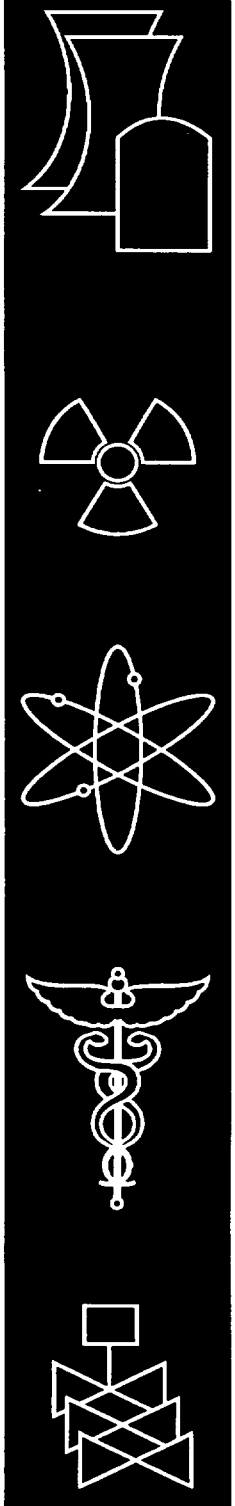
NUREG/CR-6700
ORNL/TM-2000/284



Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel

Oak Ridge National Laboratory

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at www.nrc.gov/NRC/ADAMS/index.html.
Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   P. O. Box 37082
   Washington, DC 20402–9328
   www.access.gpo.gov/su_docs
   202–512–1800
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–533–6847 or, locally, 703–805–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: Office of the Chief Information Officer,
Reproduction and Distribution
Services Section
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: DISTRIBUTION@nrc.gov
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address www.nrc.gov/NRC/NUREGS/indexnum.html are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

NUREG/CR-6700
ORNL/TM-2000/284

Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel

Manuscript Completed: December 2000
Date Published: January 2001

Prepared by
I.C. Gauld, J.C. Ryman

Oak Ridge National Laboratory
Managed by UT-Battelle, LLC
Oak Ridge, TN 37831-6370

D.D. Ebert, NRC Project Manager

Prepared for
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC Job Code W6479

ABSTRACT

This report investigates trends in the radiological decay properties and changes in relative nuclide importance associated with increasing enrichments and burnup for spent LWR fuel as they affect the areas of criticality safety, thermal analysis (decay heat), and shielding analysis of spent fuel transport and storage casks. To facilitate identifying the changes in the spent fuel compositions that most directly impact these application areas, the dominant nuclides in each area have been identified and ranked by importance. The importance is investigated as a function of increasing burnup to assist in identifying the key changes in spent fuel characteristics between conventional- and extended-burnup regimes. Studies involving both pressurized-water-reactor (PWR) fuel assemblies and boiling-water-reactor (BWR) assemblies are included. This study is seen to be a necessary first step in identifying the high-burnup spent fuel characteristics that may adversely affect the accuracy of current computational methods and data, assess the potential impact on previous guidance on isotopic source terms and decay-heat values, and thus help identify areas for methods and data improvement. Finally, several recommendations on the direction of possible future code validation efforts for high-burnup spent fuel predictions are presented.

CONTENTS

LIST OF FIGURES

LIST OF FIGURES (continued)

LIST OF TABLES

LIST OF TABLES (continued)

ACKNOWLEDGMENTS

The authors acknowledge the contributions of O. W. Hermann, B. L. Broadhead, and J. C. Wagner to discussions of assembly design characteristics, assembly models, and nuclide ranking. The efforts of W. C. Carter in preparing the manuscript, Kay Lichtenwalter in producing the plots, and C. H. Shappert in editing the manuscript are greatly appreciated. The document was reviewed by O. W. Hermann and C. E. Sanders.

1 INTRODUCTION

The present direction in the commercial nuclear power industry is towards the use of higher initial ^{235}U enrichments, more complex fuel assembly designs, and more complex core loading and fuel management schemes, leading to significantly higher-burnup fuel assemblies than seen in the past. With this increase in enrichment and burnup comes a need to reassess the accuracy of the codes and nuclear data used to predict the spent fuel isotopics and decay characteristics. This step is particularly important considering that most of the code validation performed in these areas has involved older assembly designs having low to moderate enrichments and discharge burnup. With the exception of spent fuel isotopic measurements that are now being planned or are under way, validation for extended enrichment and high-burnup fuel will likely be limited in the near term by a lack of experimental data. This has led to a situation where the enrichments and burnup from operating commercial light-water-reactor (LWR) fuels now extend beyond the range where available experimental data can be used to validate computational methods.

This report investigates the trends in the radiological-decay properties and changes in nuclide importance associated with increasing enrichments and burnup for spent LWR fuel as they affect the areas of criticality safety, thermal analysis (decay heat), and shielding analysis of spent fuel transport and storage casks. To facilitate identifying the changes in the spent fuel compositions that most directly impact these application areas, the dominant nuclides in each area have been identified and ranked by importance. The nuclide importance is investigated as a function of increasing burnup to assist in identifying the changing spent fuel characteristics between conventional- and extended-burnup fuel. Studies involving typical pressurized-water-reactor (PWR) fuel assemblies and boiling-water-reactor (BWR) assembly designs are included, although a detailed assessment of the impact of advanced fuel assembly designs on spent fuel properties was beyond the scope of this study.

1.1 Objectives

The objective of this study is to identify the changes in the spent fuel compositions and decay properties associated with extended enrichment and high-burnup spent fuel as they impact cask design, safety and licensing analyses. This task is accomplished primarily by using nuclide importance rankings to provide a measure of a nuclide's increasing or decreasing relative importance in each of the regimes for the application areas of criticality safety, decay-heat generation, and radiation shielding. This study is seen as a first step in evaluating the potential challenges that high-enrichment and high-burnup fuel presents to current computational methods and nuclear data by (1) considerably reducing the number of isotopes that need to be considered, and (2) identifying the key isotopes that show the greatest variation in importance for extended regimes. This information is important to help identify the high-burnup spent fuel characteristics that may adversely affect the accuracy of current computational methods and data, assess the potential impact on previous guidance on isotopic source terms and decay heating, and thus help identify areas for methods and data improvement.

This information may also be useful in assessing the applicability of existing experimental validation data for conventional burnup fuel to the high-burnup regime by identifying the degree of similarity, or difference, in the nuclide-importance rankings between high-burnup regimes and experimental-data regimes.

Numerous studies have been published that document the validation of computer codes for the prediction of spent fuel isotopics and radiological-decay characteristics for conventional-burnup fuels,[1–7] and spent fuel properties in this regime are believed to be relatively well known. The present report does not attempt to compile or summarize the earlier studies, but instead focuses on investigating and identifying the unique characteristics associated with high-burnup fuel.

Finally, several recommendations on the direction of possible future code validation efforts for high-burnup spent fuel predictions are presented.

2 ANALYSIS APPROACH

2.1 Areas of Interest

The nuclide importance rankings presented in this work cover the areas of criticality safety, thermal analysis (decay-heat generation), and radiation shielding as they apply to licensing evaluations for spent fuel transport and interim storage casks. The nominal time range of interest is from 2 to 100 years after discharge from a reactor. For criticality-safety studies, neutron-absorption fractions were used to rank the importance of the individual actinide and fission-product absorbers. The nuclide-absorption rankings were derived using total-absorption cross sections for the nuclides in the fuel and the assembly structures, and include both fission and capture processes. Although absorption fractions are only an approximate measure of importance to criticality, rankings using absorption fractions have been found to be similar to those obtained using more rigorous measures of nuclide-importance.[8] The fractional contributions to the total decay-heat generation were used to obtain nuclide importance rankings to thermal analyses. Similarly, the fractional contribution of individual nuclides to the neutron and photon dose rate outside of shielded transport and storage casks were used for shielding rankings. For the dose-rate calculations, several cask designs were evaluated since the nuclide importance is dependent, not only on the source term, but also on cask shielding materials and configuration.

These studies are largely an extension of previous ranking studies, reported in Ref. 9, to higher-enrichments and higher-burnup regimes for each of the application areas and focus primarily on the changes in nuclide importance with increasing burnup, rather than cooling time. In addition, the present study limits the range of cooling times to only those relevant to transport and interim storage facilities.

2.2 Range of Study

The spent fuel characteristics were studied for several combinations of enrichment and burnup representative of current and projected near-term spent fuel assembly designs. To obtain an estimate of burnup typically achieved for a given initial enrichment, the LWR Characteristic Database was reviewed.[10] This database contains historical enrichment and burnup data (up to 1993), as well as projected data (up to 2030) for LWR fuel assemblies. The historical data for PWR and BWR fuel, and the projected data up to 2005, are illustrated in Figure 1. Each data point represents a group of 100 assemblies or more. Groups with fewer than 100 assemblies were excluded from the figure so as to better represent the majority of assemblies. Data beyond 2005 were excluded since they contained a large number of low-burnup assemblies that are thought to represent partially burned assemblies from decommissioned plants.

Figure 1 indicates that the enrichment/burnup relationships for PWR and BWR assemblies are very similar. The estimated burnup for 5-wt %-enriched fuel is about 62 GWd/t.* For this study the highest enrichment considered was 5 wt % with a maximum burnup of 70 GWd/t for both reactor-type fuels. The projected burnups derived from the database were also reviewed against more recent commercial reactor burnup data from several plants that used newer fuel assembly designs than may have been considered in the LWR database projections. The assembly burnups from these plants were found to be similar to the values shown in Figure 1. The initial enrichment values and burnup ranges considered in this study are listed in Table 1.

* GWd/t = gigawatt days per metric tonne uranium.

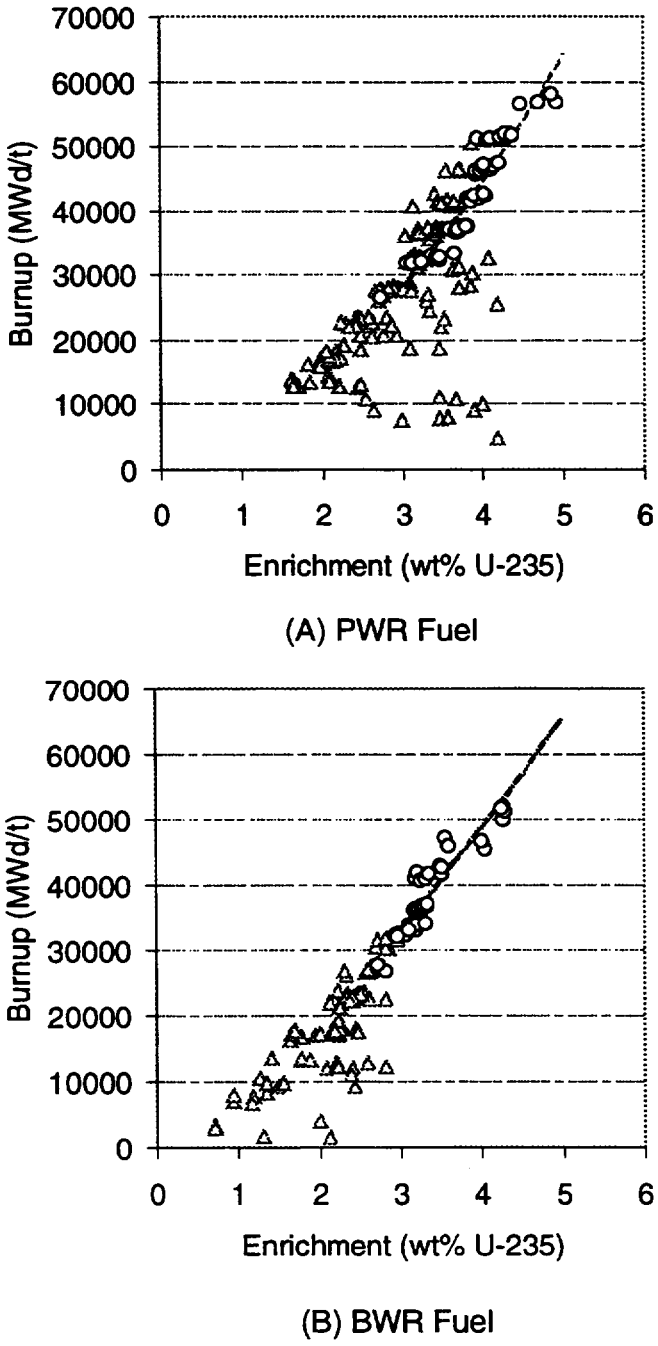


Figure 1 Historical and projected LWR spent fuel discharge burnup

Table 1 Burnup and enrichment ranges used in high-burnup ranking studies

Enrichment (wt % ^{235}U)	Burnup range studied (GWd/t)	Estimated industry-average burnup (GWd/t)
3.0	20–40	30
4.0	20–50	45
5.0	20–70	62

2.3 Depletion-Analysis Models

The basis for the ranking studies is a common set of spent fuel compositions calculated for the different enrichment and burnup combinations. These common compositions were applied to each of the application areas. Fuel assembly models were developed for both PWR and BWR types, and depletion calculations were performed using the SAS2 depletion analysis module of SCALE[11] and the ENDF/B-V 44-group burnup cross-section library.[12] The depletion calculations were performed using a constant specific power of 35 MW/t as a reference with the exposure periods being adjusted to give the desired burnup. The sensitivity of the results to the specific power was investigated by performing additional calculations at 25 MW/t and is discussed in the results sections. The initial concentrations of the minor U isotopes ^{234}U and ^{236}U were determined from the ^{235}U enrichment using the formulas in Ref. 2.

Assembly structural materials and impurities were also included, based primarily on specifications in Ref. 2. A nominal cobalt impurity level of 4600 ppm (0.46 wt %) in the grid spacers was assumed. The activation of ^{60}Co is an important component of the dose rate and, to a much lesser extent, decay heating. The impurity level used in this analysis is relatively high compared with levels associated with newer assembly designs. The levels used in the analysis of newer designs is typically about 1200 ppm, with actual impurity levels likely being at the several-hundred-ppm level. Therefore, the fractional contribution from ^{60}Co reported here is likely overestimated for the current generation of higher-enrichment and higher-burnup assemblies now in use. A reduction in the ^{60}Co contribution will result in a proportionally higher contribution from other nuclides. However, the trends in the nuclide importance will be largely unaffected.

The assembly model used to generate spent fuel compositions for each of the PWR enrichment and burnup combinations in this study was based on a Westinghouse 17 × 17 assembly model. The assembly model did not contain burnable poison rods. The assembly model was originally developed to generate cross sections for spent fuel depletion analyses[13] and included the structural cladding and Inconel grid spacer components. The calculated responses therefore included contributions from the actinides, fission products, plus activation or neutron capture in structural assembly components. The BWR model was based on a General Electric (GE) 8 × 8 assembly design previously used in the development of the NRC Regulatory Guide 3.54 for computing decay-heat generation rates in spent fuel.[2] The BWR assembly model incorporated 4 gadolinium poison rods, each rod containing 4-wt % Gd_2O_3, a gadolina enrichment value that is representative of assemblies with higher enrichment uranium. Calculations were also performed with 5- and 6-wt % gadolinia. The effect of the gadolina poison loading on the individual nuclide importance rankings was found to be minimal. Consequently, only the results for the 4-wt %-Gd_2O_3 fuel are reported. Design specifications for the reactor assembly models used to generate the spent fuel isotopic compositions are summarized in Table 2, and assembly structural masses are listed in Table 3. Typical SAS2 input files used to model the PWR and BWR assemblies are listed in the Appendix of this report.

Table 2 Fuel assembly specifications

Parameter	PWR assembly data	BWR assembly data
Assembly design data		
Designer	Westinghouse	GE
Lattice	17 × 17	8 × 8
Assembly pitch, cm	21.4173	15.24
Number of fuel rods	264	63
Number of instrument tubes	1	1
Number of guide tubes	24	– [a]
Number of rods containing poison	–	4
Fuel type	UO_2	UO_2 and UO_2 + 4-wt % Gd_2O_3
Average fuel density, g/cm^3	10.32 (0.942 TD)	9.871 (0.901 TD)
Effective fuel temperature, K	811	840
Clad type	Zircaloy-4	Zircaloy-4
Clad temperature, K	620	620
Fuel rod data		
Fuel rod outer diameter, cm	0.81915	1.0795
Gap outer diameter, cm	0.83566	–
Clad outer diameter, cm	0.94966	1.25222
Fuel rod pitch, cm	1.25984	1.6256
Guide tube data		
Inner radius, cm	0.57150	–
Outer radius, cm	0.61214	–
Guide tube material	Zircaloy-4	–
Moderator data		
Average density, g/cm^3	0.7295	0.4323
Average boron concentration, ppm	550	–
Moderator temperature, K	570	558

[a] Data not applicable.

Table 3 Assembly structural material masses

Element	PWR data (g/kg U)	BWR data (g/kg U)
O	135	135
Cr	5.9	2.4
Mn	0.33	0.15
Fe	13	6.6
Co	0.075	0.024
Ni	9.9	2.4
Zr	221	516
Nb	0.71	0
Sn	3.6	8.7

The depletion analysis models used in this study are based on conventional LWR assembly designs. However, as the enrichments and discharge burnups extend towards higher design values the assemblies generally become more complex, employing larger burnable poison loadings, removable poison rods, and variable enrichments. Also, modern reactor operations involve more complex core loading schemes, control rod insertion, power-shaping rods, and may include extended operating regimes such as temperature, void fractions, and higher moderator soluble boron concentrations. An analysis of the impact of advanced fuel assembly designs was beyond the scope of this study, given the wide range of PWR and BWR assembly designs in use and the potentially different effects the designs may have on each of the respective application areas considered in this report (criticality safety, thermal analysis, and shielding). Importantly, the isotopic importance rankings for the PWR assembly and BWR assembly (included burnable poison rods) designs used in this study were found too remarkably similar. Also, the results for the BWR assembly changed very little with increasing burnable poison concentrations. Therefore, while the overall impact on the responses is expected to vary with assembly design, the relative nuclide importances, the focus of thus study, are expected to be applicable for a wide range of conventional and modern fuel assembly designs.

2.4 Cask Models

Three different transport/storage cask designs were used to study dose-rate characteristics and to develop shielding-importance rankings for individual radionuclides. These designs included a steel, a concrete, and a lead cask. The cask design specifications were the same as those used previously in the ranking studies of Ref. 9. The steel transport cask incorporated a 27-cm carbon-steel shield and an outer 13-cm resin-neutron shield. The concrete storage cask consisted of a 50-cm-thick concrete shield, and the lead transfer cask used 12.7-cm lead and outer 13-cm resin-neutron shield. In all cask models, a homogenized dry fuel region was used. Since the fractional contribution of a radionuclide to the dose rate is dependent on the specific cask design and shielding materials, the importance values calculated using the cask designs presented here are only intended to capture the broad features and trends for several shielding materials common to many transport and storage cask designs.

2.5 Shielding Analyses

Neutron and gamma-ray dose rates at the cask surface were calculated for each of the cask models using the one-dimensional discrete ordinates shielding module SAS1.[14] The transport calculations were performed

using the coupled SCALE 27-neutron and 18-gamma-group cross-section library.[12] The dose rates were calculated by SAS1 for each of the potentially important radionuclides in the fuel assembly, for each cask type, based on a unit concentration of each individual nuclide in the respective cask designs. All actinides and fission products having a half-life greater than 30 days, and a gamma-ray energy release exceeding 10 eV/disintegration were selected. In addition, the activation product ^{60}Co was selected. The procedure generated neutron- and gamma-dose-rate conversion factors (dose rate per unit concentration) for 109 radionuclides.

The neutron and gamma source spectra for each radionuclide were obtained from ORIGEN-S.[15] Neutron source spectra included spontaneous fission, and (alpha,n) neutrons, assuming a uranium oxide matrix. Photon spectra were based on data from the Master Photon Library.[16] The dose-rate conversion factors were subsequently combined with calculated spent fuel nuclide concentrations to obtain the dose rates from the individual and/or aggregate radionuclides in the fuel. The contributions from each radionuclide were sorted to obtain the fractional contribution of each nuclide to the total dose rate and the radionuclide ranking. Examples of the SAS1 and ORIGEN-S input files used to generate the dose-rate conversion factors are listed in the Appendix.

2.6 Ranking Studies

The fractional contributions of individual nuclides to the decay heat, neutron absorption, and total radiation dose rate were used to rank the nuclides by importance in each of the respective application areas. Processing of the nuclide inventories and ranking by importance was performed using the OPUS code (ORIGEN-S post-processing utility for SCALE) currently being developed as a SCALE system utility. OPUS reads the nuclide concentrations calculated by ORIGEN-S, converts the data to response units requested by the user, and automatically sorts the nuclides in the order of descending importance. In addition the user may request specific nuclides, and output may be generated for standard radiological quantities such as activity and decay heat, or an arbitrary response specified by the user may be applied. Such a procedure was used to generate the dose rate importance rankings by using the dose-rate conversion factors for each of the dominant actinides, fission products, and activation products as calculated by SAS1. Examples of the OPUS input and output files are listed in the Appendix.

3 CRITICALITY SAFETY

The nuclide importance rankings to criticality safety analyses were evaluated for the dominant actinides and fission products in extended burnup fuels. The rankings are based on fractional neutron absorption. A previous ranking study[9] examined PWR fuel in the range of 3-wt % ^{235}U, 20-GWd/t burnup to 4.5-wt % ^{235}U, 50-GWd/t burnup, with an emphasis on rankings as a function of cooling time. This study investigates both PWR and BWR fuel assemblies up to 5-wt % ^{235}U and 70 GWd/t burnup, with an emphasis on the variation of the importance ranking with increasing burnup.

The total actinide, fission product, and structural material absorption fractions as a function of burnup are shown in Figure 2 for 5 wt % PWR fuel and cooling times of 5 and 100 years. The relative contribution of the fission products increases nearly linearly with burnup, with a corresponding decrease in the actinide contribution. The fission products represent about 6% of the total absorptions at 20 GWd/t, increasing to about 15% at 70 GWd/t. The variation with cooling time is very minor, as seen in Figure 2, due to the fact that the majority of the dominant actinides and fission products are either long-lived or stable. A comparison of the PWR with the BWR assembly results indicated only minor differences in the aggregate fission product and actinide absorption fractions. The only noticeable differences between the assembly types are that the BWR assembly consistently has slightly greater absorption by the light elements than the PWR assembly (e.g., 3.5% compared with 2.5%), and correspondingly less absorption by actinides. This difference is consistent with the BWR assembly model having more structural-material (mass)-per-unit-uranium mass than the PWR assembly.

Note that the relative importance of the actinides and fission products is problem dependent due to the influence the environment has on the effective absorption cross sections. Typically, in a cask environment the importance of the fission products is greater than in a reactor configuration. Therefore, the important features in the results are the relative trends in the rankings, rather than the absolute fractional absorption values that depend on the specific application.

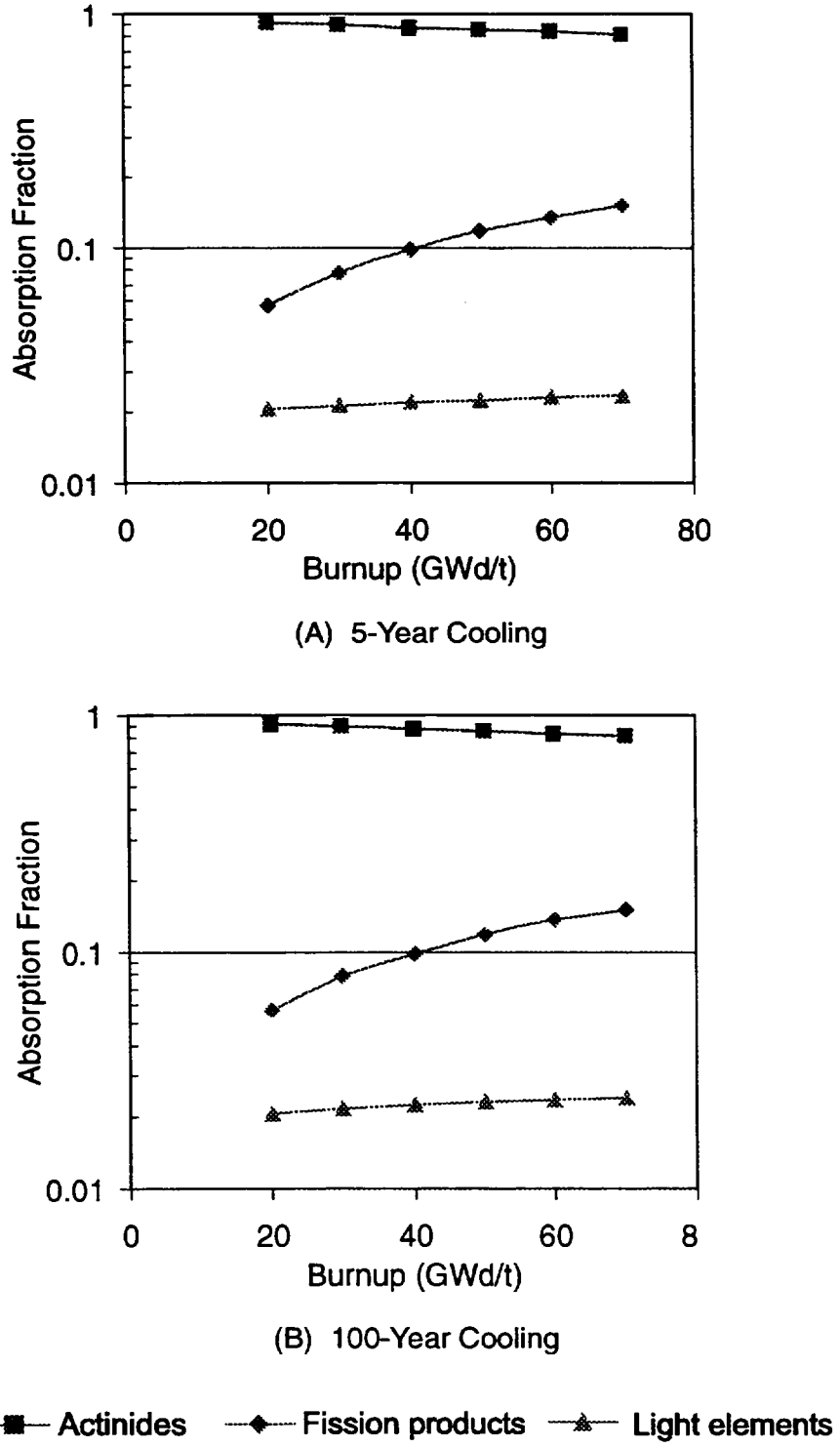


Figure 2 Absorption fractions for 5-wt % PWR fuel as a function of burnup

3.1 Criticality-Safety Rankings

The fractional contributions of individual actinides and fission-product nuclides to the neutron absorption were generated to obtain the radionuclide importance to criticality-safety analyses. The actinide and fission-product plots are presented separately due to the large number of nuclides that contribute greater than 0.1% to the total neutron absorption (12 actinides and 25 fission products) relative to the other application areas considered in this study. In terms of nuclide rankings, the PWR and BWR results were found to be very similar; therefore, plots are shown for only the PWR assembly.

The nuclide importance, expressed as the fractional contribution of each nuclide to the total neutron-absorption rate, is illustrated in Figures 3 and 4 (actinides) and Figures 5 and 6 (fission products) for 5-wt % PWR fuel. The fractional contributions and rankings of actinides in the PWR fuel are given in Table 4 for decay times of 5 and 100 years. The corresponding contributions and rankings for fission products are given in Table 5. The BWR actinide rankings are listed in Table 6, and fission product rankings are located in Table 7. Note that where the ranking order of nuclides is different for the BWR than the PWR, that the absorption values are nearly identical, so that the general trend is still the same.

The dominant actinide absorbers (absorption fraction >0.1) are ^{238}U, whose absorption fraction remains fairly constant (at about 0.28) as burnup increases: ^{239}Pu, whose fraction increases up to about 50 GWd/t, then levels out; and ^{235}U, whose fraction decreases rapidly with burnup as it is depleted by fission. The next group of moderately absorbing actinides (0.01 < absorption fraction < 0.1) include ^{240}Pu, ^{241}Am, ^{236}U, and ^{241}Pu (at shorter decay times). The rate of increase of importance for these actinides is fairly rapid up to about 40 GWd/t, then somewhat less at higher burnups. The importance of the remaining actinides (<0.01 absorption fraction) generally increases fairly rapidly throughout the entire burnup range and for all cooling times. The exception is ^{234}U, whose importance decreases slowly with burnup at short cooling times, stays fairly constant at 50-years cooling, and increases slightly with burnup at 100-years cooling.

The relative total actinide absorptions in the PWR assembly (82.3%) for high-enrichment and high-burnup fuel are seen to be greater than those for the BWR assembly (80.7%). This difference is attributed to the larger amount of structural-material-per-unit-fuel mass in the BWR assembly that results in greater structural material absorption and consequently a lower relative contribution from the actinides and fission-product nuclide groups. Similar differences are seen for all enrichments and burnups.

The importance of the individual fission-product absorbers is much less than that of the actinides, with the dominant nuclides (^{103}Rh, ^{143}Nd, and ^{149}Sm) all having individual absorption fractions <0.02. However, there are a large number of absorbing fission products, and as a group they generally represent between about 6 and 15% of the total absorption.

Most fission-product absorbers exhibit a steady rate of increase (quasi-linear) in importance with burnup over the entire burnup range considered. Two nuclides whose importance changes less rapidly are ^{149}Sm, whose importance is nearly constant beyond 30 GWd/t, and ^{147}Sm, whose importance is fairly constant after 50 GWd/t. This slower importance change is due to a rapid buildup of these fission products to equilibrium levels during irradiation, after which their concentrations in the fuel remain near constant. The concentrations of most other absorbing fission products steadily increase during irradiation.

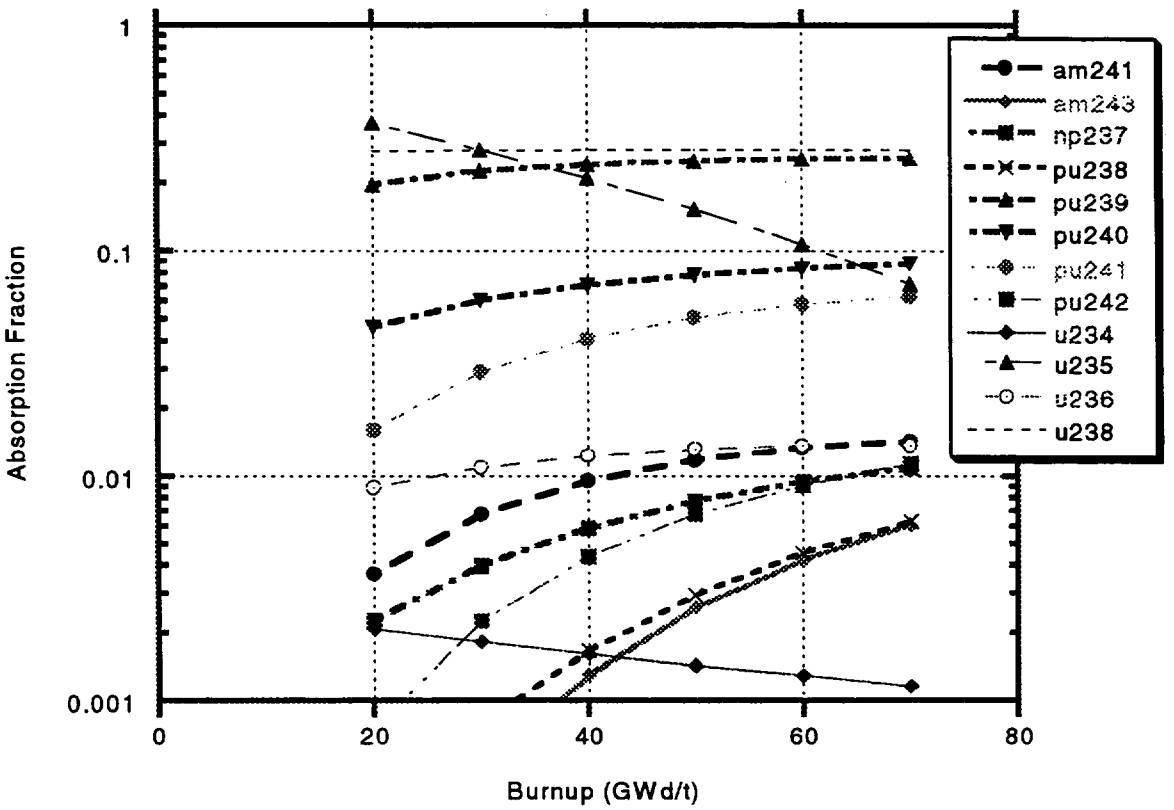


Figure 3 Fraction of total neutron absorptions from actinides for 5-wt % PWR fuel, 5-year cooling

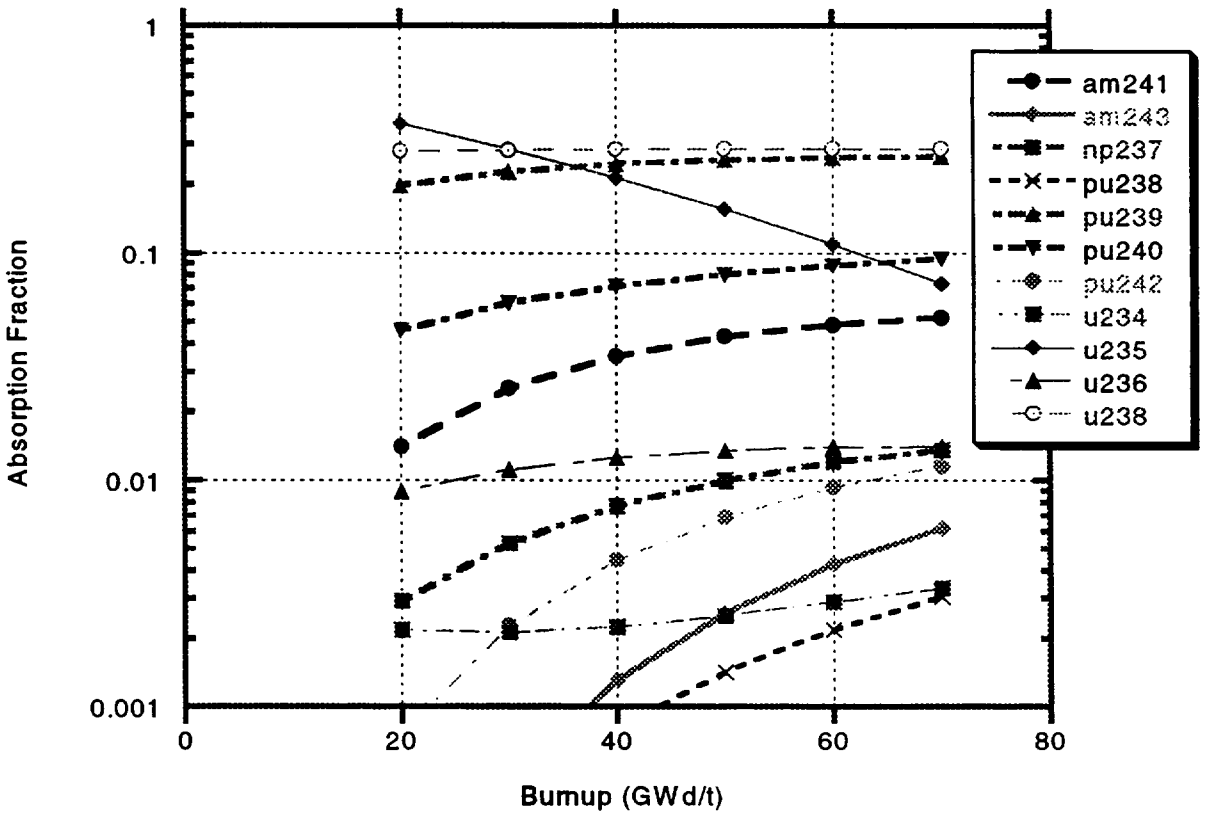


Figure 4 Fraction of total neutron absorptions from actinides for 5-wt % PWR fuel, 100-year cooling

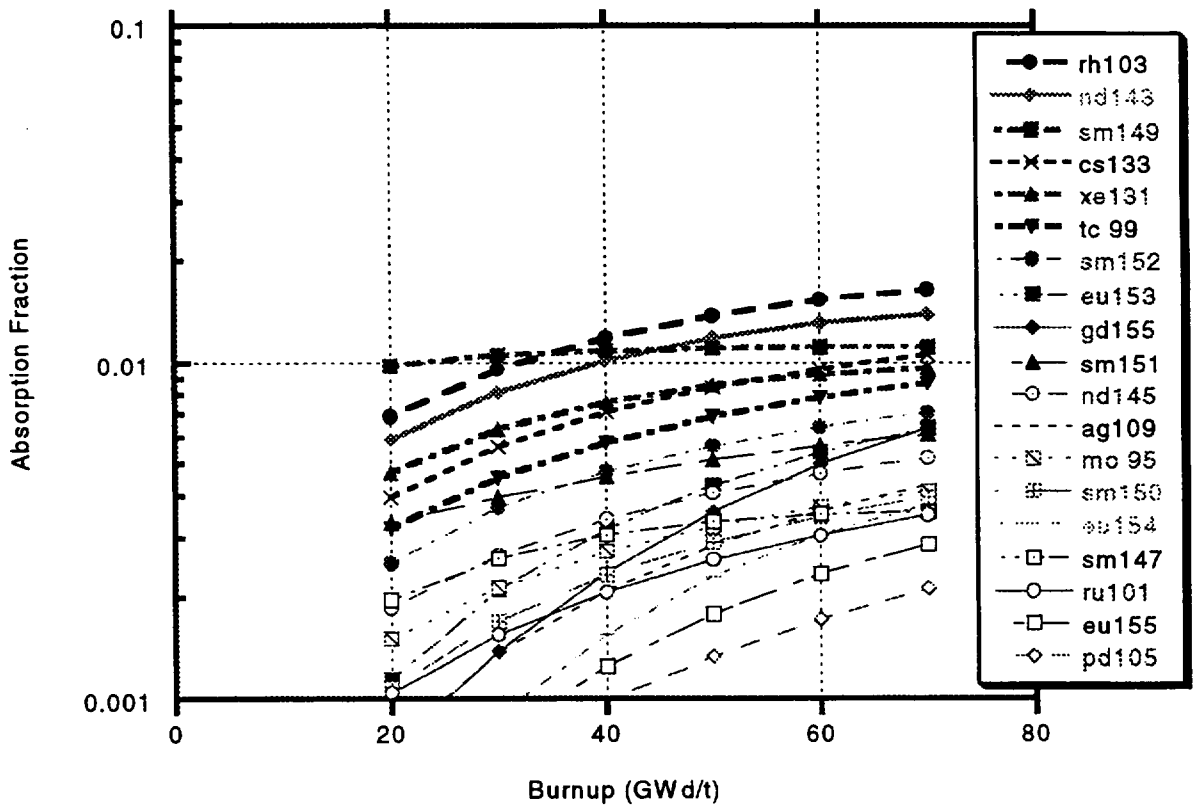


Figure 5 Fraction of total neutron absorptions from fission products for 5-wt % PWR fuel, 5-year cooling

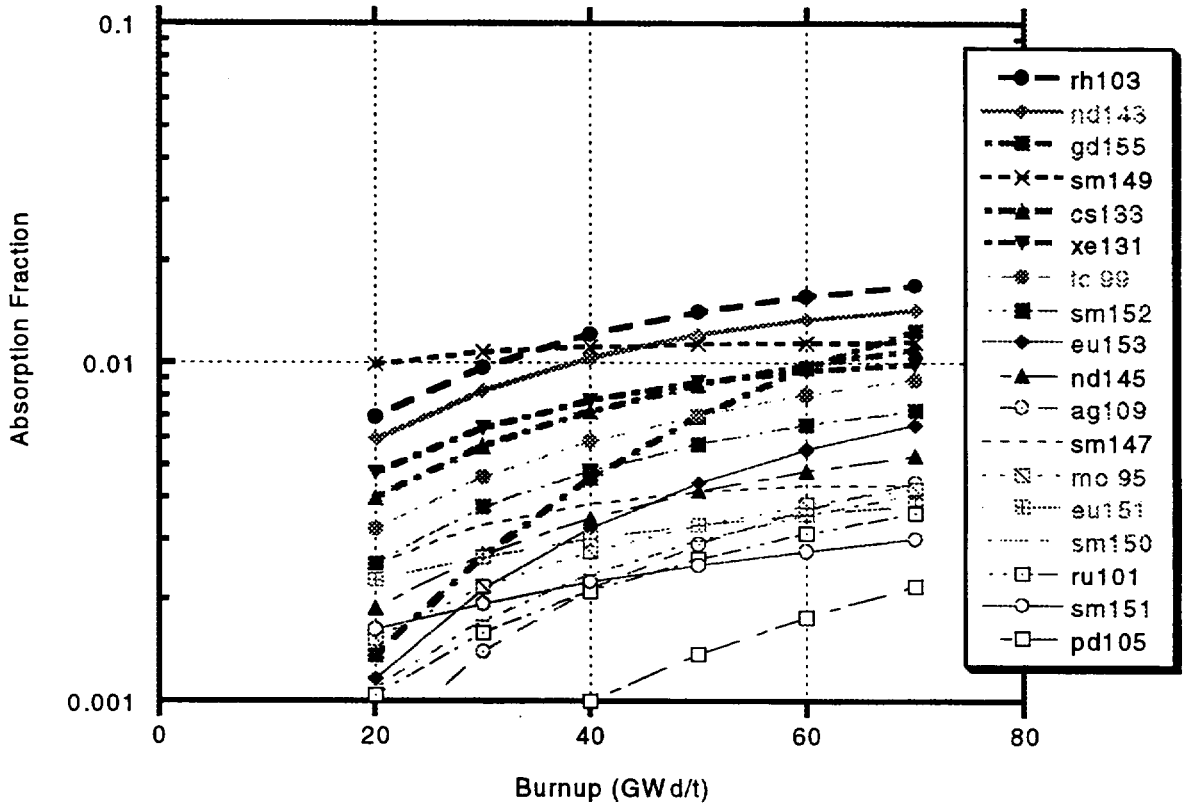


Figure 6 Fraction of total neutron absorptions from fission products for 5-wt % PWR fuel, 100-year cooling

Table 4 Criticality-safety rankings for dominant actinides in PWR fuel

	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Am-241	0.63	7	1.36	6	0.82	7	1.40	6	0.95	7	1.43	6
Am-243	0.03	12	0.50	10	0.08	12	0.56	10	0.13	12	0.61	11
Np-237	0.26	8	0.73	9	0.42	8	0.91	9	0.59	8	1.08	9
Pu-238	0.05	11	0.41	11	0.10	11	0.52	11	0.17	10	0.63	10
Pu-239	25.23	2	27.66	2	24.71	2	26.66	2	24.14	2	25.82	2
Pu-240	6.48	4	9.17	3	6.87	4	8.97	3	7.04	4	8.80	3
Pu-241	3.06	5	6.51	4	3.70	5	6.39	4	4.08	5	6.30	5
Pu-242	0.18	9	1.05	7	0.32	9	1.09	8	0.44	9	1.13	8
U-234	0.14	10	0.08	12	0.15	10	0.10	12	0.16	11	0.12	12
U-235	22.88	3	5.22	5	21.69	3	6.31	5	20.97	3	7.15	4
U-236	0.75	6	0.94	8	1.01	6	1.17	7	1.23	6	1.37	7
U-238	30.88	1	30.35	1	29.27	1	29.09	1	28.00	1	28.00	1
Total	90.59		84.12		89.17		83.34		87.94		82.62	
100-Year cooling												
Am-241	2.43	5	5.11	5	3.09	5	5.16	5	3.53	5	5.21	5
Am-243	0.03	10	0.51	9	0.08	10	0.57	9	0.13	10	0.62	9
Np-237	0.38	7	0.99	7	0.58	7	1.18	7	0.77	7	1.36	7
Pu-238	0.02	11	0.20	11	0.05	11	0.25	11	0.08	11	0.30	11
Pu-239	25.50	2	28.31	2	25.01	2	27.24	2	24.45	2	26.35	2
Pu-240	6.51	4	9.74	3	6.94	4	9.60	3	7.14	4	9.49	3
Pu-242	0.19	8	1.08	6	0.32	8	1.12	8	0.44	8	1.15	8
U-234	0.16	9	0.22	10	0.19	9	0.28	10	0.23	9	0.34	10
U-235	23.21	3	5.38	4	22.04	3	6.49	4	21.32	3	7.34	4
U-236	0.77	6	0.97	8	1.03	6	1.20	6	1.25	6	1.41	6
U-238	31.29	1	31.13	1	29.71	1	29.79	1	28.44	1	28.63	1
Total	90.52		83.78		89.08		83.04		87.85		82.36	

[a] Nuclides that contribute more than 0.1% to the total absorption rate.

Table 5 Criticality-safety rankings for dominant fission products in PWR fuel

	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Ag-109	0.12	16	0.40	12	0.17	15	0.42	12	0.21	15	0.43	12
Cs-133	0.44	5	0.89	4	0.59	5	0.98	4	0.71	5	1.06	4
Eu-153	0.16	11	0.54	10	0.24	11	0.59	9	0.32	10	0.64	8
Eu-154	0.07	18	0.32	15	0.11	18	0.35	15	0.16	18	0.38	15
Eu-155	0.06	19	0.24	18	0.09	19	0.27	18	0.12	19	0.29	18
Gd-155	0.15	13	0.61	8	0.19	13	0.63	8	0.24	13	0.64	9
Gd-157	0.03	24	0.10	23	0.04	24	0.11	23	0.05	24	0.11	23
La-139	0.04	22	0.09	24	0.05	23	0.10	24	0.06	23	0.11	24
Mo-95	0.16	12	0.33	14	0.22	12	0.37	13	0.27	12	0.41	13
Nd-143	0.76	3	1.23	3	0.90	3	1.32	2	1.02	3	1.40	2
Nd-145	0.21	9	0.42	11	0.28	9	0.47	11	0.34	9	0.52	11
Pd-105	0.05	20	0.18	19	0.08	20	0.20	19	0.10	20	0.21	19
Pd-108	0.03	23	0.15	20	0.05	22	0.16	20	0.07	22	0.17	20
Pm-147	0.13	15	0.15	21	0.15	17	0.15	21	0.16	17	0.15	21
Pr-141	0.05	21	0.12	22	0.07	21	0.13	22	0.08	21	0.14	22
Rh-103	0.84	2	1.55	1	1.04	2	1.61	1	1.19	1	1.65	1
Ru-101	0.11	17	0.27	17	0.16	16	0.31	17	0.21	16	0.35	17
Sm-147	0.20	10	0.30	16	0.26	10	0.33	16	0.31	11	0.36	16
Sm-149	1.17	1	1.24	2	1.12	1	1.17	3	1.09	2	1.12	3
Sm-150	0.14	14	0.34	13	0.19	14	0.37	14	0.23	14	0.40	14
Sm-151	0.37	6	0.56	9	0.42	7	0.59	10	0.46	8	0.61	10
Sm-152	0.31	8	0.63	7	0.40	8	0.67	7	0.47	7	0.71	7
Tc-99	0.35	7	0.72	6	0.47	6	0.80	6	0.58	6	0.87	6
Xe-131	0.52	4	0.86	5	0.66	4	0.92	5	0.76	4	0.97	5
Total	6.85		13.21		8.49		14.15		9.86		15.01	

[a] Nuclides that contribute more than 0.1% to the total absorption rate.

Table 5 (continued)

	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
100-Year cooling												
Ag-109	0.12	16	0.41	11	0.17	16	0.43	11	0.21	16	0.44	11
Cs-133	0.45	5	0.91	5	0.60	5	1.00	5	0.72	5	1.09	5
Eu-151	0.23	10	0.33	15	0.27	11	0.36	15	0.30	12	0.38	15
Eu-153	0.17	13	0.56	9	0.25	12	0.61	9	0.32	11	0.65	9
Gd-155	0.28	8	1.19	4	0.37	8	1.22	3	0.46	8	1.24	3
Gd-157	0.03	22	0.10	21	0.04	22	0.11	21	0.05	22	0.11	21
La-139	0.04	20	0.09	22	0.05	21	0.10	22	0.06	21	0.11	22
Mo-95	0.16	14	0.34	14	0.22	13	0.38	13	0.28	13	0.42	13
Nd-143	0.77	3	1.26	3	0.92	3	1.36	2	1.03	3	1.43	2
Nd-145	0.21	11	0.43	10	0.28	10	0.48	10	0.35	10	0.53	10
Pd-105	0.05	18	0.18	18	0.08	18	0.20	18	0.10	18	0.22	18
Pd-108	0.03	21	0.15	19	0.05	20	0.16	19	0.07	20	0.18	19
Pr-141	0.05	19	0.12	20	0.07	19	0.13	20	0.08	19	0.15	20
Rh-103	0.85	2	1.59	1	1.06	2	1.65	1	1.21	1	1.69	1
Ru-101	0.11	17	0.27	17	0.16	17	0.32	16	0.21	17	0.36	16
Sm-147	0.26	9	0.37	12	0.33	9	0.41	12	0.38	9	0.44	12
Sm-149	1.18	1	1.27	2	1.14	1	1.20	4	1.11	2	1.15	4
Sm-150	0.14	15	0.35	13	0.19	15	0.38	14	0.24	14	0.41	14
Sm-151	0.18	12	0.28	16	0.20	14	0.29	17	0.22	15	0.30	17
Sm-152	0.32	7	0.65	8	0.41	7	0.69	8	0.48	7	0.72	8
Tc-99	0.36	6	0.74	7	0.48	6	0.82	7	0.59	6	0.89	7
Xe-131	0.53	4	0.89	6	0.67	4	0.94	6	0.77	4	0.99	6
Total	6.89		13.49		8.54		14.40		9.91		15.22	

[a] Nuclides that contribute more than 0.1% to the total absorption rate.

Table 6 Criticality-safety rankings for dominant actinides in BWR fuel

	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Am-241	0.61	7	1.31	6	0.79	7	1.34	6	0.91	7	1.36	6
Am-243	0.04	12	0.51	10	0.08	12	0.56	10	0.13	12	0.61	11
Np-23	0.27	8	0.75	9	0.43	8	0.92	9	0.59	8	1.09	9
Pu-238	0.06	11	0.45	11	0.11	11	0.56	11	0.18	10	0.67	10
Pu-239	4.69	2	27.21	2	24.14	2	26.21	2	23.56	2	25.36	2
Pu-240	6.25	4	8.95	3	6.57	4	8.75	3	6.72	4	8.59	3
Pu-241	2.99	5	6.41	4	3.59	5	6.30	4	3.95	5	6.21	5
Pu-242	0.19	9	1.04	7	0.32	9	1.07	8	0.43	9	1.10	8
U-234	0.14	10	0.09	12	0.15	10	0.10	12	0.16	11	0.12	12
U-235	3.42	3	5.11	5	22.35	3	6.15	5	21.65	3	6.95	4
U-236	0.76	6	0.93	8	1.00	6	1.15	7	1.20	6	1.34	7
U-238	30.05	1	29.74	1	28.39	1	28.48	1	27.12	1	27.38	1
Total	89.47		82.63		87.94		81.77		86.64		80.99	
100-Year cooling												
Am-241	2.38	5	4.94	5	2.98	5	4.98	5	3.38	5	5.01	5
Am-243	0.04	10	0.52	9	0.08	10	0.57	9	0.13	10	0.62	9
Np-237	0.39	7	1.00	7	0.58	7	1.18	7	0.77	7	1.35	7
Pu-238	0.03	11	0.22	11	0.05	11	0.27	11	0.09	11	0.33	11
Pu-239	24.94	2	27.87	2	24.43	2	26.81	2	23.86	2	25.92	2
Pu-240	6.28	4	9.52	3	6.64	4	9.38	3	6.82	4	9.27	3
Pu-242	0.19	8	1.06	6	0.33	8	1.10	8	0.44	8	1.13	8
U-234	0.16	9	0.22	10	0.19	9	0.28	10	0.23	9	0.34	10
U-235	23.75	3	5.27	4	22.70	3	6.33	4	22.01	3	7.14	4
U-236	0.77	6	0.97	8	1.02	6	1.19	6	1.23	6	1.38	6
U-238	30.44	1	30.52	1	28.81	1	29.20	1	27.54	1	28.04	1
Total	89.40		82.25		87.85		81.43		86.55		80.69	

[a] Nuclides that contribute more than 0.1% to the total absorption rate.

Table 7 Criticality-safety rankings for dominant fission products in BWR fuel

	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Ag-109	0.13	16	0.41	12	0.17	15	0.43	12	0.21	16	0.45	12
Cs-133	0.45	5	0.90	4	0.59	5	0.99	4	0.72	5	1.07	4
Eu-153	0.17	11	0.56	10	0.25	11	0.62	9	0.33	10	0.66	8
Eu-154	0.07	18	0.33	15	0.12	18	0.37	15	0.16	17	0.40	15
Eu-155	0.07	19	0.26	18	0.10	19	0.28	18	0.13	19	0.30	18
Gd-155	0.14	14	0.63	8	0.19	14	0.64	8	0.23	14	0.66	9
Gd-157	0.03	24	0.10	23	0.04	24	0.10	24	0.04	24	0.11	24
La-139	0.04	22	0.10	24	0.05	23	0.11	23	0.06	23	0.12	23
Mo-95	0.17	12	0.34	14	0.23	12	0.38	14	0.28	12	0.42	13
Nd-143	0.77	3	1.30	2	0.94	3	1.40	2	1.07	3	1.49	2
Nd-145	0.21	9	0.44	11	0.29	9	0.49	11	0.35	9	0.54	11
Pd-105	0.06	20	0.19	19	0.08	20	0.21	19	0.10	20	0.22	19
Pd-108	0.04	23	0.15	20	0.05	22	0.17	20	0.07	22	0.18	20
Pm-147	0.14	15	0.15	21	0.15	17	0.15	21	0.16	18	0.15	22
Pr-141	0.05	21	0.13	22	0.07	21	0.14	22	0.09	21	0.16	21
Rh-103	0.87	2	1.59	1	1.07	2	1.65	1	1.22	1	1.69	1
Ru-101	0.12	17	0.28	17	0.17	16	0.33	17	0.22	15	0.37	16
Sm-147	0.21	10	0.31	16	0.27	10	0.34	16	0.32	11	0.37	17
Sm-149	1.19	1	1.30	3	1.15	1	1.24	3	1.12	2	1.19	3
Sm-150	0.14	13	0.36	13	0.19	13	0.39	13	0.24	13	0.42	14
Sm-151	0.37	6	0.57	9	0.42	7	0.60	10	0.46	8	0.63	10
Sm-152	0.32	8	0.65	7	0.41	8	0.69	7	0.48	7	0.72	7
Tc-99	0.36	7	0.73	6	0.48	6	0.81	6	0.58	6	0.88	6
Xe-131	0.53	4	0.87	5	0.66	4	0.92	5	0.76	4	0.97	5
Total	7.01		13.67		8.70		14.65		10.11		15.53	

[a] Nuclides that contribute more than 0.1% to the total absorption rate.

Table 7 (continued)

	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
100-Year cooling												
Ag-109	0.13	16	0.42	11	0.18	16	0.44	11	0.22	17	0.46	11
Cs-133	0.45	5	0.92	5	0.60	5	1.02	5	0.73	5	1.10	5
Eu-151	0.23	10	0.34	15	0.27	11	0.36	15	0.30	12	0.38	15
Eu-153	0.17	13	0.58	9	0.26	12	0.63	9	0.33	11	0.68	9
Gd-155	0.26	9	1.22	4	0.36	8	1.25	4	0.45	8	1.28	3
Gd-157	0.03	22	0.10	21	0.04	22	0.11	22	0.04	22	0.11	22
La-139	0.04	20	0.10	22	0.05	21	0.11	21	0.07	21	0.12	21
Mo-95	0.17	14	0.35	14	0.23	13	0.39	14	0.28	13	0.43	13
Nd-143	0.78	3	1.33	2	0.95	3	1.44	2	1.09	3	1.53	2
Nd-145	0.22	11	0.45	10	0.29	10	0.51	10	0.36	10	0.56	10
Pd-105	0.06	18	0.19	18	0.08	18	0.21	18	0.10	18	0.23	18
Pd-108	0.04	21	0.16	19	0.05	20	0.17	19	0.07	20	0.18	19
Pr-141	0.05	19	0.13	20	0.07	19	0.15	20	0.09	19	0.16	20
Rh-103	0.88	2	1.63	1	1.09	2	1.69	1	1.24	1	1.73	1
Ru-101	0.12	17	0.29	16	0.17	17	0.34	16	0.22	16	0.38	16
Sm-147	0.27	8	0.38	12	0.34	9	0.42	12	0.40	9	0.45	12
Sm-149	1.21	1	1.33	3	1.17	1	1.27	3	1.14	2	1.21	4
Sm-150	0.14	15	0.37	13	0.20	15	0.40	13	0.24	14	0.43	14
Sm-151	0.18	12	0.28	17	0.21	14	0.30	17	0.23	15	0.31	17
Sm-152	0.32	7	0.66	8	0.41	7	0.70	8	0.49	7	0.74	8
Tc-99	0.36	6	0.75	7	0.49	6	0.83	7	0.59	6	0.90	7
Xe-131	0.53	4	0.89	6	0.67	4	0.95	6	0.77	4	0.99	6
Total	7.03		13.95		8.74		14.90		10.15		15.75	

[a] Nuclides that contribute more than 0.1% to the total absorption rate.

The fission product exhibiting the most rapid increase with burnup is the stable nuclide ^{155}Gd. Its precursor ^{155}Eu has a 4.75-year half-life, and so ^{155}Gd has not reached equilibrium even for a 70-GWd/t exposure. The importance of ^{155}Gd increases by an order of magnitude between 20 and 60 GWd/t for cooling times greater than about 5 years, and it is one of the larger fission-product absorbers in high-burnup fuel. Another stable nuclide, ^{153}Eu, exhibits a similar, although less-pronounced effect, as its precursor ^{153}Gd and has a 241.6-day half-life. However, ^{153}Eu ranks considerably lower than ^{155}Gd in high-burnup fuel for cooling times more than 5 years.

The criticality safety nuclide rankings were recomputed for the PWR fuel using a SAS2 model in which the specific power was 25 MW/t. In general, the nuclide rankings were not affected, except that (rarely) the order of two nuclides could be reversed. In those cases, the absorption rates for the nuclides whose order changed were essentially identical. The only noticeable difference in the actinides for decay times from 5 to 100 years was an increase of about 6% in the absorption fraction for ^{238}Pu. The amount of ^{238}Pu increases because its precursor ^{237}U (produced both by capture from ^{235}U to ^{236}U and from (n,2n) from ^{238}U) undergoes relatively more decays to ^{237}Np when the power (and therefore flux) is lower. The biggest changes in fission-product absorption rates were for the stable ^{149}Sm (–11%) and ^{157}Gd (–15%) nuclides that have large capture cross sections.

4 DECAY HEAT

The fractional contributions of individual radionuclides to decay-heat production in a fuel assembly were evaluated for extended-burnup fuel. These relative contributions, expressed as the percent of the total decay-heat production, were used to rank individual nuclides by their importance to thermal analysis. A previous ranking study[9] examined PWR fuel in the range of 3-wt % ^{235}U, 20-GWd/t burnup to 4.5-wt %^{235}U, 50-GWd/t burnup, with an emphasis on ranking as a function of cooling time. In this study, both PWR and BWR fuel assemblies up to 5-wt % ^{235}U, 70-GWd/t burnup were studied, with an emphasis on the variation of nuclide importance to decay heat with increasing burnup.

The decay-heat-rate profiles (W/t) as a function of burnup are shown in Figures 7 and 8 for 5-wt % ^{235}U PWR fuel at cooling times of 5 and 100 years. The decay-heat rates for the 5-wt % ^{235}U BWR fuel were found to have similar burnup profiles and essentially the same decay-heat values and are therefore not illustrated. The figure shows that the actinides become an increasingly larger component of the total decay-heat rate with burnup. The fission product component increases linearly with burnup, while the actinide component displays a quasi-exponential increase, except at 100-years decay, where it is nearly linear with burnup.

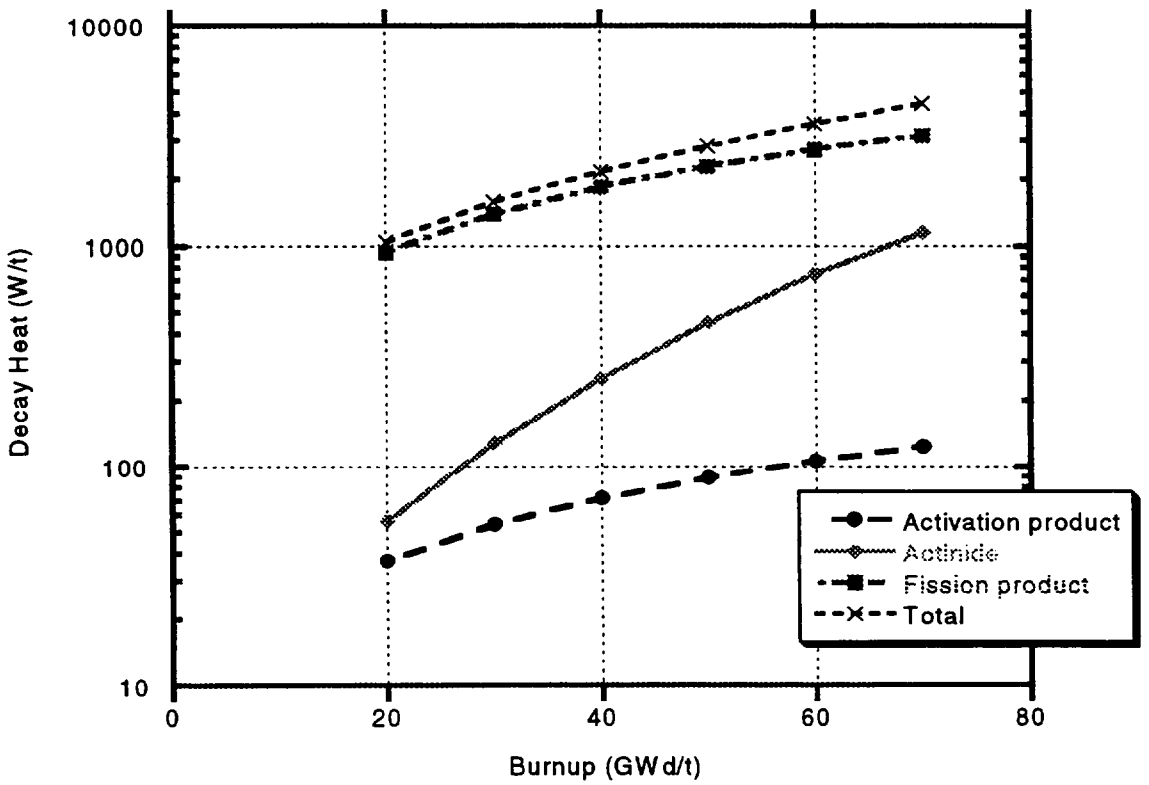


Figure 7 Decay-heat-rate profiles for 5-wt % ^{235}U PWR fuel, 5-year cooling

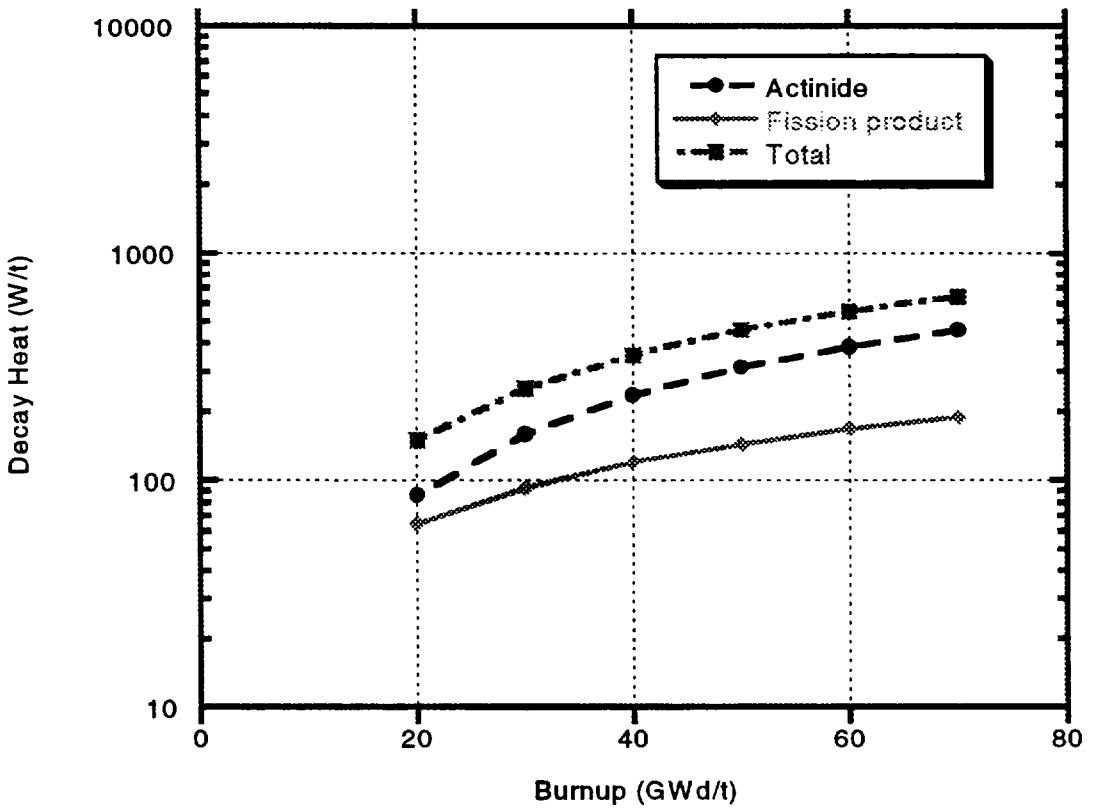


Figure 8 Decay-heat-rate profiles for 5-wt % ^{235}U PWR fuel, 100-year cooling

4.1 Decay-Heat Rankings

The fractional contributions of actinide, fission product, and light-element nuclides to decay-heat generation were computed for the enrichment and burnup combinations and cooling times of interest to obtain the radionuclide importance used in the ranking procedure. In terms of relative rankings, the PWR and BWR results are essentially identical; therefore, plots are shown for only the PWR assembly.

The nuclide importance, expressed as the fractional contribution of each nuclide to the total decay-heat rate, is illustrated in Figures 9 and 10 for 5-wt % ^{235}U PWR fuel. The trends for the 3-wt % ^{235}U and 4-wt % ^{235}U PWR assemblies are quite similar, and therefore only the results for 5-wt % enrichment are plotted. The fractional contributions and rankings for decay-heat generation for the PWR and BWR fuel assemblies are given in Table 8 and Table 9, respectively, for decay times of 5 and 100 years. The tables include enrichments of 3, 4, and 5 wt % and a range of burnup values.

The PWR results for 3 wt %, 20 GWd/t are consistent with previous investigations in Ref. 9. The dominant radionuclides in this regime, in order of descending importance rankings are ^{90}Y, ^{137m}Ba, ^{134}Cs, ^{106}Rh, ^{144}Pr, ^{137}Cs, ^{90}Sr, and ^{60}Co. The most dramatic change in the importance rankings with increasing burnup is the increase in the actinide component, attributed to ^{244}Cm and ^{238}Pu. This increase is readily seen in Figures 9 and 10. A significant increase is also observed in the contribution from ^{134}Cs, although it is only important at cooling times less than about 10 years. At cooling times of about 100 years the majority of the decay heating is produced by relatively few radionuclides: ^{241}Am, ^{238}Pu, ^{137m}Ba, and ^{90}Y. The rankings of these nuclides change with burnup; however, their aggregate contribution to the total decay-heat rate remains relatively constant. Combined, they generate between 80 and 85% of the total decay heat. The largest change in radionuclide importance occurs for ^{238}Pu, which contributes 7% of the decay heat for 3 wt %, 20 GWd/t (ranks fourth), and 28% for 5 wt %, 70 GWd/t (ranks second).

The decay-heat nuclide rankings were recomputed for the PWR using a SAS2 model in which the specific power was 25 MW/t. In general, the nuclide rankings were not affected, except that (rarely) the order of two nuclides was reversed. In those cases, the decay-heat rates for the nuclides whose order changed were essentially identical. The noticeable difference in the actinides for decay times from 5 to 100 years includes an increase of about 5% in the decay heat for ^{238}Pu, increases as large as 39% for ^{242}Cm, and increases as large as 24% for ^{243}Cm. The amount of ^{238}Pu increases because its precursor ^{237}U (produced both by capture from ^{235}U to ^{236}U and from (n,2n) from ^{238}U) undergoes relatively more decays to ^{237}Np when the power (and therefore flux) is lower. The amount of decay heat from ^{242}Cm and ^{243}Cm is not great, so the large percentage increases are not significant. Furthermore, the total actinide decay heat increases no more than about 2%. The total decay heat from the fission products is, of course, lower for the 25 MW/t case, because the decay heat is nearly proportional to the assembly-specific power for short cooling times. However, the order of fission-product nuclide rankings is essentially unaffected.

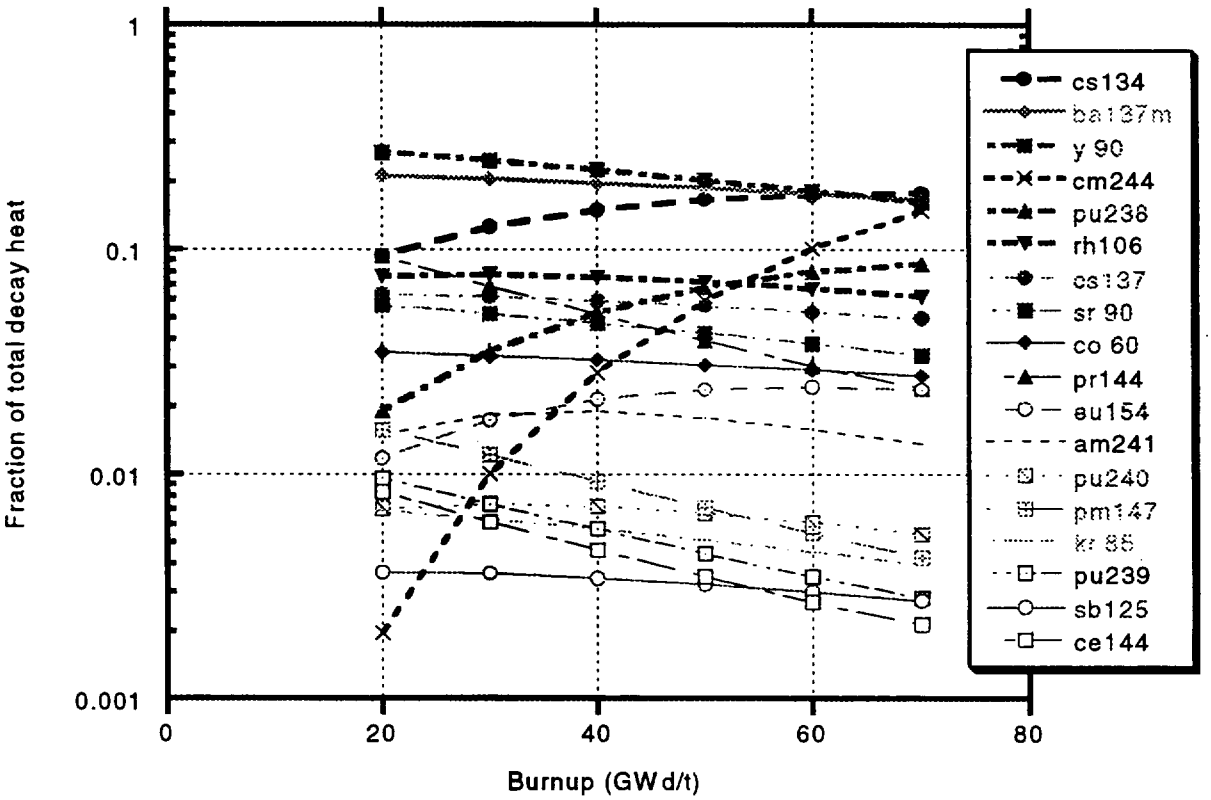


Figure 9 Fraction of decay-heat generation for 5-wt % ^{235}U PWR fuel, 5-year cooling

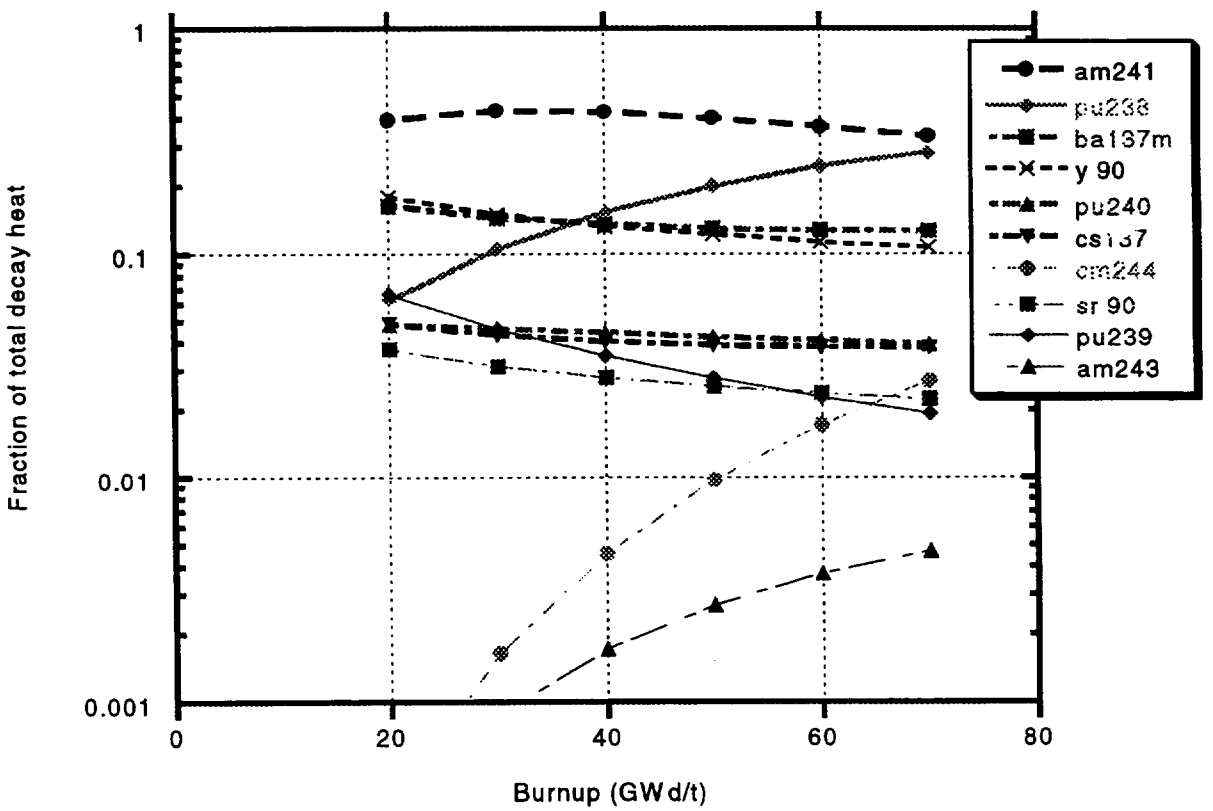


Figure 10 Fraction of decay-heat generation for 5-wt % ^{235}U PWR fuel, 100-year cooling

Table 8 Radionuclides decay-heat rankings for PWR fuel

	Burnup / enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Am-241	1.96	10	1.54	12	1.96	10	1.44	12	1.90	12	1.36	12
Ba-137m	20.10	2	16.81	2	20.01	2	16.68	2	19.75	2	16.51	2
Ce-144	0.75	15	0.29	18	0.58	17	0.25	18	0.46	17	0.21	18
Cm-244	0.69	16	13.25	4	1.66	12	14.10	4	2.84	10	14.91	4
Co-60	4.68	8	3.80	8	3.81	9	3.18	9	3.23	9	2.74	9
Cs-134	10.88	3	17.45	1	13.43	3	17.70	1	15.01	3	17.74	1
Cs-137	6.02	6	5.03	7	5.99	6	4.99	7	5.92	5	4.95	7
Eu-154	1.41	11	2.26	11	1.84	11	2.32	11	2.15	11	2.38	11
Kr-85	0.60	17	0.39	15	0.59	16	0.40	15	0.57	15	0.40	15
Pm-147	1.36	12	0.53	14	1.11	13	0.47	14	0.93	13	0.43	14
Pr-144	8.42	5	3.24	9	6.49	5	2.77	10	5.15	7	2.40	10
Pu-238	2.38	9	6.43	6	3.83	8	7.59	5	5.24	6	8.66	5
Pu-239	0.87	14	0.34	16	0.68	15	0.31	16	0.57	16	0.28	16
Pu-240	0.94	13	0.68	13	0.82	14	0.60	13	0.72	14	0.54	13
Pu-241	0.19	19	0.14	19	0.18	19	0.13	19	0.17	19	0.12	19
Rh-106	10.08	4	8.57	5	8.74	4	7.19	6	7.55	4	6.17	6
Sb-125	0.41	18	0.34	17	0.38	18	0.30	17	0.34	18	0.27	17
Sr-90	4.84	7	3.20	10	4.78	7	3.32	8	4.72	8	3.39	8
Y-90	23.08	1	15.27	3	22.82	1	15.83	3	22.48	1	16.15	3
Total (W/t)	1090		3170		1630		3790		2180		4430	
100-Year cooling												
Am-241	46.58	1	39.22	1	44.98	1	35.98	1	42.48	1	33.32	1
Am-243	0.08	10	0.50	10	0.13	10	0.49	10	0.17	10	0.47	10
Ba-137m	13.93	2	12.99	3	13.69	2	12.84	3	13.47	3	12.62	3
Cm-244	0.11	9	2.42	7	0.27	9	2.56	7	0.46	9	2.69	7
Cs-137	4.17	7	3.89	6	4.10	7	3.85	6	4.04	6	3.78	6
Pu-238	7.00	4	21.09	2	11.13	4	24.80	2	15.20	2	28.09	2
Pu-239	5.43	6	2.39	8	4.20	6	2.13	9	3.49	7	1.92	9
Pu-240	5.79	5	4.90	5	4.99	5	4.35	5	4.43	5	3.92	5
Sr-90	2.90	8	2.14	9	2.83	8	2.21	8	2.78	8	2.24	8
Y-90	13.84	3	10.22	4	13.52	3	10.55	4	13.28	4	10.68	4
Total (W/t)	175		456		265		549		356		645	

[a] Greater than 0.1% of total decay-heat-generation rate for any burnup/enrichment combination.

Table 9 Radionuclides decay heat rankings for BWR fuel

	Burnup / Enrichment											
	20 GWd/t 3 wt %		50 GWd/t 3 wt %		30 GWd/t 4 wt %		60 GWd/t 4 wt %		40 GWd/t 5 wt %		70 GWd/t 5 wt %	
Nuclide[a]	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Am-241	1.94	9	1.43	11	1.88	10	1.32	11	1.79	11	1.24	11
Ba-137m	20.72	2	17.18	2	20.48	2	17.04	2	20.18	2	16.87	2
Ce-144	0.78	15	0.30	18	0.60	17	0.25	18	0.47	17	0.22	18
Cm-244	0.78	16	13.94	4	1.80	11	14.68	4	2.99	9	15.38	4
Co-60	1.59	10	1.30	12	1.29	12	1.09	12	1.09	12	0.94	12
Cs-134	11.40	3	17.97	1	13.84	3	18.15	1	15.33	3	18.13	1
Cs-137	6.21	6	5.15	7	6.14	6	5.10	7	6.05	5	5.05	7
Eu-154	1.49	11	2.25	10	1.90	9	2.29	10	2.19	10	2.32	10
Kr-85	0.62	17	0.40	15	0.61	16	0.41	15	0.59	15	0.41	15
Pm-147	1.39	12	0.53	14	1.13	13	0.48	14	0.95	13	0.43	14
Pr-144	8.70	5	3.32	8	6.68	5	2.84	9	5.29	7	2.46	9
Pu-238	2.56	8	6.49	6	3.99	8	7.56	5	5.35	6	8.54	5
Pu-239	0.87	14	0.33	17	0.67	15	0.29	17	0.55	16	0.26	17
Pu-240	0.97	13	0.71	13	0.84	14	0.63	13	0.75	14	0.57	13
Pu-241	0.19	19	0.13	19	0.18	19	0.12	19	0.16	19	0.11	19
Rh-106	10.32	4	8.71	5	8.82	4	7.29	6	7.56	4	6.25	6
Sb-125	0.42	18	0.35	16	0.38	18	0.31	16	0.35	18	0.28	16
Sr-90	5.00	7	3.29	9	4.92	7	3.41	8	4.85	8	3.48	8
Y-90	23.83	1	15.70	3	23.47	1	16.28	3	23.12	1	16.62	3
Total (W/t)	1060		3090		1590		3710		2130		4330	
100-Year cooling												
Am-241	45.56	1	37.18	1	43.30	1	33.93	1	40.55	1	31.31	1
Am-243	0.08	10	0.51	10	0.13	10	0.49	10	0.17	10	0.47	10
Ba-137m	14.19	2	13.48	3	14.08	2	13.37	3	13.94	3	13.20	3
Cm-244	0.13	9	2.58	7	0.29	9	2.72	7	0.49	9	2.84	7
Cs-137	4.25	7	4.04	6	4.22	6	4.01	6	4.18	6	3.95	6
Pu-238	7.46	4	21.60	2	11.64	4	25.20	2	15.71	2	28.34	2
Pu-239	5.36	6	2.29	8	4.11	7	2.03	9	3.40	7	1.83	9
Pu-240	5.94	5	5.21	5	5.18	5	4.64	5	4.65	5	4.20	5
Sr-90	0.01	8	2.24	9	2.93	8	2.32	8	2.90	8	2.36	8
Y-90	14.13	3	10.66	4	13.96	3	11.06	4	13.82	4	11.25	4
Total (W/t)	172		439		257		526		343		616	

[a] Greater than 0.1 % of total decay-heat-generation rate for any burnup/enrichment combination.

5 RADIATION SHIELDING

The fractional contributions of individual radionuclides to the radiation dose rates from a transport/storage cask were evaluated for extended-burnup fuel and used to rank nuclides and obtain their importance to shielding analyses. Three different cask models were used to assess radionuclide importance: a carbon steel transport cask, a lead transfer cask, and a concrete storage cask. The variation in nuclide importance to shielding as a function of cooling time has been previously studied in Ref. 9. This study does not attempt to compile these earlier findings but instead focuses primarily on the changes in nuclide importance for extended-burnup fuels.

The typical variation in the neutron and gamma dose rate as a function of increasing burnup is shown in Figures 11 and 12 for the steel cask model and 5-wt %-enrichment fuel. The figure illustrates the rapid increase in the neutron dose rate component with burnup relative to the gamma-ray component. While the gamma dose rate (D_g) increases nearly linearly with burnup, the neutron dose rate (D_n) increases approximately as the burnup (B) to the power of 4 ($D_n \propto B^4$), indicating that neutrons represent an increasingly important component of the dose rate relative to gamma rays in cask shielding analyses involving high-burnup fuel.

The profiles in Figures 11 and 12 are representative of the other cask designs and fuel enrichments, although lower-enrichment fuel produces higher absolute dose rates, attributed to the fact that lower-enrichment fuel (lower fissile content) requires exposure to a greater neutron fluence per metric tonne (t) of uranium to achieve the same burnup as higher-enrichment fuel. As a result, for a given burnup the actinide inventory generated by neutron capture of ^{238}U increases by an amount nearly inversely proportional to the enrichment. The effect of enrichment is most evident at longer cooling times and higher burnups, where the actinides tend to become increasingly important.

The relative importance of neutrons with burnup is illustrated in Figures 13–15, which shows the neutron dose fraction (fraction of the total dose rate) as a function of burnup for 5-wt % PWR fuel for concrete, steel, and lead cask models. The different cask models show similar trends, although the magnitude of the neutron dose component varies by cask design. For the steel cask model, the neutron component increases dramatically with burnup, from less than 5% at 20 GWd/t to nearly 40% at 60 GWd/t after a 20-year cooling time. For the lead cask, the increase is from about 5% at 20 GWd/t to more than 60% of the total dose rate at 60 GWd/t. For the concrete cask the relative increases are similar; however, the neutron component is significantly lower (e.g., <10%) due to the greater effectiveness of concrete as a neutron shield.

5.1 Neutron Sources

The neutron source term was investigated further due to the observed rapid increase in importance for high-burnup fuel. The spontaneous fission and (α,n) neutron source contributions to the total neutron source term are shown in Figures 16 and 17 and illustrate the dramatic increase in the spontaneous fission source with increasing burnup. The (α,n) source increases nearly linearly with burnup and represents an important component only at relatively low burnup (<40 GWd/t) and cooling times near 100 years. The (α,n) and spontaneous-fission components contribute nearly equally at low burnup (20 – 30 GWd/t) and about 100-years cooling, with the (α,n) component decreasing to about 10% of the total neutron source at higher burnup (70 GWd/t). The neutron-source components as a function of *cooling time* for burnups of 27 and 60 GWd/t are illustrated in Figures 18 and 19 and show the relatively limited range of cooling times for which the (α,n) component can become important (range between about 80 and 1000 years). For high-burnup fuel the (α,n) source is a minor component of the neutron source term for all cooling times up to 10^5 years.

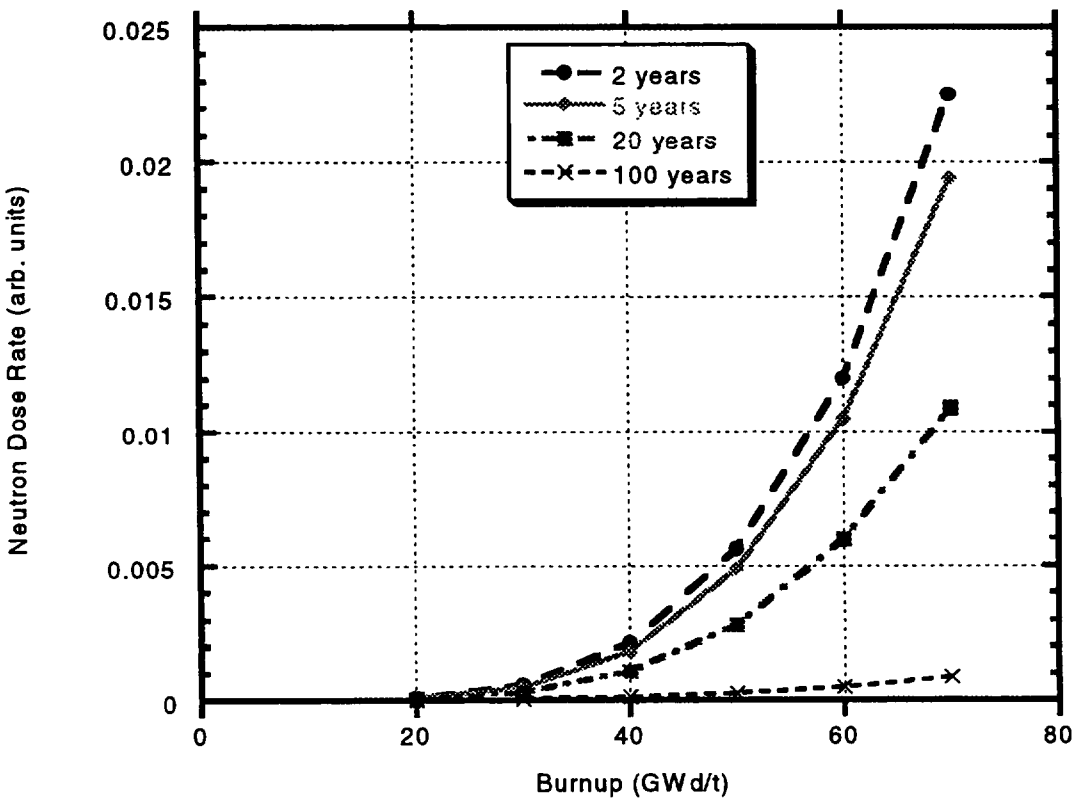


Figure 11 Neutron dose rates for steel cask with 5-wt % PWR fuel

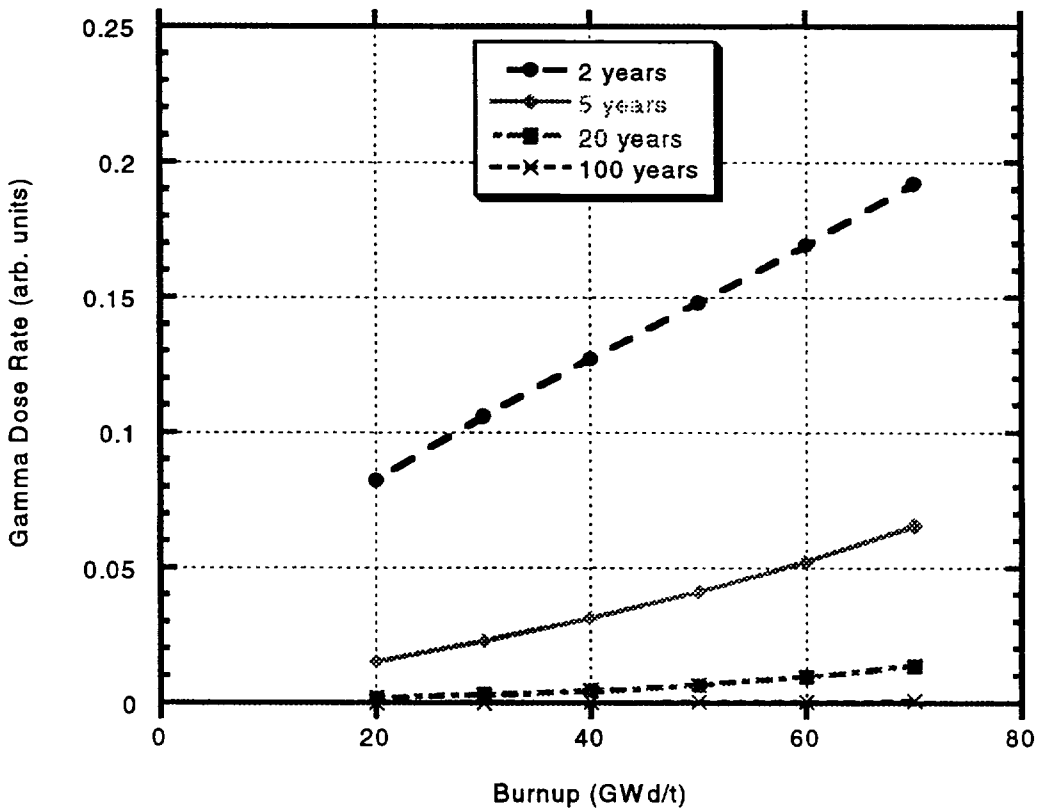


Figure 12 Gamma dose rates for steel cask with 5-wt % PWR fuel

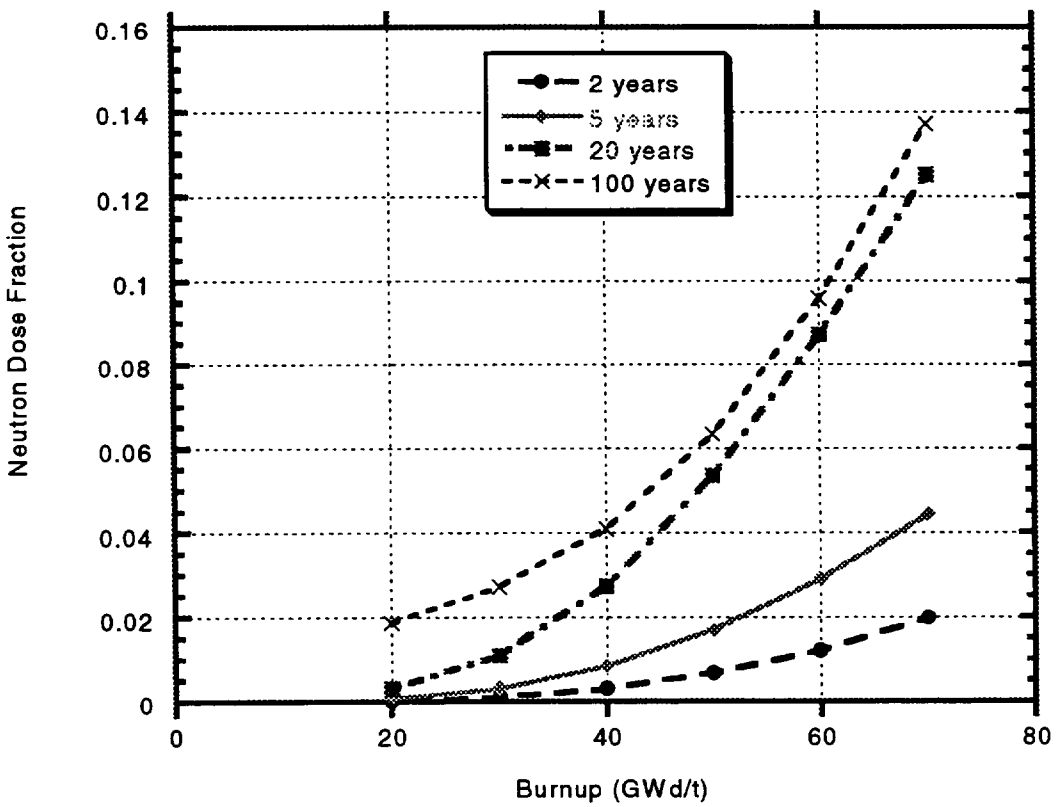


Figure 13 Relative contribution of neutrons to the total dose rate for 5-wt % PWR fuel, concrete storage cask

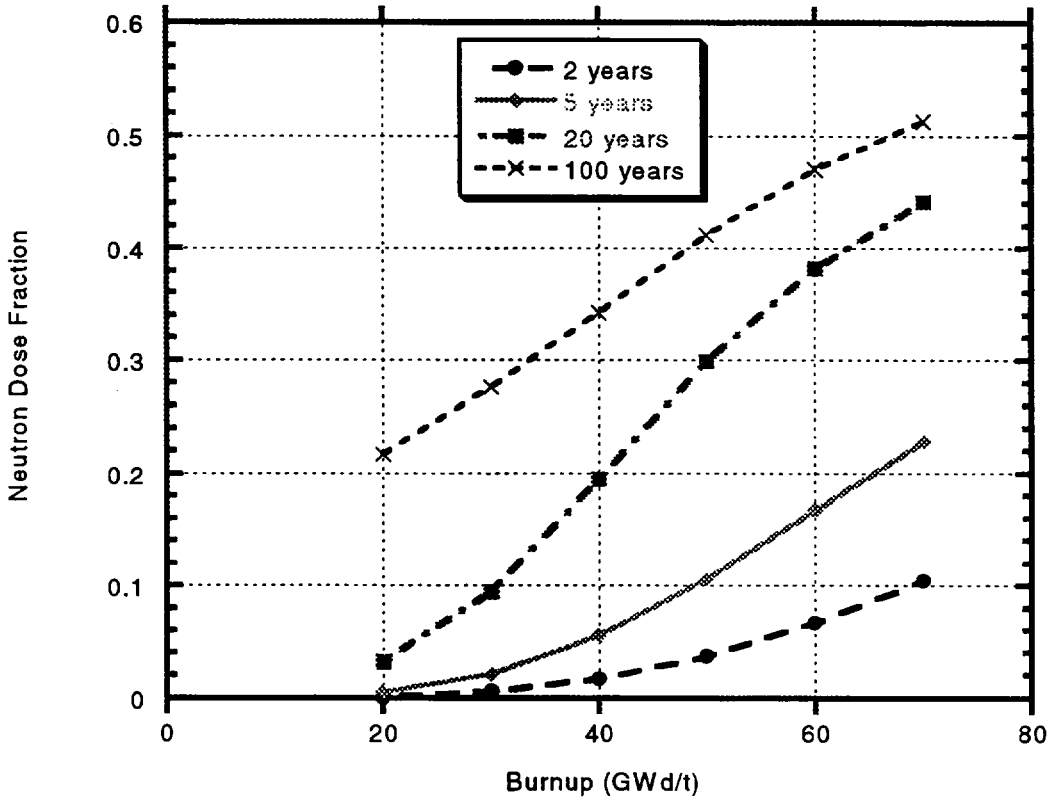


Figure 14 Relative contribution of neutrons to the total dose rate for 5-wt % PWR fuel, steel transport cask

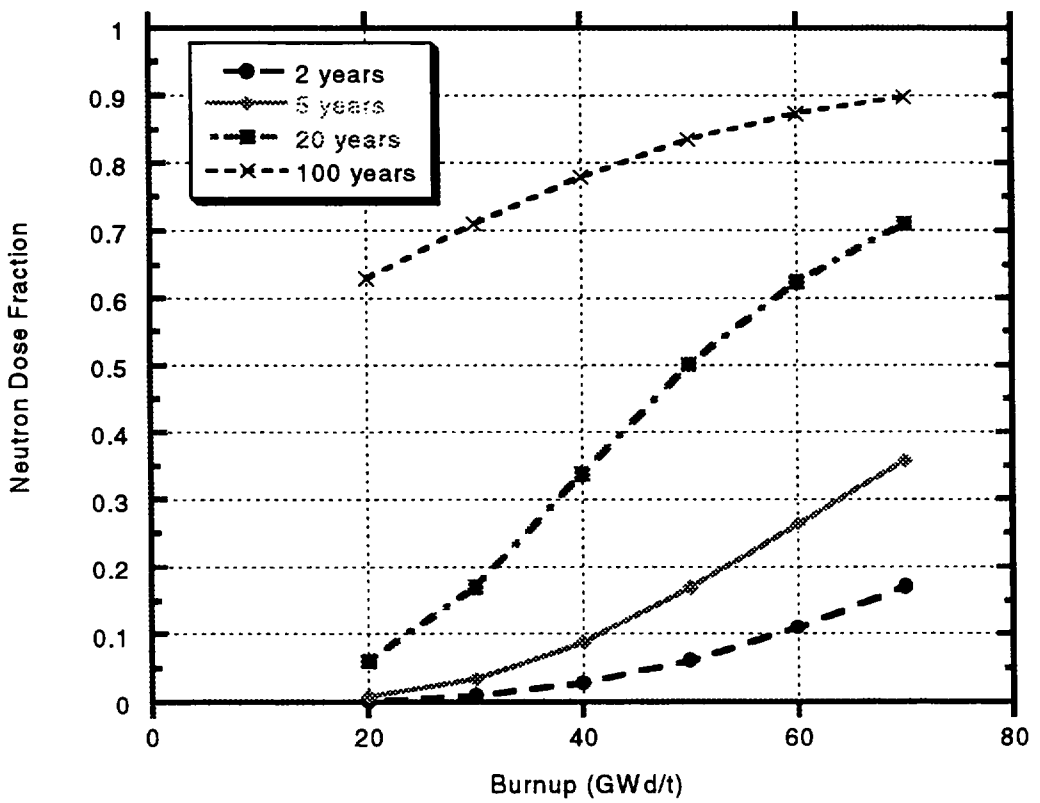


Figure 15 Relative contribution of neutrons to the total dose rate for 5-wt % PWR fuel, lead cask

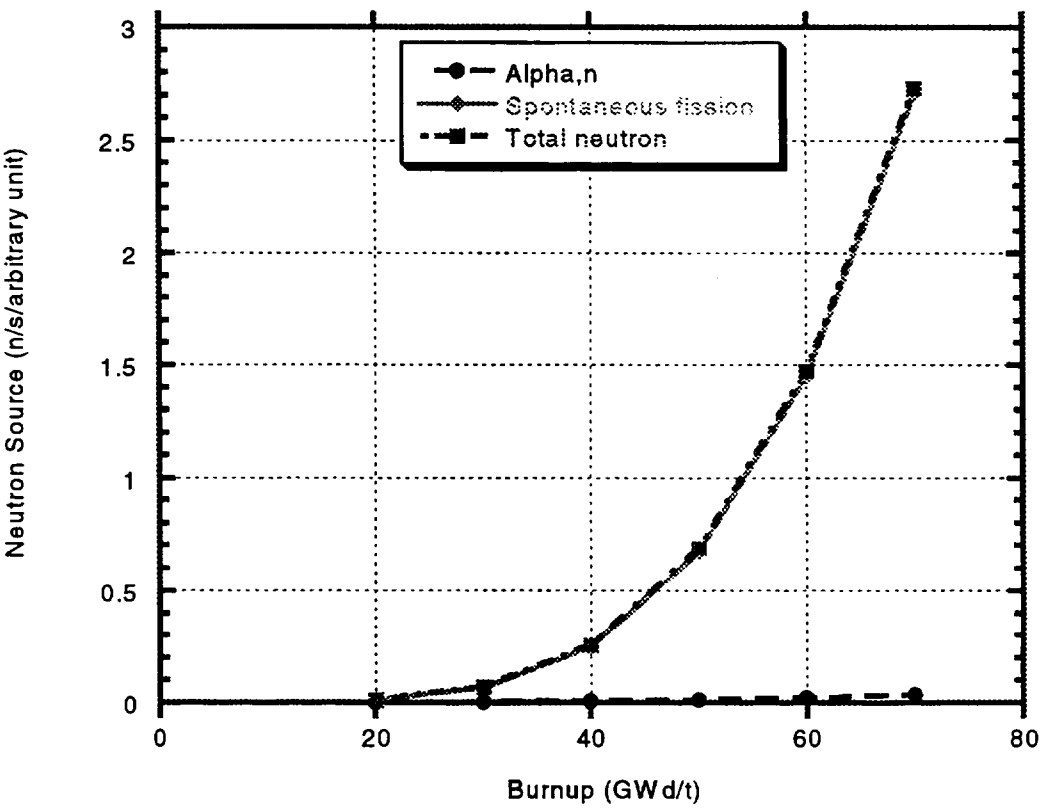


Figure 16 Contribution of (α,n) and spontaneous fission neutron sources for 5-wt % PWR fuel, 5-year cooling

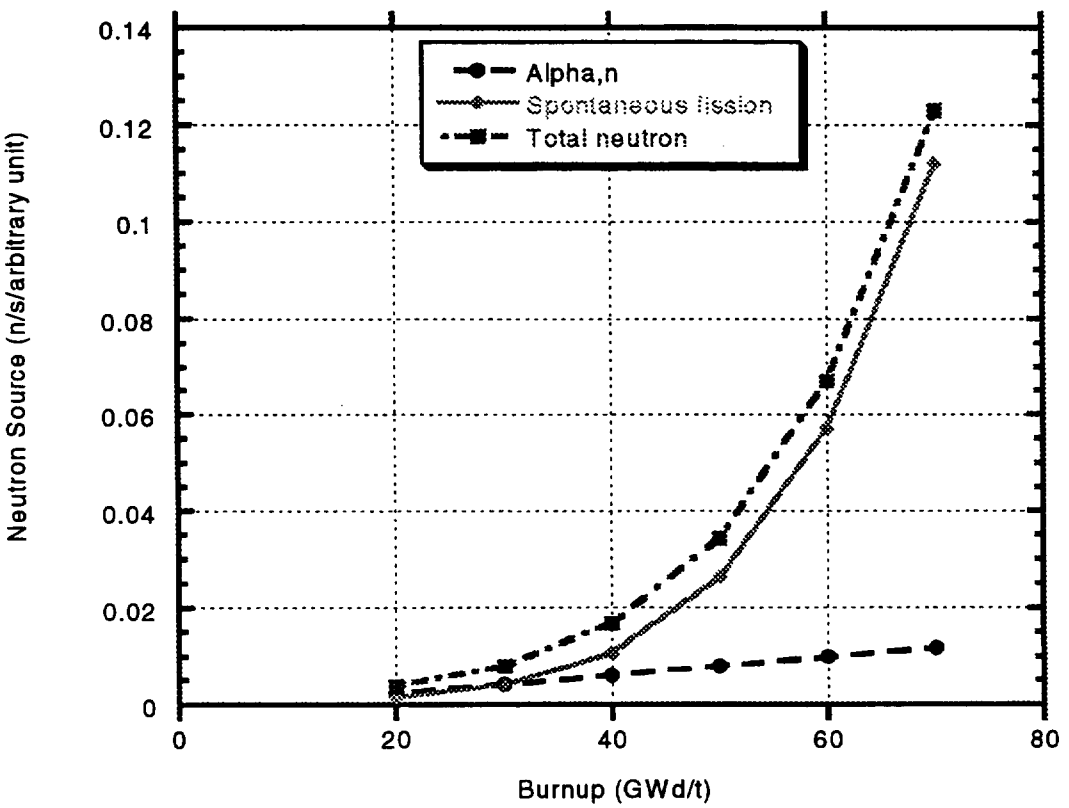


Figure 17 Contribution of (α,n) and spontaneous fission neutron sources for 5-wt % PWR fuel, 100-year cooling

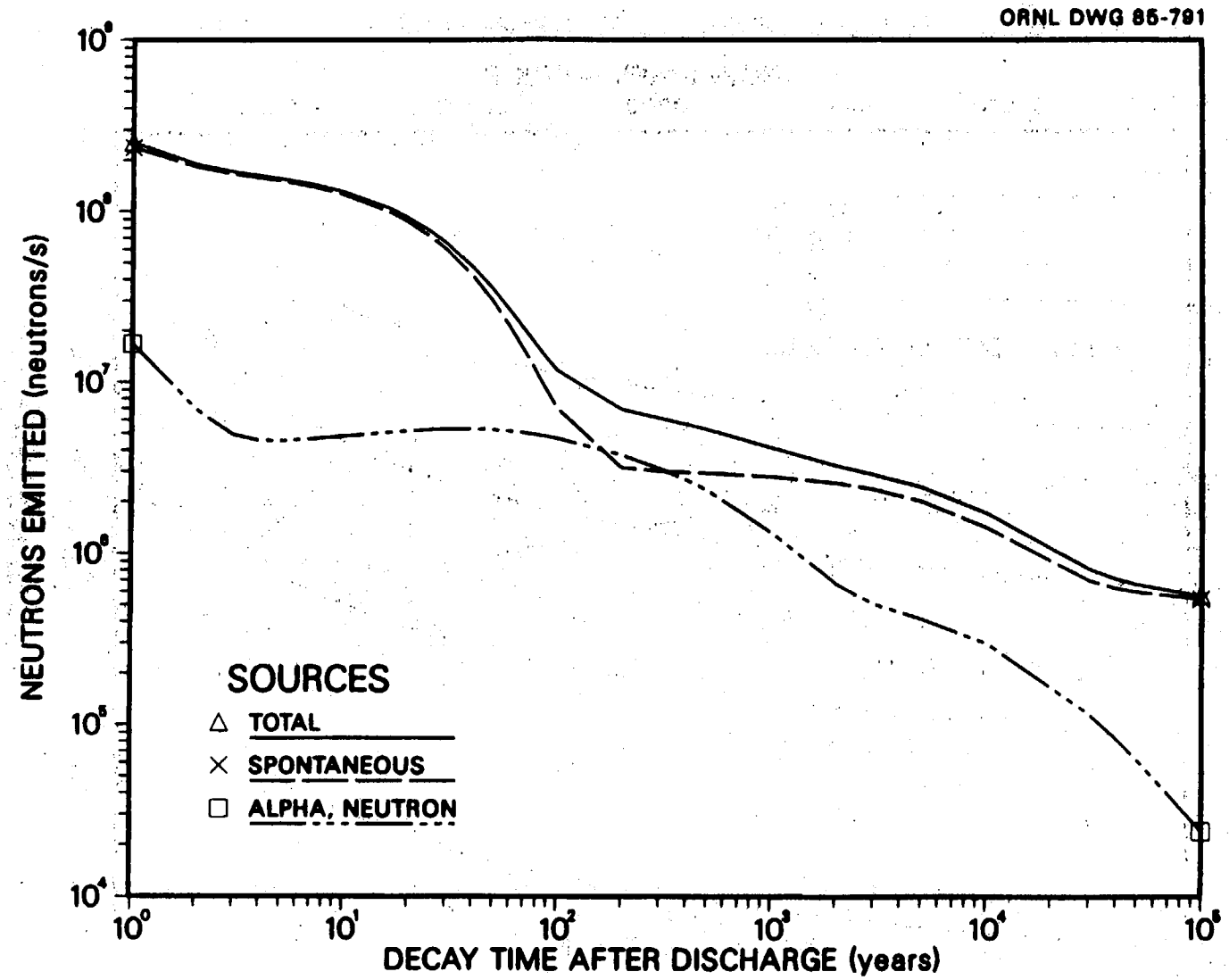


Figure 18 Neutron source from 1 metric tonne (t) of PWR fuel; 27.5 GWd/t (from Ref. 17)

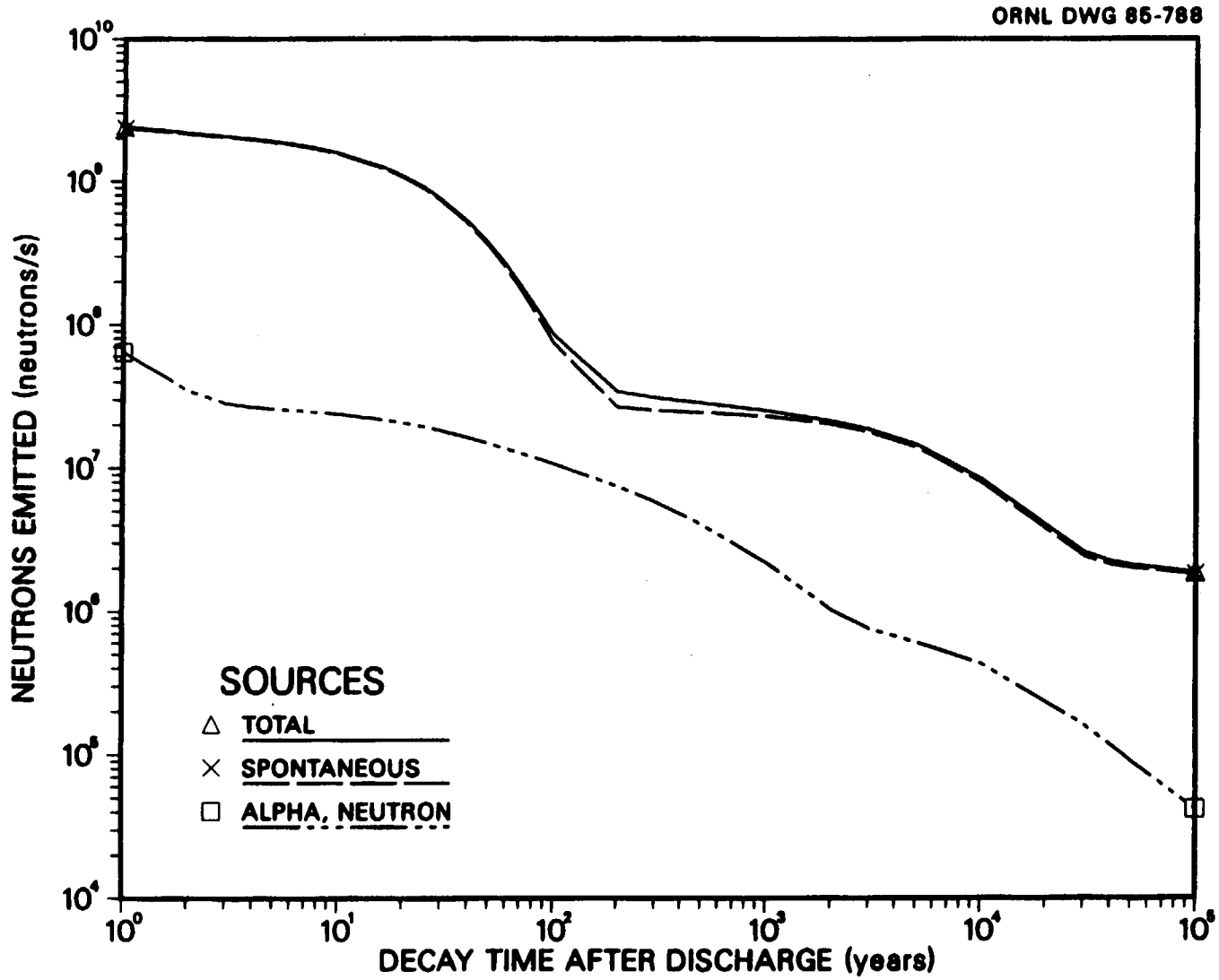


Figure 19 Neutron source from 1 metric tonne (t) of PWR fuel; 60 GWd/t (from Ref. 17)

The neutron source was further evaluated to identify the major actinides that contribute to the spontaneous-fission neutron source. The relative contributions of the dominant actinides are shown in Figures 20 and 21. A common feature for all cooling times investigated is the dominance of ^{244}Cm at high burnup and the significant increase in the relative contribution of ^{244}Cm (and ^{246}Cm) and the decreasing relative importance of the Pu isotopes to the dose rate with increasing burnup. At lower burnups the contribution from ^{240}Pu is significant (particularly at 100-years cooling, where it is the dominant nuclide) although its relative importance diminishes rapidly with increasing burnup as the curium isotopes build up.

These results demonstrate the high relative importance of neutrons, specifically the spontaneous fission neutrons, to cask shielding studies for high-burnup fuel and cooling times greater than 5 years. For shorter cooling times the fission products represent the higher fraction of the dose. However, neutrons only become a dominant fraction of the total dose rate for the steel and lead cask designs analyzed in this study, which were less effective than concrete in shielding neutrons. The concrete storage cask has the smallest neutron dose rate component (<20% of the total dose for all burnups and cooling times), and the lead cask has the largest neutron dose component (>60% after 100-years cooling).

5.2 Radionuclide Rankings

The fractional contribution of individual radionuclides to the total dose rate for the three different cask designs was calculated for the enrichment and burnup combinations and cooling-time ranges of interest to obtain the importances used in the ranking process. A review of the dominant nuclides for each of the application regimes (cask design, fuel enrichment, burnup and cooling time) found that the majority of the dose rate was attributed to relatively few nuclides for all regimes studied. The nuclides listed in Table 10 (11 actinides, 8 fission products, and 1 activation product) account for more than 99% of the total dose rate for all cask designs and fuel regimes (enrichments and burnup) analyzed.

Table 10 Most important nuclides in cask shielding calculations

U-238	Pu-242	Cm-246	Y-90	Ba-137m
Pu-238	Am-241	Cm-248	Rh-106	Eu-152
Pu-239	Cm-242	Cf-252	Ag-110m	Eu-154
Pu-240	Cm-244	Co-60	Cs-134	Pr-144

The nuclide importance, expressed as the fractional contribution of each nuclide to the total dose rate at the cask surface, is illustrated in Figures 22 and 23 (steel cask), Figures 24 and 25 (concrete cask), and Figures 26 and 27 (lead cask) for 5-wt % PWR spent fuel. The results are presented as a function of increasing burnup and for cooling times of 5 years and 100 years. The fractional contributions and ranking for the dominant radionuclides (>1%) are listed in Table 11 for moderate- and extended-enrichment and high-burnup PWR fuel (5-wt % and 70-GWd/t) for 5- and 100-year cooling times. The lower-burnup results (3-wt % and 20-GWd/t) are included for comparison and as verification with previous calculations in Ref. 9. A comparison of the dominant nuclides and their contributions to the total dose rate for the lower-enrichment and low-burnup fuel is consistent with previous ranking studies that assessed more conventional-burnup fuel.

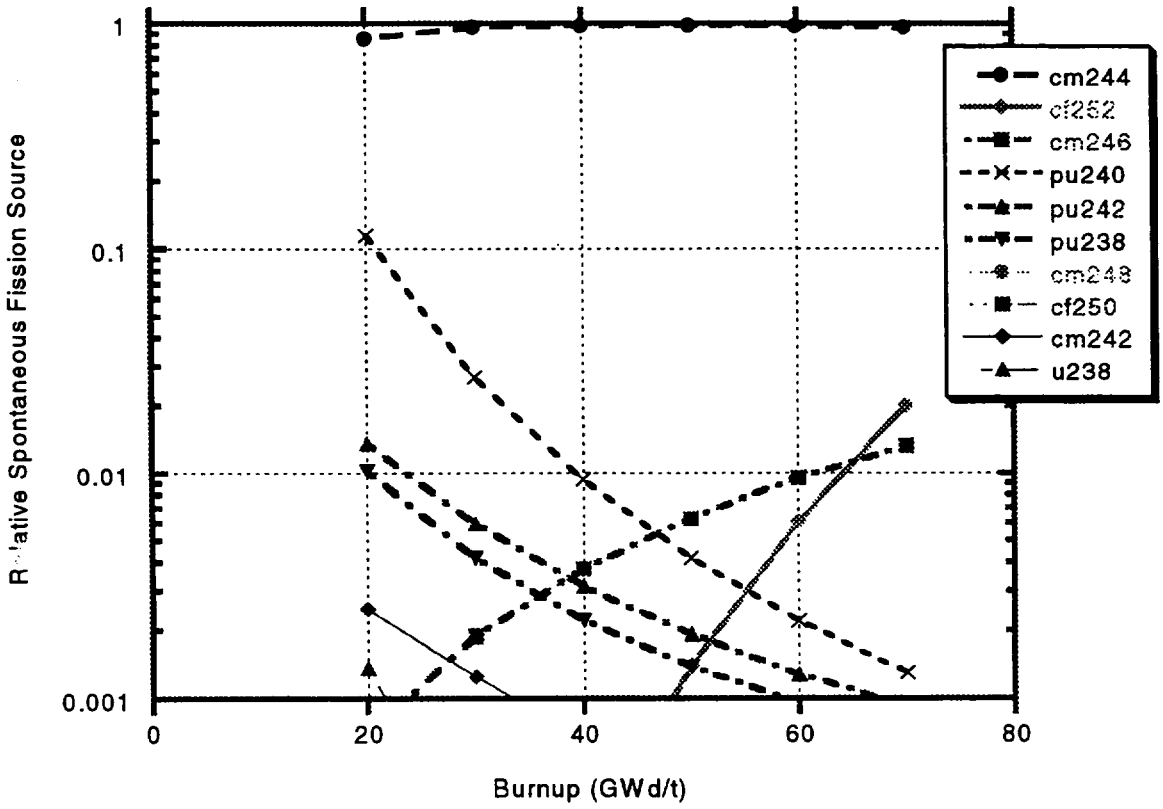


Figure 20 Relative importance of the dominant spontaneous fission actinides for 5-wt % PWR fuel, 5-year cooling

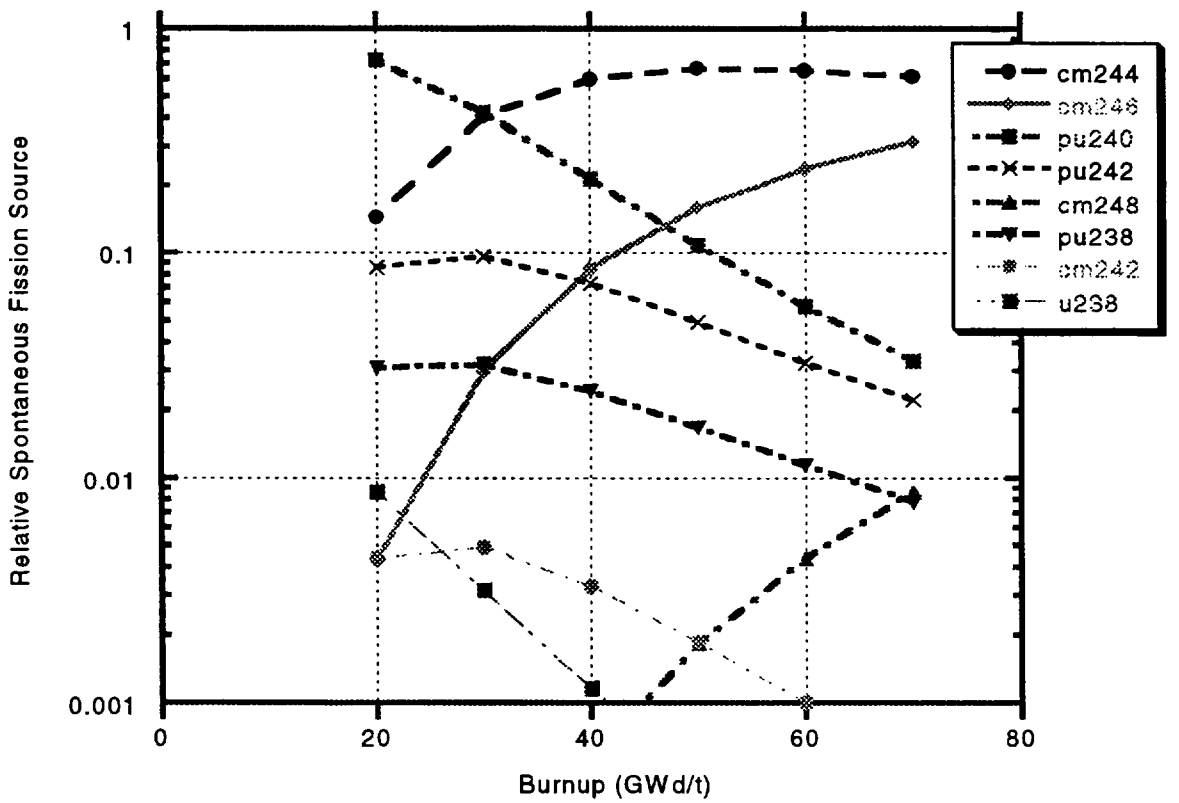


Figure 21 Relative importance of the dominant spontaneous fission actinides for 5-wt % PWR fuel, 100-year cooling

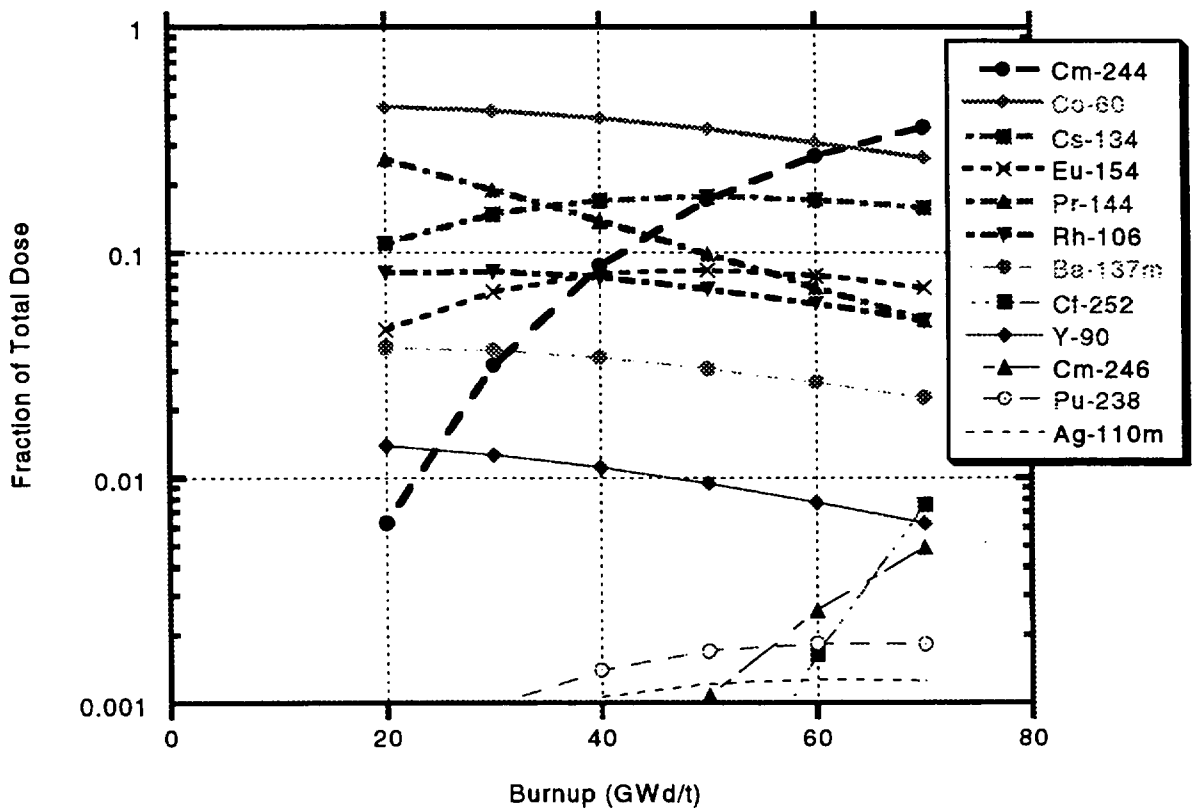


Figure 22 Fraction of total dose from dominant nuclides in 5-wt % PWR fuel; steel transport cask, 5-year cooling

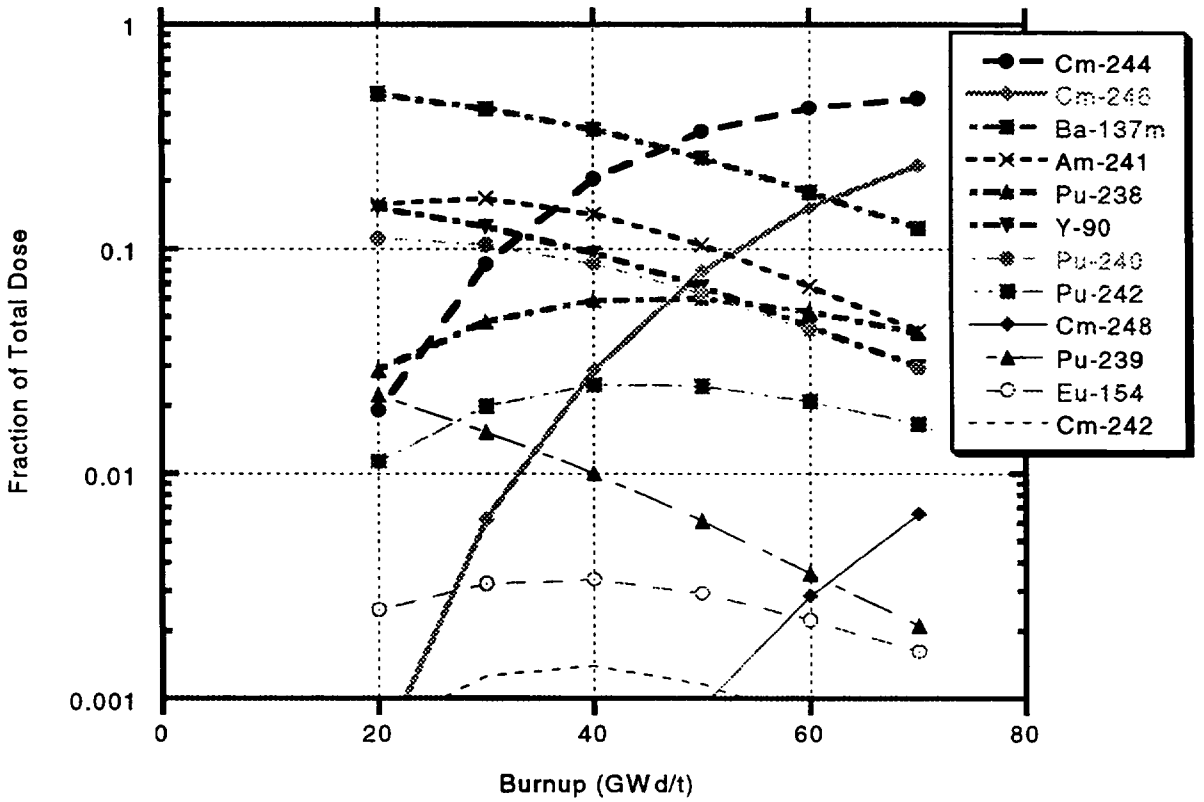


Figure 23 Fraction of total dose from dominant nuclides in 5-wt % PWR fuel; steel transport cask, 100-year cooling

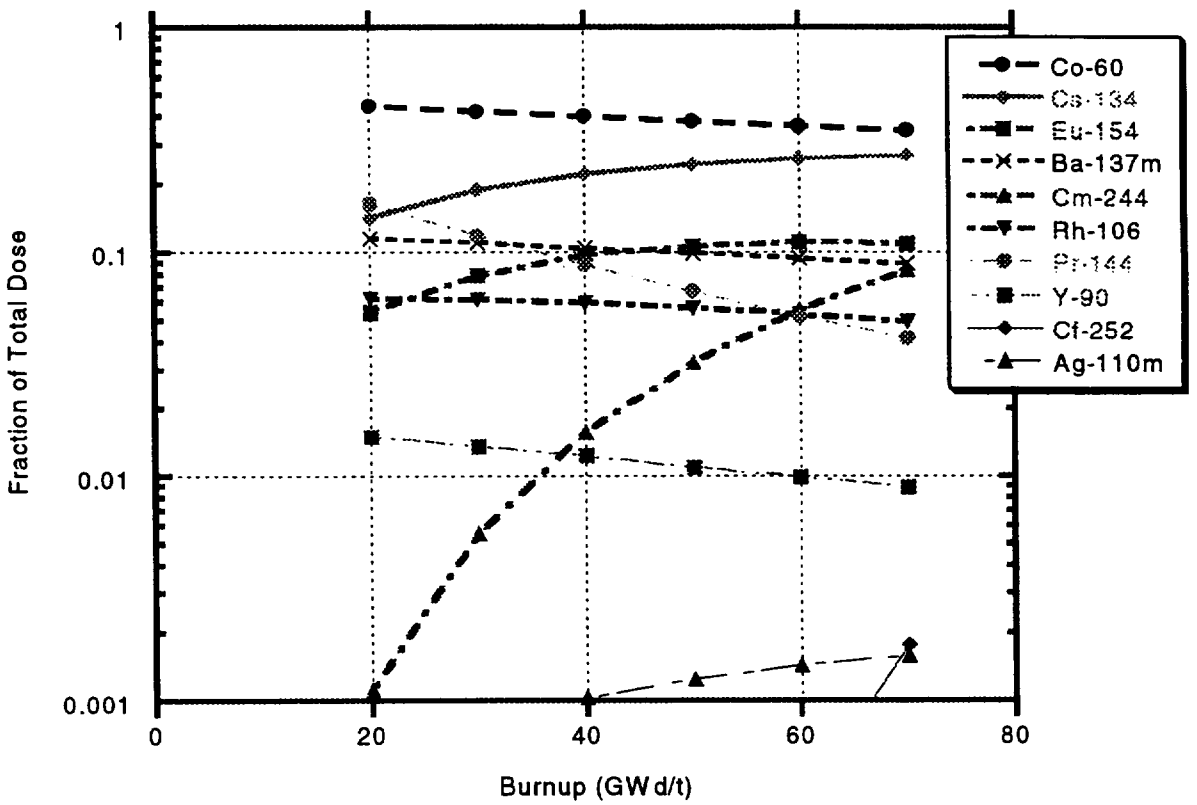


Figure 24 Fraction of total dose from dominant nuclides in 5-wt % PWR fuel; concrete storage cask, 5-year cooling

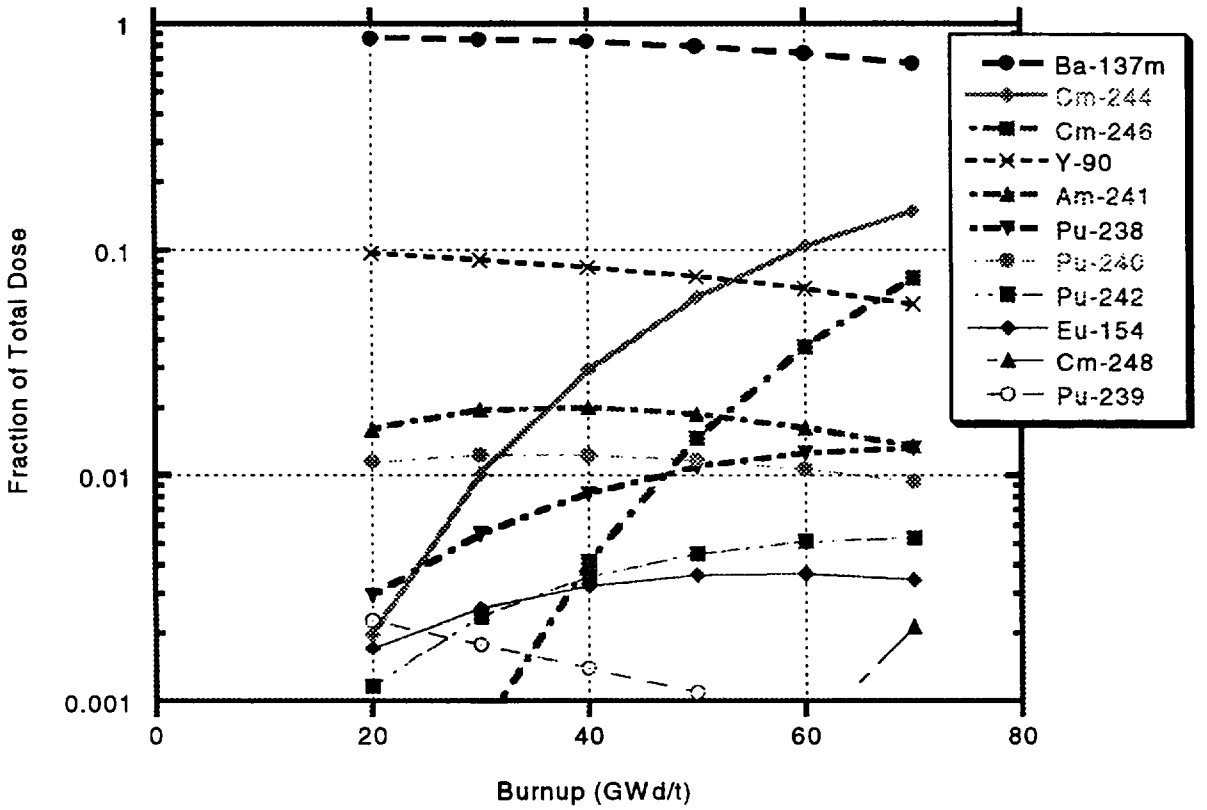


Figure 25 Fraction of total dose from dominant nuclides in 5-wt % PWR fuel; concrete storage cask, 100-year cooling

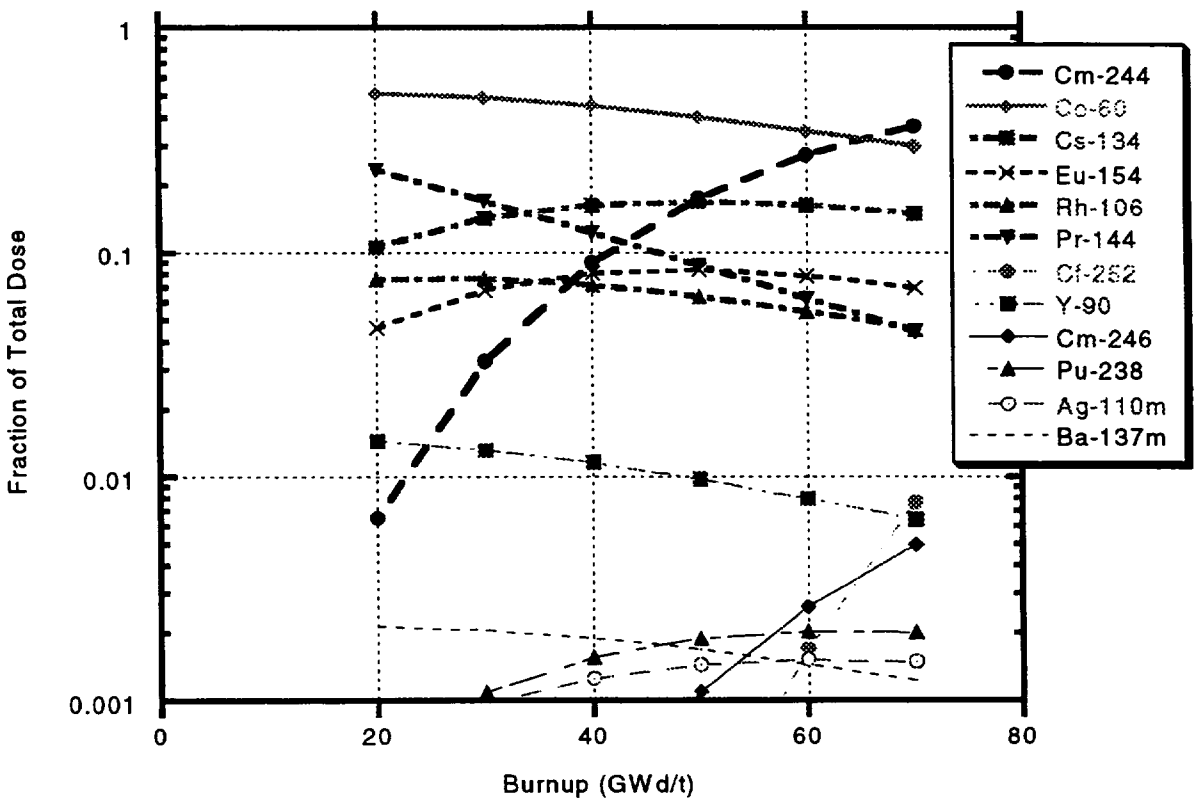


Figure 26 Fraction of total dose from dominant nuclides in 5-wt % PWR fuel; lead cask, 5-year cooling

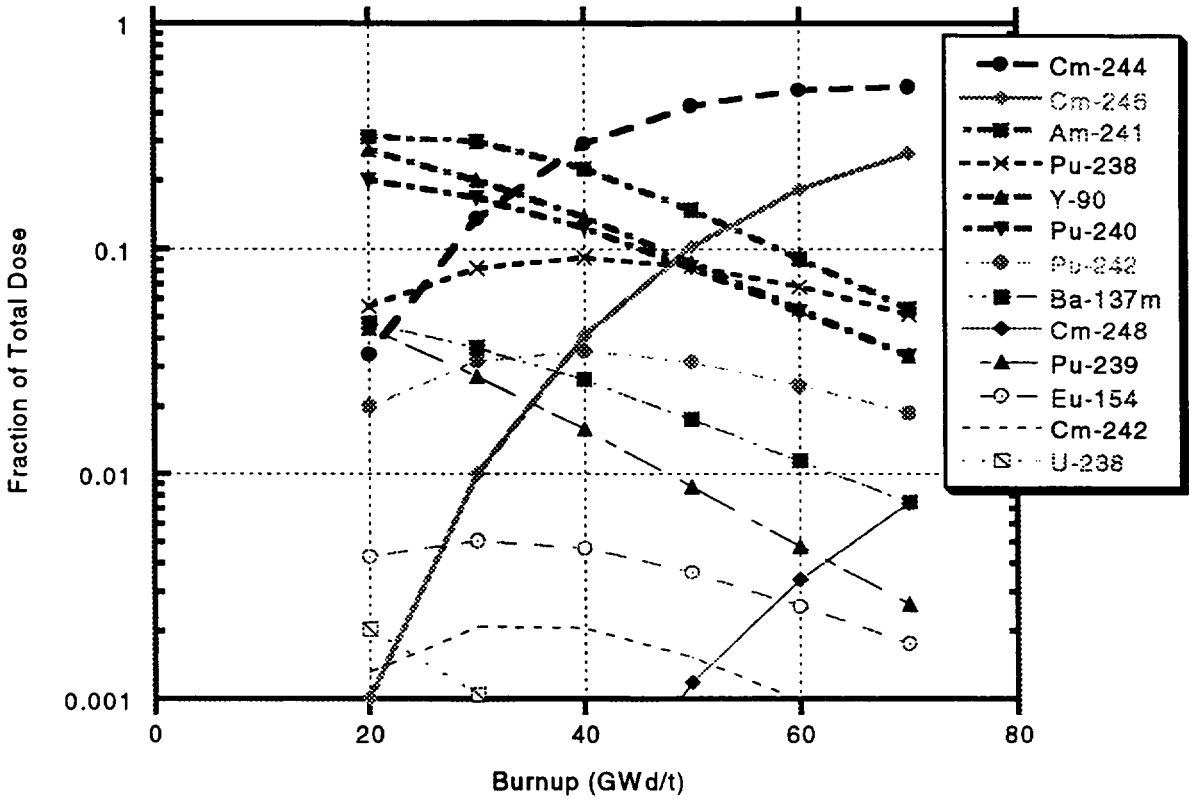


Figure 27 Fraction of total dose from dominant nuclides in 5-wt % PWR fuel; lead cask, 100-year cooling

Table 11 Shielding rankings for dominant radionuclides in PWR fuel

	Steel cask				Concrete cask				Lead cask			
	20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %		20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %		20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %	
Nuclides[a]	Percent[b]	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Ba-137m	3.0	6	2.3	7	9.2	4	8.9	4	0.2	8	0.1	8
Cm-244	1.8	7	36.1	1	0.3	8	8.3	5	1.9	6	36.6	1
Co-60	49.8	1	26.4	2	50.3	1	34.6	1	56.4	1	29.9	2
Cs-134	10.7	3	15.8	3	14.0	2	26.7	2	10.2	3	15.0	3
Eu-154	4.6	5	7.0	4	5.5	6	10.9	3	4.5	5	6.9	4
Pr-144	19.6	2	5.0	5	12.4	3	4.2	7	17.3	2	4.4	6
Rh-106	9.0	4	5.0	6	6.9	5	5.0	6	8.2	4	4.5	5
Y-90	1.0	8	0.6	8	1.1	7	0.9	8	1.0	7	0.6	7
100-Year cooling												
Am-241	18.6	2	4.3	4	2.2	3	1.3	5	2.9	1	5.4	3
Ba-137m	42.0	1	12.4	3	85.6	1	66.9	1	3.6	6	0.7	8
Cm-244	6.0	5	46.6	1	0.7	5	14.8	2	9.5	4	52.3	1
Cm-246	0.3	9	23.7	2	0.0	9	7.5	3	0.5	9	26.6	2
Pu-238	3.2	6	4.2	5	0.4	6	1.3	6	5.6	5	5.2	4
Pu-239	1.8	8	0.2	9	0.2	8	0.1	9	3.2	8	0.3	9
Pu-240	13.3	3	2.9	7	1.6	4	0.9	7	21.5	2	3.3	6
Pu-242	2.2	7	1.7	8	0.3	7	0.5	8	3.5	7	1.9	7
Y-90	11.9	4	3.0	6	8.7	2	5.8	4	18.9	3	3.4	5

[a] Dominant nuclides that contribute more than 1% to the total dose rate.

[b] Indicates percentage contribution of the nuclide to the total dose rate for the cask configuration.

The effect of assembly type (PWR or BWR) on the characteristics of high-burnup spent fuel was investigated by regenerating the nuclide importance rankings for an 8 × 8 BWR fuel assembly that incorporated four burnable poison rods, each containing 4.0-wt % Gd_2O_3. The results of the shielding importance rankings for the moderate-enrichment and moderate-burnup fuel (3-wt % and 20-GWd/t) and higher-enrichment and higher-burnup fuel (5-wt % and 70-GWd/t) are listed in Table 12. The BWR fuel results for 100-years cooling are almost identical to those for the PWR fuel. After 5-years cooling, the rankings are affected by the larger ^{60}Co contribution in the PWR assembly compared to the BWR assembly due to the significantly larger cobalt impurity level associated with the PWR assembly model. In addition, ^{252}Cf appears in the list of dominant BWR nuclides due to its larger production in the BWR assembly. A number of nuclides, most notably ^{252}Cf, ^{248}Cm, and ^{246}Cm, were found to exhibit large relative changes (up to 60%) in their fractional contributions to the total dose rate but have little impact on the total dose rate due to the generally low importance of these nuclides. All other changes in nuclide composition between the assembly types resulted in changes that were less than 1% for the cask models studied.

The most dramatic change in the relative radionuclide importance with burnup is the increasing importance of ^{244}Cm to the total dose rate. At shorter cooling times, the dose rate is dominated by fission products and the ^{60}Co activation product (note, however, that the trace level of ^{59}Co used in this study was highly conservative). Contributions from ^{134}Cs and ^{154}Eu, both generated by fission product capture as opposed to direct production from fission, generally increase with burnup, while ^{144}Pr exhibits a steady near-exponential decrease in importance with increasing burnup due to the increasing contribution from other radionuclides. At longer cooling times (>5 years) the contribution from ^{244}Cm becomes significant in high-burnup fuel. The rapid increase in the ^{244}Cm component results in the decrease in the relative contribution of many fission products and ^{60}Co.

A significant trend in the shielding results is the dramatic increase in the neutron source, and consequently the neutron component of the total dose rate for higher-burnup spent fuel. The increase with burnup is dependent on both the cooling time and the cask design. In high-burnup fuel, spontaneous-fission neutrons generated primarily by ^{244}Cm dominate the neutron source for cooling times between about 2 and 50 years. At the shorter cooling times, ^{242}Cm becomes increasingly important, and at 100 years contributions from ^{240}Pu (at low burnup) and ^{246}Cm (at high burnup) represent a significant fraction of the spontaneous-fission neutron source and thus the neutron dose rate. At the longer cooling times (>50 years) the (α,n) component of the neutron source increases and becomes significant only for relatively low-burnup fuel.

At short cooling times, the fission products ^{144}Pr, ^{106}Rh, ^{134}Cs, and activation product ^{60}Co dominate the total dose rate. However, the impurity cobalt level assumed in the assembly structural material is likely many times higher than that used in current assembly designs. Therefore, the fractional contribution from ^{60}Co is likely significantly overestimated by this study when considering newer fuel assembly designs.

Table 12 Shielding rankings for dominant radionuclides in BWR fuel

	Steel cask				Concrete cask				Lead cask			
	20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %		20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %		20 GWd/t 3.0 wt %		70 GWd/t 5.0 wt %	
Nuclides[a]	Percent[b]	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank	Percent	Rank
5-Year cooling												
Ba-137m	4.5	6	2.7	7	13.9	4	11.6	4	0.3	8	0.2	9
Cf-252	0.0	9	1.3	8	0.0	9	0.3	9	0.0	9	1.4	7
Cm-244	3.0	7	43.9	1	0.5	8	10.9	5	3.3	6	45.8	1
Co-60	24.5	2	10.7	3	24.9	1	15.1	2	29.7	1	12.5	3
Cs-134	16.2	3	19.1	2	21.3	2	34.9	1	16.5	3	18.7	2
Eu-154	7.0	5	8.1	4	8.4	6	13.5	3	7.4	5	8.2	4
Pr-144	29.3	1	6.1	5	18.7	3	5.4	7	27.6	2	5.5	6
Rh-106	13.4	4	6.0	6	10.3	5	6.4	6	13.0	4	5.6	5
Y-90	1.5	8	0.8	9	1.6	7	1.2	8	1.6	7	0.8	8
100-Year cooling												
Am-241	17.8	2	3.8	5	2.1	3	1.2	6	31.5	1	4.8	4
Ba-137m	42.0	1	12.2	3	85.6	1	66.7	1	3.6	6	0.7	8
Cm-244	6.6	5	46.2	1	0.8	5	14.8	2	10.4	4	51.7	1
Cm-246	0.4	9	25.0	2	0.0	9	8.1	3	0.6	9	28.0	2
Pu-238	3.4	6	4.0	4	0.4	6	1.3	5	5.8	5	4.9	3
Pu-239	1.8	8	0.2	9	0.2	8	0.1	9	3.2	8	0.2	9
Pu-240	13.4	3	3.0	6	1.6	4	0.9	7	21.6	2	3.4	5
Pu-242	2.2	7	1.6	8	0.3	7	0.5	8	3.4	7	1.8	7
Y-90	11.9	4	3.0	7	8.7	2	5.8	4	19.0	3	3.3	6

[a] Dominant nuclides that contribute more than 1% to the total dose rate.

[b] Indicates percentage contribution of the nuclide to the total dose rate for the cask configuration.

6 SUMMARY

This report has presented the importance rankings of the various actinides, fission products, and activation products that are important to criticality safety, decay-heat analysis, and cask-shielding analysis. The work is largely an extension of previous ranking studies (Ref. 9) to assess higher fuel enrichment and burnup regimes, and has focused on the changes in nuclide importance with increasing burnup, rather than cooling time, in the respective analysis areas. In addition, the present study has restricted the decay times to between 2 and 100 years, the regime relevant to spent fuel transport and interim storage.

Table 13 and Table 14 present summaries of the actinides, fission products, and activation products that have importance rankings greater than 0.1% in any of the application areas. Nuclides having a relative importance greater than 10% of the total are designated as high ranking (H), those between 1% and 10% are medium ranking (M), and those less than 1% are low ranking (L). These designations are provided for a reference low-burnup fuel having 3-wt % ^{235}U and 20-GWd/t and high-burnup fuel having 5-wt % ^{235}U and 70-GWd/t to give a quick indication of large changes in rank with increasing burnup. A reference cooling time of 20 years was used in generating these rankings. Thus, the tables do not include some of the short-lived radionuclides important in decay heat and shielding analyses and do not indicate large changes in nuclide ranking which do not result in a change in the ranking group (i.e., groups are very broad). These tables should therefore only be used as an approximate guide to nuclide importance for longer cooling times (about 20 years).

One of the most significant changes in the spent fuel characteristics associated with high burnup, relative to conventional-burnup fuel, is the dramatic increase in the ^{244}Cm concentration in extended-burnup fuel, which has an important impact for both decay-heat rates and shielding applications. For decay heating the contribution of ^{244}Cm increases from typically about 1% of the total at low-to-moderate burnup and cooling times less than 50 years, to over 10% of the total at high burnup. A large increase in the importance of ^{238}Pu to decay heat is also observed for high-burnup fuel, and it can contribute from 10% to about 30% of the total decay heat for high-burnup fuel. The effect of the increased ^{244}Cm inventory is also responsible for the large increase in the neutron source term (spontaneous fission) at high burnup relative to the gamma rays. The analysis of a steel transport cask indicates that the contribution from ^{244}Cm can increase from 3% at 20 GWd/t to over 60% of the total dose rate at 60 GWd/t after 20-years cooling. Consequently, existing validation data for low- and moderate-burnup fuel may not be representative of extended burnup fuel since it may not accurately represent the contribution from ^{244}Cm and other potentially important radionuclides. Any calculational uncertainties or biases based on validation studies for lower-burnup fuel need to be reviewed with respect to the changes in the underlying fuel compositions with enrichment and burnup that contribute to the response.

Table 13 Summary of actinide rankings by analysis area (20-year cooling time)[a]

	Criticality		Decay heat		Shielding[b]	
Isotope	20 GWd/t 3.0 wt %	70 GWd/t 5.0 wt %	20 GWd/t 3.0 wt %	70 GWd/t 5.0 wt %	20 GWd/t 3.0 wt %	70 GWd/t 5.0 wt %
U-235	H	H				
U-236	L	M				
U-238	H	H				
Np-237	L	M				
Pu-238	L	L	M	H	L	L
Pu-239	H	H	M	L		
Pu-240	M	M	M	M	L	L
Pu-241	L	M	L	L		
Pu-242		M			L	L
Am-241	L	M	M	M	L	L
Am-243				L		
Cm-244			L	H	L	H
Cm-246						M

[a] H = high, M = medium, and L = low ranking.

[b] Shielding ranking based on results for the steel transport cask model.

Table 14 Summary of fission- and activation-product rankings by analysis area (20-year cooling time)[a]

	Criticality		Decay heat		Shielding[b]	
Nuclide	20 GWd/t 3.0 wt %	70 GWd/t 5.0 wt %	20 GWd/t 3.0 wt %	70 GWd/t 5.0 wt %	20 GWd/t 3.0 wt %	70 GWd/t 5.0 wt %
Co-60[c]			L	L	H	H
Kr-85			L	L		
Sr-90			M	M		
Y-90			H	H	M	M
Mo-95	L	L				
Tc-99	L	L				
Ru-101		L				
Rh-103	L	M				
Pd-105		L				
Pd-108		L				
Ag-109		L				
Xe-131	L	L				
Cs-133	L	M				
Cs-134			L	L	L	L
Cs-137			M	M		
Ba-137m			H	H	H	M
Pr-141		L				
Nd-143	L	M				
Nd-145	L	L				
Sm-147	L	L				
Sm-149	M	M				
Sm-150	L	L				
Sm-151	L	L				
Sm-152	L	L				
Eu-153	L	L				
Eu-154			L	M	M	L
Gd-155	L	M				

[a] H = high, M = medium, and L = low ranking.

[b] Shielding ranking based on results for the steel transport cask model.

[c] Importance rankings based on highly conservative initial cobalt levels and are likely overestimated in this study.

7 CONCLUSIONS AND RECOMMENDATIONS

The importance rankings in this report are designed to facilitate validation efforts related to the prediction of spent fuel isotopics and radiological decay properties of high-burnup spent fuel by identifying the nuclides that have the greatest impact in the application areas of interest and illustrate the variation in nuclide importance with burnup. These results also have potential application to establishing the degree of applicability of existing validation data to the high-burnup regime and may be useful to help identify the challenges that high-burnup fuel presents to existing computational methods and nuclear data.

The availability of experimental data for code validation in the high-enrichment and high-burnup regimes is likely to be limited in the near term to isotopic assay measurements that are either underway or are being planned in support of criticality burnup credit activities in the United States and in other countries. Through these activities the experimental assay programs are seeking to compile a comprehensive database of spent fuel isotopics that is representative of the fuel types and assembly designs from operating LWRs.

Because the experimental database for decay heat and radiation source-term validation data is likely to remain limited to low- and moderate-enrichment and burnup fuels in the near term, a separate-effects approach to code validation currently used in burnup credit may be a valuable, if not necessary, option for validating code predictions of spent fuel radiological decay properties in the high burnup regime. This approach relies on independently validating the depletion analysis methods used to predict the spent fuel compositions using isotopic assay data, and validating the isotopic response (e.g., isotopic reactivity worth in burnup credit applications, or energy release rate per decay for decay heat applications). Such a strategy for validating decay heat generation and radiation source-term predictions would be simplified by the fact that relatively few radioisotopes are responsible for the responses at high burnup. For example, the decay-heat generation rate for high-burnup fuel is dominated by fewer than 10 radionuclides that are responsible for about 95% of the total decay heat at 5-years cooling. After 100 years just four radionuclides account for about 85% of the decay heating. A similar trend is observed for the radionuclides important to radiation shielding.

A separate-effects approach would also be simplified by the fact that the nuclide decay data parameters important in these application areas (recoverable energy per decay for decay heating and radiation emission spectra for source terms) are already relatively well established by measurements. Consequently, the accuracy of computer code predictions could be established to a high degree of confidence using measured isotopic inventory data alone. Unfortunately, many of the nuclides important to the burnup credit are for the most part different than those important to decay-heat and radiation-source-term analysis. However, some overlap does exist.

It is recommended that consideration be given to expanding the isotopic assay measurements currently being planned within the United States in support of the burnup credit activities to include additional radionuclides important to decay heating and radiation-source-term applications. A review of the findings from this study suggests that relatively few new radionuclides are needed to provide a relatively comprehensive characterization of high-burnup spent fuel in the respective application areas. Also, several key radionuclides exist as parent-daughter pairs (e.g., ^{137}Cs–^{137m}Ba, ^{90}Sr–^{90}Y, ^{144}Ce–^{144}Pr) and therefore only one radionuclide in these pairs would require measurement since they exist in secular equilibrium (i.e., the activities of the parent and daughter product are equal). The selected parent or daughter could be chosen to optimize the coverage of different application areas.

For decay-heat generation the recommended additional nuclides (those not generally included in burnup credit isotopic assay experiments) include: ^{137m}Ba, ^{90}Y, ^{106}Rh, ^{134}Cs, ^{144}Pr, ^{154}Eu, ^{244}Cm. For cask shielding analyses only one additional nuclide, ^{246}Cm, not already identified as being important to burnup credit or decay heating, would be needed to accurately represent the radiation source term for high burnup spent fuel.

Longer-term acquisition of decay-heat measurements for high-burnup fuel is currently being pursued through ORNL participation in a Swedish program to measure decay-heat levels for high-burnup and modern design PWR and BWR assemblies being performed for the Swedish Nuclear Fuel and Waste Management Co., SKB. However, it may be several years before validation data are made available. Currently, no known programs are being planned to measure radiation source terms for high-burnup fuel assemblies. Experiments have been proposed as part of the REBUS program (Belgonucleaire) to measure neutron source emission rates from high-burnup MOX and possibly UO_2 fuel samples. However, since the major neutron emitters in these samples will be ^{242}Cm and ^{244}Cm (due to the relatively short cooling times of these samples) it is not clear that such measurements would be preferable to isotopic measurements since the uncertainties in the REBUS measurements could be large. Since the neutron emission spectra for these isotopes are well known, the acquisition of inventory data for the major spontaneous-fission actinides could provide a more accurate validation of the neutron source. Radionuclide inventory measurements could provide an accurate method for the validation of both neutron and gamma-ray source terms, and would allow validation over a potentially wide range of burnups and cooling times.

8 REFERENCES

1. J. W. Roddy and J. C. Mailen, *Radiological Characteristics of Light-Water-Reactor Spent Fuel: A Literature Survey of Experimental Data*, ORNL/TM-10105, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, December 1987.
2. O. W. Hermann, C. V. Parks, and J.-P. Renier, *Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data*, NUREG/CR-5625 (ORNL/TM-6698), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, September 1994.
3. J. C. Ryman, O. W. Hermann, C. C. Webster, and C. V. Parks, *Fuel Inventory and Afterheat Power Studies of Uranium-Fueled Pressurized Water Reactor Fuel Assemblies Using the SAS2 and ORIGEN-S Modules of SCALE with an ENDF/B-V Updated Cross-Section Library*, NUREG/CR-2397 (ORNL/CSD-90), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, September 1982.
4. O. W. Hermann, S. M. Bowman, M. C. Brady, and C. V. Parks, *Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analysis*, ORNL/TM-12667, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, March 1995.
5. M. D. DeHart and O. W. Hermann, *An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel*, ORNL/TM-13317, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, September 1996.
6. O. W. Hermann and M. D. DeHart, *Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel*, ORNL/TM-13315, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, September 1998.
7. B. L. Broadhead, J. S. Tang, R. L. Childs, C. V. Parks, and H. Taniuchi, "Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments," EPRI TR-104329, May 1995.
8. M. D. DeHart, *Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages*, ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, May 1996.
9. B. L. Broadhead, M. D. DeHart, J. C. Ryman, J. S. Tang, and C. V. Parks, *Investigation of Nuclide Importance to Functional Requirements Related to Transport and Long-Term Storage of LWR Spent Fuel*, ORNL/TM-12742, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, June 1995.
10. U.S. Department of Energy, "Characteristics of Potential Repository Wastes," DOE/RW-0184-R1 (Vols. 1–4), July 1992.
11. O. W. Hermann and C. V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," Vol. 1, Sect. S2 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.
12. W. C. Jordan and S. M. Bowman, "SCALE Cross-Section Libraries," Vol. 3, Sect. M4 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.

13. S. M. Bowman and L. C. Leal, "ORIGEN-ARP: Automatic Rapid Process for Spent Fuel Depletion, Decay, and Source Term Analysis," Vol. 1, Sect. D1 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.

14. J. R. Knight, C. V. Parks, S. M. Bowman, L. M. Petrie and J. A. Bucholz, "SAS1: A One-Dimensional Shielding Analysis Module," Vol. 1, Sect. S1 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.

15. O. W. Hermann and R. M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," in Vol. 2, Sect. F7 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.

16. J. C. Ryman and O. W. Hermann, "ORIGEN-S Data Libraries," Vol. 3, Sect. M6 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.

17. C. V. Parks, B. L. Broadhead, O. W. Hermann, J. S. Tank, S. N. Cramer, J. C. Gauthey, B. L. Kirk, and R. W. Roussin, *Assessment of Shielding Analysis Methods, Codes, and Data for Spent Fuel Transport/Storage Applications*, ORNL/CSD/TM-246, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, July 1988.

APPENDIX A

COMPUTER CODE INPUT AND OUTPUT LISTINGS

APPENDIX A

COMPUTER CODE INPUT AND OUTPUT LISTINGS

Listing A.1 Example of SAS2 input listing used to generate PWR isotopic compositions

```
'  $HOME/nrc/ranking/pwr/pwr4_60.in
'
=sas2     parm='skipshipdata'
 17x17 pwr westinghouse assembly
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'   mixtures of fuel-pin-unit-cell:
'
44group        latticecell
uo2  1 den=10.32   1  811
 92234  0.0356 92235 4.0  92236 0.0184 92238 95.946 end
'
zr-94  1 0 1-20 811  end
mo-94  1 0 1-20 811  end
nb-95  1 0 1-20 811  end
mo-95  1 0 1-20 811  end
tc-99  1 0 1-20 811  end
ru-106 1 0 1-20 811  end
rh-103 1 0 1-20 811  end
rh-105 1 0 1-20 811  end
sn-126 1 0 1-20 811  end
xe-131 1 0 1-20 811  end
cs-134 1 0 1-20 811  end
cs-135 1 0 1-20 811  end
cs-137 1 0 1-20 811  end
ce-144 1 0 1-20 811  end
pr-143 1 0 1-20 811  end
nd-143 1 0 1-20 811  end
nd-144 1 0 1-20 811  end
nd-145 1 0 1-20 811  end
nd-146 1 0 1-20 811  end
nd-147 1 0 1-20 811  end
nd-148 1 0 1-20 811  end
nd-150 1 0 1-20 811  end
sm-147 1 0 1-20 811  end
sm-149 1 0 1-20 811  end
sm-150 1 0 1-20 811  end
sm-151 1 0 1-20 811  end
sm-152 1 0 1-20 811  end
pm-147 1 0 1-20 811  end
pm-148 1 0 1-20 811  end
pm-149 1 0 1-20 811  end
eu-151 1 0 1-20 811  end
eu-153 1 0 1-20 811  end
eu-154 1 0 1-20 811  end
eu-155 1 0 1-20 811  end
gd-155 1 0 1-20 811  end
gd-157 1 0 1-20 811  end
gd-158 1 0 1-20 811  end
gd-160 1 0 1-20 811  end
zirc2  2 1  620   end
h2o    3 den=0.7295     1  570 end
boron  3 den=0.7295 550-6  570 end
'
```

```
'  550 ppm boron (wt) in moderator
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
end comp
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'      fuel-pin-cell geometry:
'
squarepitch   1.25984 0.81915 1 3 0.94966 2 0.83566 0 end
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'    assembly and cycle parameters:
'
npin/assm=264  fuelngth=790.28  ncycles=8  nlib/cyc=1
printlevel=4  lightel=9  inplevel=1  ortube=0.61214 srtube=0.5715
numinstr=1 end
power=35    burn=75.0    down=0.0   end
power=35    burn=75.0    down=0.0   end
power=35    burn=150.0   down=0.0   end
power=35    burn=300.0   down=0.0   end
power=35    burn=300.0   down=0.0   end
power=35    burn=300.0   down=0.0   end
power=35    burn=300.0   down=0.0   end
power=35    burn=214.3   down=0.0   end
  o  135  cr  5.9  mn 0.33
  fe 13.  co 0.075 ni 9.9
  zr 221  nb  0.71 sn  3.6
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
end
=shell
cp ft71f001 $RTNDIR/pwr4e_60g.sas2.ft71f001
end
=origens
0$$  a11 71  e t
decay case
3$$ 15 a3 1 27   a16 0  a33 18 e t
35$$ 0 t
'  56$$ a13 -2 reads data from position 2 on ft71f001
56$$  0 10  a6 1  a10  0  a13 -2  a14 5  1 a17 2 e
57** 0 e t
spent fuel 100-year decay
60** 1 2 3 5 10 20 30 50 70 100
61** f 0.000001
65$$ a25 1 a46 1 e
81$$ 2 0 26 1  e
82$$  10r 2
83**  1.e+7        8.e+6        6.5e+6       5.e+6        4.e+6
 3.e+6        2.5e+6       2.e+6        1.66e+6      1.33e+6
 1.e+6        8.e+5        6.e+5        4.e+5        3.e+5
 2.e+5        1.e+5        5.e+4        1.e+4
84**  2.e+7       6.434e+6     3.e+6        1.85e+6      1.4e+6
 9.e+5        4.e+5        1.e+5        1.7e+4       3.e+3
 5.5e+2       1.e+2        3.e+1        1.e+1        3.04999e+0
 1.77e+0      1.29999e+0   1.12999e+0   1.e+0        8.e-1
 4.e-1        3.25e-1      2.25e-1      9.999985e-2  5.e-2
 3.e-2        9.999998e-3  1.e-5
 6t
4% 235u, 60 gwd/mtu pwr spent fuel source
4% 235u, 60 gwd/mtu pwr spent fuel source
4% 235u, 60 gwd/mtu pwr spent fuel source
4% 235u, 60 gwd/mtu pwr spent fuel source
4% 235u, 60 gwd/mtu pwr spent fuel source
4% 235u, 60 gwd/mtu pwr spent fuel source
4% 235u, 60 gwd/mtu pwr spent fuel source
4% 235u, 60 gwd/mtu pwr spent fuel source
4% 235u, 60 gwd/mtu pwr spent fuel source
4% 235u, 60 gwd/mtu pwr spent fuel source
'  write time steps 1 - 10 in positions
'  3 - 12 on ft71f001
```

```
56$$ 0 -10 a10  1  e t
56$$ f0 t
end
=shell
cp ft15f001 $RTNDIR/pwr4e_60g.ft15f001
cp ft71f001 $RTNDIR/pwr4e_60g.ft71f001
end
```

Listing A.2 Example of SAS2 input listing used to generate BWR isotopic compositions

```
'  $HOME/nrc/ranking/bwr/4.0_gd/bwr5e_4gd_70g.in
   '
   =sas2     parm='skipshipdata'
    8x8 bwr assembly
   ' Fuel enrichment 5.0 wt % of u-235, Fuel Temperature 840K
   ' Moderator Temperature 558K, Moderator Density 0.4323 g/cc
   '
   ' 9 cycles and one library per cycle
   ' 35 MW/MTU specific power - 70 GWd/MTU burnup
   '
   ' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
   '
   '   mixtures of fuel-pin-unit-cell:
   '
   44group        latticecell
   uo2  1 den=9.871  1  840
    92234  0.0445 92235 5.0  92236 0.0230 92238 94.9325 end
   '
   zr-94  1 0 1-20 840  end
   mo-94  1 0 1-20 840  end
   nb-95  1 0 1-20 840  end
   mo-95  1 0 1-20 840  end
   tc-99  1 0 1-20 840  end
   ru-106 1 0 1-20 840  end
   rh-103 1 0 1-20 840  end
   rh-105 1 0 1-20 840  end
   sn-126 1 0 1-20 840  end
   xe-131 1 0 1-20 840  end
   cs-134 1 0 1-20 840  end
   cs-135 1 0 1-20 840  end
   cs-137 1 0 1-20 840  end
   ce-144 1 0 1-20 840  end
   pr-143 1 0 1-20 840  end
   nd-143 1 0 1-20 840  end
   nd-144 1 0 1-20 840  end
   nd-145 1 0 1-20 840  end
   nd-146 1 0 1-20 840  end
   nd-147 1 0 1-20 840  end
   nd-148 1 0 1-20 840  end
   nd-150 1 0 1-20 840  end
   sm-147 1 0 1-20 840  end
   sm-149 1 0 1-20 840  end
   sm-150 1 0 1-20 840  end
   sm-151 1 0 1-20 840  end
   sm-152 1 0 1-20 840  end
   pm-147 1 0 1-20 840  end
   pm-148 1 0 1-20 840  end
   pm-149 1 0 1-20 840  end
   eu-151 1 0 1-20 840  end
   eu-153 1 0 1-20 840  end
   eu-154 1 0 1-20 840  end
   eu-155 1 0 1-20 840  end
   gd-155 1 0 1-20 840  end
   gd-157 1 0 1-20 840  end
   gd-158 1 0 1-20 840  end
   gd-160 1 0 1-20 840  end
   zirc2  2 1  620   end
   h2o    3 den=0.4323     1  558 end
   '
   '
   ' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
   '
   '   mixtures of larger unit cell
   '
   uo2  9  den=9.871 1 840
    92234 0.0445 92235 5.0 92236 0.0230 92238 94.9325 end
```

```
arbm-gdburn  9.871 7 0 1 1
            64154  2.18  64155 14.80  64156 20.47
            64157 15.65  64158 24.84  64160 21.86
             8016 150.0  9  0.040  840  end
'
'             ....above is 4.0 wt% gadolinium (as gd2o3) in
'                 the burnable poison pins of bwr assembly
'
zirc2  10  1 588  end
'             ....above is zircalloy casing around assembly
b-10    11 0 7.15-6  552  end
h2o     11 0.669     552  end
'             ....above is channel moderator at higher density
end comp
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'      fuel-pin-cell geometry:
'
squarepitch   1.6256 1.0795  1 3  1.25222  2  end
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'    assembly and cycle parameters:
'
npin/assm=63  fuelngth=1993.26  ncycles=9   nlib/cyc=1
printlevel=4  lightel=8  inplevel=2  numzones=6   end
9 0.53975 2 0.62611 3 0.91715 500 3.6398 10 3.8103 11 4.3261
'           ..these mixturs & radii place gadolinium pin at center
'             of 1/4 of assembly fuel, casing & channel mod.
power=35    burn=75.0    down=0.0   end
power=35    burn=75.0    down=0.0   end
power=35    burn=150.0   down=0.0   end
power=35    burn=300.0   down=0.0   end
power=35    burn=300.0   down=0.0   end
power=35    burn=300.0   down=0.0   end
power=35    burn=300.0   down=0.0   end
power=35    burn=300.0   down=0.0   end
power=35    burn=200.0   down=0.0   end
  o  135  cr  2.4   mn 0.15
  fe 6.6  co  0.024 ni 2.4
  zr 516  sn  8.7
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
end
=shell
cp ft71f001 $RTNDIR/bwr5e_4gd_70g.sas2.ft71f001
end
=origens
0$$  a11 71  e t
decay case
3$$ 15 a3 1 27 a16 0  a33 18 e t
35$$ 0 t
'  56$$ a13 -2 reads data from position 2 on ft71f001
56$$  0 10  a6 1  a10  0  a13 -2  a14 5  1 a17 2 e
57** 0 e t
spent fuel 100-year decay
60** 1 2 3 5 10 20 30 50 70 100
61** f 0.000001
65$$ a25 1 a46 1 e
81$$ 2 0 26 1  e
82$$  10r 2
83**  1.e+7        8.e+6        6.5e+6       5.e+6        4.e+6
 3.e+6        2.5e+6       2.e+6        1.66e+6      1.33e+6
 1.e+6        8.e+5        6.e+5        4.e+5        3.e+5
 2.e+5        1.e+5        5.e+4        1.e+4
84**  2.e+7       6.434e+6     3.e+6        1.85e+6      1.4e+6
 9.e+5        4.e+5        1.e+5        1.7e+4       3.e+3
 5.5e+2       1.e+2        3.e+1        1.e+1        3.04999e+0
 1.77e+0      1.29999e+0   1.12999e+0   1.e+0        8.e-1
```

```
 4.e-1          3.25e-1        2.25e-1        9.999985e-2  5.e-2
 3.e-2          9.999998e-3  1.e-5
 6t
5% 235u, 70 gwd/mtu bwr spent fuel source
5% 235u, 70 gwd/mtu bwr spent fuel source
5% 235u, 70 gwd/mtu bwr spent fuel source
5% 235u, 70 gwd/mtu bwr spent fuel source
5% 235u, 70 gwd/mtu bwr spent fuel source
5% 235u, 70 gwd/mtu bwr spent fuel source
5% 235u, 70 gwd/mtu bwr spent fuel source
5% 235u, 70 gwd/mtu bwr spent fuel source
5% 235u, 70 gwd/mtu bwr spent fuel source
5% 235u, 70 gwd/mtu bwr spent fuel source
'  write time steps 1 - 10 in positions
'  3 - 12 on ft71f001
56$$ 0 -10 a10  1  e t
56$$ f0 t
end
=shell
cp ft15f001 $RTNDIR/bwr5e_4gd_70g.ft15f001
cp ft71f001 $RTNDIR/bwr5e_4gd_70g.ft71f001
end
```

Listing A.3 ORIGEN-S and SAS1 shielding analysis input files (steel cask model)

```
'  $HOME/nrc/ranking/dose/tn24p_pb210.in
'
'  this run computes dose factors for 1 g-atom of pb-210
'
=origens
0$$ a11 71  e t
decay case for 1 gram-atom pb-210
3$$ 21 1 1 27  a16 0  a33 18  e t
35$$ 0 t
56$$ a2 1  a6 1  a10  0  a13 1  a15 3 e
57** 0 e t
unit photon source
1 gram-atom pb-210
60** 0.
61** f.05
65$$
'gram-atoms   grams    curies     watts-all  watts-gamma
     3z     1  0  0        3z         3z         3z         6z
     3z     1  0  0        3z         3z         3z         6z
     3z     1  0  0        3z         3z         3z         6z
81$$ 2 0 26 1  e
82$$  2
83**  1.e+7        8.e+6        6.5e+6       5.e+6        4.e+6
 3.e+6        2.5e+6       2.e+6        1.66e+6      1.33e+6
 1.e+6        8.e+5        6.e+5        4.e+5        3.e+5
 2.e+5        1.e+5        5.e+4        1.e+4
84**  2.e+7        6.434e+6     3.e+6        1.85e+6      1.4e+6
 9.e+5        4.e+5        1.e+5        1.7e+4       3.e+3
 5.5e+2       1.e+2        3.e+1        1.e+1        3.04999e+0
 1.77e+0      1.29999e+0   1.12999e+0   1.e+0        8.e-1
 4.e-1        3.25e-1      2.25e-1      9.999985e-2  5.e-2
 3.e-2        9.999998e-3  1.e-5
73$$  822100
74**  1
75$$  2 t
surry photon source timestep 1
56$$ 2z a10 1 f0 t
56$$ f0 t
end
=sas1        parm='size=300000'
1-d model of tn24p w/ 24 uncons. fuel assemblies smeared over 72.25 cm
27n-18couple          infhommedium
' mixture 1, homogenized fuel zone: fuel pins, clad, & basket
uo2  1 0.192 293 92235 3.0 92238 97.0 end
zircalloy  1 0.0647 293 end
arbmbal1   0.170 2  1  0  0  5000  2.5  13027  97.5  1  end
  mixture 2, boral basket
arbmbal2   2.65  2  1  0  0  5000  2.5  13027  97.5  2  end
carbonsteel  3  end
' mixture 4, radial neutron shield (includes fins)
arbm:resin 1.505 6 1 0 1  5000 1.05 1001 5.05 6012 35.13 8016 41.73
 13027 14.93 29000 2.11 4  1.0 293 5010 18.3022 5011 81.6978 end
cu    4  0.0476  end
ss304      5  end
end comp
end
last
tn24p with 24 unconsolided fuel assemblies
cylinder
1  72.25  60  +1 0 0.0 71 1 5.998+6 11.04
2  72.75  2  0
3  99.75  40  0
4  113.0  25 0
5  114.0  4  0
 end zone
```

```
 ndetec=1
read xsdose
' 2 det. @ 1° ans 2' from surface(114.0)
  365.76 116.54 182.88
end
=shell
grep \(total\) _out* > tmp
sed 's/total/pb-210 /; s/_out0005:/822100/' tmp >> ddata
rm _out*
cp ddata $RTNDIR/pb210_dose_factors
end
```

Listing A.4 Example of input listing for total dose for steel cask

```
'  $HOME/nrc/ranking/pwr/tot_tn24p_pwr_4e_60g.in
'
'  this case begins with the concentrations in position 5 of ft71f001
'  which has 12 positions.  position 1 has the end of irradiation
'  concentrations from the sas2 case.  position 2 has the concentrations
'  from the end of the sas2 decay case (zero length).  positions
'  3 through 12 are from the origen-s case run separately.
'
=shell
cp $RTNDIR/pwr4e_60g.ft71f001 ft71f001
end
=sas1        parm='size=300000'
1-d model of tn24p w/ 24 uncons. fuel assemblies smeared over 72.25 cm
27n-18couple         infhommedium
' mixture 1, homogenized fuel zone: fuel pins, clad, & basket
uo2  1 0.192 293 92235 3.0 92238 97.0 end
zircalloy  1 0.0647 293 end
arbmbal1   0.170 2  1  0  0  5000  2.5  13027  97.5  1  end
' mixture 2, boral basket
arbmbal2   2.65  2  1  0  0  5000  2.5  13027  97.5  2  end
carbonsteel  3  end
' mixture 4, radial neutron shield (includes fins)
arbm:resin 1.505 6 1 0 1  5000 1.05 1001 5.05 6012 35.13 8016 41.73
 13027 14.93 29000 2.11 4  1.0 293 5010 18.3022 5011 81.6978 end
cu    4  0.0476  end
ss304      5  end
end comp
end

tn24p with 24 unconsolided fuel assemblies
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  72.25  60  +1 0 0.0 71 5 5.998+6 11.04
2  72.75  2  0
3  99.75  40  0
4  113.0  25 0
5  114.0  4  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1° ans 2' from surface(114.0)
  365.76 116.54 182.88
end

tn24p with 24 unconsolided fuel assemblies
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  72.25  60  +1 0 0.0 71 6 5.998+6 11.04
2  72.75  2  0
3  99.75  40  0
4  113.0  25 0
5  114.0  4  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1° ans 2' from surface(114.0)
  365.76 116.54 182.88
end

tn24p with 24 unconsolided fuel assemblies
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  72.25  60  +1 0 0.0 71 8 5.998+6 11.04
2  72.75  2  0
3  99.75  40  0
4  113.0  25 0
```

```
5  114.0  4  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1' ans 2' from surface(114.0)
  365.76 116.54 182.88
end
last
 tn24p with 24 unconsolided fuel assemblies
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  72.25  60  +1 0 0.0 71 12 5.998+6 11.04
2  72.75  2  0
3  99.75  40  0
4  113.0  25 0
5  114.0  4  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1' ans 2' from surface(114.0)
  365.76 116.54 182.88
end
```

Listing A.5 Example of input listing for total dose for concrete cask

```
'  $HOME/nrc/ranking/pwr/tot_vsc_pwr_4e_60g.in
'
'  this case begins with the concentrations in position 5 of ft71f001
'  which has 12 positions.  position 1 has the end of irradiation
'  concentrations from the sas2 case.  position 2 has the concentrations
'  from the end of the sas2 decay case (zero length).  positions
'  3 through 12 are from the origen-s case run separately.
'
=shell
cp $RTNDIR/pwr4e_60g.ft71f001 ft71f001
end
=sas1        parm='size=300000'
1-d radial model of vsc with consolidated fuel smeared over 63. cm
27n-18couple  infhommedium
' mixture 1, homogenized fuel zone: fuel pins, clad, & basket
uo2  1 .377 293 92235 3.0 92238 97.0 end
zircalloy  1  0.109  end
ss304  1 .102  end
ss304  2  end
' mixture 3, ordinary concrete
h     3  0. 7.77-3  end
o     3  0. 4.39-2  end
na    3  0. 1.05-3  end
mg    3  0. 1.49-4  end
al    3  0. 2.39-3  end
si    3  0. 1.58-2  end
s     3  0. 5.64-5  end
k     3  0. 6.93-4  end
ca    3  0. 2.92-3  end
fe    3  0. 3.13-4  end
  end comp
end

vsc with consolidated fuel
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  63.0  60  +1 0 0.0 71 5 4.56+6 15.64
0  66.04  2  0
2  67.31  3  0
0  74.93  5   0
2  83.19  20   0
3  133.35  40  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1° ans 2' from surface(133.35)
  365.76 135.89 182.88
end

vsc with consolidated fuel
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  63.0  60  +1 0 0.0 71 6 4.56+6 15.64
0  66.04  2  0
2  67.31  3  0
0  74.93  5   0
2  83.19  20   0
3  133.35  40  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1° ans 2' from surface(133.35)
  365.76 135.89 182.88
end

vsc with consolidated fuel
cylinder
```

```
' source position 5/6/8/12 =(3/5/20/100 years)
1  63.0  60  +1 0 0.0 71 8 4.56+6 15.64
0  66.04  2  0
2  67.31  3  0
0  74.93  5   0
2  83.19  20   0
3  133.35  40  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1° ans 2' from surface(133.35)
  365.76 135.89 182.88
end
last
vsc with consolidated fuel
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  63.0  60  +1 0 0.0 71 12 4.56+6 15.64
0  66.04  2  0
2  67.31  3  0
0  74.93  5   0
2  83.19  20   0
3  133.35  40  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1° ans 2' from surface(133.35)
  365.76 135.89 182.88
end
```

Listing A.6 Example of input listing for total dose for lead cask

```
'  $HOME/nrc/ranking/bwr/4.0_gd_rerun/tot_pb_bwr_5e_4gd_70g.in
'
'  this case begins with the concentrations in position 5 of ft71f001
'  which has 12 positions.  position 1 has the end of irradiation
'  concentrations from the sas2 case.  position 2 has the concentrations
'  from the end of the sas2 decay case (zero length).  positions
'  3 through 12 are from the origen-s case run separately.
'
=shell
cp $RTNDIR/bwr5e_4gd_70g.ft71f001 ft71f001
end
=sas1        parm='size=300000'
1-d model of lead cask with 24 uncons. assemblies smeared over 72.25 cm
27n-18couple         infhommedium
' mixture 1, homogenized fuel zone: fuel pins, clad, & basket
uo2  1 0.192 293 92235 3.0 92238 97.0 end
zircalloy  1 0.0647 293 end
arbmbal1   0.170 2  1  0  0  5000  2.5  13027  97.5  1  end
' mixture 2, boral basket
arbmbal2   2.65  2  1  0  0  5000  2.5  13027  97.5  2  end
pb  3  end
' mixture 4, radial neutron shield (includes fins)
arbm:resin 1.505 6 1 0 1  5000 1.05 1001 5.05 6012 35.13 8016 41.73
 13027 14.93 29000 2.11 4  1.0 293 5010 18.3022 5011 81.6978 end
cu    4  0.0476  end
ss304      5  end
end comp
end

lead cask similar to tn24p with 18.6 cm lead replacing 27 cm c-steel
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  72.25  60  +1 0 0.0 71 5 5.998+6 11.04
2  72.75  2  0
3  85.45  40  0
4  98.7  25 0
5  99.7  4  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1° ans 2' from surface(99.7)
  365.76 102.24 182.88
end

lead cask similar to tn24p with 18.6 cm lead replacing 27 cm c-steel
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  72.25  60  +1 0 0.0 71 6 5.998+6 11.04
2  72.75  2  0
3  85.45  40  0
4  98.7  25 0
5  99.7  4  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1° ans 2' from surface(99.7)
  365.76 102.24 182.88
end

lead cask similar to tn24p with 18.6 cm lead replacing 27 cm c-steel
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  72.25  60  +1 0 0.0 71 8 5.998+6 11.04
2  72.75  2  0
3  85.45  40  0
4  98.7  25 0
5  99.7  4  0
```

```
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1" ans 2' from surface(99.7)
  365.76 102.24 182.88
end
last
lead cask similar to tn24p with 18.6 cm lead replacing 27 cm c-steel
cylinder
' source position 5/6/8/12 =(3/5/20/100 years)
1  72.25  60  +1 0 0.0 71 12 5.998+6 11.04
2  72.75  2  0
3  85.45  40  0
4  98.7  25 0
5  99.7  4  0
 end zone
 ndetec=1
read xsdose
' 2 det. @ 1" ans 2' from surface(99.7)
  365.76 102.24 182.88
end
```

Listing A.7 Example of OPUS input file for absorption rankings

```
'  $HOME/nrc/ranking/pwr/opus_abs_act_pwr4e_60g.in
'
'  this case begins with the concentrations in position 2 of ft71f001
'  which has 12 positions.  position 1 has the end of irradiation
'  concentrations from the sas2 case.  position 2 has the concentrations
'  from the end of the sas2 decay case (zero length).  positions
'  3 through 12 are from the origen-s case run separately.
'
'  (1) first link after =shell is to link
'      latest opus executable
'
'  (2) link below would use pwr 33 GWd/MTU
'      library but this does not use the
'      correct cross sections for absorption
' ln -s $DATA/pwr33gwd            ft33f001
'
'      note that any library can be used to
'      get the other data needed for opus--
'      it takes the first library found in a
'      multiburnup library
'
'  (3) second link after =shell uses last
'      temporary library from ft15f001 in
'      the sas2 run--ft15f001 is the unit
'      for temporary storage of binary
'      libraries before they are added to the
'      back of the multiburnup library on
'      ft33f001
'
'  (3) third link after =shell uses ft71f001
'      saved from origen-s decay case run as
'      a separate job after sas2 run that
'      saved 12 positions on ft71f001
'
=shell
ln -s /home/2ul/bin/opus     opus
ln -s $RTNDIR/pwr4e_60g.ft15f001 ft15f001
ln -s $RTNDIR/pwr4e_60g.ft71f001 ft71f001
end
=opus
default=no
libunit=15
units=absorption
time=year
typarms=nucl
tmax=100.1
libtyp=act
nposition= 2 3 4 5 6 7 8 9 10 11 12 end
xlabel=cooling time (years)
dbug=no
nrank=12
aymnuc=  am-241  am-243  np-237  pu-238  pu-239  pu-240
         pu-241  pu-242  u-234   u-235   u-236   u-238
         end
end
'  save plot data file
=shell
cp _plot000  $RTNDIR/_plot.abs_act_pwr4e_60g
end
```

Listing A.8 Example of OPUS nuclide importance listing output for absorption rankings

```
spent fuel 100-year decay
cooling time (years)
absorption fractions
nuclide
    11    13
          0.000E+00 1.000E+00 2.000E+00 3.000E+00 5.000E+00 1.000E+01 2.000E+01 3.000E+01 5.000E+01 7.001E+01 1.000E+02
 u238     2.856E-01 2.895E-01 2.899E-01 2.903E-01 2.909E-01 2.922E-01 2.939E-01 2.950E-01 2.964E-01 2.971E-01 2.979E-01
 pu239    2.574E-01 2.653E-01 2.657E-01 2.660E-01 2.666E-01 2.677E-01 2.692E-01 2.702E-01 2.713E-01 2.719E-01 2.724E-01
 pu240    8.698E-02 8.840E-02 8.874E-02 8.907E-02 8.966E-02 9.089E-02 9.269E-02 9.390E-02 9.521E-02 9.575E-02 9.601E-02
 u235     6.195E-02 6.280E-02 6.290E-02 6.298E-02 6.313E-02 6.340E-02 6.380E-02 6.408E-02 6.442E-02 6.465E-02 6.490E-02
 am241    2.018E-03 4.657E-03 7.152E-03 9.530E-03 1.396E-02 2.330E-02 3.626E-02 4.400E-02 5.092E-02 5.253E-02 5.164E-02
 u236     1.146E-02 1.162E-02 1.164E-02 1.165E-02 1.168E-02 1.173E-02 1.181E-02 1.186E-02 1.193E-02 1.198E-02 1.203E-02
 pu242    1.074E-02 1.089E-02 1.090E-02 1.092E-02 1.094E-02 1.099E-02 1.105E-02 1.109E-02 1.114E-02 1.117E-02 1.120E-02
 np237    8.792E-03 9.056E-03 9.072E-03 9.089E-03 9.122E-03 9.213E-03 9.436E-03 9.698E-03 1.028E-02 1.088E-02 1.178E-02
 pu241    7.987E-02 7.715E-02 7.362E-02 7.024E-02 6.391E-02 5.040E-02 3.127E-02 1.937E-02 7.404E-03 2.825E-03 6.654E-04
 am243    5.466E-03 5.545E-03 5.553E-03 5.559E-03 5.571E-03 5.591E-03 5.619E-03 5.635E-03 5.650E-03 5.654E-03 5.654E-03
 pu238    4.990E-03 5.295E-03 5.314E-03 5.290E-03 5.221E-03 5.041E-03 4.685E-03 4.347E-03 3.729E-03 3.193E-03 2.527E-03
 u234     8.716E-04 9.088E-04 9.359E-04 9.628E-04 1.016E-03 1.144E-03 1.387E-03 1.611E-03 2.009E-03 2.349E-03 2.769E-03
 total    8.188E-01 8.329E-01 8.332E-01 8.334E-01 8.334E-01 8.331E-01 8.324E-01 8.320E-01 8.314E-01 8.309E-01 8.304E-01
```

Listing A.9 Example of OPUS input for total decay-heat calculations

```
'  $HOME/nrc/ranking/bwr/4.0_gd_rerun/opus_watts_all_bwr5e_4gd_70g.in
'
'  this case begins with the concentrations in position 2 of ft71f001
'  which has 12 positions.  position 1 has the end of irradiation
'  concentrations from the sas2 case.  position 2 has the concentrations
'  from the end of the sas2 decay case (zero length).  positions
'  3 through 12 are from the origen-s case run separately.
'
'  (1) first link after =shell is to link Ian's
'      latest opus executable
'
'  (2) link below would use pwr 33 GWd/MTU
'      library but this does not use the
'      correct cross sections for absorption
' ln -s $DATA/pwr33gwd            ft33f001
'
'      note that any library can be used to
'      get the other data needed for opus--
'      it takes the first library found in a
'      multiburnup library
'
'  (3) second link after =shell uses last
'      temporary library from ft15f001 in
'      the sas2 run--ft15f001 is the unit
'      for temporary storage of binary
'      libraries before they are added to the
'      back of the multiburnup library on
'      ft33f001
'
'  (3) third link after =shell uses ft71f001
'      saved from origen-s decay case run as
'      a separate job after sas2 run that
'      saved 12 positions on ft71f001
'
=shell
ln -s /home/2ul/bin/opus     opus
ln -s $RTNDIR/bwr5e_4gd_70g.ft15f001 ft15f001
ln -s $RTNDIR/bwr5e_4gd_70g.ft71f001 ft71f001
end
=opus
default=no
libunit=15
units=watts
time=year
typarms=nucl
tmax=100.1
libtyp=all
nposition= 2 3 4 5 6 7 8 9 10 11 12 end
xlabel=cooling time (years)
dbug=no
nrank=25
symnuc= ag-110m am-241  am-243  ba-137m cs-134  cs-137
        ce-144  cm-242  cm-243  cm-244  eu-154  kr-85
        np-239  pm-147  pr-144  pu-238  pu-239  pu-240
        pu-241  rh-106  ru-106  sb-125  sr-90   y-90
        co-60   end
end
'  save plot data file
=shell
cp _plot000  $RTNDIR/_plot.watts_all_bwr5e_4gd_70g
end
```

Listing A.10 Example of OPUS nuclide importance listing output for decay-heat rankings

```
spent fuel 100-year decay
cooling time (years)
thermal power, watts / mthm
nuclide
    11    26
          0.000E+00 1.000E+00 2.000E+00 3.000E+00 5.000E+00 1.000E+01 2.000E+01 3.000E+01 5.000E+01 7.001E+01 1.000E+02
 pu238    3.636E+02 3.774E+02 3.776E+02 3.753E+02 3.695E+02 3.553E+02 3.283E+02 3.034E+02 2.591E+02 2.212E+02 1.746E+02
 ba137m   8.254E+02 8.008E+02 7.825E+02 7.647E+02 7.301E+02 6.505E+02 5.163E+02 4.097E+02 2.581E+02 1.626E+02 8.129E+01
 y90      8.549E+02 7.936E+02 7.742E+02 7.554E+02 7.191E+02 6.358E+02 4.970E+02 3.885E+02 2.374E+02 1.451E+02 6.930E+01
 am241    8.282E+00 1.830E+01 2.783E+01 3.690E+01 5.372E+01 8.903E+01 1.375E+02 1.661E+02 1.913E+02 1.968E+02 1.929E+02
 cm244    8.048E+02 7.758E+02 7.466E+02 7.186E+02 6.656E+02 5.496E+02 3.747E+02 2.555E+02 1.188E+02 5.522E+01 1.750E+01
 cs137    2.455E+02 2.399E+02 2.344E+02 2.291E+02 2.187E+02 1.949E+02 1.547E+02 1.227E+02 7.732E+01 4.871E+01 2.435E+01
 sr90     1.705E+02 1.664E+02 1.624E+02 1.584E+02 1.508E+02 1.333E+02 1.042E+02 8.147E+01 4.978E+01 3.042E+01 1.453E+01
 pu240    2.418E+01 2.425E+01 2.432E+01 2.439E+01 2.451E+01 2.479E+01 2.519E+01 2.545E+01 2.574E+01 2.584E+01 2.585E+01
 pu239    1.111E+01 1.130E+01 1.130E+01 1.130E+01 1.129E+01 1.129E+01 1.129E+01 1.129E+01 1.129E+01 1.128E+01 1.127E+01
 am243    2.934E+00 2.936E+00 2.936E+00 2.936E+00 2.935E+00 2.934E+00 2.931E+00 2.928E+00 2.923E+00 2.917E+00 2.909E+00
 eu154    1.506E+02 1.389E+02 1.281E+02 1.182E+02 1.006E+02 6.719E+01 2.998E+01 1.338E+01 2.664E+00 5.304E-01 4.713E-02
 kr85     2.453E+01 2.300E+01 2.156E+01 2.021E+01 1.776E+01 1.285E+01 6.731E+00 3.526E+00 9.674E-01 2.654E-01 3.815E-02
 cm243    2.778E+00 2.711E+00 2.646E+00 2.582E+00 2.460E+00 2.178E+00 1.708E+00 1.339E+00 8.233E-01 5.062E-01 2.440E-01
 pu241    6.113E+00 5.825E+00 5.550E+00 5.288E+00 4.801E+00 3.771E+00 2.326E+00 1.435E+00 5.461E-01 2.078E-01 4.884E-02
 cm242    4.026E+03 8.558E+02 1.814E+02 3.874E+01 2.191E+00 4.691E-01 4.458E-01 4.244E-01 3.846E-01 3.486E-01 3.008E-01
 np239    5.500E+04 2.306E-01 2.306E-01 2.306E-01 2.306E-01 2.305E-01 2.302E-01 2.300E-01 2.296E-01 2.292E-01 2.285E-01
 co60     7.855E+01 6.887E+01 6.038E+01 5.294E+01 4.069E+01 2.108E+01 5.657E+00 1.518E+00 1.093E-01 7.872E-03 1.521E-04
 cs134    4.215E+03 3.012E+03 2.152E+03 1.537E+03 7.848E+02 1.461E+02 5.067E+00 1.757E-01 2.112E-04 2.538E-07 1.058E-11
 sb125    4.266E+01 3.322E+01 2.577E+01 1.999E+01 1.203E+01 3.380E+00 2.668E-01 2.105E-02 1.311E-04 8.169E-07 4.016E-10
 pm147    6.723E+01 5.362E+01 4.117E+01 3.161E+01 1.864E+01 4.972E+00 3.540E-01 2.520E-02 1.278E-04 6.476E-07 2.337E-10
 rh106    8.916E+03 4.129E+03 2.089E+03 1.057E+03 2.706E+02 8.973E+00 9.869E-03 1.085E-05 1.312E-11 1.587E-17 0.000E+00
 pr144    9.111E+03 3.718E+03 1.529E+03 6.288E+02 1.064E+02 1.251E+00 1.732E-04 2.397E-08 4.590E-16 0.000E+00 0.000E+00
 ru106    5.062E+01 2.561E+01 1.296E+01 6.557E+00 1.679E+00 5.566E-02 6.121E-05 6.732E-08 8.141E-14 9.844E-20 1.309E-28
 ag110m   1.747E+02 6.338E+01 2.300E+01 8.345E+00 1.099E+00 6.914E-03 2.737E-07 1.083E-11 1.697E-20 0.000E+00 0.000E+00
 ce144    8.062E+02 3.315E+02 1.364E+02 5.608E+01 9.484E+00 1.116E-01 1.544E-05 2.137E-09 4.093E-17 7.840E-25 0.000E+00
 total    2.119E+06 1.625E+04 9.584E+03 6.676E+03 4.328E+03 2.923E+03 2.206E+03 1.790E+03 1.238E+03 9.027E+02 6.160E+02
```

Listing A.11 Example of OPUS input for dose rankings

```
'  $HOME/nrc/ranking/pwr/opus_tn24_total_pwr4e_60g.in
'
'  this case begins with the concentrations in position 5 of ft71f001
'  which has 12 positions.  position 1 has the end of irradiation
'  concentrations from the sas2 case.  position 2 has the concentrations
'  from the end of the sas2 decay case (zero length).  positions
'  3 through 12 are from the origen-s case run separately.
'
=shell
ln -s /home/2ul/bin/opus     opus
cp $RTNDIR/pwr4e_60g.ft71f001 ft71f001
end
=opus
default=no
units=gatom
timeunit=year
typarms=nucl
nrank=21
tmin=99
tmax=101
libtyp=all
' source position 5/6/8/12 = (3/5/20/100 years)
npos=5 6 8 12 end
title=pwr spent fuel - tn24p cask
xlabel=cooling time (years)
dbug=no
symnuc= ag-110m am-241 ba-137m cf-252 cm-242 cm-244
        cm-246  cm-248 co-60 cs-134 eu-152 eu-154
        nb-94 pr-144 pu-238 pu-239 pu-240 pu-242
        rh-106 u-238 y-90   end
response=
 270600   1.94259E-01
 822100   5.25375E-13
 832101   9.61070E-10
 882260   1.53387E-06
 882280   0.00000E+00
 892270   1.69048E-06
 902280   2.09268E-03
 902290   3.72571E-07
 902300   3.00311E-08
 902320   7.66638E-14
 912310   8.96271E-08
 922320   5.34966E-05
 922330   1.60745E-08
 922340   9.92199E-09
 922350   2.56493E-12
 922360   9.15256E-11
 922380   3.81300E-11
 932350   3.78970E-08
 932360   4.38647E-13
 932370   1.07664E-09
 942360   1.82302E-03
 942370   1.35276E-06
 942380   5.47615E-05
 942390   1.36725E-07
 942400   3.41827E-06
 942420   4.90282E-06
 942440   5.36144E-06
 952410   9.64205E-06
 952421   5.66585E-07
 952430   4.94377E-07
 962410   6.18662E-04
 962420   7.09660E-02
 962430   1.88070E-04
 962440   3.22196E-02
 962450   5.85597E-07
 962460   2.57231E-02
```

```
962470   1.82676E-10
962480   1.19614E-01
962500   1.81793E+01
982490   2.32673E-05
982500   3.25167E+01
982510   5.76646E-06
982520   6.88558E+03
982540   3.66252E+06
992540   9.24555E-03
992550   2.43050E+02
290670   1.52006E-06
300690   6.49474E-03
300691   1.33177E-01
300710   7.94277E+02
300711   1.76353E-01
360850   1.10008E-07
380871   4.69315E-03
390891   6.42057E+04
390900   2.03180E-01
390901   5.79942E-13
390910   5.63071E-03
410931   0.00000E+00
410940   1.75813E-06
400950   2.62999E-02
410950   4.99094E-02
410951   2.30020E-07
430980   2.11386E-09
441030   4.12748E-03
451031   0.00000E+00
451060   5.66393E+04
451061   4.22436E+03
471081   6.12461E-05
481090   0.00000E+00
471101   8.81071E-01
481131   4.72834E-10
491141   1.53620E-03
481151   2.13727E-02
501191   0.00000E+00
501230   4.54650E-03
521230   0.00000E+00
521231   1.40786E-17
511240   1.66303E+01
511250   4.21292E-04
521251   6.72294E-20
501260   4.08677E-17
511260   1.08200E+00
511261   4.98629E+02
521271   4.45711E-08
541270   8.57585E-06
521291   3.83956E-03
531290   4.05650E-30
561330   1.27031E-07
551340   3.03953E-02
561351   4.86599E-07
561371   7.64905E+02
571380   1.54096E-14
581390   1.21477E-17
581410   2.00932E-08
591420   2.46229E+01
581440   4.29357E-13
591440   3.23616E+03
591441   1.47287E+00
611450   5.87808E-18
621450   7.43073E-09
611460   3.17862E-04
611470   1.20024E-15
611481   5.16083E-01
621510   0.00000E+00
631520   2.67835E-02
641530   1.14169E-18
631540   2.15569E-02
```

```
 631550   7.56315E-16
 651600   5.38109E-01
 661660   4.40653E-09
 671661   3.91505E-05
 681690   1.36797E-10
 701690   4.70836E-06
 691700   2.31590E-06
 681710   6.65187E-01
 691710   6.88323E-30
 681720   2.83019E-09 end
end
'  save plot data file
=shell
cp _plot000  $RTNDIR/_plot_tn24_total_pwr4e_60g
end
```

Listing A.12 Example of OPUS nuclide importance listing output for dose rankings

```
pwr spent fuel - tn24p cask
cooling time (years)
arbitrary user input  response
nuclide
     4    22
           3.000E+00 5.000E+00 2.000E+01 1.000E+02
 cm244     2.694E-02 2.495E-02 1.405E-02 6.561E-04
 cm246     2.997E-04 2.996E-04 2.990E-04 2.955E-04
 ba137m    1.750E-03 1.671E-03 1.181E-03 1.860E-04
 am241     1.307E-05 1.909E-05 4.912E-05 6.900E-05
 pu238     1.185E-04 1.167E-04 1.037E-04 5.516E-05
 pu240     4.573E-05 4.593E-05 4.702E-05 4.803E-05
 y90       4.736E-04 4.508E-04 3.116E-04 4.345E-05
 pu242     2.678E-05 2.678E-05 2.678E-05 2.678E-05
 cm248     6.773E-06 6.777E-06 6.782E-06 6.781E-06
 pu239     3.471E-06 3.471E-06 3.470E-06 3.464E-06
 eu154     5.901E-03 5.021E-03 1.497E-03 2.353E-06
 cm242     8.756E-05 4.996E-06 1.049E-06 7.080E-07
 u238      1.464E-07 1.464E-07 1.464E-07 1.464E-07
 co60      2.914E-02 2.240E-02 3.113E-03 8.374E-08
 eu152     7.557E-06 6.810E-06 3.121E-06 4.867E-08
 nb94      1.655E-11 1.655E-11 1.654E-11 1.650E-11
 cf252     5.554E-04 3.289E-04 6.452E-06 5.060E-15
 cs134     2.261E-02 1.154E-02 7.451E-05 1.555E-16
 rh106     1.663E-02 4.256E-03 1.552E-07 3.319E-31
 pr144     2.516E-02 4.255E-03 6.928E-09 1.360E-41
 ag110m    7.617E-04 1.003E-04 2.498E-11 0.000E+00
 total     1.305E-01 7.552E-02 2.078E-02 1.395E-03
```

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-6700
ORNL/TM-2000/284

2. TITLE AND SUBTITLE

Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	2001

4. FIN OR GRANT NUMBER

W6479

5. AUTHOR(S)

I.C. Gauld, J.C. Ryman

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Oak Ridge National Laboratory
Managed by UT-Battelle, LLC
Oak Ridge, TN 37831-6370

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

D.D. Ebert, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This report investigated trends in the radiological decay properties and changes in relative nuclide importance associated with increasing enrichments and burnup for spent LWR fuel as they affect the areas of criticality safety, thermal analysis (decay heat), and shielding analysis of spent fuel transport and storage casks. To facilitate identifying the changes in the spent fuel compositions that most directly impact these application areas, the dominant nuclides in each area have been identified and ranked by importance. The importance is investigated as a function of increasing burnup to assist in identifying the key changes in spent fuel characteristics between contentional- and extended-burnup regimes. Studies involving both pressurized-water reactor (PWR) fuel assemblies and boiling-water-reactor (BWR) assemblies are included. This information is seen to be an essential step in identifying the high-burnup spent fuel characteristics that may adversely affect the accuracy of current computational methods and data, assess the potential impact on previous guidance on isotopic source terms and decay-heat values, and thus help identify areas for methods and data improvement. Finally, several recommendations on the direction of possible future code validation efforts for high-burnup spent fuel predictions are presented.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

high burnup LWR fuel, spent fuel transport and storage systems, calculation of spent fuel inventories, decay heat, and radiation source terms for cask shielding

13. AVAILABILITY STATEMENT

unlimited

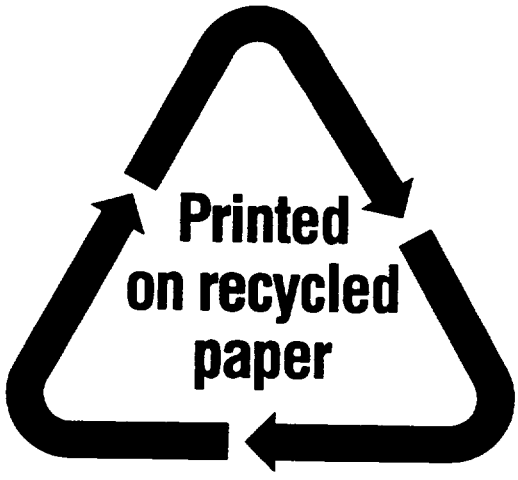
14. SECURITY CLASSIFICATION

(This Page) unclassified

(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE



Federal Recycling Program