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Source Term Estimates for DOE Spent Nuclear Fuels

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Source Term Estimates for DOE Spent Nuclear Fuels

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Source Term Estimates

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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

This analysis summarizes information on U.S. Department of Energy (DOE) spent nuclear fuels that currently reside, or will be consolidated, at one of three DOE sites prior to final disposition. Valuable information and suggestions have been provided by personnel at each of these sites as well as the Yucca Mountain Project. The following individuals have been particularly helpful.

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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.

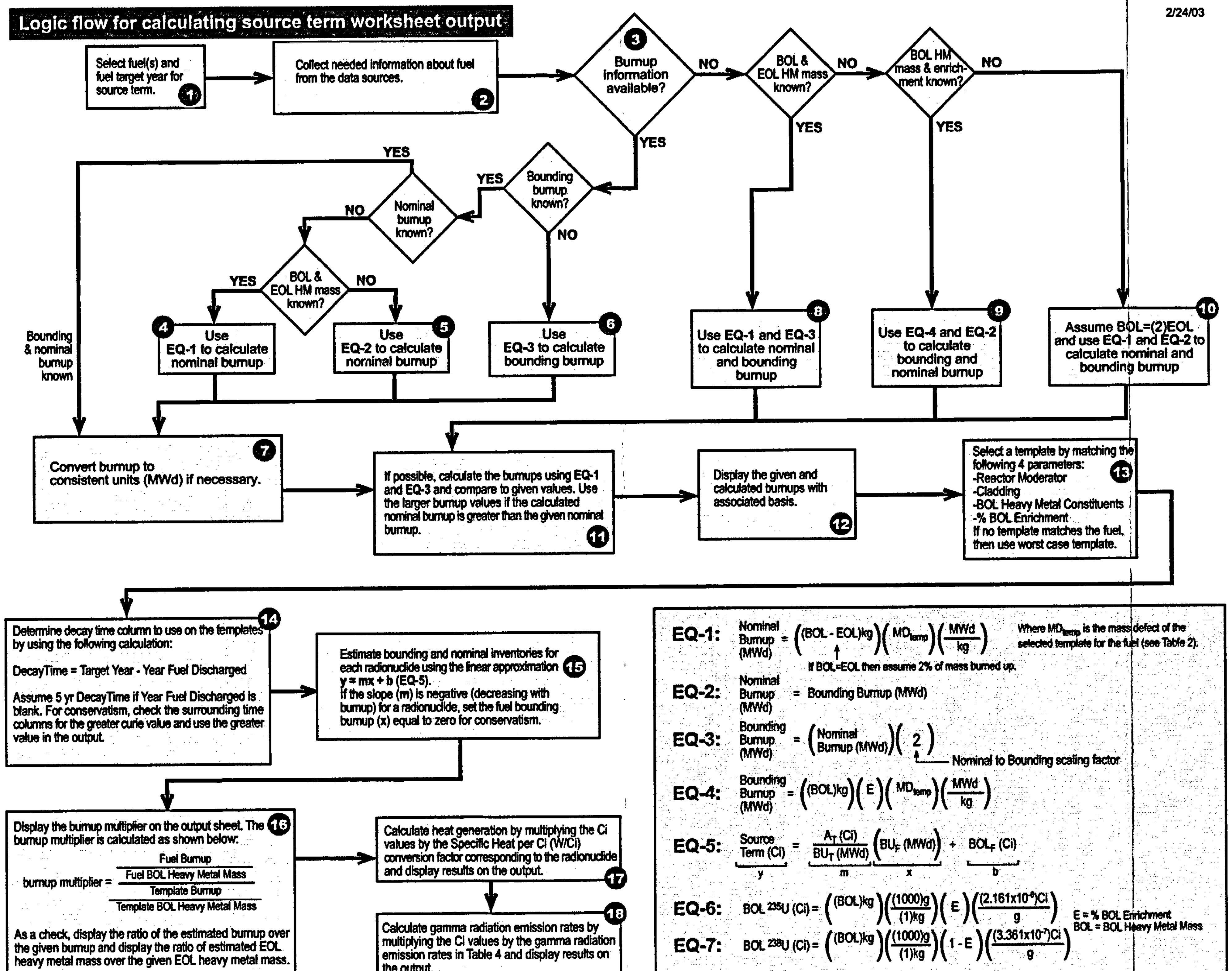

Figure 1. Logic flow for calculating source term worksheet output.

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.

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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: $Burnup_A(MWd) = Burnup_S(MWd/MTIHM)(BOL(kg))(1MT/1000kg)$. 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.

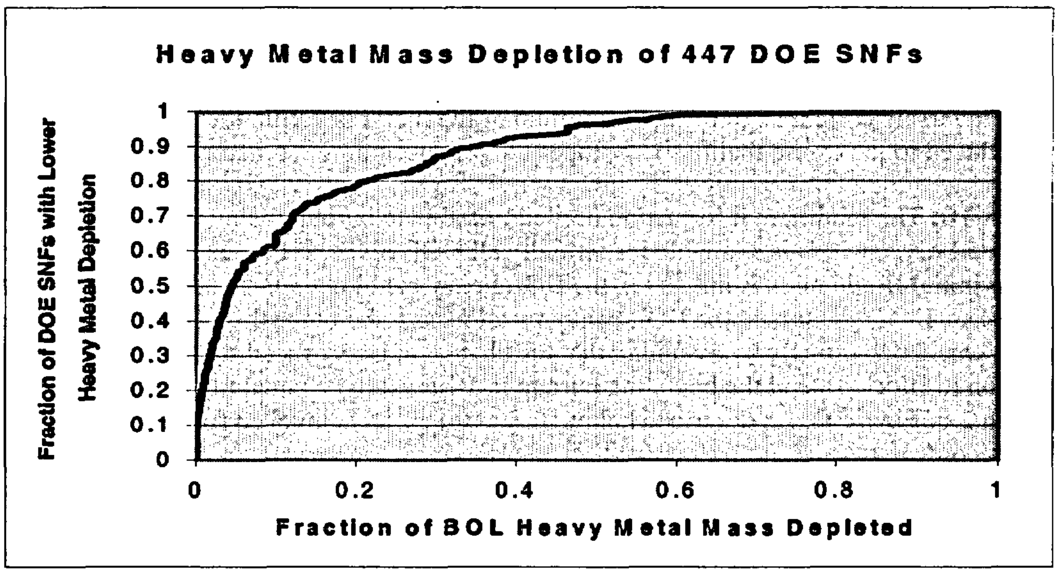

Figure 2. Heavy metal mass depletion of 447 DOE spent nuclear fuels.

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).

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																	
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: Up to 1×10^4 yrs	Total System Performance Assessment (TSPA)[a] Dose Contribution: $> 1 \times 10^4$ to 10^8 yrs	Preclosure Safety Analysis (PSA)[b]
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
Tl208*			
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.

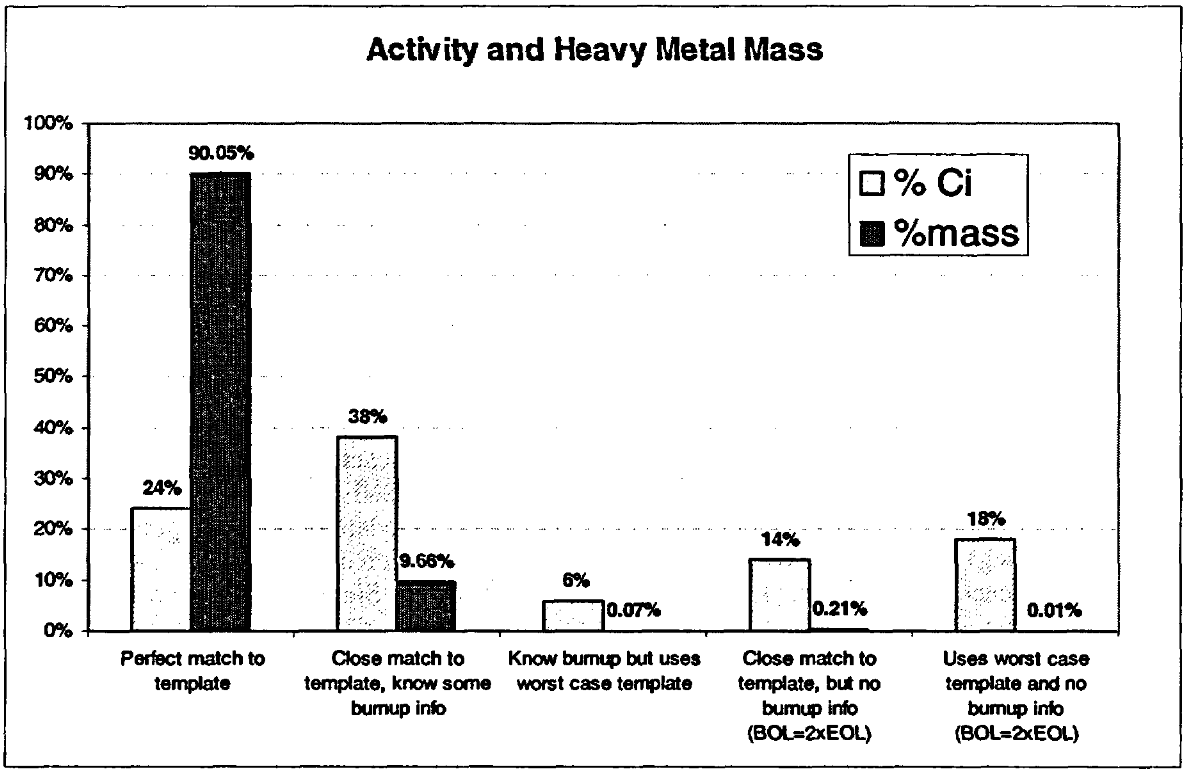

Figure 3. Activity and quantity of SNF relating to known information about SNF.

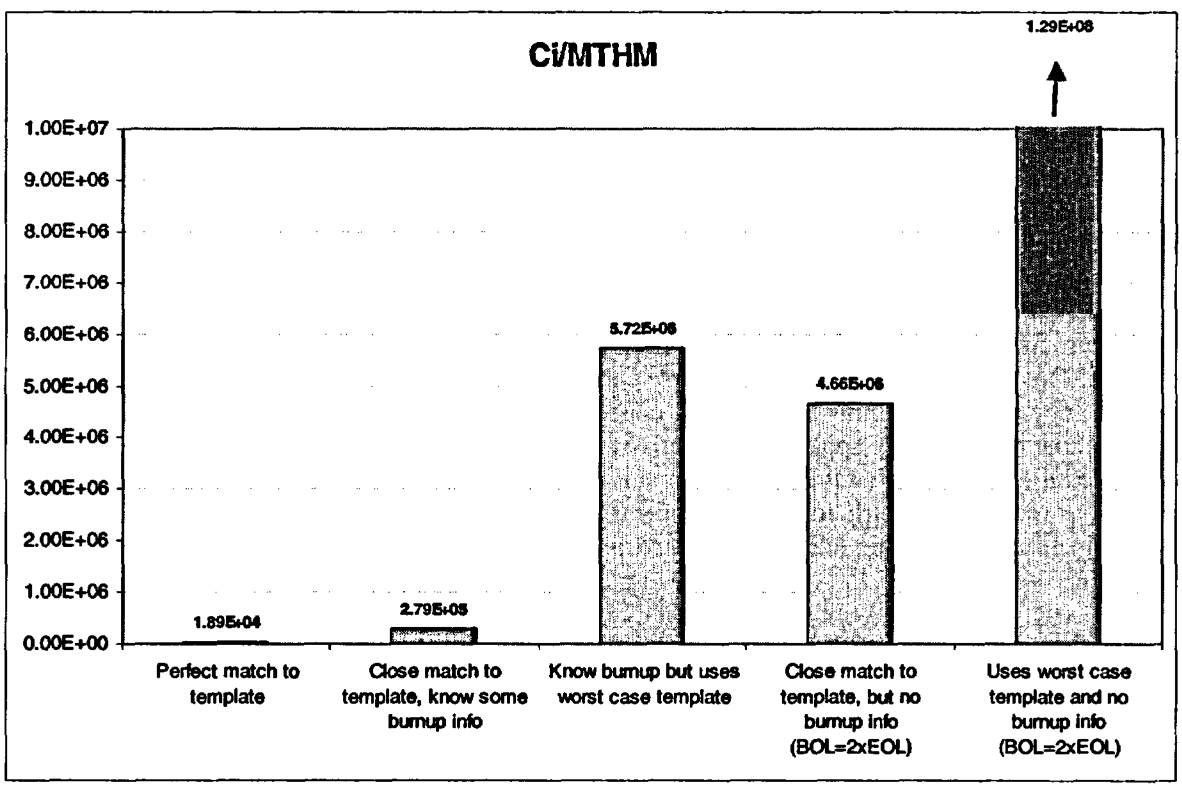

Figure 4. Activity per MTHM relating to known information about SNF.

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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.

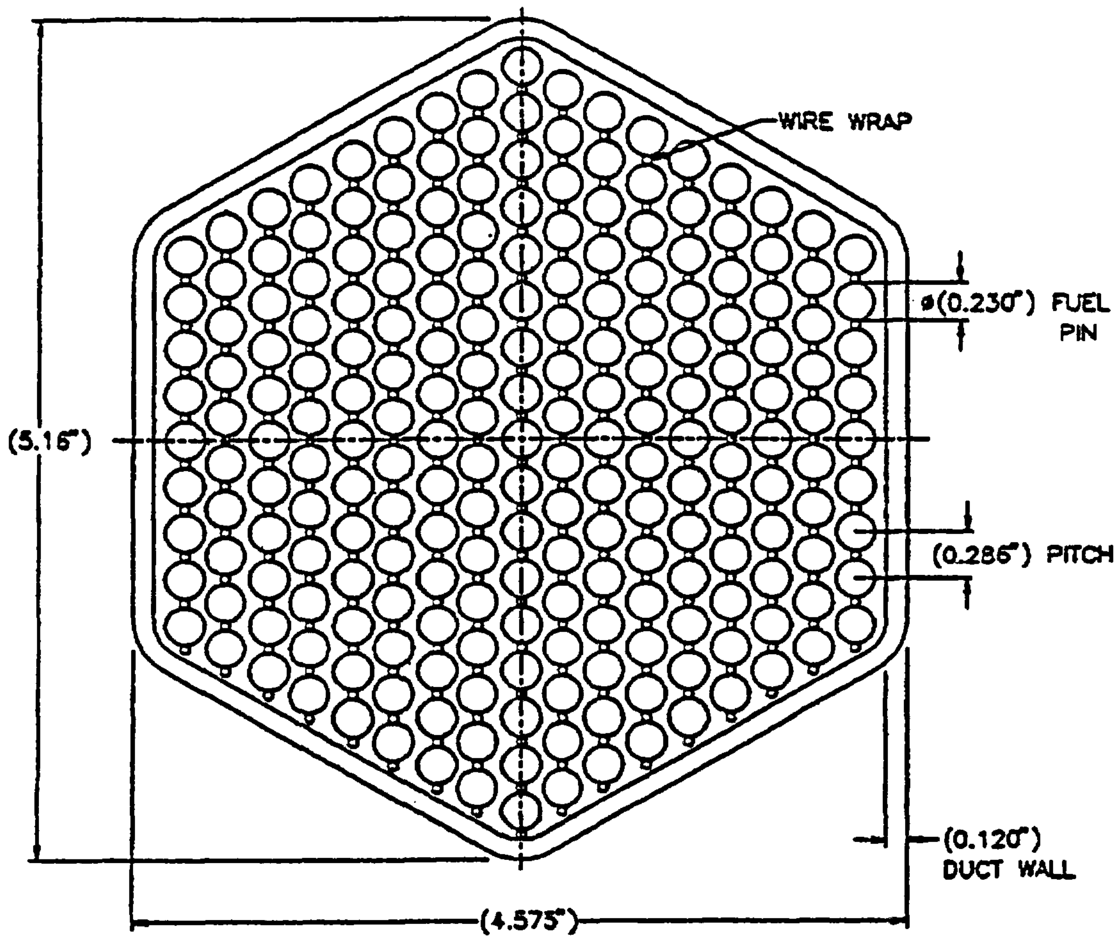

Figure 1. Fast Flux Test Facility fuel pin bundle cross section.

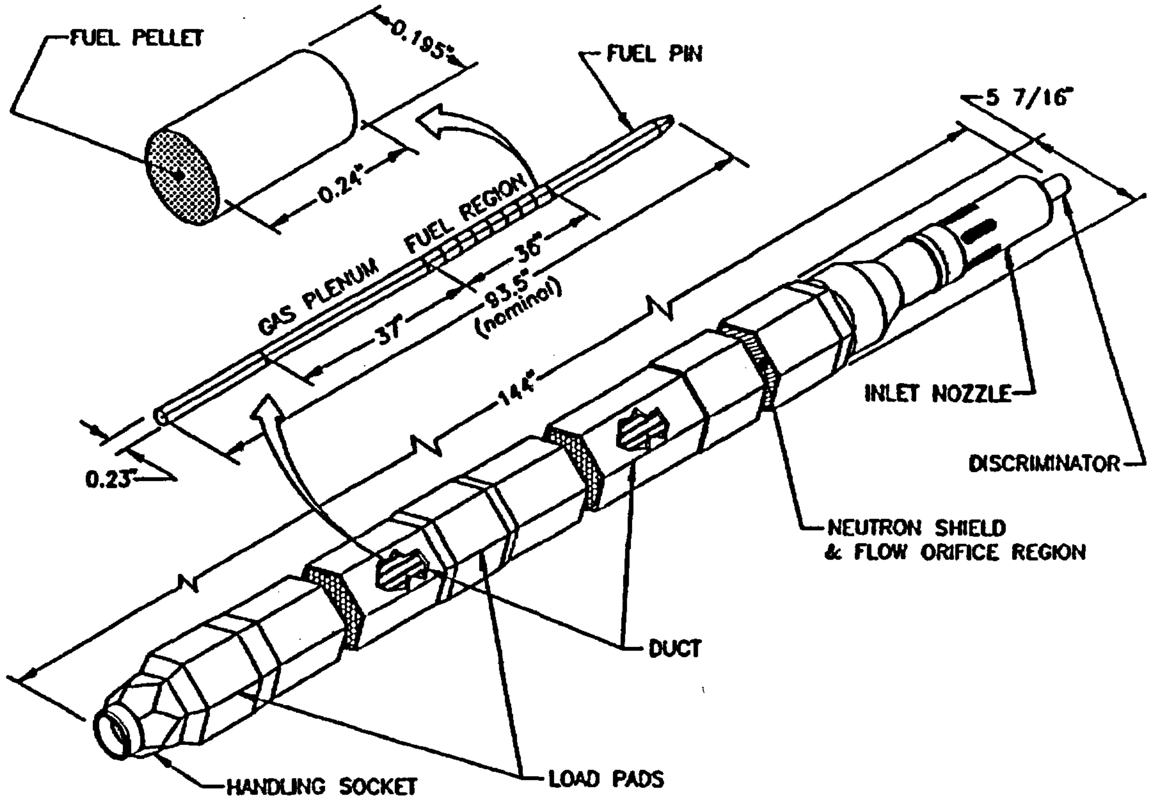

Figure 2. Fast Flux Test Facility standard driver fuel assembly.

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 × 2.454 × 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

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.

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3. J. R. Dietrich and W. H. Zinn, *Solid Fuel Reactors*, General Nuclear Engineering Corporation, Addison-Wesley Publishing Company, Inc., September 1958.

4. 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.

5. E. P. Alexanderson and H. A. Wagner, "FERMI-I New Age for Nuclear Power," the American Nuclear Society, 1979.

6. Oak Ridge National Laboratory, "Characteristics of Potential Repository Wastes," DOE/RW-0184-V1-R1, Volume 1, July 1992.

7. H. C. Hecker, "Summary of the Nuclear Design and Performance of the Light Water Breeder Reactor (LWBR)," WAPD-TM-1326, June 1979.

8. 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.

9. E. A. Avallone and T. Baumeister III, *MARK'S Standard Handbook for Mechanical Engineers*, Ninth Edition.

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

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

Cumulative Operational (days)	Cumulative Burnup (MWD)	Total Core Power (MW_{th})
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.

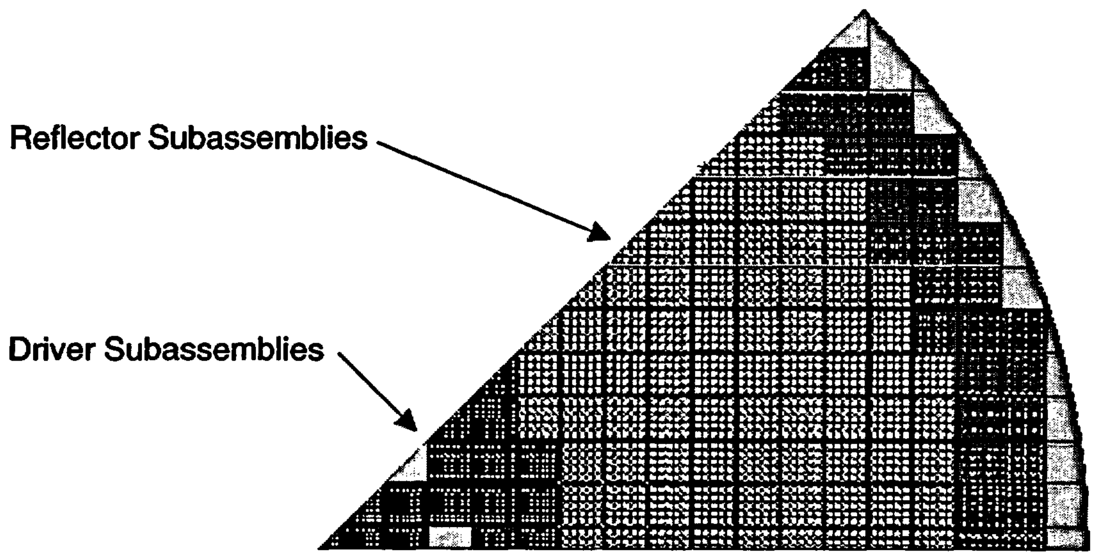

Figure 1. MCNP 1/8-core model cross-sectional view of the FERMI core.

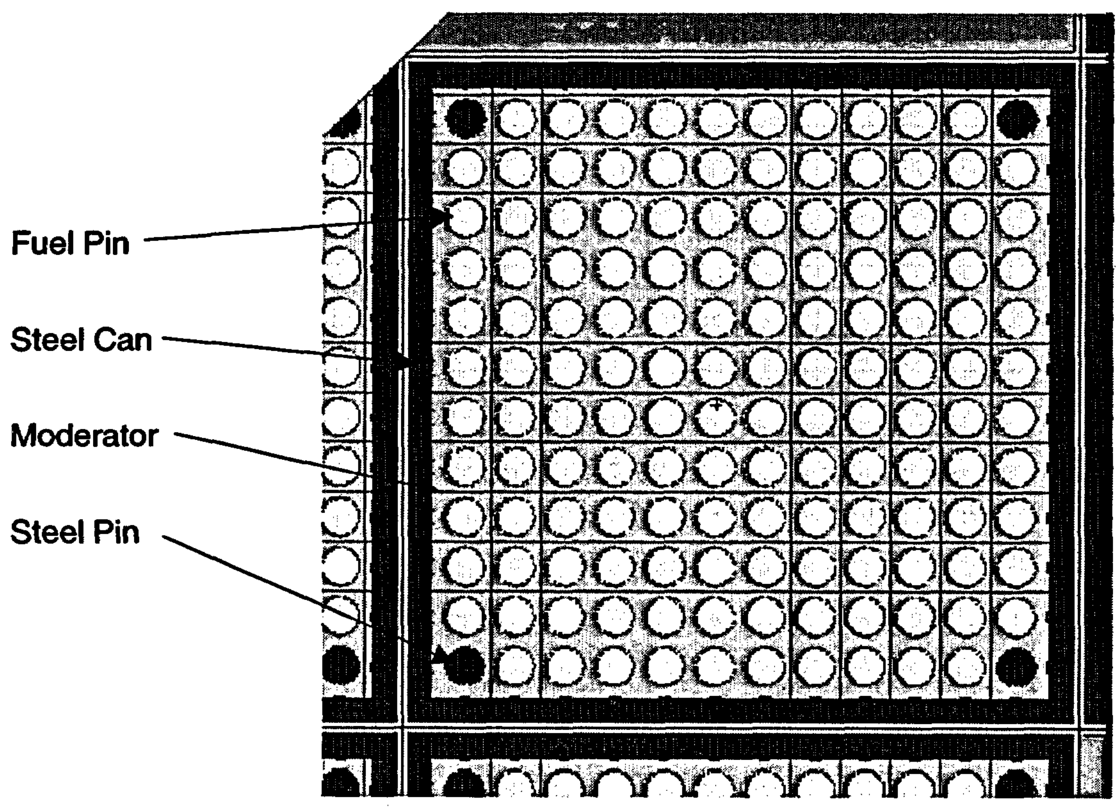

Figure 2. MCNP model cross-sectional view of a FERMI driver fuel assembly.

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.

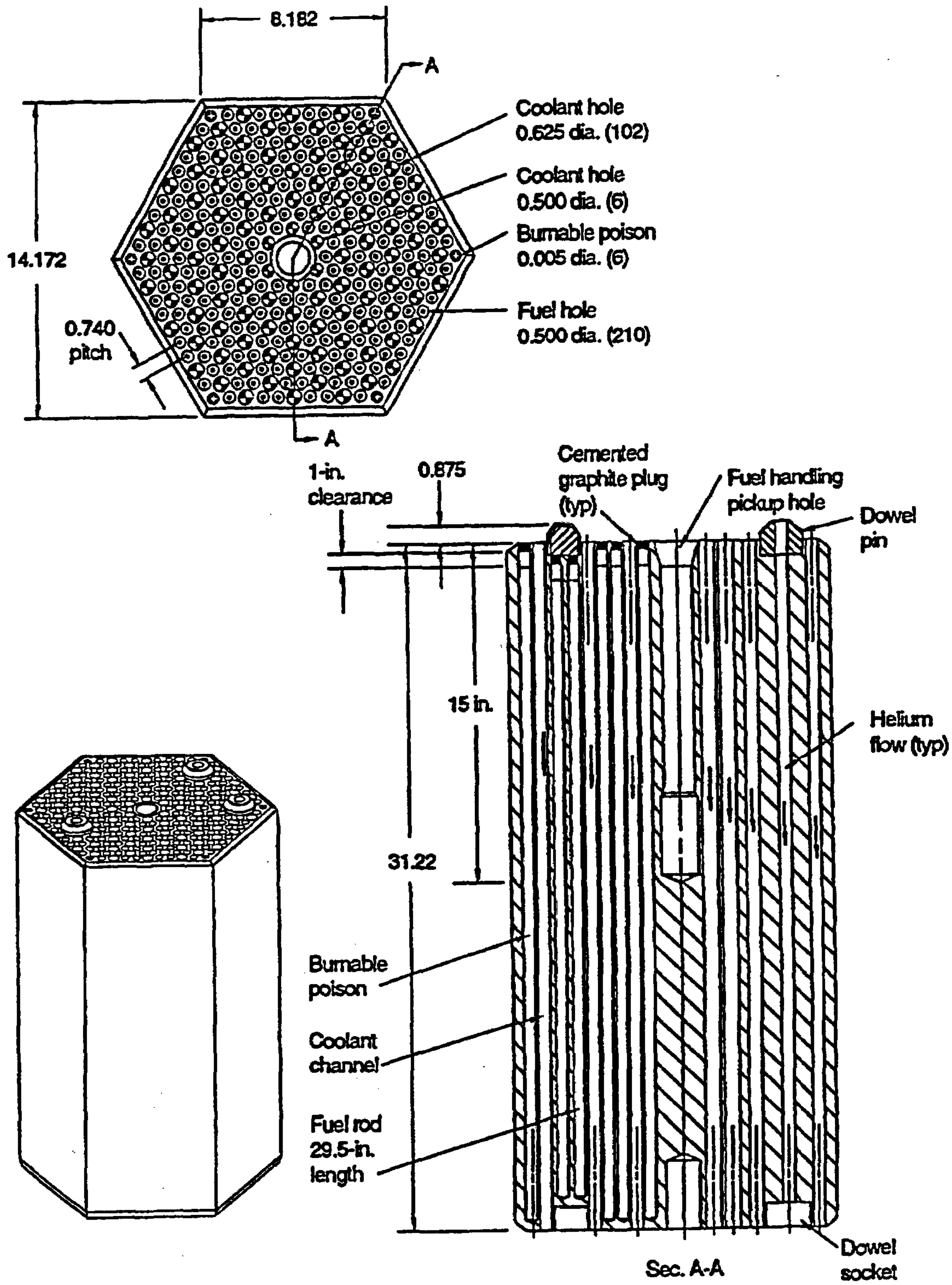

Figure 1. Standard FSV fuel element.

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.

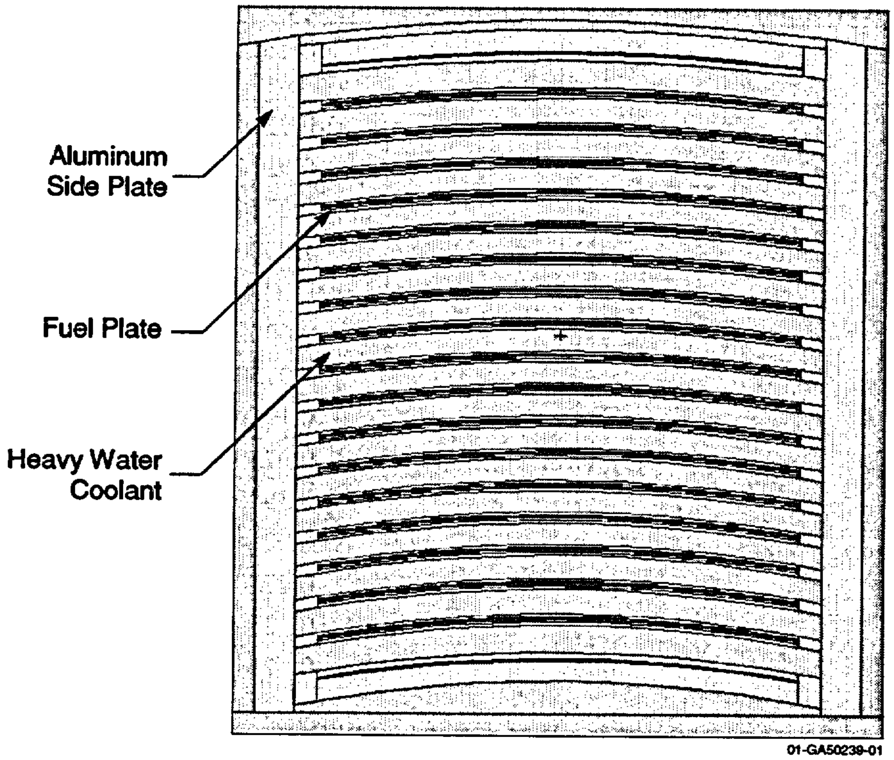

Figure 1. High Flux Beam Reactor curved-plate fuel assembly.

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.

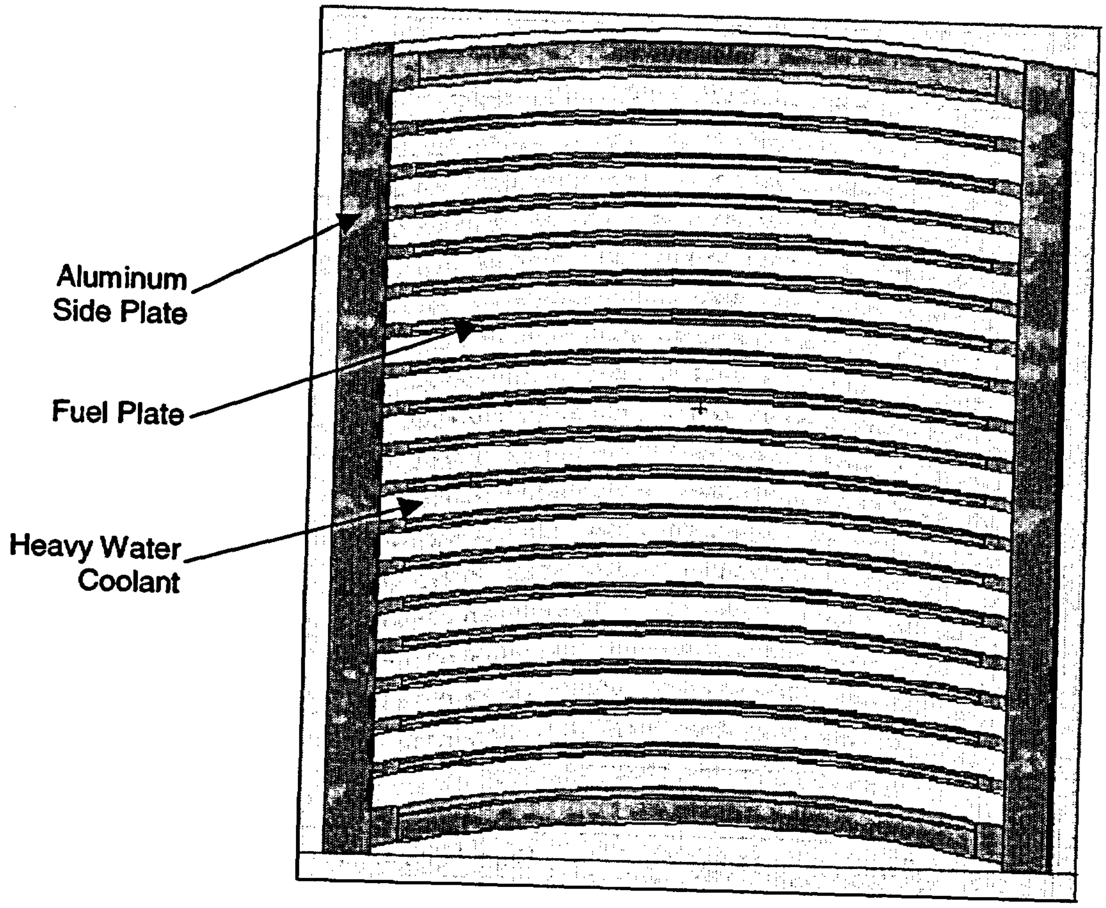

Figure 1. Curved-plate fuel assembly used in the analysis.

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.

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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.

4. E. A. Avallone and T. Baumeister III, "MARK'S Standard Handbook for Mechanical Engineers," Ninth Edition.

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.

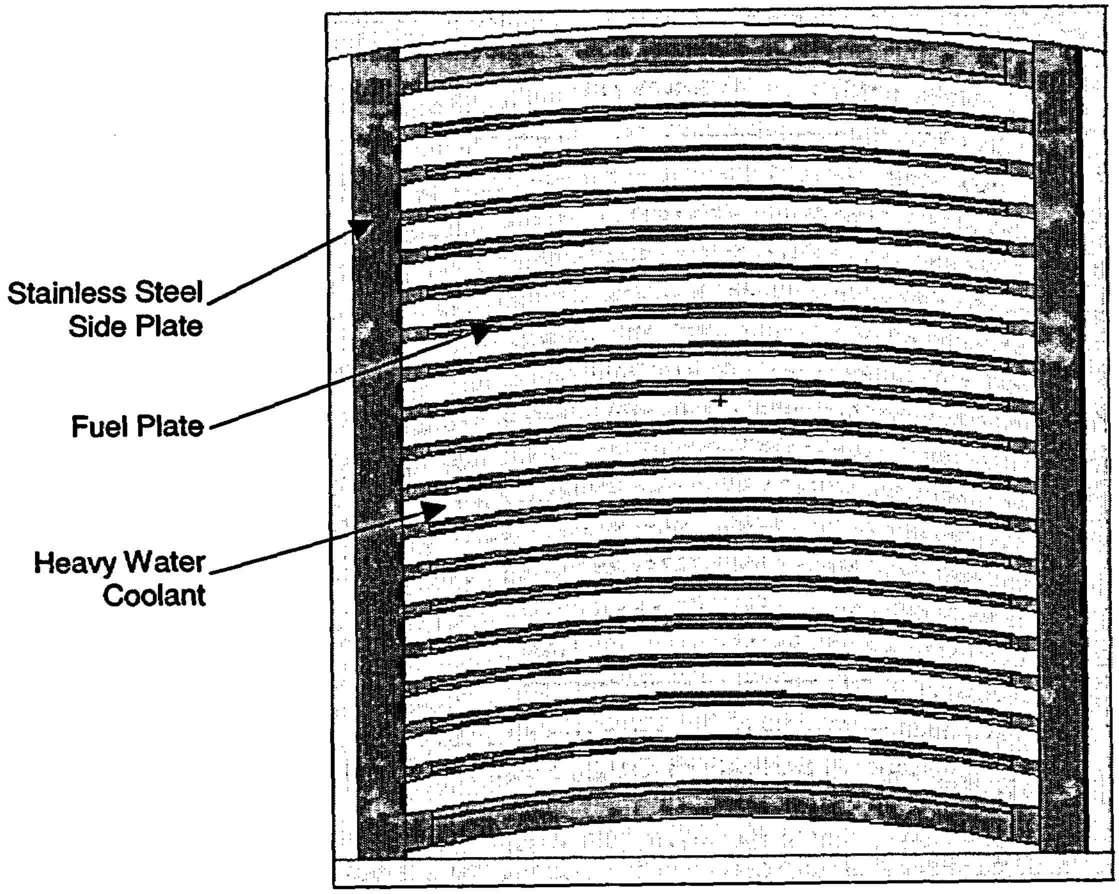

Figure 1. Curved-plate fuel element used in the analysis.

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

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

Table 1. Zircaloy-4 material constituent and impurity concentrations.

Constituent or Impurity	Impurity (ppm)
H	25
Li	
B	0.5
C	270
N	80
O	950
Na	
Al	75
Si	120
P	100
S	35
Cl	
K	
Ca	
Sc	
Ti	50
V	50
Cr	1250
Mn	50
Fe	2250
Co	20
Ni	70
Cu	50
Zn	100
Ga	
As	
Se	
Br	
Rb	
Sr	
Y	
Zr	979069
Nb	70
Mo	50

Constituent or Impurity	Impurity (ppm)
Ag	
Cd	0.5
Sn	16000
Sb	
Cs	
Ba	
La	
Ce	
Sm	10
Gd	5
Eu	
Tb	
Dy	
Ho	
Yb	
Lu	
Hf	35
Ta	200
W	100
Pb	100
Th	7
U	3.5

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 entries 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.

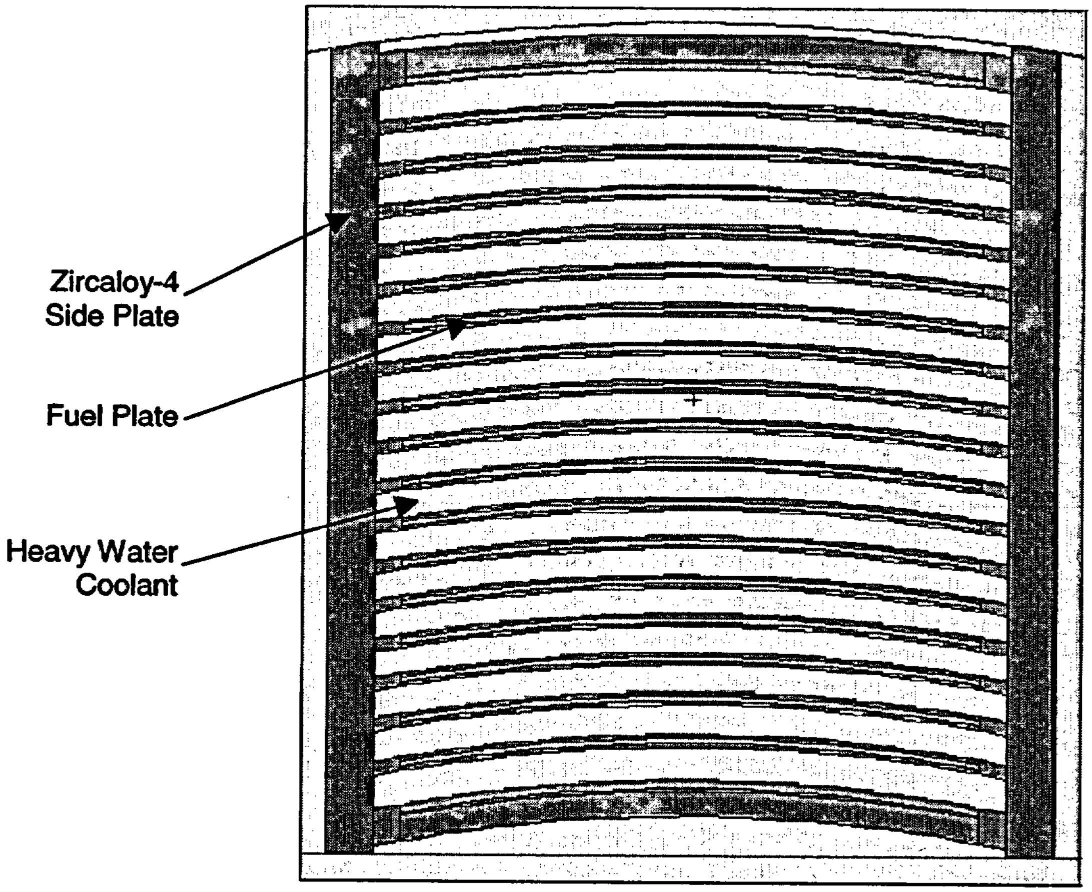

Figure 1. Curved-plate fuel element used in the analysis.

Template 11

Zircaloy-4 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:	Zircaloy-4
Burnup:	5.4 g U-235 depleted
Burnup:	5 MWd/single element
Burnup:	31.5% 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	6.411E-01	4.843E-01	3.657E-01	2.763E-01	2.087E-01	1.190E-01	5.128E-02	2.209E-02	9.520E-03	3.098E-03
BE 10	8.087E-10	8.087E-10	8.087E-10	8.087E-10	8.087E-10	8.087E-10	8.087E-10	8.087E-10	8.087E-10	8.087E-10
C 14	5.652E-03	5.649E-03	5.645E-03	5.642E-03	5.639E-03	5.632E-03	5.622E-03	5.611E-03	5.601E-03	5.588E-03
CL 36	4.189E-10	4.189E-10	4.188E-10	4.188E-10	4.188E-10	4.188E-10	4.188E-10	4.188E-10	4.188E-10	4.188E-10
CR 51	6.221E-19	8.975E-39	1.295E-58	1.868E-78	2.695E-98	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54	6.789E-04	1.182E-05	2.058E-07	3.582E-09	6.236E-11	1.890E-14	9.973E-20	5.262E-25	2.776E-30	2.550E-37
FE 55	8.548E-01	2.254E-01	5.944E-02	1.567E-02	4.133E-03	2.874E-04	5.269E-06	9.661E-08	1.771E-09	8.564E-12
FE 59	2.225E-13	1.351E-25	8.200E-38	4.978E-50	3.021E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60	1.666E+01	8.632E+00	4.472E+00	2.317E+00	1.200E+00	3.221E-01	4.478E-02	6.227E-03	8.658E-04	6.236E-05
NI 59	1.163E-04	1.163E-04	1.162E-04	1.162E-04	1.162E-04	1.162E-04	1.162E-04	1.162E-04	1.162E-04	1.162E-04
NI 63	1.619E-02	1.559E-02	1.502E-02	1.446E-02	1.393E-02	1.292E-02	1.154E-02	1.030E-02	9.203E-03	7.916E-03
ZN 65	6.459E-03	3.596E-05	2.002E-07	1.115E-09	6.205E-12	1.923E-16	3.319E-23	5.728E-30	9.886E-37	9.499E-46
SE 79	6.264E-05	6.263E-05	6.263E-05	6.263E-05	6.262E-05	6.262E-05	6.261E-05	6.260E-05	6.259E-05	6.257E-05
KR 85	1.330E+00	9.623E-01	6.965E-01	5.041E-01	3.648E-01	1.911E-01	7.246E-02	2.747E-02	1.041E-02	2.858E-03
RB 87	4.100E-09	4.100E-09	4.100E-09	4.100E-09	4.100E-09	4.100E-09	4.100E-09	4.100E-09	4.100E-09	4.100E-09
SR 89	6.372E-09	8.270E-20	1.073E-30	1.393E-41	1.808E-52	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	1.291E+01	1.146E+01	1.017E+01	9.031E+00	8.018E+00	6.319E+00	4.422E+00	3.094E+00	2.165E+00	1.345E+00
Y 90	1.291E+01	1.146E+01	1.017E+01	9.033E+00	8.020E+00	6.321E+00	4.423E+00	3.095E+00	2.166E+00	1.345E+00
Y 91	2.435E-07	9.771E-17	3.922E-26	1.574E-35	6.320E-45	1.018E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93	1.371E-03	1.371E-03	1.371E-03	1.371E-03	1.371E-03	1.371E-03	1.371E-03	1.371E-03	1.371E-03	1.371E-03
ZR 95	3.233E-06	8.263E-15	2.112E-23	5.399E-32	1.380E-40	9.017E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M	3.180E-04	5.395E-04	7.111E-04	8.442E-04	9.473E-04	1.089E-03	1.203E-03	1.256E-03	1.281E-03	1.295E-03
NB 94	1.879E-04	1.879E-04	1.878E-04	1.878E-04	1.878E-04	1.877E-04	1.876E-04	1.875E-04	1.874E-04	1.873E-04

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
NB 95	7.178E-06	1.835E-14	4.689E-23	1.199E-31	3.063E-40	2.002E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	2.398E-08	6.130E-17	1.567E-25	4.005E-34	1.024E-42	6.689E-60	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93	2.900E-06	2.897E-06	2.894E-06	2.891E-06	2.888E-06	2.883E-06	2.874E-06	2.865E-06	2.857E-06	2.846E-06
TC 99	2.207E-03	2.207E-03	2.207E-03	2.207E-03	2.207E-03	2.207E-03	2.206E-03	2.206E-03	2.206E-03	2.206E-03
RU103	4.294E-12	4.344E-26	4.395E-40	4.446E-54	4.498E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	1.483E+00	4.764E-02	1.530E-03	4.916E-05	1.579E-06	1.629E-09	5.399E-14	1.789E-18	5.930E-23	6.313E-29
RH103M	3.871E-12	3.916E-26	3.962E-40	4.008E-54	4.055E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	1.483E+00	4.764E-02	1.530E-03	4.916E-05	1.579E-06	1.629E-09	5.399E-14	1.789E-18	5.930E-23	6.313E-29
PD107	5.642E-06	5.642E-06	5.642E-06	5.642E-06	5.642E-06	5.642E-06	5.642E-06	5.642E-06	5.642E-06	5.642E-06
AG110	2.289E-06	1.444E-08	9.110E-11	5.748E-13	3.626E-15	1.443E-19	3.624E-26	9.101E-33	2.285E-39	3.621E-48
AG110M	1.721E-04	1.086E-06	6.850E-09	4.321E-11	2.726E-13	1.085E-17	2.725E-24	6.843E-31	1.718E-37	2.722E-46
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	1.775E-03	1.400E-03	1.104E-03	8.705E-04	6.865E-04	4.269E-04	2.093E-04	1.026E-04	5.032E-05	1.946E-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	1.185E-13	5.572E-26	2.620E-38	1.232E-50	5.790E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	1.672E-13	1.318E-24	1.039E-35	8.201E-47	6.468E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	1.747E-13	1.377E-24	1.086E-35	8.570E-47	6.759E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	5.964E-18	2.804E-30	1.318E-42	6.197E-55	2.913E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	1.213E-01	6.919E-04	3.947E-06	2.253E-08	1.285E-10	4.183E-15	7.771E-22	1.443E-28	2.681E-35	2.841E-44
SN121M	2.137E-03	1.994E-03	1.861E-03	1.736E-03	1.619E-03	1.410E-03	1.145E-03	9.298E-04	7.552E-04	5.722E-04
SN123	3.057E-04	1.695E-08	9.396E-13	5.210E-17	2.888E-21	8.877E-30	1.513E-42	2.578E-55	4.393E-68	4.149E-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	6.028E-05	6.027E-05	6.027E-05	6.027E-05	6.027E-05	6.026E-05	6.026E-05	6.025E-05	6.024E-05	6.024E-05
SB124	1.015E-09	7.477E-19	5.508E-28	4.059E-37	2.991E-46	1.624E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	1.725E+00	4.935E-01	1.412E-01	4.041E-02	1.156E-02	9.468E-04	2.218E-05	5.198E-07	1.218E-08	8.166E-11
SB126	8.439E-06	8.438E-06	8.438E-06	8.438E-06	8.437E-06	8.437E-06	8.436E-06	8.435E-06	8.434E-06	8.433E-06
SB126M	6.028E-05	6.027E-05	6.027E-05	6.027E-05	6.027E-05	6.026E-05	6.026E-05	6.025E-05	6.024E-05	6.024E-05
TE123M	7.104E-08	1.811E-12	4.614E-17	1.175E-21	2.997E-26	1.946E-35	3.222E-49	5.334E-63	8.829E-77	3.725E-95
TE125M	4.208E-01	1.204E-01	3.446E-02	9.859E-03	2.821E-03	2.310E-04	5.413E-06	1.268E-07	2.972E-09	1.993E-11
TE127	2.436E-05	2.204E-10	1.995E-15	1.805E-20	1.633E-25	1.337E-35	9.907E-51	7.339E-66	5.438E-81	0.000E+00
TE127M	2.487E-05	2.250E-10	2.036E-15	1.843E-20	1.667E-25	1.365E-35	1.011E-50	7.493E-66	5.551E-81	0.000E+00
TE129	3.746E-16	1.628E-32	7.072E-49	3.073E-65	1.335E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	5.755E-16	2.500E-32	1.086E-48	4.720E-65	2.051E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	3.751E-06	3.751E-06	3.751E-06	3.751E-06	3.751E-06	3.751E-06	3.751E-06	3.751E-06	3.751E-06	3.751E-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	9.499E-01	1.769E-01	3.294E-02	6.135E-03	1.142E-03	3.962E-05	2.559E-07	1.652E-09	1.067E-11	1.283E-14
CS135	3.957E-05	3.957E-05	3.957E-05	3.957E-05	3.957E-05	3.957E-05	3.957E-05	3.957E-05	3.957E-05	3.957E-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

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
CS137	1.432E+01	1.275E+01	1.136E+01	1.012E+01	9.019E+00	7.158E+00	5.061E+00	3.579E+00	2.531E+00	1.594E+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	1.354E+01	1.207E+01	1.075E+01	9.576E+00	8.532E+00	6.771E+00	4.788E+00	3.386E+00	2.394E+00	1.508E+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	8.101E-15	9.977E-32	1.229E-48	1.513E-65	1.864E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	4.409E-09	4.409E-09	4.409E-09	4.409E-09	4.409E-09	4.409E-09	4.409E-09	4.409E-09	4.409E-09	4.409E-09
CE144	4.119E+00	4.795E-02	5.582E-04	6.498E-06	7.564E-08	1.025E-11	1.617E-17	2.552E-23	4.026E-29	7.393E-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	4.119E+00	4.795E-02	5.582E-04	6.498E-06	7.565E-08	1.025E-11	1.617E-17	2.552E-23	4.026E-29	7.394E-37
PR144M	4.942E-02	5.754E-04	6.698E-06	7.797E-08	9.077E-10	1.230E-13	1.941E-19	3.062E-25	4.831E-31	8.872E-39
ND144	2.305E-13	2.320E-13	2.320E-13	2.320E-13	2.320E-13	2.320E-13	2.320E-13	2.320E-13	2.320E-13	2.320E-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	1.084E-04	9.104E-05	7.489E-05	6.158E-05	5.063E-05	3.422E-05	1.902E-05	1.057E-05	5.875E-06	2.685E-06
PM147	1.378E+01	3.676E+00	9.810E-01	2.618E-01	6.986E-02	4.975E-03	9.453E-05	1.796E-06	3.414E-08	1.731E-10
PM148M	2.329E-13	1.134E-26	5.524E-40	2.690E-53	1.310E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	1.312E-14	6.389E-28	3.111E-41	1.515E-54	7.379E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	4.249E-05	1.027E-06	2.482E-08	6.000E-10	1.450E-11	8.472E-15	1.196E-19	1.689E-24	2.386E-29	8.143E-36
SM147	1.328E-09	1.575E-09	1.642E-09	1.659E-09	1.664E-09	1.665E-09	1.666E-09	1.666E-09	1.666E-09	1.666E-09
SM151	3.907E-02	3.760E-02	3.617E-02	3.481E-02	3.349E-02	3.101E-02	2.763E-02	2.461E-02	2.193E-02	1.880E-02
EU152	5.019E-04	3.890E-04	3.015E-04	2.337E-04	1.811E-04	1.088E-04	5.065E-05	2.358E-05	1.097E-05	3.962E-06
EU154	3.810E-01	2.546E-01	1.702E-01	1.137E-01	7.600E-02	3.395E-02	1.013E-02	3.025E-03	9.029E-04	1.802E-04
EU155	2.080E-01	1.034E-01	5.142E-02	2.556E-02	1.271E-02	3.140E-03	3.859E-04	4.743E-05	5.827E-06	3.560E-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	4.445E-04	2.378E-06	1.273E-08	6.809E-11	3.643E-13	1.043E-17	1.598E-24	2.448E-31	3.750E-38	3.074E-47
TB160	4.354E-10	1.085E-17	2.703E-25	6.736E-33	1.679E-40	1.042E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	4.308E-16	4.308E-16	4.308E-16	4.308E-16	4.308E-16	4.308E-16	4.308E-16	4.308E-16	4.308E-16	4.308E-16
TL207	2.179E-09	4.306E-09	6.515E-09	8.793E-09	1.113E-08	1.594E-08	2.342E-08	3.111E-08	3.893E-08	4.946E-08
TL208	2.494E-07	2.806E-07	2.766E-07	2.659E-07	2.541E-07	2.310E-07	2.003E-07	1.737E-07	1.506E-07	1.247E-07
PB210	4.912E-11	2.904E-10	8.600E-10	1.874E-09	3.432E-09	8.502E-09	2.192E-08	4.336E-08	7.356E-08	1.282E-07
PB211	2.185E-09	4.318E-09	6.533E-09	8.818E-09	1.116E-08	1.599E-08	2.349E-08	3.120E-08	3.904E-08	4.960E-08
PB212	6.940E-07	7.809E-07	7.698E-07	7.399E-07	7.071E-07	6.430E-07	5.574E-07	4.833E-07	4.191E-07	3.471E-07
BI211	2.185E-09	4.318E-09	6.533E-09	8.818E-09	1.116E-08	1.599E-08	2.349E-08	3.120E-08	3.904E-08	4.960E-08
BI212	6.940E-07	7.809E-07	7.698E-07	7.399E-07	7.071E-07	6.430E-07	5.574E-07	4.833E-07	4.191E-07	3.471E-07
PO212	4.447E-07	5.003E-07	4.932E-07	4.741E-07	4.530E-07	4.120E-07	3.572E-07	3.097E-07	2.685E-07	2.224E-07
PO215	2.185E-09	4.318E-09	6.533E-09	8.818E-09	1.116E-08	1.599E-08	2.349E-08	3.120E-08	3.904E-08	4.960E-08
PO216	6.940E-07	7.809E-07	7.698E-07	7.399E-07	7.071E-07	6.430E-07	5.574E-07	4.833E-07	4.191E-07	3.471E-07
RN219	2.185E-09	4.318E-09	6.533E-09	8.818E-09	1.116E-08	1.599E-08	2.349E-08	3.120E-08	3.904E-08	4.960E-08
RN220	6.940E-07	7.809E-07	7.698E-07	7.399E-07	7.071E-07	6.430E-07	5.574E-07	4.833E-07	4.191E-07	3.471E-07
FR223	3.015E-11	5.953E-11	9.004E-11	1.215E-10	1.538E-10	2.203E-10	3.238E-10	4.301E-10	5.381E-10	6.838E-10

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
RA223	2.185E-09	4.318E-09	6.533E-09	8.818E-09	1.116E-08	1.599E-08	2.349E-08	3.120E-08	3.904E-08	4.960E-08
RA224	6.940E-07	7.809E-07	7.698E-07	7.399E-07	7.071E-07	6.430E-07	5.574E-07	4.833E-07	4.191E-07	3.471E-07
RA226	8.271E-10	2.770E-09	5.849E-09	1.006E-08	1.541E-08	2.948E-08	5.902E-08	9.861E-08	1.482E-07	2.297E-07
RA228	2.755E-09	4.050E-09	4.823E-09	5.283E-09	5.557E-09	5.819E-09	5.932E-09	5.956E-09	5.961E-09	5.963E-09
AC227	2.185E-09	4.314E-09	6.525E-09	8.805E-09	1.114E-08	1.597E-08	2.346E-08	3.116E-08	3.899E-08	4.955E-08
TH227	2.155E-09	4.259E-09	6.443E-09	8.696E-09	1.101E-08	1.577E-08	2.316E-08	3.077E-08	3.850E-08	4.891E-08
TH228	6.939E-07	7.803E-07	7.691E-07	7.393E-07	7.065E-07	6.429E-07	5.574E-07	4.833E-07	4.191E-07	3.471E-07
TH229	2.333E-09	4.492E-09	6.651E-09	8.809E-09	1.097E-08	1.528E-08	2.174E-08	2.820E-08	3.464E-08	4.323E-08
TH230	6.348E-07	1.162E-06	1.689E-06	2.217E-06	2.744E-06	3.798E-06	5.380E-06	6.961E-06	8.542E-06	1.065E-05
TH231	2.536E-05	2.536E-05	2.536E-05	2.536E-05	2.536E-05	2.536E-05	2.537E-05	2.537E-05	2.537E-05	2.537E-05
TH232	5.962E-09	5.962E-09	5.962E-09	5.963E-09	5.963E-09	5.963E-09	5.963E-09	5.963E-09	5.963E-09	5.963E-09
TH234	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04
PA231	1.528E-08	1.796E-08	2.064E-08	2.332E-08	2.601E-08	3.137E-08	3.941E-08	4.745E-08	5.548E-08	6.619E-08
PA233	2.661E-05	2.667E-05	2.677E-05	2.689E-05	2.703E-05	2.735E-05	2.789E-05	2.845E-05	2.902E-05	2.978E-05
PA234M	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04
PA234	1.399E-07	1.399E-07	1.399E-07	1.399E-07	1.399E-07	1.399E-07	1.399E-07	1.399E-07	1.399E-07	1.399E-07
U232	8.088E-07	7.834E-07	7.503E-07	7.161E-07	6.828E-07	6.203E-07	5.369E-07	4.647E-07	4.022E-07	3.317E-07
U233	4.576E-06	4.577E-06	4.577E-06	4.578E-06	4.578E-06	4.579E-06	4.581E-06	4.582E-06	4.584E-06	4.586E-06
U234	1.171E-02	1.172E-02	1.172E-02	1.172E-02	1.172E-02	1.172E-02	1.172E-02	1.172E-02	1.172E-02	1.172E-02
U235	2.536E-05	2.536E-05	2.536E-05	2.536E-05	2.536E-05	2.536E-05	2.537E-05	2.537E-05	2.537E-05	2.537E-05
U236	1.331E-04	1.331E-04	1.331E-04	1.331E-04	1.331E-04	1.331E-04	1.331E-04	1.331E-04	1.331E-04	1.332E-04
U237	7.593E-07	5.969E-07	4.692E-07	3.688E-07	2.899E-07	1.792E-07	8.703E-08	4.227E-08	2.053E-08	7.840E-09
U238	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04	1.076E-04
NP237	2.661E-05	2.667E-05	2.677E-05	2.689E-05	2.703E-05	2.735E-05	2.789E-05	2.845E-05	2.902E-05	2.978E-05
PU236	4.654E-07	1.380E-07	4.092E-08	1.214E-08	3.600E-09	3.190E-10	1.095E-11	2.914E-12	2.704E-12	2.698E-12
PU237	3.435E-18	3.019E-30	2.653E-42	2.332E-54	2.050E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	4.071E-02	3.913E-02	3.762E-02	3.616E-02	3.476E-02	3.212E-02	2.854E-02	2.535E-02	2.252E-02	1.923E-02
PU239	9.380E-02	9.379E-02	9.377E-02	9.376E-02	9.374E-02	9.372E-02	9.368E-02	9.364E-02	9.360E-02	9.354E-02
PU240	4.191E-02	4.188E-02	4.186E-02	4.184E-02	4.182E-02	4.177E-02	4.171E-02	4.164E-02	4.157E-02	4.149E-02
PU241	3.095E+00	2.433E+00	1.913E+00	1.504E+00	1.182E+00	7.303E-01	3.548E-01	1.723E-01	8.370E-02	3.196E-02
PU242	1.020E-05	1.020E-05	1.020E-05	1.020E-05	1.020E-05	1.020E-05	1.020E-05	1.019E-05	1.019E-05	1.019E-05
PU244	1.453E-13	1.453E-13	1.453E-13	1.453E-13	1.453E-13	1.453E-13	1.453E-13	1.453E-13	1.453E-13	1.453E-13
AM241	2.945E-02	5.118E-02	6.804E-02	8.107E-02	9.109E-02	1.046E-01	1.144E-01	1.177E-01	1.178E-01	1.158E-01
AM242M	2.030E-05	1.984E-05	1.939E-05	1.895E-05	1.853E-05	1.770E-05	1.653E-05	1.544E-05	1.442E-05	1.316E-05
AM242	2.019E-05	1.974E-05	1.929E-05	1.886E-05	1.843E-05	1.761E-05	1.645E-05	1.536E-05	1.434E-05	1.309E-05
AM243	1.032E-05	1.031E-05	1.031E-05	1.030E-05	1.030E-05	1.029E-05	1.028E-05	1.026E-05	1.025E-05	1.023E-05
CM242	6.346E-05	1.635E-05	1.596E-05	1.560E-05	1.525E-05	1.457E-05	1.360E-05	1.270E-05	1.186E-05	1.083E-05
CM243	5.221E-06	4.623E-06	4.094E-06	3.625E-06	3.210E-06	2.517E-06	1.748E-06	1.213E-06	8.425E-07	5.180E-07

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	**5**	**10**	**15**	**20**	**25**	**35**	**50**	**65**	**80**	**100**
CM244	1.647E-04	1.360E-04	1.123E-04	9.277E-05	7.661E-05	5.225E-05	2.943E-05	1.657E-05	9.334E-06	4.341E-06
CM245	2.442E-09	2.441E-09	2.440E-09	2.439E-09	2.438E-09	2.436E-09	2.433E-09	2.430E-09	2.427E-09	2.423E-09
CM246	1.230E-10	1.229E-10	1.228E-10	1.227E-10	1.226E-10	1.225E-10	1.222E-10	1.219E-10	1.217E-10	1.213E-10
CM247	4.224E-17	4.224E-17	4.224E-17	4.224E-17	4.224E-17	4.224E-17	4.224E-17	4.224E-17	4.224E-17	4.224E-17
*SUBTOTAL***	1.054E+02	6.580E+01	5.169E+01	4.316E+01	3.706E+01	2.832E+01	1.957E+01	1.372E+01	9.692E+00	6.151E+00
TOTAL***	1.054E+02	6.580E+01	5.170E+01	4.317E+01	3.706E+01	2.832E+01	1.957E+01	1.372E+01	9.692E+00	6.152E+00

* 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 12

Fuel-Specific Source Term Calculations
Advanced Test 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 generic source term for the Advanced Test Reactor spent nuclear fuel elements currently stored at the INEEL. The data sources for the analysis are documented in Reference 1 and the INEEL calculational methodology is described in detail in Reference 2.

Advanced Test Reactor Data

The Advanced Test Reactor (ATR) is a 250-MW_{th} rated light-water reactor designed specifically to study the effects of intense radiation on reactor fuels and materials. The core contains nine individual test irradiation flux traps in a 3 × 3 array within a four-leaf clover or serpentine driver core configuration as shown in Figure 1. The serpentine driver core is composed of 40 high-enriched, 48-in. active length U-Al_x plate-type fuel elements. The core driver elements are light water-cooled and beryllium-reflected. Hafnium absorber drums located in the beryllium reflector coupled with hafnium shim rods control the local power levels in each quadrant of the core. The beryllium reflector is contained within an aluminum tank, and the entire reactor core is enclosed in a stainless steel reactor pressure vessel.

Each driver fuel element contains 19 curved aluminum-clad fuel plates. Figure 2 shows the geometrical configuration of a fuel element along with pertinent dimensions. The fuel meat is an intermetallic uranium/aluminum compound with each successive plate (wider arc plate) containing proportionally more uranium (Table 1).

In a fresh ATR element, the uranium enrichment is nominally 93.15 wt% U-235. However, for the source term calculations here, in order to maximize the production of higher order actinides, the maximum U-234 and U-236 impurity concentrations have been used that results in an effective enrichment of approximately 92 wt%. This assumption is not conservative with regard to criticality safety.

There are two different types of aluminum used in the ATR elements. One is a high purity Aluminum-1100 and is used exclusively in the fuel meat. The other is a lower purity Aluminum-6061T and is used everywhere else in the fuel element (clad, end boxes, side plates). For the source term calculations, all aluminum in the ATR element is assumed to be Aluminum-6061T. This results in slightly higher impurity concentrations, which in turn produces slightly higher activation or a slightly more conservative source term. Table 2 lists the impurities and their concentrations (Reference 3).

The data below give specific dimensions, materials, loadings, densities, enrichment, etc. for the ATR driver element used in the burnup calculation for the source term generation.

Fuel Element:

Fuel Meat:	U-Al_x Enriched uranium in Aluminum-1100 Average Density = ~4.00 g/cc
Clad:	Aluminum-6061T Density = 2.70 g/cc
Loading:	1075.0 g/element U-235 BOL (nominal) 69.93 g/element U-238 BOL (nominal) 13.87 g/element U-234 BOL (nominal) 8.09 g/element U-236 BOL (nominal) 92.13% effective enrichment U-235 BOL (calculated) 93.15% nominal enrichment U-235 BOL (ATR)

Active Fuel Length:	48.0 in.
Fuel Element Length:	66.0 in. (5.5 ft end-to-end of the end boxes)
Structural Materials:	2,797.36 g/element aluminum side plates 1,174.32 g/element aluminum in the fuel meat 3,766.74 g/element aluminum clad 1,200.00 g/element upper/lower aluminum end boxes 8,938.42 g/element total aluminum

Core Coolant Water Temperature:
Inlet: <125°F
Outlet: 160°F (average)

Core Coolant Water Pressure:
Inlet: 355 psi (guage)
Outlet: 255 psi

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single ATR driver fuel element. In addition, for the ORIGEN2 (Reference 4) depletion calculation, conservative and detailed impurity concentrations were added for aluminum (Al-1100 and Al-6061T) based on the estimated aluminum masses including the cladding, fuel meat, side plates, and end boxes.

Burnup

The burnup chosen for this template is based on a 35.95% burnup of the initial U-235. This burnup is equivalent to 367.2 MWd, 314,683 MWd/MTU, and 463.3 g U-235 depleted per element and represents the upper end of typical ATR fuel element burnups. The assumption of maximum burnup is conservative for the buildup of fission products, activation products, and minor actinides in the source term and nonconservative with regard to criticality safety.

Cross-Section Development

The neutron cross sections used in the burnup or depletion calculation for the source term generation of a single ATR fuel element are based on a special ORIGEN2 cross section library developed for beginning-of-life (BOL) conditions. These cross sections are spectrally and spatially weighted over an ATR element and are used extensively in ATR depletion calculations.

The special ATR ORIGEN2 library cross sections have recently been compared to cross sections independently generated using the INEEL Monte Carlo method (Reference 2). The development process utilized a full model of the ATR core and depleted an element in position No.2 in the Northeast reactor lobe (NOTE: in Figure 1, the NORTH direction vector is towards the top of the page). Both cross section sets were found to be in excellent agreement at BOL. The INEEL Monte Carlo cross sections were further calculated as a function of burnup and were found to be relatively insensitive to burnups approaching one half the maximum burnup (367.2 MWd), giving further justification to using the BOL cross sections to perform the depletion analysis.

Advanced Test Reactor Single Element Exposure History

Table 3 summarizes the hypothetical power or exposure history used in the burnup or source term calculations for a single ATR fuel element. Although the three cycles, 15-day or 30-day shutdowns, and element powers are hypothetical, they are based on both typical and conservative assumptions.

In general, typical ATR elements are in the core over their multiyear lifetime for 2–5 cycles at variable cycle lengths and power levels. Cycle lengths typically range from 20–45 days, but there are some infrequent shorter cycles. Also, typical ATR elements may operate at different power levels from cycle-to-cycle depending on core position, element reactivity, and core lobe power splits. For example, elements around the serpentine configuration may range anywhere in power from 1–10 MW_{th}. Actual elements normally experience much longer cooling times between cycles (months to years), rather than the relatively short 15 or 30-days between cycles used here for the hypothetical power history. Hence, the hypothetical three cycles followed by either a 15-day or 30-day shutdown will add conservatism to the source term calculation.

The hypothetical element powers over the three cycles were intentionally selected to produce a very high burnup element. In fact, the 3-cycle exposure results in a 367.2 MWd total accumulated burnup for the single ATR element. When compared to current ATR spent fuel inventory records, the hypothetical burnup here represents a maximum exposure or maximum burnup element. Typical ATR end-of-life (EOL) elements do not exceed this exposure; therefore, this hypothetical power history for a single ATR element is bounding for ATR spent fuel elements currently in the spent fuel inventory. The assumption of maximum burnup is conservative for the buildup of fission products, activation products, and minor actinides in the source term and nonconservative with regard to criticality safety.

Burnup Calculation

The ORIGEN2 computer code was used to perform the depletion or burnup calculation for the ATR driver fuel element. The radionuclide inventory or source term template that follows is for a single ATR driver fuel element. The fuel element component masses and impurities (fuel meat, uranium, clad, end fixtures), neutron cross sections, burnup, and hypothetical power history and power level as discussed above are input data for the ORIGEN2 calculation. The radionuclide concentrations are given as a function of decay 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. Advance Test Reactor drawings: 401570, 401571, 401572, 401573, 401574, 401575, 401576, and 401577.

2. 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.

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

4. 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. ATR element plate dimensions and uranium loading.

Plate No.	Plate Thickness (in.)	Fuel Meat Thickness (in.)	Clad Thickness (in.)	Inside Fuel Meat Radius (cm)	U-235 Loading (g/plate)
19	0.1	0.02	0.04	13.68	52.6
18	0.05	0.02	0.015	13.29	53.8
17	0.05	0.02	0.015	12.97	65.9
16	0.05	0.02	0.015	12.64	64.0
15	0.05	0.02	0.015	12.32	76.3
14	0.05	0.02	0.015	11.99	73.8
13	0.05	0.02	0.015	11.67	71.4
12	0.05	0.02	0.015	11.34	69.0
11	0.05	0.02	0.015	11.02	66.6
10	0.05	0.02	0.015	10.69	64.2
9	0.05	0.02	0.015	10.37	61.8
8	0.05	0.02	0.015	10.04	59.4
7	0.05	0.02	0.015	9.72	57.0
6	0.05	0.02	0.015	9.39	54.6
5	0.05	0.02	0.015	9.07	52.1
4	0.05	0.02	0.015	8.75	40.4
3	0.05	0.02	0.015	8.42	38.7
2	0.05	0.02	0.015	8.09	29.1
1	0.08	0.02	0.03	7.73	24.3

Table 2. ATR 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 3. Hypothetical power history for a maximum burnup ATR driver fuel element.

Duration (days)	Cumulative Duration (days)	Time-Averaged Power (MW_{th})
15	15	10.0
15	30	0.0
30	60	5.0
30	90	0.0
30	120	2.4
1825	1945	0.0
1825	3770	0.0
1825	5595	0.0
1825	7420	0.0
1825	9245	0.0
3650	12895	0.0
5475	18370	0.0
5475	23845	0.0
5475	29320	0.0
7300	36620	0.0

The three-cycle exposure in the table above is followed by ten dates representing 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 or decay times designated for the template methodology.

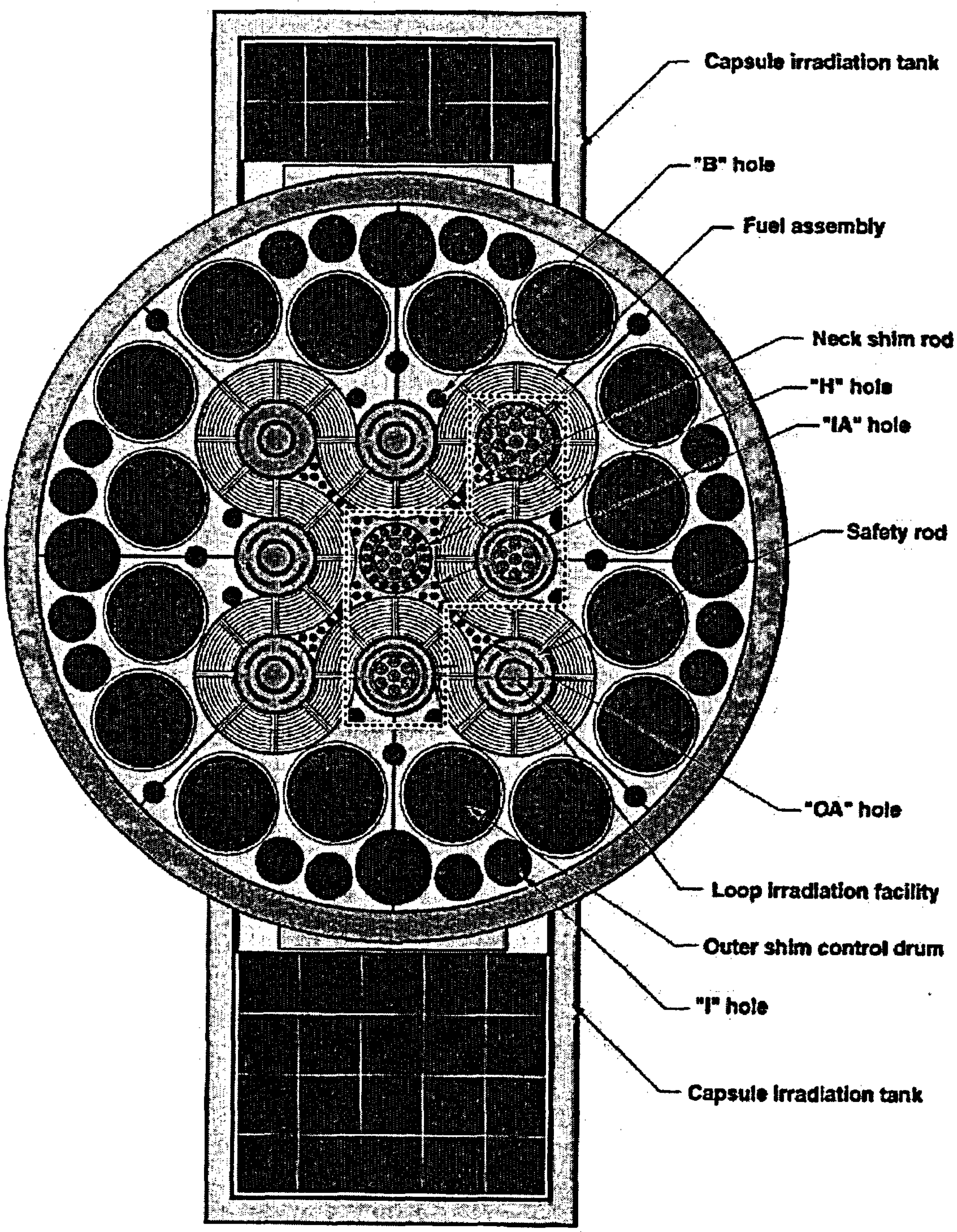

Figure 1. Cross-sectional view of the Advanced Test Reactor core.

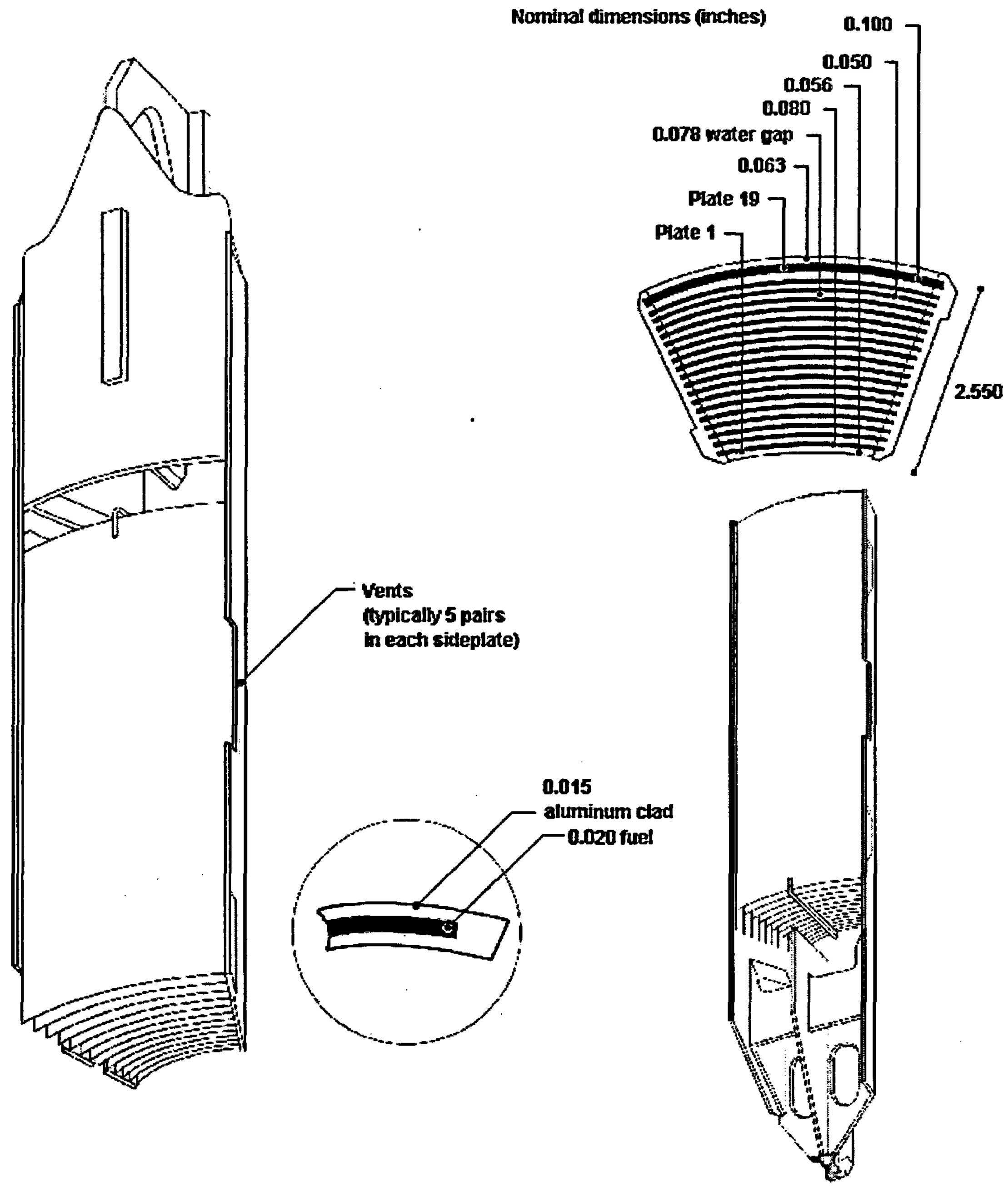

Figure 2. The configuration of an Advanced Test Reactor fuel element.

Advanced Test Reactor Element

Aluminum Cladding, 60 to 100% Enriched U-235 Fuel

Reactor Moderator/Coolant:	Light Water
Fuel Meat:	U-Al_x
Clad:	Aluminum
Burnup:	367.2 MWd/element (maximum element burnup)
Burnup:	35.95 % U-235 burnup (fissioned)
Burnup:	43.10% U-235 depletion (fissioned and transmuted)
Basis of Calculation:	Single element
BOL U-235:	1075.00 g U-235 per element
BOL U-238:	69.93 g U-238 per element
BOL U-234:	13.87 g U-234 per element
BOL U-236:	8.09 g U-236 per element
BOL Total U per element:	1166.89 g U per element
BOL Fuel Enrichment:	92.13 wt%

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H-3	3.892E+00	2.940E+00	2.221E+00	1.678E+00	1.267E+00	7.233E-01	3.118E-01	1.344E-01	5.796E-02	1.887E-02
BE-10	3.600E-08	3.600E-08	3.600E-08	3.600E-08	3.600E-08	3.600E-08	3.600E-08	3.600E-08	3.600E-08	3.600E-08
C-14	2.102E-06	2.101E-06	2.099E-06	2.098E-06	2.097E-06	2.095E-06	2.091E-06	2.087E-06	2.083E-06	2.079E-06
CL-36	4.819E-30	4.819E-30	4.819E-30	4.819E-30	4.819E-30	4.819E-30	4.819E-30	4.818E-30	4.818E-30	4.818E-30
CR-51	3.839E-18	5.715E-38	8.507E-58	1.266E-77	1.885E-97	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN-54	4.247E-02	7.414E-04	1.294E-05	2.259E-07	3.944E-09	1.202E-12	6.396E-18	3.403E-23	1.811E-28	1.682E-35
FE-55	2.476E+00	6.535E-01	1.725E-01	4.552E-02	1.201E-02	8.369E-04	1.539E-05	2.829E-07	5.201E-09	2.524E-11
FE-59	1.368E-12	8.464E-25	5.238E-37	3.241E-49	2.006E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO-60	1.403E-02	7.270E-03	3.768E-03	1.953E-03	1.012E-03	2.719E-04	3.786E-05	5.271E-06	7.338E-07	5.295E-08
NI-59	4.667E-04	4.667E-04	4.667E-04	4.667E-04	4.666E-04	4.666E-04	4.665E-04	4.665E-04	4.664E-04	4.663E-04
NI-63	7.098E-02	6.835E-02	6.583E-02	6.340E-02	6.105E-02	5.663E-02	5.058E-02	4.518E-02	4.035E-02	3.471E-02
ZN-65	2.029E-01	1.134E-03	6.335E-06	3.540E-08	1.978E-10	6.174E-15	1.077E-21	1.878E-28	3.276E-35	3.193E-44
SE-79	4.750E-03	4.750E-03	4.749E-03	4.749E-03	4.749E-03	4.748E-03	4.747E-03	4.747E-03	4.746E-03	4.745E-03
KR-85	1.050E+02	7.603E+01	5.504E+01	3.984E+01	2.884E+01	1.512E+01	5.734E+00	2.176E+00	8.254E-01	2.267E-01
RB-87	3.188E-07	3.188E-07	3.188E-07	3.188E-07	3.188E-07	3.188E-07	3.188E-07	3.188E-07	3.188E-07	3.188E-07
SR-89	1.021E-06	1.348E-17	1.780E-28	2.350E-39	3.103E-50	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR-90	1.010E+03	8.966E+02	7.961E+02	7.068E+02	6.276E+02	4.947E+02	3.463E+02	2.424E+02	1.696E+02	1.054E+02
Y-90	1.010E+03	8.968E+02	7.963E+02	7.070E+02	6.277E+02	4.948E+02	3.463E+02	2.424E+02	1.697E+02	1.054E+02
Y-91	3.816E-05	1.555E-14	6.334E-24	2.580E-33	1.051E-42	1.745E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR-93	2.445E-02	2.445E-02	2.445E-02	2.445E-02	2.445E-02	2.445E-02	2.445E-02	2.445E-02	2.445E-02	2.445E-02

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	**5**	**10**	**15**	**20**	**25**	**35**	**50**	**65**	**80**	**100**
ZR-95	2.534E-04	6.565E-13	1.701E-21	4.407E-30	1.142E-38	7.665E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-93M	5.422E-03	9.424E-03	1.253E-02	1.493E-02	1.680E-02	1.936E-02	2.142E-02	2.239E-02	2.283E-02	2.308E-02
NB-94	2.626E-07	2.626E-07	2.626E-07	2.625E-07	2.625E-07	2.624E-07	2.622E-07	2.621E-07	2.620E-07	2.618E-07
NB-95	5.625E-04	1.458E-12	3.776E-21	9.784E-30	2.535E-38	1.702E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-95M	1.880E-06	4.870E-15	1.262E-23	3.269E-32	8.471E-41	5.686E-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	1.551E-01	1.551E-01	1.551E-01	1.551E-01	1.551E-01	1.551E-01	1.551E-01	1.551E-01	1.551E-01	1.551E-01
RU-103	5.452E-10	5.639E-24	5.832E-38	6.031E-52	6.237E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU-106	6.997E+01	2.253E+00	7.253E-02	2.335E-03	7.519E-05	7.794E-08	2.601E-12	8.683E-17	2.898E-21	3.114E-27
RH-103M	4.915E-10	5.083E-24	5.257E-38	5.437E-52	5.623E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH-106	6.997E+01	2.253E+00	7.253E-02	2.335E-03	7.519E-05	7.794E-08	2.601E-12	8.683E-17	2.898E-21	3.114E-27
PD-107	1.830E-04	1.830E-04	1.830E-04	1.830E-04	1.830E-04	1.830E-04	1.830E-04	1.830E-04	1.830E-04	1.830E-04
AG-110	3.393E-04	2.148E-06	1.360E-08	8.609E-11	5.450E-13	2.184E-17	5.542E-24	1.406E-30	3.568E-37	5.732E-46
AG-110M	2.551E-02	1.615E-04	1.022E-06	6.473E-09	4.098E-11	1.642E-15	4.167E-22	1.057E-28	2.683E-35	4.310E-44
AG-111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-113M	1.215E-01	9.579E-02	7.555E-02	5.959E-02	4.699E-02	2.923E-02	1.434E-02	7.035E-03	3.451E-03	1.335E-03
CD-113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-115M	1.133E-11	5.431E-24	2.604E-36	1.248E-48	5.983E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114	1.089E-13	8.740E-25	7.015E-36	5.631E-47	4.520E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114M	1.138E-13	9.132E-25	7.330E-36	5.883E-47	4.723E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-115M	7.963E-16	3.817E-28	1.830E-40	8.771E-53	4.205E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-119M	2.773E-02	1.588E-04	9.092E-07	5.207E-09	2.981E-11	9.773E-16	1.835E-22	3.444E-29	6.466E-36	6.950E-45
SN-121M	1.096E-03	1.022E-03	9.539E-04	8.901E-04	8.305E-04	7.229E-04	5.873E-04	4.770E-04	3.875E-04	2.937E-04
SN-123	3.516E-03	1.962E-07	1.095E-11	6.112E-16	3.411E-20	1.063E-28	1.848E-41	3.212E-54	5.585E-67	5.419E-84
SN-125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-126	4.250E-03	4.250E-03	4.250E-03	4.250E-03	4.250E-03	4.249E-03	4.249E-03	4.248E-03	4.248E-03	4.247E-03
SB-124	8.220E-09	6.144E-18	4.593E-27	3.432E-36	2.566E-45	1.433E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB-125	2.311E+01	6.618E+00	1.896E+00	5.428E-01	1.555E-01	1.275E-02	2.995E-04	7.038E-06	1.654E-07	1.112E-09
SB-126	5.950E-04	5.950E-04	5.950E-04	5.950E-04	5.949E-04	5.949E-04	5.948E-04	5.948E-04	5.947E-04	5.946E-04
SB-126M	4.250E-03	4.250E-03	4.250E-03	4.250E-03	4.250E-03	4.249E-03	4.249E-03	4.248E-03	4.248E-03	4.247E-03
TE-123M	4.118E-07	1.057E-11	2.714E-16	6.966E-21	1.788E-25	1.178E-34	1.994E-48	3.372E-62	5.705E-76	2.477E-94
TE-125M	5.639E+00	1.614E+00	4.625E-01	1.325E-01	3.794E-02	3.112E-03	7.309E-05	1.717E-06	4.033E-08	2.714E-10
TE-127	2.511E-03	2.290E-08	2.089E-13	1.905E-18	1.738E-23	1.445E-33	1.097E-48	8.322E-64	6.314E-79	4.369E-99
TE-127M	2.563E-03	2.338E-08	2.132E-13	1.945E-18	1.774E-23	1.476E-33	1.120E-48	8.496E-64	6.446E-79	4.461E-99
TE-129	5.027E-14	2.241E-30	9.993E-47	4.455E-63	1.986E-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
TE-129M	7.723E-14	3.443E-30	1.535E-46	6.844E-63	3.051E-79	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-129	2.765E-04	2.765E-04	2.765E-04	2.765E-04	2.765E-04	2.765E-04	2.765E-04	2.765E-04	2.765E-04	2.765E-04
I-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-134	1.788E+02	3.334E+01	6.217E+00	1.159E+00	2.161E-01	7.511E-03	4.867E-05	3.154E-07	2.044E-09	2.470E-12
CS-135	1.266E-03	1.266E-03	1.266E-03	1.266E-03	1.266E-03	1.266E-03	1.266E-03	1.266E-03	1.266E-03	1.266E-03
CS-136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-137	1.055E+03	9.396E+02	8.372E+02	7.459E+02	6.646E+02	5.275E+02	3.731E+02	2.639E+02	1.866E+02	1.176E+02
BA-136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-137M	9.976E+02	8.889E+02	7.920E+02	7.056E+02	6.287E+02	4.990E+02	3.530E+02	2.496E+02	1.766E+02	1.113E+02
BA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-141	1.251E-12	1.583E-29	2.002E-46	2.532E-63	3.203E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-142	3.290E-07	3.290E-07	3.290E-07	3.290E-07	3.290E-07	3.290E-07	3.290E-07	3.290E-07	3.290E-07	3.290E-07
CE-144	3.914E+02	4.570E+00	5.337E-02	6.232E-04	7.277E-06	9.922E-10	1.580E-15	2.515E-21	4.004E-27	7.445E-35
PR-143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR-144	3.914E+02	4.570E+00	5.337E-02	6.232E-04	7.277E-06	9.922E-10	1.580E-15	2.515E-21	4.005E-27	7.445E-35
PR-144M	4.697E+00	5.484E-02	6.404E-04	7.478E-06	8.732E-08	1.191E-11	1.896E-17	3.018E-23	4.805E-29	8.934E-37
ND-144	1.672E-11	1.686E-11	1.686E-11	1.686E-11	1.686E-11	1.686E-11	1.686E-11	1.686E-11	1.686E-11	1.686E-11
ND-147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-147	9.346E+02	2.496E+02	6.667E+01	1.781E+01	4.756E+00	3.393E-01	6.465E-03	1.232E-04	2.347E-06	1.195E-08
PM-148M	3.851E-11	1.915E-24	9.524E-38	4.737E-51	2.356E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-148	2.169E-12	1.079E-25	5.364E-39	2.668E-52	1.327E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM-145	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM-147	6.700E-08	8.379E-08	8.827E-08	8.947E-08	8.979E-08	8.990E-08	8.991E-08	8.991E-08	8.991E-08	8.991E-08
SM-151	4.154E+00	3.997E+00	3.846E+00	3.701E+00	3.561E+00	3.298E+00	2.938E+00	2.618E+00	2.332E+00	2.000E+00
EU-152	2.244E-02	1.739E-02	1.348E-02	1.045E-02	8.102E-03	4.869E-03	2.268E-03	1.056E-03	4.921E-04	1.777E-04
EU-154	3.013E+01	2.014E+01	1.346E+01	9.001E+00	6.017E+00	2.689E+00	8.034E-01	2.400E-01	7.171E-02	1.432E-02
EU-155	1.437E+01	7.149E+00	3.556E+00	1.769E+00	8.797E-01	2.176E-01	2.678E-02	3.296E-03	4.055E-04	2.482E-05
EU-156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD-153	5.951E-04	3.195E-06	1.716E-08	9.213E-11	4.947E-13	1.427E-17	2.209E-24	3.421E-31	5.297E-38	4.404E-47
TB-160	4.593E-08	1.158E-15	2.921E-23	7.367E-31	1.858E-38	1.182E-53	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL-206	4.206E-15	4.206E-15	4.206E-15	4.206E-15	4.206E-15	4.206E-15	4.206E-15	4.206E-15	4.206E-15	4.206E-15

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

Radionuclide	5	10	15	20	25	35	50	65	80	100
TL-207	1.776E-08	5.329E-08	1.042E-07	1.681E-07	2.432E-07	4.202E-07	7.354E-07	1.090E-06	1.469E-06	1.995E-06
TL-208	1.178E-05	1.618E-05	1.715E-05	1.690E-05	1.628E-05	1.484E-05	1.285E-05	1.112E-05	9.627E-06	7.949E-06
PB-210	1.994E-10	1.381E-09	4.347E-09	9.779E-09	1.826E-08	4.631E-08	1.216E-07	2.430E-07	4.150E-07	7.277E-07
PB-211	1.781E-08	5.344E-08	1.045E-07	1.686E-07	2.439E-07	4.214E-07	7.375E-07	1.093E-06	1.473E-06	2.001E-06
PB-212	3.280E-05	4.502E-05	4.772E-05	4.703E-05	4.530E-05	4.129E-05	3.576E-05	3.095E-05	2.679E-05	2.212E-05
BI-211	1.781E-08	5.344E-08	1.045E-07	1.686E-07	2.439E-07	4.214E-07	7.375E-07	1.093E-06	1.473E-06	2.001E-06
BI-212	3.280E-05	4.502E-05	4.772E-05	4.703E-05	4.530E-05	4.129E-05	3.576E-05	3.095E-05	2.679E-05	2.212E-05
PO-212	2.101E-05	2.885E-05	3.058E-05	3.013E-05	2.902E-05	2.646E-05	2.291E-05	1.983E-05	1.717E-05	1.417E-05
PO-215	1.781E-08	5.344E-08	1.045E-07	1.686E-07	2.439E-07	4.214E-07	7.375E-07	1.093E-06	1.473E-06	2.001E-06
PO-216	3.280E-05	4.502E-05	4.772E-05	4.703E-05	4.530E-05	4.129E-05	3.576E-05	3.095E-05	2.679E-05	2.212E-05
RN-219	1.781E-08	5.344E-08	1.045E-07	1.686E-07	2.439E-07	4.214E-07	7.375E-07	1.093E-06	1.473E-06	2.001E-06
RN-220	3.280E-05	4.502E-05	4.772E-05	4.703E-05	4.530E-05	4.129E-05	3.576E-05	3.095E-05	2.679E-05	2.212E-05
FR-223	2.457E-10	7.370E-10	1.440E-09	2.323E-09	3.361E-09	5.810E-09	1.017E-08	1.507E-08	2.030E-08	2.759E-08
RA-223	1.781E-08	5.344E-08	1.045E-07	1.686E-07	2.439E-07	4.214E-07	7.375E-07	1.093E-06	1.473E-06	2.001E-06
RA-224	3.280E-05	4.502E-05	4.772E-05	4.703E-05	4.530E-05	4.129E-05	3.576E-05	3.095E-05	2.679E-05	2.212E-05
RA-226	3.720E-09	1.397E-08	3.075E-08	5.407E-08	8.392E-08	1.632E-07	3.309E-07	5.570E-07	8.413E-07	1.311E-06
RA-228	3.338E-13	1.098E-12	2.120E-12	3.296E-12	4.563E-12	7.239E-12	1.139E-11	1.558E-11	1.979E-11	2.540E-11
AC-227	1.781E-08	5.341E-08	1.043E-07	1.684E-07	2.435E-07	4.210E-07	7.369E-07	1.092E-06	1.471E-06	1.999E-06
TH-227	1.756E-08	5.271E-08	1.030E-07	1.662E-07	2.405E-07	4.156E-07	7.273E-07	1.078E-06	1.452E-06	1.973E-06
TH-228	3.280E-05	4.500E-05	4.768E-05	4.699E-05	4.526E-05	4.129E-05	3.576E-05	3.095E-05	2.679E-05	2.212E-05
TH-229	3.710E-10	6.921E-10	1.049E-09	1.442E-09	1.871E-09	2.837E-09	4.556E-09	6.598E-09	8.963E-09	1.262E-08
TH-230	3.227E-06	6.258E-06	9.294E-06	1.233E-05	1.538E-05	2.148E-05	3.066E-05	3.986E-05	4.909E-05	6.144E-05
TH-231	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03
TH-232	1.467E-12	2.869E-12	4.271E-12	5.674E-12	7.076E-12	9.880E-12	1.409E-11	1.829E-11	2.250E-11	2.811E-11
TH-234	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05
PA-231	1.880E-07	3.279E-07	4.678E-07	6.077E-07	7.476E-07	1.027E-06	1.446E-06	1.865E-06	2.284E-06	2.843E-06
PA-233	3.506E-03	3.506E-03	3.507E-03	3.508E-03	3.509E-03	3.511E-03	3.516E-03	3.520E-03	3.525E-03	3.531E-03
PA-234M	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05
PA-234	2.851E-08	2.851E-08	2.851E-08	2.851E-08	2.851E-08	2.851E-08	2.851E-08	2.851E-08	2.851E-08	2.851E-08
U-232	4.405E-05	4.829E-05	4.790E-05	4.620E-05	4.420E-05	4.020E-05	3.481E-05	3.013E-05	2.608E-05	2.152E-05
U-233	6.419E-07	7.184E-07	7.950E-07	8.717E-07	9.483E-07	1.102E-06	1.332E-06	1.562E-06	1.793E-06	2.101E-06
U-234	6.735E-02	6.746E-02	6.756E-02	6.765E-02	6.775E-02	6.792E-02	6.816E-02	6.836E-02	6.855E-02	6.877E-02
U-235	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03	1.323E-03
U-236	5.689E-03	5.689E-03	5.689E-03	5.689E-03	5.689E-03	5.689E-03	5.689E-03	5.689E-03	5.689E-03	5.689E-03
U-237	6.195E-06	4.871E-06	3.829E-06	3.011E-06	2.367E-06	1.463E-06	7.110E-07	3.456E-07	1.679E-07	6.417E-08

Radionuclide	DECAY TIMES (years out of core) (Activities* in Ci/element) 5	10	15	20	25	35	50	65	80	100
U-238	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05	2.193E-05
NP-237	3.506E-03	3.506E-03	3.507E-03	3.508E-03	3.509E-03	3.511E-03	3.516E-03	3.520E-03	3.525E-03	3.531E-03
PU-236	2.331E-04	6.917E-05	2.053E-05	6.095E-06	1.811E-06	1.619E-07	6.794E-09	2.741E-09	2.634E-09	2.631E-09
PU-237	8.648E-15	7.746E-27	6.939E-39	6.216E-51	5.568E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU-238	7.546E+00	7.254E+00	6.973E+00	6.703E+00	6.444E+00	5.955E+00	5.290E+00	4.699E+00	4.174E+00	3.565E+00
PU-239	1.573E-01	1.573E-01	1.573E-01	1.572E-01	1.572E-01	1.572E-01	1.571E-01	1.570E-01	1.570E-01	1.569E-01
PU-240	8.960E-02	8.956E-02	8.952E-02	8.948E-02	8.944E-02	8.935E-02	8.922E-02	8.908E-02	8.894E-02	8.875E-02
PU-241	2.525E+01	1.985E+01	1.561E+01	1.227E+01	9.649E+00	5.964E+00	2.899E+00	1.409E+00	6.846E-01	2.616E-01
PU-242	1.334E-04	1.334E-04	1.334E-04	1.334E-04	1.334E-04	1.334E-04	1.334E-04	1.334E-04	1.334E-04	1.334E-04
PU-244	2.396E-11	2.396E-11	2.396E-11	2.396E-11	2.396E-11	2.396E-11	2.396E-11	2.396E-11	2.396E-11	2.396E-11
AM-241	2.337E-01	4.109E-01	5.484E-01	6.548E-01	7.366E-01	8.466E-01	9.272E-01	9.542E-01	9.553E-01	9.390E-01
AM-242M	1.668E-04	1.630E-04	1.594E-04	1.558E-04	1.523E-04	1.455E-04	1.359E-04	1.269E-04	1.185E-04	1.082E-04
AM-242	1.660E-04	1.622E-04	1.586E-04	1.550E-04	1.515E-04	1.448E-04	1.352E-04	1.263E-04	1.179E-04	1.076E-04
AM-243	5.479E-04	5.476E-04	5.474E-04	5.471E-04	5.469E-04	5.464E-04	5.456E-04	5.448E-04	5.441E-04	5.430E-04
CM-242	5.028E-04	1.344E-04	1.312E-04	1.282E-04	1.253E-04	1.197E-04	1.118E-04	1.044E-04	9.752E-05	8.902E-05
CM-243	8.694E-05	7.699E-05	6.818E-05	6.038E-05	5.347E-05	4.193E-05	2.912E-05	2.023E-05	1.405E-05	8.639E-06
CM-244	1.911E-02	1.579E-02	1.304E-02	1.077E-02	8.894E-03	6.067E-03	3.418E-03	1.926E-03	1.085E-03	5.049E-04
CM-245	1.124E-06	1.124E-06	1.123E-06	1.123E-06	1.123E-06	1.122E-06	1.120E-06	1.119E-06	1.118E-06	1.116E-06
CM-246	7.669E-08	7.663E-08	7.657E-08	7.652E-08	7.646E-08	7.635E-08	7.618E-08	7.602E-08	7.585E-08	7.563E-08
CM-247	8.015E-14	8.015E-14	8.015E-14	8.015E-14	8.015E-14	8.015E-14	8.015E-14	8.015E-14	8.015E-14	8.015E-14
SUBTOTAL**	6.336E+03	4.066E+03	3.399E+03	2.961E+03	2.612E+03	2.052E+03	1.438E+03	1.011E+03	7.122E+02	4.473E+02
TOTAL***	6.336E+03	4.066E+03	3.399E+03	2.961E+03	2.612E+03	2.052E+03	1.438E+03	1.011E+03	7.122E+02	4.473E+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 15

Fuel-Specific Source Term Calculations Pathfinder 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 Pathfinder superheater spent nuclear fuel element. The data sources for the analysis are documented in References 1 through 5, and the INEEL calculational methodology is described in detail in Reference 6.

Pathfinder Data

The Pathfinder core consisted of a superheater region of elements surrounded by an annular boiler region of elements. See Figure 1 for a detailed sketch of the superheater fuel element materials and geometry. The geometric detail of the superheater fuel element was fully incorporated into the MCNP geometry model as was the boiler fuel element geometry. Although the boiler fuel element is not considered in the source term calculation, it is part of the partial core model used in the MCNP neutron transport calculation. Figures 2 through 5 show cross-sectional views of the 3-D MCNP geometry model. Figure 2 shows the superheater element. Figure 3 shows the superheater lattice. Figure 4 shows a partial core cross section with superheater elements in the center of the core surrounded by boiler region elements and an outer water reflector region. Figure 5 shows an axial cross-sectional view of the core.

The data below and resulting source term calculation are for an average burnup superheater fuel element.

Superheater Fuel Element:

	Fuel Meat:	UO_2 + 316L Stainless Steel Cermet
		Density = 8.1799 g/cc
	Clad:	316L Stainless Steel
		Density = 8.03 g/cc
	Loading:	120.4 g/element U-235 BOL
		0.05 g/element U-236 BOL
		8.37 g/element U-238 BOL
		Enrichment 93.5% U-235 BOL
		Inner fuel tube: 51.2 g/element U-235 BOL
		Outer fuel tube: 69.2 g/element U-235 BOL
Active Fuel Length:		72.0 in.
Fuel Element Length:		74.5 in.
Boron-Al_2O_3 Poison:		Length = 72.5 in.
		Pellet Radius: 0.511 cm
		Natural boron
		Loading = 1.4808 g boron/rod
		Density of Al_2O_3 = 2.59 g/cc (70% TD)

Boiler Fuel Element:

UO_2 fuel meat
2.2 and 3.2 wt% U-235 enrichment
Upper core half pellet radius = 0.448 cm
Upper core half clad thickness = 0.028 in.
Lower core half pellet radius = 0.400 cm
Lower core half clad thickness = 0.026 in.
UO_2 Density = 10.41 g/cc

Superheater Core Power Fraction:	15% (conservative)
No. of Superheater Elements in the Core:	409 total

Water Temperature:

Boiler Inlet	486°F
Steam Region	626°F

Water Pressure:

Boiler Inlet	642 psia
Steam Region	0.01665 g/cc

Aluminum-6061 per Element:	205.501 g
Stainless Steel-316L per Element:	3228.5 g
Stainless Steel 304 per Element:	456.03 g

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single Pathfinder fuel element. In addition, for the ORIGEN2 (Reference 7) depletion calculation, conservative and detailed impurity concentrations were added for the UO_2, stainless steel 316L, stainless steel 304, and aluminum (Al-6061). Table 1 lists the impurities and their concentrations (References 8 through 12).

Burnup

The burnup chosen for this template is 6.46% U-235 depletion, 6.01 MWd/element, or approximately 7.78 g of U-235 depleted for a single Pathfinder fuel element. This burnup is reasonable for an average superheater element and the depletion accounts for buildup of fission products, activation products, and minor actinides in the source term, but non-conservative with regard to criticality safety, in particular U-235 and U-238 end-of-life concentrations.

Cross-Section Development

The MCNP model was used to develop neutron cross sections specifically for the Pathfinder superheater elements. These cross sections are in turn used in the superheater fuel element ORIGEN2 depletion calculation.

The neutron cross sections used in the burnup or depletion calculation for the source term generation of a single Pathfinder fuel element are based on the methodology described in Reference 6. Cross sections from a standard ORIGEN2 light water reactor library were updated once using the specially developed beginning-of-life (BOL) cross sections for the Pathfinder. The updated cross sections take into account the unique Pathfinder neutron flux spatial and spectral characteristics to ensure accurate calculation of the fission product and actinide production as a function of burnup.

Pathfinder Exposure History

Table 2 summarizes the detailed power or exposure history used in the burnup or source term calculations for a single Pathfinder superheater 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 7) was used to perform the depletion or burnup calculation for the Pathfinder fuel element. The radionuclide inventory or source term template is for a single Pathfinder superheater 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. Lockheed Martin Idaho Technologies Company, *Pathfinder Fuel Summary Report*, INEL/INT-97-00127, February 1997.

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3. Northern States Power Company Pathfinder Atomic Power Plant, *Six Month Report No. 3—May 19, 1967 to November 19, 1967*, NSP-6603, June 27, 1967.

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5. Northern States Power Company Pathfinder Atomic Power Plant, *Six Month Report No. 1—May 19, 1966 to November 19, 1966*, TID-23646. January 9, 1967.

6. 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.

7. 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.

8. A. G. Croft, M. A. Bjerke, G. W. Morrison, and L. M. Petrie, *Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code*, ORNL/TM-6051, Oak Ridge National Laboratory.

9. J. C. Evans et al., *Long-Lived Activation Products in Reactor Materials*, NUREG/CR-3474, August 1984.

10. E. A. Avallone and T. Baumeister III, *MARK'S Standard Handbook for Mechanical Engineers*, Ninth Edition.

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

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

Table 1. Pathfinder fuel assembly material impurity concentrations.

Constituent or Impurity	UO_2 Concentration (ppm)	Stainless Steel-316L Concentration (ppm)	Stainless Steel-304 Concentration (wt%)	Aluminum-6061 Concentration (wt%)
H				0.02143
Li	1	0.18	0.13	
Be				
B	1			
C	89.4	0.03 wt%	0.08 wt%	0.02143
N	25	357	525	
O	134454			0.02143
F	10.7			
Na	15	6	37	
Mg	2			1
Al	16.7	50	200	97.15499
Si	12.1	1 wt%	1 wt%	0.6
P	35	0.045 wt%		
S		0.03 wt%		
Cl	5.3		130	
K		3	3	
Ca	2	14	19	
Sc			0.03	
Ti	1	200	600	0.075
V	3	630	690	
Cr	4	17.3 wt%	18.4 wt%	0.195
Mn	1.7	2 wt%	1.53 wt%	0.075
Fe	18	64.24 wt%	68.99 wt%	0.35
Co	1	1630	2570	
Ni	24	13.2 wt%	10 wt%	0.02143
Cu	1	2900	8150	0.275
Zn	40.3	71	2230	0.125
Ga		60	450	
As		95	1010	
Se		9	70	
Br		2	8	
Rb			10	
Sr		0.23	0.2	
Y		5	5	
Zr		6	20	0.02143
Nb		64	300	
Mo	10	2.16 wt%	5500	

Constituent or Impurity	UO_2 Concentration (ppm)	Stainless Steel-316L Concentration (ppm)	Stainless Steel-304 Concentration (wt%)	Aluminum-6061 Concentration (wt%)
Ag	0.1	5	2	
Cd	25			
In	2			
Sn	4			0.02143
Sb		13	17	
Cs			0.3	
Ba			500	
La		0.2	2.1	
Ce			550	
Pr				
Nd				
Sm		0.2	0.15	
Eu		0.07	0.02	
Gd				
Tb		9	0.71	
Dy			1	
Ho		1	1	
Er				
Tm				
Yb		2	2	
Lu		0.8	0.8	
Hf			2	
Ta				
W	2	218	520	
Tl				
Pb	1	30	139	0.02143
Bi	0.4			
Th			1	
U		5	2	

Table 2. Assumed power or exposure history for a single Pathfinder fuel element.

Duration (days)	Cumulative Duration (days)	Time-Averaged Power (MW_{th})
226	226	1.14E-05
184	410	0.00E+00
65	475	1.14E-05
74	549	2.32E-05
42	591	1.28E-03
31	622	9.62E-04
30	652	1.22E-02
31	683	0.00E+00
31	714	9.46E-03
8	722	3.07E-03
20	742	0.00E+00
2	744	1.06E-02
31	775	2.82E-02
24	799	1.77E-02
10	809	0.00E+00
26	835	2.06E-02
31	866	1.70E-03
30	896	8.03E-03
18	914	8.48E-03
25	939	0.00E+00
1	940	1.20E-02
1	941	0.00E+00
9	950	3.34E-02
2	952	0.00E+00
5	957	3.31E-02
3	960	0.00E+00
1	961	4.85E-02
1	962	0.00E+00
1	963	6.77E-02
1	964	0.00E+00
7	971	4.34E-02
3	974	2.67E-02
1	975	0.00E+00
3	978	4.46E-02
2	980	0.00E+00
4	984	2.17E-02
3	987	0.00E+00
1	988	5.01E-02
3	991	3.69E-02
2	993	4.43E-02
3	996	3.33E-02
4	1000	5.91E-02
1	1001	0.00E+00
1	1002	2.96E-02
4	1006	4.25E-02
1	1007	0.00E+00
7	1014	5.38E-02
1	1015	0.00E+00
3	1018	4.62E-02
2	1020	0.00E+00
3	1023	3.41E-02
4	1027	5.30E-02
3	1030	0.00E+00
4	1034	5.30E-02
1825	2859	0.00E+00
1825	4684	0.00E+00
1825	6509	0.00E+00
1825	8334	0.00E+00
1825	10159	0.00E+00
3650	13809	0.00E+00
5475	19284	0.00E+00
5475	24759	0.00E+00
5475	30234	0.00E+00
7300	37534	0.00E+00

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.

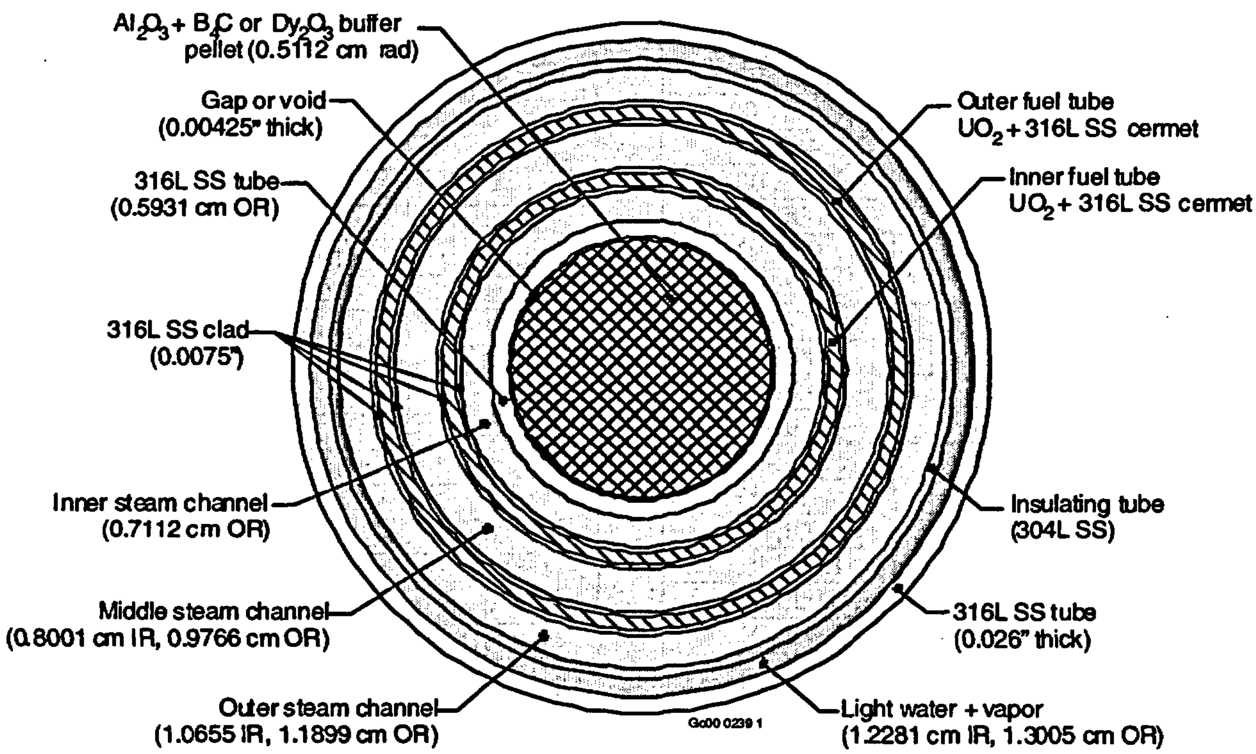

Figure 1. Cross sectional view of an actual Pathfinder superheater element.

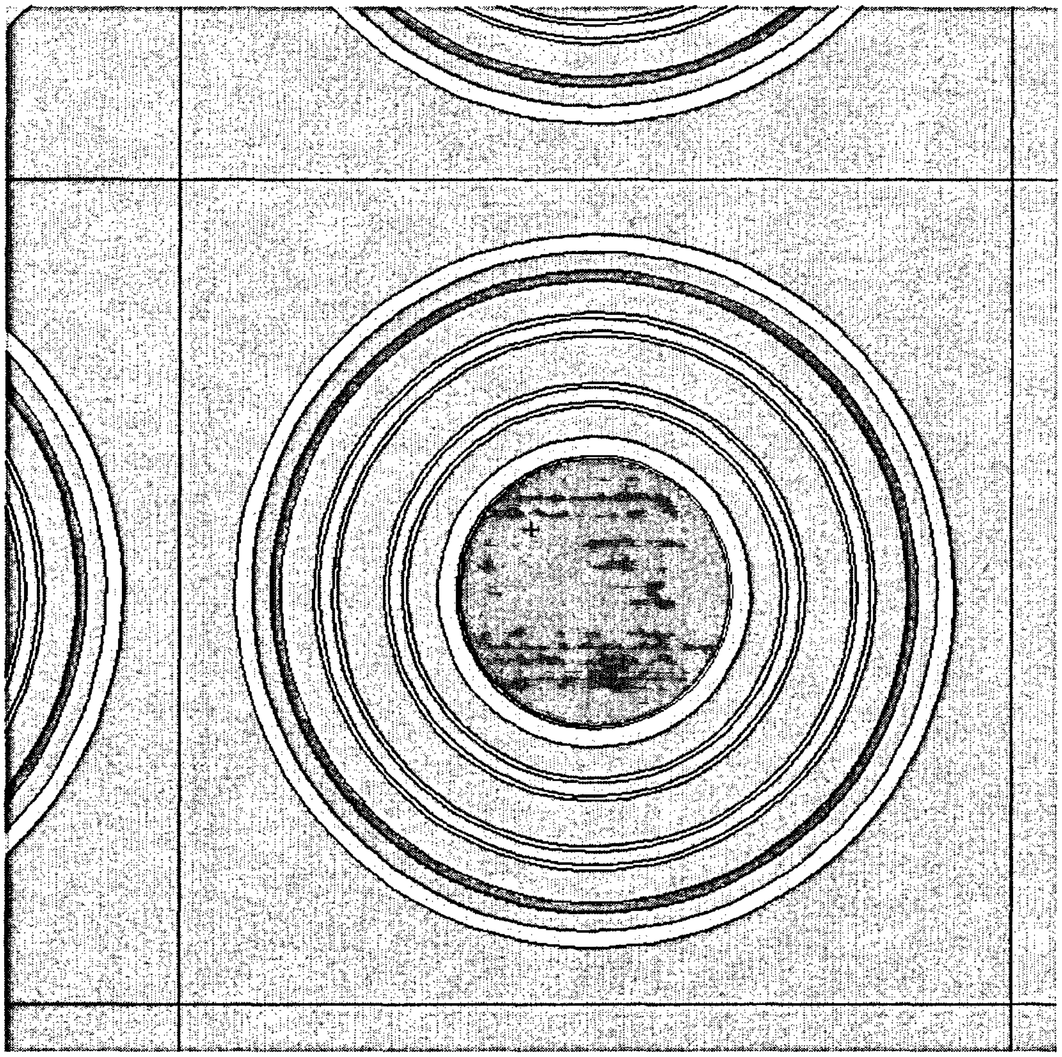

Figure 2. MCNP model representation of a Pathfinder superheater element.

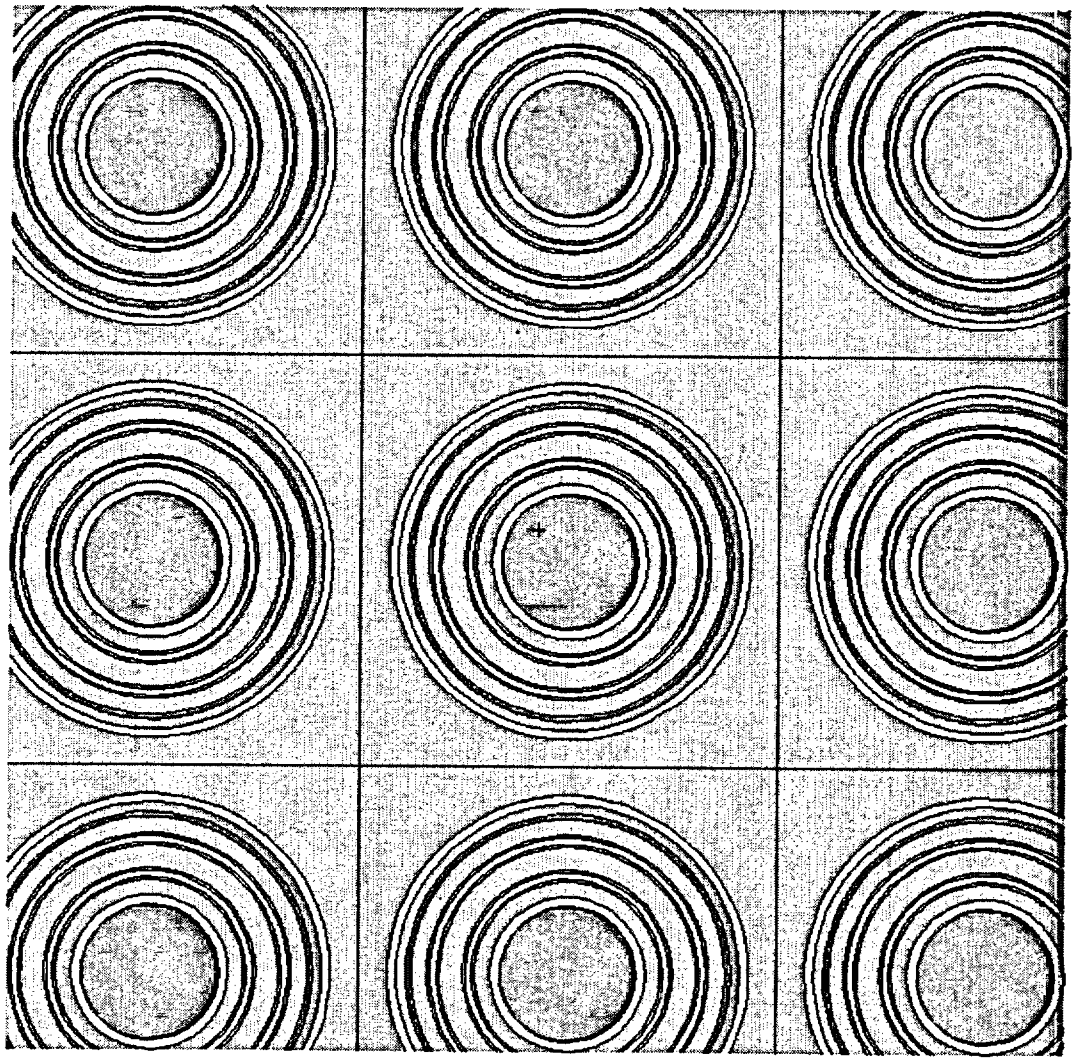

Figure 3. MCNP model representation of a Pathfinder superheater lattice.

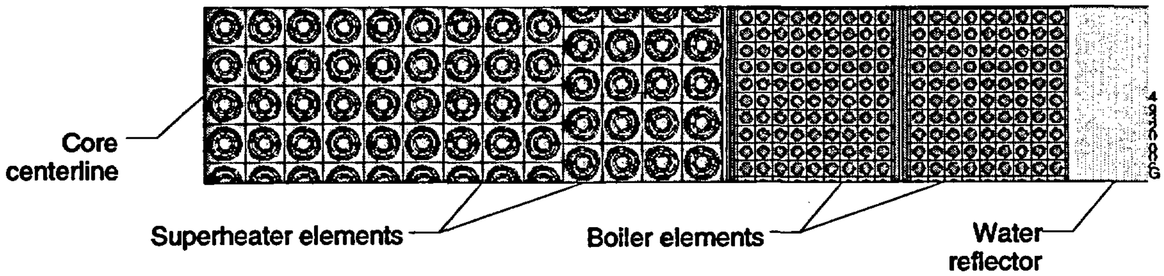

Figure 4. MCNP model representation of a section of the Pathfinder core.

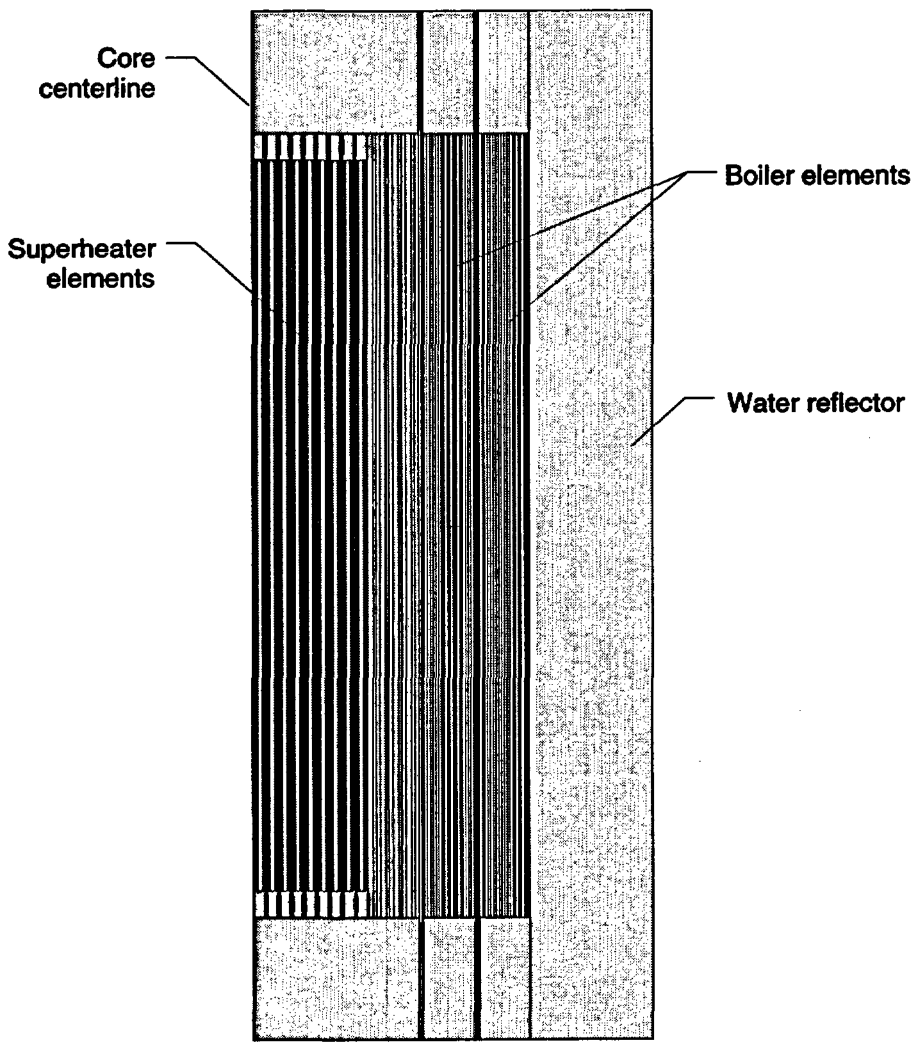

Figure 5. MCNP model axial view of the Pathfinder core.

Pathfinder Superheater Element

Stainless Steel Cladding, 60 to 100% Enriched U-235 Fuel

Reactor Moderator/Coolant:	Light Water
Fuel Meat:	UO_2-316L Stainless Steel (cermet)
Clad:	316L Stainless Steel
Burnup:	6.01 MWd/element (average element burnup)
Burnup:	5.25% U-235 burnup (amount fissioned)
Burnup:	6.46% U-235 depletion (amount fissioned and transmuted)
Core Power Fraction:	15.00% Superheater core power fraction (max assumed)
Basis of Calculation	Single superheater element with double annuli
BOL U-235:	120.40 grams U-235 per element (design basis)
BOL U-238:	8.37 grams U-238 per element
BOL U-234:	0.00 grams U-234 per element
BOL U-236:	0.05 grams U-236 per element
BOL Total U per element:	128.82 grams U per element
BOL Fuel Enrichment:	93.5 wt%

DECAY TIMES (years)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H 3	8.128E-02	6.140E-02	4.638E-02	3.504E-02	2.647E-02	1.511E-02	6.512E-03	2.808E-03	1.210E-03	3.942E-04
BE 10	7.557E-10	7.557E-10	7.557E-10	7.557E-10	7.557E-10	7.557E-10	7.557E-10	7.557E-10	7.557E-10	7.557E-10
C 14	1.388E-03	1.387E-03	1.387E-03	1.386E-03	1.385E-03	1.383E-03	1.381E-03	1.378E-03	1.376E-03	1.372E-03
CL 36	7.369E-06	7.369E-06	7.369E-06	7.369E-06	7.369E-06	7.369E-06	7.369E-06	7.368E-06	7.368E-06	7.368E-06
CR 51	1.481E-17	2.204E-37	3.281E-57	4.884E-77	7.270E-97	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54	2.539E-01	4.432E-03	7.738E-05	1.351E-06	2.358E-08	7.187E-12	3.824E-17	2.035E-22	1.083E-27	1.005E-34
FE 55	2.190E+01	5.780E+00	1.526E+00	4.026E-01	1.063E-01	7.403E-03	1.361E-04	2.502E-06	4.601E-08	2.233E-10
FE 59	9.776E-12	6.049E-24	3.744E-36	2.317E-48	1.434E-60	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60	2.197E+01	1.138E+01	5.900E+00	3.058E+00	1.585E+00	4.258E-01	5.928E-02	8.253E-03	1.149E-03	8.291E-05
NI 59	1.302E-02	1.302E-02	1.302E-02	1.302E-02	1.302E-02	1.302E-02	1.302E-02	1.301E-02	1.301E-02	1.301E-02
NI 63	1.762E+00	1.697E+00	1.634E+00	1.574E+00	1.516E+00	1.406E+00	1.256E+00	1.121E+00	1.002E+00	8.617E-01
ZN 65	2.803E-03	1.566E-05	8.751E-08	4.889E-10	2.732E-12	8.528E-17	1.488E-23	2.595E-30	4.526E-37	4.411E-46
SE 79	7.950E-05	7.950E-05	7.949E-05	7.949E-05	7.948E-05	7.947E-05	7.946E-05	7.945E-05	7.944E-05	7.942E-05
KR 85	1.724E+00	1.248E+00	9.033E-01	6.539E-01	4.734E-01	2.481E-01	9.412E-02	3.571E-02	1.355E-02	3.721E-03
RB 87	5.427E-09	5.427E-09	5.427E-09	5.427E-09	5.427E-09	5.427E-09	5.427E-09	5.427E-09	5.427E-09	5.427E-09
SR 89	1.391E-08	1.837E-19	2.425E-30	3.202E-41	4.228E-52	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	1.674E+01	1.487E+01	1.320E+01	1.172E+01	1.041E+01	8.203E+00	5.742E+00	4.019E+00	2.813E+00	1.748E+00
Y 90	1.675E+01	1.487E+01	1.320E+01	1.172E+01	1.041E+01	8.205E+00	5.743E+00	4.020E+00	2.813E+00	1.748E+00

DECAY TIMES (years)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
Y 91	4.983E-07	2.030E-16	8.270E-26	3.369E-35	1.372E-44	2.278E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93	4.090E-04	4.090E-04	4.090E-04	4.090E-04	4.090E-04	4.090E-04	4.090E-04	4.090E-04	4.090E-04	4.090E-04
ZR 95	3.324E-06	8.612E-15	2.231E-23	5.781E-32	1.498E-40	1.006E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M	9.386E-05	1.601E-04	2.115E-04	2.513E-04	2.822E-04	3.246E-04	3.588E-04	3.747E-04	3.821E-04	3.862E-04
NB 94	3.416E-05	3.415E-05	3.414E-05	3.414E-05	3.413E-05	3.412E-05	3.410E-05	3.409E-05	3.407E-05	3.405E-05
NB 95	7.379E-06	1.912E-14	4.954E-23	1.283E-31	3.325E-40	2.232E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	2.466E-08	6.389E-17	1.655E-25	4.289E-34	1.111E-42	7.459E-60	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93	2.431E-04	2.428E-04	2.426E-04	2.424E-04	2.421E-04	2.416E-04	2.409E-04	2.402E-04	2.395E-04	2.386E-04
TC 99	2.804E-03	2.804E-03	2.804E-03	2.804E-03	2.804E-03	2.804E-03	2.804E-03	2.803E-03	2.803E-03	2.803E-03
RU103	7.847E-12	8.116E-26	8.393E-40	8.681E-54	8.978E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	9.935E-01	3.199E-02	1.030E-03	3.316E-05	1.068E-06	1.107E-09	3.694E-14	1.233E-18	4.115E-23	4.422E-29
RH103M	7.074E-12	7.316E-26	7.567E-40	7.826E-54	8.093E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	9.935E-01	3.199E-02	1.030E-03	3.316E-05	1.068E-06	1.107E-09	3.694E-14	1.233E-18	4.115E-23	4.422E-29
PD107	2.885E-06	2.885E-06	2.885E-06	2.885E-06	2.885E-06	2.885E-06	2.885E-06	2.885E-06	2.885E-06	2.885E-06
AG110	6.377E-06	4.037E-08	2.556E-10	1.618E-12	1.024E-14	4.105E-19	1.042E-25	2.643E-32	6.706E-39	1.077E-47
AG110M	4.795E-04	3.035E-06	1.922E-08	1.216E-10	7.701E-13	3.087E-17	7.832E-24	1.987E-30	5.042E-37	8.100E-46
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	1.828E-03	1.441E-03	1.137E-03	8.966E-04	7.072E-04	4.399E-04	2.158E-04	1.059E-04	5.193E-05	2.009E-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	1.519E-13	7.283E-26	3.492E-38	1.674E-50	8.023E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	4.047E-15	3.249E-26	2.608E-37	2.093E-48	1.680E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	4.229E-15	3.395E-26	2.725E-37	2.187E-48	1.756E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	1.066E-17	5.108E-30	2.448E-42	1.174E-54	5.626E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	1.774E-04	1.015E-06	5.815E-09	3.329E-11	1.906E-13	6.250E-18	1.173E-24	2.202E-31	4.135E-38	4.445E-47
SN121M	1.547E-05	1.444E-05	1.346E-05	1.256E-05	1.173E-05	1.021E-05	8.291E-06	6.735E-06	5.471E-06	4.146E-06
SN123	4.457E-05	2.488E-09	1.388E-13	7.749E-18	4.325E-22	1.347E-30	2.342E-43	4.072E-56	7.080E-69	6.871E-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	6.908E-05	6.908E-05	6.908E-05	6.907E-05	6.907E-05	6.907E-05	6.906E-05	6.905E-05	6.904E-05	6.903E-05
SB124	4.254E-10	3.179E-19	2.377E-28	1.777E-37	1.328E-46	7.420E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	3.523E-01	1.009E-01	2.889E-02	8.276E-03	2.370E-03	1.944E-04	4.567E-06	1.073E-07	2.520E-09	1.696E-11
SB126	9.671E-06	9.671E-06	9.671E-06	9.670E-06	9.670E-06	9.669E-06	9.668E-06	9.667E-06	9.666E-06	9.665E-06
SB126M	6.908E-05	6.908E-05	6.908E-05	6.907E-05	6.907E-05	6.907E-05	6.906E-05	6.905E-05	6.904E-05	6.903E-05
TE123M	1.573E-08	4.038E-13	1.037E-17	2.662E-22	6.832E-27	4.502E-36	7.616E-50	1.288E-63	2.180E-77	9.464E-96
TE125M	8.595E-02	2.462E-02	7.049E-03	2.019E-03	5.782E-04	4.743E-05	1.114E-06	2.618E-08	6.148E-10	4.136E-12
TE127	3.223E-05	2.940E-10	2.681E-15	2.446E-20	2.231E-25	1.856E-35	1.408E-50	1.068E-65	8.106E-81	5.609-101
TE127M	3.291E-05	3.001E-10	2.738E-15	2.497E-20	2.277E-25	1.894E-35	1.437E-50	1.091E-65	8.275E-81	5.727-101

	DECAY TIMES (years) (Activities* in Ci/element)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
TE129	7.493E-16	3.341E-32	1.490E-48	6.641E-65	2.961E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	1.151E-15	5.132E-32	2.288E-48	1.020E-64	4.549E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	4.399E-06	4.399E-06	4.399E-06	4.399E-06	4.399E-06	4.399E-06	4.399E-06	4.399E-06	4.399E-06	4.399E-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	4.331E-01	8.075E-02	1.505E-02	2.807E-03	5.233E-04	1.819E-05	1.179E-07	7.638E-10	4.950E-12	5.981E-15
CS135	1.822E-04	1.822E-04	1.822E-04	1.822E-04	1.822E-04	1.822E-04	1.822E-04	1.822E-04	1.822E-04	1.822E-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	1.743E+01	1.553E+01	1.384E+01	1.233E+01	1.099E+01	8.721E+00	6.168E+00	4.362E+00	3.085E+00	1.944E+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	1.649E+01	1.469E+01	1.309E+01	1.166E+01	1.039E+01	8.250E+00	5.835E+00	4.127E+00	2.919E+00	1.839E+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	1.915E-14	2.422E-31	3.064E-48	3.875E-65	4.901E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	6.111E-09	6.111E-09	6.111E-09	6.111E-09	6.111E-09	6.111E-09	6.111E-09	6.111E-09	6.111E-09	6.111E-09
CE144	5.635E+00	6.580E-02	7.683E-04	8.971E-06	1.048E-07	1.428E-11	2.274E-17	3.621E-23	5.765E-29	1.072E-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	5.635E+00	6.580E-02	7.683E-04	8.972E-06	1.048E-07	1.428E-11	2.274E-17	3.621E-23	5.765E-29	1.072E-36
PR144M	6.762E-02	7.896E-04	9.220E-06	1.077E-07	1.257E-09	1.714E-13	2.729E-19	4.345E-25	6.918E-31	1.286E-38
ND144	2.524E-13	2.544E-13	2.545E-13	2.545E-13	2.545E-13	2.545E-13	2.545E-13	2.545E-13	2.545E-13	2.545E-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	1.981E-07	1.666E-07	1.371E-07	1.127E-07	9.268E-08	6.267E-08	3.484E-08	1.937E-08	1.077E-08	4.925E-09
PM147	1.929E+01	5.151E+00	1.376E+00	3.675E-01	9.815E-02	7.002E-03	1.334E-04	2.542E-06	4.844E-08	2.465E-10
PM148M	4.160E-13	2.069E-26	1.029E-39	5.116E-53	2.544E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	2.343E-14	1.165E-27	5.795E-41	2.882E-54	1.433E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	8.242E-08	1.997E-09	4.840E-11	1.173E-12	2.842E-14	1.669E-17	2.375E-22	3.379E-27	4.809E-32	1.658E-38
SM147	1.484E-09	1.831E-09	1.923E-09	1.948E-09	1.955E-09	1.957E-09	1.957E-09	1.957E-09	1.957E-09	1.957E-09
SM151	3.133E-01	3.015E-01	2.901E-01	2.792E-01	2.686E-01	2.487E-01	2.216E-01	1.975E-01	1.759E-01	1.508E-01
EU152	8.228E-03	6.379E-03	4.945E-03	3.833E-03	2.972E-03	1.786E-03	8.317E-04	3.875E-04	1.805E-04	6.517E-05
EU154	4.509E-02	3.014E-02	2.015E-02	1.347E-02	9.004E-03	4.024E-03	1.203E-03	3.591E-04	1.073E-04	2.143E-05
EU155	2.772E-01	1.379E-01	6.859E-02	3.412E-02	1.697E-02	4.198E-03	5.166E-04	6.357E-05	7.823E-06	4.788E-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	3.431E-05	1.842E-07	9.892E-10	5.312E-12	2.852E-14	8.225E-19	1.274E-25	1.972E-32	3.054E-39	2.539E-48
TB160	5.925E-08	1.494E-15	3.767E-23	9.502E-31	2.396E-38	1.524E-53	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	1.143E-13	1.143E-13	1.143E-13	1.143E-13	1.143E-13	1.143E-13	1.143E-13	1.143E-13	1.143E-13	1.143E-13

DECAY TIMES (years)
(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TL207	4.620E-09	1.180E-08	2.170E-08	3.394E-08	4.815E-08	8.136E-08	1.400E-07	2.056E-07	2.756E-07	3.727E-07
TL208	1.456E-07	1.606E-07	1.569E-07	1.503E-07	1.434E-07	1.302E-07	1.127E-07	9.753E-08	8.442E-08	6.971E-08
PB210	1.174E-14	4.855E-14	1.414E-13	3.119E-13	5.790E-13	1.467E-12	3.875E-12	7.804E-12	1.344E-11	2.383E-11
PB211	4.633E-09	1.183E-08	2.177E-08	3.403E-08	4.829E-08	8.159E-08	1.404E-07	2.062E-07	2.764E-07	3.738E-07
PB212	4.052E-07	4.469E-07	4.367E-07	4.183E-07	3.990E-07	3.623E-07	3.136E-07	2.714E-07	2.350E-07	1.940E-07
BI211	4.633E-09	1.183E-08	2.177E-08	3.403E-08	4.829E-08	8.159E-08	1.404E-07	2.062E-07	2.764E-07	3.738E-07
BI212	4.052E-07	4.469E-07	4.367E-07	4.183E-07	3.990E-07	3.623E-07	3.136E-07	2.714E-07	2.350E-07	1.940E-07
PO212	2.596E-07	2.863E-07	2.798E-07	2.680E-07	2.557E-07	2.321E-07	2.009E-07	1.739E-07	1.505E-07	1.243E-07
PO215	4.633E-09	1.183E-08	2.177E-08	3.403E-08	4.829E-08	8.159E-08	1.404E-07	2.062E-07	2.764E-07	3.738E-07
PO216	4.052E-07	4.469E-07	4.367E-07	4.183E-07	3.990E-07	3.623E-07	3.136E-07	2.714E-07	2.350E-07	1.940E-07
RN219	4.633E-09	1.183E-08	2.177E-08	3.403E-08	4.829E-08	8.159E-08	1.404E-07	2.062E-07	2.764E-07	3.738E-07
RN220	4.052E-07	4.469E-07	4.367E-07	4.183E-07	3.990E-07	3.623E-07	3.136E-07	2.714E-07	2.350E-07	1.940E-07
FR223	6.392E-11	1.631E-10	3.000E-10	4.690E-10	6.654E-10	1.125E-09	1.936E-09	2.843E-09	3.810E-09	5.153E-09
RA223	4.633E-09	1.183E-08	2.177E-08	3.403E-08	4.829E-08	8.159E-08	1.404E-07	2.062E-07	2.764E-07	3.738E-07
RA224	4.052E-07	4.469E-07	4.367E-07	4.183E-07	3.990E-07	3.623E-07	3.136E-07	2.714E-07	2.350E-07	1.940E-07
RA226	1.202E-13	4.419E-13	9.691E-13	1.704E-12	2.650E-12	5.180E-12	1.061E-11	1.803E-11	2.749E-11	4.335E-11
RA228	2.767E-11	3.664E-11	4.200E-11	4.521E-11	4.713E-11	4.898E-11	4.984E-11	5.008E-11	5.019E-11	5.029E-11
AC227	4.632E-09	1.182E-08	2.174E-08	3.399E-08	4.822E-08	8.151E-08	1.403E-07	2.060E-07	2.761E-07	3.734E-07
TH227	4.570E-09	1.167E-08	2.147E-08	3.356E-08	4.762E-08	8.046E-08	1.385E-07	2.034E-07	2.726E-07	3.686E-07
TH228	4.052E-07	4.465E-07	4.363E-07	4.179E-07	3.987E-07	3.622E-07	3.136E-07	2.714E-07	2.350E-07	1.940E-07
TH229	9.227E-12	1.765E-11	2.614E-11	3.469E-11	4.332E-11	6.076E-11	8.743E-11	1.147E-10	1.426E-10	1.807E-10
TH230	1.017E-10	1.963E-10	2.923E-10	3.897E-10	4.883E-10	6.891E-10	9.987E-10	1.317E-09	1.644E-09	2.090E-09
TH231	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04
TH232	4.988E-11	4.991E-11	4.993E-11	4.995E-11	4.998E-11	5.003E-11	5.010E-11	5.017E-11	5.024E-11	5.034E-11
TH234	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06
PA231	4.030E-08	6.605E-08	9.179E-08	1.175E-07	1.433E-07	1.947E-07	2.719E-07	3.490E-07	4.261E-07	5.289E-07
PA233	6.898E-06	6.898E-06	6.898E-06	6.899E-06	6.900E-06	6.902E-06	6.905E-06	6.908E-06	6.911E-06	6.916E-06
PA234M	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06
PA234	3.635E-09	3.635E-09	3.635E-09	3.635E-09	3.635E-09	3.635E-09	3.635E-09	3.635E-09	3.635E-09	3.635E-09
U232	4.671E-07	4.474E-07	4.271E-07	4.072E-07	3.881E-07	3.526E-07	3.052E-07	2.642E-07	2.287E-07	1.886E-07
U233	1.778E-08	1.793E-08	1.808E-08	1.823E-08	1.838E-08	1.868E-08	1.914E-08	1.959E-08	2.004E-08	2.064E-08
U234	2.089E-06	2.120E-06	2.150E-06	2.179E-06	2.207E-06	2.259E-06	2.330E-06	2.393E-06	2.449E-06	2.515E-06
U235	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04	2.435E-04
U236	9.730E-05	9.730E-05	9.730E-05	9.730E-05	9.730E-05	9.730E-05	9.730E-05	9.730E-05	9.730E-05	9.730E-05
U237	4.464E-07	3.509E-07	2.759E-07	2.169E-07	1.706E-07	1.054E-07	5.123E-08	2.490E-08	1.210E-08	4.623E-09
U238	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06	2.796E-06

DECAY TIMES (years)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
NP237	6.898E-06	6.898E-06	6.898E-06	6.899E-06	6.900E-06	6.902E-06	6.905E-06	6.908E-06	6.911E-06	6.916E-06
PU236	8.420E-08	2.499E-08	7.416E-09	2.201E-09	6.535E-10	5.796E-11	1.950E-12	4.857E-13	4.474E-13	4.463E-13
PU237	6.292E-19	5.636E-31	5.048E-43	4.522E-55	4.051E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	2.248E-03	2.161E-03	2.078E-03	1.997E-03	1.920E-03	1.774E-03	1.576E-03	1.400E-03	1.244E-03	1.062E-03
PU239	4.017E-03	4.016E-03	4.015E-03	4.015E-03	4.014E-03	4.013E-03	4.011E-03	4.010E-03	4.008E-03	4.006E-03
PU240	5.236E-04	5.233E-04	5.230E-04	5.227E-04	5.224E-04	5.219E-04	5.211E-04	5.202E-04	5.194E-04	5.183E-04
PU241	1.820E-02	1.431E-02	1.125E-02	8.843E-03	6.952E-03	4.298E-03	2.089E-03	1.015E-03	4.933E-04	1.885E-04
PU242	1.185E-08	1.185E-08	1.185E-08	1.185E-08	1.185E-08	1.185E-08	1.185E-08	1.185E-08	1.185E-08	1.185E-08
PU244	3.988E-17	3.988E-17	3.988E-17	3.988E-17	3.988E-17	3.988E-17	3.988E-17	3.988E-17	3.988E-17	3.988E-17
AM241	1.696E-04	2.973E-04	3.964E-04	4.730E-04	5.319E-04	6.111E-04	6.692E-04	6.886E-04	6.894E-04	6.776E-04
AM242M	5.862E-08	5.730E-08	5.601E-08	5.475E-08	5.352E-08	5.113E-08	4.776E-08	4.460E-08	4.165E-08	3.803E-08
AM242	5.833E-08	5.702E-08	5.573E-08	5.448E-08	5.325E-08	5.088E-08	4.752E-08	4.438E-08	4.145E-08	3.784E-08
AM243	5.938E-09	5.935E-09	5.932E-09	5.929E-09	5.926E-09	5.921E-09	5.913E-09	5.904E-09	5.896E-09	5.885E-09
CM242	1.147E-07	4.720E-08	4.611E-08	4.507E-08	4.406E-08	4.208E-08	3.930E-08	3.670E-08	3.428E-08	3.129E-08
CM243	3.100E-09	2.745E-09	2.431E-09	2.153E-09	1.907E-09	1.495E-09	1.038E-09	7.212E-10	5.009E-10	3.081E-10
CM244	4.388E-08	3.624E-08	2.993E-08	2.472E-08	2.042E-08	1.393E-08	7.848E-09	4.422E-09	2.491E-09	1.159E-09
CM245	3.390E-13	3.388E-13	3.387E-13	3.386E-13	3.384E-13	3.381E-13	3.377E-13	3.373E-13	3.369E-13	3.364E-13
CM246	2.485E-15	2.483E-15	2.481E-15	2.480E-15	2.478E-15	2.474E-15	2.469E-15	2.463E-15	2.458E-15	2.451E-15
CM247	4.323E-22	4.323E-22	4.323E-22	4.323E-22	4.323E-22	4.323E-22	4.323E-22	4.323E-22	4.323E-22	4.323E-22
Subtotal**	1.493E+02	8.620E+01	6.519E+01	5.390E+01	4.634E+01	3.578E+01	2.516E+01	1.792E+01	1.285E+01	8.322E+00
TOTAL***	1.493E+02	8.621E+01	6.520E+01	5.391E+01	4.634E+01	3.578E+01	2.516E+01	1.792E+01	1.285E+01	8.323E+00

* 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 21

Fuel-Specific Source Term Calculations
LWBR Seed Module

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 Light Water Breeder Reactor (LWBR) spent nuclear fuel seed module. The data sources for the analysis are documented here, and the INEEL calculational methodology is described in detail in Reference 1.

Light Water Breeder Reactor History

The LWBR was a full-scale power production reactor with a core design power rating of 236 MW_{th}. The LWBR core operated over a time period extending from August 26, 1977, to December 2, 1982 (5+ years). The reactor design was conceived by the Westinghouse Atomic Power Development Laboratory (WAPD) and stationed at the Shippingport Atomic Power Station (APS).

The LWBR was a pressurized, light water moderated and cooled thermal reactor with zirconium-clad UO_2-ThO_2 fuel rods (Reference 6). The beginning-of-life (BOL) UO_2 was fully enriched in U-233 (>98%) and provided a total core fissile inventory of approximately 501 kg. In addition, to the UO_2-ThO_2 fuel rods, the entire fueled active core region was reflected radially or circumferentially around the core, and both above and below with ThO_2 fuel rods. The ThO_2 rods in the outer reflector regions and the very large ThO_2 loading in the active core were designed to reduce neutron leakage and breed U-233.

The LWBR core consisted of 39 modules (12 seed, 3 standard blanket Type I, 3 standard/power flattening blanket Type II, 6 standard/power flattening blanket Type III, 9 reflector Type IV, and 6 reflector Type V). These modules were in the core for the full operating period (5+ years). Over its lifetime, the core generated 29,047 EFPH (full power = 236 MW).

The reactor core design was complex; the primary design goal was to maximize the U-233 breeding ratio. The seed modules were moveable; operated on pneumatic pistons for reactivity control, supplanting the need for neutron absorbing control rods. The different fueled module types had different numbers of rods, rod diameters, and lattice pitch. In addition, each module type had complex radial and axial urania/thoria loadings. Reference 6 provides a good description of the different module loadings and geometries. In addition, Reference 6 contains an extensive listing of additional references from which data were obtained and used in the depletion calculations here.

Light Water Breeder Reactor Seed and Reactor Data

The LWBR reactor core and fuel elements are described in some detail in Reference 6. Data from this reference have been used to develop reactor physics models needed to develop neutron cross sections for the fuel depletion and radionuclide inventory analysis.

The LWBR seed modules consisted of 619 fuel rods arranged in 15 hexagonal rings in a Zircaloy-4 hexagonal can. The 9.59-in. flat-to-flat can had a 0.08 in. thickness. The fuel rod meat or fuel pellet material was either a urania-thoria (UO_2-ThO_2) binary composition or a pure thoria (ThO_2) composition. In the seed module, there were two binary compositions with different uranium loadings.

The fuel rod clad was Zircaloy-4. The fuel pellet and pin diameters were 0.262 in. and 0.306 in., respectively. The uranium metal was high enriched at 98.23 wt% U-233 at BOL. For modeling purposes, the thoria properties listed below for the binary pellet are also used for the pure thoria pellet.

The following data provide specific seed module dimensions, materials, densities, enrichment, etc., which are typical for an LWBR seed module (Reference 6). The BOL data below were used in the fuel depletion calculations for the LWBR seed module source term generation.

Seed Module:	15 concentric hexagonal rings of fuel rods	
No. of rods:	619 fuel rods per seed module (631 lattice positions)	
Types of rods:	4 (1 high zone and 3 low zones)	
High Zone rods:	331 per module	
Low Zone rods:	288 per module	
Fuel Pellet Diameter:	0.262 in.	
Fuel Rod Meat Length:	84 in.	
Fuel Rod Pitch:	0.370 in. (hot)	
Uranium Enrichment:	98.23 wt % U-233	
	1.29 wt % U-234	
	0.09 wt % U-235	
	0.02 wt % U-236	
	0.37 wt % U-238	
Binary Fuel Rod Meat:	Urania-thoria (UO_2-ThO_2)	
UO_2 Density:	10.96 g/cc (100% Theoretical Density [TD])	
ThO_2 Density:	10.03 g/cc (100% TD)	
UO_2 Density:	10.52 g/cc (96% TD)	
ThO_2 Density:	9.93 g/cc (99% TD)	
UO_2 Fraction:	4.337 wt% (Low Zone)	
UO_2 Fraction:	5.202 wt% (High Zone)	
U-233 Mass:	16,877.36 g	per seed module
U-234 Mass:	221.64 g	per seed module
U-235 Mass:	15.46 g	per seed module
U-236 Mass:	3.44 g	per seed module
U-238 Mass:	63.57 g	per seed module
Th-232 Mass:	442,731.04 g	per seed module
Clad:	Zircaloy-4	
Clad Pin Outer Diameter:	0.306 in.	
Clad Thickness:	0.022 in.	
Clad Pin Length:	118 in.	
Clad Density:	6.44 g/cc	
Total Zircaloy Mass:	154,237.0 g per module (based conservatively on 631 fuel rods)	

Can Geometry:	Hexagonal
Can Dimensions:	9.59 in. flat-to-flat
	0.08 in. wall thickness
Can Length:	118 in.
Can Material:	Zircaloy-4
Can Density:	6.44 g/cc
Can Mass:	33,114.6 g per module
Total Zircaloy-4 Mass:	187,351.6 g per module
Coolant:	Light water
Coolant Temperature:	531°F
Coolant Pressure:	2000 psig
Coolant Density:	0.6583 g/cc

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for the material components in a single LWBR seed module. In addition, for the ORIGEN2 (Reference 2) depletion calculation, conservative and detailed impurity concentrations were added for the zircaloy clad (Reference 3) and the urania-thoria fuel pellets (Reference 4). Table 1 lists the impurities and corresponding concentrations used in the calculations. Note: The impurities in the thoria pellets were assumed to be the same as in the urania-thoria pellets.

Burnup

The LWBR module burnup or depletion analysis was performed using the ORIGEN2 computer code (Reference 2). Three basic inputs are required to perform this analysis, which include: (1) neutron cross sections specific to the LWBR seed module, (2) power history or average burnup for a single LWBR seed module over its operating lifetime, and (3) BOL seed module fissile and fertile masses. The neutron cross sections are discussed below and were incorporated as an update into a standard ORIGEN2 pressurized water reactor (PWR) cross-section library.

The complete LWBR power history is given in References 4 and 8. These data have been used to derive a simplified power history and burnup for an average seed module. The derived data used in the ORIGEN2 code as input are given in Table 2 in terms of seed module power (MW_{th}) and cumulative or total seed module burnup (MWD).

The fissile and fertile BOL isotopic masses by module are derived from data found in Reference 6 and are given in the "LWBR Seed and Reactor Data" section above. Module masses are based on the following information: (1) total number of fuel rods for that particular module, (2) fuel rod radius and length, (3) number of fuel rods in a given radial loading zone, (4) axial loading step lengths, and (5) radial and axial step percent mass loadings of UO_2 in UO_2-ThO_2. From these data, the BOL uranium isotopic masses and Thorium-232 mass can be determined for each module type. In addition, the oxide fuel impurity masses are also included in the calculations. The impurity levels were obtained from LWBR UO_2-ThO_2 fuel fabrication specifications.

Cross-Section Development

The primary goal of the reactor physics analysis was to develop BOL neutron cross sections specifically for the LWBR seed module. This required that MCNP4B (Reference 5) computer models be developed to represent the module geometry and the complex radial and axial fuel loadings in both the seed and standard blanket modules. The cross sections were required as input data for the ORIGEN2

depletion or burnup calculations. (Note: No standard ORIGEN2 neutron cross-section libraries were available for this unique reactor type). The MCNP neutron cross-section generation methodology is documented in Reference 1, and validation work to support the physics and depletion methodology and its predictive capability is given Reference 7 specifically for the LWBR.

A fully explicit, three-dimensional MCNP4B computer model of the seed, standard blanket, and top and bottom thoria reflectors was developed specifically to generate three BOL cross-section library updates. The model was essentially an infinite lattice model of seed and standard reflector blankets with exact numbers of fuel rods, rod diameters, clad thicknesses, lattice pitch, fuel and clad materials, and the complex radial and axial urania-thoria loadings.

Figure 1 shows an x-y cross-sectional view of the infinite lattice model as drawn by the MCNP4B computer code. The complex geometry and radial binary fuel loading patterns of the UO_2-ThO_2 pellets are fully visible from the color scheme as are the variable rod diameters and pitches between the seed rods and the standard blanket rods. The axially stepped binary and thoria fuel loading patterns for the seed and standard blanket modules are given Reference 6, and although not shown in Figure 1, are appropriately modeled in the MCNP model.

It should be noted that only BOL neutron cross sections were generated and used in the burnup calculations, i.e., burnup-dependent cross sections were not calculated at various time steps throughout the burnup calculations as was done in the validation work. Based on the validation work, good radionuclide inventories could be obtained with BOL cross sections only. Part of the reason for this is because during reactor operation, the seed modules burned their U-233 fissile inventory while the thoria bred new U-233. Therefore, since both the thoria and U-233 inventories remained relatively constant over the LWBR lifetime, the neutron cross sections would be weakly dependent on burnup, and here it is assumed that the BOL cross sections are reasonable estimates over the entire LWBR lifetime.

Light Water Breeder Reactor Seed Module Exposure History

Table 2 summarizes the LWBR single seed module power or exposure history used in the depletion calculations. The power history is based on data in References 4 and 8. 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 LWBR seed module. The seed module 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 LWBR seed module.

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. 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.

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3. Oak Ridge National Laboratory, *Characteristics of Potential Repository Wastes*, DOE/RW-0184-V1-R1, Volume 1, July 1992, Oak Ridge, Tennesee.

4. Bettis Atomic Power Laboratory, *Summary of the Nuclear Design and Performance of the Light water Breeder Reactor (LWBR)*, WAPD-TM-1326, June 1979.

5. "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.

6. G. L. Olson, R. K. McCardell, and D. Illum, *Fuel Summary Report: Shippingport Light Water Breeder Reactor*, INEEL/EXT-98-00799, August 1998.

7. J. W. Sterbentz, *Validation Work to Support the Idaho National Engineering and Environmental Laboratory Calculational Burnup Methodology Using Shippingport Light Water Breeder Reactor (LWBR) Spent Fuel Assay Data*, INEEL/EXT-99-00581, August 1999.

8. Bettis Atomic Power Laboratory, "Shippingport Operations with the LWBR," WAPD-TM-1542, March 1986.

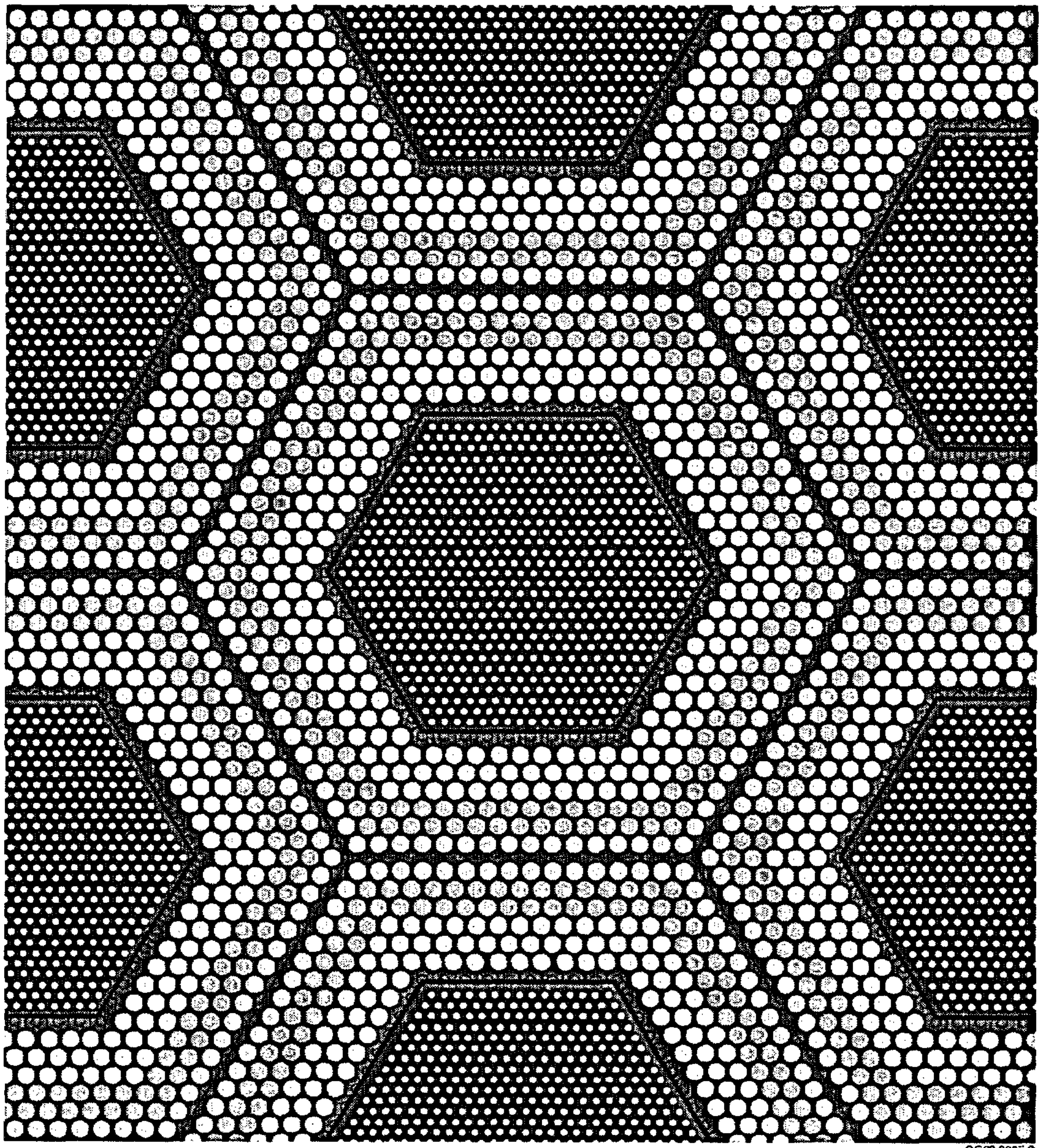

Figure 1. Infinite lattice model representation (MCNP) of the LWBR seed/standard blanket modules.

Table 1. Zircaloy-4 and UO_2-ThO_2 material constituent and impurity concentrations.

Constituent or Impurity	Zircaloy-4 Concentration (wt%)	UO2-ThO2 Concentration (ppm)
H	0.002497	
B	0.00005	1
C	0.026968	200
N	0.00799	50
O	0.094887	134454
Mg		100
Al	0.007491	500
Si	0.011986	300
P	0.009988	
S	0.003496	
Cl		15
Ca		20
Ti	0.004994	20
V	0.004994	25
Cr	0.124851	100
Mn	0.004994	10
Fe	0.224731	300
Co	0.001998	10
Ni	0.006992	200
Cu	0.004994	40
Zn	0.009988	
Zr	97.789992	
Nb	0.006992	
Mo	0.004994	100
Cd	0.000050	
Sn	1.598089	
Sm	0.000999	
Gd	0.000499	
Hf	0.003496	
Ta	0.019976	
W	0.009988	
Hg		1
Pb	0.009988	
Th	0.000699	
U	0.000350	

Table 2. Light Water Breeder Reactor seed module power history.

Operating Start Date	Operating End Date	Operational (days)	Cumulative Operating (days)	Seed Module Power (MW_{th})	Cumulative Burnup (MWD)
26-Aug-77	30-Sep-77	35	35	2.731274	95.59
30-Sep-77	31-Dec-77	92	127	5.971203	644.95
31-Dec-77	31-Mar-78	90	217	7.897081	1355.68
31-Mar-78	30-Jun-78	91	308	5.969611	1898.92
30-Jun-78	30-Sep-78	92	400	6.767031	2521.48
30-Sep-78	31-Dec-78	92	492	7.217013	3185.45
31-Dec-78	31-Mar-79	90	582	7.549443	3864.90
31-Mar-79	30-Jun-79	91	673	0.000000	3864.90
30-Jun-79	30-Sep-79	92	765	5.202659	4343.54
30-Sep-79	31-Dec-79	92	857	6.666736	4956.88
31-Dec-79	31-Mar-80	91	948	5.068303	5418.10
31-Mar-80	30-Jun-80	91	1039	5.999137	5964.02
30-Jun-80	30-Sep-80	92	1131	6.289765	6542.68
30-Sep-80	31-Dec-80	92	1223	2.911629	6810.55
31-Dec-80	31-Mar-81	90	1313	6.354118	7382.42
31-Mar-81	30-Jun-81	91	1404	3.731881	7722.02
30-Jun-81	30-Sep-81	92	1496	6.338568	8305.17
30-Sep-81	31-Dec-81	92	1588	5.039344	8768.79
31-Dec-81	31-Mar-82	90	1678	5.325345	9248.07
31-Mar-82	30-Jun-82	91	1769	5.670859	9764.12
30-Jun-82	30-Sep-82	92	1861	5.467423	10267.12
30-Sep-82	2-Dec-82	63	1924	0.031986	10269.14
2-Dec-82	2-Dec-87	1826.25	3750.25	0	10269.14
2-Dec-87	2-Dec-92	1826.25	5576.50	0	10269.14
2-Dec-92	2-Dec-97	1826.25	7402.75	0	10269.14
2-Dec-97	2-Dec-02	1826.25	9229.00	0	10269.14
2-Dec-02	2-Dec-07	1826.25	11055.25	0	10269.14
2-Dec-07	2-Dec-17	3652.50	14707.75	0	10269.14
2-Dec-17	2-Dec-32	5478.75	20186.50	0	10269.14
2-Dec-32	2-Dec-47	5478.75	25665.25	0	10269.14
2-Dec-47	2-Dec-62	5478.75	31144.00	0	10269.14
2-Dec-62	2-Dec-82	7305.00	38449.00	0	10269.14

The ten dates with zero associated power represent the ten different cooling or decay dates after exposure or post-December 1982. 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.

LWBR Seed Module

Zircaloy Cladding, 60 to 100% Enriched U-233 Fuel

Reactor Moderator/Coolant:	Light Water
Fuel Meat:	UO2-ThO2 (Urania-Thoria)
Clad:	Zircaloy-4
Burnup:	10,269.14 MWD
Burnup:	27, 749.70 MWD/MTHM
Basis of Calculation:	Single seed module
BOL U-233	16,877.36 g U-233 per seed module
BOL U-234	221.64 g U-234 per seed module
BOL U-235:	15.46 g U-235 per seed module
BOL U-236	3.44 g U-236 per seed module
BOL U-238:	63.57 g U-238 per seed module
BOL Th-232:	442,731.04 g Th-232 per seed module
BOL Fuel Enrichment:	98.23 wt% U-233

DECAY TIMES (years out of core)

(Activities* in Ci/seed module)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H 3	1.177E+02	8.889E+01	6.714E+01	5.071E+01	3.830E+01	2.185E+01	9.414E+00	4.056E+00	1.748E+00	5.687E-01
BE 10	2.437E-06	2.437E-06	2.437E-06	2.437E-06	2.437E-06	2.437E-06	2.437E-06	2.437E-06	2.437E-06	2.437E-06
C 14	9.526E-01	9.520E-01	9.515E-01	9.509E-01	9.503E-01	9.492E-01	9.475E-01	9.457E-01	9.440E-01	9.417E-01
CL 36	1.859E-02	1.859E-02	1.859E-02	1.859E-02	1.859E-02	1.859E-02	1.859E-02	1.859E-02	1.859E-02	1.859E-02
CR 51	3.260E-18	4.703E-38	6.786E-58	9.790E-78	1.412E-97	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54	5.381E-01	9.368E-03	1.631E-04	2.839E-06	4.943E-08	1.498E-11	7.905E-17	4.171E-22	2.201E-27	2.022E-34
FE 55	7.030E+01	1.854E+01	4.888E+00	1.289E+00	3.399E-01	2.363E-02	4.333E-04	7.945E-06	1.457E-07	7.043E-10
FE 59	2.760E-12	1.675E-24	1.017E-36	6.173E-49	3.747E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60	4.641E+02	2.404E+02	1.246E+02	6.453E+01	3.343E+01	8.972E+00	1.247E+00	1.734E-01	2.411E-02	1.737E-03
NI 59	6.503E-02	6.503E-02	6.503E-02	6.502E-02	6.502E-02	6.502E-02	6.501E-02	6.500E-02	6.499E-02	6.498E-02
NI 63	8.701E+00	8.379E+00	8.069E+00	7.771E+00	7.484E+00	6.941E+00	6.199E+00	5.537E+00	4.945E+00	4.253E+00
ZN 65	1.948E-01	1.085E-03	6.039E-06	3.362E-08	1.872E-10	5.802E-15	1.001E-21	1.728E-28	2.982E-35	2.865E-44
SE 79	3.638E-01	3.638E-01	3.638E-01	3.637E-01	3.637E-01	3.637E-01	3.636E-01	3.636E-01	3.635E-01	3.634E-01
KR 85	4.474E+03	3.238E+03	2.344E+03	1.696E+03	1.228E+03	6.431E+02	2.438E+02	9.244E+01	3.505E+01	9.617E+00
RB 87	1.429E-05	1.429E-05	1.429E-05	1.429E-05	1.429E-05	1.429E-05	1.429E-05	1.429E-05	1.429E-05	1.429E-05

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

Radionuclide	5	10	15	20	25	35	50	65	80	100
SR 89	1.585E-06	2.057E-17	2.670E-28	3.466E-39	4.498E-50	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	3.131E+04	2.779E+04	2.468E+04	2.191E+04	1.945E+04	1.533E+04	1.073E+04	7.506E+03	5.252E+03	3.263E+03
Y 90	3.131E+04	2.780E+04	2.468E+04	2.191E+04	1.945E+04	1.533E+04	1.073E+04	7.508E+03	5.254E+03	3.264E+03
Y 91	5.778E-05	2.319E-14	9.310E-24	3.737E-33	1.500E-42	2.417E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93	8.719E-01	8.719E-01	8.719E-01	8.719E-01	8.719E-01	8.719E-01	8.718E-01	8.718E-01	8.718E-01	8.718E-01
ZR 95	4.013E-04	1.026E-12	2.621E-21	6.701E-30	1.713E-38	1.119E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M	2.721E-01	3.972E-01	4.942E-01	5.693E-01	6.275E-01	7.077E-01	7.721E-01	8.021E-01	8.161E-01	8.239E-01
NB 94	2.796E-02	2.795E-02	2.795E-02	2.795E-02	2.794E-02	2.793E-02	2.792E-02	2.790E-02	2.789E-02	2.787E-02
NB 95	8.908E-04	2.277E-12	5.821E-21	1.487E-29	3.803E-38	2.485E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	2.976E-06	7.608E-15	1.945E-23	4.970E-32	1.271E-40	8.302E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93	4.334E-03	4.330E-03	4.325E-03	4.321E-03	4.317E-03	4.308E-03	4.295E-03	4.283E-03	4.270E-03	4.253E-03
TC 99	3.341E+00	3.341E+00	3.341E+00	3.341E+00	3.341E+00	3.340E+00	3.340E+00	3.340E+00	3.340E+00	3.340E+00
RU103	2.716E-10	2.748E-24	2.780E-38	2.812E-52	2.845E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	3.759E+02	1.207E+01	3.879E-01	1.246E-02	4.002E-04	4.129E-07	1.368E-11	4.535E-16	1.503E-20	1.600E-26
RH103M	2.449E-10	2.477E-24	2.506E-38	2.535E-52	2.565E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	3.759E+02	1.207E+01	3.879E-01	1.246E-02	4.002E-04	4.129E-07	1.368E-11	4.535E-16	1.503E-20	1.600E-26
PD107	3.543E-03	3.543E-03	3.543E-03	3.543E-03	3.543E-03	3.543E-03	3.543E-03	3.543E-03	3.543E-03	3.543E-03
AG110	7.814E-03	4.930E-05	3.110E-07	1.962E-09	1.238E-11	4.927E-16	1.237E-22	3.107E-29	7.802E-36	1.236E-44
AG110M	5.875E-01	3.707E-03	2.338E-05	1.475E-07	9.308E-10	3.705E-14	9.303E-21	2.336E-27	5.866E-34	9.294E-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	3.574E+00	2.818E+00	2.222E+00	1.752E+00	1.382E+00	8.592E-01	4.213E-01	2.066E-01	1.013E-01	3.917E-02
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	1.621E-11	7.620E-24	3.582E-36	1.684E-48	7.918E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	7.035E-11	5.548E-22	4.377E-33	3.452E-44	2.723E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	7.351E-11	5.798E-22	4.573E-33	3.607E-44	2.845E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	1.138E-15	5.352E-28	2.516E-40	1.183E-52	5.562E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	1.551E+01	8.850E-02	5.048E-04	2.881E-06	1.643E-08	5.351E-13	9.939E-20	1.846E-26	3.429E-33	3.634E-42
SN121M	6.553E-01	6.114E-01	5.704E-01	5.322E-01	4.966E-01	4.322E-01	3.511E-01	2.851E-01	2.316E-01	1.755E-01
SN123	2.416E-02	1.339E-06	7.425E-11	4.117E-15	2.282E-19	7.015E-28	1.195E-40	2.036E-53	3.471E-66	3.279E-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	4.093E-01	4.092E-01	4.092E-01	4.092E-01	4.092E-01	4.092E-01	4.091E-01	4.091E-01	4.090E-01	4.090E-01
SB124	7.506E-08	5.531E-17	4.075E-26	3.002E-35	2.212E-44	1.201E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	1.521E+03	4.351E+02	1.245E+02	3.562E+01	1.019E+01	8.347E-01	1.956E-02	4.583E-04	1.074E-05	7.200E-08

DECAY TIMES (years out of core)

(Activities* in Ci/seed module)

Radionuclide	5	10	15	20	25	35	50	65	80	100
SB126	5.730E-02	5.729E-02	5.729E-02	5.729E-02	5.729E-02	5.728E-02	5.728E-02	5.727E-02	5.727E-02	5.726E-02
SB126M	4.093E-01	4.092E-01	4.092E-01	4.092E-01	4.092E-01	4.092E-01	4.091E-01	4.091E-01	4.090E-01	4.090E-01
TE123M	7.010E-05	1.786E-09	4.553E-14	1.160E-18	2.958E-23	1.921E-32	3.180E-46	5.263E-60	8.712E-74	3.676E-92
TE125M	3.710E+02	1.062E+02	3.038E+01	8.692E+00	2.487E+00	2.036E-01	4.772E-03	1.118E-04	2.619E-06	1.756E-08
TE127	2.778E-02	2.514E-07	2.275E-12	2.058E-17	1.862E-22	1.525E-32	1.130E-47	8.370E-63	6.201E-78	4.157E-98
TE127M	2.836E-02	2.566E-07	2.322E-12	2.101E-17	1.901E-22	1.557E-32	1.153E-47	8.545E-63	6.331E-78	4.244E-98
TE129	1.088E-13	4.728E-30	2.054E-46	8.927E-63	3.879E-79	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	1.672E-13	7.264E-30	3.156E-46	1.371E-62	5.958E-79	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	1.628E-02	1.628E-02	1.628E-02	1.628E-02	1.628E-02	1.628E-02	1.628E-02	1.628E-02	1.628E-02	1.628E-02
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	6.303E+03	1.174E+03	2.186E+02	4.071E+01	7.581E+00	2.629E-01	1.698E-03	1.096E-05	7.081E-08	8.516E-11
CS135	2.941E-01	2.941E-01	2.941E-01	2.941E-01	2.941E-01	2.941E-01	2.941E-01	2.941E-01	2.941E-01	2.941E-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	3.034E+04	2.703E+04	2.408E+04	2.146E+04	1.911E+04	1.517E+04	1.073E+04	7.586E+03	5.364E+03	3.379E+03
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	2.870E+04	2.557E+04	2.278E+04	2.030E+04	1.808E+04	1.435E+04	1.015E+04	7.176E+03	5.074E+03	3.196E+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
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	1.005E-12	1.238E-29	1.525E-46	1.878E-63	2.313E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	1.019E-05	1.019E-05	1.019E-05	1.019E-05	1.019E-05	1.019E-05	1.019E-05	1.019E-05	1.019E-05	1.019E-05
CE144	2.107E+03	2.453E+01	2.855E-01	3.324E-03	3.870E-05	5.244E-09	8.274E-15	1.305E-20	2.059E-26	3.782E-34
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	2.107E+03	2.453E+01	2.855E-01	3.324E-03	3.870E-05	5.244E-09	8.274E-15	1.305E-20	2.059E-26	3.782E-34
PR144M	2.528E+01	2.943E-01	3.427E-03	3.989E-05	4.644E-07	6.293E-11	9.929E-17	1.566E-22	2.471E-28	4.539E-36
ND144	4.212E-10	4.220E-10	4.220E-10	4.220E-10	4.220E-10	4.220E-10	4.220E-10	4.220E-10	4.220E-10	4.220E-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
PM145	1.075E-02	8.903E-03	7.322E-03	6.020E-03	4.949E-03	3.346E-03	1.859E-03	1.033E-03	5.744E-04	2.625E-04
PM147	7.457E+03	1.990E+03	5.310E+02	1.417E+02	3.781E+01	2.693E+00	5.117E-02	9.724E-04	1.848E-05	9.370E-08
PM148M	2.200E-10	1.071E-23	5.217E-37	2.541E-50	1.237E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	1.239E-11	6.034E-25	2.939E-38	1.431E-51	6.969E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	1.466E-03	3.543E-05	8.563E-07	2.070E-08	5.003E-10	2.923E-13	4.127E-18	5.828E-23	8.230E-28	2.809E-34

DECAY TIMES (years out of core)

(Activities* in Ci/seed module)

Radionuclide	5	10	15	20	25	35	50	65	80	100
SM147	1.120E-06	1.254E-06	1.290E-06	1.300E-06	1.302E-06	1.303E-06	1.303E-06	1.303E-06	1.303E-06	1.303E-06
SM151	1.402E+02	1.349E+02	1.298E+02	1.249E+02	1.202E+02	1.113E+02	9.918E+01	8.836E+01	7.872E+01	6.748E+01
EU152	2.065E+00	1.600E+00	1.240E+00	9.612E-01	7.450E-01	4.475E-01	2.084E-01	9.700E-02	4.516E-02	1.630E-02
EU154	9.913E+02	6.625E+02	4.428E+02	2.960E+02	1.978E+02	8.834E+01	2.637E+01	7.872E+00	2.350E+00	4.688E-01
EU155	4.493E+02	2.234E+02	1.111E+02	5.520E+01	2.744E+01	6.784E+00	8.336E-01	1.024E-01	1.258E-02	7.689E-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	2.054E-01	1.099E-03	5.879E-06	3.146E-08	1.683E-10	4.818E-15	7.382E-22	1.131E-28	1.732E-35	1.419E-44
TB160	3.487E-07	8.690E-15	2.166E-22	5.396E-30	1.345E-37	8.349E-53	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	1.518E-12	1.518E-12	1.518E-12	1.518E-12	1.518E-12	1.518E-12	1.518E-12	1.518E-12	1.518E-12	1.518E-12
TL207	2.525E-01	3.969E-01	5.200E-01	6.249E-01	7.144E-01	8.555E-01	9.982E-01	1.087E+00	1.142E+00	1.183E+00
TL208	1.723E+02	1.794E+02	1.734E+02	1.657E+02	1.580E+02	1.434E+02	1.241E+02	1.074E+02	9.297E+01	7.675E+01
PB210	1.657E-04	1.471E-04	1.338E-04	1.252E-04	1.208E-04	1.226E-04	1.463E-04	1.892E-04	2.466E-04	3.410E-04
PB211	2.532E-01	3.980E-01	5.214E-01	6.267E-01	7.164E-01	8.579E-01	1.001E+00	1.090E+00	1.145E+00	1.187E+00
PB212	4.795E+02	4.993E+02	4.827E+02	4.612E+02	4.397E+02	3.991E+02	3.454E+02	2.989E+02	2.587E+02	2.136E+02
BI211	2.532E-01	3.980E-01	5.214E-01	6.267E-01	7.164E-01	8.579E-01	1.001E+00	1.090E+00	1.145E+00	1.187E+00
BI212	4.795E+02	4.993E+02	4.827E+02	4.612E+02	4.397E+02	3.991E+02	3.454E+02	2.989E+02	2.587E+02	2.136E+02
PO212	3.072E+02	3.199E+02	3.093E+02	2.955E+02	2.817E+02	2.557E+02	2.213E+02	1.915E+02	1.658E+02	1.369E+02
PO215	2.532E-01	3.980E-01	5.214E-01	6.267E-01	7.164E-01	8.579E-01	1.001E+00	1.090E+00	1.145E+00	1.187E+00
PO216	4.795E+02	4.993E+02	4.827E+02	4.612E+02	4.397E+02	3.991E+02	3.454E+02	2.989E+02	2.587E+02	2.136E+02
RN219	2.532E-01	3.980E-01	5.214E-01	6.267E-01	7.164E-01	8.579E-01	1.001E+00	1.090E+00	1.145E+00	1.187E+00
RN220	4.795E+02	4.993E+02	4.827E+02	4.612E+02	4.397E+02	3.991E+02	3.454E+02	2.989E+02	2.587E+02	2.136E+02
FR223	3.492E-03	5.485E-03	7.185E-03	8.634E-03	9.870E-03	1.182E-02	1.380E-02	1.502E-02	1.578E-02	1.636E-02
RA223	2.532E-01	3.980E-01	5.214E-01	6.267E-01	7.164E-01	8.579E-01	1.001E+00	1.090E+00	1.145E+00	1.187E+00
RA224	4.795E+02	4.993E+02	4.827E+02	4.612E+02	4.397E+02	3.991E+02	3.454E+02	2.989E+02	2.587E+02	2.136E+02
RA226	2.659E-05	4.499E-05	6.416E-05	8.411E-05	1.048E-04	1.486E-04	2.200E-04	2.982E-04	3.833E-04	5.073E-04
RA228	3.075E-02	3.757E-02	4.163E-02	4.406E-02	4.550E-02	4.688E-02	4.748E-02	4.760E-02	4.763E-02	4.764E-02
AC227	2.530E-01	3.975E-01	5.206E-01	6.257E-01	7.152E-01	8.567E-01	9.998E-01	1.088E+00	1.143E+00	1.185E+00
TH227	2.497E-01	3.925E-01	5.142E-01	6.181E-01	7.066E-01	8.460E-01	9.872E-01	1.075E+00	1.129E+00	1.170E+00
TH228	4.793E+02	4.989E+02	4.823E+02	4.608E+02	4.393E+02	3.990E+02	3.454E+02	2.989E+02	2.587E+02	2.136E+02
TH229	1.158E-01	1.768E-01	2.377E-01	2.986E-01	3.594E-01	4.810E-01	6.632E-01	8.452E-01	1.027E+00	1.269E+00
TH230	8.339E-03	8.716E-03	9.094E-03	9.471E-03	9.849E-03	1.060E-02	1.174E-02	1.287E-02	1.400E-02	1.551E-02
TH231	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04
TH232	4.764E-02	4.764E-02	4.764E-02	4.764E-02	4.764E-02	4.764E-02	4.764E-02	4.764E-02	4.764E-02	4.764E-02

DECAY TIMES (years out of core)

(Activities* in Ci/seed module)

Radionuclide	5	10	15	20	25	35	50	65	80	100
TH234	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05
PA231	1.235E+00	1.235E+00	1.234E+00	1.234E+00	1.234E+00	1.234E+00	1.233E+00	1.233E+00	1.233E+00	1.232E+00
PA233	1.268E-03	1.269E-03	1.272E-03	1.274E-03	1.278E-03	1.285E-03	1.296E-03	1.309E-03	1.321E-03	1.338E-03
PA234M	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05
PA234	2.362E-08	2.362E-08	2.362E-08	2.362E-08	2.362E-08	2.362E-08	2.362E-08	2.362E-08	2.362E-08	2.362E-08
U232	5.184E+02	4.941E+02	4.708E+02	4.487E+02	4.276E+02	3.884E+02	3.361E+02	2.909E+02	2.518E+02	2.077E+02
U233	1.292E+02	1.292E+02	1.292E+02	1.292E+02	1.292E+02	1.292E+02	1.292E+02	1.292E+02	1.292E+02	1.292E+02
U234	8.398E+00	8.397E+00	8.397E+00	8.397E+00	8.397E+00	8.397E+00	8.397E+00	8.397E+00	8.397E+00	8.396E+00
U235	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04	6.271E-04
U236	1.363E-03	1.363E-03	1.363E-03	1.363E-03	1.363E-03	1.363E-03	1.363E-03	1.363E-03	1.363E-03	1.363E-03
U237	1.569E-05	1.233E-05	9.694E-06	7.621E-06	5.990E-06	3.702E-06	1.798E-06	8.734E-07	4.243E-07	1.620E-07
U238	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05	1.817E-05
NP237	1.268E-03	1.269E-03	1.272E-03	1.274E-03	1.278E-03	1.285E-03	1.296E-03	1.309E-03	1.321E-03	1.338E-03
PU236	1.673E-04	4.960E-05	1.471E-05	4.364E-06	1.296E-06	1.164E-07	5.678E-09	2.791E-09	2.715E-09	2.713E-09
PU237	1.720E-15	1.512E-27	1.329E-39	1.168E-51	1.026E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	5.525E+00	5.311E+00	5.106E+00	4.909E+00	4.719E+00	4.362E+00	3.876E+00	3.444E+00	3.060E+00	2.615E+00
PU239	2.829E-01	2.828E-01	2.828E-01	2.828E-01	2.827E-01	2.826E-01	2.825E-01	2.824E-01	2.823E-01	2.821E-01
PU240	1.658E-01	1.660E-01	1.661E-01	1.661E-01	1.662E-01	1.662E-01	1.661E-01	1.659E-01	1.657E-01	1.654E-01
PU241	6.395E+01	5.027E+01	3.952E+01	3.106E+01	2.442E+01	1.509E+01	7.330E+00	3.560E+00	1.729E+00	6.604E-01
PU242	4.191E-04	4.191E-04	4.191E-04	4.191E-04	4.192E-04	4.192E-04	4.193E-04	4.193E-04	4.194E-04	4.194E-04
PU244	5.399E-11	5.399E-11	5.399E-11	5.399E-11	5.399E-11	5.399E-11	5.399E-11	5.399E-11	5.399E-11	5.399E-11
AM241	7.541E-01	1.202E+00	1.549E+00	1.817E+00	2.023E+00	2.299E+00	2.500E+00	2.564E+00	2.563E+00	2.517E+00
AM242M	1.745E-02	1.706E-02	1.668E-02	1.630E-02	1.593E-02	1.522E-02	1.422E-02	1.328E-02	1.240E-02	1.132E-02
AM242	1.737E-02	1.698E-02	1.659E-02	1.622E-02	1.585E-02	1.515E-02	1.415E-02	1.321E-02	1.234E-02	1.126E-02
AM243	3.208E-03	3.206E-03	3.205E-03	3.203E-03	3.202E-03	3.199E-03	3.194E-03	3.190E-03	3.185E-03	3.179E-03
CM242	2.021E-02	1.405E-02	1.373E-02	1.342E-02	1.312E-02	1.253E-02	1.170E-02	1.092E-02	1.020E-02	9.313E-03
CM243	6.518E-03	5.772E-03	5.111E-03	4.525E-03	4.007E-03	3.142E-03	2.182E-03	1.515E-03	1.052E-03	6.467E-04
CM244	4.580E-01	3.782E-01	3.123E-01	2.579E-01	2.130E-01	1.453E-01	8.181E-02	4.608E-02	2.595E-02	1.207E-02

Radionuclide	DECAY TIMES (years out of core) (Activities* in Ci/seed module) 5	10	15	20	25	35	50	65	80	100
CM245	7.162E-05	7.159E-05	7.157E-05	7.154E-05	7.151E-05	7.145E-05	7.136E-05	7.127E-05	7.119E-05	7.107E-05
CM246	4.751E-06	4.747E-06	4.744E-06	4.741E-06	4.737E-06	4.730E-06	4.720E-06	4.709E-06	4.699E-06	4.685E-06
CM247	1.677E-11	1.677E-11	1.677E-11	1.677E-11	1.677E-11	1.677E-11	1.677E-11	1.677E-11	1.677E-11	1.677E-11
Subtotal**	1.531E+05	1.208E+05	1.044E+05	9.197E+04	8.149E+04	6.443E+04	4.565E+04	3.252E+04	2.329E+04	1.505E+04
Total***	1.531E+05	1.208E+05	1.044E+05	9.197E+04	8.151E+04	6.444E+04	4.565E+04	3.253E+04	2.330E+04	1.506E+04

* 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 24

Fuel-Specific Source Term Calculations Pressurized Water 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 generic source term for pressurized water reactor (PWR) or low-enriched uranium oxide spent nuclear fuel. The data sources for the analysis are documented in References 1 and 2, and the INEEL calculational methodology is described in detail in Reference 3.

Pressurized Water Reactor Data

PWR characteristics are based on the Westinghouse PWR 17 × 17 fuel assembly and corresponding fuel rod dimensions and materials (Reference 1).

The uranium enrichment is chosen to be 3.2 wt% U-235. In addition, small amounts of U-234 and U-236 impurities are also included in the beginning-of-life (BOL) uranium composition in order to maximize and account for production of other actinides.

The cladding material is assumed to be Zircaloy-4. Assembly structural hardware, including the spacers and top and bottom tie grids, is also included in the activation analysis. The spacers are composed of Zircaloy-4 and Inconel-718 pieces. The top and bottom tie grids are stainless steel-304 (SS304). The total assembly spacer and grid masses are based on estimates found in Reference 2 data for a 15 × 15 PWR assembly. The assumption here is that the 15 × 15 hardware or structural materials are the same and of similar mass to those expected for a 17 × 17 assembly. The 15 × 15 assembly hardware total masses are then divided by 289 (17 × 17) in order to get the associated structural masses per rod for the 17 × 17 assembly. The mass basis for the PWR template here is a single fuel rod from a 17 × 17 assembly. Table 1 lists the impurities and their concentrations for the materials considered in the depletion/activation analysis, namely, UO_2, Zircaloy-4, Inconel-718, and SS-304.

The data below give specific dimensions, materials, loadings, densities, enrichment, etc., for the PWR fuel rod used in the burnup calculation for the source term generation.

Fuel Element:	Single rod in a 17 × 17 assembly
Pitch:	1.25 cm
Fuel Pellet Diameter:	0.819 cm
Gas Gap Thickness:	0.0082 cm
Clad Thickness:	0.0572 cm
Fuel Meat:	UO_2 3.2 wt% U-235 enrichment (BOL) Average Density = 10.412 g/cc
Clad:	Zircaloy-4 Density = 6.44 g/cc

Loading:	56.61	g/rod U-235 (BOL)
	1,711.63	g/rod U-238 (BOL)
	0.70	g/rod U-234 (BOL)
	0.17	g/rod U-236 (BOL)
	1,769.11	g/rod TOTAL U
	2,006.26	g/rod UO2 (fuel meat)
	1.7691E-3	MTU/rod
Active Fuel Length:	144.0 in.	
Fuel Rode Length:	168.0 in.	

Assembly Structural Material/Masses:

456.40	g/rod Zircaloy-4 (clad + spacer)
2.20	g/rod Inconel-718 (spacer material)
62.78	g/rod SS-304 (top/bottom tie grid plates)

Coolant/Moderator:	Light Water
Coolant Temperature:	600°F
Coolant Pressure:	2250 psi
Coolant Density:	0.6965 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 PWR fuel rod. Impurities in these materials were also included in the ORIGEN2 (Reference 4) depletion calculation in order to maximize the induced activation (see Table 1).

Burnup

The burnup chosen for this template is 35,000 MWd/MTU or 61.92 MWd for the single rod containing 1.7691E-3 MTU. The burnup period is assumed to be continuous over a 3-year period or 1,096 days. The fuel rod operates at a constant power over the burnup period with no refueling shutdowns. The relatively high burnup (35,000 MWd/MTU) 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 isotope concentrations, in particular U-235).

Cross-Section Development

The neutron cross sections used in the burnup or depletion calculation for the source term generation of a single PWR fuel rod are based on the methodology described in Reference 3. Cross sections from a standard ORIGEN2 PWR library were updated six times over the 3-year burnup period to ensure accurate production and activity levels for actinides, fission products, and activation products. The first update developed cross sections for BOL conditions followed by five subsequent updates every 180-days of fuel rod exposure. These cross-section updates take into account changes in the neutron flux spectrum and spatial profiles as a function of burnup. A simple PWR unit cell lattice was used to determine volume-averaged flux and reaction rate profiles for the cross-section development. The basic library updated was a standard ORIGEN2 PWR cross-section library.

Pressurized Water Reactor Single Rod Exposure History

Table 2 summarizes the hypothetical power or exposure history used in the burnup calculations for a single PWR fuel rod in a 17 × 17 PWR assembly. No refueling shutdowns are assumed in the calculation. Following the 3-year exposure, the radionuclide activities are decayed for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100 years. The relatively high burnup is conservative for the buildup of fission products, activation products, and minor actinides in the source term and nonconservative with regard to criticality safety. The 35,000 MWd/MTU exposure or burnup for the fuel is again relatively high and represents an upper burnup limit for most commercial spent nuclear fuels stored at the INEEL.

Burnup Calculation

The ORIGEN2 computer code was used to perform the depletion or burnup calculation for the PWR fuel rod. The radionuclide inventory or source term template that follows is for a single PWR 17 × 17 fuel rod. The fuel rod component masses and impurities (fuel meat, uranium, clad, spacers, and end fixtures), neutron cross sections, burnup, and hypothetical power history and power level as presented above are used as input data for the ORIGEN2 calculation. The radionuclide concentrations are given as a function of decay time in the template table.

The 145 radionuclides listed in the template here represent greater than 99.9% of the total curie inventory relative to the 684 activation products, 880 fission products, and 127 actinide/daughter isotopes in the complete ORIGEN2 output.

References

1. J. J. Duderstadt and L. J. Hamilton, *Nuclear Reactor Analysis*, Appendix H, John Wiley & Sons, Inc., New York, 1976.

2. R. K. McCardell, *Characteristics of Commercial Nuclear Materials Stored in the TAN Pool*, INEL/INT-98-00767, September 1998.

3. 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.

4. 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. Material constituent and impurity concentrations for the various materials in a pressurized water reactor fuel rod/assembly.

Constituent or Impurity	UO2 Concentration (ppm)	Zircaloy-4 Concentration (ppm)	Inconel-718 Concentration (wt%)	Stainless Steel-304 Concentration (ppm)
H		25		
Li	1			0.13
Be				
B	1	0.5		
C	89.4	270	0.08	0.08 wt%
N	25	80		525
O	134454	950		
F	10.7			
Na	15			37
Mg	2			
Al	16.7	75	0.70	200
Si	12.1	120	0.70	1.00 wt%
P	35	100		
S		35	0.01	
Cl	5.3			130
K				3
Ca	2			19
Sc				0.03
Ti	1	50	2.30	600
V	3	50		690
Cr	4	1250	15.50	18.40 wt%
Mn	1.7	50	1.00	1.53 wt%
Fe	18	2250	5.90	68.99 wt%
Co	1	20	0.006488	2570
Ni	24	70	73.304	10.00 wt%
Cu	1	50	0.50	8150
Zn	40.3	100		2230
Ga				450
As				1010
Se				70
Br				8
Rb				10
Sr				0.2
Y				5
Zr		979069		20
Nb		70		300

Constituent or Impurity	UO2 Concentration (ppm)	Zircaloy-4 Concentration (ppm)	Inconel-718 Concentration (wt%)	Stainless Steel-304 Concentration (ppm)
Mo	10	50		5500
Ag	0.1			2
Cd	25	0.5		
In	2			
Sn	4	16000		
Sb				17
Cs				0.3
Ba				500
La				2.1
Ce				550
Pr				
Nd				
Sm		10		0.15
Eu				0.02
Gd		5		
Tb				0.71
Dy				1
Ho				1
Er				
Tm				
Yb				2
Lu				0.8
Hf		35		2
Ta		200		
W	2	100		520
Tl				
Pb	1	100		139
Bi	0.4			
Th		7		1
U	1000000	3.5		2

Table 2. Hypothetical power history for a 35,000 MWd/MTU burnup PWR fuel rod.

Duration (days)	Cumulative Duration (days)	Time-Averaged Power (MWth)
180	180	0.0565
180	360	0.0565
180	540	0.0565
180	720	0.0565
180	900	0.0565
196	1096	0.0565
1825	2921	0.0
1825	4746	0.0
1825	6571	0.0
1825	8396	0.0
1825	10221	0.0
3650	13871	0.0
5475	19346	0.0
5475	24821	0.0
5475	30296	0.0
7300	37596	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 or decay times designated for the template methodology.

Pressurized Water Reactor Element

Zirconium Cladding, 0 to 5% Enriched U-235 Fuel

Reactor Moderator/Coolant:	Light Water
Fuel Meat:	UO_2
Clad:	Zircaloy-4
Burnup:	35000 MWd/MTU
Burnup:	61.92 MWd/single rod (high burnup)
Burnup:	73.4 % U-235 depletion (fissioned and transmuted)
Basis of Calculation:	Single rod in a 17x17 assembly
BOL U-235:	56.61 g U-235 per rod
BOL U-238:	1711.63 g U-238 per rod
BOL U-234:	0.70 g U-234 per rod
BOL U-236:	0.17 g U-236 per rod
BOL Total U per element:	1769.11 g U per rod
BOL Fuel Enrichment:	3.2 wt% U-235

DECAY TIMES (years out of core)

(Activities* in Ci/rod)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H-3	6.915E+00	5.224E+00	3.946E+00	2.981E+00	2.253E+00	1.286E+00	5.541E-01	2.388E-01	1.030E-01	3.354E-02
BE-10	1.058E-08	1.058E-08	1.058E-08	1.058E-08	1.058E-08	1.058E-08	1.058E-08	1.058E-08	1.058E-08	1.058E-08
C-14	2.977E-03	2.975E-03	2.974E-03	2.972E-03	2.970E-03	2.966E-03	2.961E-03	2.956E-03	2.950E-03	2.943E-03
CL-36	4.973E-05	4.973E-05	4.973E-05	4.973E-05	4.972E-05	4.972E-05	4.972E-05	4.972E-05	4.972E-05	4.972E-05
CR-51	1.372E-18	2.042E-38	3.040E-58	4.526E-78	6.737E-98	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN-54	8.462E-02	1.477E-03	2.579E-05	4.502E-07	7.860E-09	2.396E-12	1.275E-17	6.781E-23	3.608E-28	3.351E-35
FE-55	7.901E+00	2.085E+00	5.504E-01	1.453E-01	3.834E-02	2.671E-03	4.910E-05	9.028E-07	1.660E-08	8.055E-11
FE-59	1.098E-12	6.797E-25	4.206E-37	2.603E-49	1.611E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO-60	8.171E+00	4.235E+00	2.195E+00	1.138E+00	5.896E-01	1.584E-01	2.205E-02	3.070E-03	4.275E-04	3.085E-05
NI-59	4.844E-03	4.844E-03	4.844E-03	4.843E-03	4.843E-03	4.843E-03	4.842E-03	4.841E-03	4.841E-03	4.840E-03
NI-63	6.543E-01	6.302E-01	6.069E-01	5.845E-01	5.629E-01	5.220E-01	4.663E-01	4.165E-01	3.720E-01	3.200E-01
ZN-65	2.761E-03	1.543E-05	8.619E-08	4.815E-10	2.690E-12	8.399E-17	1.465E-23	2.555E-30	4.457E-37	4.344E-46
SE-79	7.666E-04	7.666E-04	7.666E-04	7.665E-04	7.665E-04	7.664E-04	7.663E-04	7.661E-04	7.660E-04	7.659E-04
KR-85	1.214E+01	8.789E+00	6.363E+00	4.606E+00	3.334E+00	1.747E+00	6.630E-01	2.515E-01	9.542E-02	2.621E-02
RB-87	3.849E-08	3.849E-08	3.849E-08	3.849E-08	3.849E-08	3.849E-08	3.849E-08	3.849E-08	3.849E-08	3.849E-08
SR-89	1.631E-08	2.154E-19	2.843E-30	3.754E-41	4.957E-52	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR-90	1.159E+02	1.029E+02	9.134E+01	8.110E+01	7.201E+01	5.676E+01	3.973E+01	2.781E+01	1.946E+01	1.210E+01
Y-90	1.159E+02	1.029E+02	9.136E+01	8.112E+01	7.202E+01	5.678E+01	3.974E+01	2.781E+01	1.947E+01	1.210E+01
Y-91	6.717E-07	2.736E-16	1.115E-25	4.542E-35	1.850E-44	3.071E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR-93	3.620E-03	3.620E-03	3.620E-03	3.620E-03	3.620E-03	3.620E-03	3.620E-03	3.620E-03	3.620E-03	3.620E-03

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

Radionuclide	5	10	15	20	25	35	50	65	80	100
ZR-95	6.572E-06	1.702E-14	4.412E-23	1.143E-31	2.961E-40	1.988E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-93M	9.813E-04	1.534E-03	1.962E-03	2.294E-03	2.551E-03	2.905E-03	3.190E-03	3.323E-03	3.384E-03	3.419E-03
NB-94	8.828E-05	8.826E-05	8.825E-05	8.823E-05	8.822E-05	8.819E-05	8.814E-05	8.810E-05	8.805E-05	8.799E-05
NB-95	1.459E-05	3.781E-14	9.795E-23	2.537E-31	6.575E-40	4.414E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-95M	4.875E-08	1.263E-16	3.273E-25	8.480E-34	2.197E-42	1.474E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO-93	2.257E-05	2.255E-05	2.253E-05	2.250E-05	2.248E-05	2.244E-05	2.237E-05	2.230E-05	2.224E-05	2.215E-05
TC-99	2.437E-02	2.437E-02	2.437E-02	2.437E-02	2.437E-02	2.437E-02	2.437E-02	2.437E-02	2.437E-02	2.436E-02
RU-103	2.668E-11	2.759E-25	2.853E-39	2.951E-53	3.052E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU-106	3.400E+01	1.095E+00	3.525E-02	1.135E-03	3.654E-05	3.787E-08	1.264E-12	4.219E-17	1.408E-21	1.513E-27
RH-103M	2.405E-11	2.487E-25	2.572E-39	2.660E-53	2.751E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH-106	3.400E+01	1.095E+00	3.525E-02	1.135E-03	3.654E-05	3.788E-08	1.264E-12	4.219E-17	1.408E-21	1.513E-27
PD-107	2.296E-04	2.296E-04	2.296E-04	2.296E-04	2.296E-04	2.296E-04	2.296E-04	2.296E-04	2.296E-04	2.296E-04
AG-110	8.434E-04	5.339E-06	3.380E-08	2.140E-10	1.355E-12	5.430E-17	1.378E-23	3.496E-30	8.870E-37	1.425E-45
AG-110M	6.342E-02	4.014E-04	2.541E-06	1.609E-08	1.019E-10	4.082E-15	1.036E-21	2.628E-28	6.670E-35	1.072E-43
AG-111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-113M	9.764E-02	7.701E-02	6.073E-02	4.790E-02	3.778E-02	2.350E-02	1.153E-02	5.655E-03	2.774E-03	1.073E-03
CD-113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-115M	1.632E-12	7.823E-25	3.750E-37	1.797E-49	8.618E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114	1.075E-12	8.627E-24	6.924E-35	5.558E-46	4.461E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114M	1.123E-12	9.014E-24	7.236E-35	5.808E-46	4.661E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-115M	1.129E-16	5.412E-29	2.594E-41	1.244E-53	5.961E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-119M	8.362E-02	4.788E-04	2.741E-06	1.570E-08	8.987E-11	2.947E-15	5.532E-22	1.039E-28	1.950E-35	2.096E-44
SN-121M	1.768E-03	1.649E-03	1.539E-03	1.436E-03	1.340E-03	1.167E-03	9.475E-04	7.696E-04	6.252E-04	4.738E-04
SN-123	4.105E-04	2.291E-08	1.279E-12	7.136E-17	3.983E-21	1.241E-29	2.158E-42	3.751E-55	6.522E-68	6.328E-85
SN-125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-126	1.561E-03	1.561E-03	1.561E-03	1.561E-03	1.561E-03	1.561E-03	1.561E-03	1.560E-03	1.560E-03	1.560E-03
SB-124	2.599E-09	1.943E-18	1.452E-27	1.085E-36	8.113E-46	4.533E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB-125	9.405E+00	2.694E+00	7.715E-01	2.210E-01	6.328E-02	5.190E-03	1.220E-04	2.864E-06	6.729E-08	4.526E-10
SB-126	2.185E-04	2.185E-04	2.185E-04	2.185E-04	2.185E-04	2.185E-04	2.185E-04	2.185E-04	2.184E-04	2.184E-04
SB-126M	1.561E-03	1.561E-03	1.561E-03	1.561E-03	1.561E-03	1.561E-03	1.561E-03	1.560E-03	1.560E-03	1.560E-03
TE-123M	1.816E-06	4.663E-11	1.197E-15	3.072E-20	7.887E-25	5.197E-34	8.792E-48	1.487E-61	2.516E-75	1.093E-93
TE-125M	2.295E+00	6.572E-01	1.883E-01	5.390E-02	1.544E-02	1.266E-03	2.975E-05	6.988E-07	1.641E-08	1.104E-10
TE-127	2.330E-04	2.125E-09	1.938E-14	1.768E-19	1.613E-24	1.341E-34	1.018E-49	7.723E-65	5.860E-80	4.055-100
TE-127M	2.379E-04	2.170E-09	1.979E-14	1.805E-19	1.646E-24	1.370E-34	1.039E-49	7.884E-65	5.982E-80	4.140-100
TE-129	2.279E-15	1.016E-31	4.530E-48	2.020E-64	9.005E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Radionuclide	5	10	15	20	25	35	50	65	80	100
	DECAY TIMES (years out of core) (Activities* in Ci/rod)									
TE-129M	3.501E-15	1.561E-31	6.959E-48	3.103E-64	1.383E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-129	6.086E-05	6.086E-05	6.086E-05	6.086E-05	6.086E-05	6.086E-05	6.086E-05	6.086E-05	6.086E-05	6.086E-05
I-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-134	5.977E+01	1.114E+01	2.077E+00	3.873E-01	7.221E-02	2.510E-03	1.626E-05	1.054E-07	6.830E-10	8.253E-13
CS-135	8.937E-04	8.937E-04	8.937E-04	8.937E-04	8.937E-04	8.937E-04	8.937E-04	8.937E-04	8.937E-04	8.937E-04
CS-136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-137	1.730E+02	1.542E+02	1.374E+02	1.224E+02	1.090E+02	8.656E+01	6.122E+01	4.330E+01	3.063E+01	1.930E+01
BA-136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-137M	1.637E+02	1.459E+02	1.300E+02	1.158E+02	1.032E+02	8.189E+01	5.792E+01	4.096E+01	2.897E+01	1.826E+01
BA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-141	3.156E-14	3.992E-31	5.050E-48	6.387E-65	8.078E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-142	5.022E-08	5.022E-08	5.022E-08	5.022E-08	5.022E-08	5.022E-08	5.022E-08	5.022E-08	5.022E-08	5.022E-08
CE-144	2.277E+01	2.659E-01	3.105E-03	3.625E-05	4.233E-07	5.772E-11	9.190E-17	1.463E-22	2.330E-28	4.331E-36
PR-143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR-144	2.277E+01	2.659E-01	3.105E-03	3.625E-05	4.233E-07	5.772E-11	9.190E-17	1.463E-22	2.330E-28	4.331E-36
PR-144M	2.732E-01	3.191E-03	3.726E-05	4.350E-07	5.080E-09	6.926E-13	1.103E-18	1.756E-24	2.796E-30	5.197E-38
ND-144	2.777E-12	2.786E-12	2.786E-12	2.786E-12	2.786E-12	2.786E-12	2.786E-12	2.786E-12	2.786E-12	2.786E-12
ND-147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-145	2.823E-05	2.351E-05	1.934E-05	1.590E-05	1.308E-05	8.843E-06	4.917E-06	2.734E-06	1.520E-06	6.949E-07
PM-147	6.231E+01	1.664E+01	4.445E+00	1.187E+00	3.171E-01	2.262E-02	4.310E-04	8.213E-06	1.565E-07	7.965E-10
PM-148M	3.611E-12	1.796E-25	8.931E-39	4.441E-52	2.209E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-148	2.034E-13	1.011E-26	5.030E-40	2.502E-53	1.244E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM-145	6.700E-06	1.624E-07	3.935E-09	9.534E-11	2.310E-12	1.357E-15	1.931E-20	2.747E-25	3.909E-30	1.348E-36
SM-147	6.632E-09	7.751E-09	8.050E-09	8.130E-09	8.152E-09	8.159E-09	8.159E-09	8.159E-09	8.159E-09	8.159E-09
SM-151	7.859E-01	7.562E-01	7.277E-01	7.002E-01	6.738E-01	6.239E-01	5.558E-01	4.952E-01	4.412E-01	3.783E-01
EU-152	8.270E-03	6.411E-03	4.970E-03	3.852E-03	2.987E-03	1.794E-03	8.359E-04	3.894E-04	1.814E-04	6.550E-05
EU-154	1.401E+01	9.366E+00	6.261E+00	4.186E+00	2.799E+00	1.251E+00	3.735E-01	1.116E-01	3.334E-02	6.659E-03
EU-155	7.232E+00	3.597E+00	1.789E+00	8.900E-01	4.426E-01	1.095E-01	1.348E-02	1.659E-03	2.041E-04	1.249E-05
EU-156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD-153	8.191E-04	4.398E-06	2.361E-08	1.268E-10	6.810E-13	1.964E-17	3.041E-24	4.708E-31	7.290E-38	6.062E-47
TB-160	5.592E-08	1.410E-15	3.557E-23	8.970E-31	2.262E-38	1.439E-53	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL-206	3.867E-11	3.867E-11	3.867E-11	3.867E-11	3.867E-11	3.867E-11	3.867E-11	3.867E-11	3.867E-11	3.867E-11

	DECAY TIMES (years out of core) (Activities* in Ci/rod)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
TL-207	8.190E-09	1.456E-08	2.049E-08	2.606E-08	3.131E-08	4.104E-08	5.424E-08	6.635E-08	7.778E-08	9.236E-08
TL-208	9.550E-06	1.217E-05	1.263E-05	1.237E-05	1.189E-05	1.082E-05	9.372E-06	8.113E-06	7.023E-06	5.799E-06
PB-210	2.969E-11	1.006E-10	2.493E-10	5.007E-10	8.770E-10	2.082E-09	5.268E-09	1.042E-08	1.783E-08	3.158E-08
PB-211	8.213E-09	1.460E-08	2.055E-08	2.613E-08	3.140E-08	4.116E-08	5.439E-08	6.654E-08	7.800E-08	9.262E-08
PB-212	2.658E-05	3.388E-05	3.516E-05	3.442E-05	3.308E-05	3.013E-05	2.609E-05	2.258E-05	1.955E-05	1.614E-05
BI-211	8.213E-09	1.460E-08	2.055E-08	2.613E-08	3.140E-08	4.116E-08	5.439E-08	6.654E-08	7.800E-08	9.262E-08
BI-212	2.658E-05	3.388E-05	3.516E-05	3.442E-05	3.308E-05	3.013E-05	2.609E-05	2.258E-05	1.955E-05	1.614E-05
PO-212	1.703E-05	2.171E-05	2.253E-05	2.205E-05	2.119E-05	1.930E-05	1.671E-05	1.447E-05	1.252E-05	1.034E-05
PO-215	8.213E-09	1.460E-08	2.055E-08	2.613E-08	3.140E-08	4.116E-08	5.439E-08	6.654E-08	7.800E-08	9.262E-08
PO-216	2.658E-05	3.388E-05	3.516E-05	3.442E-05	3.308E-05	3.013E-05	2.609E-05	2.258E-05	1.955E-05	1.614E-05
RN-219	8.213E-09	1.460E-08	2.055E-08	2.613E-08	3.140E-08	4.116E-08	5.439E-08	6.654E-08	7.800E-08	9.262E-08
RN-220	2.658E-05	3.388E-05	3.516E-05	3.442E-05	3.308E-05	3.013E-05	2.609E-05	2.258E-05	1.955E-05	1.614E-05
FR-223	1.133E-10	2.012E-10	2.832E-10	3.601E-10	4.326E-10	5.672E-10	7.498E-10	9.172E-10	1.075E-09	1.277E-09
RA-223	8.213E-09	1.460E-08	2.055E-08	2.613E-08	3.140E-08	4.116E-08	5.439E-08	6.654E-08	7.800E-08	9.262E-08
RA-224	2.658E-05	3.388E-05	3.516E-05	3.442E-05	3.308E-05	3.013E-05	2.609E-05	2.258E-05	1.955E-05	1.614E-05
RA-226	2.939E-10	7.807E-10	1.514E-09	2.502E-09	3.750E-09	7.054E-09	1.412E-08	2.384E-08	3.634E-08	5.752E-08
RA-228	1.835E-10	2.414E-10	2.761E-10	2.967E-10	3.091E-10	3.210E-10	3.264E-10	3.279E-10	3.284E-10	3.290E-10
AC-227	8.211E-09	1.458E-08	2.052E-08	2.609E-08	3.135E-08	4.110E-08	5.434E-08	6.646E-08	7.790E-08	9.253E-08
TH-227	8.100E-09	1.440E-08	2.026E-08	2.577E-08	3.097E-08	4.059E-08	5.365E-08	6.562E-08	7.692E-08	9.135E-08
TH-228	2.657E-05	3.386E-05	3.513E-05	3.439E-05	3.305E-05	3.012E-05	2.608E-05	2.258E-05	1.955E-05	1.614E-05
TH-229	1.156E-09	1.833E-09	2.516E-09	3.204E-09	3.899E-09	5.306E-09	7.466E-09	9.687E-09	1.197E-08	1.513E-08
TH-230	1.692E-07	2.821E-07	3.986E-07	5.184E-07	6.416E-07	8.974E-07	1.303E-06	1.732E-06	2.181E-06	2.809E-06
TH-231	3.258E-05	3.259E-05	3.259E-05	3.259E-05	3.260E-05	3.260E-05	3.262E-05	3.263E-05	3.264E-05	3.265E-05
TH-232	3.270E-10	3.271E-10	3.272E-10	3.273E-10	3.275E-10	3.277E-10	3.280E-10	3.284E-10	3.287E-10	3.292E-10
TH-234	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04
PA-231	4.975E-08	5.320E-08	5.664E-08	6.008E-08	6.352E-08	7.040E-08	8.072E-08	9.104E-08	1.014E-07	1.151E-07
PA-233	5.948E-04	6.001E-04	6.078E-04	6.173E-04	6.282E-04	6.530E-04	6.946E-04	7.385E-04	7.830E-04	8.417E-04
PA-234M	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04
PA-234	7.268E-07	7.268E-07	7.268E-07	7.268E-07	7.268E-07	7.268E-07	7.268E-07	7.268E-07	7.268E-07	7.268E-07
U-232	3.362E-05	3.567E-05	3.507E-05	3.374E-05	3.225E-05	2.933E-05	2.539E-05	2.198E-05	1.903E-05	1.570E-05
U-233	1.429E-06	1.442E-06	1.455E-06	1.468E-06	1.482E-06	1.510E-06	1.554E-06	1.601E-06	1.650E-06	1.721E-06
U-234	2.469E-03	2.550E-03	2.628E-03	2.703E-03	2.775E-03	2.911E-03	3.096E-03	3.261E-03	3.407E-03	3.576E-03
U-235	3.258E-05	3.259E-05	3.259E-05	3.259E-05	3.260E-05	3.260E-05	3.262E-05	3.263E-05	3.264E-05	3.265E-05
U-236	4.683E-04	4.684E-04	4.686E-04	4.687E-04	4.688E-04	4.691E-04	4.695E-04	4.699E-04	4.704E-04	4.709E-04
U-237	5.791E-05	4.553E-05	3.580E-05	2.814E-05	2.213E-05	1.368E-05	6.647E-06	3.231E-06	1.570E-06	6.001E-07

	DECAY TIMES (years out of core) (Activities* in Ci/rod)									
Radionuclide	**5**	**10**	**15**	**20**	**25**	**35**	**50**	**65**	**80**	**100**
U-238	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04	5.591E-04
NP-237	5.948E-04	6.001E-04	6.078E-04	6.173E-04	6.282E-04	6.530E-04	6.946E-04	7.385E-04	7.830E-04	8.417E-04
PU-236	1.340E-04	3.977E-05	1.180E-05	3.503E-06	1.041E-06	9.267E-08	3.521E-09	1.190E-09	1.129E-09	1.128E-09
PU-237	8.206E-15	7.350E-27	6.584E-39	5.898E-51	5.283E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU-238	5.847E+00	5.621E+00	5.404E+00	5.196E+00	4.995E+00	4.617E+00	4.103E+00	3.646E+00	3.241E+00	2.769E+00
PU-239	7.202E-01	7.201E-01	7.201E-01	7.200E-01	7.199E-01	7.197E-01	7.194E-01	7.191E-01	7.188E-01	7.184E-01
PU-240	9.197E-01	9.238E-01	9.271E-01	9.298E-01	9.319E-01	9.348E-01	9.370E-01	9.376E-01	9.373E-01	9.361E-01
PU-241	2.361E+02	1.856E+02	1.459E+02	1.147E+02	9.020E+01	5.575E+01	2.710E+01	1.317E+01	6.401E+00	2.446E+00
PU-242	3.979E-03	3.979E-03	3.979E-03	3.979E-03	3.979E-03	3.979E-03	3.979E-03	3.979E-03	3.979E-03	3.979E-03
PU-244	1.203E-09	1.203E-09	1.203E-09	1.203E-09	1.203E-09	1.203E-09	1.203E-09	1.203E-09	1.203E-09	1.203E-09
AM-241	2.420E+00	4.075E+00	5.359E+00	6.351E+00	7.114E+00	8.139E+00	8.887E+00	9.134E+00	9.140E+00	8.981E+00
AM-242M	2.038E-02	1.992E-02	1.947E-02	1.903E-02	1.860E-02	1.777E-02	1.660E-02	1.550E-02	1.448E-02	1.322E-02
AM-242	2.028E-02	1.982E-02	1.937E-02	1.894E-02	1.851E-02	1.769E-02	1.652E-02	1.543E-02	1.441E-02	1.315E-02
AM-243	3.885E-02	3.883E-02	3.882E-02	3.880E-02	3.878E-02	3.874E-02	3.869E-02	3.863E-02	3.858E-02	3.851E-02
CM-242	6.123E-02	1.642E-02	1.603E-02	1.567E-02	1.531E-02	1.463E-02	1.366E-02	1.276E-02	1.191E-02	1.088E-02
CM-243	3.220E-02	2.852E-02	2.526E-02	2.237E-02	1.981E-02	1.553E-02	1.079E-02	7.492E-03	5.203E-03	3.200E-03
CM-244	9.561E+00	7.897E+00	6.522E+00	5.387E+00	4.449E+00	3.035E+00	1.710E+00	9.635E-01	5.428E-01	2.526E-01
CM-245	1.223E-03	1.222E-03	1.222E-03	1.221E-03	1.221E-03	1.220E-03	1.218E-03	1.217E-03	1.215E-03	1.213E-03
CM-246	1.796E-04	1.795E-04	1.793E-04	1.792E-04	1.791E-04	1.788E-04	1.784E-04	1.780E-04	1.776E-04	1.771E-04
CM-247	6.257E-10	6.257E-10	6.257E-10	6.257E-10	6.257E-10	6.257E-10	6.257E-10	6.257E-10	6.257E-10	6.257E-10
SUBTOTAL**	1.130E+03	7.795E+02	6.452E+02	5.510E+02	4.760E+02	3.611E+02	2.449E+02	1.701E+02	1.207E+02	7.877E+01
TOTAL***	1.130E+03	7.795E+02	6.452E+02	5.510E+02	4.761E+02	3.612E+02	2.449E+02	1.702E+02	1.207E+02	7.879E+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 26

Fuel-Specific Source Term Calculations Aluminum-Clad TRIGA 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 generic source term for aluminum-clad TRIGA (Training, Research, and Isotope General Atomics) spent nuclear fuel elements currently stored at the INEEL. The data sources for the analysis are documented in References 1 and 2, and the INEEL calculational methodology is described in detail in Reference 3.

TRIGA Data

TRIGA reactors are light-water-cooled reactors designed for training, research, and isotope production. One type of fuel element used in a TRIGA reactor is an aluminum-clad, uranium-zirconium-hydride (U-Zr-H) fuel element. The enriched uranium is homogeneously mixed in the ZrH matrix. The cylindrical active fuel region in each aluminum-clad element is approximately 1.4-in. in diameter and 14 or 15 in. in length. Figure 1 shows a typical aluminum-clad fuel element with dimensions and materials. The data below give specific dimensions, materials, loadings, densities, enrichment, etc., for the aluminum-clad element used in the burnup calculation for the source term generation.

Fuel Element:

Fuel Meat:	U-Zr-H	
	Zr:H ratio is 1.0	
	Density = 6.28 g/cc	
Clad:	Aluminum-1100	
	Density = 2.70 g/cc	
Loading:	36.0	g/element U-235 BOL
	144.0	g/element U-238 BOL
	180.0	g/element U BOL
	2070.0	g/element ZrH in fuel meat
	8.0	weight % U in U-$ZrH_{1.0}$
	20%	enrichment U-235 BOL
	280.0	g/element aluminum cladding
	450.0	g/element graphite top/bottom end reflectors
	19.7	g/element Sm_2O_3 (two poison disks)

Active Fuel Length: 14 in.
Fuel Element Length: 28 in. (approximate)

Water Temperature: 77.5°F
Water Pressure: 14.7 psia

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single aluminum-clad TRIGA fuel element. In addition, for the ORIGEN2 (Reference 4) depletion or burnup calculation, conservative and detailed impurity concentrations were added for the aluminum-clad, zirconium-hydride, and graphite end reflector masses. Table 1 gives the impurity concentrations for these three materials.

Burnup

Reference 1 is a parametric study and includes radionuclide inventories or source terms for eight different burnups ranging up to 19.44%. The burnup chosen for this template is based on the 19.44% burnup of the initial U-235 or the maximum burnup used in the parametric study. This burnup is equivalent to 6.65 MWd, 36,944 MWd/MTU, and 8.07 g of U-235 depleted per element and represents the upper end of typical aluminum-clad TRIGA fuel element burnups. The assumption of maximum burnup is conservative for the buildup of fission products, activation products, and minor actinides in the source term and nonconservative with regard to criticality safety.

Cross-Section Development

An MCNP4A (Reference 5) partial core model of a MARK I TRIGA reactor core was used to generate neutron cross sections specifically for the aluminum-clad TRIGA fuel element. The MCNP4A one-twelfth core model is shown in Figure 2. The cross sections are spectrally and spatially weighted over all the fuel elements shown in Figure 2. These cross sections are in turn used in the fuel element ORIGEN2 depletion calculation.

Parametric TRIGA Single Element Exposure History

Table 2 summarizes the single element exposure history of the aluminum-clad TRIGA fuel element from Reference 1. The burnup period is a hypothetical 4-year continuous exposure. TRIGA fuel elements typically remain in the core for much longer periods of time relative to the assumed 4-year in-core residency. Therefore, typical fuel elements would have more time to decay away their source term; therefore, the 4-year assumption is expected to produce a conservative source term.

Burnup Calculation

The ORIGEN2 computer code was used to perform the depletion or burnup calculation for the aluminum-clad TRIGA fuel element. The radionuclide inventory or source term template that follows is for a single aluminum-clad TRIGA fuel element. The fuel element component masses and impurities (fuel meat, uranium, clad, burnable poison, end fixtures), neutron cross sections, burnup, and hypothetical power history and power level as discussed above are input data for the ORIGEN2 calculation. The radionuclide concentrations are given as a function of decay 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, *Radionuclide Mass Inventory, Activity, Decay Heat, and Dose Rate Parametric Data for TRIGA Spent Nuclear Fuels*, INEL-96/0482, Idaho National Engineering Laboratory, March 1997.

2. N. Tomsio, *Characterization of TRIGA Fuel*, ORNL/Sub//86-22047/3, GA-C18542, GA Technologies, October 1986.

3. 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.

4. 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.

5. "MCNP4A: Monte Carlo N-Particle Transport Code System," LA-12625M, contributed by Los Alamos National Laboratory, Los Alamos, New Mexico, 1994, and distributed as package CCC-200 by Oak Ridge National Laboratory.

Table 1. Material constituent and impurity concentrations for the various materials in an aluminum-clad TRIGA fuel element.

Constituent or Impurity	Graphite Concentration (ppm)	ZrH Concentration (wt%)	Aluminum-1100 Concentration (wt%)
H		1.0628	
Li	0.45		
Be	0.005		
B	2.5	0.00005	
C	100 wt%	0.026968	
N		0.00799	
O		0.094887	
Na	10.4		
Mg	1		
Al	4.1	0.007491	99.3
Si	26	0.011986	0.25
P	1	0.009988	
S	9.4	0.003496	
Cl	3		
K	3		
Ca	22.5		
Sc	0.01		
Ti	16	0.004994	
V	18.9	0.004994	
Cr	1	0.124851	
Mn	1	0.004994	0.025
Fe	11.1	0.224731	0.25
Co	4	0.001998	
Ni	4.6	0.006992	
Cu	0.47	0.004994	0.125
Zn	1	0.009988	0.05
Rb	1		
Sr	0.47		
Zr	0.5	98.9082	
Nb	1.74	0.006992	
Mo	1	0.004994	
Ag	0.5		
Cd	0.5	0.000050	
In	1		
Sn	1	1.598089	
Sb	1		
Cs	1		
Ba	2.9		
La	1.38		

Table 1. (continued).

Constituent or Impurity	Graphite Concentration (ppm)	ZrH Concentration (wt%)	Aluminum-1100 Concentration (wt%)
Ce	0.56		
Pr	0.64		
Nd	0.36		
Sm	0.61	0.000999	
Gd	0.08	0.000499	
Tb	0.26		
Dy	0.16		
Ho	0.08		
Er	0.04		
Tm	0.04		
Yb	0.06		
Lu	0.02		
Hf	0.17	0.003496	
Ta	0.35	0.019976	
W	25.5	0.009988	
Tl	1		
Pb	6.9	0.009988	
Bi	1		
Th		0.000699	
U		0.000350	

Table 2. Hypothetical power history for a maximum burnup aluminum-clad TRIGA fuel element

Duration (days)	Cumulative Duration (days)	Time-Averaged Power (MWth)
365	365	0.004555
365	730	0.004555
365	1095	0.004555
365	1460	0.004555
1825	3285	0.0
1825	5110	0.0
1825	6935	0.0
1825	8760	0.0
1825	10585	0.0
3650	14235	0.0
5475	19710	0.0
5475	25185	0.0
5475	30660	0.0
7300	37960	0.0

The ten decay times following the hypothetical 4-year exposure are for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100-year cooling time periods designated for the template methodology.

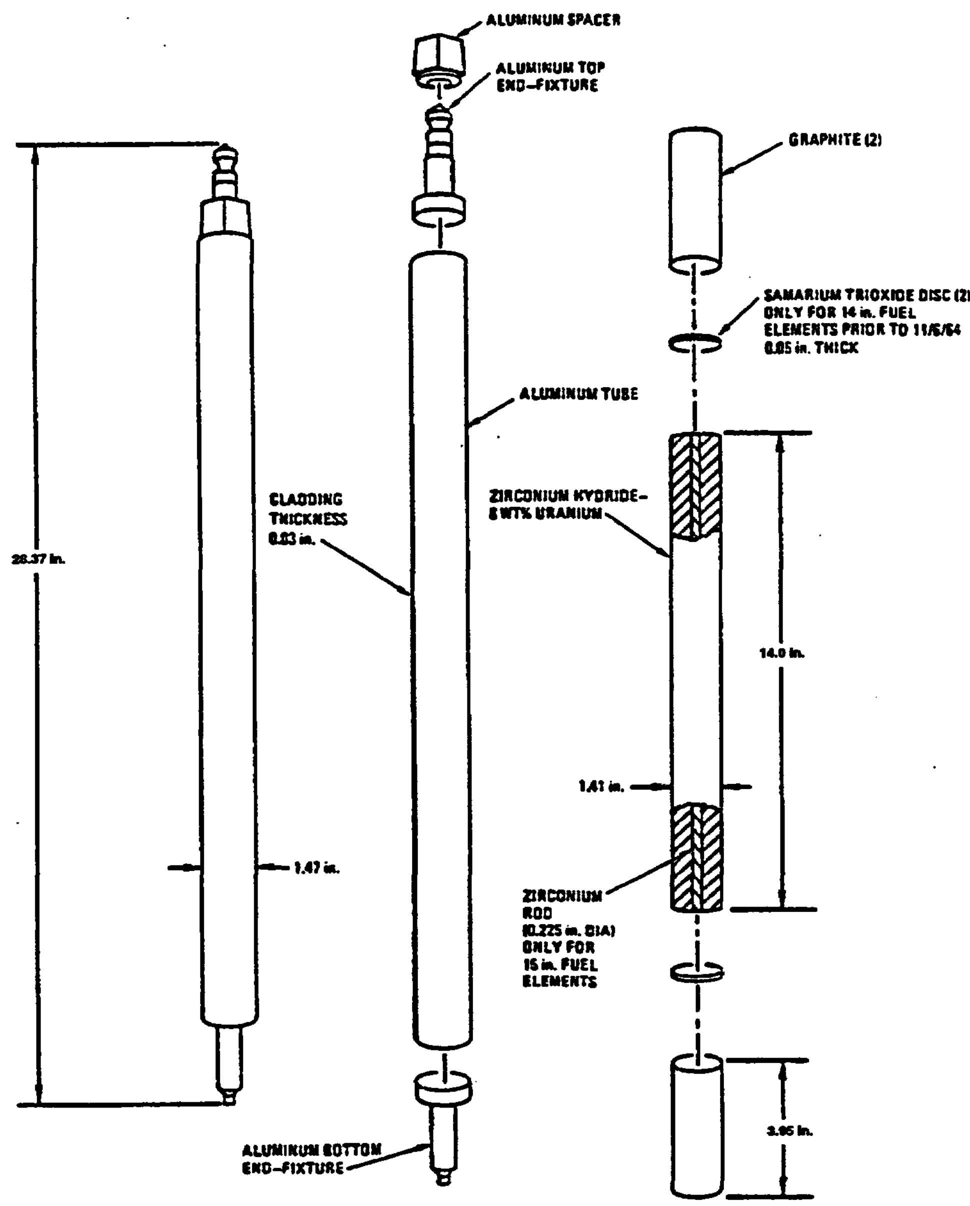

Figure 1. A typical aluminum-clad Mark I TRIGA fuel element.

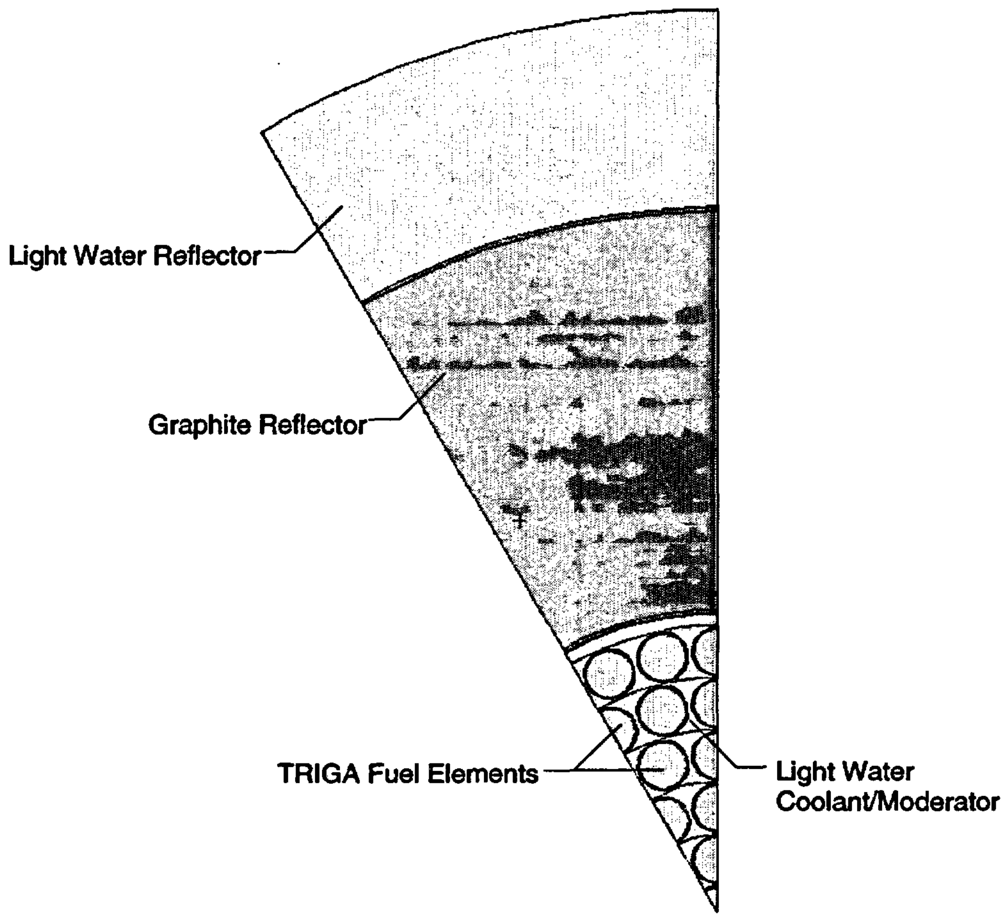

Figure 2. MCNP partial core model of a Mark I TRIGA reactor core.

TRIGA Element

Aluminum Cladding, 10 to 20% Enriched U-235 Fuel

Reactor Moderator/Coolant:	Light Water
Fuel Meat:	U-Zr-$H_{1.0}$
Clad:	Aluminum
Burnup:	6.65 MWd/element (maximum element burnup)
Burnup:	19.44% U-235 burnup (amount fissioned)
Burnup:	8.07 g U-235 depletion (amount fissioned and transmuted)
Basis of Calculation:	Single element
BOL U-235:	36.0 g U-235 per element (design basis)
BOL U-238:	144.0 g U-238 per element
BOL Total U per element:	180.0 g U per element
BOL Fuel Enrichment:	20.0 wt%

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H-3	7.185E-02	5.427E-02	4.100E-02	3.097E-02	2.339E-02	1.335E-02	5.757E-03	2.482E-03	1.070E-03	3.485E-04
BE-10	1.565E-07	1.565E-07	1.565E-07	1.565E-07	1.565E-07	1.565E-07	1.565E-07	1.565E-07	1.565E-07	1.565E-07
C-14	2.880E-04	2.878E-04	2.877E-04	2.875E-04	2.873E-04	2.870E-04	2.865E-04	2.859E-04	2.854E-04	2.847E-04
CL-36	2.861E-07	2.861E-07	2.861E-07	2.861E-07	2.861E-07	2.861E-07	2.861E-07	2.861E-07	2.861E-07	2.860E-07
CR-51	1.398E-20	2.081E-40	3.097E-60	4.611E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN-54	5.494E-04	9.591E-06	1.674E-07	2.923E-09	5.103E-11	1.555E-14	8.274E-20	4.402E-25	2.342E-30	2.176E-37
FE-55	5.771E-02	1.523E-02	4.020E-03	1.061E-03	2.800E-04	1.951E-05	3.587E-07	6.594E-09	1.212E-10	5.883E-13
FE-59	6.039E-15	3.737E-27	2.312E-39	1.431E-51	8.855E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO-60	2.062E-01	1.069E-01	5.540E-02	2.872E-02	1.488E-02	3.998E-03	5.566E-04	7.750E-05	1.079E-05	7.786E-07
NI-59	7.771E-06	7.770E-06	7.770E-06	7.770E-06	7.769E-06	7.769E-06	7.768E-06	7.767E-06	7.766E-06	7.764E-06
NI-63	9.698E-04	9.340E-04	8.995E-04	8.662E-04	8.342E-04	7.737E-04	6.911E-04	6.173E-04	5.514E-04	4.743E-04
ZN-65	3.987E-04	2.228E-06	1.245E-08	6.954E-11	3.885E-13	1.213E-17	2.116E-24	3.690E-31	6.437E-38	6.273E-47
SE-79	8.604E-05	8.603E-05	8.603E-05	8.602E-05	8.602E-05	8.601E-05	8.599E-05	8.598E-05	8.597E-05	8.595E-05
KR-85	1.677E+00	1.214E+00	8.790E-01	6.363E-01	4.606E-01	2.414E-01	9.158E-02	3.475E-02	1.318E-02	3.620E-03
RB-87	5.715E-09	5.715E-09	5.715E-09	5.715E-09	5.715E-09	5.715E-09	5.715E-09	5.715E-09	5.715E-09	5.715E-09
SR-89	2.298E-09	3.034E-20	4.007E-31	5.290E-42	6.984E-53	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR-90	1.729E+01	1.536E+01	1.363E+01	1.210E+01	1.075E+01	8.472E+00	5.930E+00	4.151E+00	2.905E+00	1.805E+00
Y-90	1.730E+01	1.536E+01	1.364E+01	1.211E+01	1.075E+01	8.475E+00	5.932E+00	4.152E+00	2.906E+00	1.806E+00
Y-91	8.752E-08	3.565E-17	1.452E-26	5.917E-36	2.411E-45	4.000E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR-93	5.433E-04	5.433E-04	5.433E-04	5.433E-04	5.433E-04	5.433E-04	5.433E-04	5.433E-04	5.433E-04	5.433E-04
ZR-95	6.734E-07	1.744E-15	4.521E-24	1.172E-32	3.034E-41	2.037E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-93M	1.542E-04	2.355E-04	2.986E-04	3.475E-04	3.854E-04	4.376E-04	4.795E-04	4.991E-04	5.081E-04	5.132E-04

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
NB-94	2.386E-05	2.386E-05	2.385E-05	2.385E-05	2.385E-05	2.384E-05	2.383E-05	2.381E-05	2.380E-05	2.379E-05
NB-95	1.495E-06	3.874E-15	1.004E-23	2.600E-32	6.737E-41	4.523E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-95M	4.996E-09	1.294E-17	3.353E-26	8.690E-35	2.252E-43	1.512E-60	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO-93	5.903E-07	5.897E-07	5.891E-07	5.885E-07	5.880E-07	5.868E-07	5.851E-07	5.833E-07	5.816E-07	5.793E-07
TC-99	2.934E-03	2.934E-03	2.934E-03	2.934E-03	2.934E-03	2.934E-03	2.934E-03	2.934E-03	2.933E-03	2.933E-03
RU-103	1.281E-12	1.325E-26	1.370E-40	1.417E-54	1.466E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU-106	6.234E-01	2.007E-02	6.463E-04	2.081E-05	6.700E-07	6.945E-10	2.318E-14	7.736E-19	2.582E-23	2.775E-29
RH-103M	1.155E-12	1.195E-26	1.235E-40	1.278E-54	1.321E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH-106	6.234E-01	2.007E-02	6.463E-04	2.081E-05	6.700E-07	6.945E-10	2.318E-14	7.737E-19	2.582E-23	2.775E-29
PD-107	4.271E-06	4.271E-06	4.271E-06	4.271E-06	4.271E-06	4.271E-06	4.271E-06	4.271E-06	4.271E-06	4.271E-06
AG-110	9.707E-07	6.146E-09	3.890E-11	2.463E-13	1.559E-15	6.250E-20	1.585E-26	4.024E-33	1.021E-39	1.640E-48
AG-110M	7.299E-05	4.621E-07	2.925E-09	1.852E-11	1.173E-13	4.699E-18	1.193E-24	3.025E-31	7.676E-38	1.233E-46
AG-111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-113M	2.179E-03	1.718E-03	1.355E-03	1.069E-03	8.430E-04	5.244E-04	2.572E-04	1.262E-04	6.191E-05	2.395E-05
CD-113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-115M	3.075E-14	1.474E-26	7.065E-39	3.387E-51	1.624E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114	1.644E-14	1.320E-25	1.060E-36	8.503E-48	6.825E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114M	1.718E-14	1.379E-25	1.107E-36	8.885E-48	7.132E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-115M	2.159E-18	1.035E-30	4.960E-43	2.378E-55	1.140E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-119M	1.902E-02	1.089E-04	6.236E-07	3.570E-09	2.044E-11	6.702E-16	1.258E-22	2.362E-29	4.434E-36	4.767E-45
SN-121M	4.467E-04	4.168E-04	3.889E-04	3.629E-04	3.386E-04	2.947E-04	2.394E-04	1.945E-04	1.580E-04	1.197E-04
SN-123	2.183E-05	1.219E-09	6.802E-14	3.797E-18	2.119E-22	6.600E-31	1.148E-43	1.996E-56	3.469E-69	3.367E-86
SN-125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-126	8.139E-05	8.138E-05	8.138E-05	8.138E-05	8.138E-05	8.137E-05	8.136E-05	8.135E-05	8.134E-05	8.133E-05
SB-124	9.863E-12	7.372E-21	5.510E-30	4.119E-39	3.079E-48	1.720E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB-125	6.261E-01	1.793E-01	5.135E-02	1.471E-02	4.212E-03	3.455E-04	8.116E-06	1.907E-07	4.479E-09	3.013E-11
SB-126	1.139E-05	1.139E-05	1.139E-05	1.139E-05	1.139E-05	1.139E-05	1.139E-05	1.139E-05	1.139E-05	1.139E-05
SB-126M	8.139E-05	8.138E-05	8.138E-05	8.138E-05	8.138E-05	8.137E-05	8.136E-05	8.135E-05	8.134E-05	8.133E-05
TE-123M	5.982E-10	1.536E-14	3.943E-19	1.012E-23	2.598E-28	1.712E-37	2.896E-51	4.899E-65	8.289E-79	3.599E-97
TE-125M	1.527E-01	4.375E-02	1.253E-02	3.588E-03	1.028E-03	8.430E-05	1.980E-06	4.652E-08	1.093E-09	7.352E-12
TE-127	8.452E-06	7.709E-11	7.031E-16	6.413E-21	5.849E-26	4.866E-36	3.692E-51	2.801E-66	2.126E-81	0.000E+00
TE-127M	8.629E-06	7.871E-11	7.179E-16	6.547E-21	5.972E-26	4.968E-36	3.769E-51	2.860E-66	2.170E-81	0.000E+00
TE-129	1.160E-16	5.172E-33	2.306E-49	1.028E-65	4.583E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-129M	1.782E-16	7.945E-33	3.542E-49	1.579E-65	7.041E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-129	4.908E-06	4.908E-06	4.908E-06	4.908E-06	4.908E-06	4.908E-06	4.908E-06	4.908E-06	4.908E-06	4.908E-06

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	**5**	**10**	**15**	**20**	**25**	**35**	**50**	**65**	**80**	**100**
I-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-134	6.894E-01	1.285E-01	2.396E-02	4.468E-03	8.329E-04	2.895E-05	1.876E-07	1.216E-09	7.879E-12	9.520E-15
CS-135	2.098E-04	2.098E-04	2.098E-04	2.098E-04	2.098E-04	2.098E-04	2.098E-04	2.098E-04	2.098E-04	2.098E-04
CS-136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-137	1.833E+01	1.633E+01	1.455E+01	1.296E+01	1.155E+01	9.169E+00	6.485E+00	4.586E+00	3.244E+00	2.044E+00
BA-136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-137M	1.734E+01	1.545E+01	1.376E+01	1.226E+01	1.093E+01	8.674E+00	6.135E+00	4.339E+00	3.069E+00	1.934E+00
BA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-141	2.761E-15	3.492E-32	4.417E-49	5.587E-66	7.066E-83	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-142	5.891E-09	5.891E-09	5.891E-09	5.891E-09	5.891E-09	5.891E-09	5.891E-09	5.891E-09	5.891E-09	5.891E-09
CE-144	2.304E+00	2.691E-02	3.142E-04	3.669E-06	4.284E-08	5.841E-12	9.300E-18	1.481E-23	2.358E-29	4.383E-37
PR-143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR-144	2.304E+00	2.691E-02	3.142E-04	3.669E-06	4.284E-08	5.841E-12	9.301E-18	1.481E-23	2.358E-29	4.383E-37
PR-144M	2.765E-02	3.229E-04	3.770E-06	4.403E-08	5.141E-10	7.010E-14	1.116E-19	1.777E-25	2.829E-31	5.260E-39
ND-144	2.873E-13	2.882E-13	2.882E-13	2.882E-13	2.882E-13	2.882E-13	2.882E-13	2.882E-13	2.882E-13	2.882E-13
ND-147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-145	8.891E-03	7.387E-03	6.076E-03	4.996E-03	4.108E-03	2.778E-03	1.545E-03	8.587E-04	4.775E-04	2.183E-04
PM-147	1.381E+01	3.688E+00	9.849E-01	2.631E-01	7.027E-02	5.013E-03	9.551E-05	1.820E-06	3.468E-08	1.765E-10
PM-148M	1.009E-13	5.016E-27	2.495E-40	1.241E-53	6.170E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-148	5.681E-15	2.825E-28	1.405E-41	6.987E-55	3.475E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM-145	1.706E-03	4.135E-05	1.002E-06	2.428E-08	5.884E-10	3.455E-13	4.916E-18	6.996E-23	9.955E-28	3.433E-34
SM-147	6.419E-08	6.444E-08	6.451E-08	6.452E-08	6.453E-08	6.453E-08	6.453E-08	6.453E-08	6.453E-08	6.453E-08
SM-151	1.810E+00	1.742E+00	1.676E+00	1.612E+00	1.552E+00	1.437E+00	1.280E+00	1.141E+00	1.016E+00	8.711E-01
EU-152	5.693E-02	4.413E-02	3.421E-02	2.652E-02	2.056E-02	1.236E-02	5.755E-03	2.680E-03	1.249E-03	4.508E-04
EU-154	8.971E+00	5.997E+00	4.009E+00	2.680E+00	1.792E+00	8.007E-01	2.392E-01	7.146E-02	2.135E-02	4.264E-03
EU-155	2.918E+00	1.451E+00	7.220E-01	3.591E-01	1.786E-01	4.419E-02	5.438E-03	6.691E-04	8.235E-05	5.040E-06
EU-156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD-153	1.521E-04	8.172E-07	4.388E-09	2.356E-11	1.266E-13	3.648E-18	5.649E-25	8.747E-32	1.355E-38	1.126E-47
TB-160	1.431E-10	3.609E-18	9.101E-26	2.295E-33	5.788E-41	3.681E-56	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL-206	1.688E-12	1.688E-12	1.688E-12	1.688E-12	1.688E-12	1.688E-12	1.688E-12	1.688E-12	1.688E-12	1.688E-12
TL-207	2,692E-09	5,352E-09	8.560E-09	1.223E-08	1.631E-08	2.541E-08	4.083E-08	5.765E-08	7.534E-08	9.970E-08
TL-208	1.209E-07	1.294E-07	1.262E-07	1.210E-07	1.155E-07	1.049E-07	9.089E-08	7.876E-08	6.825E-08	5.645E-08

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

Radionuclide	5	10	15	20	25	35	50	65	80	100
PB-210	2.012E-14	5.487E-14	1.253E-13	2.431E-13	4.200E-13	9.972E-13	2.589E-12	5.305E-12	9.409E-12	1.745E-11
PB-211	2.699E-09	5.367E-09	8.584E-09	1.227E-08	1.635E-08	2.548E-08	4.094E-08	5.781E-08	7.555E-08	9.998E-08
PB-212	3.364E-07	3.603E-07	3.513E-07	3.367E-07	3.214E-07	2.920E-07	2.530E-07	2.192E-07	1.899E-07	1.571E-07
BI-211	2.699E-09	5.367E-09	8.584E-09	1.227E-08	1.635E-08	2.548E-08	4.094E-08	5.781E-08	7.555E-08	9.998E-08
BI-212	3.364E-07	3.603E-07	3.513E-07	3.367E-07	3.214E-07	2.920E-07	2.530E-07	2.192E-07	1.899E-07	1.571E-07
PO-212	2.155E-07	2.308E-07	2.251E-07	2.157E-07	2.059E-07	1.871E-07	1.621E-07	1.404E-07	1.217E-07	1.007E-07
PO-215	2.699E-09	5.367E-09	8.584E-09	1.227E-08	1.635E-08	2.548E-08	4.094E-08	5.781E-08	7.555E-08	9.998E-08
PO-216	3.364E-07	3.603E-07	3.513E-07	3.367E-07	3.214E-07	2.920E-07	2.530E-07	2.192E-07	1.899E-07	1.571E-07
RN-219	2.699E-09	5.367E-09	8.584E-09	1.227E-08	1.635E-08	2.548E-08	4.094E-08	5.781E-08	7.555E-08	9.998E-08
RN-220	3.364E-07	3.603E-07	3.513E-07	3.367E-07	3.214E-07	2.920E-07	2.530E-07	2.192E-07	1.899E-07	1.571E-07
FR-223	3.723E-11	7.399E-11	1.183E-10	1.691E-10	2.253E-10	3.512E-10	5.645E-10	7.970E-10	1.041E-09	1.378E-09
RA-223	2.699E-09	5.367E-09	8.584E-09	1.227E-08	1.635E-08	2.548E-08	4.094E-08	5.781E-08	7.555E-08	9.998E-08
RA-224	3.364E-07	3.603E-07	3.513E-07	3.367E-07	3.214E-07	2.920E-07	2.530E-07	2.192E-07	1.899E-07	1.571E-07
RA-226	1.546E-13	3.814E-13	7.197E-13	1.181E-12	1.776E-12	3.411E-12	7.118E-12	1.254E-11	1.990E-11	3.302E-11
RA-228	9.560E-10	1.207E-09	1.357E-09	1.447E-09	1.500E-09	1.551E-09	1.573E-09	1.578E-09	1.579E-09	1.579E-09
AC-227	2.698E-09	5.362E-09	8.573E-09	1.225E-08	1.633E-08	2.545E-08	4.090E-08	5.775E-08	7.547E-08	9.988E-08
TH-227	2.662E-09	5.293E-09	8.465E-09	1.210E-08	1.613E-08	2.513E-08	4.038E-08	5.702E-08	7.451E-08	9.860E-08
TH-228	3.363E-07	3.600E-07	3.510E-07	3.364E-07	3.211E-07	2.919E-07	2.530E-07	2.192E-07	1.899E-07	1.571E-07
TH-229	5.691E-10	9.808E-10	1.392E-09	1.804E-09	2.215E-09	3.038E-09	4.271E-09	5.503E-09	6.734E-09	8.374E-09
TH-230	8.096E-11	1.300E-10	1.845E-10	2.444E-10	3.094E-10	4.540E-10	7.045E-10	9.914E-10	1.311E-09	1.781E-09
TH-231	6.040E-05	6.040E-05	6.040E-05	6.040E-05	6.040E-05	6.040E-05	6.041E-05	6.041E-05	6.041E-05	6.041E-05
TH-232	1.579E-09	1.579E-09	1.579E-09	1.579E-09	1.579E-09	1.579E-09	1.579E-09	1.579E-09	1.579E-09	1.579E-09
TH-234	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05
PA-231	1.753E-08	2.392E-08	3.030E-08	3.669E-08	4.308E-08	5.584E-08	7.498E-08	9.411E-08	1.132E-07	1.387E-07
PA-233	9.598E-06	9.618E-06	9.646E-06	9.681E-06	9.721E-06	9.810E-06	9.960E-06	1.012E-05	1.028E-05	1.049E-05
PA-234M	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05
PA-234	6.259E-08	6.259E-08	6.259E-08	6.259E-08	6.259E-08	6.259E-08	6.259E-08	6.259E-08	6.259E-08	6.259E-08
U-232	3.725E-07	3.581E-07	3.423E-07	3.265E-07	3.112E-07	2.827E-07	2.447E-07	2.118E-07	1.834E-07	1.513E-07
U-233	8.731E-07	8.733E-07	8.735E-07	8.736E-07	8.738E-07	8.742E-07	8.748E-07	8.754E-07	8.760E-07	8.769E-07
U-234	1.027E-06	1.152E-06	1.273E-06	1.389E-06	1.501E-06	1.711E-06	1.997E-06	2.252E-06	2.478E-06	2.742E-06
U-235	6.040E-05	6.040E-05	6.040E-05	6.040E-05	6.040E-05	6.040E-05	6.041E-05	6.041E-05	6.041E-05	6.041E-05
U-236	8.457E-05	8.457E-05	8.458E-05	8.458E-05	8.458E-05	8.458E-05	8.459E-05	8.460E-05	8.460E-05	8.461E-05
U-237	2.051E-07	1.613E-07	1.268E-07	9.969E-08	7.838E-08	4.845E-08	2.355E-08	1.144E-08	5.561E-09	2.125E-09
U-238	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05	4.814E-05
NP-237	9.598E-06	9.618E-06	9.646E-06	9.681E-06	9.721E-06	9.810E-06	9.960E-06	1.012E-05	1.028E-05	1.049E-05

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	**5**	**10**	**15**	**20**	**25**	**35**	**50**	**65**	**80**	**100**
PU-236	1.173E-07	3.483E-08	1.034E-08	3.069E-09	9.121E-10	8.206E-11	3.991E-12	1.950E-12	1.897E-12	1.895E-12
PU-237	1.346E-18	1.206E-30	1.080E-42	9.678E-55	8.669E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU-238	8.987E-03	8.640E-03	8.306E-03	7.985E-03	7.676E-03	7.094E-03	6.303E-03	5.600E-03	4.976E-03	4.250E-03
PU-239	3.787E-02	3.786E-02	3.786E-02	3.785E-02	3.785E-02	3.784E-02	3.782E-02	3.780E-02	3.779E-02	3.776E-02
PU-240	1.506E-02	1.505E-02	1.504E-02	1.504E-02	1.503E-02	1.501E-02	1.499E-02	1.496E-02	1.494E-02	1.491E-02
PU-241	8.362E-01	6.574E-01	5.169E-01	4.064E-01	3.195E-01	1.975E-01	9.598E-02	4.665E-02	2.267E-02	8.661E-03
PU-242	2.035E-06	2.035E-06	2.035E-06	2.035E-06	2.035E-06	2.035E-06	2.035E-06	2.035E-06	2.035E-06	2.035E-06
PU-244	4.531E-15	4.531E-15	4.531E-15	4.531E-15	4.531E-15	4.531E-15	4.531E-15	4.531E-15	4.531E-15	4.531E-15
AM-241	9.166E-03	1.502E-02	1.957E-02	2.308E-02	2.577E-02	2.939E-02	3.203E-02	3.289E-02	3.290E-02	3.232E-02
AM-242M	1.325E-05	1.296E-05	1.266E-05	1.238E-05	1.210E-05	1.156E-05	1.080E-05	1.008E-05	9.417E-06	8.597E-06
AM-242	1.319E-05	1.289E-05	1.260E-05	1.232E-05	1.204E-05	1.150E-05	1.074E-05	1.003E-05	9.370E-06	8.554E-06
AM-243	1.551E-06	1.550E-06	1.550E-06	1.549E-06	1.548E-06	1.547E-06	1.545E-06	1.542E-06	1.540E-06	1.537E-06
CM-242	3.330E-05	1.067E-05	1.043E-05	1.019E-05	9.961E-06	9.513E-06	8.884E-06	8.297E-06	7.749E-06	7.074E-06
CM-243	1.824E-06	1.615E-06	1.430E-06	1.267E-06	1.122E-06	8.797E-07	6.110E-07	4.243E-07	2.947E-07	1.812E-07
CM-244	2.095E-05	1.730E-05	1.429E-05	1.180E-05	9.749E-06	6.650E-06	3.747E-06	2.111E-06	1.189E-06	5.535E-07
CM-245	2.407E-10	2.406E-10	2.405E-10	2.404E-10	2.403E-10	2.401E-10	2.398E-10	2.395E-10	2.392E-10	2.388E-10
CM-246	7.528E-12	7.522E-12	7.517E-12	7.511E-12	7.506E-12	7.495E-12	7.478E-12	7.462E-12	7.446E-12	7.424E-12
CM-247	1.887E-18	1.887E-18	1.887E-18	1.887E-18	1.887E-18	1.887E-18	1.887E-18	1.887E-18	1.887E-18	1.887E-18
SUBTOTAL**	1.081E+02	7.800E+01	6.469E+01	5.559E+01	4.851E+01	3.764E+01	2.630E+01	1.862E+01	1.330E+01	8.573E+00
TOTAL***	1.081E+02	7.801E+01	6.469E+01	5.560E+01	4.852E+01	3.764E+01	2.631E+01	1.863E+01	1.330E+01	8.573E+00

* 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 27

Fuel-Specific Source Term Calculations TRIGA FLIP 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 for the Training, Research, and Isotope General Atomics (TRIGA) high-enrichment Fuel Life Improvement Program (FLIP) spent nuclear fuel elements currently stored at the INEEL. The data sources are documented in References 1 and 2 and the INEEL calculational methodology is described in detail in Reference 3.

TRIGA Data

TRIGA reactors are light-water-cooled reactors designed for training, research, and isotope production. One type of fuel element used in a TRIGA reactor is a highly enriched, stainless steel-clad, uranium-zirconium-hydride (U-Zr-H) fuel element. The highly enriched uranium is homogenously mixed in the ZrH matrix. The cylindrical active fuel region in each element is approximately 1.4 in. in diameter and 15 in. in length. Figure 1 shows a typical TRIGA FLIP fuel element with dimensions and materials. The data below give specific dimensions, materials, loadings, densities, enrichment, etc., for the fuel element used in the burnup calculation for the source term generation.

Fuel Element:

Fuel Meat:	U-Zr-H		
	Zr:H ratio is 1.6		
	Density = 5.92 g/cc		
Clad:	Stainless Steel		
	Density = 7.92 g/cc		
Loading:	137.0	g/element U-235 BOL	
	59.0	g/element U-238 BOL	
	196.0	g/element U BOL	
	2060.0	g/element Zr in fuel meat	
	8.5	weight % U in $U\text{-}ZrH_{1.6}$	
	70%	enrichment U-235 BOL	
	800.0	g/element stainless steel cladding (819.414 g/element with impurities)	
	450.0	g/element carbon in reflector	
	36.0	g/element natural erbium poison	

Active Fuel Length:	15 in.
Fuel Element Length:	29 in. (approximate)
Water Temperature:	77.5°F
Water Pressure:	14.7 psia

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single TRIGA FLIP fuel element. In addition, for the ORIGEN2 (Reference 4) depletion or burnup calculation, conservative and detailed impurity

concentrations were added for the stainless steel clad, zirconium-hydride, and graphite end reflectors. Table 1 gives the impurity concentrations for these materials.

Burnup

Reference 1 is a parametric study and includes radionuclide inventories or source terms for 14 different burnups ranging up to 51.09%. The burnup chosen for this template is based on the 51.09% burnup of the initial U-235 or the maximum burnup used in the parametric study. This burnup is equivalent to 66.52 MWd, 339,388 MWd/MTU, and 81.84 grams of U-235 depleted per element and represents the upper end of typical TRIGA FLIP fuel element burnups. The assumption of maximum burnup is conservative for the buildup of fission products, activation products, and minor actinides in the source term and nonconservative with regard to criticality safety.

Cross-Section Development

An MCNP4A (Reference 5) partial core model of a MARK I TRIGA reactor core was used to generate neutron cross sections specifically for the TRIGA FLIP fuel element. The MCNP4A one-twelfth core model is shown in Figure 2 with all fuel elements modeled as TRIGA FLIP elements. The cross sections are spectrally and spatially weighted over all the fuel elements shown in Figure 2. These cross sections are in turn used in the fuel element ORIGEN2 depletion calculation.

Parametric TRIGA Single Element Exposure History

Table 2 summarizes the single element exposure history of the TRIGA FLIP fuel element from Reference 1. The burnup period is a hypothetical 4-year continuous exposure. TRIGA fuel elements typically remain in the core for much longer periods of time relative to the assumed 4-year in-core residency. Therefore, typical fuel elements would have more time to decay away their source term; therefore, the 4-year assumption is expected to produce a conservative source term.

Burnup Calculation

The ORIGEN2 computer code was used to perform the depletion or burnup calculation for the TRIGA FLIP fuel element. The radionuclide inventory or source term template that follows is for a single TRIGA FLIP fuel element. The fuel element component masses and impurities (fuel meat, uranium, clad, burnable poison, end fixtures), neutron cross sections, burnup, and hypothetical power history and power level as discussed above are input data for the ORIGEN2 calculation. The radionuclide concentrations are given as a function of decay time in the template table.

The 145 radionuclides listed in the template represent greater than 99.1% 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, *Radionuclide Mass Inventory, Activity, Decay Heat, and Dose Rate Parametric Data for TRIGA Spent Nuclear Fuels*, INEL-96/0482, Idaho National Engineering Laboratory, March 1997.

2. N. Tomsio, *Characterization of TRIGA Fuel*, ORNL/Sub//86-22047/3, GA-C18542, GA Technologies, October 1986.

3. 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.

4. 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.

5. "MCNP4A: Monte Carlo N-Particle Transport Code System," LA-12625M, contributed by Los Alamos National Laboratory, Los Alamos, New Mexico, 1994, and distributed as package CCC-200 by Oak Ridge National Laboratory.

Table 1. Material constituent and impurity concentrations for the various materials in a TRIGA FLIP fuel element.

Constituent or Impurity	Graphite Concentration (ppm)	ZrH Concentration (wt%)	Stainless Steel Concentration (ppm)
H		1.7373	
Li	0.45		0.13
Be	0.005		
B	2.5	0.00005	
C	100 wt%	0.026968	0.08 wt%
N		0.00799	525
O		0.094887	
Na	10.4		37
Mg	1		
Al	4.1	0.007491	200
Si	26	0.011986	1.00 wt%
P	1	0.009988	
S	9.4	0.003496	
Cl	3		130
K	3		3
Ca	22.5		19
Sc	0.01		0.03
Ti	16	0.004994	600
V	18.9	0.004994	690
Cr	1	0.124851	18.40 wt%
Mn	1	0.004994	1.53 wt%
Fe	11.1	0.224731	68.99 wt%
Co	4	0.001998	2570
Ni	4.6	0.006992	10.00 wt%
Cu	0.47	0.004994	8150
Zn	1	0.009988	2230
Ga			450
As			1010
Se			70
Br			8
Rb	1		10
Sr	0.47		0.2
Y			5
Zr	0.5	98.2627	20
Nb	1.74	0.006992	300
Mo	1	0.004994	5500
Ag	0.5		2

Table 1. (continued).

Constituent or Impurity	Graphite Concentration (ppm)	ZrH Concentration (wt%)	Stainless Steel Concentration (ppm)
Cd	0.5	0.000050	
In	1		
Sn	1	1.598089	
Sb	1		17
Cs	1		0.3
Ba	2.9		500
La	1.38		2.1
Ce	0.56		550
Pr	0.64		
Nd	0.36		
Sm	0.61	0.000999	0.15
Eu			0.02
Gd	0.08	0.000499	
Tb	0.26		0.71
Dy	0.16		1
Ho	0.08		1
Er	0.04		
Tm	0.04		
Yb	0.06		2
Lu	0.02		0.8
Hf	0.17	0.003496	2
Ta	0.35	0.019976	
W	25.5	0.009988	520
Tl	1		
Pb	6.9	0.009988	139
Bi	1		
Th		0.000699	1
U		0.000350	2

Table 2. Hypothetical power history for a maximum burnup TRIGA FLIP fuel element.

Duration (days)	Cumulative Duration (days)	Time-Averaged Power (MWth)
365	365	0.04555
365	730	0.04555
365	1095	0.04555
365	1460	0.04555
1825	3285	0.0
1825	5110	0.0
1825	6935	0.0
1825	8760	0.0
1825	10585	0.0
3650	14235	0.0
5475	19710	0.0
5475	25185	0.0
5475	30660	0.0
7300	37960	0.0

The ten decay times following the hypothetical 4-year exposure are for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100-year cooling time periods designated for the template methodology.

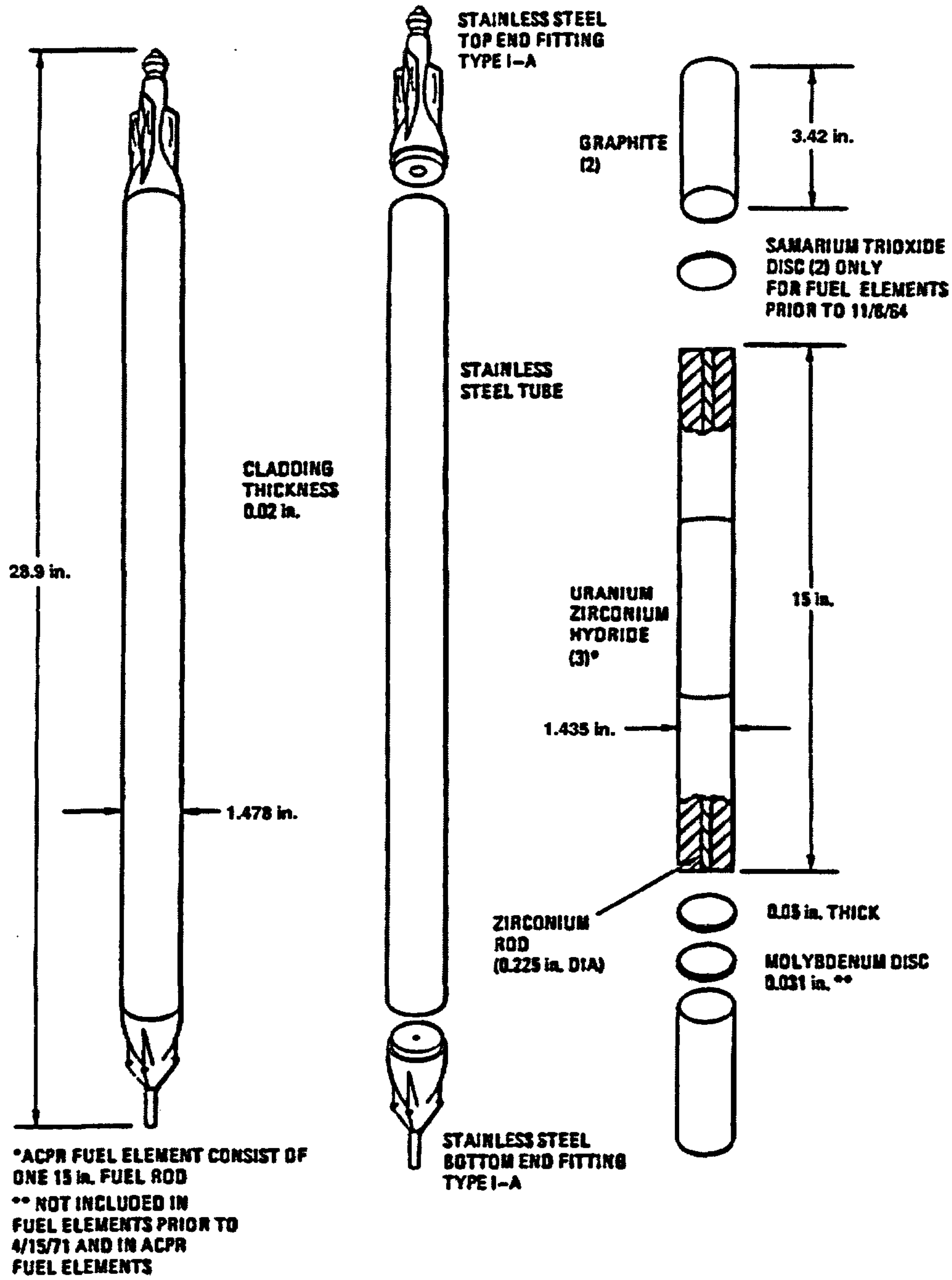

Figure 1. A typical stainless steel-clad Mark I TRIGA fuel element.

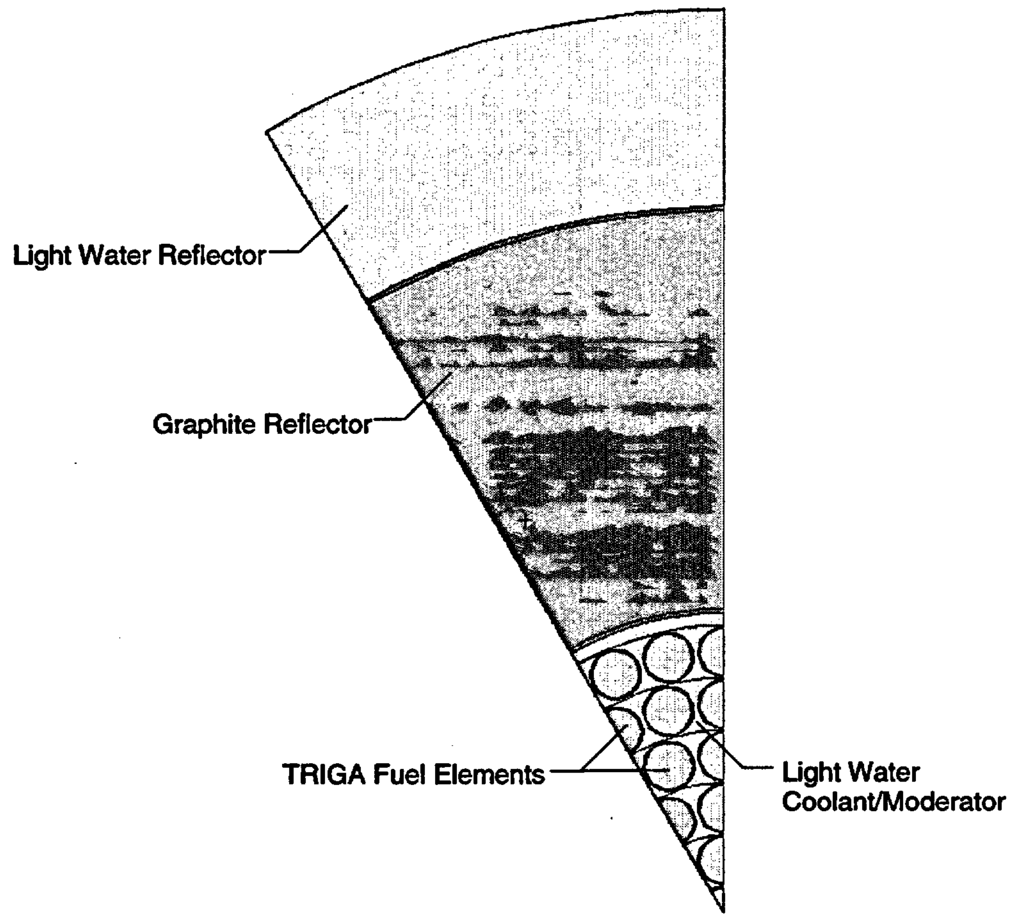

Figure 2. MCNP partial core model of a Mark I TRIGA reactor core.

TRIGA Element

FLIP, 60 to 100% Enriched U-235 Fuel

Reactor Moderator/Coolant:	Light Water
Fuel Meat:	$U\text{-}Zr\text{-}H_{1.6}$
Clad:	Stainless Steel
Burnup:	66.52 MWd/element (maximum element burnup)
Burnup:	51.09% U-235 burnup (amount fissioned)
Burnup:	81.84 g U-235 depletion (amount fissioned and transmuted)
Basis of Calculation:	Single element
BOL U-235:	137.0 g U-235 per element (design basis)
BOL U-238:	59.0 g U-238 per element
BOL Total U per element:	196.0 g U per element
BOL Fuel Enrichment:	70.0 wt%

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H-3	7.005E-01	5.292E-01	3.998E-01	3.020E-01	2.282E-01	1.302E-01	5.613E-02	2.420E-02	1.043E-02	3.398E-03
BE-10	1.313E-06	1.313E-06	1.313E-06	1.313E-06	1.313E-06	1.313E-06	1.313E-06	1.313E-06	1.313E-06	1.313E-06
C-14	8.375E-03	8.369E-03	8.364E-03	8.359E-03	8.354E-03	8.344E-03	8.329E-03	8.314E-03	8.299E-03	8.279E-03
CL-36	1.771E-04	1.771E-04	1.771E-04	1.771E-04	1.771E-04	1.771E-04	1.771E-04	1.771E-04	1.771E-04	1.770E-04
CR-51	8.242E-18	1.227E-37	1.826E-57	2.718E-77	4.046E-97	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN-54	5.368E-01	9.371E-03	1.636E-04	2.856E-06	4.986E-08	1.520E-11	8.085E-17	4.302E-22	2.289E-27	2.126E-34
FE-55	5.292E+01	1.397E+01	3.686E+00	9.730E-01	2.568E-01	1.789E-02	3.289E-04	6.047E-06	1.112E-07	5.395E-10
FE-59	6.733E-12	4.166E-24	2.578E-36	1.595E-48	9.873E-61	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO-60	8.255E+01	4.278E+01	2.218E+01	1.149E+01	5.957E+00	1.600E+00	2.228E-01	3.102E-02	4.318E-03	3.116E-04
NI-59	3.223E-02	3.223E-02	3.223E-02	3.223E-02	3.223E-02	3.222E-02	3.222E-02	3.221E-02	3.221E-02	3.220E-02
NI-63	4.106E+00	3.955E+00	3.808E+00	3.668E+00	3.532E+00	3.276E+00	2.926E+00	2.614E+00	2.335E+00	2.008E+00
ZN-65	2.221E-02	1.241E-04	6.934E-07	3.874E-09	2.165E-11	6.757E-16	1.179E-22	2.056E-29	3.586E-36	3.495E-45
SE-79	8.536E-04	8.535E-04	8.535E-04	8.534E-04	8.534E-04	8.533E-04	8.532E-04	8.530E-04	8.529E-04	8.527E-04
KR-85	1.660E+01	1.202E+01	8.699E+00	6.297E+00	4.559E+00	2.389E+00	9.063E-01	3.438E-01	1.304E-01	3.583E-02
RB-87	5.649E-08	5.649E-08	5.649E-08	5.649E-08	5.649E-08	5.649E-08	5.649E-08	5.649E-08	5.649E-08	5.649E-08
SR-89	2.078E-08	2.744E-19	3.624E-30	4.785E-41	6.318E-52	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR-90	1.706E+02	1.514E+02	1.344E+02	1.194E+02	1.060E+02	8.355E+01	5.848E+01	4.093E+01	2.865E+01	1.780E+01
Y-90	1.706E+02	1.515E+02	1.345E+02	1.194E+02	1.060E+02	8.357E+01	5.849E+01	4.094E+01	2.866E+01	1.781E+01
Y-91	7.992E-07	3.256E-16	1.326E-25	5.403E-35	2.201E-44	3.653E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR-93	5.170E-03	5.170E-03	5.170E-03	5.170E-03	5.170E-03	5.170E-03	5.170E-03	5.169E-03	5.169E-03	5.169E-03
ZR-95	6.280E-06	1.627E-14	4.216E-23	1.092E-31	2.830E-40	1.900E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-93M	1.466E-03	2.240E-03	2.841E-03	3.306E-03	3.667E-03	4.163E-03	4.562E-03	4.749E-03	4.836E-03	4.884E-03

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
NB-94	5.086E-04	5.085E-04	5.084E-04	5.084E-04	5.083E-04	5.081E-04	5.078E-04	5.076E-04	5.073E-04	5.070E-04
NB-95	1.394E-05	3.613E-14	9.360E-23	2.425E-31	6.284E-40	4.218E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-95M	4.660E-08	1.208E-16	3.128E-25	8.104E-34	2.100E-42	1.410E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO-93	2.140E-04	2.138E-04	2.136E-04	2.134E-04	2.132E-04	2.127E-04	2.121E-04	2.115E-04	2.108E-04	2.100E-04
TC-99	2.682E-02	2.682E-02	2.682E-02	2.682E-02	2.682E-02	2.682E-02	2.681E-02	2.681E-02	2.681E-02	2.681E-02
RU-103	1.208E-11	1.249E-25	1.292E-39	1.336E-53	1.381E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU-106	6.880E+00	2.215E-01	7.132E-03	2.296E-04	7.394E-06	7.664E-09	2.558E-13	8.538E-18	2.850E-22	3.062E-28
RH-103M	1.089E-11	1.126E-25	1.164E-39	1.204E-53	1.245E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH-106	6.880E+00	2.215E-01	7.132E-03	2.296E-04	7.394E-06	7.665E-09	2.558E-13	8.538E-18	2.850E-22	3.062E-28
PD-107	4.694E-05	4.694E-05	4.694E-05	4.694E-05	4.694E-05	4.694E-05	4.694E-05	4.694E-05	4.694E-05	4.694E-05
AG-110	8.759E-05	5.545E-07	3.510E-09	2.222E-11	1.407E-13	5.639E-18	1.431E-24	3.630E-31	9.212E-38	1.480E-46
AG-110M	6.585E-03	4.169E-05	2.639E-07	1.671E-09	1.058E-11	4.240E-16	1.076E-22	2.729E-29	6.926E-36	1.113E-44
AG-111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-113M	2.366E-02	1.866E-02	1.472E-02	1.161E-02	9.155E-03	5.695E-03	2.794E-03	1.371E-03	6.724E-04	2.601E-04
CD-113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-115M	2.911E-13	1.396E-25	6.689E-38	3.206E-50	1.537E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114	1.148E-12	9.216E-24	7.397E-35	5.937E-46	4.766E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114M	1.200E-12	9.629E-24	7.729E-35	6.204E-46	4.980E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-115M	2.043E-17	9.794E-30	4.695E-42	2.251E-54	1.079E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-119M	1.829E-01	1.048E-03	5.999E-06	3.435E-08	1.967E-10	6.449E-15	1.210E-21	2.273E-28	4.266E-35	4.586E-44
SN-121M	3.765E-03	3.513E-03	3.278E-03	3.058E-03	2.854E-03	2.484E-03	2.018E-03	1.639E-03	1.332E-03	1.009E-03
SN-123	2.127E-04	1.187E-08	6.626E-13	3.698E-17	2.064E-21	6.431E-30	1.118E-42	1.944E-55	3.380E-68	3.280E-85
SN-125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-126	8.042E-04	8.042E-04	8.042E-04	8.041E-04	8.041E-04	8.041E-04	8.040E-04	8.039E-04	8.038E-04	8.037E-04
SB-124	1.148E-09	8.584E-19	6.416E-28	4.795E-37	3.585E-46	2.003E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB-125	5.752E+00	1.647E+00	4.718E-01	1.351E-01	3.870E-02	3.174E-03	7.457E-05	1.752E-06	4.115E-08	2.769E-10
SB-126	1.126E-04	1.126E-04	1.126E-04	1.126E-04	1.126E-04	1.126E-04	1.126E-04	1.125E-04	1.125E-04	1.125E-04
SB-126M	8.042E-04	8.042E-04	8.042E-04	8.041E-04	8.041E-04	8.041E-04	8.040E-04	8.039E-04	8.038E-04	8.037E-04
TE-123M	6.604E-07	1.695E-11	4.352E-16	1.117E-20	2.868E-25	1.890E-34	3.197E-48	5.409E-62	9.150E-76	3.974E-94
TE-125M	1.403E+00	4.020E-01	1.151E-01	3.297E-02	9.442E-03	7.745E-04	1.819E-05	4.274E-07	1.004E-08	6.754E-11
TE-127	8.178E-05	7.459E-10	6.803E-15	6.205E-20	5.660E-25	4.708E-35	3.572E-50	2.711E-65	2.057E-80	1.423-100
TE-127M	8.349E-05	7.615E-10	6.946E-15	6.335E-20	5.778E-25	4.807E-35	3.647E-50	2.767E-65	2.100E-80	1.453-100
TE-129	1.045E-15	4.659E-32	2.077E-48	9.261E-65	4.129E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-129M	1.605E-15	7.157E-32	3.191E-48	1.423E-64	6.343E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-129	4.742E-05	4.742E-05	4.742E-05	4.742E-05	4.742E-05	4.742E-05	4.742E-05	4.742E-05	4.742E-05	4.742E-05

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
I-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-134	4.354E+01	8.116E+00	1.514E+00	2.821E-01	5.260E-02	1.828E-03	1.184E-05	7.678E-08	4.975E-10	6.011E-13
CS-135	1.314E-03	1.314E-03	1.314E-03	1.314E-03	1.314E-03	1.313E-03	1.313E-03	1.313E-03	1.313E-03	1.313E-03
CS-136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-137	1.821E+02	1.622E+02	1.445E+02	1.288E+02	1.147E+02	9.108E+01	6.442E+01	4.556E+01	3.222E+01	2.031E+01
BA-136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-137M	1.723E+02	1.535E+02	1.367E+02	1.218E+02	1.085E+02	8.616E+01	6.094E+01	4.310E+01	3.048E+01	1.921E+01
BA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-141	2.429E-14	3.072E-31	3.886E-48	4.914E-65	6.216E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-142	5.961E-08	5.961E-08	5.961E-08	5.961E-08	5.961E-08	5.961E-08	5.961E-08	5.961E-08	5.961E-08	5.961E-08
CE-144	2.238E+01	2.613E-01	3.051E-03	3.563E-05	4.160E-07	5.673E-11	9.032E-17	1.438E-22	2.290E-28	4.257E-36
PR-143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR-144	2.238E+01	2.613E-01	3.051E-03	3.563E-05	4.161E-07	5.673E-11	9.032E-17	1.438E-22	2.290E-28	4.257E-36
PR-144M	2.685E-01	3.136E-03	3.662E-05	4.276E-07	4.993E-09	6.807E-13	1.084E-18	1.726E-24	2.748E-30	5.108E-38
ND-144	3.266E-12	3.274E-12	3.274E-12	3.274E-12	3.274E-12	3.274E-12	3.274E-12	3.274E-12	3.274E-12	3.274E-12
ND-147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-145	7.672E-05	6.381E-05	5.249E-05	4.316E-05	3.549E-05	2.400E-05	1.334E-05	7.418E-06	4.125E-06	1.886E-06
PM-147	7.518E+01	2.008E+01	5.363E+00	1.432E+00	3.826E-01	2.729E-02	5.201E-04	9.910E-06	1.888E-07	9.610E-10
PM-148M	2.623E-12	1.304E-25	6.487E-39	3.226E-52	1.604E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-148	1.477E-13	7.347E-27	3.654E-40	1.817E-53	9.037E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM-145	1.629E-05	3.948E-07	9.566E-09	2.318E-10	5.617E-12	3.299E-15	4.694E-20	6.679E-25	9.504E-30	3.277E-36
SM-147	9.760E-09	1.112E-08	1.148E-08	1.157E-08	1.160E-08	1.161E-08	1.161E-08	1.161E-08	1.161E-08	1.161E-08
SM-151	6.322E-01	6.083E-01	5.853E-01	5.633E-01	5.420E-01	5.019E-01	4.471E-01	3.984E-01	3.550E-01	3.043E-01
EU-152	1.646E-02	1.276E-02	9.891E-03	7.667E-03	5.943E-03	3.572E-03	1.664E-03	7.750E-04	3.610E-04	1.303E-04
EU-154	8.198E+00	5.481E+00	3.663E+00	2.449E+00	1.637E+00	7.318E-01	2.187E-01	6.531E-02	1.951E-02	3.897E-03
EU-155	3.528E+00	1.755E+00	8.727E-01	4.341E-01	2.159E-01	5.341E-02	6.573E-03	8.089E-04	9.954E-05	6.092E-06
EU-156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD-153	7.084E-04	3.804E-06	2.043E-08	1.097E-10	5.889E-13	1.699E-17	2.629E-24	4.072E-31	6.305E-38	5.243E-47
TB-160	1.022E-08	2.577E-16	6.501E-24	1.639E-31	4.134E-39	2.630E-54	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL-206	1.413E-11	1.413E-11	1.413E-11	1.413E-11	1.413E-11	1.413E-11	1.413E-11	1.413E-11	1.413E-11	1.413E-11
TL-207	1.037E-08	1.891E-08	2.806E-08	3.771E-08	4.779E-08	6.898E-08	1.027E-07	1.379E-07	1.741E-07	2.232E-07
TL-208	7.503E-06	9.521E-06	9.883E-06	9.676E-06	9.301E-06	8.472E-06	7.335E-06	6.350E-06	5.497E-06	4.538E-06

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
PB-210	3.226E-12	4.100E-12	7.451E-12	1.514E-11	2.943E-11	8.833E-11	2.970E-10	7.267E-10	1.465E-09	3.078E-09
PB-211	1.040E-08	1.897E-08	2.813E-08	3.781E-08	4.793E-08	6.917E-08	1.030E-07	1.383E-07	1.746E-07	2.238E-07
PB-212	2.088E-05	2.650E-05	2.751E-05	2.693E-05	2.589E-05	2.358E-05	2.042E-05	1.767E-05	1.530E-05	1.263E-05
BI-211	1.040E-08	1.897E-08	2.813E-08	3.781E-08	4.793E-08	6.917E-08	1.030E-07	1.383E-07	1.746E-07	2.238E-07
BI-212	2.088E-05	2.650E-05	2.751E-05	2.693E-05	2.589E-05	2.358E-05	2.042E-05	1.767E-05	1.530E-05	1.263E-05
PO-212	1.338E-05	1.698E-05	1.762E-05	1.725E-05	1.659E-05	1.511E-05	1.308E-05	1.132E-05	9.801E-06	8.093E-06
PO-215	1.040E-08	1.897E-08	2.813E-08	3.781E-08	4.793E-08	6.917E-08	1.030E-07	1.383E-07	1.746E-07	2.238E-07
PO-216	2.088E-05	2.650E-05	2.751E-05	2.693E-05	2.589E-05	2.358E-05	2.042E-05	1.767E-05	1.530E-05	1.263E-05
RN-219	1.040E-08	1.897E-08	2.813E-08	3.781E-08	4.793E-08	6.917E-08	1.030E-07	1.383E-07	1.746E-07	2.238E-07
RN-220	2.088E-05	2.650E-05	2.751E-05	2.693E-05	2.589E-05	2.358E-05	2.042E-05	1.767E-05	1.530E-05	1.263E-05
FR-223	1.434E-10	2.615E-10	3.877E-10	5.211E-10	6.603E-10	9.535E-10	1.420E-09	1.907E-09	2.407E-09	3.086E-09
RA-223	1.040E-08	1.897E-08	2.813E-08	3.781E-08	4.793E-08	6.917E-08	1.030E-07	1.383E-07	1.746E-07	2.238E-07
RA-224	2.088E-05	2.650E-05	2.751E-05	2.693E-05	2.589E-05	2.358E-05	2.042E-05	1.767E-05	1.530E-05	1.263E-05
RA-226	4.133E-12	1.597E-11	4.084E-11	8.342E-11	1.482E-10	3.618E-10	9.513E-10	1.951E-09	3.448E-09	6.346E-09
RA-228	9.586E-10	1.212E-09	1.363E-09	1.453E-09	1.507E-09	1.559E-09	1.581E-09	1.587E-09	1.589E-09	1.590E-09
AC-227	1.039E-08	1.895E-08	2.810E-08	3.776E-08	4.785E-08	6.909E-08	1.029E-07	1.382E-07	1.744E-07	2.236E-07
TH-227	1.025E-08	1.870E-08	2.775E-08	3.729E-08	4.727E-08	6.822E-08	1.016E-07	1.364E-07	1.722E-07	2.207E-07
TH-228	2.088E-05	2.648E-05	2.748E-05	2.691E-05	2.586E-05	2.357E-05	2.041E-05	1.767E-05	1.530E-05	1.263E-05
TH-229	3.004E-09	5.147E-09	7.299E-09	9.457E-09	1.162E-08	1.598E-08	2.256E-08	2.921E-08	3.593E-08	4.499E-08
TH-230	3.216E-09	8.123E-09	1.526E-08	2.455E-08	3.591E-08	6.449E-08	1.209E-07	1.921E-07	2.762E-07	4.062E-07
TH-231	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04
TH-232	1.586E-09	1.586E-09	1.586E-09	1.587E-09	1.587E-09	1.587E-09	1.588E-09	1.589E-09	1.589E-09	1.590E-09
TH-234	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05
PA-231	6.209E-08	7.470E-08	8.731E-08	9.992E-08	1.125E-07	1.377E-07	1.755E-07	2.133E-07	2.511E-07	3.014E-07
PA-233	8.056E-04	8.063E-04	8.072E-04	8.084E-04	8.097E-04	8.128E-04	8.178E-04	8.232E-04	8.286E-04	8.358E-04
PA-234M	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05
PA-234	2.381E-08	2.381E-08	2.381E-08	2.381E-08	2.381E-08	2.381E-08	2.381E-08	2.381E-08	2.381E-08	2.381E-08
U-232	2.624E-05	2.789E-05	2.744E-05	2.640E-05	2.524E-05	2.295E-05	1.987E-05	1.720E-05	1.489E-05	1.228E-05
U-233	4.538E-06	4.556E-06	4.573E-06	4.591E-06	4.608E-06	4.644E-06	4.697E-06	4.750E-06	4.804E-06	4.876E-06
U-234	8.363E-05	1.343E-04	1.830E-04	2.298E-04	2.748E-04	3.596E-04	4.750E-04	5.774E-04	6.684E-04	7.742E-04
U-235	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04	1.193E-04
U-236	9.030E-04	9.030E-04	9.030E-04	9.030E-04	9.031E-04	9.031E-04	9.031E-04	9.031E-04	9.032E-04	9.032E-04
U-237	7.017E-06	5.517E-06	4.337E-06	3.410E-06	2.681E-06	1.657E-06	8.054E-07	3.914E-07	1.902E-07	7.269E-08
U-238	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05	1.832E-05
NP-237	8.056E-04	8.063E-04	8.072E-04	8.084E-04	8.097E-04	8.128E-04	8.178E-04	8.232E-04	8.286E-04	8.358E-04

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
PU-236	1.067E-04	3.166E-05	9.396E-06	2.789E-06	8.285E-07	7.395E-08	2.987E-09	1.132E-09	1.084E-09	1.082E-09
PU-237	3.072E-15	2.751E-27	2.465E-39	2.208E-51	1.978E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU-238	3.647E+00	3.506E+00	3.371E+00	3.240E+00	3.115E+00	2.878E+00	2.557E+00	2.272E+00	2.018E+00	1.724E+00
PU-239	9.350E-02	9.349E-02	9.347E-02	9.346E-02	9.345E-02	9.342E-02	9.338E-02	9.334E-02	9.330E-02	9.325E-02
PU-240	7.665E-02	7.674E-02	7.680E-02	7.685E-02	7.689E-02	7.692E-02	7.690E-02	7.684E-02	7.675E-02	7.661E-02
PU-241	2.860E+01	2.249E+01	1.768E+01	1.390E+01	1.093E+01	6.756E+00	3.283E+00	1.596E+00	7.754E-01	2.963E-01
PU-242	3.320E-04	3.320E-04	3.320E-04	3.320E-04	3.320E-04	3.320E-04	3.320E-04	3.320E-04	3.320E-04	3.320E-04
PU-244	4.447E-11	4.447E-11	4.447E-11	4.447E-11	4.447E-11	4.447E-11	4.447E-11	4.447E-11	4.447E-11	4.447E-11
AM-241	3.036E-01	5.040E-01	6.595E-01	7.796E-01	8.720E-01	9.960E-01	1.086E+00	1.116E+00	1.117E+00	1.097E+00
AM-242M	1.627E-03	1.590E-03	1.554E-03	1.519E-03	1.485E-03	1.419E-03	1.325E-03	1.237E-03	1.156E-03	1.055E-03
AM-242	1.618E-03	1.582E-03	1.546E-03	1.512E-03	1.477E-03	1.412E-03	1.318E-03	1.231E-03	1.150E-03	1.050E-03
AM-243	2.061E-03	2.060E-03	2.059E-03	2.058E-03	2.057E-03	2.055E-03	2.052E-03	2.049E-03	2.046E-03	2.042E-03
CM-242	5.433E-03	1.311E-03	1.279E-03	1.251E-03	1.222E-03	1.167E-03	1.090E-03	1.018E-03	9.510E-04	8.682E-04
CM-243	2.544E-03	2.253E-03	1.995E-03	1.767E-03	1.565E-03	1.227E-03	8.523E-04	5.919E-04	4.111E-04	2.528E-04
CM-244	2.728E-01	2.253E-01	1.861E-01	1.537E-01	1.270E-01	8.661E-02	4.880E-02	2.749E-02	1.549E-02	7.209E-03
CM-245	2.599E-05	2.598E-05	2.597E-05	2.596E-05	2.595E-05	2.593E-05	2.590E-05	2.586E-05	2.583E-05	2.579E-05
CM-246	3.089E-06	3.087E-06	3.085E-06	3.083E-06	3.080E-06	3.076E-06	3.069E-06	3.062E-06	3.056E-06	3.047E-06
CM-247	7.681E-12	7.681E-12	7.681E-12	7.681E-12	7.681E-12	7.681E-12	7.681E-12	7.681E-12	7.681E-12	7.681E-12
SUBTOTAL**	1.083E+03	7.579E+02	6.237E+02	5.358E+02	4.679E+02	3.641E+02	2.544E+02	1.793E+02	1.271E+02	8.087E+01
TOTAL***	1.093E+03	7.601E+02	6.245E+02	5.363E+02	4.684E+02	3.644E+02	2.546E+02	1.795E+02	1.272E+02	8.096E+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 28

Fuel-Specific Source Term Calculations Stainless Steel-Clad TRIGA 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 generic source term for stainless steel-clad TRIGA (Training, Research, and Isotope General Atomics) spent nuclear fuel elements currently stored at the INEEL. The data sources for the analysis are documented in References 1 and 2, and the INEEL calculational methodology is described in detail in Reference 3.

TRIGA Data

TRIGA reactors are light-water-cooled reactors designed for training, research, and isotope production. One type of fuel element used in a TRIGA reactor is a stainless steel-clad, uranium-zirconium-hydride (U-Zr-H) fuel element. The enriched uranium is homogenously mixed in the ZrH matrix. The cylindrical active fuel region in each stainless steel-clad element is approximately 1.4 in. in diameter and 15 in. in length. Figure 1 shows a typical stainless steel-clad fuel element with dimensions and materials. The data below give specific dimensions, materials, loadings, densities, enrichment, etc., for the stainless steel-clad element used in the burnup calculation for the source term generation.

Fuel Element:

Fuel Meat:	U-Zr-H		
	Zr:H ratio is 1.7		
	Density = 5.76 g/cc		
Clad:	Stainless Steel		
	Density = 7.92 g/cc		
Loading:	39.0	g/element U-235 BOL	
	156.0	g/element U-238 BOL	
	195.0	g/element U BOL	
	2088.0	g/element ZrH in fuel meat	
	8.5	weight % U in U-$ZrH_{1.7}$	
	20%	enrichment U-235 BOL	
	800.0	g/element stainless steel cladding (819.414 g/element with impurities)	
	450.0	g/element graphite top/bottom end reflectors	
	8.38	g/element molybdenum (single poison disc)	

Active Fuel Length: 15 in.
Fuel Element Length: 29 in. (approximate)

Water Temperature: 77.5°F
Water Pressure: 14.7 psia

From the above data (materials, enrichments, and densities), material masses and number densities were calculated for all the material components in a single stainless steel-clad TRIGA fuel element. In addition, for the ORIGEN2 (Reference 4) depletion or burnup calculation, conservative and

detailed impurity concentrations were added for the stainless steel-clad, zirconium-hydride, and graphite end reflector masses. Table 1 gives the impurity concentrations for these three materials.

Burnup

Reference 1 is a parametric study and includes radionuclide inventories or source terms for eight different burnups ranging up to 17.95%. The burnup chosen for this template is based on the 17.95% burnup of the initial U-235 or the maximum burnup used in the parametric study. This burnup is equivalent to 6.65 MWd, 34,103 MWd/MTU, and 8.08 g U-235 depleted per element and represents the upper end of typical stainless steel-clad TRIGA fuel element burnups. The assumption of maximum burnup is conservative for the buildup of fission products, activation products, and minor actinides in the source term and nonconservative with regard to criticality safety.

Cross-Section Development

An MCNP4A (Reference 5) partial core model of a MARK I TRIGA reactor core was used to generate neutron cross sections specifically for the stainless steel-clad TRIGA fuel element. The MCNP4A one-twelfth core model is shown in Figure 2. The cross sections are spectrally and spatially weighted over all the fuel elements shown in Figure 2. These cross sections are in turn used in the fuel element ORIGEN2 depletion calculation.

Parametric TRIGA Single Element Exposure History

Table 2 summarizes the single element exposure history of the stainless steel-clad TRIGA fuel element from Reference 1. The burnup period is a hypothetical 4-year continuous exposure. TRIGA fuel elements typically remain in the core for much longer periods of time relative to the assumed 4-year in-core residency. Therefore, typical fuel elements would have more time to decay away their source term; therefore, the 4-year assumption is expected to produce a conservative source term.

Burnup Calculation

The ORIGEN2 computer code was used to perform the depletion or burnup calculation for the stainless steel-clad TRIGA fuel element. The radionuclide inventory or source term template that follows is for a single stainless steel-clad TRIGA fuel element. The fuel element component masses and impurities (fuel meat, uranium, clad, burnable poison, end fixtures), neutron cross sections, burnup, and hypothetical power history and power level as discussed above are input data for the ORIGEN2 calculation. The radionuclide concentrations are given as a function of decay 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, *Radionuclide Mass Inventory, Activity, Decay Heat, and Dose Rate Parametric Data for TRIGA Spent Nuclear Fuels*, INEL-96/0482, Idaho National Engineering Laboratory, March 1997.

2. N. Tomsio, *Characterization of TRIGA Fuel*, ORNL/Sub//86-22047/3, GA-C18542, GA Technologies, October 1986.

3. 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.

4. 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.

5. "MCNP4A: Monte Carlo N-Particle Transport Code System," LA-12625M, contributed by Los Alamos National Laboratory, Los Alamos, New Mexico, 1994, and distributed as package CCC-200 by Oak Ridge National Laboratory.

Table 1. Material constituent and impurity concentrations for the various materials in a stainless steel-clad TRIGA fuel element.

Constituent or Impurity	Graphite Concentration (ppm)	ZrH Concentration (wt%)	Stainless Steel Concentration (ppm)
H		1.8439	
Li	0.45		0.13
Be	0.005		
B	2.5	0.00005	
C	100 wt%	0.026968	0.08 wt%
N		0.00799	525
O		0.094887	
Na	10.4		37
Mg	1		
Al	4.1	0.007491	200
Si	26	0.011986	1.00 wt%
P	1	0.009988	
S	9.4	0.003496	
Cl	3		130
K	3		3
Ca	22.5		19
Sc	0.01		0.03
Ti	16	0.004994	600
V	18.9	0.004994	690
Cr	1	0.124851	18.40 wt%
Mn	1	0.004994	1.53 wt%
Fe	11.1	0.224731	68.99 wt%
Co	4	0.001998	2570
Ni	4.6	0.006992	10.00 wt%
Cu	0.47	0.004994	8150
Zn	1	0.009988	2230
Ga			450
As			1010
Se			70
Br			8
Rb	1		10
Sr	0.47		0.2
Y			5
Zr	0.5	98.1560	20
Nb	1.74	0.006992	300
Mo	1	0.004994	5500
Ag	0.5		2

Table 1. (continued).

Constituent or Impurity	Graphite Concentration (ppm)	ZrH Concentration (wt%)	Stainless Steel Concentration (ppm)
Cd	0.5	0.000050	
In	1		
Sn	1	1.598089	
Sb	1		17
Cs	1		0.3
Ba	2.9		500
La	1.38		2.1
Ce	0.56		550
Pr	0.64		
Nd	0.36		
Sm	0.61	0.000999	0.15
Eu			0.02
Gd	0.08	0.000499	
Tb	0.26		0.71
Dy	0.16		1
Ho	0.08		1
Er	0.04		
Tm	0.04		
Yb	0.06		2
Lu	0.02		0.8
Hf	0.17	0.003496	2
Ta	0.35	0.019976	
W	25.5	0.009988	520
Tl	1		
Pb	6.9	0.009988	139
Bi	1		
Th		0.000699	1
U		0.000350	2

Table 2. Hypothetical power history for a maximum burnup stainless steel-clad TRIGA fuel element

Duration (days)	Cumulative Duration (days)	Time-Averaged Power (MWth)
365	365	0.004555
365	730	0.004555
365	1095	0.004555
365	1460	0.004555
1825	3285	0.0
1825	5110	0.0
1825	6935	0.0
1825	8760	0.0
1825	10585	0.0
3650	14235	0.0
5475	19710	0.0
5475	25185	0.0
5475	30660	0.0
7300	37960	0.0

The ten decay times following the hypothetical 4-year exposure are for 5, 10, 15, 20, 25, 35, 50, 65, 80, and 100-year cooling time periods designated for the template methodology.

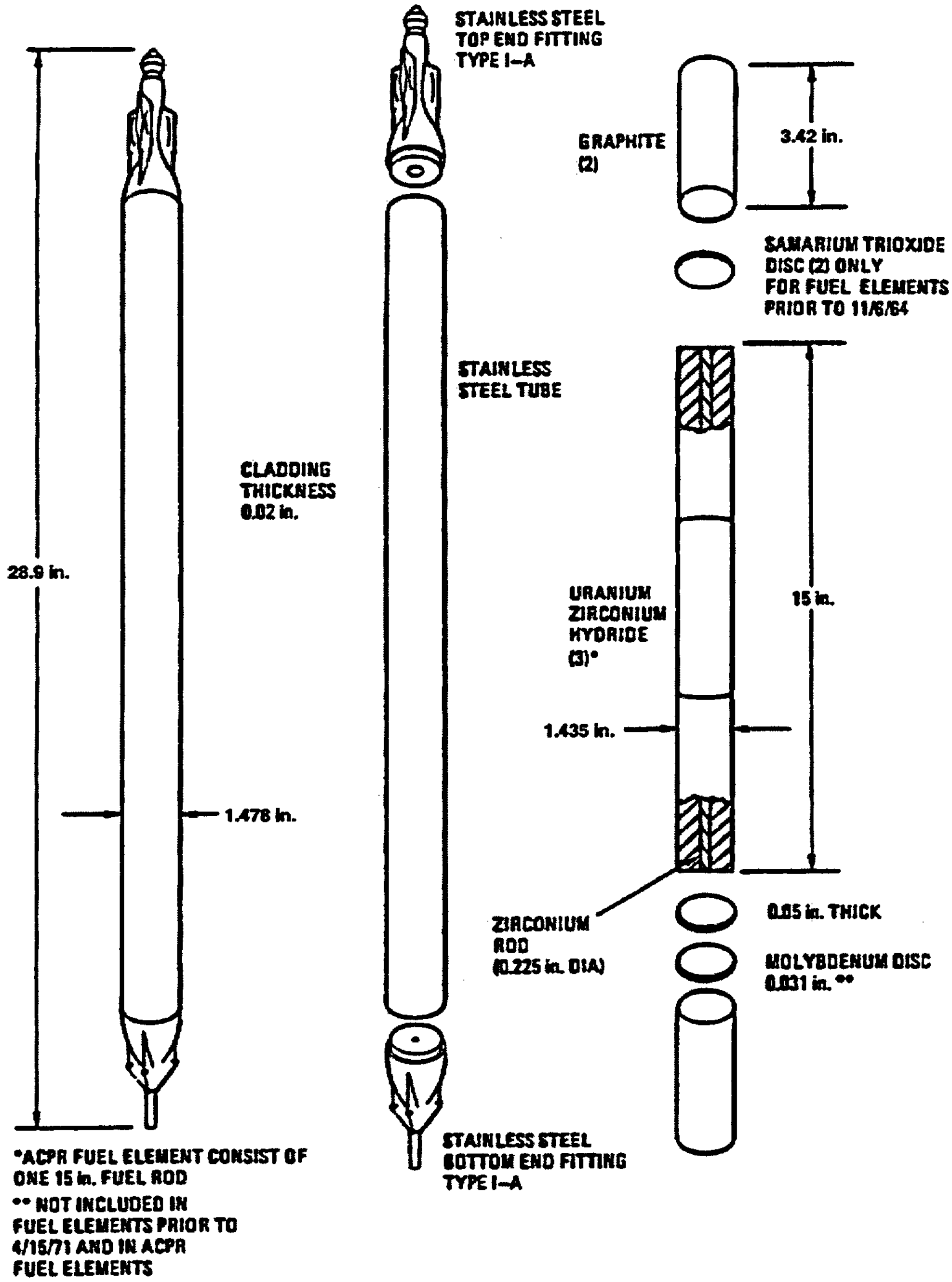

Figure 1. A typical stainless steel-clad Mark I TRIGA fuel element.

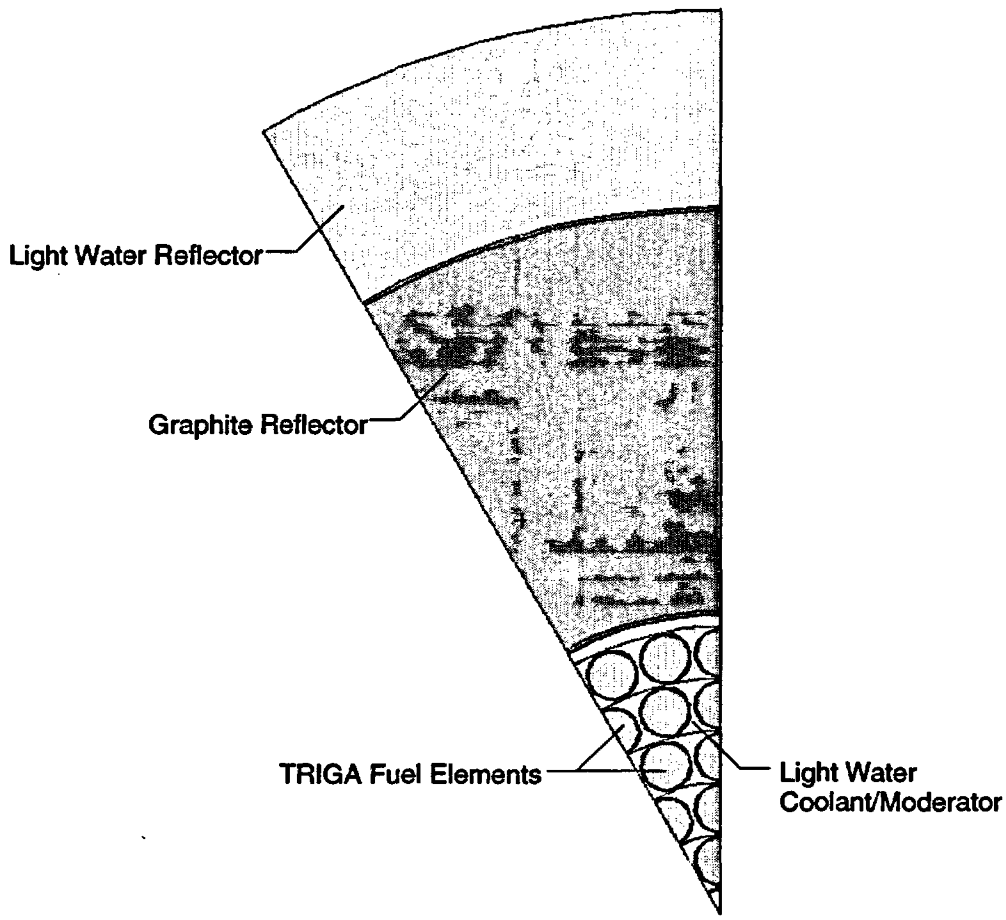

Figure 2. MCNP partial core model of a Mark I TRIGA reactor core.

TRIGA Element

Stainless Steel Cladding, 10 to 20% Enriched U-235 Fuel

Reactor Moderator/Coolant:	Light Water
Fuel Meat:	U-Zr-$H_{1.7}$
Clad:	Stainless Steel
Burnup:	6.65 MWd/element (maximum element burnup)
Burnup:	17.95% U-235 burnup (amount fissioned)
Burnup:	8.08 g U-235 depletion (amount fissioned and transmuted)
Basis of Calculation:	Single element
BOL U-235:	39.0 grams U-235 per element (design basis)
BOL U-238:	156.0 grams U-238 per element
BOL Total U per element:	195.0 grams U per element
BOL Fuel Enrichment:	20.0 wt%

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H-3	7.389E-02	5.582E-02	4.217E-02	3.185E-02	2.406E-02	1.373E-02	5.921E-03	2.552E-03	1.100E-03	3.583E-04
BE-10	1.334E-07	1.334E-07	1.334E-07	1.334E-07	1.334E-07	1.334E-07	1.334E-07	1.334E-07	1.334E-07	1.334E-07
C-14	8.559E-04	8.554E-04	8.548E-04	8.543E-04	8.538E-04	8.528E-04	8.512E-04	8.497E-04	8.481E-04	8.461E-04
CL-36	1.870E-05	1.870E-05	1.870E-05	1.870E-05	1.870E-05	1.870E-05	1.870E-05	1.870E-05	1.870E-05	1.870E-05
CR-51	6.851E-19	1.020E-38	1.518E-58	2.260E-78	3.364E-98	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN-54	4.834E-02	8.439E-04	1.473E-05	2.572E-07	4.490E-09	1.368E-12	7.281E-18	3.874E-23	2.061E-28	1.914E-35
FE-55	5.131E+00	1.354E+00	3.574E-01	9.433E-02	2.490E-02	1.734E-03	3.189E-05	5.863E-07	1.078E-08	5.231E-11
FE-59	5.296E-13	3.277E-25	2.028E-37	1.255E-49	7.766E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO-60	8.538E+00	4.425E+00	2.293E+00	1.189E+00	6.161E-01	1.655E-01	2.304E-02	3.208E-03	4.466E-04	3.223E-05
NI-59	3.612E-03	3.612E-03	3.611E-03	3.611E-03	3.611E-03	3.611E-03	3.610E-03	3.610E-03	3.609E-03	3.609E-03
NI-63	4.280E-01	4.122E-01	3.970E-01	3.823E-01	3.682E-01	3.415E-01	3.050E-01	2.724E-01	2.433E-01	2.093E-01
ZN-65	1.938E-03	1.083E-05	6.050E-08	3.380E-10	1.889E-12	5.896E-17	1.028E-23	1.794E-30	3.129E-37	3.049E-46
SE-79	8.657E-05	8.656E-05	8.656E-05	8.655E-05	8.655E-05	8.654E-05	8.653E-05	8.651E-05	8.650E-05	8.648E-05
KR-85	1.680E+00	1.216E+00	8.802E-01	6.372E-01	4.613E-01	2.417E-01	9.171E-02	3.479E-02	1.320E-02	3.625E-03
RB-87	5.922E-09	5.922E-09	5.922E-09	5.922E-09	5.922E-09	5.922E-09	5.922E-09	5.922E-09	5.922E-09	5.922E-09
SR-89	2.308E-09	3.048E-20	4.025E-31	5.314E-42	7.016E-53	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR-90	1.732E+01	1.538E+01	1.366E+01	1.212E+01	1.077E+01	8.487E+00	5.940E+00	4.157E+00	2.910E+00	1.808E+00
Y-90	1.733E+01	1.538E+01	1.366E+01	1.213E+01	1.077E+01	8.489E+00	5.941E+00	4.159E+00	2.911E+00	1.809E+00
Y-91	8.787E-08	3.580E-17	1.458E-26	5.942E-36	2.421E-45	4.017E-64	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR-93	5.294E-04	5.294E-04	5.294E-04	5.294E-04	5.293E-04	5.293E-04	5.293E-04	5.293E-04	5.293E-04	5.292E-04
ZR-95	6.666E-07	1.727E-15	4.475E-24	1.159E-32	3.004E-41	2.017E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-93M	1.502E-04	2.295E-04	2.910E-04	3.386E-04	3.756E-04	4.263E-04	4.672E-04	4.862E-04	4.951E-04	5.001E-04

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

Radionuclide	5	10	15	20	25	35	50	65	80	100
NB-94	5.404E-05	5.403E-05	5.402E-05	5.401E-05	5.400E-05	5.398E-05	5.395E-05	5.393E-05	5.390E-05	5.386E-05
NB-95	1.479E-06	3.834E-15	9.934E-24	2.574E-32	6.669E-41	4.477E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB-95M	4.946E-09	1.281E-17	3.319E-26	8.601E-35	2.228E-43	1.496E-60	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO-93	6.219E-05	6.213E-05	6.207E-05	6.201E-05	6.194E-05	6.182E-05	6.164E-05	6.146E-05	6.127E-05	6.103E-05
TC-99	2.942E-03	2.942E-03	2.942E-03	2.942E-03	2.942E-03	2.942E-03	2.942E-03	2.942E-03	2.942E-03	2.941E-03
RU-103	1.277E-12	1.321E-26	1.366E-40	1.413E-54	1.461E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU-106	6.097E-01	1.963E-02	6.320E-04	2.035E-05	6.552E-07	6.792E-10	2.267E-14	7.566E-19	2.525E-23	2.714E-29
RH-103M	1.152E-12	1.191E-26	1.232E-40	1.274E-54	1.317E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH-106	6.097E-01	1.963E-02	6.320E-04	2.035E-05	6.552E-07	6.792E-10	2.267E-14	7.566E-19	2.525E-23	2.714E-29
PD-107	4.165E-06	4.165E-06	4.165E-06	4.165E-06	4.165E-06	4.165E-06	4.165E-06	4.165E-06	4.165E-06	4.165E-06
AG-110	1.079E-06	6.829E-09	4.323E-11	2.737E-13	1.733E-15	6.944E-20	1.763E-26	4.471E-33	1.135E-39	1.822E-48
AG-110M	8.110E-05	5.134E-07	3.250E-09	2.057E-11	1.303E-13	5.221E-18	1.325E-24	3.362E-31	8.530E-38	1.370E-46
AG-111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-113M	2.149E-03	1.695E-03	1.337E-03	1.054E-03	8.314E-04	5.172E-04	2.537E-04	1.245E-04	6.106E-05	2.362E-05
CD-113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CD-115M	3.028E-14	1.451E-26	6.958E-39	3.335E-51	1.599E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114	1.220E-14	9.789E-26	7.858E-37	6.307E-48	5.062E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-114M	1.274E-14	1.023E-25	8.211E-37	6.590E-48	5.290E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN-115M	2.126E-18	1.019E-30	4.885E-43	2.342E-55	1.123E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-119M	1.626E-02	9.310E-05	5.330E-07	3.052E-09	1.747E-11	5.730E-16	1.076E-22	2.019E-29	3.791E-36	4.075E-45
SN-121M	3.857E-04	3.598E-04	3.358E-04	3.133E-04	2.923E-04	2.545E-04	2.067E-04	1.680E-04	1.364E-04	1.034E-04
SN-123	2.044E-05	1.141E-09	6.368E-14	3.555E-18	1.984E-22	6.180E-31	1.074E-43	1.868E-56	3.248E-69	3.152E-86
SN-125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN-126	8.091E-05	8.091E-05	8.090E-05	8.090E-05	8.090E-05	8.089E-05	8.088E-05	8.088E-05	8.087E-05	8.086E-05
SB-124	3.777E-11	2.823E-20	2.110E-29	1.577E-38	1.179E-47	6.588E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB-125	5.772E-01	1.653E-01	4.735E-02	1.356E-02	3.884E-03	3.186E-04	7.483E-06	1.758E-07	4.129E-09	2.778E-11
SB-126	1.133E-05	1.133E-05	1.133E-05	1.133E-05	1.133E-05	1.132E-05	1.132E-05	1.132E-05	1.132E-05	1.132E-05
SB-126M	8.091E-05	8.091E-05	8.090E-05	8.090E-05	8.090E-05	8.089E-05	8.088E-05	8.088E-05	8.087E-05	8.086E-05
TE-123M	3.346E-09	8.588E-14	2.204E-18	5.659E-23	1.453E-27	9.574E-37	1.620E-50	2.739E-64	4.635E-78	2.013E-96
TE-125M	1.408E-01	4.033E-02	1.155E-02	3.309E-03	9.475E-04	7.772E-05	1.826E-06	4.288E-08	1.008E-09	6.778E-12
TE-127	8.388E-06	7.650E-11	6.978E-16	6.364E-21	5.805E-26	4.829E-36	3.664E-51	2.780E-66	2.109E-81	1.460-101
TE-127M	8.563E-06	7.811E-11	7.124E-16	6.498E-21	5.926E-26	4.930E-36	3.741E-51	2.838E-66	2.153E-81	1.490-101
TE-129	1.157E-16	5.160E-33	2.301E-49	1.026E-65	4.573E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-129M	1.778E-16	7.927E-33	3.534E-49	1.576E-65	7.025E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-129	4.900E-06	4.900E-06	4.900E-06	4.900E-06	4.900E-06	4.900E-06	4.900E-06	4.900E-06	4.900E-06	4.900E-06

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
I-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
XE-133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-134	6.021E-01	1.123E-01	2.093E-02	3.903E-03	7.275E-04	2.529E-05	1.638E-07	1.062E-09	6.882E-12	8.315E-15
CS-135	2.141E-04	2.141E-04	2.141E-04	2.141E-04	2.141E-04	2.141E-04	2.141E-04	2.141E-04	2.141E-04	2.141E-04
CS-136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CS-137	1.833E+01	1.633E+01	1.455E+01	1.296E+01	1.155E+01	9.169E+00	6.485E+00	4.587E+00	3.244E+00	2.044E+00
BA-136M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-137M	1.734E+01	1.545E+01	1.377E+01	1.226E+01	1.093E+01	8.674E+00	6.135E+00	4.339E+00	3.069E+00	1.934E+00
BA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-141	2.771E-15	3.504E-32	4.432E-49	5.606E-66	7.090E-83	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-142	7.084E-09	7.084E-09	7.084E-09	7.084E-09	7.084E-09	7.084E-09	7.084E-09	7.084E-09	7.084E-09	7.084E-09
CE-144	2.308E+00	2.695E-02	3.147E-04	3.675E-06	4.291E-08	5.850E-12	9.315E-18	1.483E-23	2.361E-29	4.390E-37
PR-143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PR-144	2.308E+00	2.695E-02	3.147E-04	3.675E-06	4.291E-08	5.851E-12	9.315E-18	1.483E-23	2.361E-29	4.390E-37
PR-144M	2.770E-02	3.234E-04	3.776E-06	4.410E-08	5.149E-10	7.021E-14	1.118E-19	1.780E-25	2.834E-31	5.268E-39
ND-144	2.861E-13	2.870E-13	2.870E-13	2.870E-13	2.870E-13	2.870E-13	2.870E-13	2.870E-13	2.870E-13	2.870E-13
ND-147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-145	8.156E-06	6.776E-06	5.574E-06	4.583E-06	3.769E-06	2.548E-06	1.417E-06	7.878E-07	4.380E-07	2.003E-07
PM-147	1.398E+01	3.735E+00	9.976E-01	2.665E-01	7.117E-02	5.077E-03	9.674E-05	1.843E-06	3.512E-08	1.788E-10
PM-148M	8.755E-14	4.354E-27	2.165E-40	1.077E-53	5.356E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM-148	4.931E-15	2.453E-28	1.220E-41	6.066E-55	3.017E-68	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM-145	1.561E-06	3.782E-08	9.165E-10	2.221E-11	5.382E-13	3.160E-16	4.497E-21	6.399E-26	9.105E-31	3.140E-37
SM-147	1.801E-09	2.052E-09	2.119E-09	2.137E-09	2.142E-09	2.143E-09	2.144E-09	2.144E-09	2.144E-09	2.144E-09
SM-151	1.517E-01	1.460E-01	1.404E-01	1.352E-01	1.300E-01	1.204E-01	1.073E-01	9.560E-02	8.518E-02	7.302E-02
EU-152	9.765E-03	7.570E-03	5.869E-03	4.549E-03	3.526E-03	2.119E-03	9.871E-04	4.598E-04	2.142E-04	7.735E-05
EU-154	1.022E-01	6.828E-02	4.565E-02	3.052E-02	2.040E-02	9.118E-03	2.723E-03	8.138E-04	2.431E-04	4.856E-05
EU-155	1.948E-01	9.689E-02	4.820E-02	2.397E-02	1.192E-02	2.950E-03	3.630E-04	4.466E-05	5.496E-06	3.364E-07
EU-156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
GD-153	3.421E-05	1.837E-07	9.866E-10	5.298E-12	2.845E-14	8.203E-19	1.271E-25	1.967E-32	3.045E-39	2.532E-48
TB-160	4.416E-10	1.114E-17	2.809E-25	7.082E-33	1.786E-40	1.136E-55	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL-206	1.436E-12	1.436E-12	1.436E-12	1.436E-12	1.436E-12	1.436E-12	1.436E-12	1.436E-12	1.436E-12	1.436E-12
TL-207	2.810E-09	5.653E-09	9.117E-09	1.311E-08	1.756E-08	2.753E-08	4.450E-08	6.306E-08	8.260E-08	1.095E-07
TL-208	1.054E-07	1.127E-07	1.098E-07	1.052E-07	1.004E-07	9.126E-08	7.908E-08	6.853E-08	5.941E-08	4.915E-08

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

Radionuclide	5	10	15	20	25	35	50	65	80	100
PB-210	1.703E-14	4.913E-14	1.142E-13	2.230E-13	3.857E-13	9.129E-13	2.350E-12	4.772E-12	8.392E-12	1.541E-11
PB-211	2.818E-09	5.669E-09	9.143E-09	1.315E-08	1.760E-08	2.760E-08	4.462E-08	6.324E-08	8.283E-08	1.098E-07
PB-212	2.932E-07	3.135E-07	3.056E-07	2.928E-07	2.795E-07	2.540E-07	2.201E-07	1.907E-07	1.653E-07	1.368E-07
BI-211	2.818E-09	5.669E-09	9.143E-09	1.315E-08	1.760E-08	2.760E-08	4.462E-08	6.324E-08	8.283E-08	1.098E-07
BI-212	2.932E-07	3.135E-07	3.056E-07	2.928E-07	2.795E-07	2.540E-07	2.201E-07	1.907E-07	1.653E-07	1.368E-07
PO-212	1.879E-07	2.009E-07	1.958E-07	1.876E-07	1.791E-07	1.627E-07	1.410E-07	1.222E-07	1.059E-07	8.765E-08
PO-215	2.818E-09	5.669E-09	9.143E-09	1.315E-08	1.760E-08	2.760E-08	4.462E-08	6.324E-08	8.283E-08	1.098E-07
PO-216	2.932E-07	3.135E-07	3.056E-07	2.928E-07	2.795E-07	2.540E-07	2.201E-07	1.907E-07	1.653E-07	1.368E-07
RN-219	2.818E-09	5.669E-09	9.143E-09	1.315E-08	1.760E-08	2.760E-08	4.462E-08	6.324E-08	8.283E-08	1.098E-07
RN-220	2.932E-07	3.135E-07	3.056E-07	2.928E-07	2.795E-07	2.540E-07	2.201E-07	1.907E-07	1.653E-07	1.368E-07
FR-223	3.887E-11	7.816E-11	1.260E-10	1.812E-10	2.426E-10	3.805E-10	6.152E-10	8.718E-10	1.142E-09	1.514E-09
RA-223	2.818E-09	5.669E-09	9.143E-09	1.315E-08	1.760E-08	2.760E-08	4.462E-08	6.324E-08	8.283E-08	1.098E-07
RA-224	2.932E-07	3.135E-07	3.056E-07	2.928E-07	2.795E-07	2.540E-07	2.201E-07	1.907E-07	1.653E-07	1.368E-07
RA-226	1.411E-13	3.510E-13	6.635E-13	1.087E-12	1.631E-12	3.109E-12	6.416E-12	1.119E-11	1.760E-11	2.892E-11
RA-228	1.018E-09	1.286E-09	1.445E-09	1.541E-09	1.597E-09	1.651E-09	1.675E-09	1.680E-09	1.681E-09	1.681E-09
AC-227	2.817E-09	5.664E-09	9.131E-09	1.313E-08	1.758E-08	2.757E-08	4.458E-08	6.317E-08	8.274E-08	1.097E-07
TH-227	2.779E-09	5.591E-09	9.017E-09	1.297E-08	1.736E-08	2.722E-08	4.401E-08	6.236E-08	8.169E-08	1.083E-07
TH-228	2.931E-07	3.133E-07	3.053E-07	2.925E-07	2.793E-07	2.539E-07	2.201E-07	1.907E-07	1.653E-07	1.368E-07
TH-229	5.293E-10	9.119E-10	1.294E-09	1.677E-09	2.059E-09	2.823E-09	3.969E-09	5.114E-09	6.258E-09	7.781E-09
TH-230	7.486E-11	1.203E-10	1.700E-10	2.238E-10	2.816E-10	4.083E-10	6.243E-10	8.684E-10	1.137E-09	1.530E-09
TH-231	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.687E-05	6.687E-05
TH-232	1.681E-09	1.681E-09	1.681E-09	1.681E-09	1.681E-09	1.681E-09	1.681E-09	1.681E-09	1.681E-09	1.682E-09
TH-234	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05
PA-231	1.854E-08	2.561E-08	3.268E-08	3.975E-08	4.682E-08	6.095E-08	8.214E-08	1.033E-07	1.245E-07	1.527E-07
PA-233	8.248E-06	8.264E-06	8.287E-06	8.315E-06	8.347E-06	8.420E-06	8.541E-06	8.670E-06	8.799E-06	8.971E-06
PA-234M	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05
PA-234	6.785E-08	6.785E-08	6.785E-08	6.785E-08	6.785E-08	6.785E-08	6.785E-08	6.785E-08	6.785E-08	6.785E-08
U-232	3.241E-07	3.113E-07	2.974E-07	2.836E-07	2.704E-07	2.456E-07	2.126E-07	1.840E-07	1.593E-07	1.314E-07
U-233	8.114E-07	8.115E-07	8.117E-07	8.119E-07	8.120E-07	8.124E-07	8.129E-07	8.134E-07	8.139E-07	8.146E-07
U-234	9.621E-07	1.059E-06	1.152E-06	1.241E-06	1.327E-06	1.489E-06	1.710E-06	1.906E-06	2.080E-06	2.284E-06
U-235	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.686E-05	6.687E-05	6.687E-05
U-236	8.440E-05	8.441E-05	8.441E-05	8.441E-05	8.441E-05	8.442E-05	8.442E-05	8.443E-05	8.443E-05	8.444E-05
U-237	1.663E-07	1.308E-07	1.028E-07	8.083E-08	6.355E-08	3.928E-08	1.909E-08	9.278E-09	4.509E-09	1.723E-09
U-238	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05	5.219E-05
NP-237	8.248E-06	8.264E-06	8.287E-06	8.315E-06	8.347E-06	8.420E-06	8.541E-06	8.670E-06	8.799E-06	8.971E-06

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

Radionuclide	5	10	15	20	25	35	50	65	80	100
PU-236	9.008E-08	2.673E-08	7.935E-09	2.356E-09	7.002E-10	6.302E-11	3.091E-12	1.524E-12	1.483E-12	1.482E-12
PU-237	9.582E-19	8.583E-31	7.689E-43	6.887E-55	6.170E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU-238	6.905E-03	6.638E-03	6.382E-03	6.135E-03	5.898E-03	5.451E-03	4.843E-03	4.303E-03	3.823E-03	3.265E-03
PU-239	3.677E-02	3.677E-02	3.676E-02	3.676E-02	3.675E-02	3.674E-02	3.672E-02	3.671E-02	3.669E-02	3.667E-02
PU-240	1.415E-02	1.414E-02	1.413E-02	1.413E-02	1.412E-02	1.410E-02	1.408E-02	1.406E-02	1.404E-02	1.401E-02
PU-241	6.780E-01	5.331E-01	4.191E-01	3.295E-01	2.591E-01	1.601E-01	7.782E-02	3.782E-02	1.838E-02	7.023E-03
PU-242	1.538E-06	1.538E-06	1.538E-06	1.538E-06	1.538E-06	1.538E-06	1.538E-06	1.538E-06	1.538E-06	1.538E-06
PU-244	2.551E-15	2.551E-15	2.551E-15	2.551E-15	2.551E-15	2.551E-15	2.551E-15	2.551E-15	2.551E-15	2.551E-15
AM-241	7.438E-03	1.219E-02	1.587E-02	1.872E-02	2.090E-02	2.384E-02	2.598E-02	2.668E-02	2.668E-02	2.621E-02
AM-242M	9.396E-06	9.185E-06	8.978E-06	8.775E-06	8.578E-06	8.196E-06	7.654E-06	7.148E-06	6.676E-06	6.095E-06
AM-242	9.349E-06	9.139E-06	8.933E-06	8.732E-06	8.535E-06	8.155E-06	7.616E-06	7.113E-06	6.643E-06	6.064E-06
AM-243	9.825E-07	9.820E-07	9.816E-07	9.811E-07	9.807E-07	9.797E-07	9.784E-07	9.770E-07	9.756E-07	9.738E-07
CM-242	2.435E-05	7.568E-06	7.391E-06	7.224E-06	7.062E-06	6.744E-06	6.298E-06	5.882E-06	5.494E-06	5.015E-06
CM-243	1.193E-06	1.057E-06	9.360E-07	8.289E-07	7.340E-07	5.756E-07	3.998E-07	2.777E-07	1.928E-07	1.186E-07
CM-244	1.128E-05	9.315E-06	7.694E-06	6.355E-06	5.248E-06	3.580E-06	2.017E-06	1.137E-06	6.404E-07	2.980E-07
CM-245	1.101E-10	1.101E-10	1.100E-10	1.100E-10	1.099E-10	1.098E-10	1.097E-10	1.096E-10	1.094E-10	1.093E-10
CM-246	3.142E-12	3.140E-12	3.138E-12	3.135E-12	3.133E-12	3.129E-12	3.122E-12	3.115E-12	3.108E-12	3.099E-12
CM-247	6.602E-19	6.602E-19	6.602E-19	6.602E-19	6.602E-19	6.602E-19	6.602E-19	6.602E-19	6.602E-19	6.602E-19
SUBTOTAL**	1.086E+02	7.508E+01	6.143E+01	5.270E+01	4.610E+01	3.597E+01	2.521E+01	1.778E+01	1.259E+01	7.978E+00
TOTAL***	1.086E+02	7.509E+01	6.143E+01	5.271E+01	4.610E+01	3.597E+01	2.521E+01	1.778E+01	1.259E+01	7.979E+00

* 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.

Fuel-Specific Source Term Calculations
Hypothetical 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 hypothetical spent nuclear fuel element. This single-element source term is intended to be a bounding source term for spent nuclear fuel elements with no specific fuel category and those fuels currently without a developed template. This fuel is a hypothetical construction of reactor materials and heavy metal constituents designed specifically with the intention of maximizing activation products, actinides, and fission products. A description of the construction data is given below, and the INEEL calculational methodology is described in detail in Reference 1.

Hypothetical Fuel Data

The hypothetical fuel was chosen to be a ternary oxide fuel composed of urania (UO_2), plutonia (PuO_2), and thoria (ThO_2). The urania, plutonia, and thoria oxide volume percentages are 40%, 40%, and 20%, respectively. The uranium metal is assumed to be 50% enriched in U-235, the plutonium metal is reactor grade, and the thorium is 100% Th-232. This heavy metal oxide composition was designed specifically to maximize higher order actinide concentrations in the spent fuel.

The fuel pellet and inner clad are based on the dimensions of a Pressurized Water Reactor (PWR) Westinghouse 17 × 17 fuel assembly fuel rod. The ternary fuel rod here is assumed to have an inner and an outer clad. The inner clad is assumed to be Stainless Steel-304, and the outer clad is assumed to be Inconel X-750. The dual clads are intended to maximize the structural material mass and, therefore, the activation products. The dimensions for the fuel pellet and clads are given below.

In order to further increase the concentration and spectrum of activation products, cylindrical graphite reflectors are assumed to be located at the top and bottom of the active fuel.

Although the dimensions of the fuel element are similar to a PWR element, the neutron cross sections selected for the burnup calculation are based on a high-temperature graphite reactor in order to maximize the production of activation and transmutation products. Of course, a high-temperature reactor would not have metallic clad fuel rods. So again, it should be remembered that this fuel and reactor are strictly hypothetical.

The fuel element described above is a hypothetical construct intended to produce a maximum or bounding source term. The following data provide the specific fuel element dimensions, materials, densities, enrichment, etc., of the hypothetical fuel element described above.

Fuel Element:	Cylindrical rod
Length:	144.0 in.
Fuel Pellet:	Oxide Ceramic UO_2-PuO_2-ThO_2 matrix
Fuel Pellet Radius:	0.4095 cm

Uranium Enrichment:	4.0	wt % U-234
	50.0	wt % U-235
	6.0	wt % U-236
	40.0	wt % U-238
Plutonium Enrichment:	1.8	wt % Pu-238
	60.0	wt % Pu-239
	20.9	wt % Pu-240
	11.9	wt % Pu-241
	5.4	wt % Pu-242
Thorium Enrichment:	100.0	wt % Th-232
Heavy Metal Loading:	29.7592	g/element U-234 (BOL)
	371.9897	g/element U-235 (BOL)
	44.6388	g/element U-236 (BOL)
	297.5918	g/element U-238 (BOL)
	743.9795	g/element TOTAL U (BOL)
	14.0261	g/element Pu-238 (BOL)
	467.5381	g/element Pu-239 (BOL)
	162.8591	g/element Pu-240 (BOL)
	92.7284	g/element Pu-241 (BOL)
	42.0784	g/element Pu-242 (BOL)
	779.2301	g/element TOTAL Pu (BOL)
	345.4439	g/element Th-232 (BOL)
	1,868.6536	g/element total heavy metal (BOL)
	1.86865E-3	Total MTIHM/element (BOL)

Inner Clad Material:	SS-304
Inner Clad Density:	8.02 g/cc
Inner Clad Mass:	486.03 g
Inner Clad Radius:	0.4695 cm
Outer Clad Material:	Inconel X-750
Outer Clad Density:	8.3 g/cc
Outer Clad Mass:	571.66 g
Outer Clad Radius:	0.5295 cm
End Reflector Material:	H451 Graphite
End Reflector Mass:	450.0 g

From the above data (materials, enrichments, and densities), material masses 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 UO_2-PuO_2-ThO_2, SS-304, Inconel X-750, and H-451 fuel element materials. Impurities for the UO_2-PuO_2-ThO_2 fuel composition are based on the available UO_2 uranium metal impurities, and it is assumed to be the same for the plutonium and thorium metal as well. Table 1 lists the impurities and their concentrations for the mentioned materials.

Burnup

The burnup chosen for this template is 62.5 MWd or 33,447 MWd/MTIHM for a single hypothetical fuel element.

Cross-Section Development

The neutron cross sections used in the fuel burnup calculation and the source term generation are based on a standard High Temperature Gas Reactor (HTGR) cross-section library. This library comes with the ORIGEN2 computer code package. The selection of the HTGR cross section was based on a parametric study involving several cross-section libraries representing different reactor types (energy spectral characteristics). Included in the study were libraries for a fast reactor, heavy water reactor (CANDU), breeder water reactor, pressurized water reactor, Advanced Water Reactor, and the HTGR. The HTGR neutron cross sections produced the highest concentrations of higher-order actinides (plutonium, americium, and curium). Because of the hypothetical nature of the reactor, no attempt was made to update the HTGR cross sections as a function of burnup.

Single Element Exposure History

Table 2 summarizes the assumed 3-year constant 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 was used to perform the depletion or burnup calculation for a single fuel element. The fuel element masses and impurities (uranium, plutonium, thorium, steel, inconel, and graphite), 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 99.8% 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.

Table 1. Hypothetical fuel element material impurity concentrations.

Constituent or Impurity	Uranium Metal Concentration (ppm)	SS-304 Concentration (wt%)	Inconel X-750 Concentration (ppm)	Graphite Concentration (ppm)
H		0.0007		
Li	1	0.13 ppm		0.45
Be				0.005
B	1	0.0005		2.5
C	89.4	0.07	800	100 wt%
N	25	0.047	400	
O		0.015		
F	10.7			
Na	15	37 ppm		10.4
Mg	2			1
Al	16.7	0.01	10000	4.1
Si	12.1	0.6	5000	26
P	35	0.0375	400	1
S		0.02	100	9.4
Cl	5.3	130 ppm		3
K		3 ppm		3
Ca	2	19 ppm		22.5
Sc		0.03 ppm		0.01
Ti	1	0.05	27500	16
V	3	0.05		18.9
Cr	4	18.8	170000	1
Mn	1.7	1.41	10000	1
Fe	18	68.8	90000	11.1
Co	1	0.17	10000	4
Ni	24	9.23	651400	4.6
Cu	1	0.25	5000	0.47
Zn	40.3	0.01		1
Ga		450 ppm		
As		0.01		
Se		0.02		
Br		8 ppm		
Rb		10 ppm		1
Sr		0.2 ppm		0.47
Y		5 ppm		
Zr		20 ppm		0.5
Nb		0.012	9500	1.74
Mo	10	0.37	400	1

Table 1. (continued).

Constituent or Impurity	Uranium Metal Concentration (ppm)	SS-304 Concentration (wt%)	Inconel X-750 Concentration (ppm)	Graphite Concentration (ppm)
Ag	0.1	2 ppm		0.5
Cd	25			0.5
In	2			1
Sn	4	0.01		1
Sb		0.01		1
Cs		0.3 ppm		1
Ba		500 ppm		2.9
La		2.1 ppm		1.38
Ce		550 ppm		0.56
Pr				0.64
Nd				0.36
Sm		0.15 ppm		0.61
Eu		0.02 ppm		
Gd				0.08
Tb		0.71 ppm		0.26
Dy		1 ppm		0.16
Ho		1 ppm		0.08
Er				0.04
Tm				0.04
Yb		2 ppm		0.06
Lu		0.8 ppm		0.02
Hf		2 ppm		0.17
Ta			9500	0.35
W	2	520 ppm		25.5
Tl				1
Pb	1	0.002		6.9
Bi	0.4			1
Th		1 ppm		
U		2 ppm		

Table 2. Hypothetical fuel power history for a 62.5 MWd burnup.

Duration (days)	Cumulative Duration (days)	Time-Averaged Power (MWth)
365.25	365.25	0.05704
365.25	730.50	0.05704
365.25	1095.75	0.05704
1826.25	2922.00	0.0
1826.25	4748.25	0.0
1826.25	6574.50	0.0
1826.25	8400.75	0.0
1826.25	10227.00	0.0
3652.50	13879.50	0.0
5478.75	19358.25	0.0
5478.75	24837.00	0.0
5478.75	30315.75	0.0
7305.00	37620.75	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.

Hypothetical Fuel Element

Stainless Steel/Inconel Cladding, Uranium-Plutonium-Thorium Oxide Fuel

Fuel Meat:	Oxide ceramic UO_2-PuO_2-ThO_2 matrix
Inner Clad:	SS304
Outer Clad	Inconel X-750
Burnup:	62.5 MWd/element (maximum element burnup)
Basis of Calculation:	Single element (single rod)
BOL U-235:	371.9897 grams U-235 per element
BOL U-238:	297.5918 grams U-238 per element
BOL U-234:	29.7592 grams U-234 per element
BOL U-236:	44.6388 grams U-236 per element
BOL Total U:	743.9795 grams U per element
BOL Pu-238:	14.0261 grams Pu-238 per element
BOL Pu-239:	467.5381 grams Pu-239 per element
BOL Pu-240:	162.8591 grams Pu-240 per element
BOL Pu-241:	92.7284 grams Pu-241 per element
BOL Pu-242:	42.0784 grams Pu-242 per element
BOL Total Pu:	779.2301 grams Pu per element
BOL Th-232:	345.4439 grams th-232 per element
BOL U-235 Fuel Enrichment:	50 wt%
BOL Pu-239 Fuel Enrichment:	60 wt%

DECAY TIMES (years out of core)

(Activities* in Ci/element)

Radionuclide	5	10	15	20	25	35	50	65	80	100
H 3	9.328E-01	7.046E-01	5.322E-01	4.019E-01	3.036E-01	1.732E-01	7.461E-02	3.215E-02	1.385E-02	4.507E-03
BE 10	1.775E-07	1.775E-07	1.775E-07	1.775E-07	1.775E-07	1.775E-07	1.775E-07	1.775E-07	1.775E-07	1.775E-07
C 14	9.719E-04	9.713E-04	9.708E-04	9.702E-04	9.696E-04	9.684E-04	9.667E-04	9.649E-04	9.632E-04	9.608E-04
CL 36	1.711E-05	1.711E-05	1.711E-05	1.711E-05	1.711E-05	1.711E-05	1.711E-05	1.711E-05	1.710E-05	1.710E-05
CR 51	1.334E-18	1.924E-38	2.776E-58	4.005E-78	5.778E-98	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN 54	5.235E-02	9.115E-04	1.587E-05	2.762E-07	4.809E-09	1.458E-12	7.691E-18	4.058E-23	2.141E-28	1.967E-35
FE 55	5.403E+00	1.425E+00	3.757E-01	9.906E-02	2.612E-02	1.816E-03	3.330E-05	6.106E-07	1.120E-08	5.413E-11
FE 59	5.842E-13	3.546E-25	2.153E-37	1.307E-49	7.932E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CO 60	3.631E+01	1.881E+01	9.746E+00	5.049E+00	2.616E+00	7.020E-01	9.760E-02	1.357E-02	1.887E-03	1.359E-04
NI 59	2.430E-02	2.430E-02	2.430E-02	2.429E-02	2.429E-02	2.429E-02	2.429E-02	2.428E-02	2.428E-02	2.428E-02
NI 63	2.893E+00	2.786E+00	2.683E+00	2.584E+00	2.488E+00	2.308E+00	2.061E+00	1.841E+00	1.644E+00	1.414E+00
ZN 65	1.943E-04	1.082E-06	6.022E-09	3.353E-11	1.866E-13	5.785E-18	9.984E-25	1.723E-31	2.974E-38	2.857E-47
SE 79	6.428E-04	6.428E-04	6.427E-04	6.427E-04	6.427E-04	6.426E-04	6.425E-04	6.424E-04	6.423E-04	6.421E-04
KR 85	9.283E+00	6.718E+00	4.863E+00	3.519E+00	2.547E+00	1.334E+00	5.059E-01	1.918E-01	7.271E-02	1.995E-02
RB 87	2.847E-08	2.847E-08	2.847E-08	2.847E-08	2.847E-08	2.847E-08	2.847E-08	2.847E-08	2.847E-08	2.847E-08

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

Radionuclide	5	10	15	20	25	35	50	65	80	100
SR 89	1.455E-08	1.889E-19	2.452E-30	3.182E-41	4.130E-52	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR 90	8.478E+01	7.527E+01	6.683E+01	5.933E+01	5.267E+01	4.151E+01	2.905E+01	2.033E+01	1.422E+01	8.837E+00
Y 90	8.481E+01	7.529E+01	6.684E+01	5.934E+01	5.268E+01	4.153E+01	2.906E+01	2.033E+01	1.423E+01	8.839E+00
Y 91	6.036E-07	2.423E-16	9.726E-26	3.904E-35	1.567E-44	2.525E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZR 93	2.791E-03	2.791E-03	2.791E-03	2.791E-03	2.791E-03	2.791E-03	2.791E-03	2.791E-03	2.791E-03	2.791E-03
ZR 95	6.008E-06	1.536E-14	3.925E-23	1.003E-31	2.564E-40	1.676E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 93M	7.534E-04	1.180E-03	1.511E-03	1.768E-03	1.967E-03	2.240E-03	2.460E-03	2.562E-03	2.610E-03	2.637E-03
NB 94	9.911E-04	9.909E-04	9.907E-04	9.906E-04	9.904E-04	9.901E-04	9.896E-04	9.891E-04	9.885E-04	9.879E-04
NB 95	1.334E-05	3.409E-14	8.714E-23	2.227E-31	5.693E-40	3.720E-57	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NB 95M	4.457E-08	1.139E-16	2.912E-25	7.443E-34	1.902E-42	1.243E-59	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MO 93	1.282E-05	1.280E-05	1.279E-05	1.278E-05	1.277E-05	1.274E-05	1.270E-05	1.267E-05	1.263E-05	1.258E-05
TC 99	2.705E-02	2.705E-02	2.705E-02	2.705E-02	2.705E-02	2.705E-02	2.705E-02	2.705E-02	2.704E-02	2.704E-02
RU103	2.683E-11	2.714E-25	2.746E-39	2.778E-53	2.810E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU106	4.723E+01	1.517E+00	4.873E-02	1.565E-03	5.028E-05	5.188E-08	1.719E-12	5.698E-17	1.888E-21	2.010E-27
RH103M	2.419E-11	2.447E-25	2.475E-39	2.504E-53	2.533E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RH106	4.723E+01	1.517E+00	4.873E-02	1.565E-03	5.028E-05	5.188E-08	1.719E-12	5.698E-17	1.888E-21	2.010E-27
PD107	4.142E-04	4.142E-04	4.142E-04	4.142E-04	4.142E-04	4.142E-04	4.142E-04	4.142E-04	4.142E-04	4.142E-04
AG110	2.636E-04	1.662E-06	1.049E-08	6.618E-11	4.175E-13	1.661E-17	4.173E-24	1.048E-30	2.632E-37	4.169E-46
AG110M	1.982E-02	1.250E-04	7.886E-07	4.975E-09	3.139E-11	1.250E-15	3.138E-22	7.878E-29	1.979E-35	3.135E-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	7.053E-02	5.562E-02	4.386E-02	3.458E-02	2.727E-02	1.696E-02	8.315E-03	4.077E-03	1.999E-03	7.730E-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	8.304E-13	3.904E-25	1.836E-37	8.629E-50	4.057E-62	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114	2.717E-14	2.144E-25	1.691E-36	1.333E-47	1.052E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN114M	2.839E-14	2.240E-25	1.766E-36	1.393E-47	1.099E-58	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
IN115M	5.824E-17	2.738E-29	1.287E-41	6.052E-54	2.845E-66	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SN119M	1.236E-03	7.054E-06	4.024E-08	2.297E-10	1.310E-12	4.265E-17	7.922E-24	1.472E-30	2.734E-37	2.897E-46
SN121M	4.948E-04	4.617E-04	4.307E-04	4.019E-04	3.749E-04	3.263E-04	2.650E-04	2.153E-04	1.748E-04	1.325E-04
SN123	2.207E-04	1.224E-08	6.784E-13	3.761E-17	2.085E-21	6.409E-30	1.092E-42	1.861E-55	3.171E-68	2.995E-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	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.393E-03	1.393E-03
SB124	4.256E-10	3.135E-19	2.310E-28	1.702E-37	1.254E-46	6.808E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SB125	6.364E+00	1.821E+00	5.211E-01	1.491E-01	4.266E-02	3.493E-03	8.187E-05	1.918E-06	4.494E-08	3.013E-10
SB126	1.952E-04	1.952E-04	1.952E-04	1.952E-04	1.951E-04	1.951E-04	1.951E-04	1.951E-04	1.951E-04	1.950E-04
SB126M	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.394E-03	1.393E-03	1.393E-03

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
TE123M	2.881E-08	7.341E-13	1.871E-17	4.768E-22	1.215E-26	7.893E-36	1.306E-49	2.162E-63	3.580E-77	1.511E-95
TE125M	1.553E+00	4.443E-01	1.271E-01	3.638E-02	1.041E-02	8.525E-04	1.997E-05	4.680E-07	1.096E-08	7.353E-11
TE127	2.084E-04	1.886E-09	1.706E-14	1.544E-19	1.397E-24	1.144E-34	8.474E-50	6.278E-65	4.651E-80	0.000E+00
TE127M	2.128E-04	1.925E-09	1.742E-14	1.576E-19	1.426E-24	1.168E-34	8.652E-50	6.410E-65	4.749E-80	0.000E+00
TE129	2.111E-15	9.172E-32	3.985E-48	1.732E-64	7.524E-81	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE129M	3.243E-15	1.409E-31	6.122E-48	2.660E-64	1.156E-80	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I129	6.993E-05	6.993E-05	6.993E-05	6.993E-05	6.993E-05	6.993E-05	6.993E-05	6.993E-05	6.993E-05	6.993E-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	9.325E+00	1.736E+00	3.234E-01	6.023E-02	1.121E-02	3.889E-04	2.512E-06	1.622E-08	1.047E-10	1.260E-13
CS135	2.882E-03	2.882E-03	2.882E-03	2.882E-03	2.882E-03	2.882E-03	2.882E-03	2.882E-03	2.882E-03	2.882E-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	1.771E+02	1.578E+02	1.405E+02	1.252E+02	1.115E+02	8.854E+01	6.260E+01	4.427E+01	3.130E+01	1.972E+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	1.675E+02	1.492E+02	1.330E+02	1.184E+02	1.055E+02	8.376E+01	5.922E+01	4.188E+01	2.961E+01	1.865E+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	3.077E-14	3.790E-31	4.667E-48	5.748E-65	7.078E-82	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE142	4.827E-08	4.827E-08	4.827E-08	4.827E-08	4.827E-08	4.827E-08	4.827E-08	4.827E-08	4.827E-08	4.827E-08
CE144	2.163E+01	2.517E-01	2.931E-03	3.412E-05	3.972E-07	5.383E-11	8.492E-17	1.340E-22	2.114E-28	3.882E-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	2.163E+01	2.518E-01	2.931E-03	3.412E-05	3.972E-07	5.383E-11	8.492E-17	1.340E-22	2.114E-28	3.882E-36
PR144M	2.595E-01	3.021E-03	3.517E-05	4.094E-07	4.766E-09	6.459E-13	1.019E-18	1.608E-24	2.536E-30	4.658E-38
ND144	1.998E-12	2.006E-12	2.006E-12	2.006E-12	2.006E-12	2.006E-12	2.006E-12	2.006E-12	2.006E-12	2.006E-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	1.748E-07	1.455E-07	1.196E-07	9.837E-08	8.088E-08	5.467E-08	3.039E-08	1.689E-08	9.386E-09	4.289E-09
PM147	1.324E+02	3.533E+01	9.428E+00	2.516E+00	6.714E-01	4.781E-02	9.085E-04	1.726E-05	3.281E-07	1.664E-09
PM148M	2.514E-12	1.224E-25	5.962E-39	2.903E-52	1.414E-65	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PM148	1.416E-13	6.895E-27	3.358E-40	1.635E-53	7.964E-67	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SM145	3.949E-08	9.545E-10	2.307E-11	5.576E-13	1.348E-14	7.874E-18	1.112E-22	1.570E-27	2.217E-32	7.568E-39
SM147	1.418E-08	1.656E-08	1.719E-08	1.736E-08	1.741E-08	1.742E-08	1.742E-08	1.742E-08	1.742E-08	1.742E-08
SM151	5.445E+00	5.239E+00	5.042E+00	4.851E+00	4.668E+00	4.322E+00	3.850E+00	3.430E+00	3.056E+00	2.620E+00
EU152	7.184E-02	5.568E-02	4.315E-02	3.345E-02	2.593E-02	1.557E-02	7.250E-03	3.375E-03	1.571E-03	5.671E-04
EU154	2.401E+00	1.604E+00	1.072E+00	7.166E-01	4.789E-01	2.140E-01	6.386E-02	1.907E-02	5.691E-03	1.135E-03

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

Radionuclide	5	10	15	20	25	35	50	65	80	100
EU155	1.047E+01	5.206E+00	2.588E+00	1.287E+00	6.397E-01	1.581E-01	1.943E-02	2.388E-03	2.934E-04	1.792E-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	1.962E-04	1.050E-06	5.619E-09	3.006E-11	1.609E-13	4.605E-18	7.055E-25	1.081E-31	1.656E-38	1.357E-47
TB160	1.330E-08	3.314E-16	8.258E-24	2.057E-31	5.127E-39	3.184E-54	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TL206	4.953E-12	4.953E-12	4.953E-12	4.953E-12	4.953E-12	4.953E-12	4.953E-12	4.953E-12	4.953E-12	4.953E-12
TL207	3.320E-05	5.457E-05	7.281E-05	8.838E-05	1.017E-04	1.226E-04	1.439E-04	1.572E-04	1.656E-04	1.720E-04
TL208	1.981E-03	2.113E-03	2.053E-03	1.964E-03	1.874E-03	1.703E-03	1.476E-03	1.279E-03	1.109E-03	9.179E-04
PB210	1.948E-09	7.501E-09	1.886E-08	3.774E-08	6.563E-08	1.536E-07	3.815E-07	7.438E-07	1.256E-06	2.192E-06
PB211	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
PB212	5.512E-03	5.882E-03	5.714E-03	5.467E-03	5.216E-03	4.739E-03	4.107E-03	3.560E-03	3.087E-03	2.555E-03
BI211	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
BI212	5.512E-03	5.882E-03	5.714E-03	5.467E-03	5.216E-03	4.739E-03	4.107E-03	3.560E-03	3.087E-03	2.555E-03
PO212	3.532E-03	3.769E-03	3.661E-03	3.503E-03	3.342E-03	3.037E-03	2.632E-03	2.281E-03	1.978E-03	1.637E-03
PO215	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
PO216	5.512E-03	5.882E-03	5.714E-03	5.467E-03	5.216E-03	4.739E-03	4.107E-03	3.560E-03	3.087E-03	2.555E-03
RN219	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
RN220	5.512E-03	5.882E-03	5.714E-03	5.467E-03	5.216E-03	4.739E-03	4.107E-03	3.560E-03	3.087E-03	2.555E-03
FR223	4.592E-07	7.543E-07	1.006E-06	1.221E-06	1.404E-06	1.695E-06	1.989E-06	2.173E-06	2.288E-06	2.378E-06
RA223	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
RA224	5.512E-03	5.882E-03	5.714E-03	5.467E-03	5.216E-03	4.739E-03	4.107E-03	3.560E-03	3.087E-03	2.555E-03
RA226	2.309E-08	6.006E-08	1.146E-07	1.869E-07	2.773E-07	5.133E-07	1.009E-06	1.678E-06	2.527E-06	3.943E-06
RA228	2.118E-05	2.782E-05	3.178E-05	3.414E-05	3.555E-05	3.689E-05	3.747E-05	3.760E-05	3.762E-05	3.763E-05
AC227	3.327E-05	5.466E-05	7.291E-05	8.848E-05	1.018E-04	1.228E-04	1.442E-04	1.575E-04	1.658E-04	1.723E-04
TH227	3.283E-05	5.397E-05	7.201E-05	8.741E-05	1.005E-04	1.213E-04	1.423E-04	1.555E-04	1.637E-04	1.701E-04
TH228	5.511E-03	5.877E-03	5.709E-03	5.462E-03	5.211E-03	4.739E-03	4.107E-03	3.560E-03	3.086E-03	2.555E-03
TH229	1.363E-05	2.428E-05	3.493E-05	4.558E-05	5.622E-05	7.749E-05	1.093E-04	1.412E-04	1.729E-04	2.152E-04
TH230	1.308E-05	2.116E-05	2.938E-05	3.773E-05	4.623E-05	6.359E-05	9.051E-05	1.184E-04	1.471E-04	1.865E-04
TH231	7.574E-04	7.575E-04	7.576E-04	7.578E-04	7.579E-04	7.581E-04	7.585E-04	7.589E-04	7.593E-04	7.598E-04
TH232	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05
TH234	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05
PA231	1.786E-04	1.786E-04	1.787E-04	1.788E-04	1.788E-04	1.789E-04	1.791E-04	1.793E-04	1.795E-04	1.797E-04
PA233	1.544E-03	1.908E-03	2.400E-03	2.992E-03	3.661E-03	5.160E-03	7.643E-03	1.025E-02	1.287E-02	1.634E-02
PA234M	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05
PA234	1.286E-07	1.286E-07	1.286E-07	1.286E-07	1.286E-07	1.286E-07	1.286E-07	1.286E-07	1.286E-07	1.286E-07
U232	6.108E-03	5.822E-03	5.549E-03	5.288E-03	5.040E-03	4.577E-03	3.962E-03	3.429E-03	2.968E-03	2.448E-03

	DECAY TIMES (years out of core) (Activities* in Ci/element)									
Radionuclide	5	10	15	20	25	35	50	65	80	100
U233	2.259E-02	2.259E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02
U234	1.778E-01	1.811E-01	1.842E-01	1.872E-01	1.902E-01	1.956E-01	2.031E-01	2.097E-01	2.156E-01	2.225E-01
U235	7.574E-04	7.575E-04	7.576E-04	7.578E-04	7.579E-04	7.581E-04	7.585E-04	7.589E-04	7.593E-04	7.598E-04
U236	3.196E-03	3.199E-03	3.201E-03	3.204E-03	3.207E-03	3.212E-03	3.220E-03	3.229E-03	3.237E-03	3.247E-03
U237	3.165E-03	2.488E-03	1.955E-03	1.537E-03	1.208E-03	7.467E-04	3.627E-04	1.762E-04	8.557E-05	3.268E-05
U238	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05	9.890E-05
NP237	1.544E-03	1.908E-03	2.400E-03	2.992E-03	3.661E-03	5.160E-03	7.643E-03	1.025E-02	1.287E-02	1.634E-02
PU236	3.763E-05	1.116E-05	3.309E-06	9.811E-07	2.910E-07	2.568E-08	7.713E-10	1.218E-10	1.049E-10	1.044E-10
PU237	5.975E-15	5.252E-27	4.616E-39	4.057E-51	3.566E-63	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
PU238	2.358E+02	2.268E+02	2.180E+02	2.096E+02	2.015E+02	1.863E+02	1.656E+02	1.471E+02	1.308E+02	1.118E+02
PU239	2.605E+01	2.605E+01	2.604E+01	2.604E+01	2.604E+01	2.603E+01	2.602E+01	2.601E+01	2.600E+01	2.598E+01
PU240	1.828E+01	1.829E+01	1.829E+01	1.829E+01	1.829E+01	1.829E+01	1.827E+01	1.825E+01	1.822E+01	1.819E+01
PU241	1.290E+04	1.014E+04	7.971E+03	6.266E+03	4.926E+03	3.044E+03	1.478E+03	7.181E+02	3.488E+02	1.332E+02
PU242	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01
PU244	4.492E-09	4.492E-09	4.492E-09	4.492E-09	4.492E-09	4.492E-09	4.492E-09	4.492E-09	4.492E-09	4.492E-09
AM241	1.775E+02	2.676E+02	3.374E+02	3.913E+02	4.326E+02	4.879E+02	5.278E+02	5.402E+02	5.395E+02	5.295E+02
AM242M	1.207E+00	1.180E+00	1.153E+00	1.127E+00	1.102E+00	1.053E+00	9.830E-01	9.180E-01	8.573E-01	7.826E-01
AM242	1.201E+00	1.174E+00	1.147E+00	1.122E+00	1.096E+00	1.047E+00	9.781E-01	9.135E-01	8.531E-01	7.787E-01
AM243	1.023E+00	1.022E+00	1.022E+00	1.021E+00	1.021E+00	1.020E+00	1.018E+00	1.017E+00	1.015E+00	1.013E+00
CM242	1.485E+00	9.714E-01	9.493E-01	9.279E-01	9.070E-01	8.662E-01	8.089E-01	7.554E-01	7.055E-01	6.440E-01
CM243	1.123E-01	9.944E-02	8.805E-02	7.797E-02	6.904E-02	5.414E-02	3.759E-02	2.610E-02	1.812E-02	1.114E-02
CM244	3.320E+01	2.741E+01	2.264E+01	1.870E+01	1.544E+01	1.053E+01	5.930E+00	3.340E+00	1.881E+00	8.749E-01
CM245	1.982E-03	1.982E-03	1.981E-03	1.980E-03	1.979E-03	1.977E-03	1.975E-03	1.973E-03	1.970E-03	1.967E-03
CM246	2.010E-05	2.008E-05	2.007E-05	2.005E-05	2.004E-05	2.001E-05	1.996E-05	1.992E-05	1.988E-05	1.982E-05
CM247	1.442E-11	1.442E-11	1.442E-11	1.442E-11	1.442E-11	1.442E-11	1.442E-11	1.442E-11	1.442E-11	1.442E-11
SUBTOTAL**	1.427E+04	1.125E+04	9.043E+03	7.318E+03	5.961E+03	4.042E+03	2.413E+03	1.589E+03	1.163E+03	8.834E+02
TOTAL***	1.427E+04	1.126E+04	9.044E+03	7.319E+03	5.962E+03	4.043E+03	2.414E+03	1.591E+03	1.164E+03	8.844E+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.

Worst-Case Template 29

This analysis was focused on creating a bounding source term estimate. The hypothetical fuel template was created to try to maximize the source term for all radionuclides, but radionuclide production and decay is a complex multivariable process. It is impossible to maximize all radionuclides. Consequently, Template 29 was created by normalizing all the completed templates to the same basis (Ci per MWd per kg) and selecting the highest curie content for each radionuclide. This template, shown below, is used to conservatively estimate source terms when sufficient information is not known to select one of the other templates. It is expected to be extremely conservative for any credible fuel. This worst-case template was used in the analysis for spent fuels that didn't fit any of the other completed templates. The hypothetical template was not used in the analysis (other than to derive this Worst-Case Template 29).

	DECAY TIMES (years)									
	5	10	15	20	25	35	50	65	80	100
Ac-227	3.327E-05	5.466E-05	7.291E-05	8.848E-05	1.018E-04	1.228E-04	1.442E-04	1.575E-04	1.658E-04	1.723E-04
Ag-110	2.572E-03	1.622E-05	1.024E-07	6.458E-10	4.074E-12	1.621E-16	4.072E-23	0.000E+00	0.000E+00	0.000E+00
Ag-110m	1.933E-01	1.219E-03	7.699E-06	4.855E-08	3.063E-10	1.219E-14	3.062E-21	0.000E+00	0.000E+00	0.000E+00
Ag-111	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Am-241	1.775E+02	2.676E+02	3.374E+02	3.913E+02	4.326E+02	4.879E+02	5.278E+02	5.402E+02	5.395E+02	5.295E+02
Am-242	1.201E+00	1.174E+00	1.147E+00	1.122E+00	1.096E+00	1.047E+00	9.781E-01	9.135E-01	8.531E-01	7.787E-01
Am-242m	1.207E+00	1.180E+00	1.153E+00	1.127E+00	1.102E+00	1.053E+00	9.830E-01	9.180E-01	8.573E-01	7.826E-01
Am-243	1.023E+00	1.022E+00	1.022E+00	1.021E+00	1.021E+00	1.020E+00	1.018E+00	1.017E+00	1.015E+00	1.013E+00
Ba-136m	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ba-137m	2.488E+03	2.216E+03	1.975E+03	1.759E+03	1.567E+03	1.245E+03	8.802E+02	6.226E+02	4.403E+02	2.774E+02
Ba-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Be-10	1.527E-05	1.527E-05	1.527E-05	1.527E-05	1.527E-05	1.527E-05	1.527E-05	1.527E-05	1.527E-05	1.527E-05
Bi-211	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
Bi-212	1.870E-02	1.912E-02	1.843E-02	1.760E-02	1.678E-02	1.524E-02	1.319E-02	1.141E-02	9.880E-03	8.157E-03
C-14	7.583E+00	7.577E+00	7.570E+00	7.570E+00	7.563E+00	7.556E+00	7.543E+00	7.529E+00	7.515E+00	7.495E+00
Cd-113	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cd-113m	2.758E-01	2.174E-01	1.715E-01	1.353E-01	1.067E-01	6.636E-02	3.255E-02	1.598E-02	7.834E-03	3.031E-03
Cd-115m	2.291E-11	1.099E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ce-141	2.889E-12	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ce-142	1.276E-06	1.276E-06	1.276E-06	1.276E-06	1.276E-06	1.276E-06	1.276E-06	1.276E-06	1.276E-06	1.276E-06
Ce-144	8.500E+02	9.926E+00	1.159E-01	1.353E-03	1.581E-05	8.425E-07	3.430E-15	2.939E-18	1.679E-18	0.000E+00
Cl-36	1.429E-01	1.429E-01	1.429E-01	1.429E-01	1.428E-01	1.428E-01	1.428E-01	1.428E-01	1.428E-01	1.428E-01
Cm-242	1.485E+00	9.714E-01	9.493E-01	9.279E-01	9.070E-01	8.662E-01	8.089E-01	7.554E-01	7.055E-01	6.440E-01
Cm-243	1.123E-01	9.944E-02	8.805E-02	7.797E-02	6.904E-02	5.414E-02	3.759E-02	2.610E-02	1.812E-02	1.114E-02
Cm-244	3.320E+01	2.741E+01	2.264E+01	1.870E+01	1.544E+01	1.053E+01	5.930E+00	3.340E+00	1.881E+00	8.749E-01
Cm-245	1.982E-03	1.982E-03	1.981E-03	1.980E-03	1.979E-03	1.977E-03	1.975E-03	1.973E-03	1.970E-03	1.967E-03
Cm-246	1.915E-04	1.914E-04	1.912E-04	1.911E-04	1.909E-04	1.906E-04	1.902E-04	1.898E-04	1.893E-04	1.888E-04
Cm-247	6.671E-10	6.671E-10	6.671E-10	6.671E-10	6.671E-10	6.671E-10	6.671E-10	6.671E-10	6.671E-10	6.671E-10
Co-60	9.084E+04	4.704E+04	2.437E+04	1.262E+04	6.541E+03	1.755E+03	2.441E+02	3.393E+01	4.719E+00	3.399E-01
Cr-51	1.652E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cs-134	5.083E+02	9.478E+01	1.767E+01	3.296E+00	6.143E-01	2.134E-02	1.384E-04	8.966E-07	1.838E-07	7.020E-12

	DECAY TIMES (years)									
	5	10	15	20	25	35	50	65	80	100
Cs-135	2.749E-02	2.749E-02	2.749E-02	2.749E-02	2.749E-02	2.749E-02	2.749E-02	2.749E-02	2.749E-02	2.749E-02
Cs-136	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cs-137	2.629E+03	2.343E+03	2.088E+03	1.860E+03	1.658E+03	1.316E+03	9.305E+02	6.580E+02	4.654E+02	2.933E+02
Eu-152	5.555E+00	4.306E+00	3.338E+00	2.588E+00	2.006E+00	1.206E+00	5.615E-01	2.615E-01	1.219E-01	4.398E-02
Eu-154	8.753E+02	5.851E+02	3.912E+02	2.615E+02	1.748E+02	7.812E+01	2.334E+01	6.972E+00	2.083E+00	4.160E-01
Eu-155	2.847E+02	1.416E+02	7.045E+01	3.504E+01	1.743E+01	4.312E+00	5.306E-01	6.528E-02	8.035E-03	4.918E-04
Eu-156	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Fe-55	5.450E+04	1.437E+04	3.789E+03	9.991E+02	2.634E+02	1.832E+01	3.359E-01	6.159E-03	1.129E-04	5.460E-07
Fe-59	1.215E-08	7.372E-21	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Fr-223	4.592E-07	7.543E-07	1.006E-06	1.221E-06	1.404E-06	1.695E-06	1.989E-06	2.173E-06	2.288E-06	2.378E-06
Gd-153	3.031E-02	1.622E-04	8.681E-07	4.644E-09	2.484E-11	7.113E-16	1.090E-22	0.000E+00	0.000E+00	0.000E+00
H-3	8.177E+01	6.180E+01	4.667E+01	3.526E+01	2.662E+01	1.519E+01	6.545E+00	2.820E+00	1.215E+00	3.953E-01
I-129	6.636E-04	6.636E-04	6.636E-04	6.636E-04	6.636E-04	6.636E-04	6.636E-04	6.636E-04	6.636E-04	6.636E-04
I-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
In-114	1.140E-11	8.988E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
In-114m	1.191E-11	9.391E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
In-115m	1.608E-15	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Kr-85	2.601E+02	1.883E+02	1.363E+02	9.864E+01	7.141E+01	3.743E+01	1.420E+01	5.387E+00	2.044E+00	5.613E-01
La-140	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MN-54	4.195E+01	7.304E-01	1.271E-02	2.213E-04	3.853E-06	3.147E-07	6.162E-15	1.229E-17	7.023E-18	0.000E+00
Mo-93	4.926E-02	4.921E-02	4.916E-02	4.912E-02	4.907E-02	4.897E-02	4.882E-02	4.868E-02	4.854E-02	4.834E-02
Nb-93m	2.169E-02	3.679E-02	4.849E-02	5.757E-02	6.460E-02	7.427E-02	8.204E-02	8.566E-02	8.736E-02	8.832E-02
Nb-94	6.799E-02	6.797E-02	6.797E-02	6.795E-02	6.794E-02	6.792E-02	6.788E-02	6.785E-02	6.782E-02	6.777E-02
Nb-95	1.113E-03	2.884E-12	7.473E-21	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Nb-95m	3.720E-06	9.638E-15	2.497E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Nd-144	3.807E-11	3.838E-11	3.839E-11	3.839E-11	3.839E-11	3.839E-11	3.839E-11	3.839E-11	3.839E-11	3.839E-11
Nd-147	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ni-59	2.130E+01	2.130E+01	2.130E+01	2.130E+01	2.130E+01	2.130E+01	2.129E+01	2.129E+01	2.128E+01	2.128E+01
Ni-63	3.064E+03	2.951E+03	2.842E+03	2.737E+03	2.636E+03	2.444E+03	2.183E+03	1.950E+03	1.742E+03	1.498E+03
Np-237	9.759E-03	9.760E-03	9.762E-03	9.766E-03	9.770E-03	9.779E-03	9.792E-03	1.025E-02	1.287E-02	1.634E-02
Pa-231	1.786E-04	1.786E-04	1.787E-04	1.788E-04	1.788E-04	1.789E-04	1.791E-04	1.793E-04	1.795E-04	1.797E-04

	DECAY TIMES (years)									
	5	10	15	20	25	35	50	65	80	100
Pa-233	9.759E-03	9.760E-03	9.762E-03	9.766E-03	9.770E-03	9.779E-03	9.792E-03	1.025E-02	1.287E-02	1.634E-02
Pa-234	9.541E-06	9.541E-06	9.541E-06	9.541E-06	9.541E-06	9.541E-06	9.541E-06	9.541E-06	9.541E-06	9.541E-06
Pa-234m	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03
Pb-210	7.042E-09	1.980E-08	5.865E-08	1.278E-07	2.341E-07	5.798E-07	1.495E-06	2.957E-06	5.017E-06	8.743E-06
Pb-211	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
Pb-212	1.870E-02	1.912E-02	1.843E-02	1.760E-02	1.678E-02	1.524E-02	1.319E-02	1.141E-02	9.880E-03	8.157E-03
Pd-107	1.013E-03	1.013E-03	1.013E-03	1.013E-03	1.013E-03	1.013E-03	1.013E-03	1.013E-03	1.013E-03	1.013E-03
Pm-145	8.675E-01	7.207E-01	5.928E-01	4.875E-01	4.008E-01	2.710E-01	1.507E-01	8.378E-02	4.659E-02	2.130E-02
Pm-147	2.910E+03	7.770E+02	2.076E+02	5.544E+01	1.481E+01	1.056E+00	2.012E-02	3.835E-04	1.606E-05	3.718E-08
Pm-148	3.534E-12	1.757E-25	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pm-148m	6.275E-11	3.121E-24	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Po-212	1.198E-02	1.225E-02	1.181E-02	1.128E-02	1.076E-02	9.757E-03	8.447E-03	7.310E-03	6.330E-03	5.226E-03
Po-215	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
Po-216	1.870E-02	1.912E-02	1.843E-02	1.760E-02	1.678E-02	1.524E-02	1.319E-02	1.141E-02	9.880E-03	8.157E-03
Pr-143	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pr-144	8.500E+02	9.926E+00	1.159E-01	1.353E-03	1.581E-05	8.425E-07	3.430E-15	2.939E-18	1.679E-18	0.000E+00
Pr-144m	1.020E+01	1.191E-01	1.391E-03	1.625E-05	1.896E-07	1.012E-08	4.117E-17	3.524E-20	2.014E-20	0.000E+00
Pu-236	9.558E-04	2.836E-04	8.417E-05	2.498E-05	7.422E-06	6.624E-07	2.676E-08	1.014E-08	9.710E-09	9.692E-09
Pu-237	4.701E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pu-238	2.358E+02	2.268E+02	2.180E+02	2.096E+02	2.015E+02	1.863E+02	1.656E+02	1.471E+02	1.308E+02	1.118E+02
Pu-239	2.605E+01	2.605E+01	2.604E+01	2.604E+01	2.604E+01	2.603E+01	2.602E+01	2.601E+01	2.600E+01	2.598E+01
Pu-240	1.828E+01	1.829E+01	1.829E+01	1.829E+01	1.829E+01	1.829E+01	1.827E+01	1.825E+01	1.822E+01	1.819E+01
Pu-241	1.290E+04	1.014E+04	7.971E+03	6.266E+03	4.926E+03	3.044E+03	1.478E+03	7.181E+02	3.488E+02	1.332E+02
Pu-242	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01	1.535E-01
Pu-244	5.307E-09	5.307E-09	5.307E-09	5.307E-09	5.307E-09	5.307E-09	5.307E-09	5.307E-09	5.307E-09	5.307E-09
Ra-223	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
Ra-224	1.870E-02	1.912E-02	1.843E-02	1.760E-02	1.678E-02	1.524E-02	1.319E-02	1.141E-02	9.880E-03	8.157E-03
Ra-226	5.641E-08	1.889E-07	3.989E-07	6.861E-07	1.051E-06	2.010E-06	4.025E-06	6.725E-06	1.011E-05	1.566E-05
Ra-228	2.118E-05	2.782E-05	3.178E-05	3.414E-05	3.555E-05	3.689E-05	3.747E-05	3.760E-05	3.762E-05	3.763E-05
Rb-87	8.187E-07	8.187E-07	8.187E-07	8.187E-07	8.187E-07	8.187E-07	8.187E-07	8.187E-07	8.187E-07	8.187E-07
Rh-103m	1.067E-09	1.104E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

	DECAY TIMES (years)									
	5	10	15	20	25	35	50	65	80	100
Rh-106	2.298E+02	7.379E+00	2.371E-01	7.618E-03	2.446E-04	5.345E-05	8.368E-12	8.227E-14	4.701E-14	0.000E+00
Rn-219	3.329E-05	5.473E-05	7.302E-05	8.863E-05	1.019E-04	1.230E-04	1.443E-04	1.577E-04	1.660E-04	1.725E-04
Rn-220	1.870E-02	1.912E-02	1.843E-02	1.760E-02	1.678E-02	1.524E-02	1.319E-02	1.141E-02	9.880E-03	8.157E-03
Ru-103	1.184E-09	1.224E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ru-106	2.298E+02	7.379E+00	2.371E-01	7.618E-03	2.446E-04	5.345E-05	8.368E-12	8.227E-14	4.701E-14	0.000E+00
Sb-124	2.081E-06	1.533E-15	1.129E-24	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Sb-125	1.176E+02	3.366E+01	9.629E+00	2.756E+00	7.884E-01	6.457E-02	1.513E-03	3.545E-05	4.220E-06	5.569E-09
Sb-126	1.459E-03	1.459E-03	1.459E-03	1.459E-03	1.459E-03	1.459E-03	1.458E-03	1.458E-03	1.458E-03	1.458E-03
Sb-126M	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.041E-02	1.041E-02
Se-79	1.199E-02	1.199E-02	1.199E-02	1.199E-02	1.199E-02	1.199E-02	1.199E-02	1.199E-02	1.198E-02	1.198E-02
Sm-145	1.665E-01	4.034E-03	9.776E-05	2.369E-06	5.741E-08	3.371E-11	4.797E-16	6.826E-21	0.000E+00	0.000E+00
Sm-147	6.263E-06	6.287E-06	6.294E-06	6.295E-06	6.296E-06	6.296E-06	6.296E-06	6.296E-06	6.296E-06	6.296E-06
Sm-151	1.766E+02	1.700E+02	1.635E+02	1.573E+02	1.514E+02	1.402E+02	1.249E+02	1.113E+02	9.913E+01	8.499E+01
Sn-119m	8.272E+00	4.719E-02	2.692E-04	1.536E-06	8.763E-09	2.853E-13	5.300E-20	0.000E+00	0.000E+00	0.000E+00
Sn-121m	1.457E-01	1.360E-01	1.269E-01	1.184E-01	1.104E-01	9.616E-02	7.809E-02	6.341E-02	5.150E-02	3.902E-02
Sn-123	2.085E-02	1.156E-06	6.408E-11	3.553E-15	1.970E-19	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Sn-125	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Sn-126	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.042E-02	1.041E-02	1.041E-02
Sr-89	2.098E-06	2.771E-17	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Sr-90	2.525E+03	2.243E+03	1.991E+03	1.768E+03	1.570E+03	1.237E+03	8.662E+02	6.063E+02	4.243E+02	2.637E+02
Tb-160	8.938E-06	2.254E-13	5.683E-21	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Tc-99	4.230E-01	4.230E-01	4.230E-01	4.230E-01	4.230E-01	4.230E-01	4.230E-01	4.228E-01	4.228E-01	4.228E-01
Te-123m	4.057E-04	1.035E-08	2.635E-13	6.717E-18	1.712E-22	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Te-125m	2.870E+01	8.211E+00	2.350E+00	6.724E-01	1.924E-01	1.575E-02	3.691E-04	8.647E-06	1.028E-06	1.359E-09
Te-127	4.862E-03	4.435E-08	4.044E-13	3.690E-18	3.366E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	#VALUE!
Te-127m	4.965E-03	4.527E-08	4.130E-13	3.767E-18	3.435E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	#VALUE!
Te-129	1.130E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Te-129m	1.736E-13	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Th-227	3.283E-05	5.397E-05	7.201E-05	8.741E-05	1.005E-04	1.213E-04	1.423E-04	1.555E-04	1.637E-04	1.701E-04
Th-228	1.869E-02	1.911E-02	1.842E-02	1.759E-02	1.677E-02	1.523E-02	1.319E-02	1.141E-02	9.880E-03	8.157E-03
Th-229	1.730E-05	2.623E-05	3.517E-05	4.558E-05	5.622E-05	7.749E-05	1.093E-04	1.412E-04	1.729E-04	2.152E-04

	DECAY TIMES (years)									
	5	10	15	20	25	35	50	65	80	100
Th-230	4.329E-05	7.924E-05	1.152E-04	1.512E-04	1.871E-04	2.590E-04	3.669E-04	4.747E-04	5.825E-04	7.263E-04
Th-231	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02
Th-232	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05	3.763E-05
Th-234	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03
Tl-206	1.647E-10	1.647E-10	1.647E-10	1.647E-10	1.647E-10	1.647E-10	1.647E-10	1.647E-10	1.647E-10	1.647E-10
Tl-207	3.320E-05	5.457E-05	7.281E-05	8.838E-05	1.017E-04	1.226E-04	1.439E-04	1.572E-04	1.656E-04	1.720E-04
Tl-208	6.719E-03	6.870E-03	6.624E-03	6.325E-03	6.030E-03	5.473E-03	4.737E-03	4.100E-03	3.549E-03	2.931E-03
U-232	1.978E-02	1.885E-02	1.796E-02	1.712E-02	1.631E-02	1.482E-02	1.283E-02	1.110E-02	9.605E-03	7.925E-03
U-233	2.259E-02	2.259E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02	2.258E-02
U-234	7.986E-01	7.993E-01	7.993E-01	7.993E-01	7.993E-01	7.993E-01	7.993E-01	7.993E-01	7.993E-01	7.993E-01
U-235	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02	3.673E-02
U-236	1.468E-02	1.468E-02	1.468E-02	1.468E-02	1.468E-02	1.468E-02	1.468E-02	1.468E-02	1.468E-02	1.468E-02
U-237	3.165E-03	2.488E-03	1.955E-03	1.537E-03	1.208E-03	7.467E-04	3.627E-04	1.762E-04	8.557E-05	3.268E-05
U-238	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03	7.338E-03
Xe-131M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Xe-133	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y-90	2.527E+03	2.243E+03	1.991E+03	1.768E+03	1.570E+03	1.238E+03	8.663E+02	6.064E+02	4.243E+02	2.637E+02
Y-91	7.517E-05	3.062E-14	1.248E-23	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Zn-65	1.670E+00	9.328E-03	5.211E-05	2.912E-07	1.627E-09	5.080E-14	8.859E-21	0.000E+00	0.000E+00	0.000E+00
Zr-93	9.350E-02	9.350E-02	9.350E-02	9.350E-02	9.350E-02	9.350E-02	9.350E-02	9.350E-02	9.350E-02	9.350E-02
Zr-95	5.014E-04	1.299E-12	3.366E-21	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Appendix B

Index to DOE Spent Nuclear Fuels

Intentionally Blank

Fuel Name	SNF ID	TSPA Category	DBE Category		Source Term Est. Page # 2010	2030
Hanford SNF						
AMERICIUM TARGETS	776	02. Pu/U Alloy SNF	Other	NOT INTACT	C-3	D-3
CALVERT CLIFFS 1	307	08. U Oxide SNF	Non-metals	INTACT	C-4	D-4
COOPER NUCLEAR	308	08. U Oxide SNF	Non-metals	INTACT	C-5	D-5
FFTF-DFA/TDFA	71	04. MOX SNF	Non-metals	INTACT	C-6	D-6
FFTF-DFA/TDFA PINS	323	04. MOX SNF	Non-metals	INTACT	C-7	D-7
FFTF-TFA PINS	320	04. MOX SNF	Non-metals	INTACT	C-8	D-8
FFTF-TFA PINS (AC-3)	1046	03. Pu/U Carbide SNF	Non-metals	INTACT	C-9	D-9
FFTF-TFA-AB-1	317	04. MOX SNF	Non-metals	INTACT	C-10	D-10
FFTF-TFA-ABA-1 THRU 6	318	08. U Oxide SNF	Non-metals	INTACT	C-11	D-11
FFTF-TFA-ACN-1 (MOX) PINS	321	04. MOX SNF	Non-metals	INTACT	C-12	D-12
FFTF-TFA-ACN-1 (PU/UC) PINS	865	03. Pu/U Carbide SNF	Non-metals	INTACT	C-13	D-13
FFTF-TFA-ACO-2, 4 THRU 16	329	04. MOX SNF	Non-metals	INTACT	C-14	D-14
FFTF-TFA-CRBR-3 & CRBR-5	322	04. MOX SNF	Non-metals	INTACT	C-15	D-15
FFTF-TFA-DEA-2	324	04. MOX SNF	Non-metals	INTACT	C-16	D-16
FFTF-TFA-FC-1	325	03. Pu/U Carbide SNF	Non-metals	INTACT	C-17	D-17
FFTF-TFA-MFF-1 & 1A (CDE)	330	04. MOX SNF	Non-metals	INTACT	C-18	D-18
FFTF-TFA-P0-2,4 & 5	333	04. MOX SNF	Non-metals	INTACT	C-19	D-19
FFTF-TFA-SRF-3&4	334	04. MOX SNF	Non-metals	INTACT	C-20	D-20
FFTF-TFA-UO-1	335	04. MOX SNF	Non-metals	INTACT	C-21	D-21
FFTF-TFA-WBO18 & WBO42	336	08. U Oxide SNF	Non-metals	INTACT	C-22	D-22
GE TEST	96	04. MOX SNF	Non-metals	NOT INTACT	C-23	D-23
LWR COMMERCIAL FUEL	130	08. U Oxide SNF	Non-metals	NOT INTACT	C-24	D-24
LWR SCRAP	309	08. U Oxide SNF	Non-metals	NOT INTACT	C-25	D-25
N REACTOR	991	07. U Metal SNF	Other	NOT INTACT	C-26	D-26
POINT BEACH	311	08. U Oxide SNF	Non-metals	INTACT	C-27	D-27
SHIPPINGPORT PWR C2 BLKT	193	08. U Oxide SNF	Non-metals	INTACT	C-28	D-28
SINGLE PASS REACTOR FUEL	198	07. U Metal SNF	Other	NOT INTACT	C-29	D-29
SINGLE PASS REACTOR FUEL	197	07. U Metal SNF	Other	NOT INTACT	C-30	D-30
SP-100 FUEL	777	08. U Oxide SNF	Non-metals	INTACT	C-31	D-31

					Source Term Est. Page #	
Fuel Name	SNF ID	TSPA Category	DBE Category		2010	2030
TRIGA 8.5/20 FFCR (DORF)	315	11. U Zr H SNF	Non-metals	INTACT	C-32	D-32
TRIGA STD (ALUM) HANFORD	314	11. U Zr H SNF	Non-metals	INTACT	C-33	D-33
TRIGA STD 8.5/20	233	11. U Zr H SNF	Non-metals	INTACT	C-34	D-34
TRIGA STD 8.5/20 (HANFORD)	316	11. U Zr H SNF	Non-metals	INTACT	C-35	D-35
INEEL SNF						
ACRR (PULSED CORE)	757	08. U Oxide SNF	Non-metals	INTACT	C-36	D-36
ANP	451	08. U Oxide SNF	Non-metals	INTACT	C-37	D-37
APPR (AGE-2)	6	08. U Oxide SNF	Non-metals	INTACT	C-38	D-38
ARKANSAS	7	08. U Oxide SNF	Non-metals	NOT INTACT	C-39	D-39
ARMF (PLATES)	8	09. Alum Based SNF	Stable	INTACT	C-40	D-40
ARMF/CFRMF MARK I	9	09. Alum Based SNF	Stable	INTACT	C-41	D-41
ARMF/CFRMF MARK I LL	10	09. Alum Based SNF	Stable	INTACT	C-42	D-42
ARMF/CFRMF MARK II	11	09. Alum Based SNF	Stable	INTACT	C-43	D-43
ARMF/CFRMF MARK III	12	09. Alum Based SNF	Stable	INTACT	C-44	D-44
ATR	15	09. Alum Based SNF	Stable	INTACT	C-45	D-45
ATR	16	09. Alum Based SNF	Stable	INTACT	C-46	D-46
ATR	843	09. Alum Based SNF	Stable	INTACT	C-47	D-47
BCD B-17 (TURKEY POINT 3)	19	08. U Oxide SNF	Non-metals	INTACT	C-48	D-48
BER-II TRIGA (FLIP LEU 45/20) (GERMANY)	236	11. U Zr H SNF	Non-metals	INTACT	C-49	D-49
BMI (CPI-24)	774	08. U Oxide SNF	Non-metals	NOT INTACT	C-50	D-50
BMI (CPI-38)	20	08. U Oxide SNF	Non-metals	NOT INTACT	C-51	D-51
BORAX V (SUPERHEATER)	22	08. U Oxide SNF	Non-metals	INTACT	C-52	D-52
BR-3	927	08. U Oxide SNF	Non-metals	INTACT	C-53	D-53
BR-3 FUEL	340	08. U Oxide SNF	Non-metals	INTACT	C-54	D-54
BRP-B	23	08. U Oxide SNF	Non-metals	INTACT	C-55	D-55
BRP-C	24	08. U Oxide SNF	Non-metals	INTACT	C-56	D-56
BRP-D1	25	08. U Oxide SNF	Non-metals	INTACT	C-57	D-57
BRP-D2	26	08. U Oxide SNF	Non-metals	INTACT	C-58	D-58
BRP-E	27	08. U Oxide SNF	Non-metals	INTACT	C-59	D-59
BRP-EG	28	08. U Oxide SNF	Non-metals	INTACT	C-60	D-60
BRP-EG/F	1081	08. U Oxide SNF	Non-metals	INTACT	C-61	D-61

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BRP-EP	29	04. MOX SNF	Non-metals	INTACT	C-62	D-62
BRP-F	30	08. U Oxide SNF	Non-metals	INTACT	C-63	D-63
BRP-F-PU	1082	08. U Oxide SNF	Non-metals	INTACT	C-64	D-64
CONNECTICUT YANKEE (S004)	34	08. U Oxide SNF	Non-metals	INTACT	C-65	D-65
CP-5 CONVERTER CYLINDERS	36	02. Pu/U Alloy SNF	Other	INTACT	C-66	D-66
DOE TEST & EXPERIMENTAL (ALUM)	42	10. Misc. SNF	Other	NOT INTACT	C-67	D-67
DOE TEST & EXPERIMENTAL (SST)	857	10. Misc. SNF	Other	NOT INTACT	C-68	D-68
DOE TEST & EXPERIMENTAL (ZIRC)	858	10. Misc. SNF	Other	NOT INTACT	C-69	D-69
DRCT	701	08. U Oxide SNF	Non-metals	INTACT	C-70	D-70
DRCT	756	08. U Oxide SNF	Non-metals	INTACT	C-71	D-71
DRESDEN I (E00161)	928	08. U Oxide SNF	Non-metals	INTACT	C-72	D-72
DRESDEN I (UN0064)	47	08. U Oxide SNF	Non-metals	INTACT	C-73	D-73
DRESII, HBR, BR-3, BRP, TMI	50	08. U Oxide SNF	Non-metals	NOT INTACT	C-74	D-74
EBR-II NITRIDE FUEL EXPER	363	10. Misc. SNF	Other	INTACT	C-75	D-75
EBR-II OXIDE FUEL EXPER	364	04. MOX SNF	Non-metals	INTACT	C-76	D-76
EBR-II OXIDE FUEL EXPER	345	04. MOX SNF	Non-metals	INTACT	C-77	D-77
FAST REACTOR FUEL	906	06. Th/U Oxide SNF	Non-metals	NOT INTACT	C-78	D-78
FAST REACTOR FUEL (U/PUC)	1029	03. Pu/U Carbide SNF	Non-metals	NOT INTACT	C-79	D-79
FERMI CORE I & 2 (CORE FOIL)	457	02. Pu/U Alloy SNF	Other	INTACT	C-80	D-80
FERMI CORE I & 2 (CORE SHIM)	69	02. Pu/U Alloy SNF	Other	INTACT	C-81	D-81
FERMI CORE I & 2 (DECLAD)	453	02. Pu/U Alloy SNF	Other	NOT INTACT	C-82	D-82
FERMI CORE I & 2 (SECTIONED)	454	02. Pu/U Alloy SNF	Other	INTACT	C-83	D-83
FERMI CORE I & 2 (SODIUM WORTH)	455	02. Pu/U Alloy SNF	Other	INTACT	C-84	D-84
FERMI CORE I & 2 (STD FUEL SUBASSEMBLY)	456	02. Pu/U Alloy SNF	Other	INTACT	C-85	D-85
FFTF CARBIDE FUEL EXPER.	347	03. Pu/U Carbide SNF	Non-metals	INTACT	C-86	D-86
FFTF OXIDE EXPERIMENTS	349	04. MOX SNF	Non-metals	INTACT	C-87	D-87
FSVR	86	05. Th/U Carbide SNF	Non-metals	INTACT	C-88	D-88
FSVR	85	05. Th/U Carbide SNF	Non-metals	INTACT	C-89	D-89
GA HTGR FUEL	89	05. Th/U Carbide SNF	Non-metals	NOT INTACT	C-90	D-90
GA RERTR	90	11. U Zr H SNF	Non-metals	NOT INTACT	C-91	D-91
GCRE CAN (1B-8T 1&2)	94	08. U Oxide SNF	Non-metals	NOT INTACT	C-92	D-92
GCRE PELLETS (1B-7T-1)	95	08. U Oxide SNF	Non-metals	NOT INTACT	C-93	D-93
GETR FILTERS	98	08. U Oxide SNF	Non-metals	NOT INTACT	C-94	D-94

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H. B. ROBINSON (ASSEMBLY)	383	08. U Oxide SNF	Non-metals	INTACT	C-95	D-95
H. B. ROBINSON RODS	864	08. U Oxide SNF	Non-metals	NOT INTACT	C-96	D-96
HFBR	102	09. Alum Based SNF	Stable	INTACT	C-97	D-97
HFBR	961	09. Alum Based SNF	Stable	INTACT	C-98	D-98
HFEF FISSION CHAMBERS (U METAL)	894	07. U Metal SNF	Other	NOT INTACT	C-99	D-99
HTGR (PEACH BOTTOM SCRAP)	935	05. Th/U Carbide SNF	Non-metals	INTACT	C-100	D-100
KEMA	861	06. Th/U Oxide SNF	Non-metals	NOT INTACT	C-101	D-101
LOFT CENTER FUEL MODULE (A1,A2,A3,F1)	127	08. U Oxide SNF	Non-metals	INTACT	C-102	D-102
LOFT CENTER FUEL MODULE (FP-1)	1061	08. U Oxide SNF	Non-metals	INTACT	C-103	D-103
LOFT CENTER FUEL MODULE FP-2 REMAINS	923	08. U Oxide SNF	Non-metals	NOT INTACT	C-104	D-104
LOFT CORNER FUEL MODULE	128	08. U Oxide SNF	Non-metals	INTACT	C-105	D-105
LOFT FUEL RODS	924	08. U Oxide SNF	Non-metals	NOT INTACT	C-106	D-106
LOFT SQUARE FUEL MODULE	129	08. U Oxide SNF	Non-metals	INTACT	C-107	D-107
LOOSE FUEL ROD STORAGE BASKET (LFRSB)	126	08. U Oxide SNF	Non-metals	NOT INTACT	C-108	D-108
LWR SNF SCRAP (ZR/SST)	940	08. U Oxide SNF	Non-metals	NOT INTACT	C-109	D-109
MISCELLANEOUS RSWF FUEL	366	10. Misc. SNF	Other	INTACT	C-110	D-110
MISCELLANEOUS TREAT FUEL (MOX)	369	04. MOX SNF	Non-metals	NOT INTACT	C-111	D-111
MISCELLANEOUS TREAT FUEL (U-METAL)	905	07. U Metal SNF	Other	NOT INTACT	C-112	D-112
MTR CANAL SCRAP	1062	08. U Oxide SNF	Non-metals	NOT INTACT	C-113	D-113
MURR (UALX) COLUMBIA	142	09. Alum Based SNF	Stable	INTACT	C-114	D-114
MURR (UALX) COLUMBIA	962	09. Alum Based SNF	Stable	INTACT	C-115	D-115
OCONEE	156	08. U Oxide SNF	Non-metals	INTACT	C-116	D-116
ORR	461	09. Alum Based SNF	Stable	INTACT	C-117	D-117
PATHFINDER (SUPERHEATER)	166	08. U Oxide SNF	Non-metals	INTACT	C-118	D-118
PATHFINDER (SUPERHEATER)	814	08. U Oxide SNF	Non-metals	INTACT	C-119	D-119
PBF DRIVER CORE	167	08. U Oxide SNF	Non-metals	INTACT	C-120	D-120
PEACH BOTTOM (ASSEMBLY)	385	08. U Oxide SNF	Non-metals	INTACT	C-121	D-121
PEACH BOTTOM RODS	386	08. U Oxide SNF	Non-metals	NOT INTACT	C-122	D-122
PEACH BOTTOM UNIT I CORE I	169	05. Th/U Carbide SNF	Non-metals	INTACT	C-123	D-123
PEACH BOTTOM UNIT I CORE I	170	05. Th/U Carbide SNF	Non-metals	INTACT	C-124	D-124
PEACH BOTTOM UNIT I CORE II	171	05. Th/U Carbide SNF	Non-metals	INTACT	C-125	D-125
PEACH BOTTOM UNIT I CORE II (INTACT)	206	05. Th/U Carbide SNF	Non-metals	INTACT	C-126	D-126
PNL MIXED MATERIAL EXP.DCC-1	430	08. U Oxide SNF	Non-metals	NOT INTACT	C-127	D-127

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PNL MIXED MATERIAL EXP.DCC-2	431	08. U Oxide SNF	Non-metals	NOT INTACT	C-128	D-128
PNL MIXED MATERIAL EXP.DCC-3	432	08. U Oxide SNF	Non-metals	NOT INTACT	C-129	D-129
PNL MOX FUEL	414	04. MOX SNF	Non-metals	NOT INTACT	C-130	D-130
PNL MOX FUEL	415	04. MOX SNF	Non-metals	NOT INTACT	C-131	D-131
PNL MOX FUEL 7055	416	04. MOX SNF	Non-metals	NOT INTACT	C-132	D-132
PNL MOX FUEL 7057	417	04. MOX SNF	Non-metals	NOT INTACT	C-133	D-133
PNL MOX PELLETS 7057	418	04. MOX SNF	Non-metals	NOT INTACT	C-134	D-134
PNL MOX PINS 7057	419	04. MOX SNF	Non-metals	NOT INTACT	C-135	D-135
PNL MOX STAR 3	433	04. MOX SNF	Non-metals	NOT INTACT	C-136	D-136
PNL MOX STAR 4	434	04. MOX SNF	Non-metals	NOT INTACT	C-137	D-137
PNL MOX STAR 5	435	04. MOX SNF	Non-metals	NOT INTACT	C-138	D-138
PNL MOX STAR 6	436	04. MOX SNF	Non-metals	NOT INTACT	C-139	D-139
PNL MOX STAR 7	422	04. MOX SNF	Non-metals	INTACT	C-140	D-140
PNL-3	420	04. MOX SNF	Non-metals	INTACT	C-141	D-141
PULSTAR - BUFFALO (6%RODS)	174	08. U Oxide SNF	Non-metals	INTACT	C-142	D-142
PULSTAR-N.C. STATE UNIV. (4% ASSEMBLIES)	175	08. U Oxide SNF	Non-metals	INTACT	C-143	D-143
PULSTAR-SUNY-BUFFALO (6% RODS)	176	08. U Oxide SNF	Non-metals	INTACT	C-144	D-144
RESIDUE FAILED PBF RODS	381	08. U Oxide SNF	Non-metals	NOT INTACT	C-145	D-145
ROBERT E. GINNA	182	08. U Oxide SNF	Non-metals	INTACT	C-146	D-146
ROVER (UBM)	840	03. Pu/U Carbide SNF	Non-metals	NOT INTACT	C-147	D-147
SHIPPINGPORT LWBR BLKT I	374	06. Th/U Oxide SNF	Non-metals	INTACT	C-148	D-148
SHIPPINGPORT LWBR BLKT II	375	06. Th/U Oxide SNF	Non-metals	INTACT	C-149	D-149
SHIPPINGPORT LWBR BLKT III	376	06. Th/U Oxide SNF	Non-metals	INTACT	C-150	D-150
SHIPPINGPORT LWBR REFLCT. IV	371	06. Th/U Oxide SNF	Non-metals	INTACT	C-151	D-151
SHIPPINGPORT LWBR REFLCT. V	372	06. Th/U Oxide SNF	Non-metals	INTACT	C-152	D-152
SHIPPINGPORT LWBR SCRAP	377	06. Th/U Oxide SNF	Non-metals	NOT INTACT	C-153	D-153
SHIPPINGPORT LWBR SCRAP (LINER 15718)	379	06. Th/U Oxide SNF	Non-metals	NOT INTACT	C-154	D-154
SHIPPINGPORT LWBR SEED	380	06. Th/U Oxide SNF	Non-metals	INTACT	C-155	D-155
SHIPPINGPORT PWR C1 BLKT	191	08. U Oxide SNF	Non-metals	INTACT	C-156	D-156
SHIPPINGPORT PWR C2 BLKT	192	08. U Oxide SNF	Non-metals	INTACT	C-157	D-157
SHIPPINGPORT PWR-C1-S4	194	02. Pu/U Alloy SNF	Other	INTACT	C-158	D-158
SHIPPINGPORT PWR-C2-S1	195	08. U Oxide SNF	Non-metals	INTACT	C-159	D-159
SHIPPINGPORT PWR-C2-S2	196	08. U Oxide SNF	Non-metals	INTACT	C-160	D-160

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SM-1A	201	08. U Oxide SNF	Non-metals	NOT INTACT	C-161	D-161
SNAP	203	11. U Zr H SNF	Non-metals	NOT INTACT	C-162	D-162
SODIUM LOOP SAFETY FAC.	352	04. MOX SNF	Non-metals	NOT INTACT	C-163	D-163
SODIUM LOOP SAFETY FAC.	367	04. MOX SNF	Non-metals	NOT INTACT	C-164	D-164
SPEC (ORME)	208	02. Pu/U Alloy SNF	Other	NOT INTACT	C-165	D-165
SPSS (SPERT)	213	08. U Oxide SNF	Non-metals	NOT INTACT	C-166	D-166
TMI-2	228	08. U Oxide SNF	Non-metals	NOT INTACT	C-167	D-167
TMI-2 CORE DEBRIS	914	08. U Oxide SNF	Non-metals	NOT INTACT	C-168	D-168
TMI-2 CORE DEBRIS (D-153 & 388)	229	08. U Oxide SNF	Non-metals	NOT INTACT	C-169	D-169
TORY-IIA	230	08. U Oxide SNF	Non-metals	NOT INTACT	C-170	D-170
TORY-IIC	231	08. U Oxide SNF	Non-metals	NOT INTACT	C-171	D-171
TREAT DRIVER	232	08. U Oxide SNF	Non-metals	INTACT	C-172	D-172
TRIGA 8.5/20 FFCR OSU	1039	11. U Zr H SNF	Non-metals	INTACT	C-173	D-173
TRIGA 8.5/20 FFCR UNIV. OF CAL-IRVINE	1050	11. U Zr H SNF	Non-metals	INTACT	C-174	D-174
TRIGA 8.5/20 FFCR UNIV. OF CAL-IRVINE	1052	11. U Zr H SNF	Non-metals	INTACT	C-175	D-175
TRIGA (ACPR 12/20) JAPAN	480	11. U Zr H SNF	Non-metals	INTACT	C-176	D-176
TRIGA (ACPR 12/20) PENN. STATE UNIV.	1002	11. U Zr H SNF	Non-metals	INTACT	C-177	D-177
TRIGA (ACPR 12/20) SLOVENIA	932	11. U Zr H SNF	Non-metals	INTACT	C-178	D-178
TRIGA (ACPR) ROMANIA	1077	11. U Zr H SNF	Non-metals	INTACT	C-179	D-179
TRIGA (DEMOUNTABLE) U OF AZ	971	11. U Zr H SNF	Non-metals	INTACT	C-180	D-180
TRIGA (FLIP LEU 45/20) (DAMAGED) SO. KOREA	819	11. U Zr H SNF	Non-metals	INTACT	C-181	D-181
TRIGA (FLIP LEU-I 20/20) MALAYSIA	497	11. U Zr H SNF	Non-metals	INTACT	C-182	D-182
TRIGA (FLIP LEU-I 20/20) THAILAND	496	11. U Zr H SNF	Non-metals	INTACT	C-183	D-183
TRIGA (FLIP LEU-I) BANGLADESH	470	11. U Zr H SNF	Non-metals	INTACT	C-184	D-184
TRIGA (FLIP LEU-II 20/30) PHILIPPINES	499	11. U Zr H SNF	Non-metals	INTACT	C-185	D-185
TRIGA (FLIP LEU-II 20/30) TAIWAN	498	11. U Zr H SNF	Non-metals	INTACT	C-186	D-186
TRIGA (FLIP)	729	11. U Zr H SNF	Non-metals	INTACT	C-187	D-187
TRIGA (FLIP) ANL-W	354	11. U Zr H SNF	Non-metals	INTACT	C-188	D-188
TRIGA (FLIP) ANL-W (NRAD)	884	11. U Zr H SNF	Non-metals	INTACT	C-189	D-189
TRIGA (FLIP) AUSTRIA	492	11. U Zr H SNF	Non-metals	INTACT	C-190	D-190
TRIGA (FLIP) FFCR	996	11. U Zr H SNF	Non-metals	INTACT	C-191	D-191
TRIGA (FLIP) FFCR OSU	702	11. U Zr H SNF	Non-metals	INTACT	C-192	D-192
TRIGA (FLIP) FFCR SO. KOREA	733	11. U Zr H SNF	Non-metals	INTACT	C-193	D-193

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TRIGA (FLIP) MEXICO	493	11. U Zr H SNF	Non-metals	INTACT	C-194	D-194
TRIGA (FLIP) OSU	240	11. U Zr H SNF	Non-metals	INTACT	C-195	D-195
TRIGA (FLIP) SLOVENIA	495	11. U Zr H SNF	Non-metals	INTACT	C-196	D-196
TRIGA (FLIP) SO. KOREA	494	11. U Zr H SNF	Non-metals	INTACT	C-197	D-197
TRIGA (FLIP) TEXAS A&M	239	11. U Zr H SNF	Non-metals	INTACT	C-198	D-198
TRIGA (FLIP) TEXAS A&M	241	11. U Zr H SNF	Non-metals	INTACT	C-199	D-199
TRIGA (FLIP) TEXAS A&M - DAMAGED	844	11. U Zr H SNF	Non-metals	INTACT	C-200	D-200
TRIGA (FLIP) U OF WI	1035	11. U Zr H SNF	Non-metals	INTACT	C-201	D-201
TRIGA (FLIP) UNIV OF WISCONSIN	242	11. U Zr H SNF	Non-metals	INTACT	C-202	D-202
TRIGA (FLIP) WSU	243	11. U Zr H SNF	Non-metals	INTACT	C-203	D-203
TRIGA (HIGH POWER) (HEU)	998	11. U Zr H SNF	Non-metals	INTACT	C-204	D-204
TRIGA (HIGH POWER) ROMANIA	302	11. U Zr H SNF	Non-metals	INTACT	C-205	D-205
TRIGA (HIGH POWER) ROMANIA	930	11. U Zr H SNF	Non-metals	INTACT	C-206	D-206
TRIGA 20/20 FFCR MNRC	737	11. U Zr H SNF	Non-metals	INTACT	C-207	D-207
TRIGA 30/20 FFCR MNRC	1055	11. U Zr H SNF	Non-metals	INTACT	C-208	D-208
TRIGA 8.5/20 FFCR	1003	11. U Zr H SNF	Non-metals	INTACT	C-209	D-209
TRIGA 8.5/20 FFCR AFRRI	969	11. U Zr H SNF	Non-metals	INTACT	C-210	D-210
TRIGA 8.5/20 FFCR ENGLAND	987	11. U Zr H SNF	Non-metals	INTACT	C-211	D-211
TRIGA 8.5/20 FFCR HEIDELBERG	1045	11. U Zr H SNF	Non-metals	INTACT	C-212	D-212
TRIGA 8.5/20 FFCR ITALY	730	11. U Zr H SNF	Non-metals	INTACT	C-213	D-213
TRIGA 8.5/20 FFCR MNRC	703	11. U Zr H SNF	Non-metals	INTACT	C-214	D-214
TRIGA 8.5/20 FFCR OSU	1041	11. U Zr H SNF	Non-metals	INTACT	C-215	D-215
TRIGA 8.5/20 FFCR PENN. STATE UNIV.	815	11. U Zr H SNF	Non-metals	INTACT	C-216	D-216
TRIGA 8.5/20 FFCR SLOVENIA	941	11. U Zr H SNF	Non-metals	INTACT	C-217	D-217
TRIGA 8.5/20 FFCR SO. KOREA	734	11. U Zr H SNF	Non-metals	INTACT	C-218	D-218
TRIGA 8.5/20 FFCR U OF AZ	974	11. U Zr H SNF	Non-metals	INTACT	C-219	D-219
TRIGA 8.5/20 FFCR U OF IL	448	11. U Zr H SNF	Non-metals	INTACT	C-220	D-220
TRIGA 8.5/20 FFCR U OF TX AUSTIN	825	11. U Zr H SNF	Non-metals	INTACT	C-221	D-221
TRIGA 8.5/20 FFCR ZAIRE	735	11. U Zr H SNF	Non-metals	INTACT	C-222	D-222
TRIGA STD (ACPR)	895	11. U Zr H SNF	Non-metals	INTACT	C-223	D-223
TRIGA STD (ALUM) ARRR	238	11. U Zr H SNF	Non-metals	INTACT	C-224	D-224
TRIGA STD (ALUM) AUSTRIA	462	11. U Zr H SNF	Non-metals	INTACT	C-225	D-225
TRIGA STD (ALUM) BRAZIL	471	11. U Zr H SNF	Non-metals	INTACT	C-226	D-226

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TRIGA STD (ALUM) CORNELL	1047	11. U Zr H SNF	Non-metals	INTACT	C-227	D-227
TRIGA STD (ALUM) CORNELL UNIV.	235	11. U Zr H SNF	Non-metals	INTACT	C-228	D-228
TRIGA STD (ALUM) DOW	970	11. U Zr H SNF	Non-metals	INTACT	C-229	D-229
TRIGA STD (ALUM) FINLAND	463	11. U Zr H SNF	Non-metals	INTACT	C-230	D-230
TRIGA STD (ALUM) GA	728	11. U Zr H SNF	Non-metals	INTACT	C-231	D-231
TRIGA STD (ALUM) GA	870	11. U Zr H SNF	Non-metals	INTACT	C-232	D-232
TRIGA STD (ALUM) GERMANY	465	11. U Zr H SNF	Non-metals	INTACT	C-233	D-233
TRIGA STD (ALUM) HANFORD	876	11. U Zr H SNF	Non-metals	INTACT	C-234	D-234
TRIGA STD (ALUM) HANNOVER	303	11. U Zr H SNF	Non-metals	INTACT	C-235	D-235
TRIGA STD (ALUM) HEIDELBERG	464	11. U Zr H SNF	Non-metals	INTACT	C-236	D-236
TRIGA STD (ALUM) ITALY	466	11. U Zr H SNF	Non-metals	INTACT	C-237	D-237
TRIGA STD (ALUM) ITALY	467	11. U Zr H SNF	Non-metals	INTACT	C-238	D-238
TRIGA STD (ALUM) JAPAN	481	11. U Zr H SNF	Non-metals	INTACT	C-239	D-239
TRIGA STD (ALUM) KANSAS STATE UNIV	804	11. U Zr H SNF	Non-metals	INTACT	C-240	D-240
TRIGA STD (ALUM) KSU	871	11. U Zr H SNF	Non-metals	INTACT	C-241	D-241
TRIGA STD (ALUM) MSU	878	11. U Zr H SNF	Non-metals	INTACT	C-242	D-242
TRIGA STD (ALUM) REED COLLEGE	256	11. U Zr H SNF	Non-metals	INTACT	C-243	D-243
TRIGA STD (ALUM) SLOVENIA	468	11. U Zr H SNF	Non-metals	INTACT	C-244	D-244
TRIGA STD (ALUM) SO. KOREA	483	11. U Zr H SNF	Non-metals	INTACT	C-245	D-245
TRIGA STD (ALUM) U OF IL	447	11. U Zr H SNF	Non-metals	INTACT	C-246	D-246
TRIGA STD (ALUM) U OF IL	501	11. U Zr H SNF	Non-metals	INTACT	C-247	D-247
TRIGA STD (ALUM) U OF UTAH	699	11. U Zr H SNF	Non-metals	INTACT	C-248	D-248
TRIGA STD (ALUM) UNIV. OF TEXAS	877	11. U Zr H SNF	Non-metals	INTACT	C-249	D-249
TRIGA STD (ALUM) USGS	267	11. U Zr H SNF	Non-metals	INTACT	C-250	D-250
TRIGA STD (ALUM) ZAIRE	487	11. U Zr H SNF	Non-metals	INTACT	C-251	D-251
TRIGA STD 12/20 ROMANIA	1078	11. U Zr H SNF	Non-metals	INTACT	C-252	D-252
TRIGA STD 20/20 (IFE) ENGLAND	1043	11. U Zr H SNF	Non-metals	INTACT	C-253	D-253
TRIGA STD 20/20 ARRR	780	11. U Zr H SNF	Non-metals	INTACT	C-254	D-254
TRIGA STD 20/20 MNRC	1053	11. U Zr H SNF	Non-metals	INTACT	C-255	D-255
TRIGA STD 20/20 MNRC	1054	11. U Zr H SNF	Non-metals	INTACT	C-256	D-256
TRIGA STD 20/20 SOLVENIA	731	11. U Zr H SNF	Non-metals	INTACT	C-257	D-257
TRIGA STD 30/20	995	11. U Zr H SNF	Non-metals	INTACT	C-258	D-258
TRIGA STD 30/20 MNRC	704	11. U Zr H SNF	Non-metals	INTACT	C-259	D-259

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Fuel Name	SNF ID	TSPA Category	DBE Category		2010	2030
TRIGA STD 8.5/20	252	11. U Zr H SNF	Non-metals	INTACT	C-260	D-260
TRIGA STD 8.5/20 (IFE) ITALY	929	11. U Zr H SNF	Non-metals	INTACT	C-261	D-261
TRIGA STD 8.5/20 (IFE) OSU	1040	11. U Zr H SNF	Non-metals	INTACT	C-262	D-262
TRIGA STD 8.5/20 (IFE) U OF AZ	972	11. U Zr H SNF	Non-metals	INTACT	C-263	D-263
TRIGA STD 8.5/20 (IFE) U OF AZ	973	11. U Zr H SNF	Non-metals	INTACT	C-264	D-264
TRIGA STD 8.5/20 (IFE) U OF IL	1048	11. U Zr H SNF	Non-metals	INTACT	C-265	D-265
TRIGA STD 8.5/20 (IFE) UNIV. OF CAL-IRVINE	824	11. U Zr H SNF	Non-metals	INTACT	C-266	D-266
TRIGA STD 8.5/20 (IFE) UNIV. OF CAL-IRVINE	1051	11. U Zr H SNF	Non-metals	INTACT	C-267	D-267
TRIGA STD 8.5/20 AFRRI	250	11. U Zr H SNF	Non-metals	INTACT	C-268	D-268
TRIGA STD 8.5/20 ANL-W	353	11. U Zr H SNF	Non-metals	INTACT	C-269	D-269
TRIGA STD 8.5/20 ANL-W	370	11. U Zr H SNF	Non-metals	INTACT	C-270	D-270
TRIGA STD 8.5/20 AUSTRIA	469	11. U Zr H SNF	Non-metals	INTACT	C-271	D-271
TRIGA STD 8.5/20 BRAZIL	1063	11. U Zr H SNF	Non-metals	INTACT	C-272	D-272
TRIGA STD 8.5/20 CORNELL	246	11. U Zr H SNF	Non-metals	INTACT	C-273	D-273
TRIGA STD 8.5/20 DOW	251	11. U Zr H SNF	Non-metals	INTACT	C-274	D-274
TRIGA STD 8.5/20 ENGLAND	485	11. U Zr H SNF	Non-metals	INTACT	C-275	D-275
TRIGA STD 8.5/20 FINLAND	472	11. U Zr H SNF	Non-metals	INTACT	C-276	D-276
TRIGA STD 8.5/20 GA	244	11. U Zr H SNF	Non-metals	NOT INTACT	C-277	D-277
TRIGA STD 8.5/20 GERMANY	305	11. U Zr H SNF	Non-metals	INTACT	C-278	D-278
TRIGA STD 8.5/20 GERMANY	474	11. U Zr H SNF	Non-metals	INTACT	C-279	D-279
TRIGA STD 8.5/20 HANNOVER	473	11. U Zr H SNF	Non-metals	INTACT	C-280	D-280
TRIGA STD 8.5/20 HEIDELBERG	1044	11. U Zr H SNF	Non-metals	INTACT	C-281	D-281
TRIGA STD 8.5/20 INDONESIA	475	11. U Zr H SNF	Non-metals	INTACT	C-282	D-282
TRIGA STD 8.5/20 INDONESIA	476	11. U Zr H SNF	Non-metals	INTACT	C-283	D-283
TRIGA STD 8.5/20 ITALY	477	11. U Zr H SNF	Non-metals	INTACT	C-284	D-284
TRIGA STD 8.5/20 ITALY	478	11. U Zr H SNF	Non-metals	INTACT	C-285	D-285
TRIGA STD 8.5/20 ITALY	1080	11. U Zr H SNF	Non-metals	INTACT	C-286	D-286
TRIGA STD 8.5/20 JAPAN	479	11. U Zr H SNF	Non-metals	INTACT	C-287	D-287
TRIGA STD 8.5/20 KANSAS STATE UNIV	253	11. U Zr H SNF	Non-metals	INTACT	C-288	D-288
TRIGA STD 8.5/20 MEXICO	482	11. U Zr H SNF	Non-metals	INTACT	C-289	D-289
TRIGA STD 8.5/20 MNRC	254	11. U Zr H SNF	Non-metals	INTACT	C-290	D-290
TRIGA STD 8.5/20 MSU	873	11. U Zr H SNF	Non-metals	NOT INTACT	C-291	D-291
TRIGA STD 8.5/20 PENN. STATE UNIV.	237	11. U Zr H SNF	Non-metals	INTACT	C-292	D-292

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TRIGA STD 8.5/20 REED COLLEGE	775	11. U Zr H SNF	Non-metals	INTACT	C-293	D-293
TRIGA STD 8.5/20 SLOVENIA	488	11. U Zr H SNF	Non-metals	INTACT	C-294	D-294
TRIGA STD 8.5/20 SLOVENIA	1079	11. U Zr H SNF	Non-metals	INTACT	C-295	D-295
TRIGA STD 8.5/20 SO. KOREA	484	11. U Zr H SNF	Non-metals	INTACT	C-296	D-296
TRIGA STD 8.5/20 TEXAS A&M	258	11. U Zr H SNF	Non-metals	INTACT	C-297	D-297
TRIGA STD 8.5/20 THAILAND	489	11. U Zr H SNF	Non-metals	INTACT	C-298	D-298
TRIGA STD 8.5/20 TURKEY	490	11. U Zr H SNF	Non-metals	INTACT	C-299	D-299
TRIGA STD 8.5/20 U OF AZ	59	11. U Zr H SNF	Non-metals	INTACT	C-300	D-300
TRIGA STD 8.5/20 U OF AZ	975	11. U Zr H SNF	Non-metals	INTACT	C-301	D-301
TRIGA STD 8.5/20 U OF IL	449	11. U Zr H SNF	Non-metals	INTACT	C-302	D-302
TRIGA STD 8.5/20 U OF TX AUSTIN	265	11. U Zr H SNF	Non-metals	INTACT	C-303	D-303
TRIGA STD 8.5/20 U OF UTAH	261	11. U Zr H SNF	Non-metals	INTACT	C-304	D-304
TRIGA STD 8.5/20 UC @ Berkeley	874	11. U Zr H SNF	Non-metals	NOT INTACT	C-305	D-305
TRIGA STD 8.5/20 UNIV OF MARYLAND	260	11. U Zr H SNF	Non-metals	INTACT	C-306	D-306
TRIGA STD 8.5/20 UNIV. OF CAL-IRVINE	264	11. U Zr H SNF	Non-metals	INTACT	C-307	D-307
TRIGA STD 8.5/20 UNIV. OF WISCONSIN	262	11. U Zr H SNF	Non-metals	INTACT	C-308	D-308
TRIGA STD 8.5/20 USGS	964	11. U Zr H SNF	Non-metals	INTACT	C-309	D-309
TRIGA STD 8.5/20 WSU	268	11. U Zr H SNF	Non-metals	INTACT	C-310	D-310
TRIGA STD 8.5/20 ZAIRE	486	11. U Zr H SNF	Non-metals	INTACT	C-311	D-311
TRU SCRAP SNF (MOX)	368	04. MOX SNF	Non-metals	NOT INTACT	C-312	D-312
TRU SCRAP SNF (U METAL)	904	07. U Metal SNF	Other	NOT INTACT	C-313	D-313
TURKEY POINT	271	08. U Oxide SNF	Non-metals	INTACT	C-314	D-314
US/UK FUEL PINS	356	04. MOX SNF	Non-metals	NOT INTACT	C-315	D-315
VBWR (GENEVA)	285	08. U Oxide SNF	Non-metals	INTACT	C-316	D-316
VEPCO	286	08. U Oxide SNF	Non-metals	INTACT	C-317	D-317
VEPCO	700	08. U Oxide SNF	Non-metals	INTACT	C-318	D-318
VEPCO (T-11 ASSEMBLY)	993	08. U Oxide SNF	Non-metals	INTACT	C-319	D-319
VEPCO (T-11 RODS)	1049	08. U Oxide SNF	Non-metals	INTACT	C-320	D-320
VEPCO T-11	994	08. U Oxide SNF	Non-metals	INTACT	C-321	D-321
Savannah River SNF						
ANLJ	5	09. Alum Based SNF	Stable	INTACT	C-322	D-322
ASTRA-(AUSTRIA)(LEU U308)	1058	09. Alum Based SNF	Stable	INTACT	C-323	D-323
ASTRA-(AUSTRIA)(LEU U3SI2)	712	09. Alum Based SNF	Stable	INTACT	C-324	D-324

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ASTRA-AUSTRIA (UALX-HEU)	646	09. Alum Based SNF	Stable	INTACT	C-325	D-325
ASTRA-AUSTRIA (UALX-MEU)	566	09. Alum Based SNF	Stable	INTACT	C-326	D-326
ATSR	17	09. Alum Based SNF	Stable	INTACT	C-327	D-327
BABCOCK & WILCOX SCRAP	18	04. MOX SNF	Non-metals	NOT INTACT	C-328	D-328
BER-II [HMI] (END BOXES) GERMANY	892	09. Alum Based SNF	Stable	INTACT	C-329	D-329
BER-II [HMI] (UALX HEU) GERMANY	758	09. Alum Based SNF	Stable	INTACT	C-330	D-330
BNL MEDICAL RX (BMRR)	21	09. Alum Based SNF	Stable	INTACT	C-331	D-331
BSR	31	09. Alum Based SNF	Stable	INTACT	C-332	D-332
CANDU	979	08. U Oxide SNF	Non-metals	INTACT	C-333	D-333
CVTR FUEL	37	08. U Oxide SNF	Non-metals	INTACT	C-334	D-334
DR-3 (U3O8 LEU)(DENMARK)	1059	09. Alum Based SNF	Stable	INTACT	C-335	D-335
DR-3 (U3SI2 LEU)(DENMARK)	759	09. Alum Based SNF	Stable	INTACT	C-336	D-336
DR-3 (UALX HEU)(DENMARK)	714	09. Alum Based SNF	Stable	INTACT	C-337	D-337
DRESDEN I THO2/UO2 (LEU)	44	06. Th/U Oxide SNF	Non-metals	INTACT	C-338	D-338
DRESDEN UO2 (LEU)	49	08. U Oxide SNF	Non-metals	INTACT	C-339	D-339
EBWR (6% UO2) LEU	65	08. U Oxide SNF	Non-metals	INTACT	C-340	D-340
EBWR (FUEL FOLLOWER) HEU	740	08. U Oxide SNF	Non-metals	NOT INTACT	C-341	D-341
EBWR (MOX)	63	04. MOX SNF	Non-metals	INTACT	C-342	D-342
EBWR (NORMAL UO2)	60	08. U Oxide SNF	Non-metals	INTACT	C-343	D-343
EBWR (SPIKES)	891	08. U Oxide SNF	Non-metals	INTACT	C-344	D-344
EBWR (U METAL) ENRICHED HEAVY	64	07. U Metal SNF	Other	INTACT	C-345	D-345
EBWR (U METAL) ENRICHED THIN	887	07. U Metal SNF	Other	INTACT	C-346	D-346
EBWR (U METAL) ET-11	888	07. U Metal SNF	Other	INTACT	C-347	D-347
EBWR (U METAL) NORMAL HEAVY	889	07. U Metal SNF	Other	INTACT	C-348	D-348
EBWR (U METAL) NORMAL THIN	890	07. U Metal SNF	Other	INTACT	C-349	D-349
ENEA (LEU UALX) SALUGGIA ITALY	760	09. Alum Based SNF	Stable	INTACT	C-350	D-350
ENEA (UALX HEU) SALUGGIA ITALY	574	09. Alum Based SNF	Stable	INTACT	C-351	D-351
EPRI	67	04. MOX SNF	Non-metals	NOT INTACT	C-352	D-352
ERR (ASSEMBLIES)	68	06. Th/U Oxide SNF	Non-metals	INTACT	C-353	D-353
ERR (RODS)	1057	06. Th/U Oxide SNF	Non-metals	INTACT	C-354	D-354
ESSOR (UALX-HEU) ITALY	762	09. Alum Based SNF	Stable	INTACT	C-355	D-355
FMRB (GERMANY)	577	09. Alum Based SNF	Stable	INTACT	C-356	D-356
FRG-1 (U3O8 LEU) GERMANY	581	09. Alum Based SNF	Stable	INTACT	C-357	D-357

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Fuel Name	SNF ID	TSPA Category	DBE Category		2010	2030
FRG-1 (U3SI2 LEU) GERMANY	741	09. Alum Based SNF	Stable	INTACT	C-358	D-358
FRG-1 (UALX HEU) GERMANY	742	09. Alum Based SNF	Stable	INTACT	C-359	D-359
FRJ (UALX-HEU) GERMANY	933	09. Alum Based SNF	Stable	INTACT	C-360	D-360
FRJ (UALX-MEU) GERMANY	1000	09. Alum Based SNF	Stable	INTACT	C-361	D-361
FRJ TUBES (U3O8 LEU) GERMANY	999	09. Alum Based SNF	Stable	INTACT	C-362	D-362
FRM (UALX HEU 45%) GERMANY	805	09. Alum Based SNF	Stable	INTACT	C-363	D-363
FRM (UALX HEU) GERMANY	806	09. Alum Based SNF	Stable	INTACT	C-364	D-364
FRR ASTRA (U308-LEU) AUSTRIA	556	09. Alum Based SNF	Stable	INTACT	C-365	D-365
FRR ASTRA (U3SI2 LEU) AUSTRIA	515	09. Alum Based SNF	Stable	INTACT	C-366	D-366
FRR ASTRA (UALX-HEU) AUSTRIA	654	09. Alum Based SNF	Stable	INTACT	C-367	D-367
FRR ASTRA (UALX-HEU) AUSTRIA	738	09. Alum Based SNF	Stable	INTACT	C-368	D-368
FRR FMRB (GERMANY)	1066	09. Alum Based SNF	Stable	INTACT	C-369	D-369
FRR MTR (UALX HEU) AUSTRALIA	649	09. Alum Based SNF	Stable	INTACT	C-370	D-370
FRR MTR (UALX-HEU) JAPAN	603	09. Alum Based SNF	Stable	INTACT	C-371	D-371
FRR MTR (UALX-HEU) JAPAN	605	09. Alum Based SNF	Stable	INTACT	C-372	D-372
FRR MTR (UALX-HEU) NETHERLANDS	609	09. Alum Based SNF	Stable	INTACT	C-373	D-373
FRR MTR (UALX-HEU) TAIWAN	628	09. Alum Based SNF	Stable	INTACT	C-374	D-374
FRR MTR (UALX-LEU) ARGENTINA	547	09. Alum Based SNF	Stable	INTACT	C-375	D-375
FRR MTR (UALX-LEU) JAPAN	551	09. Alum Based SNF	Stable	INTACT	C-376	D-376
FRR MTR (UALX-LEU) TAIWAN	555	09. Alum Based SNF	Stable	INTACT	C-377	D-377
FRR MTR (UALX-LEU) VENEZUELA	559	09. Alum Based SNF	Stable	INTACT	C-378	D-378
FRR MTR (UALX-MEU) JAPAN	565	09. Alum Based SNF	Stable	INTACT	C-379	D-379
FRR MTR UALX HEU CANADA	294	09. Alum Based SNF	Stable	INTACT	C-380	D-380
FRR MTR-C (U308-LEU) PERU	503	09. Alum Based SNF	Stable	INTACT	C-381	D-381
FRR MTR-C (U3SI2 LEU) CANADA	512	09. Alum Based SNF	Stable	INTACT	C-382	D-382
FRR MTR-C (U3SI2 LEU) GERMANY	517	09. Alum Based SNF	Stable	INTACT	C-383	D-383
FRR MTR-C (U3SI2 LEU) GREECE	531	09. Alum Based SNF	Stable	INTACT	C-384	D-384
FRR MTR-C (U3SI2 LEU) JAPAN	289	09. Alum Based SNF	Stable	INTACT	C-385	D-385
FRR MTR-C (U3SI2 LEU) NETHERLANDS	509	09. Alum Based SNF	Stable	INTACT	C-386	D-386
FRR MTR-C (UALX LEU) SWEDEN	523	09. Alum Based SNF	Stable	INTACT	C-387	D-387
FRR MTR-C (UALX-HEU) ARGENTINA	635	09. Alum Based SNF	Stable	INTACT	C-388	D-388
FRR MTR-C (UALX-HEU) CANADA	612	09. Alum Based SNF	Stable	INTACT	C-389	D-389
FRR MTR-C (UALX-HEU) GERMANY	579	09. Alum Based SNF	Stable	INTACT	C-390	D-390

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FRR MTR-C (UALX-HEU) JAPAN	600	09. Alum Based SNF	Stable	INTACT	C-391	D-391
FRR MTR-C (UALX-HEU) PORTUGAL	631	09. Alum Based SNF	Stable	INTACT	C-392	D-392
FRR MTR-C (UALX-HEU) TURKEY	643	09. Alum Based SNF	Stable	INTACT	C-393	D-393
FRR MTR-C (UALX-LEU) JAPAN	552	09. Alum Based SNF	Stable	INTACT	C-394	D-394
FRR MTR-C (UALX-LEU) PORTUGAL	540	09. Alum Based SNF	Stable	INTACT	C-395	D-395
FRR MTR-C1 (UALX-HEU) SWITZERLAND	656	09. Alum Based SNF	Stable	INTACT	C-396	D-396
FRR MTR-C2 (U3SI2 LEU) TURKEY	527	09. Alum Based SNF	Stable	INTACT	C-397	D-397
FRR MTR-C2 (UALX-HEU) SWITZERLAND	657	09. Alum Based SNF	Stable	INTACT	C-398	D-398
FRR MTR-O (UALX-HEU) TURKEY	642	09. Alum Based SNF	Stable	INTACT	C-399	D-399
FRR MTR-O (UALX-LEU) PORTUGAL	541	09. Alum Based SNF	Stable	INTACT	C-400	D-400
FRR MTR-S (U308-LEU) INDONESIA	502	09. Alum Based SNF	Stable	INTACT	C-401	D-401
FRR MTR-S (U308-LEU) PERU	504	09. Alum Based SNF	Stable	INTACT	C-402	D-402
FRR MTR-S (U3SI2 LEU) CANADA	513	09. Alum Based SNF	Stable	INTACT	C-403	D-403
FRR MTR-S (U3SI2 LEU) GERMANY	519	09. Alum Based SNF	Stable	INTACT	C-404	D-404
FRR MTR-S (U3SI2 LEU) GERMANY	1067	09. Alum Based SNF	Stable	INTACT	C-405	D-405
FRR MTR-S (U3SI2 LEU) GREECE	532	09. Alum Based SNF	Stable	INTACT	C-406	D-406
FRR MTR-S (U3SI2 LEU) JAPAN	506	09. Alum Based SNF	Stable	INTACT	C-407	D-407
FRR MTR-S (U3SI2 LEU) JAPAN	508	09. Alum Based SNF	Stable	INTACT	C-408	D-408
FRR MTR-S (U3SI2 LEU) NETHERLANDS	510	09. Alum Based SNF	Stable	INTACT	C-409	D-409
FRR MTR-S (U3SI2 LEU) TURKEY	528	09. Alum Based SNF	Stable	INTACT	C-410	D-410
FRR MTR-S (UALX-HEU) CANADA	720	09. Alum Based SNF	Stable	INTACT	C-411	D-411
FRR MTR-S (UALX-HEU) GERMANY	582	09. Alum Based SNF	Stable	INTACT	C-412	D-412
FRR MTR-S (UALX-HEU) GERMANY	584	09. Alum Based SNF	Stable	INTACT	C-413	D-413
FRR MTR-S (UALX-HEU) GERMANY	585	09. Alum Based SNF	Stable	INTACT	C-414	D-414
FRR MTR-S (UALX-HEU) GERMANY	588	09. Alum Based SNF	Stable	INTACT	C-415	D-415
FRR MTR-S (UALX-HEU) JAPAN	602	09. Alum Based SNF	Stable	INTACT	C-416	D-416
FRR MTR-S (UALX-HEU) NETHERLANDS	607	09. Alum Based SNF	Stable	INTACT	C-417	D-417
FRR MTR-S (UALX-HEU) NETHERLANDS	608	09. Alum Based SNF	Stable	INTACT	C-418	D-418
FRR MTR-S (UALX-HEU) PORTUGAL	632	09. Alum Based SNF	Stable	INTACT	C-419	D-419
FRR MTR-S (UALX-HEU) SWITZERLAND	658	09. Alum Based SNF	Stable	INTACT	C-420	D-420
FRR MTR-S (UALX-HEU) TURKEY	644	09. Alum Based SNF	Stable	INTACT	C-421	D-421
FRR MTR-S (UALX-LEU) JAPAN	553	09. Alum Based SNF	Stable	INTACT	C-422	D-422
FRR MTR-S (UALX-LEU) PORTUGAL	542	09. Alum Based SNF	Stable	INTACT	C-423	D-423

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FRR MTR-S (UALX-MEU) GERMANY	1068	09. Alum Based SNF	Stable	INTACT	C-424	D-424
FRR PIN CLUSTER U3SI2-LEU CANADA	660	09. Alum Based SNF	Stable	INTACT	C-425	D-425
FRR PIN CLUSTER U3SI2-LEU SO. KOREA	293	09. Alum Based SNF	Stable	INTACT	C-426	D-426
FRR PIN CLUSTER U3SI2-LEU SO. KOREA	659	09. Alum Based SNF	Stable	INTACT	C-427	D-427
FRR PIN CLUSTER UALX HEU CANADA	661	09. Alum Based SNF	Stable	INTACT	C-428	D-428
FRR PIN CLUSTER UALX HEU CANADA	662	09. Alum Based SNF	Stable	INTACT	C-429	D-429
FRR PIN CLUSTER UALX HEU CANADA	663	09. Alum Based SNF	Stable	INTACT	C-430	D-430
FRR SLOWPOKE (HEU) CANADA	665	09. Alum Based SNF	Stable	INTACT	C-431	D-431
FRR SLOWPOKE (HEU) CANADA	666	09. Alum Based SNF	Stable	INTACT	C-432	D-432
FRR SLOWPOKE (HEU) CANADA	668	09. Alum Based SNF	Stable	INTACT	C-433	D-433
FRR SLOWPOKE (HEU) CANADA	669	09. Alum Based SNF	Stable	INTACT	C-434	D-434
FRR SLOWPOKE (HEU) MONTREAL	667	09. Alum Based SNF	Stable	INTACT	C-435	D-435
FRR TARGET ARGENTINA	297	08. U Oxide SNF	Non-metals	INTACT	C-436	D-436
FRR TARGET CANADA	671	08. U Oxide SNF	Non-metals	NOT INTACT	C-437	D-437
FRR TARGET INDONESIA	672	08. U Oxide SNF	Non-metals	NOT INTACT	C-438	D-438
FRR TUBES (U3SI2 LEU) DENMARK	298	09. Alum Based SNF	Stable	INTACT	C-439	D-439
FRR TUBES (U3SI2 LEU) GERMANY	673	09. Alum Based SNF	Stable	INTACT	C-440	D-440
FRR TUBES (U3SI2 LEU) GERMANY	674	09. Alum Based SNF	Stable	INTACT	C-441	D-441
FRR TUBES (U3SI2 LEU) GERMANY	675	09. Alum Based SNF	Stable	INTACT	C-442	D-442
FRR TUBES (UALX LEU) AUSTRALIA	299	09. Alum Based SNF	Stable	INTACT	C-443	D-443
FRR TUBES (UALX-HEU) AUSTRALIA	300	09. Alum Based SNF	Stable	INTACT	C-444	D-444
FRR TUBES (UALX-HEU) AUSTRALIA	684	09. Alum Based SNF	Stable	INTACT	C-445	D-445
FRR TUBES (UALX-HEU) DENMARK	676	09. Alum Based SNF	Stable	INTACT	C-446	D-446
FRR TUBES (UALX-HEU) DENMARK	678	09. Alum Based SNF	Stable	INTACT	C-447	D-447
FRR TUBES (UALX-HEU) GERMANY	683	09. Alum Based SNF	Stable	INTACT	C-448	D-448
FRR TUBES (UALX-HEU) GERMANY	685	09. Alum Based SNF	Stable	INTACT	C-449	D-449
GCRE (1B SERIES)	745	08. U Oxide SNF	Non-metals	INTACT	C-450	D-450
GCRE (1Z SERIES)	916	08. U Oxide SNF	Non-metals	INTACT	C-451	D-451
GENTR	97	09. Alum Based SNF	Stable	INTACT	C-452	D-452
GRR (UALX HEU) GREECE	440	09. Alum Based SNF	Stable	INTACT	C-453	D-453
GRR (UALX HEU) GREECE	1069	09. Alum Based SNF	Stable	INTACT	C-454	D-454
GTRR	87	09. Alum Based SNF	Stable	INTACT	C-455	D-455
H. B. ROBINSON	99	04. MOX SNF	Non-metals	NOT INTACT	C-456	D-456

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Fuel Name	SNF ID	TSPA Category	DBE Category		2010	2030
HFBR	706	09. Alum Based SNF	Stable	INTACT	C-457	D-457
HFIR (INNER)	103	09. Alum Based SNF	Stable	INTACT	C-458	D-458
HFIR (INNER)	1083	09. Alum Based SNF	Stable	INTACT	C-459	D-459
HFIR (OUTER)	707	09. Alum Based SNF	Stable	INTACT	C-460	D-460
HFIR (OUTER)	1084	09. Alum Based SNF	Stable	INTACT	C-461	D-461
HIFAR (UALX-HEU) AUSTRALIA	680	09. Alum Based SNF	Stable	INTACT	C-462	D-462
HOR (NETHERLANDS)	713	09. Alum Based SNF	Stable	INTACT	C-463	D-463
HTRE (ANP)	105	08. U Oxide SNF	Non-metals	NOT INTACT	C-464	D-464
HWCTR 3EMT-2 (UMO)	118	02. Pu/U Alloy SNF	Other	INTACT	C-465	D-465
HWCTR DRIVER (U-ZR) HEU	117	02. Pu/U Alloy SNF	Other	INTACT	C-466	D-466
HWCTR ETWO (U METAL) LEU	867	07. U Metal SNF	Other	INTACT	C-467	D-467
HWCTR IMT (U METAL-SST) DU	113	07. U Metal SNF	Other	INTACT	C-468	D-468
HWCTR IRO (UO2) LEU	976	08. U Oxide SNF	Non-metals	INTACT	C-469	D-469
HWCTR IS (U-ZR) LEU	977	02. Pu/U Alloy SNF	Other	INTACT	C-470	D-470
HWCTR OT (UO2) LEU	283	08. U Oxide SNF	Non-metals	INTACT	C-471	D-471
HWCTR RMT & SMT (U METAL) LEU	790	07. U Metal SNF	Other	INTACT	C-472	D-472
HWCTR SOT (UO2) LEU	120	08. U Oxide SNF	Non-metals	INTACT	C-473	D-473
HWCTR SPR (U-ZR) LEU	783	02. Pu/U Alloy SNF	Other	INTACT	C-474	D-474
HWCTR SPRO (U02) ALUM LEU	115	09. Alum Based SNF	Non-metals	INTACT	C-475	D-475
HWCTR SPRO (U02) SST LEU	978	08. U Oxide SNF	Non-metals	INTACT	C-476	D-476
HWCTR SPRO (UO2) ZR LEU	772	08. U Oxide SNF	Non-metals	INTACT	C-477	D-477
HWCTR TFEN (U-ZR) LEU	880	02. Pu/U Alloy SNF	Other	INTACT	C-478	D-478
HWCTR TMT-1-2 & 1-3 (U/TH)	112	02. Pu/U Alloy SNF	Stable	INTACT	C-479	D-479
HWCTR TWNT (U METAL) LEU	791	07. U Metal SNF	Other	INTACT	C-480	D-480
IAN-R1 (COLUMBIA)	596	09. Alum Based SNF	Stable	INTACT	C-481	D-481
IAN-R1 (COLUMBIA)	803	09. Alum Based SNF	Stable	INTACT	C-482	D-482
IEA-R1 (UALX HEU) BRAZIL	954	09. Alum Based SNF	Stable	INTACT	C-483	D-483
IEA-R1 (UALX LEU) BRAZIL	545	09. Alum Based SNF	Stable	INTACT	C-484	D-484
IEA-R1 (UALX LEU) BRAZIL	1076	09. Alum Based SNF	Stable	INTACT	C-485	D-485
IOWA ST. UNIV. (HEU UALX)	792	09. Alum Based SNF	Stable	INTACT	C-486	D-486
IOWA STATE UNIVERSITY (U3SI2 LEU)	953	09. Alum Based SNF	Stable	INTACT	C-487	D-487
JEN-1 (HEU UALX) SPAIN	795	09. Alum Based SNF	Stable	INTACT	C-488	D-488
JEN-1 (UALX LEU) SPAIN	749	09. Alum Based SNF	Stable	INTACT	C-489	D-489

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Fuel Name	SNF ID	TSPA Category	DBE Category		2010	2030
JMTR	507	09. Alum Based SNF	Stable	INTACT	C-490	D-490
JMTR (UALX 45% MEU) JAPAN	886	09. Alum Based SNF	Stable	INTACT	C-491	D-491
JMTR (UALX HEU) JAPAN	123	09. Alum Based SNF	Stable	INTACT	C-492	D-492
JRR-2 (UALX-HEU 45%) JAPAN	885	09. Alum Based SNF	Stable	INTACT	C-493	D-493
JRR-2 (UALX-HEU) JAPAN	606	09. Alum Based SNF	Stable	INTACT	C-494	D-494
JRR-3M (ULAX LEU)	1056	09. Alum Based SNF	Stable	INTACT	C-495	D-495
JRR-4 (U3SI2 LEU)	1071	09. Alum Based SNF	Stable	INTACT	C-496	D-496
JRR-4 (UALX HEU)	505	09. Alum Based SNF	Stable	INTACT	C-497	D-497
JRR-4 (UALX HEU)	1070	09. Alum Based SNF	Stable	INTACT	C-498	D-498
KURR (UALX-HEU) JAPAN	601	09. Alum Based SNF	Stable	INTACT	C-499	D-499
LWR SAMPLES	134	04. MOX SNF	Non-metals	NOT INTACT	C-500	D-500
MIT	135	09. Alum Based SNF	Stable	INTACT	C-501	D-501
MIT	136	09. Alum Based SNF	Stable	INTACT	C-502	D-502
ML-1 (GCRE)	137	08. U Oxide SNF	Non-metals	INTACT	C-503	D-503
MNR (UALX-HEU) CANADA	614	09. Alum Based SNF	Stable	INTACT	C-504	D-504
MNR (UALX-HEU) CANADA	1064	09. Alum Based SNF	Stable	INTACT	C-505	D-505
MURR (UALX) COLUMBIA	144	09. Alum Based SNF	Stable	INTACT	C-506	D-506
MURR (ULAX HEU) COLUMBIA	143	09. Alum Based SNF	Stable	INTACT	C-507	D-507
N.S. SAVANNAH (UO2)	854	08. U Oxide SNF	Non-metals	INTACT	C-508	D-508
NEREIDE (FRANCE)	751	09. Alum Based SNF	Stable	INTACT	C-509	D-509
NIST	154	09. Alum Based SNF	Stable	INTACT	C-510	D-510
NIST (U308 HEU)	752	09. Alum Based SNF	Stable	INTACT	C-511	D-511
OHIO STATE (HEU)	157	09. Alum Based SNF	Stable	INTACT	C-512	D-512
OHIO STATE (LEU)	158	09. Alum Based SNF	Stable	INTACT	C-513	D-513
OMEGA WEST (204)	406	09. Alum Based SNF	Stable	INTACT	C-514	D-514
OMEGA WEST (236)	407	09. Alum Based SNF	Stable	INTACT	C-515	D-515
OMEGA WEST (250)	408	09. Alum Based SNF	Stable	INTACT	C-516	D-516
ORR (U308 HEU)	903	09. Alum Based SNF	Stable	INTACT	C-517	D-517
ORR (U3O8 HEU)	753	09. Alum Based SNF	Stable	INTACT	C-518	D-518
ORR (U3SI2 LEU)	165	09. Alum Based SNF	Stable	INTACT	C-519	D-519
ORR (U3SI2 LEU)	850	09. Alum Based SNF	Stable	INTACT	C-520	D-520
ORR (U3SI2 LEU)	944	09. Alum Based SNF	Stable	INTACT	C-521	D-521
ORR SPECIAL	163	09. Alum Based SNF	Stable	INTACT	C-522	D-522

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Fuel Name	SNF ID	TSPA Category	DBE Category		2010	2030
ORR-BW-1 (MOX)	160	04. MOX SNF	Non-metals	INTACT	C-523	D-523
PRR-1 (UALX-HEU) PHILIPPIINES	638	09. Alum Based SNF	Stable	INTACT	C-524	D-524
PRR-1 (UALX-LEU) PHILLIPPINES	558	09. Alum Based SNF	Stable	INTACT	C-525	D-525
PURDUE UNIVERSITY (U-ALX HEU)	177	09. Alum Based SNF	Stable	INTACT	C-526	D-526
PURDUE UNIVERSITY-MTR-SI	178	09. Alum Based SNF	Stable	INTACT	C-527	D-527
R-2 SVTR (U3SI2 LEU) SWEDEN	942	09. Alum Based SNF	Stable	INTACT	C-528	D-528
R-2 SVTR (UALX HEU) SWEDEN	801	09. Alum Based SNF	Stable	INTACT	C-529	D-529
RA-3 (UALX-HEU) (ARGENTINA)	634	09. Alum Based SNF	Stable	INTACT	C-530	D-530
RA-3 (UALX-HEU) (ARGENTINA)	636	09. Alum Based SNF	Stable	INTACT	C-531	D-531
RECH-1 (CHILE)	708	09. Alum Based SNF	Stable	INTACT	C-532	D-532
RHF (FRANCE)	179	09. Alum Based SNF	Stable	INTACT	C-533	D-533
RINSC	181	09. Alum Based SNF	Stable	INTACT	C-534	D-534
RINSC	180	09. Alum Based SNF	Stable	INTACT	C-535	D-535
RPI (UALX-LEU) PORTUGAL	943	09. Alum Based SNF	Stable	INTACT	C-536	D-536
RSG-GAS (U308-LEU) INDONESIA	288	09. Alum Based SNF	Stable	INTACT	C-537	D-537
RU-1 (UALX LEU) URAGUAY	557	09. Alum Based SNF	Stable	INTACT	C-538	D-538
RU-1 (UALX LEU) URAGUAY	1073	09. Alum Based SNF	Stable	INTACT	C-539	D-539
RV-1 (UALX LEU) VENEZUELA	816	09. Alum Based SNF	Stable	INTACT	C-540	D-540
SAPHIR U3SI2-LEU (SWITZERLAND)	443	09. Alum Based SNF	Stable	INTACT	C-541	D-541
SAPHIR UALX-HEU (SWITZERLAND)	444	09. Alum Based SNF	Stable	INTACT	C-542	D-542
SAPHIR ULAX MEU (SWITZERLAND)	945	09. Alum Based SNF	Stable	INTACT	C-543	D-543
SAXTON (MOX SST)	883	04. MOX SNF	Non-metals	NOT INTACT	C-544	D-544
SAXTON (MOX ZR)	787	04. MOX SNF	Non-metals	NOT INTACT	C-545	D-545
SAXTON (UO2 SST)	882	08. U Oxide SNF	Non-metals	INTACT	C-546	D-546
SAXTON (UO2 ZR)	788	08. U Oxide SNF	Non-metals	INTACT	C-547	D-547
SHIPPINGPORT PWR C1 BLKT (RODS)	189	08. U Oxide SNF	Non-metals	INTACT	C-548	D-548
SLOWPOKE (HEU) CANADA	296	09. Alum Based SNF	Stable	INTACT	C-549	D-549
SLOWPOKE (HEU) CANADA	1065	09. Alum Based SNF	Stable	INTACT	C-550	D-550
SPERT-III	209	08. U Oxide SNF	Non-metals	INTACT	C-551	D-551
THOR (UALX-HEU) TAIWAN	629	09. Alum Based SNF	Stable	INTACT	C-552	D-552
TRR-1 (UALX-HEU) THAILAND	633	09. Alum Based SNF	Stable	INTACT	C-553	D-553
UMRR (HEU) ROLLA	881	09. Alum Based SNF	Stable	INTACT	C-554	D-554
UMRR (LEU) ROLLA	146	09. Alum Based SNF	Stable	INTACT	C-555	D-555

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Fuel Name	SNF ID	TSPA Category	DBE Category		2010	2030
UNIV OF FLORIDA (ARGONAUT) HEU	272	09. Alum Based SNF	Stable	INTACT	C-556	D-556
UNIV OF FLORIDA (ARGONAUT) LEU	273	09. Alum Based SNF	Stable	INTACT	C-557	D-557
UNIV OF MASS-LOWELL (HEU)	274	09. Alum Based SNF	Stable	INTACT	C-558	D-558
UNIV OF MASS-LOWELL (LEU)	275	09. Alum Based SNF	Stable	INTACT	C-559	D-559
UNIV OF MICHIGAN	276	09. Alum Based SNF	Stable	INTACT	C-560	D-560
UNIV OF MICHIGAN (CONTROL)	1005	09. Alum Based SNF	Stable	INTACT	C-561	D-561
UNIV OF MICHIGAN (REG)	277	09. Alum Based SNF	Stable	INTACT	C-562	D-562
UNIV OF VIRGINIA (U3SI2 LEU)	952	09. Alum Based SNF	Stable	INTACT	C-563	D-563
UNIV OF VIRGINIA (ULAX HEU)	279	09. Alum Based SNF	Stable	INTACT	C-564	D-564
VBWR (UO2)	855	08. U Oxide SNF	Non-metals	INTACT	C-565	D-565
WORCESTER POLY INSTITUTE	287	09. Alum Based SNF	Stable	INTACT	C-566	D-566
ZPRL (UALX-LEU) TAIWAN	554	09. Alum Based SNF	Stable	INTACT	C-567	D-567