6, Block 10) and allows burnup estimates to proceed in the event that only EOL information is available. Because of the conservatism of this assumption, the bounding burnup is set equal to the estimated nominal burnup.

The Checks subsection provides the burnup multiplier and, when possible, the ratios of the estimated (i.e., calculated nominal and bounding) burnups and the estimated EOL heavy metal mass with those provided from the SFD. The burnup multiplier is the ratio of the specific burnup (i.e., burnup per MTIHM) of the fuel being estimated over the specific burnup of the template fuel. A burnup multiplier indicates the portion of the linear scaling that accounts for differences in specific burnup. For example, a burnup multiplier of 1 indicates that any scaling accounts for a different mass of fuel with the same specific burnup. As noted previously, error is not introduced when scaling to account for different masses of fuel. Scaling to account for different specific burnups, however, could introduce error when estimating inventories of actinides and activation products. Consequently, the magnitude of the burnup multiplier provides an indication of the potential error associated with this linear approximation. If the burnup multiplier is greater than 10 or less than 0.1, then the inventories of actinides and activation products might be suspect.

When the heavy metal masses at BOL and EOL are provided, the nominal burnup is back-calculated from the depleted heavy metal mass. This calculated nominal value as well as the estimated bounding value is compared against the burnups (nominal and bounding) given in the SFD. This ratio gives an indication of the integrity (i.e., internal self-consistency) of the input data. Similarly, the heavy metal masses in the estimated radionuclide inventory are summed and compared to the given EOL heavy metal mass of the fuel record. The ratio between the estimated and the given EOL heavy metal mass of the fuel is simply another cross-check that may alert the analyst of potential uncertainty associated with the data or the estimate. If the answer for the estimated EOL over the given EOL is 1 (or close to 1), then the estimate matches the given fuel record for EOL heavy metal mass. For spent fuel records where it is necessary to estimate the burnup based on BOL = 2*EOL, the ratio may be close to 2. If the burnup multiplier is greater than 10, the EOL ratio may be thrown off to about 2. If the "worst case" template is used, then the heavy metal mass is intentionally inflated to be conservative. In this case, the ratio may be as large as about 600. The total given EOL heavy metal for all DOE SNF included in this analysis is about 2,411 MTHM. The total estimated EOL heavy metal for 2010 nominal case is about 2,527 MTHM. This difference is due to the use of the "worst case" template (by 31 fuel records).

Appendixes C and D include for each fuel record in the SFD a Fuel Radionuclide Inventory Worksheet that estimates the radionuclide inventory and associated thermal heat generation rates and photon emission rates at 2010 and 2030 respectively. Appendix C also includes tables summarizing the total DOE SNF radionuclide inventory, which is broken down by canister type and design basis event.[22] Appendix D also includes tables summarizing the total DOE SNF radionuclide inventory, which is broken down by type and Total System Performance Assessment group.[23]

8. UNCERTAINTY AND ERROR

This report provides the results and summarizes the analytical processes employed to estimate the radiological inventories associated with DOE-owned SNFs. The accuracy of the estimates is affected both by the accuracy of the input data relied on as well as the simplifications introduced by the methodology itself. A brief discussion of the overall accuracy of the estimates and of the conservative nature of the methods used to account for missing or questionable input information is given below. A more detailed discussion of the conservatisms applied to the estimates and their affect on the uncertainty of the resulting estimates is included in Appendix E.

The source term estimate for about 90% of the DOE SNF inventory (in terms of MTHM) is based on a validated ORIGEN output (i.e., a fuel template) that was developed for that fuel type. As an example, the N-Reactor template was developed specifically to model the N-Reactor fuel and was validated against available N-reactor fuel data. Thus, applying the N-Reactor template to the N-Reactor fuel introduces minimal uncertainty. Approximately 10% of the DOE SNF inventory uses a template that, although based on another fuel type, shares parameters (i.e., reactor moderator, fuel compound, enrichment, cladding) that dominate the model with respect to generation of radionuclides. An example is the use of the N-Reactor fuel template for a uranium metal fuel such as Single Pass Reactor fuels that have a slightly different configuration (tube type fuel design vs. concentric tubes design for the N-Reactor fuels). For about 0.2% of the fuels in the inventory, the burnup information is uncertain because of missing or incomplete BOL and burnup value. For these fuels, BOL heavy metal is assumed to be two times the EOL value, which would overestimate the burnup for over 96% of the DOE SNFs (see Section 6, Block 10). This assumption produces a very conservative estimate of the burnup, which results in an increased scaling of the template radionuclide inventories. The remainder (~0.11%) of the DOE SNF uses a Worst Case template that was derived by taking the highest normalized (Ci/MTU) values for each radionuclide from all the available templates. Although such a fuel does not physically exist, this template is used to bound fuel materials in the DOE SNF inventory for which a template cannot be selected. There are two reasons for this; the unavailability of a template that adequately models the fuel or the unavailability of sufficient information to identify a proper template. An example of such case is the DOE TEST & Experimental Fuels (DOE SFD record number 42 and 857). These two records consist of various different fuels and cladding types in which much of the fuel has been destructively examined. Because it is difficult to precisely identify how much of each of the fuel materials remained after the postirradiation examination, the radionuclide inventory for these fuel records are estimated using the Worst Case template.

Figure 3 illustrates the effects of the conservative assumptions used to compensate for missing information. This figure clearly shows an inverse correlation between the available information used in the methodology and the resulting radionuclide concentrations estimated. This is expected because, as noted above, the methodology applies conservative assumptions to compensate for missing or questionable information. This same phenomenon is evident in Figure 4, which shows the radionuclide concentrations (curies per metric ton of heavy metal). Figure 4 clearly illustrates that the assumptions used to compensate for missing fuel information drive the estimates toward higher radionuclide concentrations.

As shown in Figure 3, 38% (18% + 14% + 6%) of the radionuclide inventory comes from the result of 0.31% of the total fuel. This 0.31% of the mass relied heavily on conservative assumptions to compensate for missing information. Consequently by assuming this 0.31% of fuel with sufficient information can be reasonably represented by the 99.69% of fuels with sufficient information, one may conclude that the expected value for the radionuclide inventory for DOE SNF is more reasonably about 62% of the nominal inventory estimated.

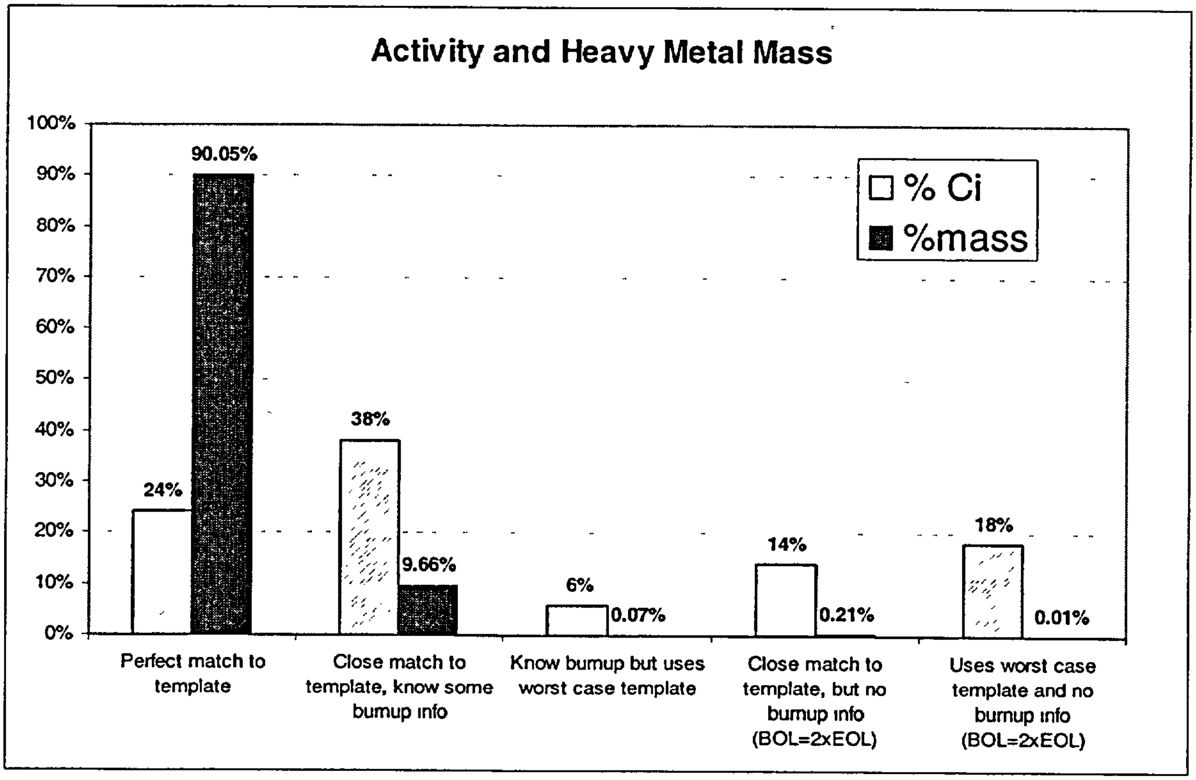

Figure 3. Activity and quantity of SNF relating to known information about SNF.

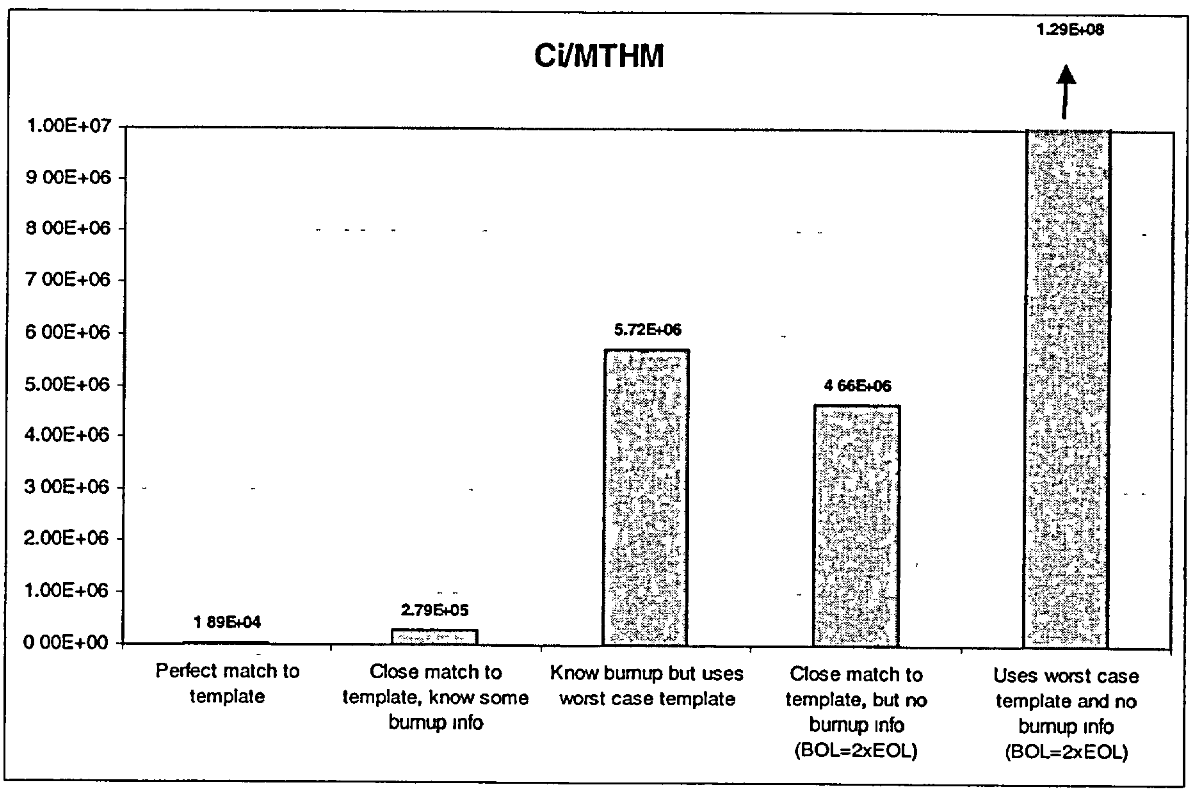

Figure 4. Activity per MTHM relating to known information about SNF.

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CONTENTS

Appendix A

Template Selection Guide and Templates

Appendix A

Template Selection Guide and Templates

On the following page a table is provided listing each of the templates as outlined in report DOE/SNF/REP-059. The templates that were not completed have a TBD in the Page Number column and have a crosswalk to an acceptable completed template. An appropriate template is selected based on the reactor moderator, the fuel compound, BOL enrichment, and cladding. When possible, a template is selected that matches all four of these parameters. If one or more of these parameters is not known, a conservative assumption may be applied. If a template matching all four parameters is not available, an alternative template may be applied in accordance with the template selection guide. When matching a fuel record to a template, the order of importance of the four criteria is: 1–reactor moderator, 2–fuel type, 3–cladding, and 4–enrichment.

Following the template table is a detailed writeup of each of the completed templates. (The page numbers of each template are shown in the template table.) Note that some templates use 365.25 days per year for the decay calculations and some templates use 365 days per year. This should produce a very minimal error in the output.

Templates and Template Selection Guide

Reactor Type (moderator)	Fuel Clad	BOL Enrichment	BOL Heavy Metal Constituents	Template No.	Template Fuel	Alternate Template No.[a]	Page No.
Fast	Stainless Steel	60 to 100%	Pu and U	1	TBD[d]	3	TBD[d]
Fast	Stainless Steel	30 to 100%	U	2	TBD	5	TBD
Fast	Stainless Steel	10 to 30%	Pu and U	3	FFTF	NA	A-5
Fast	Stainless Steel	0 to 5%	U	4	TBD	5	TBD
Fast	Zirconium	10 to 40%	U	5	Fermi	NA	A-20
Graphite	Graphite	60 to 100%	Th and U	6	Ft. St. Vrain	NA	A-39
Graphite	Zirconium	0 to 5%	U	7	N-Reactor	NA	A-51
Heavy Water	Aluminum	40 to 100%	U	8	HFBR	NA	A-64
Heavy Water	Aluminum	10 to 20%	U	9	Modified HFBR	NA	A-74
Heavy Water	Stainless Steel	0 to 5%	U	10	Modified HFBR	NA	A-84
Heavy Water	Zirconium	0 to 5%	U	11	Modified HFBR	NA	A-96
Light Water	Aluminum	60 to 100%	U	12	ATR	NA	A-108
Light Water	Aluminum	40 to 60%	U	13	TBD	12	TBD
Light Water	Aluminum	10 to 20%	U	14	TBD	12	TBD
Light Water	Stainless Steel	60 to 100%	U	15	Pathfinder	NA	A-121
Light Water	Stainless Steel	60 to 100%	Th and U	16	TBD	21	TBD
Light Water	Unclad	40 to 60%	U	17	TBD	15	TBD
Light Water	Stainless Steel	10 to 20%	U	18	TBD	15	TBD
Light Water	Stainless Steel	5 to 10%	Th and U	19	TBD	21	TBD
Light Water	Stainless Steel	0 to 5%	U	20	TBD	15	TBD
Light Water	Zirconium	60 to 100%	Th and U	21	LWBR	NA	A-134
Light Water	Zirconium	60 to 100%	U	22	TBD	15	TBD
Light Water	Zirconium	5 to 20%	U	23	TBD	24	TBD
Light Water	Zirconium	0 to 5%	U	24	PWR	NA	A-151
Light Water	Zirconium	0 to 5%	Pu and U	25	TBD	29	TBD
LW/U-Zrx[b]	Aluminum	10 to 20%	U	26	TRIGA-Al	NA	A-162
LW/U-Zrx[b]	Stainless Steel	60 to 100%	U	27	TRIGA-FLIP	NA	A-174
LW/U-Zrx[b]	Stainless Steel	10 to 20%	U	28	TRIGA-SS	NA	A-187
	Inconel and Stainless Steel		U-Pu-Th		Hypothetical	NA	A-200
All Else	Composite[c]	Composite[c]	Composite[c]	29	Worst Case	NA	A-211

a. This column specifies the available template that was used in this analysis.

b. Light water and uranium-zirconium-hydride (LW/U-Zrx) moderated reactor.

c. This template does not represent any real or postulated fuel. It includes the maximum normalized (per MWd per kg) radionuclide content for each radionuclide from each of the other templates

d. The templates with a TBD designation were not completed due to time and funding constraints.

Template 3

Fuel-Specific Source Term Calculations
Fast Flux Test Facility (FFTF) Fuel

Introduction

The Fast Flux Test Facility (FFTF) spent nuclear fuel (SNF) currently resides at the U.S. Department of Energy (DOE) Hanford Site. The total FFTF SNF inventory represents approximately 0.25% of the total uranium mass in the DOE SNF inventory.

The radionuclide inventory or source term used for the FFTF template is based on a radionuclide inventory calculated by the Hanford site personnel (References 1 and 2). The Hanford calculation represents a relatively comprehensive list of radionuclides; however, the reported inventory does not provide activity estimates for all of the radionuclides identified in the "Guide for Estimating DOE Spent Nuclear Fuel Source Terms" (Reference 3). In order to provide these additional radionuclide activity estimates, a complementary FFTF fuel assembly depletion calculation was performed by the Idaho National Engineering and Environmental Laboratory (INEEL) National SNF Program personnel.

The INEEL complementary calculation was designed to use the same input data as the Hanford calculation and match the reported Hanford radionuclide activities. Matching activities provided the verification basis for using the INEEL calculated additional radionuclides in the template inventory here. The INEEL FFTF fuel assembly depletion calculation used the same input data, namely burnup (152,230 MWd/MTHM), assembly heavy metal isotopic masses, assembly structural masses, assembly geometry, and FFTF reactor data.

In order to reproduce the Hanford calculation, the INEEL calculational methodology (Reference 4) was invoked to generate beginning-of-life (BOL) FFTF fuel assembly neutron cross sections and perform the depletion calculation. Good agreement was obtained between the Hanford and INEEL depletion calculation results and, as a consequence, the additional radionuclide activity estimates were taken directly from the INEEL output and used to supplement the Hanford data as needed.

Fast Flux Test Facility

The FFTF was a 400 MW(th), liquid sodium-cooled, fast flux test reactor, which is owned by DOE and located on the Hanford Site. The FFTF mission was to provide testing capability for US advanced reactor programs and the production of medical radioisotopes. In 1993, the FFTF was ordered into a safe shutdown condition, and in 2002 the FFTF was ordered to be permanently shut down and defueled. During its 10-year operation, the FFTF irradiated a wide variety of fuels, including the FFTF driver fuel, potential driver fuels, and related advanced fuel systems.

Fast Flux Test Facility Fuel Assembly Data

An FFTF driver assembly is 144 inches long and is a hexagonal bundle of 217 wire-wrapped fuel pins, encased in a stainless steel duct. Figures 1 and 2 show an FFTF standard driver fuel assembly with the major features and dimensions identified.

The FFTF fuel is a mixed oxide (MOX) of uranium and plutonium oxides. The uranium enrichment is 0.2% U-235 (or depleted uranium), and the plutonium enrichment is 86% Pu-239. The plutonium heavy metal mass fraction is 29% Pu/[U+Pu]. Over the course of the FFTF operation, there were four different types of driver assemblies. These assemblies differed in fissile load, but maintained the same basic physical geometry. Of these four assemblies, the assembly with the highest plutonium content (Type 4.1) was chosen for the Hanford high burnup depletion calculation.

The cladding and duct material for the driver assembly are 316 stainless steel (SS-316). The depletion calculations were performed using material masses that encompass only the active 36-in. (91.44-cm) long core region. Hence, the structural material, the small Inconel or depleted uranium spacers, and the SS-316 end fixtures were ignored because of the relatively low neutron fluence and minimal expected activation in these regions above and below the fuel column.

Selected FFTF standard driver fuel assembly design characteristics are listed below:

Fuel Bundle:	Hexagonal array of 217 wire-wrapped pins
Fuel Pin Pitch:	Triangular, 0.726-cm pin-to-pin centers
Fuel Pellet Diameter:	0.494 cm
Fuel Material:	Mixed U/Pu oxide
U Enrichment:	0.2% U-235 BOL (depleted uranium)
Pu Enrichment:	86% Pu-239 BOL
Pellet Density [%TD]:	90.4
Smeared Density [%TD]:	85.5
Oxygen/Metal Atom Ratio:	1.96
Active Fuel Length:	91.44 cm
Fuel Pin Length:	237.5 cm
Assembly Length:	365.8 cm
Cladding Outer Diameter:	5.84 mm
Cladding Thickness:	0.38 mm
Cladding Material:	Stainless Steel 316 (SS-316)
Pellet-Cladding Gap Thickness:	0.14 mm (diametral)
Wire Diameter:	1.422 mm
Wire/Duct Material:	SS-316
Coolant :	Liquid Sodium
Average Coolant Density:	0.846 g/cc (443.5°C)

Fast Flux Test Facility Assembly Fuel Compositions/Masses

Table 1 lists the Hanford-supplied FFTF fuel assembly compositions/masses that include the heavy metal uranium and plutonium isotopic masses in a single assembly. In addition, the oxygen in the MOX fuel is given along with the total SS-316 structural mass for a single assembly. These data are part of the ORIGEN input data and are used in both the Hanford and INEEL activation/depletion calculations. Note: These masses differ slightly from the descriptive text in Reference 2.

Fast Flux Test Facility Assembly Structural Constituents and Impurities

Table 2 lists the major SS-316 constituent elements and impurities needed for the activation calculation. Column 1 lists both the major constituents and impurity elements in the SS-316. Column 2 is the Hanford-supplied element weight percents for the major constituents. These Column 2 data are used in both the Hanford and INEEL calculations. The Column 3 data are the impurity concentrations in ppm per Reference 5. The INEEL calculation in addition includes these impurity data in the activation calculation.

Burnup

The burnup chosen for this template is 152,230 MWd/MTHM as specified by the Hanford calculation (Reference 2). This encompasses all the spent FFTF driver fuel assemblies as well as the Test Driver fuel assemblies. The 275 Test Driver fuel assemblies are basically the same in terms of geometry and heavy metal loading as the standard driver assemblies, with the exception of minor variations in the SS-316 cladding composition (such as titanium additions to the generic SS-316 cladding alloy). Most of the standard driver assemblies have burnups in the range of 70,000 to 90,000 MWd/MTHM. Only three FFTF experiments exceeded the 150,000 MWd/MTHM burnup value. For calculational purposes here, the FFTF fuel assembly burnup is assumed to be continuous over 928 equivalent full power days (EFPDs) at an assembly power of 5.4 MW.

The relatively high burnup (152,230 MWd/MTIHM) is conservative for the buildup of fission products, activation products, and minor actinides in the source term and nonconservative with regard to criticality safety and fissile isotope concentrations, in particular Pu-239.

Cross Sections

The Hanford calculations used here were performed in 1991. An independent review of the calculations was performed in 1999 by Hanford personnel. The Hanford reviewers were familiar with both the FFTF and the ORIGEN code (Reference 6). The calculations were found to be consistent and correct based on available information. However, the available ORIGEN output does not specifically list the cross sections used in the calculations, and therefore, they are not independently verifiable.

The corresponding INEEL depletion calculation generated BOL cross sections based on the FFTF data given above and the methodology described in Reference 4. These neutron cross sections were used in the INEEL burnup or depletion calculation for the generation of activity estimates for the additional radionuclides required for the single FFTF fuel assembly source template inventory. Cross sections for 37 actinides were updated in a standard ORIGEN2 liquid metal fast reactor library. The FFTF specific cross sections take into account neutron flux spectral and spatial characteristics of the FFTF and assembly geometry and materials.

An explicit triangular pitch unit cell model with reflective boundary conditions was developed for the MCNP4B computer code (Reference 7) to represent an FFTF fuel assembly. This model was used to calculate the volume-averaged fluxes and reaction rates for the 37 actinides. These data were then converted into 1-group cross sections for use in the ORIGEN2 depletion or activation calculation.

Burnup Calculation

Table 3 summarizes the power or exposure history used in the INEEL burnup or source term calculations for a single FFTF fuel assembly. Following the burnup or exposure period, the radionuclide activities are decayed for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100 years. These decay or cooling times correspond to the specified time periods in Reference 3. The radionuclides and their corresponding activities included in the Hanford calculations are incorporated directly into the template. The Hanford-provided radionuclide inventory is given only for decay times of 5, 10, 20, 50, and 100 years. Therefore, interpolation of these data were required in order to provide estimates for the other five decay dates. The INEEL calculation was used to supplement the Hanford data by providing activity estimates for the radionuclides not reported in Reference 2. The radionuclides/activities reported in Reference 2 and the interpolated values are designated separately in the table as are the INEEL-generated radionuclides/activities.

The goal for the FFTF template was to use the Hanford-provided data where possible and not try to reproduce this data. The simplest means of interpolation was the linear interpolation in order to fill in the other missing decay time vectors not supplied by Hanford. With the exception of a few low concentration daughter decay products, the relatively long-lived 41 radionuclides decay in a nice smooth exponential fashion. Linear interpolation is a reasonable approach to estimating the intermediate time vector activities. Any error introduced from linear interpolation results in an estimated activity that is greater than the actual value. This is in line with the basic template philosophy of erring conservatively.

The Hanford depletion calculation used the ORIGEN2 code to calculate the radionuclide concentrations that follow in the attached template. The source terms are for a single 217 pin FFTF MOX assembly. Masses of material, burnup, and power level are as indicated above. Radionuclide activities in the template are presented as a function of decay time after shutdown.

Similarly, the INEEL depletion calculation also used the ORIGEN2 computer code to calculate radionuclide inventory for a single FFTF fuel assembly. The fuel element masses and impurities (graphite, uranium, and thorium), neutron cross sections, burnup, power history, and power level as discussed above are input data for the ORIGEN2 calculation.

References

1. K. H. Bergsman, "Hanford Spent Fuel Inventory Baseline," WHC-SD-SNF-TI-001, 1994.

2. R. L. Symons, "Radiation Source Term for the Fuel Irradiated to 150 MWd/kg Burnup," RLS-91-010, 1991, to be published as HNF-5495.

3. National Spent Nuclear Fuel Program, *Guide for Estimating DOE Spent Nuclear Fuel Source Terms*, DOE/SNF/REP-059, July 2000.

4. J. W. Sterbentz and C. A. Wemple, *Calculational Burnup Methodology and Validation for the Idaho National Engineering Laboratory Spent Nuclear Fuels*, INEL-96/0304, September 1996.

5. J. C. Evans et al., *Long-Lived Activation Products in Reactor Materials*, NUREG/CR-3474, Prepared for the U.S. Nuclear Regulatory Commission by Battelle, Pacific Northwest Laboratory, Richland, WA, August 1984.

6. A. G. Croff, *ORIGEN2—A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, July 1980.

7. "MCNP4B: Monte Carlo N-Particle Transport Code System," contributed by the Transport Methods Group, Los Alamos National Laboratory and distributed by the Radiation and Safety Information Computational Center as code package CCC-660, April 1997.

Table 1. FFTF driver fuel isotopic constituents and masses in a single assembly.

Isotope/Element	Mass (g)	Heavy Metal Mass Fraction
Pu-239	8382.9	0.254660
Pu-240	1162.5	0.035316
Pu-241	115.4	0.003504
Pu-242	18.5	0.000563
Total Pu	9679.3	0.294042
Am-241	18.5	0.000561
U-235	49.5	0.001504
U-238	23170.8	0.703892
Total U	23220.3	0.705396
Total heavy metal	32918.1	1.000000
Oxygen	4347.6	
SS-316	21327.8	
Total	58593.5	

Table 2. FFTF SS-316 structural material constituent and impurity concentrations.

Constituent or Impurity	SS-316 Cladding Concentration (wt%)	SS-316 Cladding Concentration (Ref. 5) (ppm)
H		
Li		0.18
Be		
B	0.002	
C	0.06	
N	0.01	
O		
F		
Na		6
Mg		
Al	0.05	
Si	0.75	
P	0.04	
S	0.01	
Cl		
K		3
Ca		14
Sc		
Ti		200
V	0.04	
Cr	18.00	
Mn	2.00	
Fe	61.848	
Co	0.05	
Ni	14.00	
Cu	0.04	
Zn		71
Ga		60
As	0.03	
Se		9
Br		2
Rb		
Sr		0.23
Y		5
Zr		6

Table 2. (continued).

Constituent or Impurity	SS-316 Cladding Concentration (wt%)	SS-316 Cladding Concentration (Ref. 5) (ppm)
Nb	0.05	
Mo	3.00	
Ag		5
Cd		
In		
Sn		
Sb		13
Cs		
Ba		
La		0.2
Ce		
Pr		
Nd		
Sm		0.2
Eu		0.07
Gd		
Tb		9
Dy		
Ho		1
Er		
Tm		
Yb		2
Lu		0.8
Hf		
Ta	0.02	
W		218
Tl		
Pb		30
Bi		
Th		
U		5

Table 3. Assumed power and decay history for the FFTF fuel assembly used in the INEEL template depletion calculation.

Duration (days)	Cumulative Duration (days)	Time-Averaged Power (MW_{th})
928	928	5.4
1826.25	2754.25	0.0
1826.25	4580.50	0.0
1826.25	6406.75	0.0
1826.25	8233.00	0.0
1826.25	10059.25	0.0
3652.5	13711.75	0.0
5478.75	19190.50	0.0
5478.75	24669.25	0.0
5478.75	30148.00	0.0
7305.00	37453.00	0.0

The ten dates with zero associated power represent the ten different cooling or decay dates after exposure. These ten dates are specifically the 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100-year decay times in accordance with the template format.

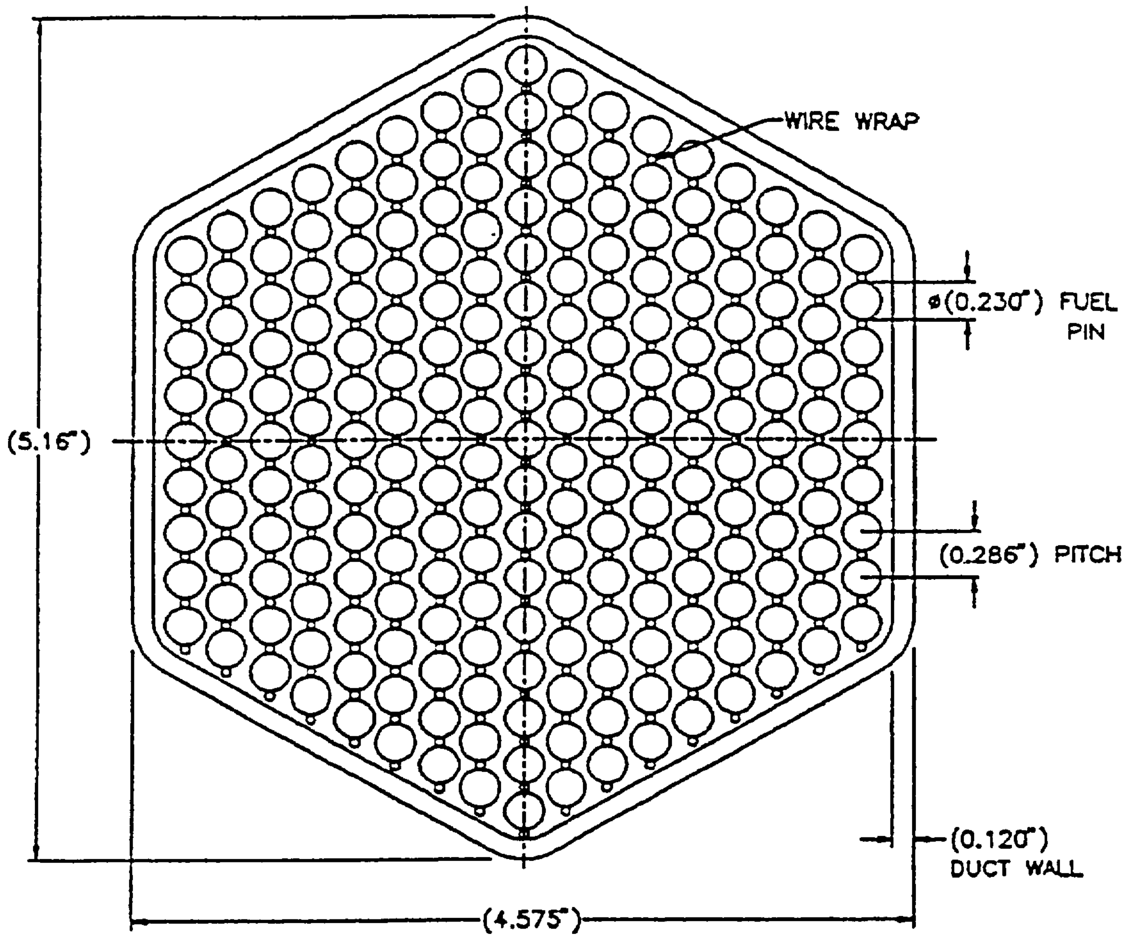

Figure 1. Fast Flux Test Facility fuel pin bundle cross section.

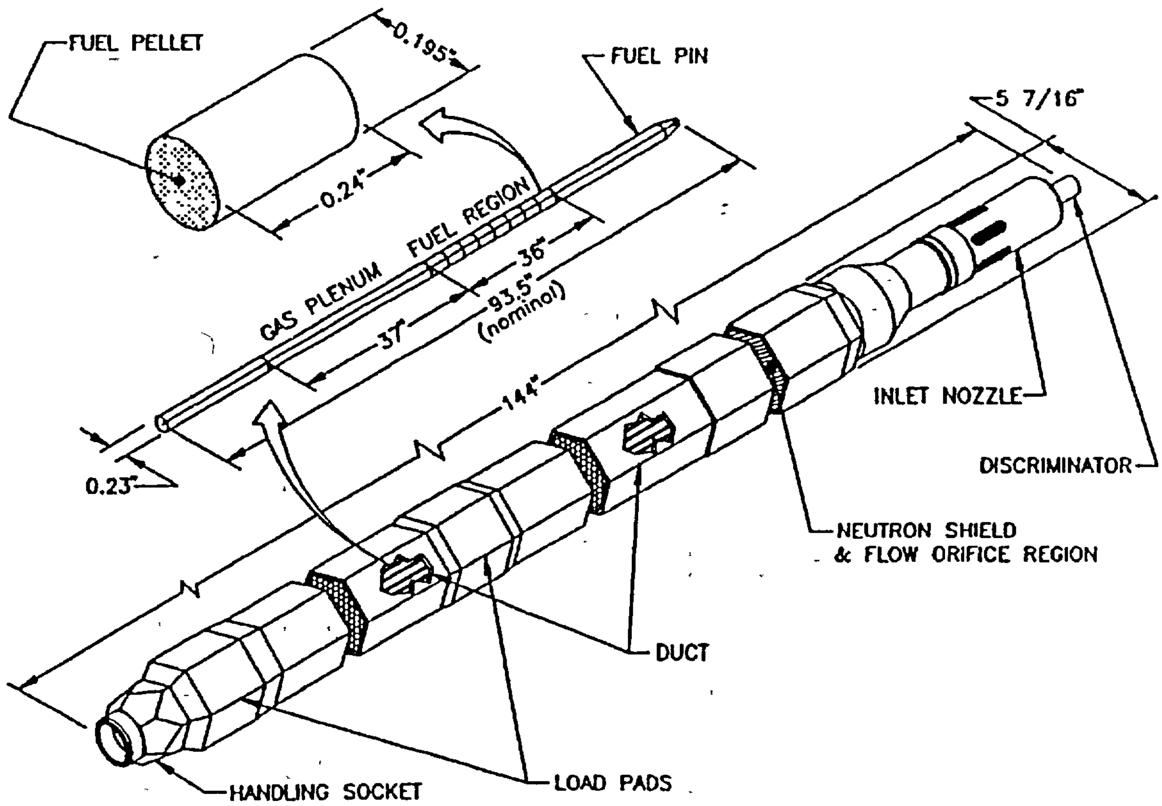

Figure 2. Fast Flux Test Facility standard driver fuel assembly.

Fast Flux Test Facility Element

Stainless Steel Cladding, MOX Fuel

Coolant:	Liquid Sodium
Fuel Meat:	MOX
Clad:	Stainless Steel 316
Burnup:	152,230.0 MWd/MTHM
Burnup:	5,011.2 MWd/single assembly (high burnup)
Basis of Calculation:	Single fuel assembly
BOL U-235:	49.5 grams U-235 per assembly
BOL U-238:	23,170.8 grams U-238 per assembly
BOL Total U per Assembly:	23,220.3 grams U per assembly
BOL Pu-239	8,382.9 grams Pu-239 per assembly
BOL Pu-240	1,162.5 grams Pu-240 per assembly
BOL Pu-241	115.4 grams Pu-241 per assembly
BOL Pu-242	18.5 grams Pu-242 per assembly
BOL Am-241	18.5 grams Am-241 per assembly
BOL Total Pu/Am per Assembly	9,697.8 grams Pu/Am per assembly
BOL U Enrichment	0.2% U-235
BOL Pu Enrichment:	86.4% Pu-239

DECAY TIMES (years out of core)

(Activities* in Ci/assembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
AC227	8.726E-10	2.297E-09	4.321E-09	6.883E-09	9.928E-09	1.729E-08	3.098E-08	4.729E-08	6.579E-08	9.343E-08
AG110	1.505E-01	9.491E-04	5.988E-06	3.778E-08	2.383E-10	9.486E-15	2.382E-21	5.982E-28	1.502E-34	2.379E-43
AG110M	1.131E+01	7.136E-02	4.502E-04	2.840E-06	1.792E-08	7.132E-13	1.791E-19	4.497E-26	1.129E-32	1.789E-41
AG111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AM241	**2.150E+02**	**3.190E+02**	*3.905E+02*	**4.620E+02**	*4.880E+02*	*5.400E+02*	**6.180E+02**	*6.186E+02*	*6.192E+02*	**6.200E+02**
AM242	**1.100E+01**	**1.070E+01**	*1.050E+01*	**1.030E+01**	*1.007E+01*	*9.620E+00*	**8.940E+00**	*8.394E+00*	*7.848E+00*	**7.120E+00**
AM242M	**1.100E+01**	**1.080E+01**	*1.055E+01*	**1.030E+01**	*1.008E+01*	*9.645E+00*	**8.990E+00**	*8.441E+00*	*7.892E+00*	**7.160E+00**
AM243	5.397E-01	5.394E-01	5.392E-01	5.389E-01	5.387E-01	5.382E-01	5.374E-01	5.366E-01	5.359E-01	5.349E-01
BA136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA137M	**1.250E+04**	**1.120E+04**	*1.003E+04*	**8.850E+03**	*8.113E+03*	*6.638E+03*	**4.425E+03**	*3.516E+03*	*2.606E+03*	**1.394E+03**
BA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BE10	1.382E-06	1.382E-06	1.382E-06	1.382E-06	1.382E-06	1.382E-06	1.382E-06	1.382E-06	1.382E-06	1.382E-06
BI211	8.729E-10	2.298E-09	4.326E-09	6.892E-09	9.941E-09	1.730E-08	3.101E-08	4.734E-08	6.586E-08	9.351E-08
BI212	4.124E-03	6.614E-03	7.349E-03	7.354E-03	7.119E-03	6.504E-03	5.632E-03	4.875E-03	4.220E-03	3.484E-03
C14	**1.310E-01**	**1.310E-01**	*1.310E-01*	**1.310E-01**	*1.308E-01*	*1.305E-01*	**1.300E-01**	*1.300E-01*	*1.300E-01*	**1.300E-01**

DECAY TIMES (years out of core)
(Activities* in Ci/assembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
CD113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113M	1.378E+01	1.087E+01	8.569E+00	6.757E+00	5.328E+00	3.313E+00	1.625E+00	7.966E-01	3.906E-01	1.510E-01
CD115M	2.210E-10	1.039E-22	4.884E-35	2.296E-47	1.080E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	2.920E-12	3.596E-29	4.429E-46	5.455E-63	6.718E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	3.496E-06	3.496E-06	3.496E-06	3.496E-06	3.496E-06	3.496E-06	3.496E-06	3.496E-06	3.496E-06	3.496E-06
CE144	**1.510E+03**	**1.760E+01**	*8.801E+00*	**2.380E-03**	*1.983E-03*	*1.190E-03*	**5.930E-15**	*4.151E-15*	*2.372E-15*	**2.710E-34**
CL36	1.716E-06	1.716E-06	1.716E-06	1.716E-06	1.716E-06	1.716E-06	1.716E-06	1.716E-06	1.716E-06	1.716E-06
CM242	**1.160E+01**	**8.880E+00**	*8.680E+00*	**8.480E+00**	*8.300E+00*	*7.940E+00*	**7.400E+00**	*6.947E+00*	*6.494E+00*	**5.890E+00**
CM243	4.224E+00	3.741E+00	3.312E+00	2.933E+00	2.597E+00	2.036E+00	1.414E+00	9.818E-01	6.817E-01	4.191E-01
CM244	**2.160E+01**	**1.780E+01**	*1.500E+01*	**1.220E+01**	*1.081E+01*	*8.030E+00*	**3.860E+00**	*2.873E+00*	*1.885E+00*	**5.690E-01**
CM245	8.881E-03	8.877E-03	8.873E-03	8.870E-03	8.866E-03	8.859E-03	8.848E-03	8.837E-03	8.826E-03	8.812E-03
CM246	4.769E-04	4.765E-04	4.762E-04	4.758E-04	4.755E-04	4.748E-04	4.737E-04	4.727E-04	4.717E-04	4.703E-04
CM247	3.033E-09	3.033E-09	3.033E-09	3.033E-09	3.033E-09	3.033E-09	3.033E-09	3.033E-09	3.033E-09	3.033E-09
CO60	**2.430E+02**	**1.260E+02**	*7.985E+01*	**3.370E+01**	*2.819E+01*	*1.718E+01*	**6.520E-01**	*4.567E-01*	*2.613E-01*	**9.080E-04**
CR51	5.593E-17	8.069E-37	1.164E-56	1.679E-76	2.423E-96	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS134	**2.410E+03**	**4.490E+02**	*2.323E+02*	**1.560E+01**	*1.300E+01*	*7.800E+00*	**6.490E-04**	*4.543E-04*	*2.596E-04*	**3.260E-11**
CS135	2.390E-01	2.390E-01	2.390E-01	2.390E-01	2.390E-01	2.390E-01	2.390E-01	2.390E-01	2.390E-01	2.390E-01
CS136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS137	**1.320E+04**	**1.180E+04**	*1.058E+04*	**9.360E+03**	*8.580E+03*	*7.019E+03*	**4.678E+03**	*3.717E+03*	*2.755E+03*	**1.473E+03**
EU152	5.655E+00	4.383E+00	3.397E+00	2.633E+00	2.040E+00	1.226E+00	5.707E-01	2.657E-01	1.237E-01	4.463E-02
EU154	**4.980E+02**	**3.330E+02**	*2.410E+02*	**1.490E+02**	*1.264E+02*	*8.110E+01*	**1.320E+01**	*9.311E+00*	*5.421E+00*	**2.350E-01**
EU155	**1.110E+03**	**5.500E+02**	*3.430E+02*	**1.360E+02**	*1.137E+02*	*6.903E+01*	**2.053E+00**	*1.438E+00*	*8.223E-01*	**1.890E-03**
EU156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FE55	**2.080E+02**	**5.480E+01**	*2.931E+01*	**3.810E+00**	*3.175E+00*	*1.906E+00*	**1.280E-03**	*8.960E-04*	*5.120E-04*	**2.080E-09**
FE59	7.510E-11	4.559E-23	2.767E-35	1.680E-47	1.020E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FR223	1.204E-11	3.169E-11	5.964E-11	9.499E-11	1.370E-10	2.385E-10	4.275E-10	6.526E-10	9.079E-10	1.289E-09
GD153	1.082E-01	5.791E-04	3.098E-06	1.658E-08	8.870E-11	2.539E-15	3.890E-22	5.959E-29	9.129E-36	7.483E-45
H3	**7.550E+01**	**5.700E+01**	*4.475E+01*	**3.250E+01**	*2.809E+01*	*1.927E+01*	**6.040E+00**	*4.338E+00*	*2.635E+00*	**3.650E-01**
I129	**6.460E-03**	**6.460E-03**	*6.460E-03*	**6.460E-03**	*6.460E-03*	*6.460E-03*	**6.460E-03**	*6.460E-03*	*6.460E-03*	**6.460E-03**
I131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	5.629E-12	4.440E-23	3.502E-34	2.762E-45	2.178E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	5.882E-12	4.639E-23	3.659E-34	2.886E-45	2.276E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	1.553E-14	7.301E-27	3.433E-39	1.614E-51	7.587E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
KR85	**6.450E+02**	**4.670E+02**	*3.555E+02*	**2.440E+02**	*2.092E+02*	*1.396E+02*	**3.510E+01**	*2.499E+01*	*1.487E+01*	**1.390E+00**

DECAY TIMES (years out of core)
(Activities* in Ci/assembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
MN54	1.690E+02	2.930E+00	1.465E+00	8.890E-04	7.408E-04	4.445E-04	2.480E-14	1.736E-14	9.920E-15	6.330E-32
MO93	2.980E-02	2.977E-02	2.974E-02	2.971E-02	2.968E-02	2.962E-02	2.953E-02	2.945E-02	2.936E-02	2.924E-02
NB93M	5.116E-02	8.077E-02	1.037E-01	1.215E-01	1.353E-01	1.543E-01	1.695E-01	1.766E-01	1.799E-01	1.817E-01
NB94	1.384E-01	1.384E-01	1.384E-01	1.384E-01	1.383E-01	1.383E-01	1.382E-01	1.381E-01	1.381E-01	1.380E-01
NB95	1.176E-03	3.005E-12	7.682E-21	1.964E-29	5.019E-38	3.279E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB95M	3.929E-06	1.004E-14	2.567E-23	6.561E-32	1.677E-40	1.096E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND144	1.410E-10	1.416E-10	1.416E-10	1.416E-10	1.416E-10	1.416E-10	1.416E-10	1.416E-10	1.416E-10	1.416E-10
ND147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NI59	1.115E+00	1.115E+00	1.115E+00	1.115E+00	1.115E+00	1.115E+00	1.114E+00	1.114E+00	1.114E+00	1.114E+00
NI63	2.090E+01	2.010E+01	1.940E+01	1.870E+01	1.807E+01	1.680E+01	1.490E+01	1.349E+01	1.208E+01	1.020E+01
NP237	1.160E-02	1.210E-02	1.270E-02	1.330E-02	1.422E-02	1.605E-02	1.880E-02	2.186E-02	2.492E-02	2.900E-02
PA231	7.779E-09	1.320E-08	1.879E-08	2.454E-08	3.047E-08	4.281E-08	6.258E-08	8.385E-08	1.066E-07	1.393E-07
PA233	9.701E-03	1.012E-02	1.068E-02	1.136E-02	1.212E-02	1.384E-02	1.668E-02	1.965E-02	2.266E-02	2.662E-02
PA234	9.062E-06	9.062E-06	9.062E-06	9.062E-06	9.062E-06	9.062E-06	9.062E-06	9.062E-06	9.062E-06	9.062E-06
PA234M	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03
PB210	1.110E-09	9.814E-10	9.566E-10	1.111E-09	1.540E-09	3.685E-09	1.213E-08	3.041E-08	6.267E-08	1.345E-07
PB211	8.729E-10	2.298E-09	4.326E-09	6.892E-09	9.941E-09	1.730E-08	3.101E-08	4.734E-08	6.586E-08	9.351E-08
PB212	4.124E-03	6.614E-03	7.349E-03	7.354E-03	7.119E-03	6.504E-03	5.632E-03	4.875E-03	4.220E-03	3.484E-03
PD107	3.263E-02	3.263E-02	3.263E-02	3.263E-02	3.263E-02	3.263E-02	3.263E-02	3.263E-02	3.263E-02	3.263E-02
PM145	2.559E-06	2.134E-06	1.755E-06	1.443E-06	1.186E-06	8.020E-07	4.458E-07	2.477E-07	1.377E-07	6.292E-08
PM147	8.270E+03	2.210E+03	1.184E+03	1.570E+02	1.308E+02	7.853E+01	5.670E-02	3.969E-02	2.268E-02	1.040E-07
PM148	7.326E-11	3.568E-24	1.737E-37	8.461E-51	4.121E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148M	1.301E-09	6.334E-23	3.085E-36	1.502E-49	7.316E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PO212	2.642E-03	4.238E-03	4.708E-03	4.712E-03	4.561E-03	4.167E-03	3.609E-03	3.123E-03	2.704E-03	2.232E-03
PO215	8.729E-10	2.298E-09	4.326E-09	6.892E-09	9.941E-09	1.730E-08	3.101E-08	4.734E-08	6.586E-08	9.351E-08
PO216	4.124E-03	6.614E-03	7.349E-03	7.354E-03	7.119E-03	6.504E-03	5.632E-03	4.875E-03	4.220E-03	3.484E-03
PR143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144	1.510E+03	1.760E+01	8.801E+00	2.380E-03	1.983E-03	1.190E-03	5.930E-15	4.151E-15	2.372E-15	2.710E-34
PR144M	1.810E+01	2.110E-01	1.055E-01	2.860E-05	2.383E-05	1.430E-05	7.110E-17	4.977E-17	2.844E-17	3.250E-36
PU236	5.526E-02	1.639E-02	4.859E-03	1.441E-03	4.276E-04	3.794E-05	1.358E-06	4.038E-07	3.789E-07	3.782E-07
PU237	2.949E-12	2.592E-24	2.278E-36	2.002E-48	1.760E-60	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	1.550E+02	1.490E+02	1.435E+02	1.380E+02	1.169E+02	7.455E+01	1.110E+01	3.081E+01	5.052E+01	7.680E+01
PU239	3.430E+02	3.430E+02	3.430E+02	3.430E+02	3.430E+02	3.430E+02	3.430E+02	3.427E+02	3.424E+02	3.420E+02
PU240	3.690E+02	3.690E+02	3.690E+02	3.690E+02	3.688E+02	3.685E+02	3.680E+02	3.674E+02	3.668E+02	3.660E+02

DECAY TIMES (years out of core)

(Activities* in Ci/assembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
PU242	1.260E-01	1.260E-01	1.260E-01	1.260E-01	1.262E-01	1.265E-01	1.270E-01	1.270E-01	1.270E-01	1.270E-01
PU244	5.621E-09	5.621E-09	5.621E-09	5.621E-09	5.621E-09	5.621E-09	5.621E-09	5.621E-09	5.621E-09	5.621E-09
RA223	8.729E-10	2.298E-09	4.326E-09	6.892E-09	9.941E-09	1.730E-08	3.101E-08	4.734E-08	6.586E-08	9.351E-08
RA224	4.124E-03	6.614E-03	7.349E-03	7.354E-03	7.119E-03	6.504E-03	5.632E-03	4.875E-03	4.220E-03	3.484E-03
RA226	6.989E-11	4.202E-10	1.283E-09	2.880E-09	5.427E-09	1.418E-08	3.935E-08	8.319E-08	1.500E-07	2.812E-07
RA228	3.162E-14	9.100E-14	1.736E-13	2.754E-13	3.941E-13	6.770E-13	1.207E-12	1.858E-12	2.630E-12	3.845E-12
RB87	1.853E-06	1.853E-06	1.853E-06	1.853E-06	1.853E-06	1.853E-06	1.853E-06	1.853E-06	1.853E-06	1.853E-06
RH103M	2.761E-09	2.793E-23	2.826E-37	2.859E-51	2.892E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	4.550E+03	1.460E+02	7.308E+01	1.510E-01	1.258E-01	7.550E-02	1.660E-10	1.162E-10	6.640E-11	1.940E-25
RN219	8.729E-10	2.298E-09	4.326E-09	6.892E-09	9.941E-09	1.730E-08	3.101E-08	4.734E-08	6.586E-08	9.351E-08
RN220	4.124E-03	6.614E-03	7.349E-03	7.354E-03	7.119E-03	6.504E-03	5.632E-03	4.875E-03	4.220E-03	3.484E-03
RU103	3.063E-09	3.098E-23	3.134E-37	3.171E-51	3.208E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	4.550E+03	1.460E+02	7.308E+01	1.510E-01	1.258E-01	7.550E-02	1.660E-10	1.162E-10	6.640E-11	1.940E-25
SB124	3.254E-07	2.398E-16	1.767E-25	1.302E-34	9.590E-44	5.206E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	1.160E+03	3.310E+02	1.791E+02	2.710E+01	2.259E+01	1.356E+01	1.490E-02	1.043E-02	5.960E-03	5.470E-08
SB126	3.081E-02	3.081E-02	3.081E-02	3.081E-02	3.081E-02	3.080E-02	3.080E-02	3.080E-02	3.079E-02	3.079E-02
SB126M	2.201E-01	2.201E-01	2.201E-01	2.201E-01	2.200E-01	2.200E-01	2.200E-01	2.200E-01	2.200E-01	2.199E-01
SE79	5.080E-02	5.080E-02	5.080E-02	5.080E-02	5.078E-02	5.075E-02	5.070E-02	5.070E-02	5.070E-02	5.070E-02
SM145	6.827E-07	1.650E-08	3.989E-10	9.641E-12	2.330E-13	1.361E-16	1.923E-21	2.715E-26	3.834E-31	1.309E-37
SM147	8.211E-07	9.674E-07	1.006E-06	1.017E-06	1.020E-06	1.021E-06	1.021E-06	1.021E-06	1.021E-06	1.021E-06
SM151	5.060E+02	4.870E+02	4.690E+02	4.510E+02	4.355E+02	4.045E+02	3.580E+02	3.238E+02	2.896E+02	2.440E+02
SN119M	3.524E-01	2.011E-03	1.147E-05	6.545E-08	3.734E-10	1.216E-14	2.258E-21	4.195E-28	7.791E-35	8.258E-44
SN121M	1.625E-01	1.516E-01	1.415E-01	1.320E-01	1.231E-01	1.072E-01	8.706E-02	7.071E-02	5.743E-02	4.351E-02
SN123	5.659E-02	3.138E-06	1.740E-10	9.644E-15	5.347E-19	1.643E-27	2.800E-40	4.772E-53	8.131E-66	7.681E-83
SN125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN126	2.201E-01	2.201E-01	2.201E-01	2.201E-01	2.200E-01	2.200E-01	2.200E-01	2.200E-01	2.200E-01	2.199E-01
SR89	1.068E-06	1.386E-17	1.799E-28	2.335E-39	3.031E-50	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR90	4.820E+03	4.280E+03	3.825E+03	3.370E+03	3.083E+03	2.510E+03	1.650E+03	1.306E+03	9.618E+02	5.030E+02
TB160	1.971E-05	4.914E-13	1.225E-20	3.051E-28	7.602E-36	4.721E-51	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC99	1.975E+00	1.975E+00	1.975E+00	1.975E+00	1.975E+00	1.975E+00	1.975E+00	1.975E+00	1.974E+00	1.974E+00
TE123M	1.210E-04	3.083E-09	7.856E-14	2.003E-18	5.103E-23	3.314E-32	5.486E-46	9.082E-60	1.504E-73	6.342E-92
TE125M	2.820E+02	8.070E+01	4.366E+01	6.610E+00	5.509E+00	3.307E+00	3.630E-03	2.541E-03	1.452E-03	1.340E-08
TE127	3.032E-02	2.743E-07	2.482E-12	2.246E-17	2.032E-22	1.664E-32	1.233E-47	9.134E-63	6.767E-78	4.537E-98
TE127M	3.095E-02	2.801E-07	2.534E-12	2.293E-17	2.075E-22	1.699E-32	1.259E-47	9.325E-63	6.909E-78	4.632E-98

DECAY TIMES (years out of core)
(Activities* in Ci/assembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TE129M	3.892E-13	1.691E-29	7.349E-46	3.193E-62	1.387E-78	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TH227	8.608E-10	2.267E-09	4.266E-09	6.797E-09	9.804E-09	1.706E-08	3.058E-08	4.668E-08	6.495E-08	9.223E-08
TH228	4.124E-03	6.611E-03	7.343E-03	7.348E-03	7.113E-03	6.503E-03	5.632E-03	4.875E-03	4.220E-03	3.484E-03
TH229	7.254E-09	7.479E-09	7.808E-09	8.247E-09	8.804E-09	1.030E-08	1.364E-08	1.852E-08	2.522E-08	3.746E-08
TH230	7.875E-08	2.630E-07	5.523E-07	9.432E-07	1.432E-06	2.690E-06	5.232E-06	8.490E-06	1.239E-05	1.849E-05
TH231	5.042E-05	5.201E-05	5.360E-05	5.519E-05	5.678E-05	5.995E-05	6.471E-05	6.947E-05	7.423E-05	8.057E-05
TH232	1.203E-13	2.298E-13	3.528E-13	4.891E-13	6.388E-13	9.783E-13	1.588E-12	2.317E-12	3.166E-12	4.484E-12
TH234	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03	6.971E-03
TL206	5.435E-16	5.435E-16	5.435E-16	5.435E-16	5.435E-16	5.435E-16	5.435E-16	5.435E-16	5.435E-16	5.435E-16
TL207	8.704E-10	2.292E-09	4.314E-09	6.873E-09	9.914E-09	1.725E-08	3.092E-08	4.720E-08	6.568E-08	9.325E-08
TL208	1.482E-03	2.376E-03	2.640E-03	2.642E-03	2.558E-03	2.337E-03	2.024E-03	1.752E-03	1.516E-03	1.252E-03
U232	6.177E-03	7.383E-03	7.479E-03	7.259E-03	6.957E-03	6.333E-03	5.483E-03	4.745E-03	4.107E-03	3.388E-03
U233	3.736E-07	5.899E-07	8.169E-07	1.058E-06	1.314E-06	1.880E-06	2.879E-06	4.070E-06	5.456E-06	7.610E-06
U234	2.898E-03	5.275E-03	7.570E-03	9.786E-03	1.193E-02	1.599E-02	2.157E-02	2.661E-02	3.115E-02	3.652E-02
U235	**5.270E-05**	**5.440E-05**	*5.615E-05*	**5.790E-05**	*5.958E-05*	*6.295E-05*	**6.800E-05**	*7.307E-05*	*7.814E-05*	**8.490E-05**
U236	4.170E-04	4.713E-04	5.255E-04	5.798E-04	6.340E-04	7.423E-04	9.045E-04	1.066E-03	1.228E-03	1.443E-03
U237	3.612E-03	2.839E-03	2.232E-03	1.754E-03	1.379E-03	8.522E-04	4.139E-04	2.011E-04	9.767E-05	3.729E-05
U238	**6.890E-03**	**6.890E-03**	*6.890E-03*	**6.890E-03**	*6.890E-03*	*6.890E-03*	**6.890E-03**	*6.890E-03*	*6.890E-03*	**6.890E-03**
XE131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y90	**4.820E+03**	**4.280E+03**	*3.825E+03*	**3.370E+03**	*3.083E+03*	*2.510E+03*	**1.650E+03**	*1.306E+03*	*9.618E+02*	**5.030E+02**
Y91	4.549E-05	1.826E-14	7.329E-24	2.942E-33	1.181E-42	1.903E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZN65	1.011E-02	5.627E-05	3.132E-07	1.744E-09	9.709E-12	3.010E-16	5.194E-23	8.963E-30	1.547E-36	1.486E-45
ZR93	1.924E-01	1.924E-01	1.924E-01	1.924E-01	1.924E-01	1.924E-01	1.924E-01	1.924E-01	1.924E-01	1.924E-01
ZR95	5.296E-04	1.354E-12	3.460E-21	8.844E-30	2.261E-38	1.477E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00

* Four decimal places of accuracy are as reported by ORIGEN2 output and are not significant for many radionuclides.
Bold text denotes data supplied by Hanford.
Italicized text denotes data interpolated from the Hanford data.

Template 5

Fuel-Specific Source Term Calculations FERMI Subassembly Fuel

Introduction

The following data have been used in the Idaho National Engineering and Environmental Laboratory (INEEL) spent nuclear fuel source term calculational methodology to generate a source term template for a single FERMI spent nuclear fuel subassembly. This single subassembly source term uses a core average burnup based on the 101 driver subassemblies from Core A-2. The data sources for the analysis are documented herein, and the INEEL calculational methodology is described in detail in Reference 1.

FERMI Reactor History

Over the lifetime of the FERMI liquid metal fast breeder reactor (LMFBR), two separate cores were operated. These two cores are designated as Cores A-1 and A-2. Both cores contained fuel subassemblies of similar design and uranium loading, but each core had a slightly different number of total driver subassemblies. Core A-1 operated from August 23, 1963, to October 6, 1966, and accumulated a total core burnup of approximately 636.7 MWD. Core A-2 operated from September 23, 1970, to December 2, 1971, and accumulated a total core burnup of approximately 5,926.0 MWD, or more than nine times Core A-1.

From the INEEL inventory record data (Reference 2), there are 104 driver fuel subassemblies of low burnup and 101 subassemblies with a relatively higher burnup currently stored at the INEEL. These data suggest that the 104 subassemblies are from Core A-1 and the 101 higher burnup subassemblies are from Core A-2.

FERMI Reactor Data

The FERMI reactor core and fuel elements are described in some detail in Reference 3. Data from this reference has been used to develop reactor physics models needed to develop neutron cross sections for the fuel depletion and radionuclide inventory analysis.

The FERMI subassemblies consist of a stainless steel can containing a 12×12 array of rods. The $2.646 \times 2.454 \times 34.594$-in. steel can has a 0.096-in. thick wall. Inside the can, the 12×12 array consists of 140 fuel pins and 4 corner pins (stainless steel). The fuel pin meat is a uranium-molybdenum (U-Mo) metal rod with 10 wt% molybdenum metal. The fuel pin clad is zircaloy and is bonded to the fuel meat. The fuel meat and pin diameters are 0.148 in. and 0.158 in., respectively. The uranium metal is medium enriched at 25.6 wt% U-235 at beginning-of-life (BOL). The pin pitch within the subassembly is 0.2 in. and the subassembly pitch in the core is approximately 2.693 in..

The following data provide specific fuel element dimensions, materials, densities, enrichment, etc. which are typical for a FERMI driver subassembly. The BOL data below were used in the fuel depletion calculation for the FERMI subassembly source term generation.

Fuel subassembly:	12 × 12 array of fuel pins in stainless steel can
No. of Rods:	140 fuel rods, 4 stainless steel corner pins
Fuel Rod Meat:	Uranium/molybdenum metal alloy
Fuel Meat Density:	17.32 g/cc (10 wt% Mo)
Fuel Rod Meat Diameter:	0.148 in.
Fuel Rod Meat Length:	30.5 in.
Uranium Enrichment:	25.6 wt % U-235
Heavy Metal Loading per rod:	34.33 g/rod U-235/rod (BOL)
	99.77 g/rod U-238/rod (BOL)
	134.10 g/rod TOTAL U

Heavy Metal Loading per subassembly:

4,806.144	g/rod U-235/subassembly (BOL)
13,967.856	g/rod U-238/subassembly (BOL)
18,774.000	g/subassembly TOTAL U

Molybdenum Metal Loading:	14.90 g/rod Mo (BOL)
Molybdenum Metal Loading:	2,086.00 g/subassembly Mo (BOL)
Clad:	Zircaloy
Clad Outer Diameter:	0.158 in.
Clad Pin Length:	32.06 in.
Clad Density:	6.44 g/cc
Clad Thickness:	0.005 in.
Total Zircaloy Mass:	1,138.48 g/subassembly
Can Dimensions:	2.646 × 2.454 × 34.594 in.
	0.96-in. wall thickness
Can Material:	Stainless Steel 304
Steel Density:	7.92 g/cc
Can Steel Mass:	4,382.89 g/subassembly
Steel Corner Pins (4) Mass:	325.35 g/subassembly
Total Steel Mass:	4,708.24 g/subassembly
Coolant:	Liquid metal sodium
Coolant Temperature:	800°F
Coolant Density:	0.85 g/cc

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for the material components in a single FERMI subassembly. In addition, for the ORIGEN2 (Reference 4) depletion calculation, conservative and detailed impurity concentrations were added for the zircaloy clad (References 6 and 7) and the Stainless Steel 304 can/corner pins

(References 8, 9, and 10). Table 1 lists the impurities and corresponding concentrations used in the calculations.

Burnup

The core burnup sustained by the Core A-2 subassemblies was chosen for this template. The Core A-2 burnup was substantially higher than Core A-1. The Core A-2 subassemblies accumulated a total core burnup of approximately 5,926.0 MWD, or an average subassembly burnup of approximately 58.67 MWD per subassembly for the 101 subassemblies in this core. This subassembly average burnup translates into a 1.4% U-235 depletion, or a 3,125.1 MWD/MTU per subassembly. This burnup is conservative with respect to the buildup of fission products, activation products, and minor actinides in the source term, but nonconservative with regard to criticality safety, in particular U-235 and U-238 end-of-life concentrations.

The Core A-2 power history profile is based on Reference 5. Table 2 gives the accumulated days over which Core A-2 operated along with the corresponding accumulated burnup in megawatt-days (MWD) and the reactor thermal power in megawatts (MW_{th}) for the 101 core subassemblies. Note in Table 2 that there are many time periods in which the reactor power is zero; these represent reactor shutdowns. Also, the single, average-burnup FERMI subassembly burnup is based on 1/101 of the total reactor power and forms the basis the FERMI template.

Cross-Section Development

The neutron cross sections used in the burnup or depletion calculation for the source term generation of a single average-burnup FERMI subassembly are based on the methodology described in Reference 3. Cross sections from a standard ORIGEN2 LMFBR library were updated once using BOL cross sections specifically developed to account for the unique FERMI neutron flux spatial and spectral characteristics to ensure accurate calculation of the fission product and actinide production as a function of burnup.

In order to calculate the BOL FERMI neutron cross sections, an explicit FERMI 1/8-core model was developed with reflective boundary conditions on the radial surfaces. The reflective surfaces created the transport effect of a full core. Figures 1 and 2 show cross sectional views of the MCNP computer model (Reference 11).

FERMI Subassembly Exposure History

Table 2 summarizes the FERMI total core power or exposure history. In the actual depletion calculation for the single, average-burnup FERMI subassembly, $1/101^{th}$ of the total core power was used as the subassembly power output in the burnup or source term calculations. Following the burnup or exposure period, the radionuclide activities are decayed for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100 years.

Burnup Calculation

The ORIGEN2 computer code was used to perform the depletion or burnup calculation for a single FERMI subassembly. The fuel subassembly masses and impurities, neutron cross sections, burnup, power history, and power level as discussed above are input data for the ORIGEN2 calculation. The ORIGEN2 output or radionuclide concentrations are given as a function of time in the attached template table representing a single average-burnup FERMI subassembly.

The 145 radionuclides listed in the template represent greater than 99.99% of the total curie inventory had all 684 activation products, 880 fission products, and 127 actinide/daughter isotopes from the ORIGEN2 output been included in the template.

References

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11. "MCNP4B: Monte Carlo N-Particle Transport Code System," contributed by the Transport Methods Group, Los Alamos National Laboratory and distributed by the Radiation and Safety Information Computational Center as code package CCC-660, April 1997.

Table 1. Zircaloy and SS-304 material constituent and impurity concentrations.

Constituent or Impurity	Zircaloy Concentration (wt%)	Stainless Steel Concentration (ppm)
H	0.002497	
Li		0.13
Be		
B	0.00005	
C	0.026968	0.08 wt%
N	0.00799	525
O	0.094887	
Na		37
Mg		
Al	0.007491	200
Si	0.011986	1.00 wt%
P	0.009988	
S	0.003496	
Cl		130
K		3
Ca		19
Sc		0.03
Ti	0.004994	600
V	0.004994	690
Cr	0.124851	18.40 wt%
Mn	0.004994	1.53 wt%
Fe	0.224731	68.99 wt%
Co	0.001998	2570
Ni	0.006992	10.00 wt%
Cu	0.004994	8150
Zn	0.009988	2230
Ga		450
As		1010
Se		70
Br		8
Rb		10
Sr		0.2
Y		5
Zr	97.789992	20
Nb	0.006992	300
Mo	0.004994	5500

Table 1. (continued).

Constituent or Impurity	Zircaloy Concentration (wt%)	Stainless Steel Concentration (ppm)
Ag		2
Cd	0.000050	
In		
Sn	1.598089	
Sb		17
Cs		0.3
Ba		500
La		2.1
Ce		550
Pr		
Nd		
Sm	0.000999	0.15
Eu		0.02
Gd	0.000499	
Tb		0.71
Dy		1
Ho		1
Er		
Tm		
Yb		2
Lu		0.8
Hf	0.003496	2
Ta	0.019976	
W	0.009988	520
Tl		
Pb	0.009988	139
Bi		
Th	0.000699	1
U	0.000350	2

Table 2. FERMI Core A.2 power history.

Cumulative Operational (days)	Cumulative Burnup (MWD)	Total Core Power (MW_{th})
1	1.3	1.30
2	11.1	9.80
4	37.1	13.00
5	37.1	0.00
6	61.7	24.60
7	68.8	7.10
8	68.8	0.00
10	137.6	34.40
16	137.6	0.00
18	289.6	76.00
19	289.6	0.00
20	336.0	46.40
22	382.8	23.40
23	382.8	0.00
27	974.1	147.83
52	974.1	0.00
59	2157.7	169.09
61	2170.9	6.60
62	2238.9	68.00
65	2238.9	0.00
67	2346.3	53.70
73	2346.3	0.00
74	2346.7	0.40
77	2495.2	49.50
103	2495.2	0.00
107	2863.6	92.10
110	2863.6	0.00
111	2864.1	0.50
112	2869.0	4.90
113	2922.8	53.80
132	2922.8	0.00
133	2925.3	2.50
135	3061.2	67.95
139	3061.2	0.00
140	3067.8	6.60
211	3067.8	0.00
213	3223.2	77.70
247	3223.2	0.00
248	3242.1	18.90
254	3242.1	0.00
258	3806.2	141.03
261	3806.2	0.00
263	3971.1	82.45
275	3971.1	0.00
279	4363.3	98.05
423	4363.3	0.00
426	4675.0	103.90
427	4675.0	0.00
435	5926.0	156.38
2261.25	5926.0	0.00
4087.5	5926.0	0.00
5913.75	5926.0	0.00
7740	5926.0	0.00
9566.25	5926.0	0.00
13218.75	5926.0	0.00
18697.5	5926.0	0.00
24176.25	5926.0	0.00
29655	5926.0	0.00
36960	5926.0	0.00

The ten dates with zero associated power represent the ten different cooling or decay dates after exposure. These ten dates are specifically the 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100-year cooling times designated for the template methodology.

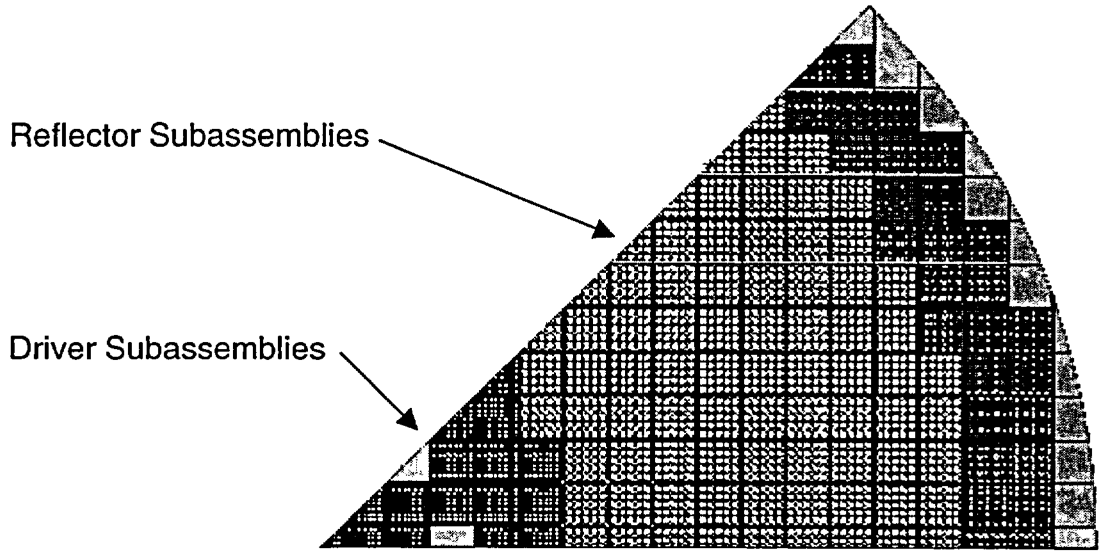

Figure 1. MCNP 1/8-core model cross-sectional view of the FERMI core.

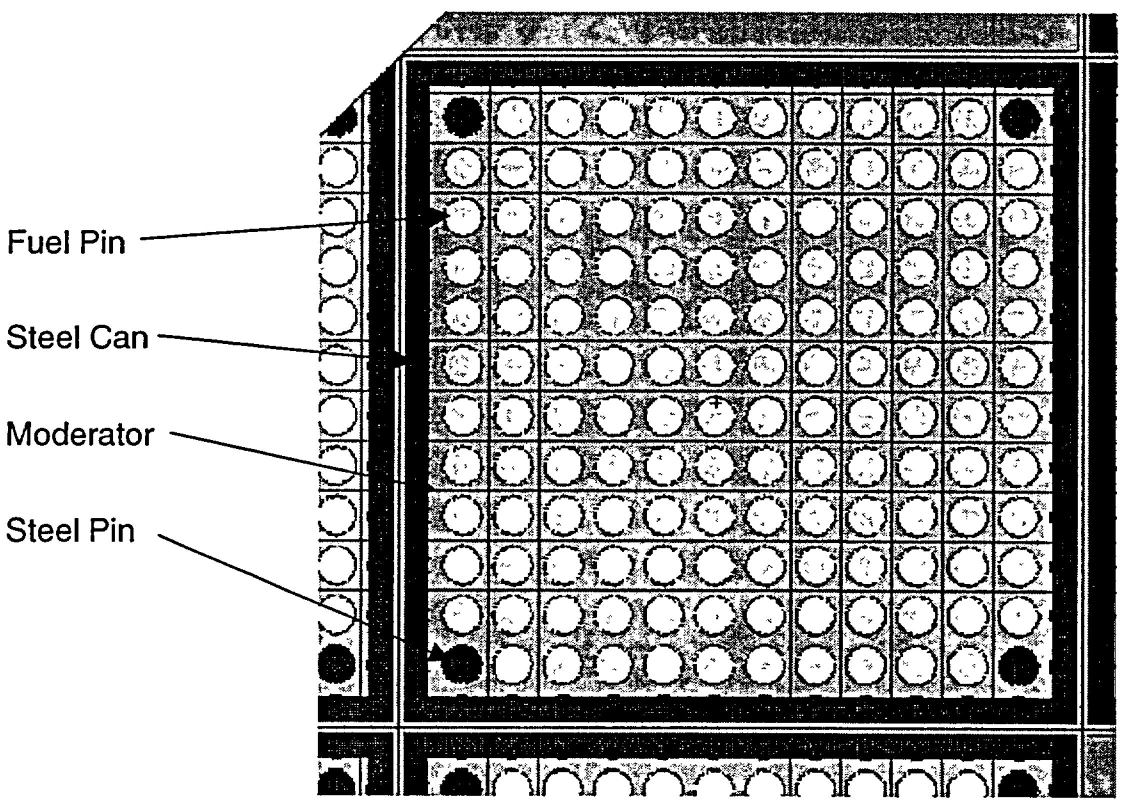

Figure 2. MCNP model cross-sectional view of a FERMI driver fuel assembly.

FERMI Subassembly Fuel

Zircaloy Cladding, 10 to 40% Enriched U-235 Fuel

Reactor Moderator/Coolant:	Liquid metal sodium
Fuel Meat:	Uranium/molybdenum metal alloy
Clad:	Zircaloy
Burnup:	3,125.2 MWD/MTU
Burnup:	58.67 MWD/subassembly (average burnup)
Burnup:	1.4% U-235 depletion (fissioned and transmuted)
Basis of Calculation:	Single subassembly
Subassemblies	101 subassemblies in core
BOL U-235:	4,806.144 g U-235 per subassembly
BOL U-238:	13,967.856 g U-238 per subassembly
BOL Total U per element:	18,774.000 g U per subassembly
BOL Fuel Enrichment:	25.6 wt% U-235

DECAY TIMES (years out of core)

(Activities* in Ci/subassembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H 3	1.118E+00	8.444E-01	6.378E-01	4.817E-01	3.639E-01	2.076E-01	8.943E-02	3.853E-02	1.660E-02	5.403E-03
BE 10	7.874E-09	7.873E-09	7.873E-09	7.873E-09	7.873E-09	7.873E-09	7.873E-09	7.873E-09	7.873E-09	7.873E-09
C 14	1.279E-03	1.278E-03	1.277E-03	1.276E-03	1.276E-03	1.274E-03	1.272E-03	1.269E-03	1.267E-03	1.264E-03
CL 36	3.238E-06	3.238E-06	3.238E-06	3.238E-06	3.238E-06	3.238E-06	3.238E-06	3.238E-06	3.238E-06	3.237E-06
CR 51	1.282E-18	1.849E-38	2.668E-58	3.849E-78	5.554E-98	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54	1.259E+00	2.191E-02	3.815E-04	6.642E-06	1.156E-07	3.505E-11	1.849E-16	9.757E-22	5.148E-27	4.729E-34
FE 55	2.474E+00	6.525E-01	1.721E-01	4.537E-02	1.196E-02	8.319E-04	1.525E-05	2.797E-07	5.128E-09	2.479E-11
FE 59	1.249E-12	7.580E-25	4.601E-37	2.793E-49	1.695E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60	6.360E+00	3.295E+00	1.707E+00	8.842E-01	4.581E-01	1.229E-01	1.709E-02	2.377E-03	3.304E-04	2.380E-05
NI 59	3.985E-03	3.985E-03	3.985E-03	3.984E-03	3.984E-03	3.984E-03	3.983E-03	3.983E-03	3.982E-03	3.982E-03
NI 63	9.690E-02	9.332E-02	8.987E-02	8.655E-02	8.335E-02	7.730E-02	6.904E-02	6.166E-02	5.507E-02	4.737E-02
ZN 65	2.294E-03	1.277E-05	7.111E-08	3.959E-10	2.204E-12	6.832E-17	1.179E-23	2.035E-30	3.512E-37	3.374E-46
SE 79	9.677E-04	9.676E-04	9.676E-04	9.675E-04	9.675E-04	9.674E-04	9.672E-04	9.671E-04	9.669E-04	9.667E-04
KR 85	1.576E+01	1.140E+01	8.254E+00	5.974E+00	4.324E+00	2.265E+00	8.587E-01	3.256E-01	1.234E-01	3.387E-02
RB 87	4.841E-08	4.841E-08	4.841E-08	4.841E-08	4.841E-08	4.841E-08	4.841E-08	4.841E-08	4.841E-08	4.841E-08

DECAY TIMES (years out of core)
(Activities* in Ci/subassembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
SR 90	1.444E+02	1.282E+02	1.138E+02	1.010E+02	8.971E+01	7.071E+01	4.948E+01	3.462E+01	2.423E+01	1.505E+01
Y 90	1.444E+02	1.282E+02	1.138E+02	1.011E+02	8.973E+01	7.072E+01	4.949E+01	3.463E+01	2.423E+01	1.505E+01
Y 91	3.555E-06	1.427E-15	5.728E-25	2.299E-34	9.229E-44	1.487E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93	3.859E-03	3.859E-03	3.859E-03	3.859E-03	3.859E-03	3.859E-03	3.859E-03	3.859E-03	3.859E-03	3.859E-03
ZR 95	2.492E-05	6.371E-14	1.628E-22	4.163E-31	1.064E-39	6.951E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M	9.163E-04	1.535E-03	2.015E-03	2.386E-03	2.674E-03	3.070E-03	3.389E-03	3.537E-03	3.607E-03	3.645E-03
NB 94	4.869E-04	4.868E-04	4.867E-04	4.866E-04	4.865E-04	4.864E-04	4.861E-04	4.859E-04	4.856E-04	4.853E-04
NB 95	5.533E-05	1.415E-13	3.615E-22	9.241E-31	2.362E-39	1.543E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	1.849E-07	4.727E-16	1.208E-24	3.087E-33	7.893E-42	5.157E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93	2.133E-03	2.130E-03	2.128E-03	2.126E-03	2.124E-03	2.120E-03	2.114E-03	2.107E-03	2.101E-03	2.093E-03
TC 99	2.631E-02	2.631E-02	2.631E-02	2.631E-02	2.631E-02	2.630E-02	2.630E-02	2.630E-02	2.630E-02	2.630E-02
RU103	7.758E-11	7.849E-25	7.940E-39	8.033E-53	8.126E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	1.653E+01	5.310E-01	1.706E-02	5.479E-04	1.760E-05	1.816E-08	6.017E-13	1.994E-17	6.610E-22	7.036E-28
RH103M	6.994E-11	7.075E-25	7.158E-39	7.241E-53	7.326E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	1.653E+01	5.310E-01	1.706E-02	5.479E-04	1.760E-05	1.816E-08	6.017E-13	1.994E-17	6.610E-22	7.036E-28
PD107	7.454E-05	7.454E-05	7.454E-05	7.454E-05	7.454E-05	7.454E-05	7.454E-05	7.454E-05	7.454E-05	7.454E-05
AG110	8.574E-06	5.409E-08	3.413E-10	2.153E-12	1.358E-14	5.406E-19	1.357E-25	3.410E-32	8.561E-39	1.356E-47
AG110M	6.447E-04	4.067E-06	2.566E-08	1.619E-10	1.021E-12	4.065E-17	1.021E-23	2.563E-30	6.437E-37	1.020E-45
AG111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113M	9.607E-02	7.576E-02	5.974E-02	4.711E-02	3.715E-02	2.310E-02	1.133E-02	5.554E-03	2.723E-03	1.053E-03
CD113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD115M	9.489E-12	4.461E-24	2.097E-36	9.861E-49	4.636E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	3.257E-15	2.569E-26	2.025E-37	1.598E-48	1.260E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	3.402E-15	2.684E-26	2.117E-37	1.670E-48	1.317E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	6.669E-16	3.135E-28	1.474E-40	6.930E-53	3.258E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	1.230E-02	7.017E-05	4.004E-07	2.284E-09	1.303E-11	4.243E-16	7.881E-23	1.464E-29	2.719E-36	2.882E-45
SN121M	6.085E-04	5.678E-04	5.297E-04	4.942E-04	4.611E-04	4.014E-04	3.260E-04	2.648E-04	2.150E-04	1.629E-04
SN123	1.442E-03	7.994E-08	4.432E-12	2.457E-16	1.362E-20	4.186E-29	7.134E-42	1.216E-54	2.071E-67	1.957E-84
SN125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN126	2.205E-03	2.204E-03	2.204E-03	2.204E-03	2.204E-03	2.204E-03	2.204E-03	2.204E-03	2.203E-03	2.203E-03

DECAY TIMES (years out of core)
(Activities* in Ci/subassembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
SB125	1.421E+01	4.065E+00	1.163E+00	3.329E-01	9.524E-02	7.799E-03	1.827E-04	4.282E-06	1.003E-07	6.726E-10
SB126	3.086E-04	3.086E-04	3.086E-04	3.086E-04	3.086E-04	3.086E-04	3.085E-04	3.085E-04	3.085E-04	3.084E-04
SB126M	2.205E-03	2.204E-03	2.204E-03	2.204E-03	2.204E-03	2.204E-03	2.204E-03	2.204E-03	2.203E-03	2.203E-03
TE123M	4.825E-09	1.230E-13	3.134E-18	7.987E-23	2.036E-27	1.322E-36	2.189E-50	3.623E-64	5.997E-78	2.530E-96
TE125M	3.466E+00	9.917E-01	2.838E-01	8.121E-02	2.324E-02	1.902E-03	4.459E-05	1.044E-06	2.448E-08	1.641E-10
TE127	6.284E-04	5.686E-09	5.145E-14	4.656E-19	4.213E-24	3.449E-34	2.556E-49	1.893E-64	1.403E-79	0.000E+00
TE127M	6.416E-04	5.805E-09	5.253E-14	4.753E-19	4.301E-24	3.522E-34	2.609E-49	1.933E-64	1.432E-79	0.000E+00
TE129	1.103E-14	4.792E-31	2.082E-47	9.047E-64	3.931E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	1.694E-14	7.362E-31	3.199E-47	1.390E-63	6.039E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	6.704E-05	6.704E-05	6.704E-05	6.704E-05	6.704E-05	6.704E-05	6.704E-05	6.704E-05	6.704E-05	6.704E-05
I131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS134	8.785E-01	1.636E-01	3.047E-02	5.673E-03	1.056E-03	3.664E-05	2.366E-07	1.528E-09	9.869E-12	1.187E-14
CS135	2.640E-03	2.640E-03	2.640E-03	2.640E-03	2.640E-03	2.640E-03	2.640E-03	2.640E-03	2.640E-03	2.640E-03
CS136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS137	1.616E+02	1.440E+02	1.283E+02	1.143E+02	1.018E+02	8.082E+01	5.714E+01	4.041E+01	2.857E+01	1.800E+01
BA136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA137M	1.529E+02	1.362E+02	1.214E+02	1.081E+02	9.632E+01	7.645E+01	5.406E+01	3.822E+01	2.703E+01	1.703E+01
BA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	1.809E-13	2.228E-30	2.744E-47	3.379E-64	4.163E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	5.505E-08	5.505E-08	5.505E-08	5.505E-08	5.505E-08	5.505E-08	5.505E-08	5.505E-08	5.505E-08	5.505E-08
CE144	4.304E+01	5.010E-01	5.833E-03	6.790E-05	7.904E-07	1.071E-10	1.690E-16	2.666E-22	4.206E-28	7.726E-36
PR143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144	4.304E+01	5.010E-01	5.833E-03	6.790E-05	7.905E-07	1.071E-10	1.690E-16	2.666E-22	4.207E-28	7.726E-36
PR144M	5.165E-01	6.012E-03	6.999E-05	8.148E-07	9.485E-09	1.285E-12	2.028E-18	3.200E-24	5.048E-30	9.271E-38
ND144	2.239E-12	2.255E-12	2.255E-12	2.255E-12	2.255E-12	2.255E-12	2.255E-12	2.255E-12	2.255E-12	2.255E-12
ND147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM145	1.609E-07	1.349E-07	1.110E-07	9.124E-08	7.502E-08	5.071E-08	2.818E-08	1.566E-08	8.706E-09	3.978E-09

DECAY TIMES (years out of core)
(Activities* in Ci/subassembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
PM148M	1.826E-12	8.893E-26	4.331E-39	2.109E-52	1.027E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	1.029E-13	5.009E-27	2.439E-40	1.188E-53	5.785E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	5.828E-08	1.409E-09	3.405E-11	8.230E-13	1.989E-14	1.162E-17	1.641E-22	2.317E-27	3.272E-32	1.117E-38
SM147	1.564E-08	1.902E-08	1.992E-08	2.016E-08	2.023E-08	2.025E-08	2.025E-08	2.025E-08	2.025E-08	2.025E-08
SM151	4.711E+00	4.533E+00	4.362E+00	4.197E+00	4.038E+00	3.739E+00	3.331E+00	2.968E+00	2.644E+00	2.266E+00
EU152	1.429E-03	1.108E-03	8.587E-04	6.656E-04	5.159E-04	3.099E-04	1.443E-04	6.717E-05	3.127E-05	1.129E-05
EU154	1.220E-01	8.151E-02	5.447E-02	3.641E-02	2.433E-02	1.086E-02	3.244E-03	9.684E-04	2.891E-04	5.767E-05
EU155	5.504E+00	2.736E+00	1.360E+00	6.763E-01	3.363E-01	8.310E-02	1.021E-02	1.255E-03	1.542E-04	9.419E-06
EU156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD153	4.863E-06	2.602E-08	1.392E-10	7.448E-13	3.986E-15	1.141E-19	1.748E-26	2.677E-33	4.102E-40	3.362E-49
TB160	4.540E-09	1.131E-16	2.819E-24	7.025E-32	1.751E-39	1.087E-54	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	3.190E-19	3.190E-19	3.190E-19	3.190E-19	3.190E-19	3.190E-19	3.190E-19	3.190E-19	3.190E-19	3.190E-19
TL207	1.264E-07	3.892E-07	7.729E-07	1.260E-06	1.834E-06	3.195E-06	5.628E-06	8.370E-06	1.130E-05	1.538E-05
TL208	3.846E-07	4.231E-07	4.126E-07	3.948E-07	3.765E-07	3.418E-07	2.959E-07	2.562E-07	2.218E-07	1.832E-07
PB210	1.038E-12	6.477E-12	1.970E-11	4.358E-11	8.055E-11	2.019E-10	5.253E-10	1.044E-09	1.778E-09	3.108E-09
PB211	1.267E-07	3.903E-07	7.750E-07	1.263E-06	1.839E-06	3.204E-06	5.643E-06	8.393E-06	1.133E-05	1.543E-05
PB212	1.070E-06	1.178E-06	1.148E-06	1.099E-06	1.048E-06	9.513E-07	8.236E-07	7.130E-07	6.174E-07	5.100E-07
BI211	1.267E-07	3.903E-07	7.750E-07	1.263E-06	1.839E-06	3.204E-06	5.643E-06	8.393E-06	1.133E-05	1.543E-05
BI212	1.070E-06	1.178E-06	1.148E-06	1.099E-06	1.048E-06	9.513E-07	8.236E-07	7.130E-07	6.174E-07	5.100E-07
PO212	6.857E-07	7.545E-07	7.357E-07	7.039E-07	6.714E-07	6.095E-07	5.277E-07	4.568E-07	3.956E-07	3.267E-07
PO215	1.267E-07	3.903E-07	7.750E-07	1.263E-06	1.839E-06	3.204E-06	5.643E-06	8.393E-06	1.133E-05	1.543E-05
PO216	1.070E-06	1.178E-06	1.148E-06	1.099E-06	1.048E-06	9.513E-07	8.236E-07	7.130E-07	6.174E-07	5.100E-07
RN219	1.267E-07	3.903E-07	7.750E-07	1.263E-06	1.839E-06	3.204E-06	5.643E-06	8.393E-06	1.133E-05	1.543E-05
RN220	1.070E-06	1.178E-06	1.148E-06	1.099E-06	1.048E-06	9.513E-07	8.236E-07	7.130E-07	6.174E-07	5.100E-07
FR223	1.749E-09	5.382E-09	1.068E-08	1.741E-08	2.535E-08	4.417E-08	7.781E-08	1.157E-07	1.562E-07	2.127E-07
RA223	1.267E-07	3.903E-07	7.750E-07	1.263E-06	1.839E-06	3.204E-06	5.643E-06	8.393E-06	1.133E-05	1.543E-05
RA224	1.070E-06	1.178E-06	1.148E-06	1.099E-06	1.048E-06	9.513E-07	8.236E-07	7.130E-07	6.174E-07	5.100E-07
RA226	1.795E-11	6.332E-11	1.364E-10	2.372E-10	3.657E-10	7.054E-10	1.421E-09	2.384E-09	3.593E-09	5.584E-09
RA228	6.568E-10	9.525E-10	1.129E-09	1.234E-09	1.297E-09	1.357E-09	1.383E-09	1.389E-09	1.391E-09	1.392E-09
AC227	1.267E-07	3.900E-07	7.743E-07	1.262E-06	1.837E-06	3.201E-06	5.639E-06	8.385E-06	1.132E-05	1.541E-05

DECAY TIMES (years out of core)
(Activities* in Ci/subassembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TH228	1.070E-06	1.177E-06	1.147E-06	1.098E-06	1.047E-06	9.512E-07	8.235E-07	7.130E-07	6.174E-07	5.100E-07
TH229	2.933E-10	5.563E-10	8.211E-10	1.088E-09	1.356E-09	1.899E-09	2.727E-09	3.572E-09	4.434E-09	5.609E-09
TH230	1.455E-08	2.741E-08	4.028E-08	5.316E-08	6.605E-08	9.185E-08	1.306E-07	1.695E-07	2.084E-07	2.603E-07
TH231	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02
TH232	1.389E-09	1.389E-09	1.389E-09	1.389E-09	1.389E-09	1.390E-09	1.390E-09	1.391E-09	1.391E-09	1.392E-09
TH234	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03
PA231	1.359E-06	2.444E-06	3.529E-06	4.614E-06	5.698E-06	7.866E-06	1.112E-05	1.437E-05	1.762E-05	2.195E-05
PA233	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04
PA234M	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03
PA234	6.096E-06	6.096E-06	6.096E-06	6.096E-06	6.096E-06	6.096E-06	6.096E-06	6.096E-06	6.096E-06	6.096E-06
U232	1.234E-06	1.176E-06	1.121E-06	1.068E-06	1.018E-06	9.246E-07	8.003E-07	6.927E-07	5.996E-07	4.946E-07
U233	5.551E-07	5.593E-07	5.636E-07	5.678E-07	5.720E-07	5.805E-07	5.932E-07	6.059E-07	6.186E-07	6.355E-07
U234	2.856E-04	2.858E-04	2.861E-04	2.863E-04	2.865E-04	2.869E-04	2.875E-04	2.881E-04	2.886E-04	2.892E-04
U235	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02	1.025E-02
U236	7.411E-04	7.411E-04	7.412E-04	7.412E-04	7.412E-04	7.412E-04	7.412E-04	7.412E-04	7.412E-04	7.412E-04
U237	2.548E-10	2.003E-10	1.574E-10	1.238E-10	9.729E-11	6.012E-11	2.920E-11	1.418E-11	6.890E-12	2.631E-12
U238	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03	4.689E-03
NP237	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04	1.942E-04
PU236	3.377E-09	1.170E-09	5.155E-10	3.214E-10	2.638E-10	2.417E-10	2.396E-10	2.395E-10	2.395E-10	2.395E-10
PU237	2.519E-16	2.214E-28	1.946E-40	1.710E-52	1.503E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	1.275E-02	1.226E-02	1.179E-02	1.133E-02	1.089E-02	1.006E-02	8.939E-03	7.940E-03	7.053E-03	6.022E-03
PU239	1.143E+00	1.143E+00	1.143E+00	1.143E+00	1.142E+00	1.142E+00	1.142E+00	1.141E+00	1.141E+00	1.140E+00
PU240	3.998E-03	3.996E-03	3.993E-03	3.991E-03	3.989E-03	3.985E-03	3.979E-03	3.972E-03	3.966E-03	3.958E-03
PU241	1.039E-03	8.164E-04	6.418E-04	5.045E-04	3.966E-04	2.451E-04	1.190E-04	5.782E-05	2.809E-05	1.072E-05
PU242	2.567E-11	2.567E-11	2.567E-11	2.567E-11	2.567E-11	2.567E-11	2.567E-11	2.567E-11	2.567E-11	2.566E-11
PU244	4.073E-22	4.073E-22	4.073E-22	4.073E-22	4.073E-22	4.073E-22	4.073E-22	4.073E-22	4.073E-22	4.073E-22
AM241	9.985E-06	1.728E-05	2.293E-05	2.730E-05	3.067E-05	3.518E-05	3.849E-05	3.959E-05	3.962E-05	3.894E-05
AM242M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AM242	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AM243	4.929E-13	4.927E-13	4.924E-13	4.922E-13	4.920E-13	4.915E-13	4.908E-13	4.901E-13	4.894E-13	4.885E-13

DECAY TIMES (years out of core)
(Activities* in Ci/subassembly)

Radionuclide	5	10	15	20	25	35	50	65	80	100
CM243	1.886E-12	1.670E-12	1.479E-12	1.309E-12	1.159E-12	9.092E-13	6.313E-13	4.383E-13	3.043E-13	1.871E-13
CM244	9.687E-14	8.000E-14	6.606E-14	5.456E-14	4.505E-14	3.073E-14	1.730E-14	9.746E-15	5.489E-15	2.553E-15
CM245	1.252E-19	1.251E-19	1.251E-19	1.250E-19	1.250E-19	1.249E-19	1.247E-19	1.246E-19	1.244E-19	1.242E-19
CM246	4.105E-23	4.102E-23	4.099E-23	4.096E-23	4.093E-23	4.087E-23	4.078E-23	4.069E-23	4.060E-23	4.048E-23
CM247	1.985E-30	1.985E-30	1.985E-30	1.985E-30	1.985E-30	1.985E-30	1.985E-30	1.985E-30	1.985E-30	1.985E-30
SUBTOTAL**	9.686E+02	6.191E+02	5.102E+02	4.422E+02	3.896E+02	3.066E+02	2.158E+02	1.525E+02	1.081E+02	6.872E+01
TOTAL***	9.686E+02	6.192E+02	5.102E+02	4.422E+02	3.896E+02	3.066E+02	2.158E+02	1.525E+02	1.081E+02	6.872E+01

* Four decimal places of accuracy are as reported by ORIGEN2 output and are not significant for many radionuclides.
** Subtotal: total activity of the 145 isotopes listed in the table.
*** Total: total activity of the ORIGEN2 output isotopes.

Template 6

Fuel-Specific Source Term Calculations
Fort Saint Vrain Graphite Fuel

Introduction

The following data have been used in the Idaho National Engineering and Environmental Laboratory (INEEL) spent nuclear fuel source term calculational methodology to generate a source term template for a single Fort Saint Vrain (FSV) spent nuclear fuel element. This single-element source term is intended to bound all 2,208 irradiated FSV high-enriched, uranium-thorium-graphite spent nuclear fuel elements. The data sources for the analysis are documented in References 1 through 6, and the INEEL calculational methodology is described in detail in Reference 7.

Fort Saint Vrain Reactor Data

The FSV reactor core and fuel elements are described in some detail in References 1 through 5. Data from these references have been used to develop reactor physics models needed to support the depletion/activation analysis.

The FSV fuel element is a hexagonal graphite block with 210 axial fuel rods and 108 helium gas channels. In addition, there is a centrally located fuel handling pickup hole and six peripheral burnable poison rods. The cylindrical fuel rods are composed of spherical fuel particles bound together in a graphite binder matrix. There are two types of spherical fuel particles, namely a fissile particle $(Th,U)C_2$ and a fertile particle ThC_2. The uranium enrichment in the fissile particle is 93.13 wt% U-235. The fertile particle contains 100% natural thorium (Th-232).

The following data provide specific fuel element dimensions, materials, densities, enrichment, etc., for a typical FSV fuel element. However, in order to achieve a bounding burnup for all FSV spent nuclear fuel elements, the uranium and thorium element loadings are based on the heaviest heavy-metal-loaded FSV elements that are installed and irradiated in the FSV core. Specifically, the heavy metal loadings are based on the element with ID number 1-5718. The beginning-of-life (BOL) data below were used in the burnup calculation for the FSV fuel element source term generation.

Fuel Element:	Hexagonal graphite block
Flat-to-flat :	14.17 in.
Length:	31.22 in.
Bulk graphite density:	1.74 g/cc
Material:	H-451 graphite
Graphite Mass:	154,794 g/element
Fuel Rod:	$(Th,U)C_2$ and ThC_2 spherical particles in a graphite binder matrix
Uranium Enrichment:	0.6389 wt% U-234
	93.133 wt% U-235
	0.2711 wt% U-236
	5.957 wt% U-238

Heavy Metal Loading:	7.978	g/element U-234 (BOL)
	1,163.000	g/element U-235 (BOL)
	3.385	g/element U-236 (BOL)
	74.388	g/element U-238 (BOL)
	1,248.752	g/element Total U
	11,454.000	g/element Th-232 (BOL)
	1.2702752E-2	Total MTIHM/element (BOL)

Coolant :	Helium gas
Coolant Temperature:	1535°F
Coolant Pressure:	700 psig
Coolant Density:	0.0021 g/cc

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single FSV fuel element. In addition, for the ORIGEN2 (Reference 8) depletion calculation, conservative and detailed impurity concentrations were added for H-451 graphite. For conservatism, a graphite mass (154.8 kg) equal to the entire fuel element volume was input along with the corresponding impurity masses for maximum activation. Table 1 lists the impurities and concentrations for graphite H-451.

Burnup

The burnup chosen for this template is 100,000 MWd/MTIHM, 1,270.275 MWd, and approximately 1,019 g of U-235 depleted for a single FSV element. Because the BOL uranium loading was 1,163 g of U-235, this burnup represents an 88% depletion of the BOL uranium. This relatively high burnup is needed to ensure the entire FSV element inventory is bounded.

Based on Reference 1 data, the entire FSV element inventory has ≤ 88% U-235 depletion with the exception of one element that has a 97% depletion. The vast majority of the FSV elements are between 40 and 70% U-235 depletion. Perhaps more importantly are the total grams of U-235 depleted. The heavily loaded template element here depletes 1,019 g of U-235. The highest FSV element depletion is approximately 800 g of U-235 (Reference 1). From this perspective, the template radionuclide inventory would definitely be bounding with approximately 20% higher levels of concentrations for fission products, activation products, and actinides other than U-235, U-238, and Th-232.

For the template analysis here, the burnup period in the analysis is assumed to start February 1, 1979, (start of Cycle 2) and end August 18, 1989, (FSV shutdown). The corresponding fuel element output power for the 100,000 MWd/MTIHM is approximately 330 kW and is assumed to be continuous over the burnup period (3,851 days) with no refueling shutdowns (see Table 2). The relatively high burnup (100,000 MWd/MTIHM) is conservative for the buildup of fission products, activation products, and minor actinides in the source term and nonconservative with regard to criticality safety, i.e., fissile concentrations of U-235.

Cross-Section Development

The neutron cross sections used in the burnup or depletion calculation for the source term generation of a single FSV fuel element are based on the methodology described in Reference 7. Cross sections from a standard ORIGEN2 high-temperature gas-cooled graphite reactor library were updated five times over the burnup period to ensure accurate FSV production and activity levels for actinides, fission products, and activation products. The first update developed cross sections for BOL conditions followed by four subsequent updates every 730 days of fuel element exposure. These cross-section updates take into account changes in the neutron flux spectrum and spatial profiles as a function of burnup and are essentially element-average cross sections. An explicit FSV fuel block (with reflective boundary conditions on the element peripheral surfaces) was used to determine volume-averaged flux and reaction rate profiles for the cross-section development (see Figure 1).

Fort Saint Vrain Single Element Exposure History

Table 2 summarizes the power or exposure history used in the burnup or source term calculations for a single FSV fuel element. Following the burnup or exposure period, the radionuclide activities are decayed for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100 years.

Burnup Calculation

The ORIGEN2 computer code (Reference 8) was used to perform the depletion or burnup calculation for the FSV fuel element. The radionuclide inventory or source term template is for a single FSV fuel element or block. The fuel element masses and impurities (graphite, uranium, and thorium), neutron cross sections, burnup, power history, and power level discussed above are input data for the ORIGEN2 calculation. The radionuclide concentrations are given as a function of time in the template table.

The 145 radionuclides listed in the template represent greater than 99.99% of the total curie inventory had all 684 activation products, 880 fission products, and 127 actinide/daughter isotopes from the ORIGEN2 output been included in the template.

References

1. Data transfer (via diskette) from Public Service of Colorado (PSC) to the Westinghouse Idaho Nuclear Company, FFA-1994-0001, "Initial and Present Nuclide Content for Segments 1-10 for INEL/WEST." Spreadsheet database of the 2,208 irradiated FSV fuel elements with BOL and EOL heavy metal masses by element for FSV segments 1–10. Responsible engineers: W.A. Grover and S.M. Goebel, April 12, 1994.

2. R. P. Morissette and N. Tomsio, "Characterization of Fort St. Vrain Fuel," ORNL/Sub/86-22047/1, GA-C18511, October 1986.

3. DOE, *Characteristics of Potential Repository Wastes*, DOE/RW-0184-R1, Volume 1, July 1982.

4. G. E Bingham, *Final Safety Analysis Report for the Irradiated Fuels Storage Facility*, ICP-1052, Allied Chemical Corporation, February 1974.

5. J. J. Saurwein, C. M. Miller, and C. A. Young, *Postirradiation Examination and Evaluation of Fort St. Vrain Fuel Element 1-0743*, GA-A16258, May 1981.

6. "Fort Saint Vrain Safety Analysis Report," Revision 8, Section 3.4, 1990.

7. J. W. Sterbentz and C. A. Wemple, *Calculational Burnup Methodology and Validation for the Idaho National Engineering Laboratory Spent Nuclear Fuels*, INEL-96/0304, September 1996.

8. A. G. Croff, *ORIGEN2—A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, July 1980.

Table 1. H-451 graphite constituent and impurity concentrations for the Fort Saint Vrain fuel element.

Constituent or Impurity	Graphite Concentration (ppm)
H	
Li	0.45
Be	0.005
B	2.5
C	100 wt%
N	
O	
Na	10.4
Mg	1
Al	4.1
Si	26
P	1
S	9.4
Cl	3
K	3
Ca	22.5
Sc	0.01
Ti	16
V	18.9
Cr	1
Mn	1
Fe	11.1
Co	4
Ni	4.6
Cu	0.47
Zn	1
Ga	
As	
Se	
Br	
Rb	1
Sr	0.47
Y	
Zr	0.5
Nb	1.74
Mo	1
Ag	0.5
Cd	0.5
In	1
Sn	1
Sb	1

Table 1. (continued).

Constituent or Impurity	Graphite Concentration (ppm)
Cs	1
Ba	2.9
La	1.38
Ce	0.56
Pr	0.64
Nd	0.36
Sm	0.61
Eu	
Gd	0.08
Tb	0.26
Dy	0.16
Ho	0.08
Er	0.04
Tm	0.04
Yb	0.06
Lu	0.02
Hf	0.17
Ta	0.35
W	25.5
Tl	1
Pb	6.9
Bi	1
Th	
U	

Table 2. Burnup or power history for a 100,000 MWd/MTIHM burnup FSV fuel element.

Duration (days)	Cumulative Duration (days)	Time-Averaged Power (MWth)
365	365	0.3299
366	731	0.3299
365	1096	0.3299
365	1461	0.3299
365	1826	0.3299
366	2192	0.3299
365	2557	0.3299
365	2922	0.3299
365	3287	0.3299
564	3851	0.3299
1825	5676	0.0
1825	7501	0.0
1825	9326	0.0
1825	11151	0.0
1825	12976	0.0
3650	16626	0.0
5475	22101	0.0
5475	27576	0.0
5475	33051	0.0
7300	40351	0.0

The ten dates with zero associated power represent the ten different cooling or decay dates after exposure. These ten dates are specifically the 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100-year cooling times designated for the template methodology.

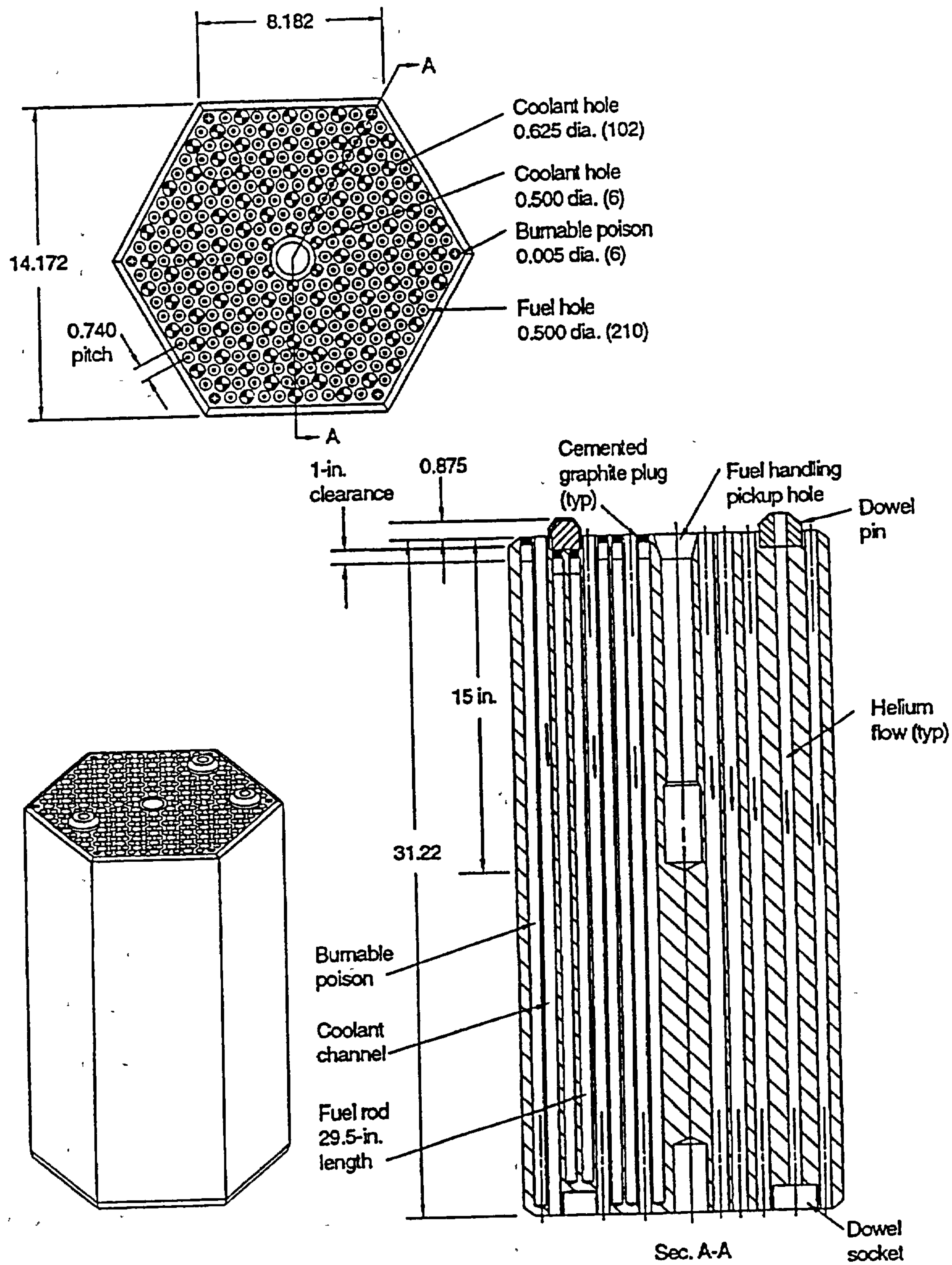

Figure 1. Standard FSV fuel element.

Fort Saint Vrain Reactor Element

Graphite Cladding, 60 to 100% Enriched U-235/Th-232 Fuel

Reactor Moderator	Graphite
Reactor Coolant:	Helium Gas
Fuel Meat:	$(Th,U)C_2$
Clad:	Graphite
Burnup:	100,000 MWd/MTIHM
Burnup:	1,270.28 MWd/element (high burnup)
Burnup:	88% U-235 depletion (fissioned and transmuted)
Basis of Calculation:	Single fuel element
BOL U-235:	1,163.000 grams U-235 per element
BOL U-238:	74.388 grams U-238 per element
BOL U-234:	7.978 grams U-234 per element
BOL U-236:	3.385 grams U-236 per element
BOL Total U:	1,248.752 grams U per element
BOL Th-232:	11,454.000 grams Th-232 per element
BOL Total Heavy Metal:	12,702.752 grams Th + U per element
BOL U Enrichment:	93.13 wt% U-235

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H 3	2.675E+01	2.020E+01	1.526E+01	1.153E+01	8.705E+00	4.966E+00	2.139E+00	9.219E-01	3.972E-01	1.293E-01
BE 10	8.043E-04	8.043E-04	8.043E-04	8.043E-04	8.043E-04	8.043E-04	8.043E-04	8.043E-04	8.043E-04	8.043E-04
C 14	2.948E-02	2.946E-02	2.945E-02	2.943E-02	2.941E-02	2.937E-02	2.932E-02	2.927E-02	2.921E-02	2.914E-02
CL 36	1.355E-03	1.355E-03	1.355E-03	1.355E-03	1.355E-03	1.355E-03	1.355E-03	1.355E-03	1.355E-03	1.355E-03
CR 51	5.382E-21	7.813E-41	1.120E-60	1.616E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54	1.207E-03	2.103E-05	3.658E-07	6.369E-09	1.109E-10	3.361E-14	1.773E-19	9.356E-25	4.937E-30	4.535E-37
FE 55	1.689E-01	4.455E-02	1.175E-02	3.097E-03	8.168E-04	5.679E-05	1.041E-06	1.909E-08	3.501E-10	1.693E-12
FE 59	1.542E-14	9.395E-27	5.681E-39	3.448E-51	2.093E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60	2.958E+01	1.532E+01	7.938E+00	4.113E+00	2.131E+00	5.718E-01	7.950E-02	1.105E-02	1.537E-03	1.107E-04
NI 59	4.721E-04	4.721E-04	4.721E-04	4.721E-04	4.721E-04	4.720E-04	4.720E-04	4.719E-04	4.718E-04	4.718E-04
NI 63	6.261E-02	6.029E-02	5.806E-02	5.592E-02	5.385E-02	4.994E-02	4.460E-02	3.984E-02	3.558E-02	3.060E-02
ZN 65	1.209E-03	6.738E-06	3.749E-08	2.087E-10	1.162E-12	3.601E-17	6.215E-24	1.073E-30	1.851E-37	1.779E-46
SE 79	2.678E-02	2.678E-02	2.678E-02	2.678E-02	2.678E-02	2.678E-02	2.677E-02	2.677E-02	2.676E-02	2.676E-02
KR 85	3.492E+02	2.528E+02	1.829E+02	1.324E+02	9.583E+01	5.020E+01	1.903E+01	7.216E+00	2.736E+00	7.506E-01
RB 87	1.328E-06	1.328E-06	1.328E-06	1.328E-06	1.328E-06	1.328E-06	1.328E-06	1.328E-06	1.328E-06	1.328E-06
SR 89	1.996E-07	2.599E-18	3.362E-29	4.363E-40	5.663E-51	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	3.285E+03	2.917E+03	2.589E+03	2.299E+03	2.041E+03	1.609E+03	1.126E+03	7.876E+02	5.512E+02	3.424E+02

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
Y 91	6.795E-06	2.736E-15	1.095E-24	4.395E-34	1.764E-43	2.842E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93	8.560E-02	8.560E-02	8.560E-02	8.560E-02	8.560E-02	8.560E-02	8.560E-02	8.560E-02	8.560E-02	8.560E-02
ZR 95	4.363E-05	1.118E-13	2.851E-22	7.287E-31	1.863E-39	1.217E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M	3.269E-02	4.363E-02	5.211E-02	5.868E-02	6.377E-02	7.078E-02	7.641E-02	7.903E-02	8.025E-02	8.093E-02
NB 94	6.255E-04	6.254E-04	6.253E-04	6.252E-04	6.251E-04	6.249E-04	6.246E-04	6.243E-04	6.239E-04	6.235E-04
NB 95	9.687E-05	2.483E-13	6.329E-22	1.618E-30	4.135E-39	2.702E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	3.237E-07	8.296E-16	2.115E-24	5.406E-33	1.382E-41	9.028E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93	1.318E-05	1.317E-05	1.315E-05	1.314E-05	1.313E-05	1.310E-05	1.306E-05	1.302E-05	1.298E-05	1.293E-05
TC 99	4.234E-01	4.234E-01	4.234E-01	4.234E-01	4.234E-01	4.234E-01	4.233E-01	4.233E-01	4.233E-01	4.233E-01
RU103	6.513E-11	6.618E-25	6.666E-39	6.744E-53	6.822E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	4.072E+01	1.309E+00	4.201E-02	1.350E-03	4.335E-05	4.472E-08	1.482E-12	4.912E-17	1.628E-21	1.733E-27
RH103M	5.872E-11	5.966E-25	6.009E-39	6.079E-53	6.150E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	4.072E+01	1.309E+00	4.201E-02	1.350E-03	4.335E-05	4.473E-08	1.482E-12	4.912E-17	1.628E-21	1.733E-27
PD107	6.831E-04	6.831E-04	6.831E-04	6.831E-04	6.831E-04	6.831E-04	6.831E-04	6.831E-04	6.831E-04	6.831E-04
AG110	1.411E-03	8.908E-06	5.616E-08	3.543E-10	2.235E-12	8.897E-17	2.234E-23	5.610E-30	1.409E-36	2.232E-45
AG110M	1.061E-01	6.698E-04	4.222E-06	2.664E-08	1.681E-10	6.689E-15	1.680E-21	4.218E-28	1.059E-34	1.678E-43
AG111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113M	4.052E-01	3.195E-01	2.520E-01	1.987E-01	1.567E-01	9.742E-02	4.777E-02	2.342E-02	1.149E-02	4.441E-03
CD113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD115M	2.404E-12	1.134E-24	5.313E-37	2.497E-49	1.174E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	3.480E-12	2.754E-23	2.164E-34	1.708E-45	1.346E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	3.635E-12	2.878E-23	2.261E-34	1.784E-45	1.407E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	1.679E-16	7.924E-29	3.711E-41	1.745E-53	8.202E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	6.468E-03	3.693E-05	2.106E-07	1.202E-09	6.854E-12	2.231E-16	4.144E-23	7.698E-30	1.430E-36	1.516E-45
SN121M	7.842E-03	7.317E-03	6.826E-03	6.369E-03	5.942E-03	5.173E-03	4.201E-03	3.412E-03	2.771E-03	2.100E-03
SN123	1.323E-03	7.344E-08	4.066E-12	2.254E-16	1.249E-20	3.841E-29	6.545E-42	1.115E-54	1.900E-67	1.795E-84
SN125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN126	2.820E-02	2.820E-02	2.820E-02	2.820E-02	2.820E-02	2.819E-02	2.819E-02	2.819E-02	2.819E-02	2.818E-02
SB124	2.230E-08	1.648E-17	1.211E-26	8.921E-36	6.573E-45	3.568E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	6.457E+01	1.848E+01	5.287E+00	1.513E+00	4.329E-01	3.545E-02	8.306E-04	1.946E-05	4.560E-07	3.058E-09
SB126	3.948E-03	3.948E-03	3.948E-03	3.948E-03	3.948E-03	3.947E-03	3.947E-03	3.946E-03	3.946E-03	3.945E-03
SB126M	2.820E-02	2.820E-02	2.820E-02	2.820E-02	2.820E-02	2.819E-02	2.819E-02	2.819E-02	2.819E-02	2.818E-02
TE123M	1.220E-05	3.113E-10	7.923E-15	2.019E-19	5.146E-24	3.342E-33	5.532E-47	9.158E-61	1.516E-74	6.395E-93
TE125M	1.575E+01	4.509E+00	1.290E+00	3.692E-01	1.056E-01	8.649E-03	2.026E-04	4.749E-06	1.112E-07	7.460E-10
TE127	1.906E-03	1.728E-08	1.561E-13	1.412E-18	1.278E-23	1.046E-33	7.753E-49	5.744E-64	4.255E-79	2.853E-99

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TE127M	1.946E-03	1.764E-08	1.594E-13	1.442E-18	1.305E-23	1.068E-33	7.915E-49	5.864E-64	4.344E-79	2.912E-99
TE129	1.897E-14	8.285E-31	3.581E-47	1.556E-63	6.761E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	2.914E-14	1.273E-30	5.502E-47	2.390E-63	1.039E-79	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	1.282E-03	1.282E-03	1.282E-03	1.282E-03	1.282E-03	1.282E-03	1.282E-03	1.282E-03	1.282E-03	1.282E-03
I131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS134	1.160E+03	2.160E+02	4.022E+01	7.490E+00	1.394E+00	4.838E-02	3.123E-04	2.018E-06	1.303E-08	1.567E-11
CS135	3.139E-02	3.139E-02	3.139E-02	3.139E-02	3.139E-02	3.139E-02	3.139E-02	3.139E-02	3.139E-02	3.139E-02
CS136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS137	3.373E+03	3.005E+03	2.677E+03	2.385E+03	2.125E+03	1.686E+03	1.192E+03	8.431E+02	5.962E+02	3.756E+02
BA136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA137M	3.191E+03	2.842E+03	2.532E+03	2.256E+03	2.010E+03	1.595E+03	1.128E+03	7.976E+02	5.640E+02	3.553E+02
BA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	2.145E-13	2.656E-30	3.253E-47	4.006E-64	4.934E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	1.178E-06	1.178E-06	1.178E-06	1.178E-06	1.178E-06	1.178E-06	1.178E-06	1.178E-06	1.178E-06	1.178E-06
CE144	1.566E+02	1.824E+00	2.123E-02	2.471E-04	2.877E-06	3.898E-10	6.150E-16	9.703E-22	1.531E-27	2.812E-35
PR143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144	1.566E+02	1.825E+00	2.123E-02	2.471E-04	2.877E-06	3.899E-10	6.151E-16	9.704E-22	1.531E-27	2.812E-35
PR144M	1.880E+00	2.189E-02	2.547E-04	2.965E-06	3.452E-08	4.678E-12	7.381E-18	1.164E-23	1.837E-29	3.374E-37
ND144	7.244E-11	7.249E-11	7.250E-11	7.250E-11	7.250E-11	7.250E-11	7.250E-11	7.250E-11	7.250E-11	7.250E-11
ND147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM145	5.580E-04	4.613E-04	3.793E-04	3.118E-04	2.564E-04	1.733E-04	9.633E-05	5.354E-05	2.975E-05	1.360E-05
PM147	5.328E+02	1.422E+02	3.794E+01	1.013E+01	2.702E+00	1.924E-01	3.656E-03	6.948E-05	1.320E-06	6.695E-09
PM148M	1.482E-11	7.245E-25	3.514E-38	1.711E-51	8.333E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	8.345E-13	4.081E-26	1.979E-39	9.638E-53	4.693E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	5.473E-05	1.323E-06	3.197E-08	7.728E-10	1.868E-11	1.091E-14	1.541E-19	2.176E-24	3.073E-29	1.049E-35
SM147	1.179E-07	1.275E-07	1.300E-07	1.307E-07	1.309E-07	1.310E-07	1.310E-07	1.310E-07	1.310E-07	1.310E-07
SM151	1.041E+01	1.002E+01	9.636E+00	9.271E+00	8.921E+00	8.260E+00	7.359E+00	6.555E+00	5.841E+00	5.006E+00
EU152	5.129E-01	3.976E-01	3.082E-01	2.388E-01	1.851E-01	1.111E-01	5.176E-02	2.410E-02	1.122E-02	4.049E-03
EU154	2.239E+02	1.497E+02	1.000E+02	6.684E+01	4.467E+01	1.995E+01	5.956E+00	1.778E+00	5.307E-01	1.059E-01
EU155	8.762E+01	4.356E+01	2.166E+01	1.077E+01	5.354E+00	1.323E+00	1.625E-01	1.997E-02	2.455E-03	1.500E-04
EU156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD153	5.970E-03	3.197E-05	1.709E-07	9.144E-10	4.893E-12	1.401E-16	2.145E-23	3.287E-30	5.036E-37	4.128E-46

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TB160	2.368E-07	5.915E-15	1.471E-22	3.665E-30	9.130E-38	5.669E-53	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	8.711E-09	8.711E-09	8.711E-09	8.711E-09	8.711E-09	8.711E-09	8.711E-09	8.711E-09	8.711E-09	8.711E-09
TL207	1.459E-03	2.130E-03	2.702E-03	3.189E-03	3.605E-03	4.260E-03	4.923E-03	5.335E-03	5.589E-03	5.783E-03
TL208	9.283E-01	9.492E-01	9.152E-01	8.739E-01	8.331E-01	7.562E-01	6.545E-01	5.665E-01	4.904E-01	4.049E-01
PB210	9.730E-07	8.953E-07	8.635E-07	8.757E-07	9.304E-07	1.162E-06	1.803E-06	2.782E-06	4.089E-06	6.338E-06
PB211	1.463E-03	2.136E-03	2.709E-03	3.198E-03	3.615E-03	4.272E-03	4.937E-03	5.350E-03	5.605E-03	5.799E-03
PB212	2.584E+00	2.642E+00	2.547E+00	2.432E+00	2.319E+00	2.105E+00	1.822E+00	1.577E+00	1.365E+00	1.127E+00
BI211	1.463E-03	2.136E-03	2.709E-03	3.198E-03	3.615E-03	4.272E-03	4.937E-03	5.350E-03	5.605E-03	5.799E-03
BI212	2.584E+00	2.642E+00	2.547E+00	2.432E+00	2.319E+00	2.105E+00	1.822E+00	1.577E+00	1.365E+00	1.127E+00
PO212	1.655E+00	1.693E+00	1.632E+00	1.558E+00	1.486E+00	1.348E+00	1.167E+00	1.010E+00	8.745E-01	7.221E-01
PO215	1.463E-03	2.136E-03	2.709E-03	3.198E-03	3.615E-03	4.272E-03	4.937E-03	5.350E-03	5.605E-03	5.799E-03
PO216	2.584E+00	2.642E+00	2.547E+00	2.432E+00	2.319E+00	2.105E+00	1.822E+00	1.577E+00	1.365E+00	1.127E+00
RN219	1.463E-03	2.136E-03	2.709E-03	3.198E-03	3.615E-03	4.272E-03	4.937E-03	5.350E-03	5.605E-03	5.799E-03
RN220	2.584E+00	2.642E+00	2.547E+00	2.432E+00	2.319E+00	2.105E+00	1.822E+00	1.577E+00	1.365E+00	1.127E+00
FR223	2.017E-05	2.943E-05	3.733E-05	4.406E-05	4.980E-05	5.887E-05	6.805E-05	7.373E-05	7.724E-05	7.994E-05
RA223	1.463E-03	2.136E-03	2.709E-03	3.198E-03	3.615E-03	4.272E-03	4.937E-03	5.350E-03	5.605E-03	5.799E-03
RA224	2.584E+00	2.642E+00	2.547E+00	2.432E+00	2.319E+00	2.105E+00	1.822E+00	1.577E+00	1.365E+00	1.127E+00
RA226	3.214E-07	5.458E-07	8.031E-07	1.094E-06	1.418E-06	2.167E-06	3.547E-06	5.239E-06	7.249E-06	1.043E-05
RA228	9.265E-04	1.024E-03	1.082E-03	1.117E-03	1.138E-03	1.157E-03	1.166E-03	1.168E-03	1.168E-03	1.168E-03
AC227	1.462E-03	2.133E-03	2.705E-03	3.193E-03	3.609E-03	4.266E-03	4.931E-03	5.343E-03	5.597E-03	5.792E-03
TH227	1.443E-03	2.106E-03	2.672E-03	3.154E-03	3.565E-03	4.213E-03	4.869E-03	5.276E-03	5.528E-03	5.719E-03
TH228	2.582E+00	2.640E+00	2.545E+00	2.430E+00	2.317E+00	2.104E+00	1.822E+00	1.577E+00	1.365E+00	1.127E+00
TH229	2.390E-03	3.624E-03	4.859E-03	6.092E-03	7.325E-03	9.789E-03	1.348E-02	1.717E-02	2.085E-02	2.575E-02
TH230	9.635E-05	1.117E-04	1.272E-04	1.429E-04	1.587E-04	1.908E-04	2.398E-04	2.899E-04	3.410E-04	4.103E-04
TH231	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04
TH232	1.168E-03	1.168E-03	1.168E-03	1.168E-03	1.168E-03	1.168E-03	1.168E-03	1.168E-03	1.168E-03	1.168E-03
TH234	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05
PA231	6.022E-03	6.021E-03	6.021E-03	6.020E-03	6.019E-03	6.018E-03	6.016E-03	6.015E-03	6.013E-03	6.010E-03
PA233	1.589E-02	1.589E-02	1.589E-02	1.590E-02	1.590E-02	1.591E-02	1.593E-02	1.595E-02	1.597E-02	1.600E-02
PA234M	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05
PA234	2.324E-08	2.324E-08	2.324E-08	2.324E-08	2.324E-08	2.324E-08	2.324E-08	2.324E-08	2.324E-08	2.324E-08
U232	2.733E+00	2.604E+00	2.482E+00	2.365E+00	2.254E+00	2.047E+00	1.772E+00	1.534E+00	1.327E+00	1.095E+00
U233	2.618E+00	2.618E+00	2.618E+00	2.617E+00	2.617E+00	2.617E+00	2.617E+00	2.617E+00	2.617E+00	2.617E+00
U234	3.395E-01	3.432E-01	3.467E-01	3.500E-01	3.532E-01	3.593E-01	3.676E-01	3.749E-01	3.814E-01	3.890E-01
U235	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04	3.102E-04

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
U236	1.096E-02	1.096E-02	1.096E-02	1.096E-02	1.096E-02	1.096E-02	1.096E-02	1.096E-02	1.096E-02	1.096E-02
U237	2.554E-05	2.008E-05	1.578E-05	1.241E-05	9.753E-06	6.027E-06	2.928E-06	1.423E-06	6.919E-07	2.649E-07
U238	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05	1.788E-05
NP237	1.589E-02	1.589E-02	1.589E-02	1.590E-02	1.590E-02	1.591E-02	1.593E-02	1.595E-02	1.597E-02	1.600E-02
PU236	5.247E-04	1.556E-04	4.614E-05	1.368E-05	4.059E-06	3.595E-07	1.215E-08	3.095E-09	2.859E-09	2.852E-09
PU237	1.727E-14	1.524E-26	1.334E-38	1.173E-50	1.031E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	2.616E+02	2.515E+02	2.417E+02	2.324E+02	2.234E+02	2.064E+02	1.833E+02	1.628E+02	1.446E+02	1.235E+02
PU239	1.725E-01	1.725E-01	1.725E-01	1.725E-01	1.725E-01	1.725E-01	1.724E-01	1.724E-01	1.723E-01	1.723E-01
PU240	2.876E-01	2.999E-01	3.100E-01	3.183E-01	3.252E-01	3.354E-01	3.447E-01	3.498E-01	3.524E-01	3.538E-01
PU241	1.041E+02	8.184E+01	6.433E+01	5.057E+01	3.976E+01	2.457E+01	1.194E+01	5.801E+00	2.820E+00	1.080E+00
PU242	4.937E-03	4.937E-03	4.937E-03	4.937E-03	4.937E-03	4.937E-03	4.937E-03	4.937E-03	4.936E-03	4.936E-03
PU244	3.070E-09	3.070E-09	3.070E-09	3.070E-09	3.070E-09	3.070E-09	3.070E-09	3.071E-09	3.071E-09	3.071E-09
AM241	1.139E+00	1.869E+00	2.435E+00	2.872E+00	3.208E+00	3.659E+00	3.987E+00	4.094E+00	4.095E+00	4.023E+00
AM242M	3.491E-03	3.412E-03	3.335E-03	3.260E-03	3.187E-03	3.045E-03	2.843E-03	2.655E-03	2.480E-03	2.264E-03
AM242	3.473E-03	3.395E-03	3.319E-03	3.244E-03	3.171E-03	3.029E-03	2.829E-03	2.642E-03	2.467E-03	2.252E-03
AM243	5.869E-02	5.866E-02	5.863E-02	5.860E-02	5.858E-02	5.852E-02	5.844E-02	5.836E-02	5.828E-02	5.817E-02
CM242	4.576E-02	2.827E-03	2.746E-03	2.684E-03	2.623E-03	2.505E-03	2.340E-03	2.185E-03	2.041E-03	1.863E-03
CM243	6.681E-02	5.916E-02	5.239E-02	4.639E-02	4.108E-02	3.221E-02	2.236E-02	1.553E-02	1.078E-02	6.628E-03
CM244	2.581E+01	2.132E+01	1.760E+01	1.454E+01	1.201E+01	8.188E+00	4.612E+00	2.597E+00	1.463E+00	6.803E-01
CM245	5.110E-03	5.108E-03	5.106E-03	5.104E-03	5.102E-03	5.098E-03	5.091E-03	5.085E-03	5.079E-03	5.071E-03
CM246	2.016E-03	2.015E-03	2.014E-03	2.012E-03	2.011E-03	2.008E-03	2.003E-03	1.999E-03	1.994E-03	1.989E-03
CM247	1.325E-08	1.325E-08	1.325E-08	1.325E-08	1.325E-08	1.325E-08	1.325E-08	1.325E-08	1.325E-08	1.325E-08
Subtotal**	1.645E+04	1.294E+04	1.116E+04	9.818E+03	8.689E+03	6.849E+03	4.830E+03	3.425E+03	2.441E+03	1.565E+03
TOTAL***	1.645E+04	1.294E+04	1.116E+04	9.818E+03	8.689E+03	6.849E+03	4.830E+03	3.425E+03	2.441E+03	1.565E+03

* Four decimal places of accuracy are as reported by ORIGEN2 output and are not significant for many radionuclides.
** Subtotal: total activity of the 145 isotopes listed in the table.
*** Total: total activity of the ORIGEN2 output isotopes.

Template 7
Fuel-Specific Source Term Calculations
N-Reactor Fuel

Introduction

The N-Reactor spent nuclear fuel (SNF) currently resides at the United States Department of Energy (DOE) Hanford Site. The SNF is stored in two water-filled pools, namely, the 105-KE Basin (KE Basin) and the 105-KW Basin (KW Basin). The combined total SNF mass of the two basins is approximately 2,100 MT. This mass represents greater than 91% of the total DOE SNF uranium mass and a significant fraction of the total DOE SNF source term as well.

The radionuclide inventory data or source term used to develop this template is based on N-Reactor radionuclide inventories previously calculated by the Hanford site and is taken directly from Reference 1. Specifically, the "Safety/Regulatory Assessment Feed" design basis radionuclide inventory (Table 3.9, Reference 1) was selected as the template basis. This particular design basis inventory represents a high burnup (16.49% Pu-240) isotopic mixture expected to yield the largest dose to people per unit of material released and thus a maximum per unit mass source term. The inventory is decayed to a single date (May 31, 1998) or 22.08 years following discharge from the reactor.

The N-Reactor radionuclide inventory data (Table 3.9, Reference 1) is a relatively comprehensive list of radionuclides and the corresponding activities in terms of Ci/MTU. However, the N-Reactor inventory does not provide radionuclide activities for all of the radionuclides identified in "Guide for Estimating DOE Spent Nuclear Fuel Source Terms" (Reference 2). Specifically missing are the actinides Th-229, Th-232, U-232, and U-233, as well as the actinide daughter decay products Ac-227, Pa-231, Pb-210, Ra-226, and Ra-228. Because the N-Reactor fuel did not contain thorium in the initial or beginning-of-life (BOL) fuel mass, the four actinides (Th-229, Th-232, U-232, and U-233) are not expected to exist in any significant quantity in irradiated N-Reactor fuel. The five daughter decay products (Ac-227, Pa-231, Pb-210, Ra-226, and Ra-228) are dependent on the decay time following reactor discharge.

In order to provide activity values for these five daughter decay products, and at the same time provide N-Reactor inventories as a function of additional decay times, the Table 3.9 data was decayed using the ORIGEN2 code (Reference 3) and various decay dates out to 100-years following reactor discharge. This way both the SNF template format is met and all the important radionuclides have an associated activity in the template as a function of decay time. However, it should be noted that the daughter decay product activities will be slightly underpredicted, because they are decayed from May 31, 1998 (Table 3.9 data) and not the discharge date.

It should be emphasized that the Idaho National Engineering and Environmental Laboratory (INEEL) spent nuclear fuel source term calculational methodology used to generate other template source terms is not used here to generate the N-Reactor source term. The N-Reactor source term is taken directly out of Table 3.9 from Reference 1. These data (Table 3.9, Reference 1) are simply decayed out to the 100-year time frame and reported in this template radionuclide inventory along with the initial Table 3.9 data which corresponds to the 22.08-year decay time. The decay calculation performed as part of the N-Reactor template development herein utilized the ORIGEN2 computer code and the standard ORIGEN2 decay libraries (Reference 3).

N-Reactor

The following description of the N-Reactor in this section is taken almost verbatim directly out of Reference 5.

The 105-N Reactor (N-Reactor) is a graphite-moderated, pressurized water-cooled reactor located in the 100-N Area of the Hanford Site. It was initially designed for plutonium production for national defense. Initial operation began in 1963. Two years later, N-Reactor was modified to produce steam to be used by the Washington Public Power Supply System to generate electricity. N-Reactor was the only dual-purpose reactor in the United States.

The core of the N-Reactor was a 1800-ton graphite block, 33 feet (10 meters) high by 33 feet (10 meters) wide by 39 feet (12 meters) long. A total of 1003 horizontal Zircaloy-2 process tubes held the fuel and contained the cooling water. The cooling water transferred the reaction heat from the 366 metric tons of uranium fuel to the secondary coolant water in steam generators. Perpendicular to the process tubes were 84 horizontal water-cooled, boron containing control rods. These rods entered the reactor from both sides and provided operating reactivity control, neutron flux shaping, and emergency shutdown control. Completely independent, backup emergency shutdown control was provided by 107 vertical channels penetrating the core that could be gravity-filled with special neutron absorbing balls.

When operating, N-Reactor produced up to 4,000 MW_{th} of heat energy and up to 13 million pounds per hour of low-pressure steam, which produced 860 MWe. The production of tritium and various nondefense target elements was also demonstrated.

N-Reactor ceased operation in 1987, but fuel remained in the core pending a possible restart. The final core was discharged in April 1989.

N-Reactor Fuel Element Data

The radionuclide inventory from Table 3.9 (Reference 1) is based on the N-Reactor MARK IV fuel assembly. This assembly consists of two concentric annular fuel elements (termed inner element and outer element). The fuel meat is uranium metal with Zircaloy-2 cladding on both inner and outer surfaces of each element. Both elements have a Beginning-of-Life (BOL) enrichment of 0.947% U-235 and a combined total uranium weight of approximately 22.7 kg or 50 lb.

The following N-reactor fuel assembly table data is based on References 1 and 4.

Outer Element Diameters	
Zircaloy Clad OD	6.160 cm
Uranium meat OD	6.032 cm
Uranium meat ID	4.422 cm
Zircaloy Clad ID	4.321 cm
Outer Element Enrichments	
U-235	0.94700 wt%
U-236	0.03920 wt%
U-238	99.0138 wt%

Inner Element Diameters	
Zircaloy Clad OD	3.249 cm
Uranium meat OD	3.096 cm
Uranium meat ID	1.321 cm
Zircaloy Clad ID	1.219 cm
Inner Element Enrichments	
U-235	0.94700 wt%
U-236	0.03920 wt%
U-238	99.0138 wt%
Assembly Uranium Mass	22.7 kg
Assembly Dimensions	
Maximum Length	66.294 cm
End Cap Thickness	0.483 cm
Fuel Assembly Max. Weight	23.4 kg

Based on the above table data and the fact that the radionuclide inventory from Table 3.9 (Reference 1) is based on a BOL uranium total mass of 11.6 MTU, the following BOL isotopic uranium masses can be estimated.

Uranium Isotope	BOL Mass (grams)
U-235	109,852.0
U-236	4,547.0
U-238	11,485,601.0
Total U	11,600,000.0

Burnup and Time Since Discharge

The K Basin inventory of N-Reactor SNF is composed of assemblies that experienced a range of burnups and were discharged from the reactor between January 1971 and April 1987. The burnups range from 0.0 to approximately 6000 megawatt-days (MWd) per MTU.

Accountability records have been used to subdivide the K Basin N-Reactor assemblies by burnup and mass in order to estimate total radionuclide inventories. The accountability record run data listing includes (1) discharge date, (2) fuel type, and (3) other information for 497 keys of fuel assemblies. Each of the 497 keys includes assemblies of the same type, same burnup, and same discharge date from the reactor.

The burnup of a N-Reactor fuel key is historically ranked by End-of-Life (EOL) Pu-240 concentration or Pu-240 weight percent of the plutonium mass. Pu-240 concentration increases with burnup or exposure time in the core and is a direct indicator of an assembly or key burnup. Typically, seven bins (<5%, 5-7%, 7-9%, 9-11%, 11-13%, 13-15%, and >15% Pu-240) are used to categorize spent N-Reactor assemblies.

In the highest burnup bin (>15% Pu-240), there are two fuel keys with maximum burnups of 16.72 % Pu-240 and 16.49% Pu-240 and discharge dates of February 20, 1976 and May 1, 1976, respectively. When decayed to May 31, 1998, the 16.49% Pu-240 fuel was found to have a mix of isotopes that produced a maximum dose to people. The total BOL uranium mass of this 16.49% Pu-240 fuel stored at the K Basins was 11.6 MTU. A complete listing of the specific radionuclide composition in Ci/MTU for this fuel key, decayed to May 31, 1998, is listed in the template column headed with a 22.08-year decay.

It should be noted that the application of the 16.49 wt% Pu-240 SNF template here to all the N-Reactor SNF K-Basin fuel inventory will produce a conservative or over-estimate of the actual total radionuclide inventory.

Cross Section Development

ORIGEN2 S.2 runs were used to generate the radionuclide inventories used to characterize the N-Reactor spent fuel. These same ORIGEN2 runs were apparently used to generate the Table 3.9 (Reference 1) radionuclide data and are based on improved cross section libraries generated by the WIMS-E computer code (Reference 6).

Fuel/Clad Impurities

Fuel and cladding material constituents, both major and minor (impurities), activate, fission, and transmute into a wide variety of radionuclides. Comprehensive lists of BOL fuel, clad, and other structural and poison materials are key to fully characterizing the SNF and determining its EOL radionuclide inventory. Pre-irradiated impurity concentrations are typically low relative to the major constituent concentrations, but can lead through activation to significant quantities of important radionuclides. Hence, it is important for a depletion or activation calculation to include as many known impurity element and their concentrations as possible.

The depletion calculation used to develop the N-Reactor radionuclide inventory (Table 3.9, Reference 1) is based on the major and minor impurity elements in the uranium metal fuel, the Zircaloy-2 clad, and the zirconium-beryllium braze material used to seal the fuel assemblies as given in Table 3.4, Reference 1. This elemental list and the associated elemental masses are for pre-irradiated conditions and given specifically in terms of the total N-Reactor fuel, clad, and braze mass, or 2,100,000 kg, 145,000 kg, and 3,000 kg, respectively. The Table 3.4 values are scaled to a parts-per-million (ppm) basis for presentation purposes and conformity to the template format. These impurities are listed in Table 1 below.

Decay Calculation

The N-Reactor radionuclide inventory template is based on the "Safety/Regulatory Assessment Design Basis" per Reference 1. This inventory is based on 16.49 wt% Pu-240 exposure fuel and is reported to have a discharge date of May 1, 1976 and is decayed to May 31, 1998. This would represent a 22.08-year decay period.

In order to further decay the radionuclide inventory out to 25, 35, 50, 65, 80, and 100 years for the template format, the ORIGEN2 computer code was used to perform the decay calculation. See Table 2 for the decay history. The first step was to modify the radionuclide inventory or source term from the Table 3.9 (Reference 1) "Safety/Regulatory Assessment Design Basis" radionuclide list (Ci/MTU) in order to load the mass vector into the ORIGEN2 decay input deck. The list was loaded as a mass (grams) vector where the Table 3.9 data was multiplied by the BOL 11.6 MTU (template basis) and divided by the appropriate radionuclide curie-to-gram (Ci/gm) conversion factor. The inventory was then decayed for the six additional decay dates with the radionuclide activities output in terms of Ci. The total Ci inventory for the 16.49% Pu-240 key was then divided by the total key BOL uranium mass (11.6 MTU) in order to convert back to the more convenient units of Ci/MTU and it is these values that are reported in the attached radionuclide inventory temple here.

References

1. M. J. Packer, "105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities," Volume 1, *Fuel*, HNF-SD-SNF-TI-009, Rev. 3, November 4, 1999.

2. National Spent Nuclear Fuel Program, *Guide for Estimating DOE Spent Nuclear Fuel Source Terms*, DOE/SNF/REP-059, July 2000.

3. A. G. Croff, *ORIGEN2—A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, July 1980.

4. Duke Engineering and Services, *Criticality Safety Evaluation Report for Spent Nuclear Fuel Processing and Storage Facilities*, HNF-SD-SNF-CSER-005, Revision 3, Fluor Daniel Northwest, Richland Washington, February 1997.

5. K. H. Bergsman, *Hanford Spent Fuel Inventory Baseline*, WHC-SD-SNF-TI-001, Rev. 0, July 15, 1994.

6. M. J. Packer, "Single Use Letter Report for the Verification and Validation of the RADNUC2A and ORIGEN S.2 Computer Codes," SNF-4503, Rev. 0, DE&S Hanford, Richland, Washington.

Table 1. N-Reactor fuel assembly material impurity concentrations.

Constituent or Impurity	Uranium Metal Concentration (ppm)	Zircaloy-2 Cladding Concentration (wt%)	Braze Filler Concentration (ppm)
H	2	25.5	47.3
Li			
Be	10	—	47333
B	0.25	0.51	0.47
C	366–738	280.7	473
N	75.2	81.4	189
O	—	—	2177
F			
Na	—	20.4	18.9
Mg		20.4	56.7
Al	705–905	76.5	566670
Si	124.3	102	236
P			
S			
Cl			
K			
Ca			
Sc			
Ti	—	51	47.3
V	—	51	47.3
Cr	65	510–1531	473–1420
Mn	25	51	56.7
Fe	301–401	717–2041	567–1987
Co	—	10.2	18.9
Ni	100.5	306–814	283–757
Cu	75.2	51	56.7
Zn			
Ga			
As			
Se			
Br			
Rb			
Sr			
Y			
Zr	358.6	100 wt%	926667
Nb			
Mo	—	51	47.3
Ag			
Cd	0.25	0.51	0.47
In			

Table 1. (continued).

Constituent or Impurity	Uranium Metal Concentration (ppm)	Zircaloy-2 Cladding Concentration (wt%)	Braze Filler Concentration (ppm)
Sn	—	12276–17379	10767–16067
Sb			
Cs			
Ba			
La			
Ce			
Pr			
Nd			
Sm			
Eu			
Gd			
Tb			
Dy			
Ho			
Er			
Tm			
Yb			
Lu			
Hf	—	204	189
Ta			
W	—	51	95
Tl			
Pb	—	102	123
Bi			
Th			
U	100 wt%	3.6	3.77

Table 2. N-Reactor decay history used in the template decay calculation.

Dates	Differential Decay Time (years)	Cumulative Decay Time (days)	Cumulative Decay Time (years)	Time-Averaged Power (MWth)
1-May-1976 (discharge)	0.0	0.0	0.0	0.0
31-May-1998 (Table 3.9 decay date)	22.08	8065.00	22.08	0.0
01-May-2001	2.92	9131.00	25.00	0.0
01-May-2011	12.92	12783.00	35.00	0.0
01-May-2026	27.92	18262.00	50.00	0.0
01-May-2041	42.92	23741.00	65.00	0.0
01-May-2056	57.92	29220.00	80.00	0.0
01-May-2076	77.92	36525.00	100.00	0.0

The radionuclide decay dates begin with May 31, 1998, or 22.08-years after discharge from the N-Reactor core followed by 25, 35, 50, 65, 80, and 100-year decay times in accordance with the template format.

N-Reactor Fuel

Zircaloy-2 Cladding, Uranium Metal Fuel

Fuel Meat:	Uranium Metal
BOL U-235 Fuel Enrichment:	0.947 wt%
Cladding:	Zircaloy-2

16.49% Pu-240 Key Data:

Burnup:	16.49% Pu-240 (maximum burnup)
BOL U-235:	109852.0 g U-235
BOL U-236:	4547.0 g U-236
BOL U-238:	11485601.0 g U-238
BOL Total U:	11.6 MTU

DECAY TIMES (years out of core)
(Activities* in Ci/MTU)

Radionuclide	5	10	15	22.08	25	35	50	65	80	100
H 3				2.610E+01	2.216E+01	1.264E+01	5.447E+00	2.347E+00	1.011E+00	3.291E-01
BE 10				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
C 14				5.530E-01	5.528E-01	5.522E-01	5.511E-01	5.501E-01	5.491E-01	5.478E-01
CL 36				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CR 51				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FE 55				5.410E-01	2.484E-01	1.728E-02	3.168E-04	5.809E-06	1.065E-07	5.148E-10
FE 59				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60				2.091E+00	1.424E+00	3.822E-01	5.314E-02	7.388E-03	1.027E-03	7.398E-05
NI 59				3.179E-02	3.179E-02	3.179E-02	3.179E-02	3.178E-02	3.178E-02	3.178E-02
NI 63				3.470E+00	3.394E+00	3.148E+00	2.811E+00	2.511E+00	2.243E+00	1.929E+00
ZN 65				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SE 79				6.541E-02	6.540E-02	6.540E-02	6.538E-02	6.537E-02	6.536E-02	6.534E-02
KR 85				3.699E+02	3.063E+02	1.604E+02	6.083E+01	2.306E+01	8.741E+00	2.399E+00
RB 87				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 89				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90				6.928E+03	6.463E+03	5.094E+03	3.565E+03	2.494E+03	1.746E+03	1.084E+03
Y 90				6.930E+03	6.465E+03	5.096E+03	3.566E+03	2.495E+03	1.746E+03	1.084E+03
Y 91				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93				2.948E-01	2.948E-01	2.948E-01	2.948E-01	2.948E-01	2.948E-01	2.948E-01
ZR 95				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M				1.929E-01	2.049E-01	2.349E-01	2.591E-01	2.703E-01	2.755E-01	2.784E-01

DECAY TIMES (years out of core)
(Activities* in Ci/MTU)

Radionuclide	5	10	15	22.08	25	35	50	65	80	100
NB 94				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC 99				2.190E+00	2.190E+00	2.190E+00	2.190E+00	2.190E+00	2.190E+00	2.190E+00
RU103				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106				2.560E-02	3.441E-03	3.553E-06	1.177E-10	3.899E-15	1.291E-19	1.375E-25
RH103M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106				2.559E-02	3.441E-03	3.553E-06	1.177E-10	3.899E-15	1.291E-19	1.375E-25
PD107				1.560E-02	1.560E-02	1.560E-02	1.560E-02	1.560E-02	1.560E-02	1.560E-02
AG110				7.172E-10	3.728E-11	1.485E-15	3.728E-22	9.353E-29	2.347E-35	3.719E-44
AG110M				5.391E-08	2.803E-09	1.117E-13	2.803E-20	7.034E-27	1.765E-33	2.797E-42
AG111				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD115M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M				6.137E-08	3.009E-09	9.810E-14	1.821E-20	3.379E-27	6.272E-34	6.647E-43
SN121M				6.272E-02	6.024E-02	5.244E-02	4.259E-02	3.459E-02	2.809E-02	2.128E-02
SN123				1.720E-16	5.638E-19	1.738E-27	2.957E-40	5.032E-53	8.563E-66	8.089E-83
SN125				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN126				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB124				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB126				1.810E-02	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB126M				1.291E-01	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE123M				1.500E-21	3.124E-24	2.035E-33	3.364E-47	5.560E-61	9.190E-75	3.878E-93
TE125M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE127				2.119E-19	2.408E-22	1.978E-32	1.463E-47	1.082E-62	8.004E-78	5.366E-98
TE127M				2.160E-19	2.459E-22	2.019E-32	1.493E-47	1.105E-62	8.172E-78	5.478E-98
TE129				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

DECAY TIMES (years out of core)
(Activities* in Ci/MTU)

Radionuclide	5	10	15	22.08	25	35	50	65	80	100
I129				5.160E-03	5.160E-03	5.160E-03	5.160E-03	5.160E-03	5.160E-03	5.160E-03
I131				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE131M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE133				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS134				6.477E+00	2.428E+00	8.425E-02	5.440E-04	3.512E-06	2.267E-08	2.728E-11
CS135				6.040E-02	6.040E-02	6.040E-02	6.040E-02	6.040E-02	6.040E-02	6.039E-02
CS136				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS137				9.655E+03	9.026E+03	7.167E+03	5.067E+03	3.584E+03	2.534E+03	1.596E+03
BA136M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA137M				9.138E+03	8.542E+03	6.780E+03	4.794E+03	3.390E+03	2.397E+03	1.509E+03
BA140				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA140				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE144				7.909E-04	5.878E-05	7.975E-09	1.258E-14	1.983E-20	3.126E-26	5.741E-34
PR143				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144				7.820E-04	5.807E-05	7.879E-09	1.242E-14	1.959E-20	3.089E-26	5.672E-34
PR144M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND144				0.000E+00	2.682E-19	2.897E-19	2.897E-19	2.897E-19	2.897E-19	2.897E-19
ND147				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM145				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM147				1.091E+02	5.041E+01	3.591E+00	6.823E-02	1.297E-03	2.463E-05	1.249E-07
PM148M				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM147				0.000E+00	1.436E-09	2.584E-09	2.671E-09	2.672E-09	2.672E-09	2.672E-09
SM151				1.020E+02	9.974E+01	9.233E+01	8.226E+01	7.328E+01	6.528E+01	5.597E+01
EU152				8.450E-01	7.282E-01	4.375E-01	2.037E-01	9.483E-02	4.415E-02	1.593E-02
EU154				1.129E+02	8.922E+01	3.987E+01	1.191E+01	3.553E+00	1.060E+00	2.116E-01
EU155				1.060E+01	7.051E+00	1.743E+00	2.141E-01	2.631E-02	3.233E-03	1.975E-04
EU156				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD153				5.190E-10	2.450E-11	7.025E-16	1.075E-22	1.647E-29	2.520E-36	2.066E-45
TB160				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

DECAY TIMES (years out of core)
(Activities* in Ci/MTU)

Radionuclide	5	10	15	22.08	25	35	50	65	80	100
TL207				0.000E+00	3.522E-08	6.234E-07	2.525E-06	5.233E-06	8.440E-06	1.317E-05
TL208				0.000E+00	1.428E-13	5.123E-12	2.063E-11	3.893E-11	5.785E-11	8.681E-11
PB210				0.000E+00	1.876E-10	1.516E-08	1.381E-07	4.552E-07	1.022E-06	2.222E-06
PB211				0.000E+00	3.532E-08	6.252E-07	2.532E-06	5.247E-06	8.464E-06	1.321E-05
PB212				0.000E+00	3.974E-13	1.426E-11	5.741E-11	1.084E-10	1.610E-10	2.416E-10
BI211				0.000E+00	3.532E-08	6.252E-07	2.532E-06	5.247E-06	8.464E-06	1.321E-05
BI212				0.000E+00	3.974E-13	1.426E-11	5.741E-11	1.084E-10	1.610E-10	2.416E-10
PO212				0.000E+00	2.547E-13	9.138E-12	3.678E-11	6.942E-11	1.032E-10	1.548E-10
PO215				0.000E+00	3.532E-08	6.252E-07	2.532E-06	5.247E-06	8.464E-06	1.321E-05
PO216				0.000E+00	3.974E-13	1.426E-11	5.741E-11	1.084E-10	1.610E-10	2.416E-10
RN219				0.000E+00	3.532E-08	6.252E-07	2.532E-06	5.247E-06	8.464E-06	1.321E-05
RN220				0.000E+00	3.974E-13	1.426E-11	5.741E-11	1.084E-10	1.610E-10	2.416E-10
FR223				0.000E+00	4.873E-10	8.629E-09	3.493E-08	7.236E-08	1.167E-07	1.822E-07
RA223				0.000E+00	3.532E-08	6.252E-07	2.532E-06	5.247E-06	8.464E-06	1.321E-05
RA224				0.000E+00	3.974E-13	1.426E-11	5.741E-11	1.084E-10	1.610E-10	2.416E-10
RA226				0.000E+00	6.381E-09	1.252E-07	5.862E-07	1.388E-06	2.533E-06	4.594E-06
RA228				0.000E+00	1.410E-12	2.046E-11	6.641E-11	1.179E-10	1.708E-10	2.416E-10
AC227				0.000E+00	3.532E-08	6.252E-07	2.531E-06	5.244E-06	8.456E-06	1.320E-05
TH227				0.000E+00	3.483E-08	6.166E-07	2.497E-06	5.175E-06	8.347E-06	1.303E-05
TH228				0.000E+00	3.974E-13	1.426E-11	5.741E-11	1.084E-10	1.610E-10	2.416E-10
TH229				0.000E+00	8.222E-11	1.638E-09	7.757E-09	1.864E-08	3.453E-08	6.399E-08
TH230				0.000E+00	1.010E-05	4.492E-05	9.776E-05	1.511E-04	2.051E-04	2.778E-04
TH231				0.000E+00	1.270E-02	1.270E-02	1.270E-02	1.271E-02	1.271E-02	1.271E-02
TH232				0.000E+00	1.031E-11	4.564E-11	9.871E-11	1.517E-10	2.049E-10	2.758E-10
TH234				0.000E+00	3.309E-01	3.309E-01	3.309E-01	3.309E-01	3.309E-01	3.309E-01
PA231				0.000E+00	7.839E-07	3.471E-06	7.500E-06	1.153E-05	1.556E-05	2.093E-05
PA233				0.000E+00	4.703E-02	4.865E-02	5.135E-02	5.421E-02	5.710E-02	6.093E-02
PA234M				0.000E+00	3.309E-01	3.309E-01	3.309E-01	3.309E-01	3.309E-01	3.309E-01
PA234				0.000E+00	4.303E-04	4.303E-04	4.303E-04	4.303E-04	4.303E-04	4.303E-04
U232				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
U233				0.000E+00	5.975E-07	2.688E-06	5.966E-06	9.422E-06	1.308E-05	1.823E-05
U234				3.840E-01	3.851E-01	3.886E-01	3.934E-01	3.978E-01	4.016E-01	4.059E-01
U235				1.270E-02	1.270E-02	1.270E-02	1.270E-02	1.271E-02	1.271E-02	1.271E-02
U236				7.159E-02	7.160E-02	7.165E-02	7.171E-02	7.177E-02	7.183E-02	7.191E-02

DECAY TIMES (years out of core)
(Activities* in Ci/MTU)

Radionuclide	5	10	15	22.08	25	35	50	65	80	100
U237				0.000E+00	1.453E-03	8.983E-04	4.362E-04	2.119E-04	1.029E-04	3.929E-05
U238				3.309E-01	3.309E-01	3.309E-01	3.309E-01	3.309E-01	3.309E-01	3.309E-01
NP237				4.660E-02	4.703E-02	4.865E-02	5.135E-02	5.421E-02	5.710E-02	6.093E-02
PU236				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU237				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238				1.330E+02	1.300E+02	1.202E+02	1.068E+02	9.483E+01	8.429E+01	7.201E+01
PU239				1.730E+02	1.730E+02	1.730E+02	1.729E+02	1.728E+02	1.728E+02	1.727E+02
PU240				1.370E+02	1.369E+02	1.368E+02	1.366E+02	1.364E+02	1.361E+02	1.359E+02
PU241				6.817E+03	5.924E+03	3.661E+03	1.778E+03	8.638E+02	4.195E+02	1.602E+02
PU242				8.716E-02	8.716E-02	8.716E-02	8.716E-02	8.716E-02	8.716E-02	8.707E-02
PU244				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
AM241				4.339E+02	4.616E+02	5.290E+02	5.783E+02	5.946E+02	5.951E+02	5.847E+02
AM242M				3.721E-01	3.672E-01	3.508E-01	3.276E-01	3.059E-01	2.857E-01	2.608E-01
AM242				3.710E-01	3.653E-01	3.491E-01	3.259E-01	3.044E-01	2.843E-01	2.595E-01
AM243				2.780E-01	2.779E-01	2.777E-01	2.773E-01	2.769E-01	2.766E-01	2.760E-01
CM242				3.081E-01	3.028E-01	2.886E-01	2.696E-01	2.517E-01	2.351E-01	2.146E-01
CM243				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CM244				4.471E+00	3.998E+00	2.727E+00	1.535E+00	8.647E-01	4.871E-01	2.266E-01
CM245				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CM246				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CM247				0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TOTAL**				4.110E+04	3.791E+04	2.908E+04	1.994E+04	1.394E+04	9.914E+03	6.467E+03

* Four decimal places of accuracy are as reported by ORIGEN2 output and are not significant for many radionuclides.
** Total: total activity of the 145 isotopes listed in the table.

Template 8

Fuel-Specific Source Term Calculations
High Flux Beam Reactor Fuel

Introduction

The following data have been used in the Idaho National Engineering and Environmental Laboratory (INEEL) spent nuclear fuel source term calculational methodology to generate a source term template for a single High Flux Beam Reactor (HFBR) spent nuclear fuel element. This single-element source term uses the highest burnup of the 240 elements stored at the INEEL and is considered to be a bounding source term for these high-enriched spent nuclear fuel elements. The data sources for the analysis are documented in References 1 through 7, and the INEEL calculational methodology is described in detail in Reference 8.

HFBR Reactor Data

The HFBR core and fuel elements are described in References 1 through 7. Data from these references have been used to develop reactor physics models for the depletion/activation analysis.

The HFBR fuel elements are plate-type elements consisting of 19 curved plates. The plates are stacked, separated by a heavy water gap (102–129 mils), and held together as a rectangular structure by two aluminum side plates (140 mils thick). Plates 1 and 19 are aluminum plates containing no fuel. The fuel meat in Plates 2 through 18 is a mixture of U_3O_8 in an aluminum matrix and clad on both sides with aluminum, as shown in Figure 1. The uranium enrichment is nominally 93% high-enriched uranium metal. The uranium isotopic data are given below.

The following data provide specific fuel element dimensions, materials, densities, enrichment, etc. which are typical for a heavy-loaded KM-type HFBR fuel element. There is also a less heavily loaded fuel element in the HFBR called a KL-type fuel element. Both the KM and KL-type fuel elements are identical with the exception of the initial beginning-of-life (BOL) loading. The BOL data below was used in the burnup calculation for the source term generation and is based on a KM-type HFBR fuel element.

Fuel Element:	Curved plate
No. of Plates:	19
Fueled Plates:	17
Fuel Plate Thickness:	50 mils
XY dimensions :	2.820 in. by 3.194 in.
Length:	23.75 in.
Fuel Rod:	U_3O_8 in an Aluminum-6061T matrix
Fuel Density:	3.608 g/cc
Fuel Meat Thickness:	21 mils
Uranium Enrichment:	0.60 wt % U-234
	93.00 wt % U-235
	0.35 wt % U-236
	6.05 wt % U-238

Heavy Metal Loading:	2.26 g/element	U-234 (BOL)
	350.61 g/element	U-235 (BOL)
	1.32 g/element	U-236 (BOL)
	22.81 g/element	U-238 (BOL)
	377.00 g/element	Total U

Clad:	Aluminum 6061T
Clad Density:	2.70 g/cc
Clad Thickness:	14.5 mils
Side Plates:	Aluminum 6061T
Side Plate Width:	140 mils
Total Aluminum Mass:	4,064.13 g/element
Coolant/Moderator :	Heavy Water (D_2O)
Coolant Temperature:	52 C
Coolant Pressure:	175.3 psig
Coolant Density:	1.09325 g/cc

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single HFBR fuel element. In addition, for the ORIGEN2 (Reference 9) depletion calculation, conservative and detailed impurity concentrations were added for the aluminum clad and structural components. Table 1 lists the Aluminum 6061T impurities and their concentrations (Reference 10).

Burnup

The burnup chosen for this template is 62.3% U-235 depletion, 164.6 MWd, and approximately 218.4 g of U-235 depleted for a single HFBR KM-type element. This is a relatively high burnup and represents the maximum burnup of the HFBR elements stored at the INEEL. This burnup is conservative with respect to the buildup of fission products, activation products, and minor actinides in the source term, but nonconservative with regard to criticality safety, in particular U-235 and U-238 end-of-life concentrations.

For the template analysis here, the burnup period is assumed to be a 3-cycle exposure. The first cycle runs for 24-days, followed by a 25-day cycle, and finally another 24-day cycle. Between Cycles 1 and 2 and Cycles 2 and 3, there is an assumed shutdown period of 14 days. At the end of the third cycle, the element is assumed to be removed from the core and the cooling or decay period begins. During the burnup period, the fuel element output power is assumed to be approximately 2.255 MW and is assumed to be the same and constant for each of the three cycles (see Table 2).

Cross-Section Development

The neutron cross sections used in the burnup or depletion calculation for the source term generation of a single HFBR fuel element are based on the methodology described in Reference 8. Cross sections from a standard ORIGEN2 heavy water reactor library were updated once using the specially developed BOL cross sections for the HFBR. The updated cross sections take into account the unique HFBR neutron flux spatial and spectral characteristics to ensure accurate calculation of the fission product and actinide production as a function of burnup.

In order to calculate the BOL HFBR neutron cross sections, an explicit HFBR fuel assembly was developed with reflective boundary conditions on the element peripheral surfaces. The reflective surfaces simulated an infinite array of HFBR fuel elements.

HFBR Single Element Exposure History

Table 2 summarizes the power or exposure history used in the burnup or source term calculations for a single HFBR fuel element. Following the burnup or exposure period, the radionuclide activities are decayed for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100 years.

Burnup Calculation

The ORIGEN2 computer code (Reference 9) was used to perform the depletion or burnup calculation for the HFBR fuel element. The radionuclide inventory or source term template is for a single HFBR fuel element or assembly. The fuel element masses and impurities, neutron cross sections, burnup, power history, and power level as discussed above are input data for the ORIGEN2 calculation. The radionuclide concentrations are given as a function of time in the template table.

The 145 radionuclides listed in the template represent greater than 99.99% of the total curie inventory had all 684 activation products, 880 fission products, and 127 actinide/daughter isotopes from the ORIGEN2 output been included in the template.

References

1. Brookhaven National Laboratory to Westinghouse Idaho Nuclear Co., “Basin Storage Fuel Receipt Criteria, Part A,” Idaho Chemical Processing Plant, May 4, 1989.

2. Paul Colsmann (Brookhaven National Laboratory) to Gary Offutt (Exxon Nuclear Idaho Co.), Letter report regarding fuel element description and data, February 11, 1983.

3. G. Price (Brookhaven National Laboratory) to G. Kinne, “MRR fuel elements in the MH-1A shipping cask,” Memorandum, February 22, 1985.

4. Mark Davis (Brookhaven National Laboratory) to J. Sawyer (Westinghouse Idaho Nuclear Co.), Letter report regarding fuel element description and data, March 15, 1985.

5. Drawing BR 51-0400-1 Rev. A, “HFBR Fuel Element Type KL Details,” Brookhaven National Laboratory, Upton, New York, August 11, 1988.

6. Drawing ME55-95 Rev. A, “HFBR Fuel Element,” Brookhaven National Laboratory, Upton, New York.

7. “Research, Training, Test and Production Reactor Directory,” 3rd edition, pages 50-57, published by the American Nuclear Society, 1988.

8. J. W. Sterbentz and C. A. Wemple, *Calculational Burnup Methodology and Validation for the Idaho National Engineering Laboratory Spent Nuclear Fuels*, INEL-96/0304, September 1996.

9. A. G. Croff, *ORIGEN2 – A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, July 1980.

10. ASTM B-209, Table 1, “Chemical Composition Limits,” 1990.

Table 1. HFBR Aluminum-6061T material constituent and impurity concentrations

Constituent or Impurity	Concentration (wt%)
H	0.02143
C	0.02143
O	0.02143
Mg	1.00000
Al	97.15499
Si	0.60000
Ti	0.07500
Cr	0.19500
Mn	0.07500
Fe	0.35000
Ni	0.02143
Cu	0.27500
Zn	0.12500
Zr	0.02143
Sn	0.02143
Pb	0.02143

Table 2. Assumed burnup or power history for a single HFBR fuel element.

Duration (days)	Cumulative Duration (days)	Time- Averaged Power (MW_{th})
24	24	2.255
14	38	0.0
25	63	2.255
14	77	0.0
24	101	2.255
1825	1926	0.0
1825	3751	0.0
1825	5576	0.0
1825	7401	0.0
1825	9226	0.0
3650	12876	0.0
5475	18351	0.0
5475	23826	0.0
5475	29301	0.0
7300	36601	0.0

The bottom ten dates with zero associated power represent the ten different cooling or decay dates after exposure. These ten dates are specifically the 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100-year cooling times designated for the template methodology.

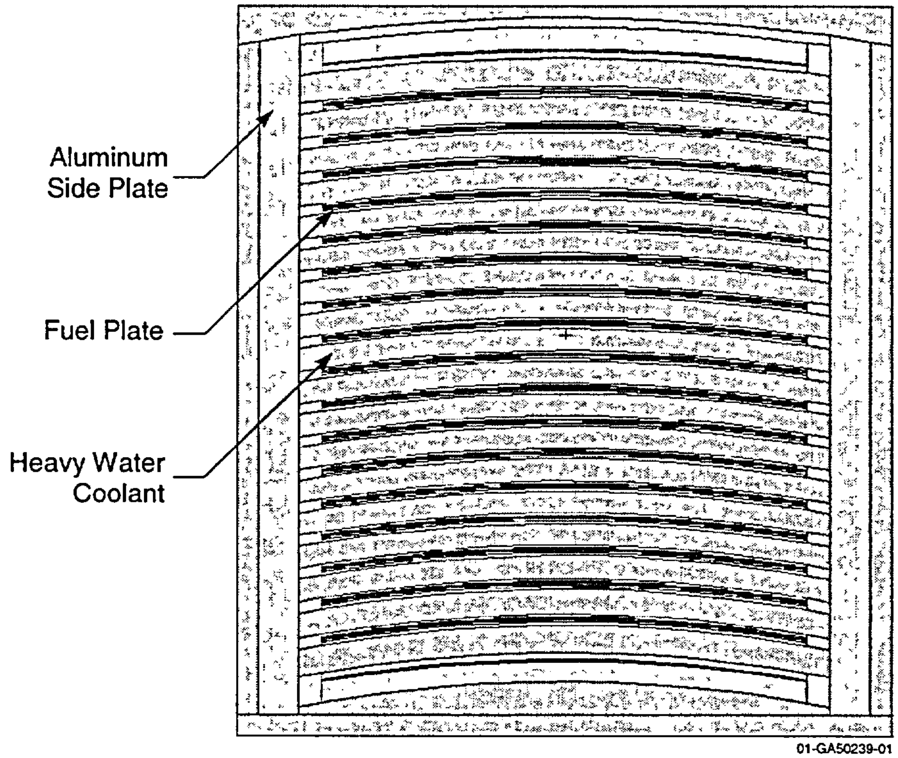

Figure 1. High Flux Beam Reactor curved-plate fuel assembly.

High Flux Beam Reactor Element

Aluminum Cladding, 40 to 100% Enriched U-235 Fuel

Reactor Moderator/Coolant:	Heavy Water
Fuel Meat:	U_3O_8 in Aluminum
Clad:	Aluminum 6061T
Burnup:	436570 MWd/MTU
Burnup:	164.6 MWd/single element (high burnup)
Burnup:	62.3% U-235 depletion (fissioned and transmuted)
Basis of Calculation:	Single fuel element
BOL U-235:	350.61 g U-235 per element
BOL U-238:	22.81 g U-238 per element
BOL U-234:	2.26 g U-234 per element
BOL U-236:	1.32 g U-236 per element
BOL Total U per element:	377.00 g U per element
BOL Fuel Enrichment:	93 wt% U-235

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H 3	1.766E+00	1.334E+00	1.008E+00	7.614E-01	5.752E-01	3.283E-01	1.415E-01	6.101E-02	2.631E-02	8.566E-03
BE 10	1.531E-08	1.531E-08	1.531E-08	1.531E-08	1.531E-08	1.531E-08	1.531E-08	1.531E-08	1.531E-08	1.531E-08
C 14	4.364E-06	4.362E-06	4.358E-06	4.356E-06	4.354E-06	4.348E-06	4.340E-06	4.333E-06	4.325E-06	4.314E-06
CL 36	7.315E-29	7.315E-29	7.315E-29	7.315E-29	7.315E-29	7.315E-29	7.314E-29	7.314E-29	7.314E-29	7.314E-29
CR 51	3.092E-17	4.602E-37	6.851E-57	1.020E-76	1.518E-96	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54	8.728E-03	1.524E-04	2.660E-06	4.644E-08	8.107E-10	2.471E-13	1.315E-18	6.994E-24	3.721E-29	3.457E-36
FE 55	1.214E+01	3.204E+00	8.457E-01	2.232E-01	5.891E-02	4.104E-03	7.545E-05	1.387E-06	2.550E-08	1.238E-10
FE 59	1.057E-11	6.542E-24	4.048E-36	2.505E-48	1.550E-60	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60	5.753E-02	2.982E-02	1.545E-02	8.010E-03	4.152E-03	1.115E-03	1.553E-04	2.162E-05	3.010E-06	2.172E-07
NI 59	1.251E-03	1.251E-03	1.251E-03	1.251E-03	1.251E-03	1.251E-03	1.251E-03	1.250E-03	1.250E-03	1.250E-03
NI 63	2.467E-01	2.376E-01	2.288E-01	2.203E-01	2.122E-01	1.968E-01	1.758E-01	1.570E-01	1.403E-01	1.206E-01
ZN 65	8.871E-01	4.956E-03	2.769E-05	1.547E-07	8.644E-10	2.699E-14	4.707E-21	8.210E-28	1.432E-34	1.396E-43
SE 79	2.031E-03	2.031E-03	2.031E-03	2.031E-03	2.031E-03	2.030E-03	2.030E-03	2.030E-03	2.029E-03	2.029E-03
KR 85	4.689E+01	3.394E+01	2.457E+01	1.779E+01	1.288E+01	6.749E+00	2.560E+00	9.713E-01	3.685E-01	1.012E-01
RB 87	1.409E-07	1.409E-07	1.409E-07	1.409E-07	1.409E-07	1.409E-07	1.409E-07	1.409E-07	1.409E-07	1.409E-07
SR 89	6.465E-07	8.536E-18	1.127E-28	1.488E-39	1.965E-50	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	4.484E+02	3.981E+02	3.535E+02	3.138E+02	2.787E+02	2.197E+02	1.537E+02	1.076E+02	7.532E+01	4.681E+01
Y 90	4.485E+02	3.982E+02	3.536E+02	3.139E+02	2.787E+02	2.197E+02	1.538E+02	1.076E+02	7.534E+01	4.682E+01
Y 91	2.270E-05	9.249E-15	3.768E-24	1.535E-33	6.253E-43	1.038E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93	1.068E-02	1.068E-02	1.068E-02	1.068E-02	1.068E-02	1.068E-02	1.068E-02	1.068E-02	1.068E-02	1.068E-02

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
ZR 95	1.503E-04	3.893E-13	1.009E-21	2.613E-30	6.771E-39	4.545E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M	2.340E-03	4.097E-03	5.458E-03	6.513E-03	7.331E-03	8.457E-03	9.362E-03	9.783E-03	9.980E-03	1.009E-02
NB 94	1.265E-07	1.265E-07	1.264E-07	1.264E-07	1.264E-07	1.264E-07	1.263E-07	1.262E-07	1.262E-07	1.261E-07
NB 95	3.336E-04	8.643E-13	2.239E-21	5.802E-30	1.503E-38	1.009E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	1.115E-06	2.888E-15	7.483E-24	1.939E-32	5.023E-41	3.372E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC 99	6.264E-02	6.264E-02	6.264E-02	6.264E-02	6.264E-02	6.264E-02	6.263E-02	6.263E-02	6.263E-02	6.262E-02
RU103	3.739E-10	3.867E-24	3.999E-38	4.136E-52	4.278E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	3.618E+01	1.165E+00	3.751E-02	1.208E-03	3.888E-05	4.031E-08	1.345E-12	4.490E-17	1.499E-21	1.611E-27
RH103M	3.371E-10	3.486E-24	3.605E-38	3.729E-52	3.856E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	3.618E+01	1.165E+00	3.751E-02	1.208E-03	3.888E-05	4.031E-08	1.345E-12	4.490E-17	1.499E-21	1.611E-27
PD107	8.699E-05	8.699E-05	8.699E-05	8.699E-05	8.699E-05	8.699E-05	8.699E-05	8.699E-05	8.699E-05	8.699E-05
AG110	2.696E-04	1.707E-06	1.081E-08	6.841E-11	4.331E-13	1.736E-17	4.404E-24	1.118E-30	2.836E-37	4.555E-46
AG110M	2.027E-02	1.283E-04	8.125E-07	5.144E-09	3.256E-11	1.305E-15	3.312E-22	8.403E-29	2.132E-35	3.425E-44
AG111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113M	5.666E-02	4.469E-02	3.524E-02	2.780E-02	2.192E-02	1.364E-02	6.690E-03	3.282E-03	1.610E-03	6.229E-04
CD113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD115M	5.021E-12	2.407E-24	1.154E-36	5.531E-49	2.651E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	6.849E-13	5.498E-24	4.412E-35	3.542E-46	2.843E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	7.157E-13	5.744E-24	4.611E-35	3.701E-46	2.971E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	3.529E-16	1.692E-28	8.110E-41	3.887E-53	1.863E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	2.449E-02	1.402E-04	8.028E-07	4.597E-09	2.632E-11	8.630E-16	1.620E-22	3.041E-29	5.709E-36	6.138E-45
SN121M	6.173E-04	5.760E-04	5.374E-04	5.014E-04	4.679E-04	4.073E-04	3.308E-04	2.687E-04	2.183E-04	1.654E-04
SN123	1.608E-03	8.974E-08	5.009E-12	2.796E-16	1.560E-20	4.861E-29	8.451E-42	1.469E-54	2.555E-67	2.479E-84
SN125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN126	1.678E-03	1.678E-03	1.678E-03	1.678E-03	1.678E-03	1.678E-03	1.678E-03	1.677E-03	1.677E-03	1.677E-03
SB124	9.331E-09	6.974E-18	5.214E-27	3.897E-36	2.913E-45	1.627E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	9.106E+00	2.608E+00	7.469E-01	2.139E-01	6.125E-02	5.025E-03	1.181E-04	2.773E-06	6.513E-08	4.382E-10
SB126	2.349E-04	2.349E-04	2.349E-04	2.349E-04	2.349E-04	2.349E-04	2.349E-04	2.348E-04	2.348E-04	2.348E-04
SB126M	1.678E-03	1.678E-03	1.678E-03	1.678E-03	1.678E-03	1.678E-03	1.678E-03	1.677E-03	1.677E-03	1.677E-03
TE123M	2.469E-06	6.339E-11	1.627E-15	4.178E-20	1.073E-24	7.066E-34	1.195E-47	2.022E-61	3.421E-75	1.485E-93
TE125M	2.222E+00	6.363E-01	1.822E-01	5.219E-02	1.494E-02	1.225E-03	2.879E-05	6.766E-07	1.589E-08	1.069E-10
TE127	1.161E-03	1.059E-08	9.657E-14	8.808E-19	8.034E-24	6.683E-34	5.071E-49	3.847E-64	2.919E-79	2.020E-99
TE127M	1.185E-03	1.081E-08	9.859E-14	8.992E-19	8.202E-24	6.823E-34	5.177E-49	3.928E-64	2.980E-79	2.063E-99
TE129	3.550E-14	1.583E-30	7.057E-47	3.146E-63	1.403E-79	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TE129M	5.454E-14	2.432E-30	1.084E-46	4.834E-63	2.155E-79	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	1.093E-04	1.093E-04	1.093E-04	1.093E-04	1.093E-04	1.093E-04	1.093E-04	1.093E-04	1.093E-04	1.093E-04
I131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS134	2.701E+02	5.036E+01	9.390E+00	1.751E+00	3.264E-01	1.134E-02	7.352E-05	4.764E-07	3.087E-09	3.730E-12
CS135	7.006E-04	7.006E-04	7.006E-04	7.006E-04	7.006E-04	7.006E-04	7.006E-04	7.006E-04	7.006E-04	7.006E-04
CS136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS137	4.739E+02	4.222E+02	3.762E+02	3.351E+02	2.986E+02	2.370E+02	1.676E+02	1.186E+02	8.386E+01	5.284E+01
BA136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA137M	4.483E+02	3.994E+02	3.558E+02	3.170E+02	2.825E+02	2.242E+02	1.586E+02	1.122E+02	7.933E+01	4.999E+01
BA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	9.182E-13	1.161E-29	1.469E-46	1.858E-63	2.350E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	1.478E-07	1.478E-07	1.478E-07	1.478E-07	1.478E-07	1.478E-07	1.478E-07	1.478E-07	1.478E-07	1.478E-07
CE144	1.865E+02	2.178E+00	2.543E-02	2.970E-04	3.468E-06	4.728E-10	7.528E-16	1.199E-21	1.908E-27	3.548E-35
PR143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144	1.865E+02	2.178E+00	2.543E-02	2.970E-04	3.468E-06	4.729E-10	7.529E-16	1.199E-21	1.908E-27	3.548E-35
PR144M	2.238E+00	2.614E-02	3.052E-04	3.564E-06	4.161E-08	5.674E-12	9.034E-18	1.438E-23	2.290E-29	4.258E-37
ND144	7.955E-12	8.023E-12	8.024E-12	8.024E-12	8.024E-12	8.024E-12	8.024E-12	8.024E-12	8.024E-12	8.024E-12
ND147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM147	2.108E+02	5.631E+01	1.504E+01	4.017E+00	1.073E+00	7.654E-02	1.458E-03	2.779E-05	5.295E-07	2.695E-09
PM148M	9.845E-12	4.896E-25	2.435E-38	1.211E-51	6.022E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	5.545E-13	2.758E-26	1.371E-39	6.820E-53	3.392E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM147	1.463E-08	1.842E-08	1.943E-08	1.970E-08	1.977E-08	1.979E-08	1.980E-08	1.980E-08	1.980E-08	1.980E-08
SM151	3.313E+00	3.188E+00	3.068E+00	2.952E+00	2.841E+00	2.630E+00	2.343E+00	2.088E+00	1.860E+00	1.595E+00
EU152	5.305E-04	4.112E-04	3.188E-04	2.471E-04	1.916E-04	1.151E-04	5.362E-05	2.498E-05	1.164E-05	4.202E-06
EU154	2.862E+01	1.914E+01	1.279E+01	8.552E+00	5.717E+00	2.555E+00	7.633E-01	2.280E-01	6.813E-02	1.361E-02
EU155	1.912E+01	9.510E+00	4.730E+00	2.353E+00	1.170E+00	2.895E-01	3.563E-02	4.384E-03	5.395E-04	3.302E-05
EU156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD153	7.804E-04	4.191E-06	2.250E-08	1.208E-10	6.488E-13	1.871E-17	2.897E-24	4.486E-31	6.946E-38	5.776E-47
TB160	4.633E-08	1.168E-15	2.947E-23	7.431E-31	1.874E-38	1.192E-53	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	1.255E-13	1.255E-13	1.255E-13	1.255E-13	1.255E-13	1.255E-13	1.255E-13	1.255E-13	1.255E-13	1.255E-13

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TL207	3.538E-09	1.099E-08	2.179E-08	3.545E-08	5.153E-08	8.957E-08	1.575E-07	2.339E-07	3.157E-07	4.295E-07
TL208	4.476E-06	7.224E-06	8.031E-06	8.037E-06	7.781E-06	7.109E-06	6.157E-06	5.330E-06	4.614E-06	3.810E-06
PB210	1.936E-11	1.384E-10	4.447E-10	1.018E-09	1.931E-09	5.044E-09	1.380E-08	2.863E-08	5.065E-08	9.276E-08
PB211	3.548E-09	1.102E-08	2.185E-08	3.554E-08	5.168E-08	8.983E-08	1.579E-07	2.346E-07	3.166E-07	4.307E-07
PB212	1.246E-05	2.010E-05	2.235E-05	2.237E-05	2.166E-05	1.979E-05	1.714E-05	1.483E-05	1.284E-05	1.060E-05
BI211	3.548E-09	1.102E-08	2.185E-08	3.554E-08	5.168E-08	8.983E-08	1.579E-07	2.346E-07	3.166E-07	4.307E-07
BI212	1.246E-05	2.010E-05	2.235E-05	2.237E-05	2.166E-05	1.979E-05	1.714E-05	1.483E-05	1.284E-05	1.060E-05
PO212	7.982E-06	1.288E-05	1.432E-05	1.433E-05	1.388E-05	1.268E-05	1.098E-05	9.504E-06	8.227E-06	6.793E-06
PO215	3.548E-09	1.102E-08	2.185E-08	3.554E-08	5.168E-08	8.983E-08	1.579E-07	2.346E-07	3.166E-07	4.307E-07
PO216	1.246E-05	2.010E-05	2.235E-05	2.237E-05	2.166E-05	1.979E-05	1.714E-05	1.483E-05	1.284E-05	1.060E-05
RN219	3.548E-09	1.102E-08	2.185E-08	3.554E-08	5.168E-08	8.983E-08	1.579E-07	2.346E-07	3.166E-07	4.307E-07
RN220	1.246E-05	2.010E-05	2.235E-05	2.237E-05	2.166E-05	1.979E-05	1.714E-05	1.483E-05	1.284E-05	1.060E-05
FR223	4.896E-11	1.520E-10	3.013E-10	4.899E-10	7.122E-10	1.238E-09	2.177E-09	3.234E-09	4.364E-09	5.938E-09
RA223	3.548E-09	1.102E-08	2.185E-08	3.554E-08	5.168E-08	8.983E-08	1.579E-07	2.346E-07	3.166E-07	4.307E-07
RA224	1.246E-05	2.010E-05	2.235E-05	2.237E-05	2.166E-05	1.979E-05	1.714E-05	1.483E-05	1.284E-05	1.060E-05
RA226	3.675E-10	1.421E-09	3.199E-09	5.736E-09	9.066E-09	1.824E-08	3.870E-08	6.788E-08	1.064E-07	1.732E-07
RA228	1.568E-13	5.237E-13	1.016E-12	1.583E-12	2.195E-12	3.487E-12	5.492E-12	7.519E-12	9.550E-12	1.226E-11
AC227	3.548E-09	1.102E-08	2.183E-08	3.550E-08	5.161E-08	8.974E-08	1.578E-07	2.344E-07	3.163E-07	4.303E-07
TH227	3.499E-09	1.087E-08	2.155E-08	3.505E-08	5.097E-08	8.859E-08	1.557E-07	2.314E-07	3.122E-07	4.248E-07
TH228	1.246E-05	2.009E-05	2.233E-05	2.235E-05	2.164E-05	1.978E-05	1.714E-05	1.483E-05	1.284E-05	1.060E-05
TH229	6.538E-11	1.714E-10	3.308E-10	5.436E-10	8.098E-10	1.502E-09	2.941E-09	4.861E-09	7.261E-09	1.121E-08
TH230	3.254E-07	6.526E-07	9.971E-07	1.358E-06	1.736E-06	2.536E-06	3.843E-06	5.264E-06	6.787E-06	8.955E-06
TH231	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04
TH232	6.979E-13	1.375E-12	2.053E-12	2.730E-12	3.408E-12	4.763E-12	6.796E-12	8.828E-12	1.086E-11	1.357E-11
TH234	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06
PA231	3.879E-08	6.903E-08	9.926E-08	1.295E-07	1.597E-07	2.201E-07	3.107E-07	4.013E-07	4.918E-07	6.125E-07
PA233	5.185E-03	5.186E-03	5.187E-03	5.189E-03	5.191E-03	5.196E-03	5.203E-03	5.212E-03	5.220E-03	5.231E-03
PA234M	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06
PA234	7.951E-09	7.951E-09	7.951E-09	7.951E-09	7.951E-09	7.951E-09	7.951E-09	7.951E-09	7.951E-09	7.951E-09
U232	1.880E-05	2.246E-05	2.275E-05	2.208E-05	2.116E-05	1.927E-05	1.668E-05	1.444E-05	1.250E-05	1.031E-05
U233	1.669E-07	2.802E-07	3.935E-07	5.068E-07	6.202E-07	8.472E-07	1.188E-06	1.529E-06	1.871E-06	2.328E-06
U234	7.076E-03	7.471E-03	7.851E-03	8.216E-03	8.567E-03	9.229E-03	1.013E-02	1.093E-02	1.164E-02	1.246E-02
U235	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04	2.859E-04
U236	2.749E-03	2.749E-03	2.749E-03	2.749E-03	2.749E-03	2.749E-03	2.749E-03	2.749E-03	2.749E-03	2.749E-03
U237	1.116E-05	8.776E-06	6.900E-06	5.425E-06	4.265E-06	2.636E-06	1.281E-06	6.227E-07	3.027E-07	1.157E-07

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
U238	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06	6.116E-06
NP237	5.185E-03	5.186E-03	5.187E-03	5.189E-03	5.191E-03	5.196E-03	5.203E-03	5.212E-03	5.220E-03	5.231E-03
PU236	1.678E-04	4.980E-05	1.478E-05	4.389E-06	1.304E-06	1.172E-07	5.596E-09	2.677E-09	2.601E-09	2.598E-09
PU237	2.498E-14	2.238E-26	2.005E-38	1.796E-50	1.609E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	2.846E+01	2.736E+01	2.630E+01	2.528E+01	2.431E+01	2.246E+01	1.995E+01	1.772E+01	1.574E+01	1.345E+01
PU239	1.145E-01	1.145E-01	1.144E-01	1.144E-01	1.144E-01	1.144E-01	1.143E-01	1.143E-01	1.142E-01	1.142E-01
PU240	6.005E-02	6.068E-02	6.118E-02	6.160E-02	6.194E-02	6.243E-02	6.285E-02	6.304E-02	6.311E-02	6.309E-02
PU241	4.550E+01	3.577E+01	2.813E+01	2.211E+01	1.739E+01	1.075E+01	5.223E+00	2.538E+00	1.234E+00	4.716E-01
PU242	5.088E-04	5.088E-04	5.088E-04	5.088E-04	5.088E-04	5.088E-04	5.088E-04	5.088E-04	5.088E-04	5.088E-04
PU244	2.820E-09	2.820E-09	2.820E-09	2.820E-09	2.820E-09	2.820E-09	2.820E-09	2.820E-09	2.820E-09	2.820E-09
AM241	4.139E-01	7.334E-01	9.812E-01	1.173E+00	1.320E+00	1.519E+00	1.664E+00	1.713E+00	1.715E+00	1.686E+00
AM242M	2.414E-04	2.359E-04	2.306E-04	2.254E-04	2.204E-04	2.105E-04	1.966E-04	1.836E-04	1.715E-04	1.566E-04
AM242	2.402E-04	2.348E-04	2.295E-04	2.243E-04	2.193E-04	2.095E-04	1.956E-04	1.827E-04	1.707E-04	1.558E-04
AM243	6.115E-03	6.112E-03	6.109E-03	6.106E-03	6.104E-03	6.098E-03	6.089E-03	6.081E-03	6.072E-03	6.061E-03
CM242	1.144E-03	1.946E-04	1.899E-04	1.856E-04	1.814E-04	1.732E-04	1.618E-04	1.511E-04	1.411E-04	1.288E-04
CM243	1.352E-03	1.197E-03	1.060E-03	9.387E-04	8.312E-04	6.519E-04	4.527E-04	3.144E-04	2.184E-04	1.343E-04
CM244	1.360E+00	1.123E+00	9.278E-01	7.663E-01	6.329E-01	4.317E-01	2.432E-01	1.370E-01	7.722E-02	3.593E-02
CM245	3.192E-04	3.191E-04	3.190E-04	3.188E-04	3.187E-04	3.184E-04	3.180E-04	3.177E-04	3.173E-04	3.168E-04
CM246	3.676E-05	3.673E-05	3.670E-05	3.668E-05	3.665E-05	3.660E-05	3.652E-05	3.644E-05	3.636E-05	3.625E-05
CM247	2.180E-10	2.180E-10	2.180E-10	2.180E-10	2.180E-10	2.180E-10	2.180E-10	2.180E-10	2.180E-10	2.180E-10
Subtotal**	2.948E+03	1.870E+03	1.569E+03	1.368E+03	1.207E+03	9.489E+02	6.671E+02	4.719E+02	3.354E+02	2.142E+02
TOTAL***	2.948E+03	1.870E+03	1.569E+03	1.369E+03	1.207E+03	9.490E+02	6.672E+02	4.719E+02	3.354E+02	2.142E+02

* Four decimal places of accuracy are as reported by ORIGEN2 output and are not significant for many radionuclides.
** Subtotal: total activity of the 145 isotopes listed in the table.
*** Total: total activity of the ORIGEN2 output isotopes.

Template 9

Representative Fuel Source Term Calculations

Introduction

The following data have been used in the Idaho National Engineering and Environmental Laboratory (INEEL) spent nuclear fuel source term calculational methodology to generate a source term template to represent an aluminum clad, 10 to 20% enriched, uranium-based fuel from a heavy water moderated reactor. No one specific fuel element in the Template 9 group of fuels was singled out for the template development. Because the spent fuels in this group are primarily MTR-type or plate-type fuels, the previously constructed High Flux Beam Reactor (HFBR) fuel element geometry model (Template 8) was modified to represent fuels in this group. Modifications included developing new cross sections and adjusting enrichment and burnup. Other differences include the fuel meat material and the number of fueled plates in the element. These data are included below. The calculational methodology used in the template development here is described in detail in Reference 1.

Reactor Data

The hypothetical element is a plate-type element consisting of 19 curved plates. The plates are stacked, separated by a heavy water gap (102–129 mils), and held together as a rectangular structure by two aluminum side plates (140 mils thick). The fuel meat in plates 1 through 19 is a uranium-aluminum-silicon matrix and is clad with aluminum, as shown in Figure 1. The uranium enrichment is nominally 15% high-enriched uranium metal and represents the midpoint of the 10–20% U-235 enrichment characteristic of the Template 9 fuel group. The uranium isotopic data are given below.

The following data provide specific fuel element dimensions, materials, densities, enrichment, etc., which are used to represent the hypothetical fuel element. The BOL data below were used in the BOL cross-section development and the burnup calculation for the source term generation.

Fuel Element:	Curved plate
No. of Plates:	19
Fueled Plates:	19
Fuel Plate Thickness:.	50 mils
XY dimensions:	2.820 in. by 3.194 in.
Length:	23.75 in.
Fuel Rod:	U-Al-Si (30% U, 68% Al, 2% Si)
Fuel Density:	3.616 g/cc
Fuel Meat Thickness:	21 mils
Uranium Enrichment:	0.60 wt % U-234
	15.00 wt % U-235
	0.35 wt % U-236
	84.05 wt % U-238
Heavy Metal Loading:	2.06 g/element U-234 (BOL)
	51.38 g/element U-235 (BOL)
	1.20 g/element U-236 (BOL)
	287.87 g/element U-238 (BOL)
	342.51 g/element Total U

Clad:	Aluminum 6061T
Clad Density:	2.70 g/cc
Clad Thickness:	14.5 mils
Side Plates:	Aluminum 6061T
Side Plate Width:	140 mils
Total Aluminum Mass:	4,064.13 g/element
Coolant/Moderator:	Heavy Water (D_2O)
Coolant Temperature:	52°C
Coolant Pressure:	175.3 psig
Coolant Density:	1.09325 g/cc

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single hypothetical fuel element. In addition, for the ORIGEN2 (Reference 2) depletion calculation, conservative and detailed impurity concentrations were added for the aluminum clad and structural components. Table 1 lists the Aluminum 6061T impurities and their concentrations (Reference 3).

Burnup

The burnup chosen for this template is 34.27% U-235 depletion or 15 MWd. Approximately 17.61 g U-235 were depleted for this single element. This burnup represents a medium range burnup for this element and its uranium loading.

For this analysis, the burnup period is assumed to be 1 year. The burnup and reactor power (approximately 0.041 MW) is further assumed to be constant and uniform over the 1-year period. At the end of the 1-year period, the element is assumed to be removed from the core, and the cooling or decay period begins. Table 2 gives the irradiation period and decay times following irradiation.

Cross-Section Development

The neutron cross sections used in the burnup or depletion calculation are based on the methodology described in Reference 1. Cross sections from a standard ORIGEN2 heavy water reactor library were updated once using the specially developed beginning-of-life (BOL) cross sections for the hypothetical fuel element. The updated cross sections take into account the unique neutron flux spatial and spectral characteristics to ensure accurate calculation of the fission product and actinide production as a function of burnup.

In order to calculate the BOL neutron cross sections, an explicit fuel assembly was developed with reflective boundary conditions on the element peripheral surfaces. The reflective surfaces simulated an infinite array of fuel elements. The fuel element model is shown in Figure 1.

Fuel Element Exposure History

Table 2 summarizes the power or exposure history used in the burnup or source term calculations for the single fuel element. Following the burnup or exposure period, the radionuclide activities are decayed for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100 years.

Burnup Calculation

The ORIGEN2 computer code (Reference 2) was used to perform the depletion or burnup calculation for the fuel element. The radionuclide inventory or source term template is for a single fuel element or assembly. The fuel element masses and impurities, neutron cross sections, burnup, power history, and power level as discussed above are input data for the ORIGEN2 calculation. The radionuclide concentrations are given as a function of time in the template table.

The 145 radionuclides listed in the template represent greater than 99.9% of the total curie inventory had all 684 activation products, 880 fission products, and 127 actinide/daughter isotopes from the ORIGEN2 output been included in the template.

References

1. J. W. Sterbentz and C. A. Wemple, *Calculational Burnup Methodology and Validation for the Idaho National Engineering Laboratory Spent Nuclear Fuels*, INEL-96/0304, September 1996.

2. A. G. Croff, *ORIGEN2—A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, July 1980.

3. ASTM B-209, Table 1, "Chemical Composition Limits," 1990.

Table 1. Aluminum-6061T material constituent and impurity concentrations.

Constituent or Impurity	Concentration (wt%)
H	0.02143
C	0.02143
O	0.02143
Mg	1.00000
Al	97.15499
Si	0.60000
Ti	0.07500
Cr	0.19500
Mn	0.07500
Fe	0.35000
Ni	0.02143
Cu	0.27500
Zn	0.12500
Zr	0.02143
Sn	0.02143
Pb	0.02143

Table 2. Assumed burnup or power history.

Condition	Time (years)	Decay Time (days)	Cumulative Duration (days)	Time-Averaged Power (MW_{th})
Irradiation	1	—	365.25	0.041
Decay	5	1825.00	2191.50	0.0
Decay	10	3652.50	4017.75	0.0
Decay	15	5478.75	5844.00	0.0
Decay	20	7305.00	7670.25	0.0
Decay	25	9131.25	9496.50	0.0
Decay	35	12783.75	13149.00	0.0
Decay	50	18262.50	18627.75	0.0
Decay	65	23741.25	24106.50	0.0
Decay	80	29220.00	29585.25	0.0
Decay	100	36525.00	36890.25	0.0

The bottom ten dates with zero associated power represent the ten different cooling or decay dates after exposure. These ten dates are specifically the 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100-year cooling times designated for the template methodology.

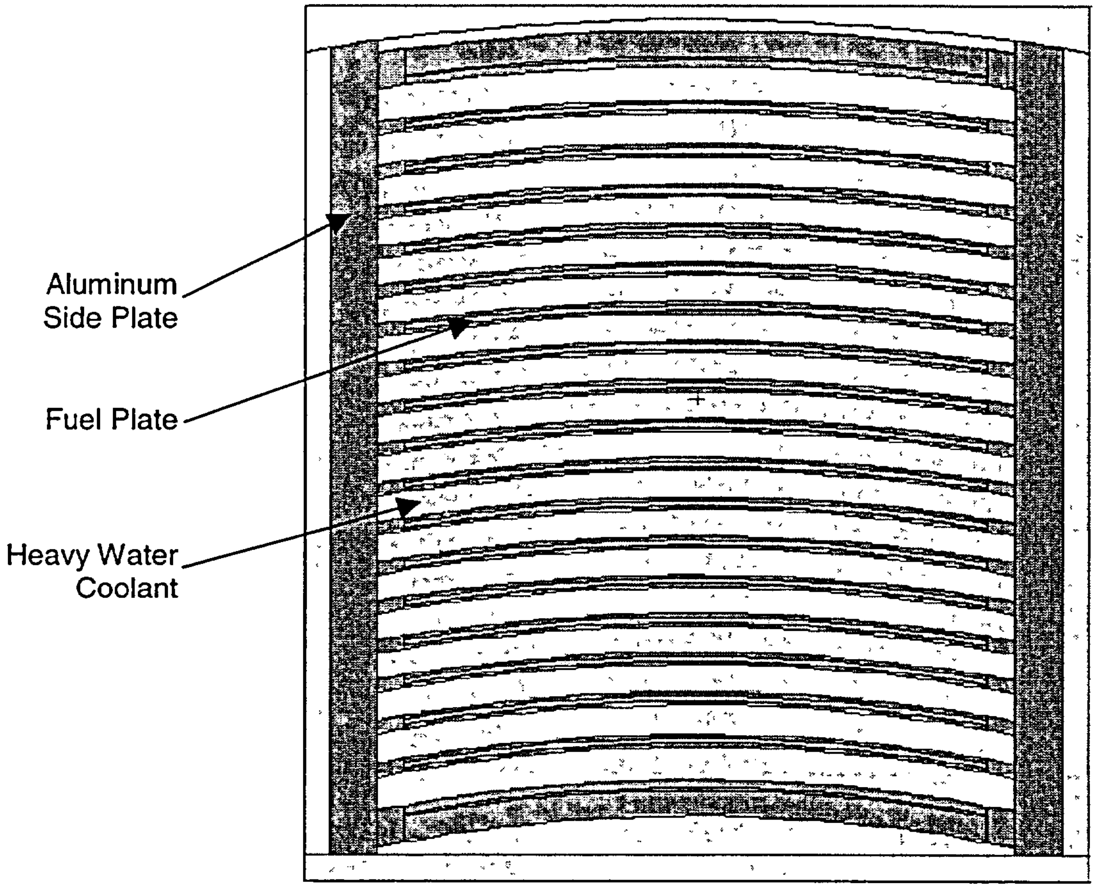

Figure 1. Curved-plate fuel assembly used in the analysis.

Representative Template 9 Reactor Element

Aluminum Cladding, 10 to 20% Enriched U-235 Fuel, Heavy Water Moderated Reactor

Reactor Moderator/Coolant:	Heavy Water
Fuel Meat:	U-Al-Si (30% U, 68% Al, 2% Si) in Aluminum
Clad:	Aluminum 6061T
Burnup:	17.61 g U-235 depleted
Burnup:	15 MWd/single element (high burnup)
Burnup:	34.27% U-235 depletion (fissioned and transmuted)
Basis of Calculation:	Single fuel element
BOL U-234:	2.06 g U-234 per element
BOL U-235:	51.38 g U-235 per element
BOL U-236:	1.20 g U-236 per element
BOL U-238:	287.87 g U-238 per element
BOL Total U per element:	342.51 g U per element
BOL Fuel Enrichment:	15 wt% U-235

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H 3	1.624E-01	1.227E-01	9.266E-02	6.998E-02	5.285E-02	3.015E-02	1.299E-02	5.597E-03	2.412E-03	7.850E-04
BE 10	1.421E-09	1.421E-09	1.421E-09	1.421E-09	1.421E-09	1.421E-09	1.421E-09	1.421E-09	1.421E-09	1.421E-09
C 14	4.451E-07	4.448E-07	4.446E-07	4.443E-07	4.440E-07	4.435E-07	4.427E-07	4.419E-07	4.411E-07	4.400E-07
CL 36	8.927E-34	8.927E-34	8.927E-34	8.927E-34	8.926E-34	8.926E-34	8.926E-34	8.926E-34	8.925E-34	8.925E-34
CR 51	9.101E-19	1.313E-38	1.894E-58	2.733E-78	3.943E-98	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54	9.224E-04	1.606E-05	2.796E-07	4.867E-09	8.473E-11	2.568E-14	1.355E-19	7.150E-25	3.772E-30	3.465E-37
FE 55	1.159E+00	3.056E-01	8.059E-02	2.125E-02	5.604E-03	3.897E-04	7.144E-06	1.310E-07	2.402E-09	1.161E-11
FE 59	3.117E-13	1.892E-25	1.149E-37	6.973E-50	4.233E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60	1.763E-03	9.134E-04	4.732E-04	2.451E-04	1.270E-04	3.408E-05	4.739E-06	6.589E-07	9.161E-08	6.599E-09
NI 59	2.094E-04	2.094E-04	2.094E-04	2.094E-04	2.094E-04	2.094E-04	2.094E-04	2.093E-04	2.093E-04	2.093E-04
NI 63	3.030E-02	2.918E-02	2.810E-02	2.707E-02	2.606E-02	2.417E-02	2.159E-02	1.928E-02	1.722E-02	1.481E-02
ZN 65	6.440E-02	3.585E-04	1.996E-06	1.111E-08	6.187E-11	1.918E-15	3.310E-22	5.712E-29	9.857E-36	9.471E-45
SE 79	1.881E-04	1.881E-04	1.881E-04	1.880E-04	1.880E-04	1.880E-04	1.880E-04	1.880E-04	1.879E-04	1.879E-04
KR 85	4.051E+00	2.932E+00	2.122E+00	1.536E+00	1.112E+00	5.822E-01	2.207E-01	8.369E-02	3.173E-02	8.706E-03
RB 87	1.253E-08	1.253E-08	1.253E-08	1.253E-08	1.253E-08	1.253E-08	1.253E-08	1.253E-08	1.253E-08	1.253E-08
SR 89	1.966E-08	2.552E-19	3.313E-30	4.299E-41	5.580E-52	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	3.944E+01	3.501E+01	3.108E+01	2.760E+01	2.450E+01	1.931E+01	1.351E+01	9.455E+00	6.616E+00	4.110E+00
Y 90	3.945E+01	3.502E+01	3.109E+01	2.760E+01	2.451E+01	1.931E+01	1.352E+01	9.458E+00	6.618E+00	4.111E+00
Y 91	7.487E-07	3.005E-16	1.206E-25	4.842E-35	1.944E-44	3.131E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
ZR 93	9.761E-04	9.761E-04	9.761E-04	9.761E-04	9.760E-04	9.760E-04	9.760E-04	9.760E-04	9.760E-04	9.760E-04
ZR 95	5.352E-06	1.368E-14	3.496E-23	8.938E-32	2.284E-40	1.493E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M	2.272E-04	3.846E-04	5.067E-04	6.013E-04	6.746E-04	7.755E-04	8.566E-04	8.944E-04	9.119E-04	9.217E-04
NB 94	2.237E-08	2.237E-08	2.237E-08	2.236E-08	2.236E-08	2.235E-08	2.234E-08	2.233E-08	2.232E-08	2.230E-08
NB 95	1.188E-05	3.037E-14	7.763E-23	1.984E-31	5.072E-40	3.314E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	3.970E-08	1.015E-16	2.594E-25	6.630E-34	1.695E-42	1.107E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TC 99	6.531E-03	6.531E-03	6.530E-03	6.530E-03	6.530E-03	6.530E-03	6.530E-03	6.529E-03	6.529E-03	6.529E-03
RU103	1.232E-11	1.246E-25	1.261E-39	1.276E-53	1.291E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	3.837E+00	1.232E-01	3.959E-03	1.272E-04	4.085E-06	4.214E-09	1.397E-13	4.629E-18	1.534E-22	1.633E-28
RH103M	1.111E-11	1.124E-25	1.137E-39	1.150E-53	1.163E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	3.837E+00	1.232E-01	3.959E-03	1.272E-04	4.085E-06	4.214E-09	1.397E-13	4.629E-18	1.534E-22	1.633E-28
PD107	1.398E-05	1.398E-05	1.398E-05	1.398E-05	1.398E-05	1.398E-05	1.398E-05	1.398E-05	1.398E-05	1.398E-05
AG110	9.694E-06	6.116E-08	3.858E-10	2.434E-12	1.536E-14	6.112E-19	1.535E-25	3.854E-32	9.679E-39	1.533E-47
AG110M	7.289E-04	4.598E-06	2.901E-08	1.830E-10	1.155E-12	4.596E-17	1.154E-23	2.898E-30	7.277E-37	1.153E-45
AG111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113M	4.962E-03	3.913E-03	3.086E-03	2.433E-03	1.919E-03	1.193E-03	5.850E-04	2.868E-04	1.407E-04	5.438E-05
CD113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD115M	2.306E-13	1.084E-25	5.098E-38	2.397E-50	1.127E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	4.232E-15	3.339E-26	2.633E-37	2.077E-48	1.637E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	4.423E-15	3.488E-26	2.752E-37	2.170E-48	1.712E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	1.621E-17	7.621E-30	3.583E-42	1.684E-54	7.919E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	1.834E-03	1.046E-05	5.969E-08	3.405E-10	1.943E-12	6.326E-17	1.175E-23	2.182E-30	4.054E-37	4.296E-46
SN121M	6.625E-05	6.181E-05	5.767E-05	5.381E-05	5.021E-05	4.370E-05	3.550E-05	2.882E-05	2.341E-05	1.774E-05
SN123	9.145E-05	5.070E-09	2.811E-13	1.558E-17	8.640E-22	2.656E-30	4.525E-43	7.711E-56	1.314E-68	1.241E-85
SN125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN126	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.708E-04	1.708E-04
SB124	6.877E-11	5.066E-20	3.733E-29	2.750E-38	2.027E-47	1.100E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	8.982E-01	2.570E-01	7.354E-02	2.105E-02	6.022E-03	4.931E-04	1.155E-05	2.707E-07	6.343E-09	4.252E-11
SB126	2.393E-05	2.393E-05	2.393E-05	2.393E-05	2.393E-05	2.393E-05	2.392E-05	2.392E-05	2.392E-05	2.392E-05
SB126M	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.709E-04	1.708E-04	1.708E-04
TE123M	2.852E-09	7.268E-14	1.853E-18	4.721E-23	1.203E-27	7.815E-37	1.293E-50	2.142E-64	3.545E-78	1.495E-96
TE125M	2.192E-01	6.271E-02	1.795E-02	5.134E-03	1.469E-03	1.203E-04	2.819E-06	6.604E-08	1.548E-09	1.038E-11
TE127	6.763E-05	6.120E-10	5.538E-15	5.011E-20	4.534E-25	3.712E-35	2.750E-50	2.038E-65	1.510E-80	1.012-100
TE127M	6.905E-05	6.248E-10	5.654E-15	5.116E-20	4.629E-25	3.790E-35	2.808E-50	2.080E-65	1.541E-80	1.033-100

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TE129	1.070E-15	4.647E-32	2.019E-48	8.774E-65	3.812E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	1.643E-15	7.140E-32	3.102E-48	1.348E-64	5.856E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	1.074E-05	1.074E-05	1.074E-05	1.074E-05	1.074E-05	1.074E-05	1.074E-05	1.074E-05	1.074E-05	1.074E-05
I131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS134	4.959E+00	9.235E-01	1.720E-01	3.203E-02	5.964E-03	2.068E-04	1.336E-06	8.626E-09	5.571E-11	6.700E-14
CS135	7.291E-05	7.291E-05	7.291E-05	7.291E-05	7.291E-05	7.291E-05	7.291E-05	7.291E-05	7.291E-05	7.291E-05
CS136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS137	4.291E+01	3.823E+01	3.406E+01	3.034E+01	2.703E+01	2.145E+01	1.517E+01	1.073E+01	7.585E+00	4.778E+00
BA136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA137M	4.059E+01	3.616E+01	3.222E+01	2.870E+01	2.557E+01	2.030E+01	1.435E+01	1.015E+01	7.176E+00	4.520E+00
BA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	2.430E-14	2.993E-31	3.686E-48	4.540E-65	5.591E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	1.333E-08	1.333E-08	1.333E-08	1.333E-08	1.333E-08	1.333E-08	1.333E-08	1.333E-08	1.333E-08	1.333E-08
CE144	1.248E+01	1.453E-01	1.691E-03	1.969E-05	2.292E-07	3.106E-11	4.900E-17	7.731E-23	1.220E-28	2.240E-36
PR143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144	1.248E+01	1.453E-01	1.691E-03	1.969E-05	2.292E-07	3.106E-11	4.900E-17	7.731E-23	1.220E-28	2.240E-36
PR144M	1.497E-01	1.743E-03	2.029E-05	2.362E-07	2.750E-09	3.727E-13	5.880E-19	9.277E-25	1.464E-30	2.688E-38
ND144	7.663E-13	7.709E-13	7.710E-13	7.710E-13	7.710E-13	7.710E-13	7.710E-13	7.710E-13	7.710E-13	7.710E-13
ND147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM147	3.655E+01	9.755E+00	2.603E+00	6.946E-01	1.854E-01	1.320E-02	2.508E-04	4.767E-06	9.058E-08	4.593E-10
PM148M	7.302E-13	3.556E-26	1.732E-39	8.434E-53	4.107E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	4.113E-14	2.003E-27	9.754E-41	4.750E-54	2.313E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM147	2.897E-09	3.554E-09	3.729E-09	3.776E-09	3.788E-09	3.793E-09	3.793E-09	3.793E-09	3.793E-09	3.793E-09
SM151	6.897E-02	6.636E-02	6.385E-02	6.144E-02	5.912E-02	5.474E-02	4.877E-02	4.345E-02	3.871E-02	3.318E-02
EU152	5.751E-04	4.457E-04	3.454E-04	2.677E-04	2.075E-04	1.247E-04	5.804E-05	2.702E-05	1.258E-05	4.540E-06
EU154	1.049E+00	7.014E-01	4.687E-01	3.133E-01	2.094E-01	9.351E-02	2.791E-02	8.333E-03	2.487E-03	4.962E-04
EU155	4.988E-01	2.480E-01	1.233E-01	6.130E-02	3.047E-02	7.532E-03	9.255E-04	1.137E-04	1.397E-05	8.537E-07
EU156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD153	1.590E-05	8.509E-08	4.553E-10	2.436E-12	1.303E-14	3.732E-19	5.716E-26	8.757E-33	1.342E-39	1.100E-48
TB160	1.073E-09	2.673E-17	6.661E-25	1.660E-32	4.136E-40	2.568E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TL206	1.333E-15	1.333E-15	1.333E-15	1.333E-15	1.333E-15	1.333E-15	1.333E-15	1.333E-15	1.333E-15	1.333E-15
TL207	8.136E-10	2.624E-09	5.305E-09	8.727E-09	1.278E-08	2.242E-08	3.968E-08	5.918E-08	8.005E-08	1.091E-07
TL208	8.648E-08	1.120E-07	1.165E-07	1.140E-07	1.096E-07	9.983E-08	8.643E-08	7.481E-08	6.475E-08	5.346E-08
PB210	4.649E-15	2.966E-14	1.119E-13	2.993E-13	6.497E-13	2.110E-12	7.340E-12	1.819E-11	3.692E-11	7.792E-11
PB211	8.158E-10	2.632E-09	5.320E-09	8.752E-09	1.282E-08	2.248E-08	3.979E-08	5.934E-08	8.027E-08	1.094E-07
PB212	2.407E-07	3.116E-07	3.241E-07	3.174E-07	3.051E-07	2.779E-07	2.406E-07	2.082E-07	1.802E-07	1.488E-07
BI211	8.158E-10	2.632E-09	5.320E-09	8.752E-09	1.282E-08	2.248E-08	3.979E-08	5.934E-08	8.027E-08	1.094E-07
BI212	2.407E-07	3.116E-07	3.241E-07	3.174E-07	3.051E-07	2.779E-07	2.406E-07	2.082E-07	1.802E-07	1.488E-07
PO212	1.542E-07	1.996E-07	2.077E-07	2.034E-07	1.955E-07	1.780E-07	1.541E-07	1.334E-07	1.155E-07	9.532E-08
PO215	8.158E-10	2.632E-09	5.320E-09	8.752E-09	1.282E-08	2.248E-08	3.979E-08	5.934E-08	8.027E-08	1.094E-07
PO216	2.407E-07	3.116E-07	3.241E-07	3.174E-07	3.051E-07	2.779E-07	2.406E-07	2.082E-07	1.802E-07	1.488E-07
RN219	8.158E-10	2.632E-09	5.320E-09	8.752E-09	1.282E-08	2.248E-08	3.979E-08	5.934E-08	8.027E-08	1.094E-07
RN220	2.407E-07	3.116E-07	3.241E-07	3.174E-07	3.051E-07	2.779E-07	2.406E-07	2.082E-07	1.802E-07	1.488E-07
FR223	1.126E-11	3.629E-11	7.334E-11	1.206E-10	1.766E-10	3.099E-10	5.487E-10	8.182E-10	1.107E-09	1.509E-09
RA223	8.158E-10	2.632E-09	5.320E-09	8.752E-09	1.282E-08	2.248E-08	3.979E-08	5.934E-08	8.027E-08	1.094E-07
RA224	2.407E-07	3.116E-07	3.241E-07	3.174E-07	3.051E-07	2.779E-07	2.406E-07	2.082E-07	1.802E-07	1.488E-07
RA226	6.467E-14	3.304E-13	9.237E-13	1.965E-12	3.571E-12	8.916E-12	2.380E-11	4.915E-11	8.719E-11	1.610E-10
RA228	1.264E-14	3.950E-14	7.493E-14	1.155E-13	1.591E-13	2.510E-13	3.934E-13	5.374E-13	6.817E-13	8.742E-13
AC227	8.158E-10	2.630E-09	5.315E-09	8.741E-09	1.280E-08	2.246E-08	3.976E-08	5.929E-08	8.019E-08	1.093E-07
TH227	8.046E-10	2.595E-09	5.246E-09	8.631E-09	1.264E-08	2.217E-08	3.925E-08	5.853E-08	7.917E-08	1.079E-07
TH228	2.407E-07	3.114E-07	3.239E-07	3.171E-07	3.048E-07	2.778E-07	2.405E-07	2.082E-07	1.802E-07	1.488E-07
TH229	9.131E-13	2.048E-12	3.746E-12	6.009E-12	8.842E-12	1.624E-11	3.175E-11	5.272E-11	7.934E-11	1.240E-10
TH230	6.632E-11	1.891E-10	3.692E-10	6.044E-10	8.925E-10	1.619E-09	3.058E-09	4.873E-09	7.023E-09	1.035E-08
TH231	7.302E-05	7.302E-05	7.302E-05	7.302E-05	7.303E-05	7.303E-05	7.303E-05	7.303E-05	7.303E-05	7.304E-05
TH232	5.304E-14	1.011E-13	1.492E-13	1.973E-13	2.454E-13	3.416E-13	4.859E-13	6.303E-13	7.747E-13	9.672E-13
TH234	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05
PA231	9.174E-09	1.690E-08	2.463E-08	3.236E-08	4.009E-08	5.553E-08	7.869E-08	1.018E-07	1.250E-07	1.558E-07
PA233	5.425E-05	5.449E-05	5.486E-05	5.531E-05	5.584E-05	5.703E-05	5.904E-05	6.116E-05	6.331E-05	6.614E-05
PA234M	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05
PA234	1.255E-07	1.255E-07	1.255E-07	1.255E-07	1.255E-07	1.255E-07	1.255E-07	1.255E-07	1.255E-07	1.255E-07
U232	3.095E-07	3.289E-07	3.234E-07	3.112E-07	2.975E-07	2.705E-07	2.342E-07	2.027E-07	1.754E-07	1.447E-07
U233	1.799E-09	2.988E-09	4.183E-09	5.387E-09	6.602E-09	9.068E-09	1.287E-08	1.681E-08	2.089E-08	2.655E-08
U234	2.075E-06	3.373E-06	4.621E-06	5.821E-06	6.975E-06	9.150E-06	1.211E-05	1.474E-05	1.707E-05	1.978E-05
U235	7.302E-05	7.302E-05	7.302E-05	7.302E-05	7.303E-05	7.303E-05	7.303E-05	7.303E-05	7.303E-05	7.304E-05
U236	1.950E-04	1.950E-04	1.950E-04	1.950E-04	1.950E-04	1.950E-04	1.951E-04	1.951E-04	1.951E-04	1.952E-04

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
U237	2.847E-06	2.238E-06	1.759E-06	1.383E-06	1.087E-06	6.716E-07	3.262E-07	1.585E-07	7.698E-08	2.939E-08
U238	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05	9.654E-05
NP237	5.425E-05	5.449E-05	5.486E-05	5.531E-05	5.584E-05	5.703E-05	5.904E-05	6.116E-05	6.331E-05	6.614E-05
PU236	1.251E-06	3.710E-07	1.100E-07	3.263E-08	9.681E-09	8.600E-10	3.178E-11	1.019E-11	9.625E-12	9.609E-12
PU237	1.460E-17	1.283E-29	1.128E-41	9.914E-54	8.713E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	9.332E-02	8.971E-02	8.624E-02	8.291E-02	7.970E-02	7.366E-02	6.544E-02	5.813E-02	5.165E-02	4.411E-02
PU239	1.548E-01	1.548E-01	1.548E-01	1.547E-01	1.547E-01	1.547E-01	1.546E-01	1.545E-01	1.545E-01	1.544E-01
PU240	8.139E-02	8.135E-02	8.131E-02	8.127E-02	8.123E-02	8.114E-02	8.101E-02	8.088E-02	8.076E-02	8.059E-02
PU241	1.160E+01	9.121E+00	7.170E+00	5.636E+00	4.431E+00	2.738E+00	1.330E+00	6.460E-01	3.138E-01	1.198E-01
PU242	4.607E-05	4.607E-05	4.607E-05	4.607E-05	4.607E-05	4.607E-05	4.607E-05	4.607E-05	4.607E-05	4.607E-05
PU244	1.680E-12	1.680E-12	1.680E-12	1.680E-12	1.680E-12	1.680E-12	1.680E-12	1.680E-12	1.680E-12	1.680E-12
AM241	1.103E-01	1.917E-01	2.549E-01	3.038E-01	3.413E-01	3.918E-01	4.288E-01	4.411E-01	4.415E-01	4.339E-01
AM242M	1.432E-04	1.400E-04	1.368E-04	1.337E-04	1.307E-04	1.249E-04	1.166E-04	1.089E-04	1.017E-04	9.284E-05
AM242	1.425E-04	1.393E-04	1.361E-04	1.330E-04	1.300E-04	1.242E-04	1.160E-04	1.084E-04	1.012E-04	9.238E-05
AM243	9.615E-05	9.610E-05	9.606E-05	9.601E-05	9.597E-05	9.588E-05	9.574E-05	9.561E-05	9.547E-05	9.529E-05
CM242	3.659E-04	1.153E-04	1.126E-04	1.101E-04	1.076E-04	1.028E-04	9.596E-05	8.962E-05	8.369E-05	7.640E-05
CM243	4.771E-05	4.225E-05	3.741E-05	3.313E-05	2.934E-05	2.300E-05	1.597E-05	1.109E-05	7.700E-06	4.734E-06
CM244	2.931E-03	2.421E-03	1.999E-03	1.651E-03	1.363E-03	9.297E-04	5.236E-04	2.949E-04	1.661E-04	7.725E-05
CM245	9.205E-08	9.201E-08	9.198E-08	9.194E-08	9.190E-08	9.183E-08	9.171E-08	9.160E-08	9.149E-08	9.134E-08
CM246	4.760E-09	4.757E-09	4.753E-09	4.750E-09	4.746E-09	4.739E-09	4.729E-09	4.718E-09	4.708E-09	4.694E-09
CM247	3.521E-15	3.521E-15	3.521E-15	3.521E-15	3.521E-15	3.521E-15	3.521E-15	3.521E-15	3.521E-15	3.521E-15
Subtotal**	2.569E+02	1.700E+02	1.421E+02	1.234E+02	1.084E+02	8.463E+01	5.895E+01	4.135E+01	2.914E+01	1.842E+01
TOTAL***	2.569E+02	1.700E+02	1.421E+02	1.234E+02	1.084E+02	8.463E+01	5.895E+01	4.134E+01	2.914E+01	1.842E+01

* Four decimal places of accuracy are as reported by ORIGEN2 output and are not significant for many radionuclides.
** Subtotal: total activity of the 145 isotopes listed in the table.
*** Total: total activity of the ORIGEN2 output isotopes.

Template 10

Representative Fuel Source Term Calculations

Introduction

The following data have been used in the Idaho National Engineering and Environmental Laboratory (INEEL) spent nuclear fuel source term calculational methodology to generate a source term template to represent a stainless steel clad, 0-5% enriched, uranium-based fuel from a heavy water-moderated reactor. Because the spent fuels in this group are primarily MTR-type or plate-fuels, the previously constructed HFBR fuel element geometry model (Template 8) was modified to represent fuels in this group. Modifications included developing new cross sections and adjusting enrichment and burnup. Other differences include the fuel meat material and the number of fueled plates in the element. These data are included below. The calculation methodology used is described in detail in Reference 1.

Reactor Data

The hypothetical element is a plate-type element consisting of 19 curved plates. The plates are stacked, separated by a heavy water gap (102–129 mils), and held together as a rectangular structure by two stainless steel side plates (140 mils thick). The fuel meat in the plates is a uranium-aluminum-silicon matrix and is clad with stainless steel, as shown in Figure 1. The uranium enrichment is nominally 5% enriched uranium metal and represents the upper end of the 0–5% U-235 enrichment characteristic of the Template 10 fuel group.

The following data provide the specific fuel element dimensions, materials, densities, enrichment, etc. The beginning-of-life (BOL) data below were used in the BOL cross-section development and the burnup calculation for the source term generation.

Fuel Element:	Curved plate
No. of Plates:	19
Fueled Plates:	19
Fuel Plate Thickness:	50 mils
Cross Sectional Dimensions:	2.820 in. by 3.194 in.
Length:	23.75 in.
Fuel Rod:	U-Al-Si (30% U, 68% Al, 2% Si)
Fuel Density:	3.616 g/cc
Fuel Meat Thickness:	21 mils
Uranium Enrichment:	0.60 wt % U-234
	5.00 wt % U-235
	0.35 wt % U-236
	94.05 wt % U-238
Heavy Metal Loading:	2.06 g/element U-234 (BOL)
	17.13 g/element U-235 (BOL)
	1.20 g/element U-236 (BOL)
	322.12 g/element U-238 (BOL)
	342.51 g/element Total U
	776.33 g/element Aluminum-6061 (Fuel Meat)
	22.83 g/element Silicon (Fuel Meat)

Clad:	Stainless Steel-304
Clad Density:	8.02 g/cc
Clad Thickness:	14.5 mils
Side Plates:	Stainless Steel-304
Side Plate Width:	140 mils
Total Stainless Steel-304 Mass:	9,765.95 g/element

Coolant/Moderator:	Heavy Water (D_2O)
Coolant Temperature:	52°C
Coolant Pressure:	175.3 psig
Coolant Density:	1.09325 g/cc

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single fuel element. In addition, for the ORIGEN2 (Reference 2) depletion calculation, conservative and detailed impurity concentrations were added for the stainless steel clad and aluminum/silicon fuel meat constituents. Tables 1 and 2 list the Stainless Steel-304 and Aluminum 6061T impurities and their concentrations, respectively, according to References 3, 4, 5 and 6.

Burnup

The burnup chosen for this template is 23.1% U-235 depletion, 5.0 MWd, and approximately 3.96 grams of U-235 depleted for this single element. This burnup represents a nominal range burnup one might expect for this element and uranium loading.

For the analysis, the burnup period is assumed to be 1 year. The burnup and reactor power (approximately 0.0137 MW) is further assumed to be constant and uniform over the 1-year period. At the end of the 1-year period, the element is removed from the core, and the cooling or decay period begins. Table 3 gives the irradiation period and decay times following irradiation.

Cross-Section Development

The neutron cross sections used in the burnup or depletion calculation are based on the methodology described in Reference 1. Cross sections from a standard ORIGEN2 heavy water reactor library were updated once using the BOL cross sections developed for the hypothetical fuel element. These updated cross sections take into account the neutron flux spatial and spectral characteristics to ensure accurate calculation of the actinide production as a function of burnup.

In order to calculate the BOL neutron cross sections, an explicit fuel element was developed with reflective boundary conditions on the element peripheral surfaces. The reflective surfaces simulated an infinite array of fuel elements. The fuel element model is shown in Figure 1.

Fuel Element Exposure History

Table 3 summarizes the power or exposure history used in the burnup or source term calculations for a single fuel element. Following the burnup or exposure period, the radionuclide activities are decayed for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100 years.

Burnup Calculation

The ORIGEN2 computer code (Reference 2) was used to perform the depletion or burnup calculation for a single fuel element. The fuel element masses and impurities, neutron cross sections, burnup, power history, and power level discussed above are input data for the ORIGEN2 calculation. The resulting radionuclide concentrations are given as a function of time in the template table.

The 145 radionuclides listed in the template represent greater than 99.9% of the total curie inventory had all 684 activation products, 880 fission products, and 127 actinide/daughter isotopes from the ORIGEN2 output been included in the template.

References

1. J. W. Sterbentz and C. A. Wemple, *Calculational Burnup Methodology and Validation for the Idaho National Engineering Laboratory Spent Nuclear Fuels*, INEL-96/0304, September 1996.

2. A. G. Croff, *ORIGEN2—A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, July 1980.

3. J. C. Evans et al., "Long-Lived Activation Products in Reactor Materials," NUREG/CR-3474, Prepared for the U.S. Nuclear Regulatory Commission by Battelle, Pacific Northwest Laboratory, Richland, WA, August 1984.

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5. F.W. Walker et al., "Nuclides and Isotopes: Chart of the Nuclides," General Electric Co., 1989.

6. John Logan, INEEL, to Tom Clements, Appendix data in a letter, "Assessment of Neutron-Activation Products in Low Level Waste Discharged from Nuclear Reactors at the Test Reactor Area; and Sent to the Radioactive Waste Management Complex for Disposal," JAL-04-99, September 9, 1999.

Table 1. Stainless Steel-304 material constituent and impurity concentrations.

Constituent or Impurity	Impurity (ppm)	Weight Fraction (wt%)
H		0.0007
Li	0.13	
B		0.0005
C		0.07
N		0.047
O		0.015
Na	37	
Al		0.01
Si		0.6
P		0.0375
S		0.02
Cl	130	
K	3	
Ca	19	
Sc	0.03	
Ti		0.05
V		0.05
Cr		18.8
Mn		1.41
Fe		68.8
Co		0.17
Ni		9.23
Cu		0.25
Zn		0.01
Ga	450	
As		0.01
Se		0.02
Br	8	
Rb	10	
Sr	0.2	
Y	5	
Zr	20	
Nb		0.012
Mo		0.37

Constituent or Impurity	Impurity (ppm)	Weight Fraction (wt%)
Ag	2	
Sn		0.01
Sb		0.01
Cs	0.3	
Ba	500	
La	2.1	
Ce	550	
Sm	0.15	
Eu	0.02	
Tb	0.71	
Dy	1	
Ho	1	
Yb	2	
Lu	0.8	
Hf	2	
W	520	
Pb		0.002
Th	1	
U	2	

Table 2. Aluminum-6061T material constituent and impurity concentrations.

Constituent or Impurity	Weight Fraction (wt%)
H	
Li	0.0005
B	0.022
C	0.02
N	0.0005
O	0.05
Na	0.00002
Mg	0.9
Al	97.39387
Si	0.65
P	0.001
S	0.002
Ti	0.02
V	0.02
Cr	0.05
Mn	0.03
Fe	0.2
Co	0.05
Ni	0.04
Cu	0.25
Zn	0.02
Ga	0.05
Sr	0.00001
Zr	0.02
Nb	0.01
Mo	0.0001
Cd	0.05
Sn	0.02
Sb	0.01
Hf	0.05
Ta	0.05
Pb	0.02

Table 3. Assumed burnup or power history for a single hypothetical fuel element.

Condition	Time (years)	Decay Time (days)	Cumulative Duration (days)	Time-Averaged Power (MW_{th})
Irradiation	1	—	365.25	0.0137
Decay	5	1825.00	2191.50	0.0
Decay	10	3652.50	4017.75	0.0
Decay	15	5478.75	5844.00	0.0
Decay	20	7305.00	7670.25	0.0
Decay	25	9131.25	9496.50	0.0
Decay	35	12783.75	13149.00	0.0
Decay	50	18262.50	18627.75	0.0
Decay	65	23741.25	24106.50	0.0
Decay	80	29220.00	29585.25	0.0
Decay	100	36525.00	36890.25	0.0

The dates with zero associated power represent the ten different cooling or decay dates after exposure. These ten dates are specifically the 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100-year cooling times designated for the template methodology.

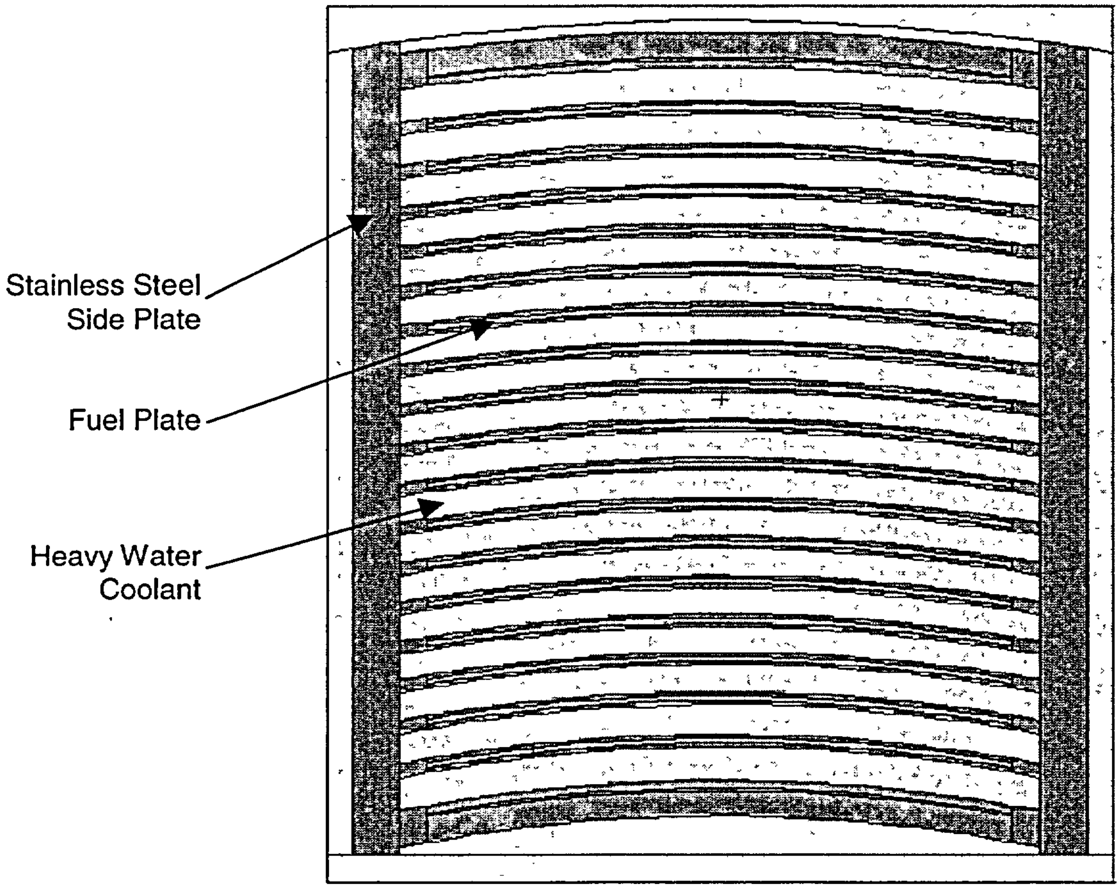

Figure 1. Curved-plate fuel element used in the analysis.

Template 10

Stainless Steel Cladding, 0 to 5% Enriched U-235 Fuel

Reactor Moderator/Coolant:	Heavy Water
Fuel Meat:	U-Al-Si (30% U, 68% Al, 2%Si) in Aluminum
Clad:	Stainless Steel-304
Burnup:	3.96 g U-235 depleted
Burnup:	5 MWd/single element
Burnup:	23.1% U-235 depletion (fissioned and transmuted)
Basis of Calculation:	Single fuel element
BOL U-235:	17.13 g U-235 per element
BOL U-238:	322.12 g U-238 per element
BOL U-234:	2.06 g U-234 per element
BOL U-236:	1.20 g U-236 per element
BOL Total U per element:	342.51 g U per element
BOL Fuel Enrichment:	5 wt% U-235

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H 3	1.199E+00	9.062E-01	6.844E-01	5.170E-01	3.904E-01	2.228E-01	9.597E-02	4.135E-02	1.782E-02	5.797E-03
BE 10	1.766E-09	1.766E-09	1.766E-09	1.766E-09	1.766E-09	1.766E-09	1.766E-09	1.766E-09	1.766E-09	1.766E-09
C 14	1.112E-01	1.111E-01	1.110E-01	1.110E-01	1.109E-01	1.108E-01	1.106E-01	1.104E-01	1.102E-01	1.099E-01
CL 36	2.095E-03	2.095E-03	2.095E-03	2.095E-03	2.094E-03	2.094E-03	2.094E-03	2.094E-03	2.094E-03	2.094E-03
CR 51	2.423E-16	3.496E-36	5.044E-56	7.277E-76	1.050E-95	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54	6.151E-01	1.071E-02	1.864E-04	3.245E-06	5.650E-08	1.712E-11	9.035E-17	4.767E-22	2.515E-27	2.311E-34
FE 55	7.991E+02	2.107E+02	5.556E+01	1.465E+01	3.863E+00	2.686E-01	4.926E-03	9.031E-05	1.656E-06	8.006E-09
FE 59	1.781E-10	1.081E-22	6.562E-35	3.983E-47	2.418E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60	1.332E+03	6.898E+02	3.574E+02	1.851E+02	9.592E+01	2.574E+01	3.579E+00	4.976E-01	6.919E-02	4.984E-03
NI 59	3.123E-01	3.123E-01	3.123E-01	3.123E-01	3.123E-01	3.123E-01	3.122E-01	3.122E-01	3.121E-01	3.121E-01
NI 63	4.493E+01	4.327E+01	4.167E+01	4.013E+01	3.865E+01	3.584E+01	3.201E+01	2.859E+01	2.554E+01	2.197E+01
ZN 65	2.108E-02	1.173E-04	6.533E-07	3.637E-09	2.025E-11	6.276E-16	1.083E-22	1.869E-29	3.226E-36	3.100E-45
SE 79	1.425E-04	1.425E-04	1.425E-04	1.425E-04	1.425E-04	1.425E-04	1.424E-04	1.424E-04	1.424E-04	1.424E-04
KR 85	1.126E+00	8.153E-01	5.901E-01	4.271E-01	3.091E-01	1.619E-01	6.138E-02	2.327E-02	8.823E-03	2.421E-03
RB 87	5.801E-09	5.801E-09	5.801E-09	5.801E-09	5.801E-09	5.801E-09	5.801E-09	5.801E-09	5.801E-09	5.801E-09
SR 89	4.876E-09	6.329E-20	8.215E-31	1.066E-41	1.384E-52	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	1.059E+01	9.402E+00	8.347E+00	7.411E+00	6.579E+00	5.186E+00	3.629E+00	2.539E+00	1.777E+00	1.104E+00
Y 90	1.059E+01	9.405E+00	8.349E+00	7.413E+00	6.581E+00	5.187E+00	3.630E+00	2.540E+00	1.777E+00	1.104E+00
Y 91	1.935E-07	7.769E-17	3.118E-26	1.252E-35	5.024E-45	8.095E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93	2.853E-04	2.853E-04	2.853E-04	2.853E-04	2.853E-04	2.853E-04	2.853E-04	2.853E-04	2.853E-04	2.853E-04

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	**5**	**10**	**15**	**20**	**25**	**35**	**50**	**65**	**80**	**100**
ZR 95	1.615E-06	4.127E-15	1.055E-23	2.697E-32	6.894E-41	4.504E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M	6.681E-05	1.128E-04	1.484E-04	1.760E-04	1.974E-04	2.268E-04	2.505E-04	2.615E-04	2.666E-04	2.694E-04
NB 94	9.969E-04	9.967E-04	9.966E-04	9.964E-04	9.962E-04	9.959E-04	9.954E-04	9.949E-04	9.944E-04	9.937E-04
NB 95	3.585E-06	9.164E-15	2.342E-23	5.988E-32	1.530E-40	9.999E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	1.198E-08	3.062E-17	7.827E-26	2.001E-34	5.113E-43	3.341E-60	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93	7.223E-04	7.216E-04	7.209E-04	7.202E-04	7.195E-04	7.180E-04	7.159E-04	7.138E-04	7.117E-04	7.088E-04
TC 99	2.168E-03	2.168E-03	2.168E-03	2.168E-03	2.168E-03	2.168E-03	2.168E-03	2.168E-03	2.167E-03	2.167E-03
RU103	5.649E-12	5.715E-26	5.782E-40	5.849E-54	5.917E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	3.370E+00	1.082E-01	3.477E-03	1.117E-04	3.587E-06	3.701E-09	1.227E-13	4.065E-18	1.347E-22	1.434E-28
RH103M	5.093E-12	5.152E-26	5.212E-40	5.273E-54	5.334E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	3.370E+00	1.082E-01	3.477E-03	1.117E-04	3.587E-06	3.701E-09	1.227E-13	4.065E-18	1.347E-22	1.434E-28
PD107	1.485E-05	1.485E-05	1.485E-05	1.485E-05	1.485E-05	1.485E-05	1.485E-05	1.485E-05	1.485E-05	1.485E-05
AG110	3.771E-05	2.379E-07	1.501E-09	9.469E-12	5.974E-14	2.377E-18	5.971E-25	1.499E-31	3.766E-38	5.965E-47
AG110M	2.835E-03	1.788E-05	1.129E-07	7.119E-10	4.492E-12	1.788E-16	4.490E-23	1.127E-29	2.831E-36	4.485E-45
AG111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD113M	3.248E-03	2.561E-03	2.020E-03	1.593E-03	1.256E-03	7.810E-04	3.829E-04	1.878E-04	9.207E-05	3.560E-05
CD113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD115M	2.192E-13	1.031E-25	4.845E-38	2.278E-50	1.071E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	9.470E-15	7.470E-26	5.892E-37	4.646E-48	3.665E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	9.895E-15	7.805E-26	6.156E-37	4.855E-48	3.830E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	1.017E-17	4.783E-30	2.248E-42	1.057E-54	4.970E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	2.970E-03	1.695E-05	9.670E-08	5.516E-10	3.148E-12	1.025E-16	1.904E-23	3.535E-30	6.567E-37	6.960E-46
SN121M	7.795E-05	7.273E-05	6.785E-05	6.331E-05	5.906E-05	5.141E-05	4.175E-05	3.391E-05	2.754E-05	2.087E-05
SN123	4.829E-05	2.677E-09	1.485E-13	8.228E-18	4.562E-22	1.402E-30	2.389E-43	4.071E-56	6.937E-69	6.554E-86
SN125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN126	8.899E-05	8.898E-05	8.898E-05	8.898E-05	8.897E-05	8.897E-05	8.896E-05	8.895E-05	8.894E-05	8.893E-05
SB124	3.051E-08	2.248E-17	1.656E-26	1.221E-35	8.994E-45	4.882E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	5.609E-01	1.605E-01	4.593E-02	1.314E-02	3.761E-03	3.080E-04	7.216E-06	1.691E-07	3.962E-09	2.656E-11
SB126	1.246E-05	1.246E-05	1.246E-05	1.246E-05	1.246E-05	1.246E-05	1.245E-05	1.245E-05	1.245E-05	1.245E-05
SB126M	8.899E-05	8.898E-05	8.898E-05	8.898E-05	8.897E-05	8.897E-05	8.896E-05	8.895E-05	8.894E-05	8.893E-05
TE123M	5.949E-06	1.517E-10	3.864E-15	9.849E-20	2.510E-24	1.631E-33	2.698E-47	4.467E-61	7.394E-75	3.119E-93
TE125M	1.368E-01	3.916E-02	1.120E-02	3.206E-03	9.177E-04	7.514E-05	1.760E-06	4.125E-08	9.665E-10	6.480E-12
TE127	3.762E-05	3.404E-10	3.080E-15	2.787E-20	2.522E-25	2.065E-35	1.530E-50	1.133E-65	8.396E-81	0.000E+00
TE127M	3.840E-05	3.475E-10	3.144E-15	2.845E-20	2.575E-25	2.108E-35	1.562E-50	1.157E-65	8.572E-81	0.000E+00
TE129	4.909E-16	2.133E-32	9.268E-49	4.027E-65	1.750E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TE129M	7.542E-16	3.277E-32	1.424E-48	6.187E-65	2.688E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	4.583E-06	4.583E-06	4.583E-06	4.583E-06	4.583E-06	4.583E-06	4.583E-06	4.583E-06	4.583E-06	4.583E-06
I131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS134	2.074E+00	3.863E-01	7.193E-02	1.339E-02	2.494E-03	8.650E-05	5.586E-07	3.608E-09	2.330E-11	2.802E-14
CS135	1.874E-05	1.874E-05	1.874E-05	1.874E-05	1.873E-05	1.873E-05	1.873E-05	1.873E-05	1.873E-05	1.873E-05
CS136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS137	1.443E+01	1.285E+01	1.145E+01	1.020E+01	9.088E+00	7.213E+00	5.100E+00	3.607E+00	2.550E+00	1.606E+00
BA136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA137M	1.365E+01	1.216E+01	1.083E+01	9.650E+00	8.597E+00	6.824E+00	4.825E+00	3.412E+00	2.412E+00	1.520E+00
BA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE141	8.081E-15	9.952E-32	1.225E-48	1.509E-65	1.859E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	1.871E-08	1.871E-08	1.871E-08	1.871E-08	1.871E-08	1.871E-08	1.871E-08	1.871E-08	1.871E-08	1.871E-08
CE144	3.740E+00	4.354E-02	5.069E-04	5.901E-06	6.869E-08	9.309E-12	1.469E-17	2.317E-23	3.656E-29	6.714E-37
PR143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR144	3.740E+00	4.354E-02	5.069E-04	5.901E-06	6.869E-08	9.310E-12	1.469E-17	2.317E-23	3.656E-29	6.714E-37
PR144M	4.488E-02	5.225E-04	6.083E-06	7.081E-08	8.243E-10	1.117E-13	1.762E-19	2.781E-25	4.387E-31	8.057E-39
ND144	2.453E-13	2.466E-13	2.466E-13	2.466E-13	2.466E-13	2.466E-13	2.466E-13	2.466E-13	2.466E-13	2.466E-13
ND147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM145	5.249E-06	4.403E-06	3.622E-06	2.978E-06	2.449E-06	1.655E-06	9.200E-07	5.113E-07	2.842E-07	1.299E-07
PM147	1.092E+01	2.915E+00	7.779E-01	2.076E-01	5.539E-02	3.945E-03	7.496E-05	1.424E-06	2.707E-08	1.373E-10
PM148M	2.218E-13	1.080E-26	5.260E-40	2.562E-53	1.247E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	1.249E-14	6.084E-28	2.963E-41	1.443E-54	7.026E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	1.976E-06	4.777E-08	1.155E-09	2.791E-11	6.745E-13	3.941E-16	5.565E-21	7.858E-26	1.110E-30	3.787E-37
SM147	8.673E-10	1.064E-09	1.116E-09	1.130E-09	1.134E-09	1.135E-09	1.135E-09	1.135E-09	1.135E-09	1.135E-09
SM151	2.626E-02	2.527E-02	2.432E-02	2.339E-02	2.251E-02	2.084E-02	1.857E-02	1.654E-02	1.474E-02	1.263E-02
EU152	1.824E-04	1.414E-04	1.095E-04	8.489E-05	6.579E-05	3.952E-05	1.841E-05	8.567E-06	3.989E-06	1.439E-06
EU154	5.170E-01	3.455E-01	2.309E-01	1.543E-01	1.031E-01	4.607E-02	1.375E-02	4.106E-03	1.226E-03	2.445E-04
EU155	2.694E-01	1.339E-01	6.659E-02	3.310E-02	1.646E-02	4.068E-03	4.999E-04	6.142E-05	7.547E-06	4.610E-07
EU156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD153	1.054E-04	5.644E-07	3.020E-09	1.616E-11	8.645E-14	2.475E-18	3.792E-25	5.809E-32	8.897E-39	7.294E-48
TB160	3.921E-08	9.770E-16	2.434E-23	6.066E-31	1.512E-38	9.387E-54	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	1.179E-15	1.179E-15	1.179E-15	1.179E-15	1.179E-15	1.179E-15	1.179E-15	1.179E-15	1.179E-15	1.179E-15

Radionuclide	DECAY TIMES (years out of core) (Activities* in Ci/element) 5	10	15	20	25	35	50	65	80	100
TL207	2.839E-09	5.552E-09	8.309E-09	1.110E-08	1.393E-08	1.965E-08	2.837E-08	3.721E-08	4.611E-08	5.803E-08
TL208	7.898E-07	9.137E-07	9.109E-07	8.788E-07	8.405E-07	7.639E-07	6.612E-07	5.724E-07	4.954E-07	4.091E-07
PB210	4.527E-11	2.678E-10	7.935E-10	1.730E-09	3.168E-09	7.854E-09	2.026E-08	4.008E-08	6.802E-08	1.186E-07
PB211	2.847E-09	5.567E-09	8.332E-09	1.113E-08	1.397E-08	1.970E-08	2.845E-08	3.731E-08	4.624E-08	5.820E-08
PB212	2.198E-06	2.543E-06	2.535E-06	2.446E-06	2.339E-06	2.126E-06	1.840E-06	1.593E-06	1.379E-06	1.139E-06
BI211	2.847E-09	5.567E-09	8.332E-09	1.113E-08	1.397E-08	1.970E-08	2.845E-08	3.731E-08	4.624E-08	5.820E-08
BI212	2.198E-06	2.543E-06	2.535E-06	2.446E-06	2.339E-06	2.126E-06	1.840E-06	1.593E-06	1.379E-06	1.139E-06
PO212	1.408E-06	1.629E-06	1.624E-06	1.567E-06	1.499E-06	1.362E-06	1.179E-06	1.021E-06	8.835E-07	7.295E-07
PO215	2.847E-09	5.567E-09	8.332E-09	1.113E-08	1.397E-08	1.970E-08	2.845E-08	3.731E-08	4.624E-08	5.820E-08
PO216	2.198E-06	2.543E-06	2.535E-06	2.446E-06	2.339E-06	2.126E-06	1.840E-06	1.593E-06	1.379E-06	1.139E-06
RN219	2.847E-09	5.567E-09	8.332E-09	1.113E-08	1.397E-08	1.970E-08	2.845E-08	3.731E-08	4.624E-08	5.820E-08
RN220	2.198E-06	2.543E-06	2.535E-06	2.446E-06	2.339E-06	2.126E-06	1.840E-06	1.593E-06	1.379E-06	1.139E-06
FR223	3.928E-11	7.675E-11	1.148E-10	1.534E-10	1.924E-10	2.716E-10	3.922E-10	5.143E-10	6.374E-10	8.023E-10
RA223	2.847E-09	5.567E-09	8.332E-09	1.113E-08	1.397E-08	1.970E-08	2.845E-08	3.731E-08	4.624E-08	5.820E-08
RA224	2.198E-06	2.543E-06	2.535E-06	2.446E-06	2.339E-06	2.126E-06	1.840E-06	1.593E-06	1.379E-06	1.139E-06
RA226	7.623E-10	2.555E-09	5.399E-09	9.292E-09	1.423E-08	2.724E-08	5.456E-08	9.118E-08	1.371E-07	2.125E-07
RA228	4.820E-10	7.091E-10	8.445E-10	9.252E-10	9.734E-10	1.019E-09	1.039E-09	1.043E-09	1.044E-09	1.045E-09
AC227	2.846E-09	5.562E-09	8.321E-09	1.112E-08	1.395E-08	1.968E-08	2.842E-08	3.727E-08	4.619E-08	5.814E-08
TH227	2.808E-09	5.491E-09	8.217E-09	1.098E-08	1.378E-08	1.943E-08	2.806E-08	3.680E-08	4.560E-08	5.739E-08
TH228	2.198E-06	2.541E-06	2.533E-06	2.444E-06	2.337E-06	2.126E-06	1.840E-06	1.593E-06	1.379E-06	1.139E-06
TH229	1.052E-09	2.016E-09	2.981E-09	3.946E-09	4.911E-09	6.843E-09	9.745E-09	1.265E-08	1.556E-08	1.946E-08
TH230	5.856E-07	1.073E-06	1.560E-06	2.048E-06	2.536E-06	3.511E-06	4.976E-06	6.440E-06	7.906E-06	9.860E-06
TH231	2.848E-05	2.848E-05	2.848E-05	2.849E-05	2.849E-05	2.849E-05	2.850E-05	2.850E-05	2.851E-05	2.851E-05
TH232	1.044E-09	1.044E-09	1.044E-09	1.044E-09	1.044E-09	1.044E-09	1.045E-09	1.045E-09	1.045E-09	1.045E-09
TH234	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04
PA231	1.975E-08	2.276E-08	2.578E-08	2.879E-08	3.180E-08	3.782E-08	4.685E-08	5.588E-08	6.491E-08	7.695E-08
PA233	8.641E-05	8.703E-05	8.794E-05	8.907E-05	9.038E-05	9.336E-05	9.837E-05	1.037E-04	1.090E-04	1.161E-04
PA234M	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04
PA234	1.373E-07	1.373E-07	1.373E-07	1.373E-07	1.373E-07	1.373E-07	1.373E-07	1.373E-07	1.373E-07	1.373E-07
U232	2.617E-06	2.589E-06	2.495E-06	2.386E-06	2.276E-06	2.068E-06	1.790E-06	1.550E-06	1.341E-06	1.106E-06
U233	2.042E-06	2.044E-06	2.046E-06	2.048E-06	2.050E-06	2.054E-06	2.060E-06	2.066E-06	2.073E-06	2.083E-06
U234	1.083E-02	1.083E-02	1.083E-02	1.084E-02	1.084E-02	1.084E-02	1.085E-02	1.086E-02	1.086E-02	1.087E-02
U235	2.848E-05	2.848E-05	2.848E-05	2.849E-05	2.849E-05	2.849E-05	2.850E-05	2.850E-05	2.851E-05	2.851E-05
U236	1.302E-04	1.302E-04	1.303E-04	1.303E-04	1.303E-04	1.303E-04	1.304E-04	1.305E-04	1.305E-04	1.306E-04
U237	7.079E-06	5.564E-06	4.374E-06	3.438E-06	2.703E-06	1.670E-06	8.113E-07	3.941E-07	1.914E-07	7.309E-08

DECAY TIMES (years out of core)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
U238	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04	1.057E-04
NP237	8.641E-05	8.703E-05	8.794E-05	8.907E-05	9.038E-05	9.336E-05	9.837E-05	1.037E-04	1.090E-04	1.161E-04
PU236	3.486E-06	1.034E-06	3.065E-07	9.092E-08	2.698E-08	2.405E-09	9.770E-11	3.753E-11	3.596E-11	3.591E-11
PU237	5.466E-17	4.804E-29	4.222E-41	3.711E-53	3.261E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	2.301E-01	2.213E-01	2.127E-01	2.045E-01	1.966E-01	1.817E-01	1.614E-01	1.434E-01	1.274E-01	1.089E-01
PU239	3.258E-01	3.258E-01	3.257E-01	3.257E-01	3.256E-01	3.255E-01	3.254E-01	3.252E-01	3.251E-01	3.249E-01
PU240	1.339E-01	1.338E-01	1.337E-01	1.337E-01	1.336E-01	1.335E-01	1.333E-01	1.331E-01	1.328E-01	1.326E-01
PU241	2.885E+01	2.268E+01	1.783E+01	1.402E+01	1.102E+01	6.808E+00	3.307E+00	1.606E+00	7.803E-01	2.979E-01
PU242	8.371E-05	8.371E-05	8.371E-05	8.371E-05	8.371E-05	8.371E-05	8.371E-05	8.371E-05	8.371E-05	8.371E-05
PU244	3.157E-12	3.157E-12	3.157E-12	3.157E-12	3.157E-12	3.157E-12	3.157E-12	3.157E-12	3.157E-12	3.157E-12
AM241	2.771E-01	4.797E-01	6.368E-01	7.583E-01	8.517E-01	9.772E-01	1.069E+00	1.100E+00	1.101E+00	1.082E+00
AM242M	5.906E-04	5.773E-04	5.643E-04	5.516E-04	5.392E-04	5.151E-04	4.811E-04	4.493E-04	4.196E-04	3.830E-04
AM242	5.877E-04	5.744E-04	5.615E-04	5.488E-04	5.365E-04	5.125E-04	4.787E-04	4.470E-04	4.175E-04	3.811E-04
AM243	2.627E-04	2.626E-04	2.625E-04	2.623E-04	2.622E-04	2.620E-04	2.616E-04	2.612E-04	2.609E-04	2.604E-04
CM242	1.142E-03	4.756E-04	4.646E-04	4.541E-04	4.438E-04	4.239E-04	3.959E-04	3.697E-04	3.452E-04	3.152E-04
CM243	1.739E-04	1.540E-04	1.363E-04	1.207E-04	1.069E-04	8.383E-05	5.820E-05	4.041E-05	2.806E-05	1.725E-05
CM244	1.156E-02	9.546E-03	7.883E-03	6.510E-03	5.376E-03	3.666E-03	2.065E-03	1.163E-03	6.550E-04	3.046E-04
CM245	6.429E-07	6.427E-07	6.424E-07	6.422E-07	6.419E-07	6.414E-07	6.406E-07	6.398E-07	6.390E-07	6.380E-07
CM246	1.877E-08	1.875E-08	1.874E-08	1.873E-08	1.871E-08	1.869E-08	1.865E-08	1.860E-08	1.856E-08	1.851E-08
CM247	2.297E-14	2.297E-14	2.297E-14	2.297E-14	2.297E-14	2.297E-14	2.297E-14	2.297E-14	2.297E-14	2.297E-14
SUBTOTAL**	2.287E+03	1.018E+03	5.157E+02	2.918E+02	1.832E+02	9.559E+01	5.841E+01	4.502E+01	3.708E+01	2.971E+01
TOTAL***	2.287E+03	1.018E+03	5.157E+02	2.919E+02	1.832E+02	9.560E+01	5.842E+01	4.503E+01	3.708E+01	2.972E+01

* Four decimal places of accuracy are as reported by ORIGEN2 output and are not significant for many radionuclides.
** Subtotal: total activity of the 145 isotopes listed in the table.
*** Total: total activity of the ORIGEN2 output isotopes.

Template 11

Representative Fuel Source Term Calculations

Introduction

The following data have been used in the Idaho National Engineering and Environmental Laboratory (INEEL) spent nuclear fuel source term calculational methodology to generate a source term template to represent a Zircaloy-4 clad, 0-5% enriched, uranium-based fuel from a heavy water-moderated reactor. Because the spent fuels in this group are primarily MTR-type or plate-fuels, the previously constructed HFBR fuel element geometry model (Template 8) was modified to represent fuels in this group. Modifications included developing new cross sections and adjusting enrichment and burnup. Other differences include the fuel meat material and the number of fueled plates in the element. These data are included below. The calculation methodology used is described in detail in Reference 1.

Reactor Data

The hypothetical element is a plate-type element consisting of 19 curved plates. The plates are stacked, separated by a heavy water gap (102–129 mils), and held together as a rectangular structure by two Zircaloy-4 side plates (140 mils thick). The fuel meat in the 19 plates is a uranium-aluminum-silicon matrix and is clad with Zircaloy-4, as shown in Figure 1. The uranium enrichment is nominally 5% enriched uranium metal and represents the upper end of the 0–5% U-235 enrichment characteristic of the Template 11 fuel group.

The following data provide the specific fuel element dimensions, materials, densities, enrichment, etc. The beginning-of-life (BOL) data below were used in the BOL cross-section development and the burnup calculation for the source term generation.

Fuel Element:	Curved plate
No. of Plates:	19
Fueled Plates:	19
Fuel Plate Thickness:	50 mils
Cross Sectional Dimensions :	2.820 in. by 3.194 in.
Length:	23.75 in.
Fuel Rod:	U-Al-Si (30% U, 68% Al, 2% Si)
Fuel Density:	3.616 g/cc
Fuel Meat Thickness:	21 mils
Uranium Enrichment:	0.60 wt % U-234
	5.00 wt % U-235
	0.35 wt % U-236
	94.05 wt % U-238
Heavy Metal Loading:	2.06 g/element U-234 (BOL)
	17.13 g/element U-235 (BOL)
	1.20 g/element U-236 (BOL)
	322.12 g/element U-238 (BOL)
	342.51 g/element Total U
	776.33 g/element Aluminum-6061 (Fuel Meat)
	22.83 g/element Silicon (Fuel Meat)

Clad:	Zircaloy-4
Clad Density:	6.44 g/cc
Clad Thickness:	14.5 mils
Side Plates:	Zircaloy-4
Side Plate Width:	140 mils
Total Zircaloy-4 Mass:	7,841.99 g/element
Coolant/Moderator :	Heavy Water (D_2O)
Coolant Temperature:	52°C
Coolant Pressure:	175.3 psig
Coolant Density:	1.09325 g/cc

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single fuel element. In addition, for the ORIGEN2 (Reference 2) depletion calculation, conservative and detailed impurity concentrations were added for the Zircaloy-4 clad and aluminum/silicon fuel meat constituents. Tables 1 and 2 list the Zircaloy-4 and Aluminum 6061T impurities and their concentrations, respectively per References 3 and 4.

Burnup

The burnup chosen for this template is 31.5% U-235 depletion, 5.0 MWd, and approximately 5.4 g U-235 depleted for this single element. This burnup represents a nominal range burnup one might expect for this element and uranium loading.

For the analysis, the burnup period is assumed to be 1 year. The burnup and reactor power (approximately 0.0137 MW) is further assumed to be constant and uniform over the 1-year period. At the end of the 1-year period, the element is removed from the core, and the cooling or decay period begins. Table 3 gives the irradiation period and decay times following irradiation.

Cross-Section Development

The neutron cross sections used in the burnup or depletion calculation are based on the methodology described in Reference 1. Cross sections from a standard ORIGEN2 heavy water reactor library were updated once using the BOL cross sections developed for the hypothetical fuel element. These updated cross sections take into account the neutron flux spatial and spectral characteristics to ensure accurate calculation of the actinide production as a function of burnup.

In order to calculate the BOL neutron cross sections, an explicit fuel element was developed with reflective boundary conditions on the element peripheral surfaces. The reflective surfaces simulated an infinite array of fuel elements. The fuel element model is shown in Figure 1.

Fuel Element Exposure History

Table 3 summarizes the power or exposure history used in the burnup or source term calculations for a single fuel element. Following the burnup or exposure period, the radionuclide activities are decayed for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100 years.

Burnup Calculation

The ORIGEN2 computer code (Reference 2) was used to perform the depletion or burnup calculation for a single fuel element. The fuel element masses and impurities, neutron cross sections, burnup, power history, and power level discussed above are input data for the ORIGEN2 calculation. The resulting radionuclide concentrations are given as a function of time in the template table.

The 145 radionuclides listed in the template represent greater than 99.9% of the total curie inventory had all 684 activation products, 880 fission products, and 127 actinide/daughter isotopes from the ORIGEN2 output been included in the template.

References

1. J. W. Sterbentz and C. A. Wemple, *Calculational Burnup Methodology and Validation for the Idaho National Engineering Laboratory Spent Nuclear Fuels*, INEL-96/0304, September 1996.

2. A. G. Croff, *ORIGEN2—A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, July 1980.

3. Oak Ridge National Laboratory, "Summary of the Nuclear Design and Performance of the Light Water Breeder Reactor (LWBR)," WAPD-TM-1326, June 1979. *Characteristics of Potential Repository Wastes*, DOE/RW-0184-V1-R1, Volume 1, Oak Ridge, TN 37831, July 1992.

4. John Logan to Tom Clements, Appendix data in a letter, "Assessment of Neutron-Activation Products in Low Level Waste Discharged from Nuclear Reactors at the Test Reactor Area; and Sent to the Radioactive Waste Management Complex for Disposal," JAL-04-99, September 9, 1999.