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Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code

S. B. Ludwig
J. P. Renier

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STANDARD- AND EXTENDED-BURNUP PWR AND BWR REACTOR MODELS FOR THE ORIGEN2 COMPUTER CODE

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NOTICE: This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

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GLOSSARY

Buildup	The creation of isotopes in nuclear reactor fuel and structural materials due to nuclear transmutation.
BWR	Boiling-water reactor.
BWR-U	Refers to an ORIGEN2 reactor model designed to use 2.75 wt % ^{235}U fuel in a wholly uranium-enriched four-cycle BWR to achieve a standard burnup of 27,500 MWd/MTIHM. The BWR-U model was first introduced in 1978 (see ref. 7).
BWR-UE	Refers to an ORIGEN2 reactor model designed to use 3.4 wt % ^{235}U fuel in a wholly uranium-enriched four-cycle BWR to achieve an extended burnup of 40,000 MWd/MTIHM. The BWR-UE model is a new model and is documented in this report.
BWR-US	Refers to an ORIGEN2 reactor model designed to use 3.0 wt % ^{235}U fuel in a wholly uranium-enriched four-cycle BWR to achieve a standard burnup of 27,500 MWd/MTIHM. The BWR-US model updates the BWR-U model and is documented in this report. The major difference between the BWR-US and the BWR-U models is the explicit use of axially varying moderator density in the BWR-US model, rather than a constant (average) density assumed in the BWR-U model.
BWR-US0	Refers to an ORIGEN2 reactor model designed to use 3.0 wt % ^{235}U fuel in a wholly uranium-enriched four-cycle BWR to achieve a standard burnup of 27,500 MWd/MTIHM. The BWR-US0 model updates the BWR-U model and is documented in this report. Both BWR-U and BWR-US0 assume a constant (average) axial moderator density.
BWR-UX	Refers to the BWR-U reactor model driven to an extended burnup of 40,000 MWd/MTIHM. This case was run solely for comparison with the new BWR-UE model, using a higher initial fuel enrichment and increased number of irradiation cycles.
CANDU	Canadian D_2O-moderated, uranium-fueled reactor (natural and slightly enriched uranium fuel designs).
Depletion calculation	Calculational irradiation of fresh reactor fuel, resulting in the prediction of the discharged fuel composition.

Extended burnup	As used in this document, extended burnup refers to the irradiation of fuel to a design burnup of 50,000 MWd/MTIHM for PWRs or 40,000 MWd/MTIHM for BWRs. This is accomplished by (1) increasing the initial enrichment of ^{235}U in fresh fuel and (2) increasing the total time the fuel is irradiated.
Fuel element	The smallest structurally discrete part of a fuel assembly that has nuclear fuel as its principal constituent; also called a fuel pin or a fuel rod.
Fuel assembly	A fuel assembly is made up of a group of fuel elements mechanically joined in an array by metal grids and loaded into a nuclear reactor core as a single unit. In addition to the fuel rods, fuel assemblies may also contain burnable-poison rods, control rods, water rods, instrumentation, and other structural materials.
GWd	Gigawatt-days = 10^9 watt-days.
HTGR	High-temperature, gas-cooled reactor.
LMFBR	Liquid-metal, fast breeder reactor (sodium cooled and moderated).
LWR	Light-water reactor (i.e., BWRs and PWRs).
MSBR	Molten-salt breeder reactor.
MTIHM	Metric tons (= 10^6 g) of initial heavy metal, interpreted as total elemental mass fuel (both fissile and nonfissile isotopes of, commonly, uranium or uranium/plutonium) before any irradiation.
MWd	Megawatt-days = 10^6 watt-days.
Pin cell	A cylindrical model of a fuel element used in a reactor physics calculation.
PUD50	Refers to an ORIGEN2 reactor model designed to use 4.15 wt % ^{235}U fuel in a wholly uranium-enriched five-cycle PWR to achieve an extended burnup of 50,000 MWd/MTIHM. The PUD50 model was first introduced in 1980 (see ref. 8).
PWR	Pressurized-water reactor.

PWR-U	Refers to an ORIGEN2 reactor model designed to use 3.2 wt % ^{235}U fuel in a wholly uranium-enriched three-cycle PWR to achieve a standard burnup of 33,000 MWd/MTIHM. The PWR-U model was first introduced in 1978 (see ref. 7).
PWR-UE	Refers to an ORIGEN2 reactor model designed to use 4.2 wt % ^{235}U fuel in a wholly uranium-enriched three-cycle PWR to achieve an extended burnup of 50,000 MWd/MTIHM. The PWR-UE model updates the PUD50 model and is documented in this report. The PWR-UE model uses three truly "extended" cycles, rather than five "normal-length" cycles to achieve extended burnup.
PWR-US	Refers to an ORIGEN2 reactor model designed to use 3.2 wt % ^{235}U fuel in a wholly uranium-enriched three-cycle PWR to achieve a standard burnup of 33,000 MWd/MTIHM. The PWR-US model updates the PWR-U model and is documented in this report.
Reactor model	In relation to the ORIGEN2 computer code, a reactor model is a one-group, zero-dimensional (point) cross-section library used by the ORIGEN2 computer code to simulate the depletion, buildup, and decay of radionuclides in a nuclear reactor. Each ORIGEN2 cross-section library is, therefore, dependent on reactor type, initial enrichment, and other fuel cycle parameters (e.g., specific power and both duration and number of irradiation cycles) from which it was developed.
Standard burnup	As used in this document, standard burnup refers to the irradiation of fuel to a design burnup of 33,000 MWd/MTIHM for PWRs or 27,500 MWd/MTIHM for BWRs.

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ABSTRACT

Multidimensional reactor physics calculations have been conducted, leading to the creation of new or revised enriched-uranium PWR and BWR reactor models for the ORIGEN2 computer code. These models are based on cross sections that have been taken directly from reactor physics codes, thus eliminating the need for making adjustments in cross sections in order to obtain accurate depletion results. A new ORIGEN2 reactor model for an extended-burnup BWR (to 40,000 MWd/MTIHM) has been prepared. Using the same computational techniques, reactor models for standard- and extended-burnup PWRs (to 33,000 MWd/MTIHM and 50,000 MWd/MTIHM, respectively) have been revised. Also, two versions of the ORIGEN2 reactor model for a standard-burnup BWR (to 27,500 MWd/MTIHM) have been prepared. The first BWR standard-burnup version, as well as the extended-burnup BWR model, were prepared assuming variation in coolant density axially within the reactor core. The second BWR standard-burnup version was prepared assuming a constant (average) coolant density within the core region. This version was prepared to provide comparison with an earlier standard-burnup BWR model based on the same coolant density assumption. Recommended fuel and structural material masses and compositions are presented. A summary of the new ORIGEN2 reactor models is given.

STANDARD- AND EXTENDED-BURNUP PWR AND BWR REACTOR MODELS FOR THE ORIGEN2 COMPUTER CODE

S. B. Ludwig
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1. INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The purpose of this report is to describe an updated set of reactor models for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) operating on uranium fuel cycles and the methods used to generate the information for these models. Since new fuel cycle schemes and reactor core designs are introduced from time to time by reactor manufacturers and fuel vendors, an effort has been made to update these reactor models periodically and to expand the data bases used by the ORIGEN2 computer code. In addition, more sophisticated computational techniques than previously available were used to calculate the resulting reactor model cross-section libraries. The PWR models were based on a Westinghouse design,[1] while the BWR models were based on a General Electric BWR/6 design.[2] The specific reactor types considered in this report are as follows (see Glossary for the definition of these and other terms): (1) PWR-US, (2) PWR-UE, (3) BWR-US, (4) BWR-US0, and (5) BWR-UE. Each reactor model includes a unique data library that may be used to simulate the buildup and depletion of isotopes in nuclear materials using the ORIGEN2 computer code.[3]

1.1.1 Background

In the late 1960s and early 1970s, the ORIGEN computer code[4] was developed by the Chemical Technology Division at Oak Ridge National Laboratory (ORNL). Initial versions of the required nuclear data libraries (half-lives, branching ratios, cross sections, fission product

yields, photon and neutron yields, etc.) and reactor models (HTGR, LWR, LMFBR, and MSBR) were also developed. The code was principally intended for use in generating spent fuel and waste characteristics (composition, thermal power, etc.) that would form the basis for the study and design of fuel reprocessing plants, shipping casks, and waste treatment and disposal facilities. Since the fuel cycle operations were being examined generically over a wide range of fuel characteristics, it was only necessary that the ORIGEN results be somewhat representative of this range.

The ORIGEN code was made available to users outside ORNL through the Radiation Shielding Information Center (at ORNL). The relative simplicity of ORIGEN, coupled with its convenient and detailed output, led to its acquisition by many organizations. The uses of ORIGEN soon shifted from generic fuel cycle applications to specific calculations for environmental impact studies, requiring relatively precise calculations of minor isotopes such as ^{3}H, ^{14}C, ^{232}U, and $^{242,244}Cm$. The attempts made to update specific aspects of ORIGEN and its data bases yielded inconsistencies and a large number of different data bases.

In 1975, a concerted program was initiated to update the ORIGEN computer code and its associated data bases and reactor models. A revised and updated version of the original ORIGEN computer code (called ORIGEN2) was released in 1980, along with an extensive User's Manual.[5,6] In recent years, the User's Manual has been updated periodically through changes made by ORNL staff considering contributions from the code's many users around the world. Versions of the code are available for IBM, CDC, VAX, UNIVAC, CRAY, and PRIME computers, as well as for IBM-compatible personal computers. ORIGEN2 appears to be adaptable to most mainframe and minicomputers.

The initial reactor models for LWRs were expanded to include specific PWR and BWR types operating with enriched uranium, uranium/plutonium, and depleted uranium/thorium fuel cycles.[7,8] ORIGEN2-compatible reactor models for the LMFBR[9] and CANDU[10] were also developed and documented. In addition, the decay data and photon libraries[11] were updated and documented. All the reactor models were

developed using simplified, one-dimensional, multigroup reactor physics calculational techniques. The resulting cross-section libraries were a significant improvement over the models included in the original ORIGEN computer code and were primarily based on the existing compilations of the ENDF/B-IV cross-section data from the National Neutron Cross Section Center at Brookhaven National Laboratory.

1.1.2 Scope

The scope of this project was to prepare an updated set of reactor models for PWRs and BWRs operating on uranium fuel cycles, using the most recent data on fuel cycle schemes and reactor core designs from the reactor manufacturers and more sophisticated reactor physics computational techniques than previously used. The fundamental need was to provide an extended-burnup BWR reactor model for ORIGEN2 (because none was previously available). The results of this project are the creation of five new reactor models and associated ORIGEN2 cross-section libraries. The BWR-US and BWR-UE are completely new reactor models, while the PWR-US, PWR-UE, and BWR-US0 are updates to previously developed ORIGEN2 reactor models.[7,8]

1.1.3 Improvements in Computational Techniques

As mentioned above, the reactor physics computational techniques employed in the preparation of the new and/or updated LWR models were more sophisticated than those techniques previously available. A summary of the previous modeling effort is described in Sect. 2.1.

The most obvious improvement in computational sophistication comes from the explicit use of multidimensional calculations in the depletion model, whereas previous modeling techniques used quasi-zero-dimensional (one-dimensional over an extremely small volume) assumptions to model the depletion of the entire batch of fuel assemblies in the core.

The addition of multidimensional computations allows explicit variation of the axial moderator density within the BWR core, whereas earlier analysis had held this value constant. The previous

calculations assumed an average saturated steam volume fraction of 0.365 at a pressure of 1050 psia, whereas the current BWR models assume this average volume fraction to be 0.392. For the current BWR-US and BWR-UE models, the moderator density is varied from 0.72 g/cm^3 (at the bottom of the fuel region) to 0.28 g/cm^3 (at the top of the fuel region), with the average density assumed to be 0.429 g/cm^3. The previous work[7] assumed the average moderator density to be 0.484 g/cm^3. Since moderation of the neutron spectrum is proportional to the density of the moderator, the neutron spectrum for the current models is moderated to a lesser extent (or "harder") than the "softer" neutron spectrum used in calculating the previous reactor models.

Also, the use of multidimensional techniques allows the explicit calculation of fuel depletion within a BWR fuel assembly containing fuel rods at different initial enrichments, rather than assuming an assembly-averaged enrichment as was the previous practice.

For the PWR, the more-sophisticated techniques now allow the variation of soluble boron concentration in the moderator, whereas previous calculations assumed an average value. All control rods are assumed to be fully withdrawn from the core at startup and the reactor is kept critical at the desired power level using the injection or removal of soluble boron in the moderator. Typically, the boron concentration is 1200 ppm at the beginning of cycle (BOC) and approximately 100 ppm at the end of cycle (EOC). The high concentration of soluble boron at BOC causes the neutron spectrum to be relatively hard; the spectrum becomes progressively softer as the boron concentration is lowered. This change in neutron-energy spectra throughout the fuel cycle is important in determining the cross sections for burnup-dependent actinides and impacts the fast-to-thermal fission ratio. Such a phenomenon can be modeled in ORIGEN2 by correctly determining the cross sections of certain actinide reactions at various burnups, accounting for the assumed concentration of soluble boron.

1.2 SUMMARY

This project involved the gathering and processing of a large amount of diverse data that led to the generation of updated ORIGEN2 reactor models for standard- and extended-burnup PWR and BWR fuel cycles. The specific types of information developed for the PWR-US, PWR-UE, BWR-US, BWR-US0, and BWR-UE reactor models are as follows:

1. 84-energy-group neutron spectra;
2. one-group, burnup-dependent cross sections for the major actinides;
3. one-group, "typical" cross sections for 1300 nuclides (including the actinides); and
4. new values for the ORIGEN flux parameters THERM, RES, and FAST.

In addition, information developed previously[7] and used in the development of the new reactor models is included in this report for completeness. This information includes:

1. recommended initial heavy-metal compositions of fresh fuel;
2. recommended initial metal compositions of fresh fuel-assembly structural materials; and
3. recommended minor constituent concentrations for both the oxide fuel and the structural materials.

Using this information and a modified version of the ORIGEN2 computer code (named ORIGEN2 Version 2.1), depletion calculations were made for each of the five reactor model types. The results of these calculations were then compared with those obtained using the old ORIGEN2 reactor models.

In general, the results of the new vs old ORIGEN2 depletion calculations agree fairly well, with noted differences caused primarily by the use of different modeling techniques or changes in evaluated cross-section data for specific isotopes resulting from the use of updated versions of the nuclear data bases. Four basic comparisons are provided in Sect. 5: (1) actinide composition of spent fuel at discharge and after a 10-year decay period, (2) fission product and activation product radioactivity of a spent fuel assembly after a 10-year decay

period, (3) selected ORIGEN2 output totals (radioactivity, thermal power, neutron source, and photon source) for spent fuel assemblies after a 10-year decay period, and (4) actinide composition vs literature (experimentally measured) values.

As expected, the results of the PWR-U and PWR-US cases agree fairly well. One major difference across all five cases is the large increase in the ^{242m}Am composition (at discharge) and the subsequent increase in the ^{242}Am and ^{242}Cm composition (after a 10-year decay period) as calculated by ORIGEN2 using the updated LWR models. The root of this increase is the ORIGEN2 cross section for ^{240}Pu, which exhibits a difference directly proportional to the difference in concentration of americium or curium isotope. A possible explanation for this difference is that the updated ORIGEN2 cross-section libraries have resulted from a neutron energy spectrum that is, on the average, "harder," and ^{240}Pu is very sensitive to the assumed energy spectrum. Also, the cross-section data for ^{240}Pu were extensively reevaluated in the update from ENDF/B-IV to ENDF/B-V. The previous models were developed using ENDF/B-IV data, but the updated models were prepared using ENDF/B-V data compilations. The result of these differences is a ^{240}Pu (n,γ) ORIGEN2 (one-group) cross section that is significantly greater than the same cross section found in the previous ORIGEN2 LWR models. A combination of (n,γ) reactions and β^- decays causes the large increase in buildup of ^{242}Cm, and, to a lesser extent, ^{244}Cm. Both ^{242}Cm and especially ^{244}Cm are important neutron sources in shielding calculations.

The agreement of results for the BWR-UX vs BWR-UE case is poor, as expected. Previously, no ORIGEN2 reactor model existed for extended-burnup BWR calculations. In many cases, the BWR-U model was used erroneously to calculate extended-burnup BWR results. This comparison has been included to provide a perspective on the differences associated with the use of an ORIGEN2 model with parameters that far exceed the ranges for which the ORIGEN2 model was designed. It is important to note that previous use of the BWR-U model (for extended-burnup calculations) may severely underestimate the actinide composition of the BWR spent fuel. This is primarily attributed to the dependency of the ORIGEN2 model on the initial ^{235}U enrichment of the fuel. By increasing

the initial fuel enrichment to achieve extended burnup, as was done in the BWR-UX case, no corresponding change was made, or, for that matter, can be made in the one-group cross sections found in the BWR-U library.

The activation product radioactivities show a wider variation of results. Most noteworthy are those for ^{60}Co, ^{93}Zr, and $^{154,155}Eu$. All four radionuclides are strong gamma-ray (as well as thermal) sources. The increase or decrease in the generation of these activation products appears to be caused by the use of revised multigroup cross-section data, the sensitivity of the multigroup cross-section data to the calculated neutron energy spectrum used to collapse these data to one group, or both. One interesting comparison to note is that the total radioactivity of the activation products for the updated BWR models increases to a lesser extent than for the PWR models. This finding is attributed to the much greater composition of gadolinium (as a burnable poison) assumed to be present in BWR spent fuel.

The total radioactivity and the total photon emission rate for spent fuel assemblies agree within 1 to 3%.

The increase in thermal power (2 to 6% for PWRs, 1 to 10% for BWRs) is important to note since the design and optimization of shipping casks and storage facilities require these data.

Another very important result is the overall increase in the neutron source strength (12 to 15% for the PWR cases, 3 to 80% for the BWR cases), which would affect the shielding design of shipping casks. The approximate 80% increase in neutron source strength in the BWR case, is, however, attributed to the erroneous previous use of the BWR-U model in extended-burnup calculations.

ORIGEN2 (with the PWR-US model) was used to simulate the burnup of fuel to a burnup of 31,364 MWd/MTIHM, using fuel of an initial enrichment of 2.561 wt % ^{235}U. These results were then compared with the experimentally measured composition of discharged spent fuel from the H. B. Robinson reactor at the same conditions of enrichment and burnup. The ORIGEN2-predicted composition shows excellent agreement with the experimental measurements. Further comparisons with experimental data will be the subject of a future report as ORNL continues to validate the ORIGEN2 computer code.

2. DESCRIPTION OF LWR CORE CHARACTERISTICS

In the United States, and indeed throughout the world, the most popular of the present generation of reactors, the light-water reactor (LWR), uses ordinary water as a moderator and coolant. Such reactors operate at very high pressures in order to achieve high operating temperatures while maintaining most or all of the water in a liquid phase. If the water is allowed to boil in the core, the reactor is referred to as a BWR. If the system pressure is kept sufficiently high to prevent bulk boiling, the reactor is known as a PWR. Table 2.1 shows typical nuclear power reactor design data for some common PWRs and BWRs. In the late 1970s, ORNL developed ORIGEN2 reactor models[7,8] for standard- and extended-burnup PWRs, as well as a standard-burnup BWR.

Recent surveys[12,13] have shown, however, that there are currently 35 different PWR fuel assembly types, ranging from array sizes of 13 x 13 to 17 x 17, and 23 different BWR fuel assembly types, ranging from array sizes of 6 x 6 to 11 x 11. Various types are manufactured by four different vendors. In some cases, the fuel assembly design is unique to a particular reactor or power plant. The ORIGEN2 reactor models described herein were developed to be representative of current LWR fuel cycle behavior, using a single "average" PWR and "average" BWR fuel assembly as models. These models use one-energy-group, zero-dimensional cross sections and a scalar, energy-independent neutron flux, thus removing the spatial- and energy-dependence of properties used in complex neutronic calculations. What remains is a fairly simple (yet extensive) simulation of the burnup, depletion, irradiation, and decay of nuclear fuel and associated structural materials based on a "generic" fuel cycle. The only variables remaining in the calculation are the type of fuel cycle, initial enrichment of the fuel, total burnup, and the specific power and irradiation/decay history of the fuel. Eliminating the spatial- and energy-dependency from the neutronics calculation, however, makes ORIGEN2, by design, most applicable for calculations of average nuclear fuel and structural material radiological properties of, say, an entire core.

Table 2.1. Typical LWR design data[a]

Reactor/fuel vendor	PWR (W)	PWR (B&W)	PWR (CE)	BWR/6 (GE)
General Data				
Thermal output, MWt	3411	3600	3800	3579
Electrical output, MWe	1150	1200	1300	1200
Efficiency, %	33.7	33.3	34.2	33.5
Fuel type	UO_2	UO_2	UO_2	UO_2
Coolant/moderator	H_2O	H_2O	H_2O	H_2O
Structural material	Zircaloy	Zircaloy	Zircaloy	Zircaloy-2
Core Data				
Active height, cm	366	363	381	376
Equivalent active diameter, cm	337	352	363	366
Fuel weight, kg	90,200	94,900	103,000	138,000
Specific power, MW/MTU	37.8	37.9	36.9	25.9
Burnup, MWd/MTU	33,000	33,000	33,000	27,500
Fuel Assemblies				
Number of assemblies	193	205	241	732
Fuel-element array	17x17	17x17	16x16	8x8
Assembly lateral dimension, cm	21.4	21.7	20.3	14.0
Assembly pitch, cm	21.5	21.8	20.7	30.5
Number of fuel elements/assembly	264	264	236	63
Total number of fuel locations/core	50,952	54,120	56,876	46,116
Fuel-Element Data				
Fuel-element pitch, cm	1.25	1.27	1.28	1.62
Fuel-element OD, cm	0.94	0.96	0.97	1.25
Clad thickness, cm	0.0572	0.0597	0.0635	0.0864
Fuel-pellet diameter, cm	0.819	0.823	0.825	1.056
Pellet-clad gap, cm	0.0082	0.010	0.0089	0.008
Initial enrichment, % ^{235}U	2.1/2.6/3.1	2.91	1.9/2.4/2.9	2.2-2.7
Equilibrium cycle enrichment	3.2	-	-	2.75
Thermal Hydraulic Data				
System pressure, bar	155	155	155	72
Average linear power density, W/cm	178	178	175	206
Inlet temperature, °C	300	300	296	269
Outlet temperature, °C	332	333	328	286
Maximum fuel temperature, °C	1788	2021	1882	1829

[a]Data derived from ref. 14.

In the ensuing years, LWR operating trends have progressed to extending the fuel burnup by increasing the initial enrichment of the fuel and using more complicated core loading and fuel management schemes. For this reason, ORNL has revised its ORIGEN2 reactor models for LWRs so as to provide an ORIGEN2 extended-burnup model for the BWR and to update the extended-burnup model for the PWR. The current standard-burnup models for the PWR and BWR have also been revised with the same computational techniques used in creating or updating the extended-burnup models in order to eliminate any inconsistency introduced by the variation in modeling techniques.

2.1 SUMMARY OF PREVIOUS MODELING EFFORTS

The fundamental objective of the past development of reactor models for the ORIGEN2 computer code[7-10] was to allow ORIGEN2 to predict the correct spent fuel compositions without having to resort to the adjustment of cross sections, which had typified earlier ORIGEN computer codes and their associated reactor models. This meant that ORIGEN2 had to be able to use cross sections that resulted from the processing of existing ENDF/B compilations. The following paragraphs are reworded from Ref. 7 to describe the basic steps taken to prepare ORIGEN2 reactor models.

The generation of the information required for these reactor models required the gathering and initial processing of existing raw cross-section data into an 84-neutron-energy-group library that could be used by a modular system of reactor physics codes, such as the AMPX system.[15] Two separate libraries were created: (1) a fuel-depletion library - containing actinides, moderators, poisons, and principal structural materials, whose presence in the reactor would have the greatest effects on the neutron spectrum and depletion characteristics; and (2) a minor-constituent library - containing fission products, minor impurities, and minor structural materials of interest in ORIGEN2 but having a minor effect on the neutron spectrum and depletion calculations. Only the first of these libraries was considered in the subsequent multigroup fuel-depletion calculations.

Following these initial steps, burnup-dependent cross sections that accounted for spatial and energy self-shielding effects were generated for each fuel type and reactor model combination being considered. The libraries resulting from this procedure were collapsed from multi-energy-group to few-energy-group structure (typically from 84-group to 5-group). The five-energy-group cross-section group structure consists of one thermal group and four fast groups spanning the range of 1.0×10^{-5} eV to 10.0 MeV (spanning the thermal-to-fast neutron energy range). These few-group, burnup-dependent libraries were then used in a diffusion-theory depletion code that predicted the composition of the spent fuel and supplied some of the cross sections required by ORIGEN2. The cross sections in the minor constituent, 84-group library (i.e., those having a minor effect on neutron spectrum and depletion) were then collapsed to one-group cross sections using a "typical" neutron spectrum for the fuel type and reactor model as calculated during the generation of the burnup-dependent cross sections. Fission product yields were obtained by flux-weighting energy-dependent yields using this same calculated neutron spectrum. Additional calculations were then performed to obtain new values of the ORIGEN flux parameters - THERM, RES, and FAST - and parameters related to the activation of plenum springs and fuel-assembly end pieces.

The depletion calculations were performed for the PWR and BWR models using the CITATION code.[16] Multigroup fuel-depletion calculations with CITATION modeled the depletion characteristics of an entire batch of reactor fuel at steady state. For the purposes of updating the ORIGEN computer code, spatial details of the depletion are not important; therefore, the basic approach was to use the simplest neutronic and fuel management model possible while still obtaining adequate depletion results.

Early in the planning stages of the ORIGEN update effort,[7] discussions indicated that the neutronic interaction of the fuel assemblies with differing burnups would have an important effect on the results of the depletion calculation. Hence, it was decided that the spatial order of the CITATION calculations would be kept as low as possible and that the simultaneous presence of fuel with different

burnups would be accounted for in the fuel management scheme. This led to the selection of zero-dimensional, three- and four-region CITATION models for the uranium-enriched PWR and BWR standard-burnup designs, respectively.

The relative neutronic homogeneity of the uranium-enriched LWRs allowed the use of spatial models that were effectively zero-dimensional. More specifically, the spatial model used was a one-dimensional, three-region (PWR) or four-region (BWR) slab with reflecting boundary conditions. The zero-dimensionality characteristic was achieved by making the volumes of each of the regions very small (0.001 cm^3), thus forcing the neutron flux to be flat and have the same energy distribution over all regions. In the PWR model, the first, second, and third regions were assigned cross-section sets corresponding to fuel burnups of 5, 16, and 27 GWd/MTIHM, respectively. Similarly, the four regions in the BWR model were assigned cross-section sets corresponding to 3, 19, 17, and 25 GWd/MTIHM. The reload fuel is charged to the region that has been assigned the cross sections corresponding to the lowest burnup and is then irradiated for 293.3 full-power days for the PWR or 265.4 full-power days for the BWR. This somewhat depleted fuel composition is subsequently moved to the region that has been assigned cross sections corresponding to the next higher burnup and is irradiated for another 293.3 or 265.4 full-power days while a fresh batch of fuel is irradiated in the vacated region. This process is continued for ten cycles, at which time the discharge composition of the fuel reaches a nearly constant value.

The models described above require some type of initial loading to begin the calculations. The three regions in the PWR model are initially loaded with fuel having enrichments of 3.1, 2.6, and 2.1 wt % ^{235}U. In the BWR model, one region is initially loaded with fuel having a uranium enrichment of 1.1 wt % ^{235}U; the remaining regions contain fuel having a uranium enrichment of 2.5 wt % ^{235}U. All reload batches have uranium enrichments of 3.2 wt % for the PWR and 2.75 wt % for the BWR. During the first cycle, cross sections are altered from the assignments described previously in that all regions were assigned cross sections corresponding to the lowest burnup for the reactor.

The CITATION depletion calculations explicitly consider all the nuclides listed in the fuel depletion cross-section library. All nuclide densities input to CITATION are pin-cell-averaged quantities for the uranium-cycle LWR models.

The power-level input to CITATION is equivalent to a specific power of 37.5 MW/MTIHM for the PWR and 25.9 MW/MTIHM for the BWR. The specific power of each batch of fuel in the CITATION model varies throughout its life, depending on its enrichment and the enrichment of the other fuel in the model at that time.

The results of the CITATION depletion calculations for the uranium-cycle LWRs consist of the following:

1. one-energy-group, effective cross sections for the nuclides in the fuel depletion nuclide library, as a function of burnup;
2. the discharge composition of various LWR reactor fuels; and
3. five-group neutron spectra as a function of burnup.

The one-energy-group, effective cross sections of the principal actinides are reformatted and incorporated into the ORIGEN2 computer code as a function of burnup.

The five-energy-group neutron spectra are used as a basis for performing pin-cell calculations to provide collapsed, one-energy-group cross sections for the fission products, activation products, and actinides for incorporation into the ORIGEN2 cross-section library.

A similar approach to that stated above was used in preparing the revised LWR reactor models and cross-section libraries that are the subject of this report. The cross-section processing calculations that resulted in the generation of the "master" cross-section library are described in Sect. 3. The fuel-depletion calculations for the five revised and updated LWR standard- and extended-burnup reactor models are described in Sect. 4. The following sections describe the basic assumptions for enriched-uranium core geometries and fuel cycles for PWRs and BWRs operated at standard and extended burnups.

2.2 PWR CORE CHARACTERISTICS

The PWR reactor core is comprised of an array of fuel assemblies

that are identical in mechanical design but different in fuel enrichment, based on the position of each assembly within the reactor core. The reference design[1] employs three enrichments in a three-region core for a standard-burnup of 33,000 MWd/MTIHM. Fuel cycle times of 6 to 18 months are possible, with an average of a 12-month cycle between core reloads/reshuffles. The core is cooled and moderated by light water at a pressure of 155 bar (2250 psia). The coolant used as the moderator contains soluble boron as a neutron poison. The concentration of boron in the coolant is varied (as required) to control relatively slow reactivity changes, including the effects of fuel burnup. Additional boron, in the form of burnable poison rods, is employed in the first core to establish the desired initial reactivity and to smooth the power level across the reactor core.

Two hundred and sixty-four (264) fuel rods are mechanically joined in a square array to form a fuel assembly. The fuel rods are supported in intervals along their length by grid assemblies that maintain the lateral spacing between the rods throughout the design life of the assembly. The grid assembly consists of an "egg-crate" arrangement of interlocked straps. The straps contain spring fingers and dimples for fuel rod support, as well as coolant mixing vanes. The fuel rods consist of slightly enriched uranium dioxide ceramic cylindrical pellets contained in slightly cold-worked Zircaloy-4 tubing that is plugged and seal-welded at the ends to encapsulate the fuel. All fuel rods are pressurized with helium during fabrication to reduce stresses and strains to increase fatigue life.

The center position in the assembly is reserved for the in-core instrumentation, while the remaining 24 nonfuel rod positions in the array are equipped with guide thimbles joined to the grids and the top and bottom nozzles. Depending on the position of the assembly in the core, the guide thimbles are used as core locations for rod cluster control assemblies, neutron source assemblies, regulating or safety control rods, burnable poison rods, or moderating (water) rods. Otherwise, the guide thimbles are fitted with plugging devices to limit bypass flow.

The following dimensions were assumed for the PWR standard- and extended-burnup fuel assembly: outside diameter of the fuel rods, 0.374 in.; diametrical gap, 0.0065 in.; Zircaloy-4 cladding thickness, 0.0225 in.; fuel pellet diameter, 0.3225 in.; pellet density, 95% of theoretical; fuel rod pitch, 0.496 in.; active fuel height, 144 in.; and fuel rod top plenum, 6.3 in. The burnable poison rods consisted of a borosilicate glass annulus encased in stainless steel-304 cladding, with the inside assumed to be dry (no moderation).

2.2.1 Standard-Burnup PWR Fuel Cycles

The first fuel cycle contained three different ^{235}U fuel enrichment regions located in a typical, checkerboard fashion. For the standard-burnup cycle, the initial fuel enrichments were 3.10, 2.60, and 2.10 wt % ^{235}U; the reload enrichment was 3.20 wt % ^{235}U. The initial core loading for the first fuel cycle was made up of fuel assemblies of the lowest two enrichments occupying the central checkerboard region, surrounded by a perimeter of assemblies at an enrichment of 3.1 wt % ^{235}U. After the first cycle, the fuel with the lowest enrichment (2.1 wt % ^{235}U) is discharged and the perimeter assemblies are shuffled to the central checkboard. Fresh fuel at 3.2 wt % ^{235}U is loaded to the perimeter of the reactor core.

After successive core reloading and reshuffling, the core loading pattern reaches equilibrium. Spent fuel (that has completed three cycles) is extracted from positions in the central core, fuel from the perimeter (that has completed one cycle) is relocated toward the central core, and fresh fuel is loaded to the perimeter of the core.

2.2.2 Extended-Burnup PWR Fuel Cycles

The first fuel cycle contained three different ^{235}U fuel enrichment regions located in the same checkerboard pattern as the standard-burnup fuel cycle. For the extended-burnup cycle, the initial fuel enrichments were 3.20, 2.70, and 2.20 wt % ^{235}U; the reload enrichment was 4.20 wt % ^{235}U.

The extended-burnup fuel cycle is divided into three cycles, as in the standard-burnup case. The only difference is that each irradiation (or burnup) cycle is longer (extended), enabling the discharged fuel to attain an extended burnup of 50,000 MWd/MTIHM. After successive core reloading and reshuffling, the core loading pattern reaches equilibrium. Spent fuel (that has completed three cycles) is extracted from positions in the central core, fuel from the perimeter (that has completed one cycle) is relocated toward the central core, and fresh fuel is loaded to the perimeter of the core.

2.3 BWR CORE CHARACTERISTICS

The BWR reactor core is comprised of an array of fuel assemblies (consisting of fuel bundles and surrounding fuel channels) that are identical in mechanical design but different in fuel enrichment. The reference design[2] employs three enrichments in a four-region core for a standard burnup of 27,500 MWd/MTIHM for the initial and equilibrium core loading. Unlike the PWR core, the loading pattern of the BWR core is centered around each cruciform control rod (or control blade), with the core shuffling resulting in a rotation of the fuel around each blade so as to maintain the lower enriched fuel (or fuel at higher burnup) closer to the center of the core. Fuel cycle times of 6 to 18 months are possible, with the cycle between core reloadings/reshufflings averaging 12 months. The core is cooled and moderated by light water at a pressure of 72 bar (1000 psia). Some of the fuel rods within an assembly contain gadolinium (~1500 ppm) as a burnable poison in groups of 10, 12, or 16 (most common) burnable poison rods (BPRs) per assembly.

Sixty-two or sixty-three fuel rods are mechanically joined in a square array to form a fuel bundle. These rods are supported in intervals along their length by grid assemblies that maintain the lateral spacing between the rods throughout the design life of the assembly. The grid assembly consists of an "egg-crate" arrangement of interlocked straps. The straps contain spring fingers and dimples for fuel rod support, as well as coolant mixing vanes. The fuel rods consist of slightly enriched uranium dioxide ceramic cylindrical pellets

contained in slightly cold-worked Zircaloy-2 tubing that is plugged and seal-welded at the ends to encapsulate the fuel. All fuel rods are pressurized with helium during fabrication to reduce stresses and strains to increase fatigue life. The remaining one or two rod positions in a BWR assembly are reserved for water rods.

A Zircaloy channel box surrounds each fuel assembly. A cruciform control rod assembly is located between each grouping of four fuel channels. Each fuel assembly contains fuel rods of three or four different ^{235}U enrichments. Certain fuel rods also contain various concentrations of gadolinium, in the form of Gd_2O_3, as a burnable neutron poison.

The following dimensions were assumed for the BWR standard- and extended-burnup fuel assembly: outside diameter of the fuel rods, 0.403 in.; diametrical gap, 0.0090 in.; Zircaloy-2 cladding thickness, 0.0225 in.; fuel pellet diameter, 0.4160 in.; pellet density, 95% of theoretical; fuel rod pitch, 0.64 in.; active fuel height, 148 in.; and fuel rod top plenum, 10.0 in. The fuel assembly channel box had the following dimensions: outside diameter, 5.21 in.; thickness, 0.120 in.; fuel assembly pitch, 6.0 in.; and water channel thickness, 0.48 in.

2.3.1 Standard-Burnup BWR Fuel Cycles

The first fuel cycle contained four different ^{235}U fuel assembly-averaged enrichment regions within the core. For the standard-burnup cycle, the initial average fuel assembly enrichments were 0.71, 1.8, 2.2, and 2.2 wt % ^{235}U, with one of the 2.2 wt % assemblies containing gadolinium in the form of BPRs. The equilibrium reload average enrichment was 3.0 wt % ^{235}U. It is important to note that the enrichments quoted above are "average" enrichments and that within each different fuel assembly are fuel elements of many different enrichments. For example, the 1.8 wt % enriched assembly contained fuel rods with ^{235}U enrichments ranging from 2.0 to 1.3 wt %, with two of the rods containing Gd_2O_3 burnable poison. The 3.0 wt % fuel assemblies contained fuel rods with enrichments ranging from 3.6 to 1.9 wt %, with five rods containing Gd_2O_3 burnable poison.

After successive core reloading and reshuffling, the core loading pattern reaches equilibrium. Spent fuel (that has completed four cycles) is extracted from positions nearer the central core, fuel nearer the perimeter (that has completed one cycle) is relocated toward the central core, and fresh fuel is loaded to the perimeter of the core.

2.3.2 Extended-Burnup BWR Fuel Cycles

The first fuel cycle contained four different ^{235}U fuel assembly-averaged enrichment regions within the core. For the extended-burnup cycle, the initial average fuel assembly enrichments were 0.71, 1.9, 2.3, and 2.3 wt % ^{235}U, with one of the 2.3 wt % assemblies containing gadolinium in the form of BPRs. The equilibrium reload average enrichment was 3.4 wt % ^{235}U. The 3.4 wt % enriched assembly consisted of fuel rods with enrichments ranging from 4.2 to 2.0 wt %, with six of the rods containing Gd_2O_3 burnable poison.

The extended-burnup fuel cycle in the BWR is divided into four burnup cycles. After successive core reloading and reshuffling, the core loading pattern reaches equilibrium. Spent fuel (that has completed four cycles) is extracted from positions nearer the central core, fuel nearer the perimeter (that has completed one cycle) is relocated toward the central core, and fresh fuel is loaded to the perimeter of the core.

2.4 CALCULATION OF AXIAL COOLANT DENSITIES

In preparing the cross-section libraries, it is important to consider certain material properties, such as the enrichment of the fuel, the geometry of the fuel assembly, and the density of the coolant (moderator). The weighted, broad-energy-group cross-section libraries are prepared by collapsing multidimensional and multiple energy group (multigroup) cross sections using multigroup neutron flux spectra summed over the geometric dimensions under consideration. The calculated neutron flux spectra are dependent on the localized material properties. Of the many material properties, the coolant density in the axial

direction may show the most significant variation, especially in BWRs, where the coolant begins to boil within the active fuel zone.

In PWRs, the axial change in water density is very small and the moderator density was assumed at average conditions of 2200 psia and 570°F. Thus, the saturation water density was 0.73 g/cm^3. An effective fuel temperature for Doppler feedback of 1000 K was used.

In BWRs, there are large variations in the coolant water density in the axial direction, and these density changes must be accounted for in reactor physics calculations. The MTH computer module[17] was used to perform multichannel thermal-hydraulics calculations at 100% of rated power, using a typical axial power shape. The calculations performed by the MTH code included estimations of two-phase, slip flow, and channel power; losses from pressure loop, friction, acceleration, and elevation; and local losses from spacers, inlet, and outlet nozzles. The BWR reactor core was axially subdivided into eight zones over the active fuel length, plus one additional zone for the fuel rod plenum (top), and two zones for the top and bottom reflector regions. Of the total coolant flowing within the reactor, 10% was assumed to bypass the active core zone. The following zone-averaged water densities were calculated, starting from the bottom of the core: 0.74, 0.60, 0.495, 0.417, 0.360, 0.310, 0.265, and 0.235 g/cm^3. The water density of the top reflector was calculated to be 0.285 g/cm^3.

3. DESCRIPTION OF THE GENERATION OF THE MASTER, MULTIGROUP CROSS-SECTION DATA BASE

This section describes the processing of evaluated cross-section data into a multigroup cross-section library that is suitable for use in reactor neutron energy spectrum and fuel-depletion calculations. This library will be applicable to a wide variety of uranium-cycle LWRs.

There are two principal steps involved in processing the evaluated cross-section data into a master library.* Using the AMPX code system,[15] the first step is to generate point cross-section data that are very closely spaced for those few isotopes ($^{235,238}U$ and $^{239-242}Pu$) whose resonance neutron absorptions are important in uranium cycle reactors. For the second step, these data, along with raw data for the remaining isotopes, are neutron-energy-spectrum-weighted <u>within</u> each of the neutron energy groups that are to be included in the master library using an input weighting function that is generally typical of thermal reactors. The result of this two-step processing sequence, which is described in detail in Sects. 3.2 and 3.3, is the AMPX "master" cross-section library. A summary description of the processing sequence is given in Table 3.1.

3.1 SOURCE OF CROSS-SECTION DATA

The cross-section data base used in the multigroup reactor physics calculations described in Sect. 4 contains all cross sections that were

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*The terminology "master" library refers to one of the cross-section library formats produced by the AMPX modular code system. Characteristics of such libraries are described in ref. 15. Master libraries are not structured for direct use in a multigroup particle transport calculation. These libraries include appropriate data to allow subsequent problem-dependent resonance self-shielding to be performed using either the Nordheim method, the Bondarenko method, or an integral transport calculation with many spatial zones. Temperature dependence of data is allowed. Any number of scattering processes can be included to any degree of anisotropic representation. Format of a master library allows a generality paralleling that of the ENDF/B point libraries.

Table 3.1. Generation of an 84-group AMPX master cross-section library

Data sources:

- ENDF/B-IV, General Purpose Files
- ENDF/B-IV, Thermal Scattering Data
- ENDF/B-V, General Purpose Files
- ENDF/B-V, Special Files (e.g., Fission Products)
- LENDL Data Files

Assumed Neutron Spectrum: fission spectrum (fission energy range)
1/E (intermediate energy range)
Maxwellian (thermal energy range)

Codes used	Output
XLACS2, XLACS3	84-group cross sections and transfer arrays weighted over the neutron spectrum for all reaction types in the data files are prepared. Resonance parameters for the resolved range are included.
NJOY87, MODER, UNRESR, WEIGHT, TABU	Bondarenko factors for the unresolved resonance range are calculated for the 84-group AMPX library.
UNITAB	Inserts Bondarenko factors into the 84-group AMPX library.
AJAX	Creates a new master library by combining data sets.
RADE	Performs a consistency check on data in the new AMPX master library.

available from the many data sources in machine-readable format. The evaluated cross-section data were derived from three principal sources: the Evaluated Nuclear Data File, Versions 4 and 5 [(ENDF/B-IV)[18] and (ENDF/B-V)[19]] and the Livermore Evaluated Nuclear Data Library (LENDL).[20] These data were divided into two groups: (1) the actinides, moderators, poisons, and principal structural materials; and (2) the fission products, minor impurities, and minor structural materials.

Evaluated cross sections from ENDF/B-IV, ENDF/B-V, and LENDL were processed into a multigroup structure of 84 neutron groups, with 34 groups allowing upscattering in the thermal energy range using the XLACS code.[21,22] This 84-group library contains the neutron data for the nuclides of neutronic importance in fuel-depletion calculations. Many of these cross sections were copied from a previously created 84-group library used in developing the earlier ORIGEN2 reactor models.[7] The neutron energy structure for the 84-group master library is given in Table 3.2. Table 3.2 also provides the corresponding neutron energy structure for the 12-group library (used in axial assembly calculations) and the 4-group library (used in burnup and depletion calculations).

The first group of nuclides is neutronically important in the reactor, and all of these nuclides were represented explicitly in the depletion calculations, which were performed using a 4-group library. One-group cross sections for nuclides in the second group, (i.e., fission products, minor impurities, and minor structural materials), were obtained by collapsing the 84-group cross sections using a neutron spectrum selected after the depletion calculation had been performed. A list of the nuclides in the first group is given in Table 3.3. Also noted in Table 3.3 is whether a nuclide is resonance or nonresonance. Table 3.4 provides a list of nuclides in the second group and includes a 1/v cross section that is used to spectrally weight the 2200-m/s cross section for the other (~1100) nuclides needed by the ORIGEN2 computer code.

Table 3.2. ORIGEN2 master library 84-group neutron energy structure

ORIGEN2 master library	12-group axial library	4-group depletion library	Upper energy (eV)
1	1	1	1.00000E+7[a]
2			7.78801E+6
3			6.06531E+6
4			4.72367E+6
5			3.67879E+6
6	2		2.86505E+6
7			2.23130E+6
8			1.73774E+6
9			1.35335E+6
10			1.05399E+6
11	3	2	8.20850E+5
12			6.39279E+5
13			4.97871E+5
14			3.87742E+5
15			3.01974E+5
16			2.35177E+5
17			1.83156E+5
18	4		1.42642E+5
19			1.11090E+5
20			8.65169E+4
21			6.73794E+4
22			4.08677E+4
23			2.47875E+4
24			1.50344E+4
25			9.11882E+3
26	5	3	5.53084E+3
27			3.35463E+3
28			2.03468E+3
29			1.23410E+3
30			7.48518E+2
31			4.53999E+2
32			2.75364E+2
33			1.67017E+2
34			1.30073E+2
35	6		1.01301E+2
36			7.88932E+1
37			6.14421E+1
38			4.78512E+1
39			3.72665E+1
40			2.90232E+1
41			2.26033E+1
42	7		1.76035E+1
43			1.37096E+1

Table 3.2 (continued)

ORIGEN2 master library	12-group axial library	4-group depletion library	Upper energy (eV)
44			1.06770E+1
45			8.31529E+0
46			6.47595E+0
47			5.04348E+0
48			3.92786E+0
49	8		3.05902E+0
50			2.38237E+0
51			1.85539E+0
52			1.43950E+0
53			1.12535E+0
54			8.33691E-1
55			7.84930E-1
56	9	4[b]	6.32500E-1
57			5.11187E-1
58			4.14983E-1
59	10		3.38908E-1
60			2.78866E-1
61			2.31511E-1
62			1.94124E-1
63			1.64513E-1
64			1.40911E-1
65			1.21896E-1
66			1.06323E-1
67	11		9.22968E-2
68			8.01604E-2
69			6.97166E-2
70			6.07832E-2
71			5.31932E-2
72			4.61091E-2
73			3.95312E-2
74			3.34592E-2
75			2.78932E-2
76	12[b]		2.28332E-2
77			1.82792E-2
78			1.42312E-2
79			1.06892E-2
80			7.65324E-3
81			5.12325E-3
82			3.09925E-3
83			1.58125E-3
84[b]			5.69249E-4

[a]Read 1.00000×10^{7} eV.
[b]Bottom energy for each group is 1.00000×10^{-5} eV.

Table 3.3. ORIGEN2 nuclides included in burnup and depletion analyses

Nuclide	Nuclide	Nuclide	Nuclide
Major structural components, neutron moderators, and poisons			
^{1}H	^{7}Li	^{10}B	^{16}O
Cr[a,b]	Mn[a]	Fe[a,b]	^{59}Co[b]
Ni[a]	Zr[b,c]	^{103}Ru	^{103}Rh[b]
^{105}Ru	^{105}Rh	^{109}Ag	^{131}Xe
^{133}Xe	^{133}Cs	^{134}Cs	^{135}I
^{135}Xe	^{143}Pr	^{143}Nd[b]	^{145}Nd[b]
^{147}Nd	^{147}Pm[b]	^{148}Pm	^{148m}Pm
^{149}Pm	^{149}Sm[b]	^{150}Sm[b]	^{151}Pm
^{151}Sm[b]	^{152}Sm[b]	^{153}Eu[b]	^{154}Eu[b]
^{154}Gd[b]	^{155}Eu	^{155}Gd[b]	^{156}Gd[b]
^{157}Gd[b]	^{158}Gd[b]	^{159}Gd	^{160}Gd
Actinides[d]			
^{232}Th	^{233}Pa	^{233}U	^{234}U
^{235}U	^{236}U	^{238}U	^{237}Np
^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu
^{242}Pu	^{243}Pu	^{241}Am	^{242}Am
^{243}Am	^{242}Cm	^{243}Cm	^{244}Cm
^{245}Cm	^{246}Cm	^{247}Cm	^{248}Cm

[a]Elemental composition in stainless steel-304.
[b]Resonance nuclide.
[c]Elemental composition in cladding.
[d]All actinides are resonance nuclides.

Table 3.4. ORIGEN2 nuclides not included in depletion analyses

Nuclide	Nuclide	Nuclide	Nuclide	Nuclide
^{72}Zn	^{72}Ge	^{73}Ge	^{74}Ge	^{75}As
^{76}Ge	^{77}Se	^{78}Se	^{80}Se	^{81}Br
^{82}Kr	^{82}Se	^{83}Kr	^{84}Kr	^{85}Rb
^{85}Kr	^{86}Kr	^{87}Rb	^{88}Sr	^{89}Sr
^{89}Y	^{90}Y	^{90}Sr	^{91}Y	^{95}Nb
^{95}Mo	^{96}Mo	^{97}Mo	^{98}Mo	^{99}Mo
^{99}Tc	^{100}Ru	^{100}Mo	^{101}Ru	^{102}Ru
^{104}Pd	^{104}Ru	^{105}Pd	^{106}Pd	^{106}Ru
^{107}Pd	^{108}Pd	^{110}Cd	^{110}Pd	^{111}Ag
^{111}Cd	^{112}Cd	^{113}Cd	^{114}Cd	^{115}In
^{115m}Cd	^{116}Sn	^{116}Cd	^{117}Sn	^{118}Sn
^{119}Sn	^{120}Sn	^{121}Sb	^{122}Te	^{122}Sn
^{123}Sn	^{123}Sb	^{124}Sb	^{124}Te	^{124}Sn
^{125}Sn	^{125}Sb	^{125}Te	^{126}Sn	^{126}Sb
^{126}Te	^{127}I	^{127m}Te	^{128}Xe	^{128}Te
^{129}I	^{129m}Te	^{130}I	^{130}Xe	^{130}Te
^{131}I	^{132}Xe	^{132}Te	^{134}Xe	^{134}Ba
^{135}Cs	^{136}Cs	^{136}Ba	^{136}Xe	^{137}Ba
^{137}Cs	^{138}Ba	^{139}La	^{140}La	^{140}Ce
^{140}Ba	^{141}Ce	^{141}Pr	^{142}Pr	^{142}Nd
^{142}Ce	^{143}Ce	^{144}Nd	^{144}Ce	^{146}Nd
^{147}Sm	^{148}Sm	^{148}Nd	^{150}Nd	^{151}Eu
^{153}Sm	^{154}Sm	^{156}Eu	^{157}Eu	^{159}Tb
^{160}Tb	^{160}Dy	^{161}Dy	^{162}Dy	1/v

3.2 RESONANCE NUCLIDE PROCESSING

The nuclides included in the first (neutronically important) group can be separated into two classes: resonance and nonresonance. For isotopes containing unresolved resonance data, the following sequence of computer modules was used to generate Bondarenko factors: NJOY87,[23] MODER,[23] UNRESR,[23] WEIGHT,[23] and TABU.[22]

For all nuclides for which ENDF or LENDL evaluations were available, the XLACS2[21] (and later the XLACS3,[22] with improved integration schemes) module was used to cast the data into an 84-group structure. For the nuclides that contained resolved resonance parameters, the NITAWL module[22] was used for the calculation of the neutron resonance self-shielding factors in the fuel rods using the Nordheim method. The resonance parameters also contain the necessary information to account for Doppler broadening at a specified temperature (assumed to be 1000 K). For the nuclides with unresolved resonance data, the calculated Bondarenko factors were included in the AMPX master library using the UNITAB module.[21] The Bondarenko factors are subsequently used by BONAMI[24] for self-shielding calculations within the fuel rods.

Specific thermal scatter data at different temperatures were added to cross-section libraries for certain nuclides, such as H in H_2O or U in UO_2. These thermal scatter data were previously generated by Gulf General Atomics, using the GASKET module[25] in combination with experimental data.

3.3 PREPARATION OF THE MASTER LIBRARY

The cross-section data from all sources were unified into an AMPX master library using the AJAX module.[21] Verification tests for consistency were performed using the RADE computer module.[21]

4. MULTIGROUP DEPLETION CALCULATIONS

The 84-energy-group AMPX master cross-section library described in Sect. 3 of this report contains cross-section information of a general nature for solving uranium-cycle thermal reactor problems. This section describes the processing of this general, multigroup library into problem-dependent, burnup-dependent, multigroup libraries and then into problem-dependent, one-group libraries used in ORIGEN2.

The three principal computer codes used in processing the cross sections are NITAWL, XSDRNPM,[24] and GPRCYCLE.[26] In fact, GPRCYCLE is a system driver to access NITAWL, XSDRNPM, and many other codes of this general reactor cycle calculational system used in development of the new and/or revised ORIGEN2 reactor models. A general description of the three principal computer codes is given in Sect. 4.1.

Four major steps are involved in processing the master cross-section library into ORIGEN2 libraries. The first of these, described in Sect. 4.2, is to perform "pin-cell" neutron energy spectrum calculations at different burnups using NITAWL and XSDRNPM, which are used to account for self-shielding effects of the nuclides in the reactor core. The second step, described in Sect. 4.3, is to perform fuel-depletion calculations with GPRCYCLE using few-group, self-shielded cross sections from XSDRNPM. The principal result of the GPRCYCLE calculation is one-group, burnup-dependent cross sections for the most neutronically important actinides. The third step, described in Sect. 4.4, is to use sophisticated reactor physics calculations for each different reactor/fuel combination to determine the multigroup neutron energy spectrum that is "typical" for each fuel type and then to use this spectrum to weight (i.e., collapse) the multigroup cross sections to prepare master, one-group cross-section libraries and calculate new values for ORIGEN flux parameters THERM, RES, and FAST. The final step, described in Sect. 4.5, involves the preparation of ORIGEN2-readable libraries from the master, one-group libraries. Inclusive in this step is the insertion of the "other" 1100 nuclide cross sections that are not available in multigroup format by the use of the ORIGEN flux parameters

THERM, RES, and FAST as weighting factors for the thermal cross sections, resonance integrals, and fission-spectrum-averaged threshold cross sections, respectively.

The buildup and depletion of nuclides during irradiation are calculated by ORIGEN2 using zero-dimensional (i.e., point) geometry and quasi-one-group cross sections developed for each fuel type/reactor model being considered. This means that ORIGEN2 cannot account for spatial or resonance self-shielding effects or changes in neutron spectra other than those embodied initially.

A far less visible, but extremely important, feature of ORIGEN2 is the variability of the actinide cross sections. Early in the development of ORIGEN2 in the late 1970s, it was determined that the use of a constant, average cross section from a sophisticated reactor physics code for the principal actinide nuclides such as $^{235,238}U$ and $^{239-242}Pu$ would not give correct depletion results. The primary explanations given for this situation were: (1) the cross sections of some of these isotopes did not vary linearly with burnup, and (2) the cross sections near the end of irradiation are more important in determining the discharge fuel composition than the cross sections near the beginning of irradiation. The solution to this problem was to obtain cross sections as a function of burnup for the principal actinides and to use them in ORIGEN2. These cross sections are incorporated in DATA statements in separate ORIGEN2 subroutines, described in Appendix C. At the beginning of each irradiation step, the burnup is estimated, the variable cross sections are calculated by interpolation, and the updated values are substituted in the matrix of differential equations being solved by ORIGEN2. If a fission cross section of a nuclide having direct fission product yields is altered (e.g., ^{235}U), then the array containing the product of the fission product yields and the fission cross section is also adjusted to reflect this change. Each reactor/fuel combination can have a different set of variable cross sections, a different set of actinides with variable cross sections, or both.

4.1 CROSS-SECTION PROCESSING CODES

The development of new and/or revised cross-section libraries and reactor models for the ORIGEN2 computer code involved a massive data processing effort. Dozens of computer codes and many data bases were used. Three of the more important codes utilized in this effort are described in Sects. 4.1.1 through 4.1.3. The third code, GPRCYCLE, is actually a system driver that linked together approximately 24 individual modules from 3 separate code packages into a new computational system for general reactor-cycle analyses.

4.1.1 NITAWL

The NITAWL computer code is used to account for resonance self-shielding effects in nuclides with resonance parameters. That is, given information about the fuel region of a pin cell, such as the Doppler temperature, moderator concentration, pin-cell dimensions, and resonance nuclide concentrations, NITAWL accounts for (1) Doppler broadening of the resonances, and (2) the fact that the effect of large resonances is diminished because relatively few neutrons are at the resonance energy. The latter effect results because the resonance itself depletes the supply of neutrons having the same energy as that of the resonance (i.e., energy self-shielding). The Nordheim integral treatment is used to account for the resonance self-shielding effects. The output of NITAWL is a multigroup cross section "working library"* in which the resonance parameters have been incorporated into the group-averaged cross sections.

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*The terminology "working library" refers to one of the cross-section library formats produced by the AMPX modular code system. Characteristics of such libraries are described in ref. 15. Working libraries are intended to be used as input to transport codes such as XSDRNPM.

4.1.2 XSDRNPM

The XSDRNPM computer code is effectively used to account for spatial and energy self-shielding effects within a fuel element or a fuel assembly. To do this, the code performs a one-dimensional, static, S_8P_3, discrete-ordinates flux calculation and then uses the flux to weight the input cross sections (from the working library) in space, energy, or both. The result is a cross-section library which accounts for the fact that the neutron energy spectrum (and thus the effective nuclide cross sections) varies significantly within the fuel material and within the moderator. This library is designated as a "weighted library." The principal input data required are a physical description of the fuel element or assembly (i.e., the dimensions of each zone), the concentration of each nuclide within each zone, and a working library from NITAWL that includes data for the specified nuclides.

4.1.3 GPRCYCLE

In addition to the NITAWL and XSDRNPM computer codes used in this study, an overall computational system called GPRCYCLE (General Purpose Reactor CYCLE) was used to access data libraries and execute computer codes, including many codes normally considered part of the well-known AMPX,[15,21-22] SCALE,[24] and VENTURE[27] packages. Using GPRCYCLE, a series of modules was executed in the proper sequence to generate weighted, multigroup cross-section libraries as a function of ^{235}U enrichment, fuel burnup, location in the core, and concentration of boron (as soluble boron in the moderator for PWRs, and as burnable poison rods in BWRs). Cross sections in the libraries are also functions of the composition of the fuel (actinides and fission products), as well as the composition of the moderator (water), burnable poison rods, control rods, and important structural materials.

A general overview of the functional sequence of reactor physics calculations performed during this study is provided in the next few paragraphs, followed by more-detailed descriptions of the four principal

steps involved in cross-section processing used to generate ORIGEN2-compatible libraries in Sects. 4.2 through 4.5.

All the calculations were performed on ORNL's IBM-3033 mainframe computer. Simplifications in the problem complexity and overall calculational methodology were made in order to reduce computational expenses. One simplification was the assumption that the reactor core was symmetrical, thus allowing neutronics calculations to be performed on part of the core, with appropriate use of boundary conditions.

Another simplification in calculational complexity was the substitution of less CPU-intensive (and, therefore, less costly) computational modules in place of more-detailed techniques. Since the ORIGEN2 code uses one-group, zero-dimensional cross-section data to simulate the neutronic behavior of a commercial reactor and calculate the buildup and decay composition of approximately 1300 radionuclides, it was decided not to perform a pin-by-pin depletion analysis. Indeed, for a PWR, a pin-by-pin fuel depletion for an entire core with ten axial zones would require over 550,000 separate calculations involving more than 200 nuclides per calculation. Therefore, zones were established to combine fuel rods of similar initial ^{235}U enrichment, burnable poison loading, batch number, and burnup. The number density of the nuclides in each zone was assumed to be average for the zone and termed "depletion-zone."

Cell-weighted, collapsed neutron cross sections (84-group to 4-group for the depletion calculations, 84-group to 12-group for the axial discrete-ordinates transport calculations) for the fuel rods, burnable poison rods, waterholes, and other regions were generated for eight axial depletion zones. These multigroup, cell-weighted cross sections were calculated for each of the "cross-section zones" and stored in an "AMPX-working-library" format. The AMPX-working-library cross sections were then converted to a CCCC-ISOTXS[28] format, isotope-dependent interface file using the CONTAC module[21] and to a CCCC-GRUPXS[28] format, group-dependent cross-section interface file using the CROSPROS module.[27]

The CCCC-GRUPXS interface files were subsequently used to perform multigroup and multidimensional neutronics diffusion calculations using

the DVENTR[27] and VENTNEUT[27] modules of the VENTURE code package. The PWR calculations assumed a two-dimensional, x-y geometry, and the BWR calculations assumed three-dimensional, x-y-z geometry. For the PWR cases, a soluble boron search (at $K_{eff} = 1.0$) was made at each depletion time, assuming that all regulating control rods were fully withdrawn.

Using the local* neutron spectra and local power densities, depletion calculations were performed for each of the depletion zones using the BURNER module.[27] This module generates a CCCC-ZNATDN[28] interface file containing zone-dependent number densities for each nuclide in each depletion zone. The BURNER module uses a decay and depletion-chain library containing approximately 90 chains and 210 nuclides. This library was transposed from the EPRI-CINDER[29,30] library into a CCCC-EXPOSE[28] interface file. Added to this library were chains for special materials, such as burnable poisons. Separate effective fission product yields for ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu were used. For each of the nuclides, the energy releases for all of the major neutron reactions were added using the JPR6 module[31] and DCRSPR[28] interface file. For each major fissile actinide, the nuclide-dependent recoverable energy per fission was used. These values are listed in Table 4.1.

Average nuclide number densities from the ZNATDN interface files for each depletion zone were used to recalculate burnup-dependent, cell-weighted neutron cross sections for the fuel rod cells and other material zones.

The nuclide number densities for the ZNATDN interface files were used as input to the ZDRN/MWA8 module.[32] This module was then used to perform the spectrum weighted collapse of the multigroup cross sections to a one-group, spectrum-averaged cross section. The nuclide number densities for each depletion zone, resulting from multidimensional, multigroup neutronics calculations, together with cell-weighted, 12-group (collapsed) cross sections, were used in axial discrete-ordinates transport calculations. For the PWR and BWR models, 11 axial zones were

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*The terminology "local" refers to the specific value associated with each depletion zone.

Table 4.1. Nuclide-dependent recoverable energy per fission

Nuclide	Recoverable energy per fission (Mev)
^{234}U	190.30
^{235}U	194.02
^{236}U	192.80
^{238}U	198.12
^{237}Np	195.10
^{238}Pu	197.80
^{239}Pu	200.05
^{240}Pu	199.80
^{241}Pu	202.22
^{242}Pu	200.62
^{243}Pu	200.00
^{241}Am	202.30
^{242}Am	200.00
^{243}Am	202.10
^{242}Cm	200.00
^{243}Cm	200.00
^{244}Cm	200.00
^{245}Cm	200.00
^{246}Cm	200.00
^{247}Cm	200.00

used (8 zones in the active fuel zone, and 1 each in the top plenum zone, top reflector region, and bottom reflector region). This calculation yielded 12-group neutron cross sections for over 210 nuclides for each reactor/fuel combination. These cross sections were, in turn, collapsed to a single group for input to the ORIGEN2 libraries.

One-group cross sections for the remaining (approximately 1100) nuclides considered in ORIGEN2 were obtained by weighting the thermal cross sections, resonance integrals, and fission-spectrum-averaged threshold cross sections using the ORIGEN flux parameters THERM, RES, and FAST with neutron cross sections from an ORIGEN2 auxiliary data base[7] containing data for the thermal (2200-m/s) capture, fission and total cross sections, resonance integrals, and threshold energies and reaction values. Fission product yields for the principal actinides and energy release values for the various neutron reaction types were incorporated and merged with the one-group cross sections to form ORIGEN2-compatible cross-section/fission-product-yield libraries for each new or revised reactor model.

4.2 PIN-CELL CALCULATIONS

Calculations were performed to cell-weight each cross-section zone based on the different characteristics of each zone. The nuclide number densities used as input were the depletion-zone averaged values, based on the fuel burnup, initial ^{235}U enrichment, cycle number (fresh fuel or reshuffled), burnable poison composition, and axial location of each cell within the core.

The cell calculations consisted of three distinct steps as depicted in Table 4.2. First, for each cell type, the BONAMI module and the appropriate Bondarenko factor data on the master library were used to compute multigroup neutron cross sections in the unresolved resonance region.

Second, for each cell type, the NITAWL module was used to modify the multigroup cross sections to account for resolved resonances, using the Nordheim integral technique, for nuclides with resonance parameters having number densities greater than 1.0×10^{-10}. The cross sections

Table 4.2. 84-group pin-cell analyses

Data sources:

- Fuel rod lattice cell geometries
- Nuclide composition from burnup analyses
- Moderator fraction from thermal-hydraulic analyses
- 84-group master cross-sections for 68 nuclides

Codes used	Output
BONAMI	Unresolved resonance processing.
NITAWL	Resolved resonance processing and generation of the AMPX working library.
XSDRNPM	84-group pin-cell analyses yielding 4-group collapsed cell-weighted cross sections and spectral factors THERM, RES, and FAST.

for resonance nuclides below this threshold were corrected via an infinite dilution assumption. The effective fuel pellet temperature to determine the effect of Doppler broadening of the resonances was assumed to be 1000 K. The NITAWL module generates cross-section libraries in an "AMPX-working-library" format and includes all the nonresonance nuclides, together with the resonance nuclides (and resonance adjustment parameters), in the group structure of the input cross-section library (i.e., the 84-group structure).

Each "cross-section-zone" data file, containing the weighted cross sections for each zone, was combined into a "super" library using the AWL module.[22] Each nuclide was indexed with a special identification number relating the region or zone number (radial or axial), batch number (e.g., first cycle), and enrichment (identifying different average initial enrichment within each assembly).

The third step is the processing of the 84-group cross sections into a weighted, collapsed library using XSDRNPM. The XSDRNPM code performs one-dimensional, discrete-ordinates neutron transport calculations to account for the spatial self-shielding effect of the fuel rods and surrounding moderator, using an S_8P_3 cylindrical geometry for each fuel cell. The output of XSDRNPM includes neutron spectra and reaction rates, which in turn were used to collapse the cross sections from the 84-group structure to a 4-group, cell-weighted structure with energy boundaries of 1.0E+7, 8.208E+5, 5.53E+3, 6.325E-1, and 1.0E-5 eV.

Another "super" library was created using the 4-group cell-weighted cross sections for each fuel cell by the AWL module. The 4-group libraries were generated at several depletion steps and stored in "AMPX-working library" format. Using the CONTAC module,[21] nuclide-dependent CCCC-ISOTXS interface files were created from this working library. Group-dependent CCCC-GRUPXS interface files were generated using the CROSPROS module, with DCRSPR and JPR6 as input. The GRUPXS interface libraries were later employed in multidimensional, few-group diffusion reactor physics calculations (DVENTR and VENTNEUT modules) and zone-dependent, depletion calculations (BURNER module).

The XSDRNPM-generated neutron spectra were also used to collapse

the 84-group cross sections to a 12-group structure for use in the ZDRN/MWA8 module (axial zone calculations).

Based on the number and location of fuel assemblies containing BPRs, three different types of assemblies are common in PWRs: 10 rods/assembly, 12 rods/assembly, and 16 rods/assembly. Only assemblies containing 16 BPRs were considered since this type constitutes the majority of all BPR assemblies. For each BPR cell, the appropriate cell-weighted, homogenized fuel and moderator mixture was assumed in order to maintain the assembly volumetric ratios of the fuel and BPR cells. A one-dimensional, discrete-ordinates calculation was performed (with XSDRNPM) in cylindrical geometry, followed by a spatial weighting and collapse of the cross sections and, finally, the merging of these cross sections with the fuel cell-weighted sets into the "super" library. The BPRs are contained in the annulus of a dry rod clad in SS 304. The burnable poison is borosilicate glass (B_2O_3), with an approximate ^{10}B loading of 0.00042 lb/ft (linear). The inner tube has an OD of 0.1685 in. and a wall thickness of 0.006 in. The outer cladding has an OD of 0.381 in. and a wall thickness of 0.0185 in.

4.3 FUEL DEPLETION CALCULATIONS

The multidimensional, multigroup reactor (burnup-depletion) calculations were performed with the DVENTR and VENTNEUT neutronics modules using two-dimensional (x-y) geometry for the PWR and three-dimensional (x-y-z) geometry for the BWR. Table 4.3 describes the basic computational sequence used to perform the burnup-depletion analyses.

The calculations were performed at rated full power, with normalization of the neutron flux in the reactor based on maintaining full power by using the individual energy releases of each nuclide. Various interface files were generated by the DVENTR module and utilized by the VENTNEUT neutronics module. DVENTR prepares the CCCC-ZNATDN (nuclide number density), CCCC-GEOM (geometry), CCCC-SEARCH (soluble boron search), and CCCC-NDXSRF (indexing ZNATDN number densities to depletion-zone and cross-section-zone) interface files.

Table 4.3. Burnup-depletion analyses

Data sources:

- PWR and BWR reactor models, including radial depletion zones and 11 axial zones
- Power history for each reactor burnup model
- 4-group cell-weighted cross sections for each depletion zone for 68 nuclides
- ORIGEN2 auxiliary data base for 1100 nuclides
- EPRI-CINDER library of fission product yields and energy releases

Codes used	Output
CONTAC, CROSPROS	Performs format conversion for 4-group depletion zone-weighted cross sections.
DVENTR, VENTNEUT	Diffusion calculation for $K_{eff} = 1.0$ for each time step.
BURNER	4-group burnup calculation for each time step, yielding new number densities to update cell-weighted cross sections.
ZDRN/MWA8	One-dimensional, 11-axial-zone transport analysis, 12-group using final number densities. Yields one-group cross sections weighted over entire reactor model; THERM, RES, and FAST weighting of ORIGEN2 auxiliary data base to produce ORIGEN2 cross-section libraries for each reactor model.

To reduce the computational requirements, the steady-state neutronics calculations were performed using a quarter-core geometry, with two reflective boundaries and two zero boundaries. The depletion calculations were made for each depletion zone, at each depletion time step. The neutronics K_{eff} calculations and the neutronics criticality calculations were carried out using the VENTNEUT and BURNER modules, respectively.

For the PWR reactor calculations, the quarter-core geometry consisted of 41 full assemblies, 14 half-assemblies, and one quarter-assembly (center of core). Each assembly had a 17 x 17 fuel rod matrix. Since the axial variation in the moderator is negligible, the reactor calculations were performed in two-dimensional (x-y) geometry. The fuel region was surrounded by an 8-in.-thick reflector of water at 570°F and 2200 psia and an appropriate soluble boron concentration. A criticality search, using $K_{eff} = 1.0$, was performed with all regulating control rods fully withdrawn. The fuel assemblies of different batches (cycles) were positioned in a checkerboard fashion, with fresh fuel located at the edge of the core and once- and twice-burned fuel located within the checkerboard. This core management scheme was used for both standard- and extended-burnup models.

The PWR depletion calculations, using the BURNER module and the ZNATDN, RZFLUX, GRUPXS, NDXSRF, and EXPOSE interface files, were performed at different depletion time steps. The BURNER module uses the matrix exponential method and creates new "depleted nuclide densities," which were used to update the ZNATDN interface files.

The updated ZNATDN files (with burnup dependency) were then used by VENTNEUT to recalculate the burnup-dependent, cell-weighted neutron cross sections. At the end of each cycle, the fuel assemblies were reshuffled, spent assemblies were discharged, and new fuel assemblies were loaded.

The nuclide number densities at various depletion times were subsequently used in ZDRN/MWA8 to calculate one-group, reactor- or zone-weighted cross sections. The number of depletion zones used per quarter core totaled 33.

For the BWR reactor calculations, the quarter core consisted of 196 full fuel assemblies. Each fuel assembly had an 8 x 8 fuel rod matrix, containing fuel rods of different ^{235}U enrichments, with several rods also containing Gd_2O_3 as a burnable poison. Since the axial variation of the water density in the moderator is large for BWRs, the multidimensional, multigroup neutronics calculations were performed in three-dimensional (x-y-z) geometry. The reactor fuel region was surrounded by an 8-in.-thick reflector and a black absorber region. The BWR quarter-core model used a 12 x 12 mesh in the x-y direction to account for the Zircaloy fuel channel and waterholes, and an axial mesh of 60 intervals over the active fuel region and top plenum.

Functionally, the BWR is loaded in 49 superzones per quarter core. Each superzone consists of four fuel assemblies (one from each cycle) and a cruciform control assembly. At the end of each cycle, one assembly is discharged, the remaining three are rotated around the superzone, and a fresh assembly is charged to the remaining position.

Due to symmetry in the superzones, 12 different depletion zones in the x-y plane and three zones in the axial direction were used. Twelve neutron cross-section-zones were created by averaging nuclide number densities over several depletion zones.

The neutronics diffusion calculations generated zone-dependent neutron fluxes, which were normalized to the rated full power and used the equilibrium, space-dependent xenon option.

Using BURNER, depletion calculations at three depletion time steps per cycle were performed. The burnup-dependent ZNATDN interface data were used to recalculate the burnup-dependent, cell-weighted cross-section files, which in turn were used in K_{eff} calculations in VENTNEUT. The fuel assemblies were reshuffled or discharged after each cycle, and fresh fuel was loaded to the core. The nuclide number densities from the ZNATDN interface files were used to calculate one-group cross sections using ZDRN/MWA8. Twelve depletion- and cross-section-zones (four in the x-y plane times three in the axial direction) were considered over the quarter core.

In the functional operation of the BWR core, approximately one-quarter of the cruciform control rods are stipulated for fine regulating

control, while the remaining rods are fully withdrawn. The BWR core is thus considered to have 12 superzones per quarter core, with each superzone having 16 fuel assemblies and one partially inserted, regulating, cruciform control-assembly.

4.4 GENERATION OF ONE-GROUP CROSS SECTIONS

The multigroup cross sections must be collapsed to a one-group cross section for use in the ORIGEN2 computer code. The burnup-dependent nuclide number densities for the different depletion zones, together with the cell-weighted, 12-group cross sections were used in an axial, discrete-ordinates ZDRN/MWA8 calculation. The result of this calculation is a one-group cross section for the approximately 200 nuclides important to core neutronics. A single master, collapsed cross-section library is created for each different reactor model.

4.5 GENERATION OF ORIGEN2 CROSS-SECTION LIBRARIES

The final step in the process is to combine the master cross-section library with one-group cross sections for the other 1100 nuclides used in ORIGEN2 and the spectrum-weighted fission product yields for the fissile isotopes, using the same techniques used to create earlier ORIGEN2 reactor models.[7] The one-group cross sections were calculated using the ORIGEN flux parameters THERM, RES, and FAST with neutron cross sections from an ORIGEN2 auxiliary data base,[7] which contains data for the thermal (2200-m/s) capture, fission, and total cross sections, resonance integrals, and threshold energies and reaction values. Fission product yields for the principal actinides were merged with the one-group cross sections to form ORIGEN2-compatible cross-section/fission-product-yield libraries for each new or revised reactor model.

5. DESCRIPTION OF REACTOR MODELS

The results of the reactor physics calculations described in Sect. 4 were specifically performed to provide the neutronic and cross-section data required by the ORIGEN2 computer code. However, the full incorporation of the updated uranium-cycle LWR models into ORIGEN2 requires the results of work not directly involved with the reactor physics calculations. The additional work can be divided into five major categories: (1) modification of the ORIGEN2 code to include the new variable cross-section subroutines that coincide with the revised cross-section libraries, (2) specification of the initial structural material and fuel compositions, (3) comparison of depletion calculations between the updated LWR cross-section libraries and the previously developed libraries, (4) a summary description of the revised reference LWR models, and (5) a brief comparison of fuel-depletion results (experimental data found in the literature) with ORIGEN2 depletion calculations using the updated LWR models.

The revised, one-group cross sections for the five updated LWR models are presented in Appendix A. The burnup-dependent variable cross sections for selected actinides are presented in Appendix B. It is important to note that the previous LWR models listed 20 cross sections as burnup-dependent, while the updated models only consider seven of these 20 as having significant variation with burnup.

Modifications of the ORIGEN2 code to incorporate the new information were fairly straightforward. First, three (dummy) subroutines named XSEC38 through XSEC40 were replaced by five new subroutines called XSEC38 through XSEC42, corresponding to the variable actinide (burnup-dependent) cross-section data for the five updated LWR models. In addition, subroutine MAIN2 was modified to allow access to these subroutines during program execution. Information about using the updated LWR models and ORIGEN2 Version 2.1 is found in Appendix C.

The specification of the input compositions to ORIGEN2 is repeated here from an earlier report[7] to document fully the input information required to use the updated LWR models. Section 5.1 presents (1) the

mass and elemental composition, including minor constituents, of the structural metals in a fuel assembly; (2) the trace element composition of oxide fuel pellets; and (3) the heavy-metal isotopic composition of the fuel.

As a test of the updated LWR models, results from comparisons between these models and the previous LWR models are presented in Section 5.2. The ORIGEN2 input files used with the updated LWR models are presented in Appendix D. A summary of the results of ORIGEN2 fuel-depletion calculations for the updated uranium-cycle LWRs is given in Section 5.3.

Finally, a brief comparison of ORIGEN2-calculated fuel-depletion vs experimental results (from literature) for a spent fuel sample from an assembly discharged from the H. B. Robinson nuclear plant[33] is presented in Section 5.4. Additional comparisons of ORIGEN2 vs experimental results will be the subject of a later report. ORNL is currently involved in a program [jointly with the Material Characterization Center (MCC) at Pacific Northwest Laboratories (PNL)] to analyze "well-characterized" spent fuel samples using spark source mass spectroscopy (SSMS) and thermal emission isotope dilution mass spectroscopy (TEIDMS) analysis techniques. The results of these analyses will provide a measure of the accuracy (or validity) of ORIGEN2 fuel-depletion predictions.

5.1 INPUT COMPOSITIONS AND MASSES[7]

Three different composition-related aspects should be considered when specifying the LWR fuel-input composition:

1. the mass and composition of the fuel-assembly structural materials associated with a given amount of fuel,
2. the trace-element concentrations in the oxide fuel pellets, and
3. the heavy-metal isotopic composition of the fuel.

5.1.1 Fuel-Assembly Structural Material Masses and Compositions

5.1.1.1 Composition

The seven structural materials used in the revised ORIGEN2 LWR models are Zircaloy-2, Zircaloy-4, Inconel 718, Inconel X-750, SS 302, SS 304, and Nicrobraze 50. The assumed elemental composition of each of these materials is given in Table 5.1.

The cobalt content of these structural materials is of particular interest because the naturally occurring ^{59}Co produces ^{60}Co, a high-energy gamma ray emitter that inhibits the detection of residual amounts of actinides in the structural materials after fuel dissolution. The cobalt is a contaminant in metals containing nickel and results from the use of nickel that has been previously alloyed with cobalt. By judiciously selecting the heats from which the metals containing nickel are taken, it appears to be possible to limit the cobalt contamination to ~0.08 wt % for stainless steels.

5.1.1.2 Structural Material Mass Distribution

The PWR and BWR fuel-assembly structural material distributions recommended for use in reactor depletion calculations are given in Table 5.2. The various materials have been grouped in three categories (i.e., fuel zone, fuel-gas plenum zone, and end fitting zone) according to their proximity to the active fuel region. The elemental compositions of all materials listed in Table 5.2 are given in Table 5.1. It should be noted that the BWR fuel channel, which encloses the square array of fuel elements to prevent crossflow between adjacent assemblies, is assumed to be discarded after only a single cycle (even though it can be used more than once.)

5.1.1.3 Fuel-Assembly Dimensions

The physical parameters of spent PWR and BWR fuel assemblies are given in Table 5.3, even though they are not directly related to ORIGEN2

Table 5.1. Assumed elemental compositions of LWR fuel-assembly structural materials

Element	Atomic number	Structural material composition (g/metric ton metal)						
		Zircaloy-2	Zircaloy-4	Inconel-718	Inconel X-750	Stainless steel 302	Stainless steel 304	Nicrobraze 50
H	1	13	13	0	0	0	0	0
B	5	0.33	0.33	0	0	0	0	50
C	6	120	120	400	399	1,500	800	100
N	7	80	80	1,300	1,300	1,300	1,300	66
O	8	950	950	0	0	0	0	43
Al	13	24	24	5,992	7,982	0	0	100
Si	14	0	0	1,997	2,993	10,000	10,000	511
P	15	0	0	0	0	450	450	103,244
S	16	35	35	70	70	300	300	100
Ti	22	20	20	7,990	24,943	0	0	100
V	23	20	20	0	0	0	0	0
Cr	24	1,000	1,250	189,753	149,660	180,000	190,000	149,709
Mn[a]	25	20	20	1,997	6,984	20,000	20,000	100
Fe	26	1,500	2,250	179,766	67,846	697,740	688,440	471
Co[a]	27	10	10	4,694	6,485	800	800	381
Ni	28	500	20	519,625	721,861	89,200	89,200	744,438
Cu	29	20	20	999	499	0	0	0
Zr[a]	40	979,630	979,110	0	0	0	0	100
Nb	41	0	0	55,458	8,980	0	0	0
Mo	42	0	0	29,961	0	0	0	0
Cd	48	0.25	0.25	0	0	0	0	0
Sn	50	16,000	16,000	0	0	0	0	0
Hf	72	78	78	0	0	0	0	0
W	74	20	20	0	0	0	0	100
U	92	0.2	0.2	0	0	0	0	0
Density, g/cm^3	--	6.56	6.56	8.19	8.30	8.02	8.02	n/a

[a]Value used in ORIGEN2 should be less than this (actual) value if the materials are not in the active fuel zone. (See ref. 7.)

Table 5.2. Assumed mass distribution for fuel-assembly structural material[a]

	PWR			BWR		
		Mass			Mass	
	Material	kg/MTHM	kg/assembly	Material	kg/MTHM	kg/assembly
Fuel zone						
Cladding	Zircaloy-4	223.0	102.9	Zircaloy-2	279.5	51.2
Fuel channel[b]	--	--	--	Zircaloy-4	227.5	41.7
Grid spacers	Inconel 718	12.8	5.9	Zircaloy-4	10.6	1.9
Grid-spacer springs	Inconel 718			Inconel X-750	1.8	0.3
Grid-brazing material	Nicrobraze 50	2.6	1.2	--	--	--
Miscellaneous	SS 304[c]	9.9	4.6	--	--	--
Fuel-gas plenum zone						
Cladding	Zircaloy-4	12.0	5.5	Zircaloy-2	25.4	4.7
Fuel channel[b]	--	--	--	Zircaloy-4	20.7	3.8
Plenum spring	SS 302	4.2	1.9	SS 302	6.0	1.1
End fitting zone						
Top end fitting	SS 304	14.8	6.8	SS 304	10.9	2.0
Bottom end fitting	SS 304	12.4	5.7	SS 304	26.1	4.8
Expansion springs	--	--	--	Inconel X-750	2.1	0.4
Total		291.7	134.5		610.6	111.9

[a]Data derived from ref. 7.
[b]Assumed to be discarded after one cycle.
[c]Distributed throughout the PWR core in sleeves and so forth.

Table 5.3. Physical characteristics of LWR fuel assemblies[a]

	BWR	PWR
Overall assembly length, m	4.470	4.059
Cross section, cm	13.9 x 13.9	21.4 x 21.4
Fuel element length, m	4.064	3.851
Active fuel height, m	3.759	3.658
Fuel element OD, cm	1.252	0.950
Fuel element array	8 x 8	17 x 17
Fuel elements per assembly	63	264
Assembly total weight, kg	275.7	657.9
Uranium/assembly, kg	183.3	461.4
UO_2/assembly, kg	208.0	523.4
Zircaloy/assembly, kg	99.5[b]	108.4[c]
Hardware/assembly, kg	12.4[d]	26.1[e]
Total metal/assembly, kg	111.9	134.5
Nominal volume/assembly, m^3	0.0864[f]	0.186[f]

[a]Data derived from ref. 7.
[b]Includes Zircaloy fuel-element spacers and fuel channel.
[c]Includes Zircaloy control-rod guide thimbles.
[d]Includes stainless steel tie-plates, Inconel springs, and plenum springs.
[e]Includes stainless steel nozzles and Inconel-718 grids.
[f]Based on overall outside dimension.

calculations. These parameters are presented because of the current interest in such information relative to the possible storage and disposal of unreprocessed spent fuel. It should be noted that the only assumed difference between the standard- and extended-burnup fuel designs is the initial heavy-metal enrichment of the oxide fuel pellets, with higher enrichment required to achieve a higher average burnup. It is also important to remember that the fuel assembly dimensions in Table 5.3 are representative of two generic assemblies: a PWR and a BWR. In reality, there are dozens of dimensional variations in fuel assembly design; some are as unique as an individual reactor, while others are more common for a group of reactors.

5.1.2 Nonactinide-Element Composition of LWR Oxide Fuels

The nonactinide elements present in fresh LWR fuels are comprised of a large number of trace elements (<100 ppm) plus the oxygen present in the actinide oxide. A typical set of nonactinide element concentrations in fresh LWR oxide fuel is given in Table 5.4. These values generally reflect actual measured concentrations instead of the maximum allowable concentrations given in purity specifications. If the concentration of a particular element has been determined to be less than a particular value, then that value is used in Table 5.4. The 2.5-ppm gadolinium concentration given here is only appropriate for PWRs. The use of gadolinium as a burnable poison mixed with the BWR oxide fuel results in this value being increased to 1573 ppm on the average for use in ORIGEN2 calculations of BWR fuel depletion.

5.1.3 Initial Heavy-Metal Composition in LWR Oxide Fuels

The average, initial heavy-metal (i.e., actinide) composition of one metric ton of fuel for the five LWRs is given in Table 5.5. Note that all five LWR models assume a once-through fuel cycle of uranium enriched (only) fuel.

Table 5.4. Assumed nonactinide composition of LWR oxide fuels[a]

Element	Atomic number	Concentration (ppm)[b]	Element	Atomic number	Concentration (ppm)[b]
Li	3	1.0	Mn	25	1.7
B	5	1.0	Fe	26	18.0
C	6	89.4	Co	27	1.0
N	7	25.0	Ni	28	24.0
O	8	134,454.0[c]	Cu	29	1.0
F	9	10.7	Zn	30	40.3
Na	11	15.0	Mo	42	10.0
Mg	12	2.0	Ag	47	0.1
Al	13	16.7	Cd	48	25.0
Si	14	12.1	In	49	2.0
P	15	35.0	Sn	50	4.0
Cl	17	5.3	Gd	64	2.5[d]
Ca	20	2.0	W	74	2.0
Ti	22	1.0	Pb	82	1.0
V	23	3.0	Bi	83	0.4
Cr	24	4.0			

[a]Data derived from ref. 7.
[b]Parts of element per million parts of heavy metal.
[c]Stoichiometric quantity of MO_2 fuel.
[d]Average of 1573 ppm of gadolinium in BWR fuel rods as a burnable poison.

Table 5.5. Assumed initial actinide composition of one metric ton of heavy-metal LWR oxide fuel

	PWR-US[a]	PWR-UE[b]	BWR-US[c]	BWR-US0[d]	BWR-UE[e]
Design burnup, MWd/MTIHM	33,000	50,000	27,500	27,500	40,000
Isotopic composition, g					
^{234}U	290	379	269	269	305
^{235}U	32,000	42,000	30,000	30,000	34,000
^{238}U	967,710	957,621	969,731	969,731	965,695
Total U	1,000,000	1,000,000	1,000,000	1,000,000	1,000,000
Total heavy metal, g	1,000,000	1,000,000	1,000,000	1,000,000	1,000,000

[a]PWR-US = standard-burnup, three-cycle PWR. PWR-U model, used for comparisons, assumes the same oxide fuel composition.

[b]PWR-UE = extended-burnup, three-cycle PWR. PUD50 model, used for comparisons, assumes five cycles and actinide composition of ^{234}U, 376 g/MTIHM; ^{235}U, 41,500 g/MTIHM; and ^{238}U, 958,124 g/MTIHM.

[c]BWR-US = standard-burnup, four-cycle BWR. BWR-U model, used for comparisons, assumes actinide composition of ^{234}U, 247 g/MTIHM; ^{235}U, 27,500 g/MTIHM; and ^{238}U, 972,253 g/MTIHM.

[d]BWR-US0 = standard-burnup, four-cycle BWR with constant (average) axial coolant density. BWR-U model, used for comparisons, uses the same coolant density and assumes actinide composition of ^{234}U, 247 g/MTIHM; ^{235}U, 27,500 g/MTIHM; and ^{238}U, 972,253 g/MTIHM.

[e]BWR-UE = extended-burnup, four-cycle BWR. BWR-UX model, used for comparisons, is the BWR-U model driven to high burnup by increasing the number of cycles and fuel enrichment. BWR-UX assumes six cycles and actinide composition of ^{234}U, 314 g/MTIHM; ^{235}U, 35,000 g/MTIHM; and ^{238}U, 964,686 g/MTIHM.

5.2 ORIGEN2 DEPLETION CALCULATIONS

This section describes and compares the results of ORIGEN2 depletion calculations using the five updated LWR models (e.g., PWR-US, PWR-UE, BWR-US, BWR-US0, and BWR-UE) with identical calculations performed using previously developed ORIGEN2 LWR models (e.g., PWR-U, PUD50, and BWR-U). Of the five updated LWR models, only the BWR-UE is actually "new" (i.e., provides a previously unavailable capability). The other LWR reactor models (i.e., PWR-US, PWR-UE, BWR-US, and BWR-US0) were developed, using the same sophisticated reactor physics calculation techniques, to update the previous models [i.e., the PWR-U (a standard-burnup PWR), PUD50 (an extended-burnup PWR), and BWR-U (a standard-burnup BWR)]. The previous BWR-U model assumed a constant average axial coolant density. For comparison purposes only, a companion to the BWR-US model (called BWR-US0) was developed using this same assumption. Also, to compare with the new extended-burnup BWR model (BWR-UE), the previous standard-burnup BWR model (BWR-U) was used to simulate an extended irradiation cycle. The only difference in this case, called BWR-UX, was the use of a higher initial ^{235}U enrichment and the addition of two average-length burnup cycles (for a total of six cycles), to achieve the 40,000-MWd/MTIHM burnup. Section 5.2.1 presents comparisons of results for spent fuel actinide composition at discharge and after a 10-year decay period as predicted by ORIGEN2 for each of the updated reactor models with the previous models. Section 5.2.2 describes comparisons of results for fission product and activation product radioactivity for spent fuel assemblies after a 10-year decay period, of the same previous-vs-updated-models combinations. Section 5.2.3 describes comparisons of results of selected ORIGEN2 output totals (i.e., radioactivity, thermal power, neutron product rate, and photon emission rate) for spent fuel assemblies after a 10-year decay period. The reader should refer to Appendix D for a description of some important differences between the cases using the previous LWR models and cases based on the updated LWR models. As expected, the agreement of results is best for cases involving LWR models that are most alike (e.g., PWR-U vs PWR-US) and poorest for cases involving LWR models that

are least alike (i.e., BWR-UX vs BWR-UE). In fact, the only reason the last comparison case was included is that past analyses[12] have attempted to characterize extended-burnup BWR spent fuel, even though use of the BWR-U model at higher enrichment to achieve an extended burnup of 40,000 MWd/MTIHM far exceeds the applicable range for which BWR-U was intended.

5.2.1 Comparison of Actinide Compositions

The following paragraphs describe comparisons of the actinide compositions (g/MTIHM) at discharge and after a 10-year decay period.

Tables 5.6 through 5.10 present the comparisons of (1) PWR-U vs PWR-US, (2) PUD50 vs PWR-UE, (3) BWR-U vs BWR-US, (4) BWR-U vs BWR-US0, and (5) BWR-UX vs BWR-UE, respectively.

As expected, the results of the PWR-U and PWR-US cases agree fairly well. One major difference across all five cases is the large increase in the ^{242m}Am composition (at discharge) and the subsequent increase in the ^{242}Am and ^{242}Cm compositions (after a 10-year decay period) as calculated by ORIGEN2 using the updated LWR models. The root of this increase is the ORIGEN2 cross section for ^{240}Pu, which exhibits a difference directly proportional to the difference in concentration of americium or curium isotope. A possible explanation for this difference is that the updated ORIGEN2 cross-section libraries have resulted from a neutron energy spectrum that is, on the average, "harder," and ^{240}Pu is very sensitive to the assumed energy spectrum. Also, the cross-section data for ^{240}Pu were extensively reevaluated in the update from ENDF/B-IV to ENDF/B-V. The previous models were developed using ENDF/B-IV data, but the updated models were prepared using ENDF/B-V data compilations. The result is an ORIGEN2 ^{240}Pu cross section that is significantly greater than that found in previous models. A combination of (n,γ) reactions and β^- decays causes the large increase in buildup of ^{242}Cm and, to a lesser extent, ^{244}Cm. Both ^{242}Cm and, especially, ^{244}Cm are important neutron sources in shielding calculations. The PUD50-vs-PWR-UE case results show mediocre agreement, caused by the use of slightly greater initial enrichment (4.15 vs 4.2 wt % ^{235}U) and less burnup cycles (five vs three) to achieve a burnup of 50,000 MWd/MTIHM.

Table 5.6. Comparison of compositions of standard-burnup PWR spent fuel at discharge and after a 10-year decay period: PWR-U vs PWR-US

	Composition at discharge (g)			Composition at 10 years (g)		
Nuclide	PWR-U[a]	PWR-US[b]	DIFF[c](%)	PWR-U[a]	PWR-US[b]	DIFF[c](%)
^{234}U	1.798E+2	1.731E+2	-3.73	1.908E+2	1.844E+2	-3.35
^{235}U	7.972E+3	7.618E+3	-4.44	7.973E+3	7.619E+3	-4.44
^{236}U	3.959E+3	3.971E+3	0.30	3.961E+3	3.973E+3	0.30
^{238}U	9.441E+5	9.444E+5	0.03	9.441E+5	9.444E+5	0.03
^{237}Np	4.311E+2	4.618E+2	7.12	4.467E+2	4.788E+2	7.19
^{239}Np	9.511E+1	9.743E+1	2.44	7.358E-5	7.868E-5	6.93
^{238}Pu	1.314E+2	1.386E+2	5.48	1.363E+2	1.413E+2	3.67
^{239}Pu	4.938E+3	5.157E+3	4.44	5.032E+3	5.254E+3	4.41
^{240}Pu	2.307E+3	2.115E+3	-8.32	2.312E+3	2.121E+3	-8.26
^{241}Pu	1.218E+3	1.244E+3	2.14	7.529E+2	7.688E+2	2.11
^{242}Pu	4.509E+2	4.626E+2	2.60	4.509E+2	4.626E+2	2.60
^{241}Am	3.254E+1	3.661E+1	12.51	4.935E+2	5.073E+2	2.80
^{242m}Am	7.460E-1	1.553E+0	108.18	7.127E-1	1.484E+0	108.22
^{242}Am	1.117E-1	9.672E-2	-13.41	8.526E-6	1.775E-5	108.19
^{243}Am	8.556E+1	9.150E+1	6.94	8.562E+1	9.156E+1	6.94
^{242}Cm	1.474E+1	1.273E+1	-13.64	1.727E-3	3.594E-3	108.11
^{243}Cm	4.090E-1	3.769E-1	-7.85	3.207E-1	2.956E-1	-7.83
^{244}Cm	2.390E+1	2.674E+1	11.88	1.630E+1	1.824E+1	11.90

[a]PWR-U = burnup, 33 GWd/MTIHM; 3.2 wt % enriched, 3-cycle, PWR-U model.
[b]PWR-US = burnup, 33 GWd/MTIHM; 3.2 wt % enriched, 3-cycle, PWR-US model.
[c]DIFF = 100 x (PWR-US - PWR-U)/PWR-U.

Table 5.7. Comparison of compositions of extended-burnup PWR spent fuel at discharge and after a 10-year decay period: PUD50 vs PWR-UE

	Composition at discharge (g)			Composition at 10 years (g)		
Nuclide	PUD50[a]	PWR-UE[b]	DIFF[c](%)	PUD50[a]	PWR-UE[b]	DIFF[c](%)
^{234}U	1.804E+2	1.783E+2	-1.16	2.063E+2	2.097E+2	1.65
^{235}U	6.782E+3	6.678E+3	-1.53	6.783E+3	6.680E+3	-1.52
^{236}U	5.607E+3	5.505E+3	-1.82	5.610E+3	5.508E+3	-1.82
^{238}U	9.234E+5	9.220E+5	-0.15	9.234E+5	9.220E+5	-0.15
^{237}Np	7.429E+2	9.176E+2	23.52	7.633E+2	9.427E+2	23.50
^{239}Np	9.867E+1	1.048E+2	6.21	1.839E-4	1.795E-4	-2.39
^{238}Pu	3.162E+2	3.971E+2	25.59	3.218E+2	3.902E+2	21.26
^{239}Pu	5.764E+3	6.802E+3	18.01	5.862E+3	6.906E+3	17.81
^{240}Pu	2.716E+3	3.233E+3	19.04	2.741E+3	3.262E+3	19.01
^{241}Pu	1.729E+3	1.591E+3	-7.98	1.069E+3	9.831E+2	-8.04
^{242}Pu	8.563E+2	7.144E+2	-16.57	8.563E+2	7.144E+2	-16.57
^{241}Am	6.945E+1	5.891E+1	-15.18	7.234E+2	6.606E+2	-8.68
^{242m}Am	1.583E+0	2.770E+0	74.98	1.513E+0	2.636E+0	74.88
^{242}Am	1.954E-1	1.362E-1	-30.30	1.810E-5	3.165E-5	74.86
^{243}Am	2.139E+2	2.089E+2	-2.34	2.140E+2	2.089E+2	-2.38
^{242}Cm	2.987E+1	2.194E+1	-26.55	3.666E-3	6.407E-3	74.77
^{243}Cm	1.151E+0	9.730E-1	-15.46	9.028E-1	7.629E-1	-15.50
^{244}Cm	9.045E+1	1.033E+2	14.21	6.170E+1	7.047E+1	14.21

[a]PUD50 = burnup, 50 GWd/MTIHM; 4.15 wt % enriched, 5-cycle, PUD50 model.
[b]PWR-UE = burnup, 50 GWd/MTIHM; 4.2 wt % enriched, 3-cycle, PWR-UE model.
[c]DIFF = 100 x (PWR-UE - PUD50)/PUD50.

Table 5.8. Comparison of compositions of standard-burnup BWR spent fuel at discharge and after a 10-year decay period: BWR-U vs BWR-US

	Composition at discharge (g)			Composition at 10 years (g)		
Nuclide	BWR-U[a]	BWR-US[b]	DIFF[c](%)	BWR-U[a]	BWR-US[b]	DIFF[c](%)
^{234}U	1.575E+2	1.720E+2	9.21	1.658E+2	1.812E+2	9.42
^{235}U	7.528E+3	9.220E+3	22.48	7.530E+3	9.221E+3	22.46
^{236}U	3.313E+3	3.538E+3	6.79	3.315E+3	3.540E+3	6.79
^{238}U	9.517E+5	9.501E+5	-0.17	9.517E+5	9.501E+5	-0.17
^{237}Np	3.376E+2	3.734E+2	10.60	3.487E+2	3.859E+2	10.67
^{239}Np	6.839E+1	6.573E+1	-3.89	5.567E-5	6.040E-5	8.50
^{238}Pu	9.841E+1	1.110E+2	12.79	1.040E+2	1.150E+2	10.58
^{239}Pu	4.763E+3	4.652E+3	-2.33	4.830E+3	4.717E+3	-2.34
^{240}Pu	2.095E+3	1.755E+3	-16.23	2.098E+3	1.759E+3	-16.16
^{241}Pu	1.091E+3	1.159E+3	6.23	6.741E+2	7.162E+2	6.25
^{242}Pu	3.714E+2	3.745E+2	1.08	3.714E+2	3.745E+2	1.08
^{241}Am	4.054E+1	4.607E+1	13.64	4.531E+2	4.844E+2	6.91
^{242m}Am	8.963E-1	2.116E+0	136.08	8.564E-1	2.022E+0	136.10
^{242}Am	9.791E-2	9.065E-2	-7.41	1.024E-5	2.418E-5	136.13
^{243}Am	6.475E+1	7.026E+1	8.51	6.478E+1	7.028E+1	8.49
^{242}Cm	1.335E+1	1.252E+1	-6.22	2.075E-3	4.894E-3	135.86
^{243}Cm	3.311E-1	3.365E-1	1.63	2.596E-1	2.638E-1	1.62
^{244}Cm	1.604E+1	1.864E+1	16.21	1.094E+1	1.272E+1	16.27

[a]BWR-U = burnup, 27.5 GWd/MTIHM; 2.75 wt % enriched, 4-cycle, BWR-U.
[b]BWR-US = burnup, 27.5 GWd/MTIHM; 3.0 wt % enriched, 4-cycle, BWR-US.
[c]DIFF = 100 x (BWR-US - BWR-U)/BWR-U.

Table 5.9. Comparison of compositions of standard-burnup BWR spent fuel at discharge and after a 10-year decay period: BWR-U vs BWR-US0

	Composition at discharge (g)			Composition at 10 years (g)		
Nuclide	BWR-U[a]	BWR-US0[b]	DIFF[c](%)	BWR-U[a]	BWR-US0[b]	DIFF[c](%)
^{234}U	1.575E+2	1.728E+2	9.71	1.658E+2	1.815E+2	8.87
^{235}U	7.528E+3	8.946E+3	18.84	7.530E+3	8.947E+3	18.82
^{236}U	3.313E+3	3.535E+3	6.70	3.315E+3	3.537E+3	6.70
^{238}U	9.517E+5	9.506E+5	-0.12	9.517E+5	9.506E+5	-0.12
^{237}Np	3.376E+2	3.635E+2	7.67	3.487E+2	3.755E+2	7.69
^{239}Np	6.839E+1	6.502E+1	-4.93	5.567E-5	5.611E-5	0.79
^{238}Pu	9.841E+1	1.043E+2	5.99	1.040E+2	1.080E+2	3.85
^{239}Pu	4.763E+3	4.513E+3	-5.25	4.830E+3	4.577E+3	-5.24
^{240}Pu	2.095E+3	1.746E+3	-16.66	2.098E+3	1.749E+3	-16.63
^{241}Pu	1.091E+3	1.094E+3	0.27	6.741E+2	6.759E+2	0.27
^{242}Pu	3.714E+2	3.643E+2	-1.91	3.714E+2	3.643E+2	-1.91
^{241}Am	4.054E+1	4.328E+1	6.76	4.531E+2	4.569E+2	0.84
^{242m}Am	8.963E-1	1.914E+0	113.54	8.564E-1	1.829E+0	113.54
^{242}Am	9.791E-2	8.526E-2	-12.92	1.024E-5	2.188E-5	113.67
^{243}Am	6.475E+1	6.526E+1	0.79	6.478E+1	6.529E+1	0.79
^{242}Cm	1.335E+1	1.173E+1	-12.13	2.075E-3	4.428E-3	113.40
^{243}Cm	3.311E-1	3.016E-1	-8.91	2.596E-1	2.365E-1	-8.90
^{244}Cm	1.604E+1	1.659E+1	3.43	1.094E+1	1.132E+1	3.47

[a]BWR-U = burnup, 27.5 GWd/MTIHM; 2.75 wt % enriched, 4-cycle, BWR-U.
[b]BWR-US0 = burnup, 27.5 GWd/MTIHM; 3.0 wt % enriched, 4-cycle, BWR-US0, (BWR-US except using constant axial coolant density).
[c]DIFF = 100 x (BWR-US0 - BWR-U)/BWR-U.

Table 5.10. Comparison of compositions of extended-burnup BWR spent fuel at discharge and after a 10-year decay period: BWR-UX vs BWR-UE

	Composition at discharge (g)			Composition at 10 years (g)		
Nuclide	BWR-UX[a]	BWR-UE[b]	DIFF[c](%)	BWR-UX[a]	BWR-UE[b]	DIFF[c](%)
^{234}U	1.731E+2	1.562E+2	-9.76	1.911E+2	1.811E+2	-5.23
^{235}U	6.159E+3	7.316E+3	18.79	6.161E+3	7.317E+3	18.76
^{236}U	4.581E+3	4.401E+3	-3.93	4.584E+3	4.403E+3	-3.95
^{238}U	9.374E+5	9.348E+5	-0.28	9.374E+5	9.348E+5	-0.28
^{237}Np	5.789E+2	6.863E+2	18.55	5.937E+2	7.040E+2	18.58
^{239}Np	6.829E+1	7.165E+1	4.92	1.219E-4	1.637E-4	34.29
^{238}Pu	2.194E+2	3.087E+2	40.70	2.250E+2	3.089E+2	37.29
^{239}Pu	4.856E+3	5.945E+3	22.43	4.924E+3	6.016E+3	22.18
^{240}Pu	2.467E+3	2.490E+3	0.93	2.480E+3	2.515E+3	1.41
^{241}Pu	1.394E+3	1.638E+3	17.50	8.612E+2	1.012E+3	17.51
^{242}Pu	6.204E+2	6.608E+2	6.51	6.204E+2	6.609E+2	6.53
^{241}Am	6.181E+1	7.769E+1	25.69	5.887E+2	6.967E+2	18.35
^{242m}Am	1.424E+0	4.045E+0	184.06	1.361E+0	3.865E+0	183.98
^{242}Am	1.505E-1	1.444E-1	-4.05	1.627E-5	4.623E-5	184.14
^{243}Am	1.419E+2	1.904E+2	34.18	1.419E+2	1.904E+2	34.18
^{242}Cm	2.276E+1	2.378E+1	4.48	3.296E-3	9.357E-3	183.89
^{243}Cm	7.218E-1	9.860E-1	36.60	5.660E-1	7.731E-1	36.59
^{244}Cm	4.815E+1	8.745E+1	81.62	3.284E+1	5.965E+1	81.64

[a]BWR-UX = burnup, 40 GWd/MTIHM; 3.5 wt % enriched, 6-cycle, BWR-U model (increased enrichment and number of cycles to attain 40 GWd/MTIHM).
[b]BWR-UE = burnup, 40 GWd/MTIHM; 3.4 wt % enriched, 4-cycle, BWR-UE model.
[c]DIFF = 100 x (BWR-UE - BWR-UX)/BWR-UX.

For the standard-burnup BWR cases (i.e., BWR-U vs BWR-US and BWR-U vs BWR-US0), the differences in results are caused by the different initial enrichment of the fuel [2.75 (previous) vs 3.0 wt % (updated)].

The agreement of results for the BWR-UX vs BWR-UE case is poor, as expected. Previously, no ORIGEN2 reactor model existed for extended-burnup BWR calculations. In many cases, the BWR-U model was used erroneously to calculate extended-burnup BWR results. This comparison has been included to provide a perspective on the differences associated with the use of an ORIGEN2 model with parameters that far exceed the ranges for which the ORIGEN2 model was designed. It is important to note that previous use of the BWR-U model (for extended-burnup calculations) may severely underestimate the actinide composition of the BWR spent fuel. This is primarily attributed to the dependency of the ORIGEN2 model on the initial ^{235}U enrichment of the fuel. By increasing the initial fuel enrichment to achieve extended burnup, as was done in the BWR-UX case, no corresponding change was made or, for that matter, can be made in the one-group cross sections found in the BWR-U library.

5.2.2 Comparison of Fission and Activation Product Radioactivities

Tables 5.11 through 5.15 show comparisons of the fission product and activation product radioactivity for spent fuel assemblies (fuel plus all structural material) after a 10-year decay period for all five cases. In general, no variation of any consequence is observed for the fission product radioactivity, and the sum totals of the fission product radioactivities for all five cases agree within 1 to 2 %. The activation product radioactivities show a wider variation of results. Most noteworthy are those for ^{60}Co, ^{93}Zr, and $^{154,155}Eu$. All four radionuclides are strong gamma-ray (as well as thermal) sources. The increase or decrease in the generation of these activation products appears to be caused by the use of revised multigroup cross-section data, the sensitivity of the multigroup cross-section data to the calculated neutron energy spectrum used to collapse these data to one group, or both.

Table 5.11. Comparison of fission product and activation product radioactivities of standard-burnup PWR spent fuel assemblies after a 10-year decay period: PWR-U vs PWR-US

	Radioactivity at 10 years (Ci/MTIHM)		
	PWR-U[a]	PWR-US[b]	DIFF[c] (%)
Fission products			
^{3}H	3.111E+02	3.094E+02	-0.55
^{85}Kr	4.854E+03	4.875E+03	0.43
^{90}Sr	5.716E+04	5.749E+04	0.58
^{90}Y	5.717E+04	5.751E+04	0.59
^{93}Zr	1.804E+00	1.810E+00	0.33
^{99}Tc	1.306E+01	1.317E+01	0.84
^{106}Ru	5.505E+02	5.465E+02	-0.73
^{106}Rh	5.505E+02	5.465E+02	-0.73
^{113m}Cd	3.475E+01	3.336E+01	-4.00
^{125}Sb	1.162E+03	1.132E+03	-2.58
^{125m}Te	2.835E+02	2.761E+02	-2.61
^{134}Cs	5.221E+03	5.478E+03	4.92
^{137}Cs	8.212E+04	8.214E+04	0.02
^{137m}Ba	7.769E+04	7.771E+04	0.03
^{144}Ce	1.488E+02	1.492E+02	0.27
^{144}Pr	1.488E+02	1.492E+02	0.27
^{147}Pm	9.483E+03	9.369E+03	-1.20
^{151}Sm	3.310E+02	3.496E+02	5.62
^{152}Eu	3.705E+00	3.875E+00	4.59
^{154}Eu	4.690E+03	4.076E+03	-13.09
^{155}Eu	1.599E+03	1.446E+03	-9.57
Total	3.035E+05	3.036E+05	0.03
Activation products			
^{3}H	1.528E+02	1.496E+02	-2.09
^{14}C	1.550E+00	1.457E+00	-6.00
^{55}Fe	3.885E+02	3.673E+02	-5.46
^{60}Co	2.122E+03	3.142E+03	48.07
^{59}Ni	5.112E+00	3.884E+00	-24.02
^{63}Ni	6.466E+02	6.065E+02	-6.20
^{93}Zr	1.267E-01	2.596E-01	104.89
^{94}Nb	1.279E+00	1.294E+00	1.17
^{125}Sb	1.245E+02	1.235E+02	-0.80
^{125m}Te	3.037E+01	3.014E+01	-0.76
^{154}Eu	1.090E-01	1.897E-02	-82.60
^{155}Eu	3.460E-02	5.552E-03	-83.95
Total	3.473E+03	4.426E+03	27.44

[a]PWR-U = burnup, 33 GWd/MTIHM; 3.2 wt % enriched, 3-cycle, PWR-U model.
[b]PWR-US = burnup, 33 GWd/MTIHM; 3.2 wt % enriched, 3-cycle, PWR-US model.
[c]DIFF = 100 x (PWR-US - PWR-U)/PWR-U.

Table 5.12. Comparison of fission product and activation product radioactivities of extended-burnup PWR spent fuel assemblies after a 10-year decay period: PUD50 vs PWR-UE

	Radioactivity at 10 years (Ci/MTIHM)		
	PUD50[a]	PWR-UE[b]	DIFF[c] (%)
Fission products			
^{3}H	4.537E+02	4.606E+02	1.52
^{85}Kr	6.746E+03	6.958E+03	3.14
^{90}Sr	8.183E+04	8.337E+04	1.88
^{90}Y	8.185E+04	8.339E+04	1.88
^{93}Zr	2.659E+00	2.677E+00	0.68
^{99}Tc	1.850E+01	1.877E+01	1.46
^{106}Ru	6.769E+02	7.148E+02	5.60
^{106}Rh	6.769E+02	7.148E+02	5.60
^{113m}Cd	5.845E+01	5.984E+01	2.38
^{125}Sb	1.481E+03	1.593E+03	7.56
^{125m}Te	3.614E+02	3.887E+02	7.55
^{134}Cs	9.690E+03	1.095E+04	13.00
^{137}Cs	1.217E+05	1.226E+05	0.74
^{137m}Ba	1.151E+05	1.160E+05	0.78
^{144}Ce	1.462E+02	1.640E+02	12.18
^{144}Pr	1.462E+02	1.640E+02	12.18
^{147}Pm	9.302E+03	9.225E+03	-0.83
^{151}Sm	4.542E+02	5.130E+02	12.95
^{152}Eu	5.764E+00	5.560E+00	-3.54
^{154}Eu	8.644E+03	8.096E+03	-6.34
^{155}Eu	3.061E+03	3.009E+03	-1.70
Total	4.424E+05	4.484E+05	1.35
Activation products			
^{3}H	1.591E+02	1.580E+02	-0.69
^{14}C	1.931E+00	1.884E+00	-2.43
^{55}Fe	4.096E+02	4.266E+02	4.15
^{60}Co	2.474E+03	3.966E+03	60.31
^{59}Ni	5.273E+00	4.732E+00	-10.26
^{63}Ni	7.844E+02	7.567E+02	-3.53
^{93}Zr	1.974E-01	3.910E-01	98.07
^{94}Nb	1.827E+00	1.960E+00	7.28
^{125}Sb	1.596E+02	1.767E+02	10.71
^{125m}Te	3.895E+01	4.312E+01	10.71
^{154}Eu	9.075E-02	2.958E-02	-67.40
^{155}Eu	3.380E-02	1.024E-02	-69.70
Total	4.035E+03	5.536E+03	37.20

[a]PUD50 = burnup, 50 GWd/MTIHM; 4.15 wt % enriched, 5-cycle, PUD50 model.
[b]PWR-UE = burnup, 50 GWd/MTIHM; 4.2 wt % enriched, 3-cycle, PWR-UE model.
[c]DIFF = 100 x (PWR-UE - PUD50)/PUD50.

Table 5.13. Comparison of fission product and activation product radioactivities of standard-burnup BWR spent fuel assemblies after a 10-year decay period: BWR-U vs BWR-US

	Radioactivity at 10 years (Ci/MTIHM)		
	BWR-U[a]	BWR-US[b]	DIFF[c] (%)
Fission products			
^{3}H	2.532E+02	2.517E+02	-0.59
^{85}Kr	3.924E+03	3.991E+03	1.71
^{90}Sr	4.700E+04	4.789E+04	1.89
^{90}Y	4.701E+04	4.790E+04	1.89
^{93}Zr	1.503E+00	1.519E+00	1.06
^{99}Tc	1.113E+01	1.114E+01	0.09
^{106}Ru	4.037E+02	3.846E+02	-4.73
^{106}Rh	4.037E+02	3.846E+02	-4.73
^{113m}Cd	2.614E+01	2.653E+01	1.49
^{125}Sb	8.520E+02	8.594E+02	0.87
^{125m}Te	2.079E+02	2.097E+02	0.87
^{134}Cs	3.713E+03	3.982E+03	7.24
^{137}Cs	6.796E+04	6.784E+04	-0.18
^{137m}Ba	6.429E+04	6.418E+04	-0.17
^{144}Ce	1.024E+02	1.036E+02	1.17
^{144}Pr	1.024E+02	1.036E+02	1.17
^{147}Pm	8.055E+03	8.102E+03	0.58
^{151}Sm	2.982E+02	3.414E+02	14.49
^{152}Eu	4.841E+00	5.939E+00	22.68
^{154}Eu	3.623E+03	3.075E+03	-15.13
^{155}Eu	1.255E+03	1.150E+03	-8.37
Total	2.495E+05	2.508E+05	0.52
Activation products			
^{3}H	1.469E+02	1.370E+02	-6.74
^{14}C	1.531E+00	1.318E+00	-13.91
^{55}Fe	2.023E+02	1.732E+02	-14.38
^{60}Co	6.657E+02	9.422E+02	41.54
^{59}Ni	1.068E+00	8.355E-01	-21.77
^{63}Ni	1.467E+02	1.246E+02	-15.06
^{93}Zr	3.019E-01	5.761E-01	90.82
^{94}Nb	3.135E-02	3.189E-02	1.72
^{125}Sb	2.521E+02	2.645E+02	4.92
^{125m}Te	6.150E+01	6.455E+01	4.96
^{154}Eu	7.327E+01	1.254E+01	-82.89
^{155}Eu	2.257E+01	3.614E+00	-83.99
Total	1.574E+03	1.725E+03	9.59

[a]BWR-U = burnup, 27.5 GWd/MTIHM; 2.75 wt % enriched, 4-cycle, BWR-U model.
[b]BWR-US = burnup, 27.5 GWd/MTIHM; 3.0 wt % enriched, 4-cycle, BWR-US model.
[c]DIFF = 100 x (BWR-US - BWR-U)/BWR-U.

Table 5.14. Comparison of fission product and activation product radioactivities of standard-burnup BWR spent fuel assemblies after a 10-year decay period: BWR-U vs BWR-US0

	Radioactivity at 10 years (Ci/MTIHM)		
	BWR-U[a]	BWR-US0[b]	DIFF[c] (%)
Fission products			
^{3}H	2.532E+02	2.505E+02	-1.07
^{85}Kr	3.924E+03	4.016E+03	2.34
^{90}Sr	4.700E+04	4.830E+04	2.77
^{90}Y	4.701E+04	4.831E+04	2.77
^{93}Zr	1.503E+00	1.528E+00	1.66
^{99}Tc	1.113E+01	1.115E+01	0.18
^{106}Ru	4.037E+02	3.788E+02	-6.17
^{106}Rh	4.037E+02	3.788E+02	-6.17
^{113m}Cd	2.614E+01	2.573E+01	-1.57
^{125}Sb	8.520E+02	8.470E+02	-0.59
^{125m}Te	2.079E+02	2.067E+02	-0.58
^{134}Cs	3.713E+03	3.864E+03	4.07
^{137}Cs	6.796E+04	6.784E+04	-0.18
^{137m}Ba	6.429E+04	6.418E+04	-0.17
^{144}Ce	1.024E+02	1.039E+02	1.46
^{144}Pr	1.024E+02	1.039E+02	1.46
^{147}Pm	8.055E+03	8.182E+03	1.58
^{151}Sm	2.982E+02	3.211E+02	7.68
^{152}Eu	4.841E+00	5.529E+00	14.21
^{154}Eu	3.623E+03	2.995E+03	-17.33
^{155}Eu	1.255E+03	1.092E+03	-12.99
Total	2.495E+05	2.514E+05	0.77
Activation products			
^{3}H	1.469E+02	1.394E+02	-5.11
^{14}C	1.531E+00	1.354E+00	-11.56
^{55}Fe	2.023E+02	1.792E+02	-11.42
^{60}Co	6.657E+02	9.431E+02	41.67
^{59}Ni	1.068E+00	8.601E-01	-19.47
^{63}Ni	1.467E+02	1.287E+02	-12.27
^{93}Zr	3.019E-01	5.480E-01	81.52
^{94}Nb	3.135E-02	3.003E-02	-4.21
^{125}Sb	2.521E+02	2.478E+02	-1.71
^{125m}Te	6.150E+01	6.047E+01	-1.67
^{154}Eu	7.327E+01	1.167E+01	-84.07
^{155}Eu	2.257E+01	3.270E+00	-85.51
Total	1.574E+03	1.716E+03	9.05

[a]BWR-U = burnup, 27.5 GWd/MTIHM; 2.75 wt % enriched, 4-cycle, BWR-U model.
[b]BWR-US0 = burnup, 27.5 GWd/MTIHM; 3.0 wt % enriched, 4-cycle, BWR-US0 model.
[c]DIFF = 100 x (BWR-US0 - BWR-U)/BWR-U.

Table 5.15. Comparison of fission product and activation product radioactivities of extended-burnup BWR spent fuel assemblies after a 10-year decay period: BWR-UX vs BWR-UE

	Radioactivity at 10 years (Ci/MTIHM)		
	BWR-UX[a]	BWR-UE[b]	DIFF[c] (%)
Fission products			
^{3}H	3.503E+02	3.661E+02	4.51
^{85}Kr	5.322E+03	5.270E+03	-0.98
^{90}Sr	6.616E+04	6.394E+04	-3.36
^{90}Y	6.617E+04	6.396E+04	-3.34
^{93}Zr	2.163E+00	2.108E+00	-2.54
^{99}Tc	1.560E+01	1.532E+01	-1.79
^{106}Ru	4.671E+02	5.200E+02	11.33
^{106}Rh	4.671E+02	5.200E+02	11.33
^{113m}Cd	3.925E+01	4.868E+01	24.03
^{125}Sb	1.023E+03	1.208E+03	18.08
^{125m}Te	2.497E+02	2.948E+02	18.06
^{134}Cs	6.152E+03	7.677E+03	24.79
^{137}Cs	9.665E+04	9.711E+04	0.48
^{137m}Ba	9.143E+04	9.186E+04	0.47
^{144}Ce	1.010E+02	1.102E+02	9.11
^{144}Pr	1.010E+02	1.102E+02	9.11
^{147}Pm	8.201E+03	7.856E+03	-4.21
^{151}Sm	3.552E+02	4.867E+02	37.02
^{152}Eu	6.099E+00	7.858E+00	28.84
^{154}Eu	6.249E+03	6.208E+03	-0.66
^{155}Eu	2.098E+03	2.423E+03	15.49
Total	3.516E+05	3.500E+05	-0.46
Activation products			
^{3}H	1.585E+02	1.482E+02	-6.50
^{14}C	2.046E+00	1.589E+00	-22.34
^{55}Fe	2.311E+02	1.260E+02	-45.48
^{60}Co	8.003E+02	6.345E+02	-20.72
^{59}Ni	1.386E+00	4.065E-01	-70.67
^{63}Ni	1.934E+02	5.967E+01	-69.15
^{93}Zr	4.024E-01	8.817E-01	119.11
^{94}Nb	4.128E-02	1.173E-02	-71.58
^{125}Sb	2.888E+02	3.766E+02	30.40
^{125m}Te	7.046E+01	9.189E+01	30.41
^{154}Eu	6.470E+01	2.010E+01	-68.93
^{155}Eu	2.215E+01	7.087E+00	-68.00
Total	1.833E+03	1.467E+03	-19.98

[a]BWR-UX = burnup, 40 GWd/MTIHM; 3.5 wt % enriched, 6-cycle, BWR-U model.
[b]BWR-UE = burnup, 40 GWd/MTIHM; 3.4 wt % enriched, 4-cycle, BWR-UE model.
[c]DIFF = 100 x (BWR-UE - BWR-UX)/BWR-UX.

One interesting comparison to note is that the total radioactivity of the activation products for the updated BWR models increases to a lesser extent than for the PWR models. This finding is attributed to the much greater composition of gadolinium (as a burnable poison) assumed to be present in BWR spent fuel.

5.2.3 Comparison of Selected ORIGEN2 Output Totals

Tables 5.16 through 5.20 show comparisons of results for selected ORIGEN2 output totals (i.e., radioactivity, thermal power, neutron production rate, and photon emission rate) for the five cases of spent fuel assemblies after a 10-year decay period.

The total radioactivity and the total photon emission rate for spent fuel assemblies agree within 1 to 3%.

The increase in thermal power (2 to 6% for PWRs, 1 to 10% for BWRs) is important to note since the design and optimization of shipping casks and storage facilities require these data.

Another very important result is the overall increase in the neutron source strength (12 to 15% for the PWR cases, 3 to 80% for the BWR cases), which would affect the shielding design of shipping casks. However, the approximate 80% increase in neutron source strength in the BWR case is attributed to the erroneous previous use of the BWR-U model in extended-burnup calculations.

5.3 SUMMARY OF THE ORIGEN2 URANIUM-CYCLE LWR MODELS

Fuel-depletion calculations have been performed for the five new and/or revised uranium-cycle LWR fuel types considered in this report - PWR-US, PWR-UE, BWR-US, BWR-US0, and BWR-UE - using input compositions given in Table 5.5. The results of these calculations, including irradiation conditions and measures of the uranium and plutonium compositions of fresh and spent fuels, are given in Table 5.21.

Table 5.16. Comparison of selected ORIGEN2 output totals of standard-burnup PWR spent fuel assemblies after a 10-year decay period: PWR-U vs PWR-US

	PWR-U[a]	PWR-US[b]	DIFF[c] (%)
Radioactivity, Ci/MTIHM			
Activation products	3.474E+03	4.427E+03	27.43
Actinide and daughters	8.386E+04	8.578E+04	2.29
Fission products	3.035E+05	3.036E+05	0.03
Total	3.909E+05	3.938E+05	0.74
Thermal power, W/MTIHM			
Activation products	3.324E+01	4.895E+01	47.26
Actinide and daughters	2.098E+02	2.191E+02	4.43
Fission products	8.965E+02	8.957E+02	-0.09
Total	1.140E+03	1.164E+03	2.11
Neutron production rate, neutrons$\cdot s^{-1} \cdot MTIHM^{-1}$			
(α,n) reaction	6.026E+06	6.317E+06	4.83
Spontaneous fission	1.855E+08	2.070E+08	11.59
Total	1.915E+08	2.133E+08	11.38
Photon emission rate, photons$\cdot s^{-1} \cdot MTIHM^{-1}$			
Activation products	1.717E+14	2.500E+14	45.60
Actinide and daughters	6.794E+13	7.065E+13	3.99
Fission products	7.492E+15	7.475E+15	-0.23
Total	7.732E+15	7.796E+15	0.83

[a]PWR-U = burnup, 33 GWd/MTIHM; 3.2 wt % enriched, 3-cycle, PWR-U model.
[b]PWR-US = burnup, 33 GWd/MTIHM; 3.2 wt % enriched, 3-cycle, PWR-US model.
[c]DIFF = 100 x (PWR-US - PWR-U)/PWR-U.

Table 5.17. Comparison of selected ORIGEN2 output totals of extended-burnup PWR spent fuel assemblies after a 10-year decay period: PUD50 vs PWR-UE

	PUD50[a]	PWR-UE[b]	DIFF[c] (%)
Radioactivity, Ci/MTIHM			
Activation products	4.036E+03	5.538E+03	37.22
Actinide and daughters	1.243E+05	1.174E+05	-5.55
Fission products	4.425E+05	4.484E+05	1.33
Total	5.708E+05	5.713E+05	0.09
Thermal power, W/MTIHM			
Activation products	3.880E+01	6.186E+01	59.43
Actinide and daughters	4.779E+02	5.398E+02	12.95
Fission products	1.340E+03	1.965E+03	46.64
Total	1.857E+03	1.965E+03	5.82
Neutron production rate, neutrons$\cdot s^{-1} \cdot$MTIHM^{-1}			
(α,n) reaction	1.418E+07	1.604E+07	13.12
Spontaneous fission	6.974E+08	7.968E+08	14.25
Total	7.115E+08	8.128E+08	14.24
Photon emission rate, photons$\cdot s^{-1} \cdot$MTIHM^{-1}			
Activation products	2.012E+14	3.170E+14	57.55
Actinide and daughters	1.312E+14	1.377E+14	4.95
Fission products	1.125E+16	1.144E+16	1.69
Total	1.158E+16	1.189E+16	2.70

[a]PUD50 = burnup, 50 GWd/MTIHM; 4.15 wt % enriched, 5-cycle, PUD50 model.
[b]PWR-UE = burnup, 50 GWd/MTIHM; 4.2 wt % enriched, 3-cycle, PWR-UE model.
[c]DIFF = 100 x (PWR-UE - PUD50)/PUD50.

Table 5.18. Comparison of selected ORIGEN2 output totals of standard-burnup BWR spent fuel assemblies after a 10-year decay period: BWR-U vs BWR-US

	BWR-U[a]	BWR-US[b]	DIFF[c] (%)
Radioactivity, Ci/MTIHM			
Activation products	1.576E+03	1.727E+03	9.58
Actinide and daughters	7.454E+04	7.928E+04	6.36
Fission products	2.495E+05	2.508E+05	0.52
Total	3.256E+05	3.318E+05	1.90
Thermal power, W/MTIHM			
Activation products	1.181E+01	1.555E+01	31.67
Actinide and daughters	1.693E+02	1.821E+02	7.56
Fission products	7.302E+02	7.334E+02	0.44
Total	9.113E+02	9.310E+02	2.16
Neutron production rate, neutrons$\cdot s^{-1} \cdot$MTIHM^{-1}			
(α,n) reaction	4.818E+06	5.215E+06	8.24
Spontaneous fission	1.251E+08	1.446E+08	15.59
Total	1.299E+08	1.498E+08	15.32
Photon emission rate, photons$\cdot s^{-1} \cdot$MTIHM^{-1}			
Activation products	7.479E+13	9.144E+13	22.26
Actinide and daughters	5.811E+13	6.282E+13	8.11
Fission products	6.096E+15	6.110E+15	0.23
Total	6.229E+15	6.264E+15	0.57

[a]BWR-U = burnup, 27.5 GWd/MTIHM; 2.75 wt % enriched, 4-cycle, BWR-U model.
[b]BWR-US = burnup, 27.5 GWd/MTIHM; 3.0 wt % enriched, 4-cycle, BWR-US model.
[c]DIFF = 100 x (BWR-US - BWR-U)/BWR-U.

Table 5.19. Comparison of selected ORIGEN2 output totals of standard-burnup BWR spent fuel assemblies after a 10-year decay period: BWR-U vs BWR-US0

	BWR-U[a]	BWR-US0[b]	DIFF[c] (%)
Radioactivity, Ci/MTIHM			
Activation products	1.576E+03	1.718E+03	9.01
Actinide and daughters	7.454E+04	7.477E+04	0.31
Fission products	2.495E+05	2.514E+05	0.76
Total	3.256E+05	3.279E+05	0.71
Thermal power, W/MTIHM			
Activation products	1.181E+01	1.550E+01	31.24
Actinide and daughters	1.693E+02	1.704E+02	0.65
Fission products	7.302E+02	7.342E+02	0.55
Total	9.113E+02	9.201E+02	0.97
Neutron production rate, neutrons$\cdot s^{-1} \cdot MTIHM^{-1}$			
(α,n) reaction	4.818E+06	4.868E+06	1.04
Spontaneous fission	1.251E+08	1.289E+08	3.04
Total	1.299E+08	1.337E+08	2.93
Photon emission rate, photons$\cdot s^{-1} \cdot MTIHM^{-1}$			
Activation products	7.479E+13	9.029E+13	20.72
Actinide and daughters	5.811E+13	5.894E+13	1.43
Fission products	6.096E+15	6.115E+15	0.31
Total	6.229E+15	6.264E+15	0.57

[a]BWR-U = burnup, 27.5 GWd/MTIHM; 2.75 wt % enriched, 4-cycle, BWR-U model.
[b]BWR-US0 = burnup, 27.5 GWd/MTIHM; 3.0 wt % enriched, 4-cycle, BWR-US0 model.
[c]DIFF = 100 x (BWR-US0 - BWR-U)/BWR-U.

Table 5.20. Comparison of selected ORIGEN2 output totals of extended-burnup BWR spent fuel assemblies after a 10-year decay period: BWR-UX vs BWR-UE

	BWR-UX[a]	BWR-UE[b]	DIFF[c] (%)
Radioactivity, Ci/MTIHM			
Activation products	1.835E+03	1.470E+03	-19.89
Actinide and daughters	9.830E+04	1.180E+05	20.04
Fission products	3.516E+05	3.500E+05	-0.46
Total	4.518E+05	4.694E+05	3.90
Thermal power, W/MTIHM			
Activation products	1.393E+01	1.124E+01	-19.31
Actinide and daughters	3.203E+02	4.604E+02	43.74
Fission products	1.049E+03	1.053E+03	0.38
Total	1.384E+03	1.525E+03	10.19
Neutron production rate, neutrons$\cdot s^{-1} \cdot$MTIHM^{-1}			
(α,n) reaction	9.371E+06	1.368E+07	45.98
Spontaneous fission	3.718E+08	6.731E+08	81.04
Total	3.812E+08	6.867E+08	80.14
Photon emission rate, photons$\cdot s^{-1} \cdot$MTIHM^{-1}			
Activation products	8.693E+13	7.607E+13	-12.49
Actinide and daughters	9.444E+13	1.268E+14	34.27
Fission products	8.778E+15	8.847E+15	0.79
Total	8.959E+15	9.050E+15	1.01

[a]BWR-UX = burnup, 40 GWd/MTIHM; 3.5 wt % enriched, 6-cycle, BWR-U model.
[b]BWR-UE = burnup, 40 GWd/MTIHM; 3.4 wt % enriched, 4-cycle, BWR-UE model.
[c]DIFF = 100 x (BWR-UE - BWR-UX)/BWR-UX.

Table 5.21. Summary of ORIGEN2 LWR model characteristics

Parameter	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Electric power, MW(e)	1,150	1,150	1,200	1,200	1,200
Thermal power, MW(t)	3,411	3,411	3,579	3,579	3,579
Average specific power,[a] MW(t)/MTIHM[b]	37.5	37.5	25.9	25.9	25.9
Average fuel burnup, MWd/MTIHM[b]	33,000	50,000	27,500	27,500	40,000
Irradiation duration, full-power days	880	1333.3	1061.8	1061.8	1544.4
Average number of refueling cycles	3	3	4	4	4
Refueling cycle length, full-power days	293.3	444.4	265.45	265.45	386.1
at 80% capacity	366.7	555.5	331.80	331.80	482.6
Charge, g/MTIHM					
^{235}U	32,000	42,000	30,000	30,000	34,000
Total U	1,000,000	1,000,000	1,000,000	1,000,000	1,000,000
Fissile Pu[c]	0	0	0	0	0
Total Pu[d]	0	0	0	0	0
Total (U + Pu)	1,000,000	1,000,000	1,000,000	1,000,000	1,000,000

Table 5.21 (continued)

Parameter	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Discharge, g/MTIHM					
^{235}U	7,618	6,678	9,220	8,946	7,316
Total U[c]	956,200	934,400	963,000	963,300	946,700
Fissile Pu[d]	6,498	8,498	5,877	5,672	7,655
Total Pu[e]	9,215	12,842	8,117	7,887	11,114
Total (U + Pu)[f]	965,415	947,242	971,117	971,187	957,814
Total heavy metal[g]	966,000	948,500	971,700	971,600	958,900

[a]Based on full power and full charge.
[b]MTIHM = metric ton of initial heavy metal.
[c]Total U = total uranium as reported in ORIGEN2 output.
[d]Fissile Pu = $^{239}Pu + ^{241}Pu + ^{239}Np$.
[e]Total Pu = $^{238}Pu + ^{239}Pu + ^{240}Pu + ^{241}Pu + ^{242}Pu + ^{239}Np$.
[f]Total (U + Pu) = sum of total uranium plus total plutonium.
[g]Total heavy metal = total of all actinides in ORIGEN2 output.

5.4 COMPARISON OF ORIGEN2 DEPLETION RESULTS WITH LITERATURE

Table 5.22 compares the composition of selected actinides as predicted by ORIGEN2 fuel-depletion calculations with experimental values found in the literature.[33] ORIGEN2 (with the PWR-US model) was used to simulate the burnup of fuel to a burnup of 31,364 MWd/MTIHM, using fuel of an initial enrichment of 2.561 wt % ^{235}U. These results were then compared with the experimentally measured composition of discharged spent fuel from the H. B. Robinson reactor at the same conditions of enrichment and burnup. The ORIGEN2-predicted composition shows excellent agreement with the experimental measurements. Further comparisons with experimental data will be the subject of a future report as ORNL continues to validate the ORIGEN2 computer code.

Table 5.22. Comparison of ORIGEN2 PWR-US fuel-depletion calculations with literature values

Nuclide	Composition (atom %)		DIFF[c] (%)
	ORIGEN2[a]	Ref. 33[b]	
^{234}U	0.01475	0.014	5.36
^{235}U	0.6002	0.653	-8.09
^{236}U	0.3419	0.347	-1.47
^{238}U	99.04	98.99	0.05
^{238}Pu	1.527	1.56	-2.16
^{239}Pu	54.52	55.79	-2.28
^{240}Pu	25.48	24.91	2.29
^{241}Pu	12.75	12.29	3.74
^{242}Pu	5.725	5.45	5.05
^{241}Am	56.64	62.4	-9.23
^{242}Am	0.3315	0.312	6.25
^{243}Am	43.03	37.29	15.39
^{242}Cm	2.792	3.28	-14.88
^{243}Cm	1.295	1.21	7.03
^{244}Cm	91.63	90.66	1.07
^{245}Cm	3.810	4.43	-14.00
^{246}Cm	0.4645	0.453	2.45
^{247}Cm	0.004412	0.004	10.30
^{248}Cm	0.0002485	0.001	-75.15

[a]ORIGEN2, PWR-US model, BU = 31,364 MWd/MTIHM, enrichment = 2.561 wt %.
[b]Ref. 33 = measured composition of H. B. Robinson fuel on 3/6/76.
[c]DIFF = 100 x (ORIGEN2 - ref. 33 value)/ref. 33 value.

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APPENDIXES

Appendix A.

ONE-GROUP, SPECTRUM-AVERAGE CROSS SECTIONS FOR URANIUM-CYCLE PWR AND BWR FUELS

The cross sections for the five new or revised LWR models (i.e., PWR-US, PWR-UE, BWR-US, BWR-US0, and BWR-UE) are listed (by nuclide and reaction type) in Tables A.1 through A.3. These tables were extracted directly from the new or revised ORIGEN2 cross-section library files by W. J. Reich.

Table A.1. One-group, spectrum-averaged cross sections for uranium cycle light-water reactors - light elements

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
H-1	N,G	2.834E-02	2.236E-02	2.494E-02	2.729E-02	1.968E-02
H-2	N,G	4.527E-05	3.571E-05	3.983E-05	4.358E-05	3.143E-05
H-2	N,2N	8.395E-04	8.733E-04	8.437E-04	8.286E-04	8.721E-04
H-3	N,G	5.125E-07	4.043E-07	4.509E-07	4.933E-07	3.559E-07
He-3	N,G	5.421E+01	5.411E+01	5.564E+01	5.566E+01	5.545E+01
He-3	N,P	4.552E+02	3.591E+02	4.005E+02	4.382E+02	3.161E+02
Li-6	N,G	2.391E-03	1.887E-03	2.104E-03	2.302E-03	1.661E-03
Li-6	N,A	7.704E+01	6.010E+01	6.731E+01	7.395E+01	5.243E+01
Li-6	N,P	8.925E-04	9.285E-04	8.971E-04	8.810E-04	9.273E-04
Li-7	N,G	3.095E-03	2.445E-03	2.726E-03	2.981E-03	2.154E-03
Li-7	N,2N	4.103E-06	4.263E-06	3.535E-06	3.685E-06	3.661E-06
Li-7	N,A	5.568E-03	5.792E-03	5.596E-03	5.495E-03	5.784E-03
Be-9	N,G	7.826E-04	6.167E-04	6.882E-04	7.532E-04	5.424E-04
Be-9	N,2N	4.644E-02	4.831E-02	4.668E-02	4.584E-02	4.825E-02
Be-9	N,A	2.693E-02	2.802E-02	2.707E-02	2.658E-02	2.798E-02
Be-10	N,G	8.541E-05	6.738E-05	7.516E-05	8.222E-05	5.931E-05
B-10	N,G	4.271E-02	3.369E-02	3.758E-02	4.111E-02	2.965E-02
B-10	N,A	3.274E+02	2.583E+02	2.881E+02	3.152E+02	2.274E+02
B-10	N,P	2.529E-03	2.627E-03	2.304E-03	2.366E-03	2.384E-03
B-11	N,G	4.389E-04	3.494E-04	3.882E-04	4.231E-04	3.094E-04
B-11	N,2N	5.477E-07	5.698E-07	5.505E-07	5.406E-07	5.690E-07
B-11	N,A	2.574E-06	2.675E-06	2.218E-06	2.312E-06	2.297E-06
C-12	N,G	2.884E-04	2.274E-04	2.538E-04	2.778E-04	2.001E-04
C-12	N,A	3.479E-04	3.616E-04	3.004E-04	3.129E-04	3.112E-04
C-13	N,G	9.717E-05	8.090E-05	8.848E-05	9.484E-05	7.414E-05
C-13	N,A	5.611E-04	5.837E-04	5.639E-04	5.538E-04	5.829E-04
C-14	N,G	8.541E-08	6.738E-08	7.516E-08	8.222E-08	5.931E-08
N-14	N,G	6.403E-03	5.054E-03	5.637E-03	6.166E-03	4.450E-03
N-14	N,A	2.546E-02	2.658E-02	2.318E-02	2.378E-02	2.409E-02
N-14	N,P	1.692E-01	1.370E-01	1.499E-01	1.628E-01	1.215E-01
N-15	N,G	2.050E-06	1.617E-06	1.804E-06	1.973E-06	1.423E-06
N-15	N,A	2.816E-06	2.930E-06	2.831E-06	2.780E-06	2.926E-06
N-15	N,P	4.208E-07	4.378E-07	4.230E-07	4.154E-07	4.372E-07
O-16	N,G	1.520E-05	1.199E-05	1.338E-05	1.464E-05	1.056E-05
O-16	N,A	2.560E-03	2.668E-03	2.270E-03	2.346E-03	2.357E-03
O-16	N,P	3.897E-06	4.054E-06	3.917E-06	3.847E-06	4.049E-06
O-17	N,A	2.005E-02	1.582E-02	1.764E-02	1.930E-02	1.392E-02
O-18	N,G	3.040E-05	2.749E-05	2.921E-05	3.034E-05	2.661E-05
O-18	N,A	3.237E-04	3.367E-04	3.253E-04	3.195E-04	3.363E-04
F-19	N,G	1.114E-03	9.418E-04	1.024E-03	1.091E-03	8.726E-04
F-19	N,A	3.697E-03	3.846E-03	3.715E-03	3.649E-03	3.841E-03
F-19	N,P	3.390E-04	3.527E-04	3.407E-04	3.346E-04	3.522E-04
Ne-20	N,G	3.160E-03	2.493E-03	2.781E-03	3.042E-03	2.194E-03
Ne-20	N,A	4.316E-03	4.490E-03	4.338E-03	4.260E-03	4.484E-03
Ne-21	N,G	5.910E-02	4.663E-02	5.201E-02	5.690E-02	4.104E-02
Ne-21	N,A	1.281E-01	1.011E-01	1.127E-01	1.233E-01	8.896E-02

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Ne-22	N,G	4.100E-03	3.234E-03	3.607E-03	3.947E-03	2.847E-03
Na-22	N,G	5.810E+03	5.281E+03	5.601E+03	5.807E+03	5.129E+03
Na-23	N,G	4.942E-02	3.995E-02	4.428E-02	4.784E-02	3.597E-02
Na-23	N,GX	3.541E-02	2.820E-02	3.134E-02	3.417E-02	2.500E-02
Na-23	N,A	1.551E-04	1.612E-04	1.343E-04	1.398E-04	1.392E-04
Na-23	N,P	4.337E-04	4.515E-04	3.788E-04	3.932E-04	3.929E-04
Mg-24	N,G	4.591E-03	3.653E-03	4.062E-03	4.429E-03	3.237E-03
Mg-24	N,A	3.239E-04	3.370E-04	3.256E-04	3.197E-04	3.365E-04
Mg-24	N,P	2.825E-04	2.939E-04	2.839E-04	2.788E-04	2.935E-04
Mg-25	N,G	1.605E-02	1.281E-02	1.423E-02	1.550E-02	1.137E-02
Mg-25	N,A	3.669E-03	3.817E-03	3.687E-03	3.621E-03	3.811E-03
Mg-25	N,P	4.305E-04	4.479E-04	4.327E-04	4.249E-04	4.473E-04
Mg-26	N,G	3.440E-03	2.751E-03	3.053E-03	3.323E-03	2.447E-03
Mg-26	N,A	6.474E-04	6.735E-04	6.507E-04	6.390E-04	6.726E-04
Mg-27	N,G	3.416E-03	2.695E-03	3.006E-03	3.289E-03	2.372E-03
Al-27	N,G	2.115E-02	1.700E-02	1.883E-02	2.046E-02	1.518E-02
Al-27	N,A	1.400E-04	1.456E-04	1.407E-04	1.382E-04	1.454E-04
Al-27	N,P	6.692E-04	6.962E-04	6.726E-04	6.605E-04	6.952E-04
Si-28	N,G	1.455E-02	1.149E-02	1.281E-02	1.401E-02	1.012E-02
Si-28	N,A	1.596E-04	1.660E-04	1.604E-04	1.575E-04	1.658E-04
Si-28	N,P	8.634E-04	8.982E-04	8.678E-04	8.522E-04	8.970E-04
Si-29	N,G	2.391E-02	1.887E-02	2.104E-02	2.302E-02	1.661E-02
Si-29	N,A	1.380E-03	1.436E-03	1.388E-03	1.363E-03	1.434E-03
Si-29	N,P	5.833E-04	6.068E-04	5.863E-04	5.757E-04	6.060E-04
Si-30	N,G	1.045E-02	8.519E-03	9.388E-03	1.014E-02	7.688E-03
Si-30	N,A	2.158E-05	2.245E-05	2.169E-05	2.130E-05	2.242E-05
Si-30	N,P	8.528E-07	8.872E-07	8.572E-07	8.418E-07	8.860E-07
Si-31	N,G	4.100E-02	3.234E-02	3.607E-02	3.947E-02	2.847E-02
P-31	N,G	1.535E-02	1.211E-02	1.350E-02	1.478E-02	1.065E-02
P-31	N,A	2.788E-04	2.901E-04	2.802E-04	2.752E-04	2.897E-04
P-31	N,P	6.692E-03	6.962E-03	6.726E-03	6.605E-03	6.952E-03
S-32	N,G	4.527E-02	3.571E-02	3.983E-02	4.358E-02	3.143E-02
S-32	N,A	3.361E-03	3.411E-03	3.335E-03	3.309E-03	3.374E-03
S-32	N,P	1.295E-02	1.347E-02	1.302E-02	1.278E-02	1.345E-02
S-33	N,2N	1.685E-06	1.753E-06	1.694E-06	1.663E-06	1.751E-06
S-33	N,A	1.196E-02	9.433E-03	1.052E-02	1.151E-02	8.303E-03
S-33	N,P	1.708E-04	1.348E-04	1.503E-04	1.644E-04	1.186E-04
S-34	N,G	2.050E-02	1.617E-02	1.804E-02	1.973E-02	1.423E-02
S-34	N,A	5.397E-04	5.615E-04	5.425E-04	5.327E-04	5.607E-04
S-34	N,P	1.358E-04	1.412E-04	1.365E-04	1.340E-04	1.410E-04
S-36	N,G	1.281E-02	1.011E-02	1.127E-02	1.233E-02	8.896E-03
Cl-35	N,G	3.619E+00	2.844E+00	3.177E+00	3.481E+00	2.496E+00
Cl-35	N,A	1.323E-03	1.375E-03	1.329E-03	1.306E-03	1.372E-03
Cl-35	N,P	4.876E-02	4.003E-02	4.383E-02	4.724E-02	3.614E-02
Cl-36	N,G	8.541E-01	6.738E-01	7.516E-01	8.222E-01	5.931E-01
Cl-37	N,G	3.875E-02	3.112E-02	3.448E-02	3.747E-02	2.777E-02
Cl-37	N,GX	4.667E-04	3.765E-04	4.165E-04	4.519E-04	3.371E-04

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Cl-37	N,A	2.590E-04	2.694E-04	2.603E-04	2.556E-04	2.690E-04
Cl-37	N,P	5.190E-05	5.399E-05	5.216E-05	5.123E-05	5.392E-05
Ar-36	N,G	4.327E-01	3.425E-01	3.816E-01	4.169E-01	3.023E-01
Ar-36	N,A	8.826E-03	9.064E-03	8.812E-03	8.700E-03	9.008E-03
Ar-36	N,P	1.538E-02	1.600E-02	1.546E-02	1.518E-02	1.598E-02
Ar-38	N,G	6.924E-02	5.481E-02	6.106E-02	6.671E-02	4.838E-02
Ar-39	N,G	5.125E+01	4.043E+01	4.509E+01	4.933E+01	3.559E+01
Ar-40	N,G	5.893E-02	4.703E-02	5.223E-02	5.690E-02	4.177E-02
Ar-40	N,A	5.381E-02	4.245E-02	4.735E-02	5.180E-02	3.737E-02
Ar-41	N,G	4.271E-02	3.369E-02	3.758E-02	4.111E-02	2.965E-02
K-39	N,G	1.678E-01	1.325E-01	1.477E-01	1.616E-01	1.167E-01
K-39	N,A	4.057E-03	4.128E-03	4.032E-03	3.995E-03	4.088E-03
K-39	N,P	1.552E-02	1.614E-02	1.560E-02	1.532E-02	1.612E-02
K-40	N,G	2.562E+00	2.021E+00	2.255E+00	2.467E+00	1.779E+00
K-40	N,A	3.328E-02	2.625E-02	2.928E-02	3.203E-02	2.311E-02
K-40	N,P	3.759E-01	2.965E-01	3.308E-01	3.619E-01	2.610E-01
K-41	N,G	1.420E-01	1.156E-01	1.275E-01	1.378E-01	1.043E-01
C-40	N,G	3.416E-02	2.695E-02	3.006E-02	3.289E-02	2.372E-02
C-40	N,A	1.662E-03	1.675E-03	1.643E-03	1.635E-03	1.653E-03
C-40	N,P	1.014E-02	1.055E-02	1.019E-02	1.001E-02	1.054E-02
C-42	N,G	5.546E-02	4.374E-02	4.879E-02	5.339E-02	3.849E-02
C-42	N,A	5.175E-03	5.384E-03	5.201E-03	5.108E-03	5.376E-03
C-42	N,P	4.534E-03	4.717E-03	4.557E-03	4.475E-03	4.710E-03
C-43	N,G	5.910E-01	4.791E-01	5.291E-01	5.729E-01	4.306E-01
C-43	N,A	1.813E-03	1.886E-03	1.822E-03	1.790E-03	1.884E-03
C-43	N,P	7.333E-04	7.629E-04	7.370E-04	7.238E-04	7.618E-04
C-44	N,G	8.790E-02	6.987E-02	7.772E-02	8.478E-02	6.186E-02
C-44	N,2N	2.482E-08	2.582E-08	2.494E-08	2.449E-08	2.578E-08
C-44	N,A	7.544E-06	7.849E-06	7.583E-06	7.446E-06	7.838E-06
C-44	N,P	1.725E-05	1.794E-05	1.733E-05	1.702E-05	1.792E-05
C-46	N,G	5.990E-02	4.728E-02	5.273E-02	5.767E-02	4.163E-02
C-48	N,G	9.395E-02	7.412E-02	8.267E-02	9.044E-02	6.524E-02
Sc-45	N,G	1.434E+00	1.130E+00	1.261E+00	1.380E+00	9.931E-01
Sc-45	N,GX	8.149E-01	6.418E-01	7.163E-01	7.841E-01	5.642E-01
Sc-45	N,A	8.028E-06	8.351E-06	8.069E-06	7.924E-06	8.340E-06
Sc-45	N,P	1.787E-05	1.859E-05	1.796E-05	1.764E-05	1.856E-05
Sc-46	N,G	6.833E-01	5.390E-01	6.012E-01	6.578E-01	4.745E-01
Ti-46	N,G	5.419E-02	4.337E-02	4.812E-02	5.236E-02	3.860E-02
Ti-46	N,A	1.428E-04	1.485E-04	1.435E-04	1.409E-04	1.483E-04
Ti-46	N,P	2.771E-03	2.883E-03	2.785E-03	2.735E-03	2.879E-03
Ti-47	N,G	1.687E-01	1.380E-01	1.519E-01	1.639E-01	1.248E-01
Ti-47	N,A	1.187E-03	1.235E-03	1.193E-03	1.171E-03	1.233E-03
Ti-47	N,P	4.748E-03	4.939E-03	4.772E-03	4.686E-03	4.932E-03
Ti-48	N,G	6.705E-01	5.298E-01	5.906E-01	6.457E-01	4.670E-01
Ti-48	N,A	2.385E-06	2.481E-06	2.397E-06	2.354E-06	2.477E-06
Ti-48	N,P	4.530E-05	4.712E-05	4.553E-05	4.471E-05	4.706E-05
Ti-49	N,G	1.995E-01	1.598E-01	1.772E-01	1.928E-01	1.423E-01

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Ti-49	N,A	1.745E-04	1.815E-04	1.754E-04	1.722E-04	1.813E-04
Ti-49	N,P	4.966E-04	5.166E-04	4.991E-04	4.901E-04	5.159E-04
Ti-50	N,G	1.614E-02	1.291E-02	1.432E-02	1.559E-02	1.149E-02
Ti-50	N,A	8.701E-08	9.052E-08	8.745E-08	8.588E-08	9.040E-08
Ti-50	N,P	1.079E-05	1.122E-05	1.084E-05	1.065E-05	1.121E-05
V-50	N,G	6.784E+00	5.520E+00	6.087E+00	6.582E+00	4.975E+00
V-51	N,G	4.282E-01	3.402E-01	3.785E-01	4.130E-01	3.011E-01
V-51	N,A	7.631E-07	7.938E-07	7.670E-07	7.532E-07	7.928E-07
V-51	N,P	2.169E-05	2.256E-05	2.180E-05	2.141E-05	2.253E-05
Cr-50	N,G	1.368E+00	1.081E+00	1.205E+00	1.318E+00	9.534E-01
Cr-50	N,A	1.403E-04	1.459E-04	1.410E-04	1.385E-04	1.457E-04
Cr-50	N,P	5.471E-03	5.691E-03	5.498E-03	5.400E-03	5.683E-03
Cr-51	N,G	6.833E-02	5.390E-02	6.012E-02	6.578E-02	4.745E-02
Cr-52	N,G	1.859E-01	1.495E-01	1.655E-01	1.795E-01	1.336E-01
Cr-52	N,2N	2.797E-05	2.906E-05	2.410E-05	2.513E-05	2.496E-05
Cr-52	N,A	2.566E-05	2.670E-05	2.236E-05	2.323E-05	2.318E-05
Cr-52	N,P	4.607E-04	4.799E-04	4.066E-04	4.208E-04	4.219E-04
Cr-53	N,G	1.569E+00	1.241E+00	1.383E+00	1.512E+00	1.095E+00
Cr-53	N,A	6.476E-04	6.737E-04	6.509E-04	6.392E-04	6.728E-04
Cr-53	N,P	2.141E-05	2.227E-05	2.152E-05	2.113E-05	2.224E-05
Cr-54	N,G	3.116E-02	2.466E-02	2.748E-02	3.002E-02	2.177E-02
Cr-54	N,A	9.927E-06	1.033E-05	9.977E-06	9.798E-06	1.031E-05
Cr-54	N,P	8.632E-07	8.980E-07	8.676E-07	8.520E-07	8.968E-07
Mn-54	N,G	8.541E-01	6.738E-01	7.516E-01	8.222E-01	5.931E-01
Mn-55	N,G	1.442E+00	1.212E+00	1.317E+00	1.401E+00	1.114E+00
Mn-55	N,A	2.899E-05	3.016E-05	2.515E-05	2.616E-05	2.607E-05
Mn-55	N,P	3.420E-04	3.562E-04	3.003E-04	3.113E-04	3.116E-04
Fe-54	N,G	1.964E-01	1.558E-01	1.735E-01	1.894E-01	1.378E-01
Fe-54	N,A	5.173E-04	5.381E-04	5.199E-04	5.106E-04	5.374E-04
Fe-54	N,P	1.426E-02	1.484E-02	1.434E-02	1.408E-02	1.482E-02
Fe-56	N,G	2.290E-01	1.833E-01	2.034E-01	2.211E-01	1.633E-01
Fe-56	N,A	8.350E-05	8.688E-05	7.273E-05	7.556E-05	7.541E-05
Fe-56	N,P	1.643E-03	1.713E-03	1.466E-03	1.513E-03	1.523E-03
Fe-57	N,G	2.160E-01	1.713E-01	1.907E-01	2.082E-01	1.514E-01
Fe-57	N,A	7.549E-04	7.853E-04	7.587E-04	7.451E-04	7.843E-04
Fe-57	N,P	8.669E-05	9.018E-05	8.713E-05	8.556E-05	9.006E-05
Fe-58	N,G	1.135E-01	9.271E-02	1.021E-01	1.102E-01	8.381E-02
Fe-58	N,A	2.158E-06	2.245E-06	2.169E-06	2.130E-06	2.242E-06
Fe-58	N,P	2.158E-06	2.245E-06	2.169E-06	2.130E-06	2.242E-06
Co-58	N,G	2.976E+02	2.635E+02	2.820E+02	2.953E+02	2.517E+02
Co-58m	N,G	1.162E+04	9.163E+03	1.022E+04	1.118E+04	8.066E+03
Co-59	N,G	4.919E+00	4.289E+00	4.529E+00	4.773E+00	3.969E+00
Co-59	N,GX	2.430E+00	2.068E+00	2.244E+00	2.385E+00	1.924E+00
Co-59	N,A	3.965E-05	4.124E-05	3.442E-05	3.579E-05	3.567E-05
Co-59	N,P	4.367E-04	4.551E-04	3.875E-04	4.004E-04	4.023E-04
Co-60	N,G	2.479E-01	2.117E-01	2.295E-01	2.436E-01	1.975E-01
Co-60m	N,G	9.578E+00	8.524E+00	9.106E+00	9.518E+00	8.170E+00

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Ni-58	N,G	3.631E-01	2.889E-01	3.211E-01	3.500E-01	2.558E-01
Ni-58	N,2N	2.327E-06	2.418E-06	2.005E-06	2.090E-06	2.077E-06
Ni-58	N,A	1.341E-03	1.397E-03	1.181E-03	1.223E-03	1.226E-03
Ni-58	N,P	2.117E-02	2.208E-02	1.904E-02	1.960E-02	1.978E-02
Ni-59	N,G	1.005E+01	8.386E+00	9.163E+00	9.814E+00	7.698E+00
Ni-59	N,A	1.025E+00	8.085E-01	9.019E-01	9.866E-01	7.117E-01
Ni-60	N,G	2.446E-01	1.941E-01	2.160E-01	2.358E-01	1.716E-01
Ni-60	N,A	1.728E-04	1.798E-04	1.737E-04	1.706E-04	1.795E-04
Ni-60	N,P	4.314E-04	4.488E-04	4.336E-04	4.258E-04	4.482E-04
Ni-61	N,G	2.243E-01	1.792E-01	1.989E-01	2.166E-01	1.593E-01
Ni-61	N,A	4.823E-03	4.005E-03	4.344E-03	4.662E-03	3.624E-03
Ni-61	N,P	3.671E-04	3.819E-04	3.689E-04	3.623E-04	3.814E-04
Ni-62	N,G	1.222E+00	9.660E-01	1.077E+00	1.177E+00	8.517E-01
Ni-62	N,A	1.403E-05	1.459E-05	1.410E-05	1.385E-05	1.458E-05
Ni-62	N,P	2.158E-05	2.245E-05	2.169E-05	2.130E-05	2.242E-05
Ni-63	N,G	1.964E+00	1.550E+00	1.729E+00	1.891E+00	1.364E+00
Ni-64	N,G	1.370E-01	1.101E-01	1.220E-01	1.325E-01	9.834E-02
Ni-65	N,G	2.077E+00	1.639E+00	1.828E+00	1.999E+00	1.443E+00
Cu-63	N,G	4.496E-01	3.683E-01	4.051E-01	4.370E-01	3.336E-01
Cu-63	N,A	1.552E-04	1.615E-04	1.560E-04	1.532E-04	1.613E-04
Cu-63	N,P	1.942E-03	2.021E-03	1.952E-03	1.917E-03	2.018E-03
Cu-65	N,G	2.176E-01	1.784E-01	1.962E-01	2.116E-01	1.617E-01
Cu-65	N,2N	1.530E-04	1.591E-04	1.537E-04	1.510E-04	1.589E-04
Cu-65	N,A	3.239E-05	3.370E-05	3.256E-05	3.197E-05	3.365E-05
Cu-65	N,P	7.793E-05	8.107E-05	7.832E-05	7.691E-05	8.096E-05
Cu-66	N,G	1.153E+01	9.096E+00	1.015E+01	1.110E+01	8.007E+00
Zn-64	N,G	9.041E-02	7.630E-02	8.304E-02	8.856E-02	7.060E-02
Zn-64	N,A	1.266E-06	9.985E-07	1.114E-06	1.219E-06	8.790E-07
Zn-66	N,G	8.207E-02	6.672E-02	7.360E-02	7.961E-02	6.010E-02
Zn-66	N,A	1.742E-06	1.375E-06	1.533E-06	1.677E-06	1.210E-06
Zn-67	N,G	9.725E-01	8.474E-01	9.119E-01	9.608E-01	8.012E-01
Zn-67	N,A	5.127E-07	4.045E-07	4.512E-07	4.936E-07	3.560E-07
Zn-68	N,G	1.450E-01	1.269E-01	1.363E-01	1.434E-01	1.203E-01
Zn-68	N,GX	1.045E-02	9.142E-03	9.823E-03	1.033E-02	8.666E-03
Zn-68	N,A	1.740E-06	1.372E-06	1.531E-06	1.675E-06	1.208E-06
Zn-70	N,G	7.089E-03	5.592E-03	6.238E-03	6.824E-03	4.923E-03
Zn-70	N,GX	7.432E-04	5.863E-04	6.540E-04	7.155E-04	5.161E-04
Ga-69	N,G	4.802E-01	4.493E-01	4.718E-01	4.838E-01	4.440E-01
Ga-71	N,G	1.040E+00	9.540E-01	1.009E+00	1.042E+00	9.317E-01
Ga-71	N,GX	3.316E-02	3.042E-02	3.216E-02	3.323E-02	2.971E-02
Ge-70	N,G	2.869E-01	2.300E-01	2.551E-01	2.773E-01	2.051E-01
Ge-70	N,GX	2.549E-02	2.044E-02	2.266E-02	2.464E-02	1.822E-02
Ge-72	N,G	1.189E-01	1.027E-01	1.101E-01	1.159E-01	9.597E-02
Ge-73	N,G	3.247E+00	3.029E+00	3.129E+00	3.194E+00	2.936E+00
Ge-74	N,G	5.366E-02	4.760E-02	5.067E-02	5.261E-02	4.543E-02
Ge-74	N,GX	1.440E-02	1.181E-02	1.299E-02	1.400E-02	1.071E-02
Ge-75	N,G	3.843E-01	3.032E-01	3.382E-01	3.700E-01	2.669E-01

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Ge-76	N,G	5.684E-02	5.578E-02	5.754E-02	5.691E-02	5.671E-02
Ge-76	N,GX	3.628E-02	3.458E-02	3.609E-02	3.675E-02	3.454E-02
As-75	N,G	2.164E+00	2.126E+00	2.176E+00	2.167E+00	2.144E+00
Se-74	N,G	1.671E+01	1.575E+01	1.650E+01	1.687E+01	1.564E+01
Se-76	N,G	7.484E+00	5.959E+00	6.623E+00	7.217E+00	5.282E+00
Se-76	N,GX	1.826E+00	1.448E+00	1.612E+00	1.760E+00	1.279E+00
Se-77	N,G	4.201E+00	3.463E+00	3.796E+00	4.072E+00	3.148E+00
Se-78	N,G	1.870E-01	1.848E-01	1.881E-01	1.863E-01	1.864E-01
Se-78	N,GX	1.133E-01	1.072E-01	1.121E-01	1.145E-01	1.066E-01
Se-79	N,G	3.371E-01	2.650E-01	2.960E-01	3.242E-01	2.326E-01
Se-80	N,G	8.774E-02	7.814E-02	8.328E-02	8.620E-02	7.502E-02
Se-80	N,GX	1.101E-02	9.556E-03	1.030E-02	1.086E-02	9.014E-03
Se-82	N,G	8.457E-03	7.837E-03	8.051E-03	8.266E-03	7.511E-03
Se-82	N,GX	4.955E-04	3.909E-04	4.361E-04	4.770E-04	3.441E-04
Br-79	N,G	4.794E+00	4.653E+00	4.786E+00	4.796E+00	4.656E+00
Br-79	N,GX	9.780E-01	9.298E-01	9.713E-01	9.900E-01	9.275E-01
Br-81	N,G	1.807E+00	1.799E+00	1.844E+00	1.819E+00	1.838E+00
Br-81	N,GX	1.228E+00	1.182E+00	1.230E+00	1.247E+00	1.188E+00
Kr-78	N,G	4.533E-01	3.720E-01	4.089E-01	4.408E-01	3.374E-01
Kr-78	N,GX	2.116E-02	1.737E-02	1.909E-02	2.058E-02	1.575E-02
Kr-80	N,G	2.787E+00	2.569E+00	2.583E+00	2.684E+00	2.386E+00
Kr-80	N,GX	7.556E-01	6.729E-01	7.187E-01	7.510E-01	6.453E-01
Kr-80	N,2N	4.316E-05	4.490E-05	4.338E-05	4.260E-05	4.484E-05
Kr-80	N,A	8.126E-06	8.464E-06	7.347E-06	7.541E-06	7.624E-06
Kr-80	N,P	1.023E-04	1.065E-04	8.946E-05	9.283E-05	9.278E-05
Kr-82	N,G	7.185E+00	6.708E+00	6.924E+00	7.097E+00	6.505E+00
Kr-82	N,GX	3.520E+00	3.156E+00	3.363E+00	3.505E+00	3.039E+00
Kr-82	N,2N	8.416E-05	8.755E-05	8.459E-05	8.307E-05	8.744E-05
Kr-82	N,A	2.422E-06	2.518E-06	2.096E-06	2.182E-06	2.172E-06
Kr-82	N,P	2.089E-05	2.174E-05	1.821E-05	1.892E-05	1.888E-05
Kr-83	N,G	2.025E+01	1.664E+01	1.820E+01	1.959E+01	1.503E+01
Kr-83	N,2N	1.446E-03	1.504E-03	1.453E-03	1.427E-03	1.502E-03
Kr-83	N,A	2.762E-06	2.872E-06	2.393E-06	2.491E-06	2.481E-06
Kr-83	N,P	7.258E-05	7.560E-05	6.391E-05	6.619E-05	6.632E-05
Kr-84	N,G	1.275E-01	1.301E-01	1.330E-01	1.289E-01	1.356E-01
Kr-84	N,GX	4.852E-02	4.683E-02	4.868E-02	4.933E-02	4.711E-02
Kr-84	N,2N	1.338E-04	1.392E-04	1.345E-04	1.321E-04	1.390E-04
Kr-84	N,A	1.618E-05	1.687E-05	1.503E-05	1.529E-05	1.561E-05
Kr-84	N,P	2.154E-06	2.239E-06	1.863E-06	1.940E-06	1.930E-06
Kr-85	N,G	1.770E-01	1.484E-01	1.622E-01	1.724E-01	1.373E-01
Kr-86	N,G	9.914E-03	8.988E-03	9.610E-03	9.824E-03	8.828E-03
Kr-86	N,2N	1.834E-04	1.908E-04	1.844E-04	1.810E-04	1.906E-04
Kr-86	N,A	1.012E-05	1.056E-05	9.402E-06	9.569E-06	9.766E-06
Kr-86	N,P	4.573E-08	4.754E-08	3.953E-08	4.116E-08	4.095E-08
Kr-87	N,G	5.125E+01	4.043E+01	4.509E+01	4.933E+01	3.559E+01
Rb-85	N,G	2.545E-01	2.539E-01	2.595E-01	2.548E-01	2.596E-01
Rb-85	N,GX	2.225E-02	2.132E-02	2.221E-02	2.257E-02	2.136E-02

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Rb-86	N,G	1.133E+00	1.066E+00	1.104E+00	1.119E+00	1.044E+00
Rb-87	N,G	7.784E-02	7.774E-02	7.859E-02	7.756E-02	7.850E-02
Rb-88	N,G	8.541E-02	6.738E-02	7.516E-02	8.222E-02	5.931E-02
Sr-84	N,G	9.673E-02	9.191E-02	9.604E-02	9.791E-02	9.166E-02
Sr-84	N,GX	2.046E-01	1.944E-01	2.031E-01	2.071E-01	1.938E-01
Sr-86	N,G	3.811E-01	3.346E-01	3.587E-01	3.737E-01	3.181E-01
Sr-86	N,GX	1.539E-01	1.386E-01	1.475E-01	1.534E-01	1.339E-01
Sr-87	N,G	4.159E+00	3.857E+00	4.060E+00	4.194E+00	3.788E+00
Sr-88	N,G	1.212E-03	1.138E-03	1.143E-03	1.178E-03	1.078E-03
Sr-89	N,G	5.092E-02	4.394E-02	4.725E-02	4.968E-02	4.118E-02
Sr-90	N,G	8.324E-02	6.728E-02	7.411E-02	8.032E-02	6.007E-02
Y-89	N,G	1.275E-01	1.051E-01	1.154E-01	1.237E-01	9.581E-02
Y-89	N,GX	9.289E-05	7.485E-05	8.284E-05	8.991E-05	6.697E-05
Y-90	N,G	4.195E-01	3.607E-01	3.897E-01	4.097E-01	3.384E-01
Y-90	N,GX	1.475E-01	1.159E-01	1.295E-01	1.418E-01	1.018E-01
Y-91	N,G	1.594E-01	1.357E-01	1.474E-01	1.556E-01	1.267E-01
Zr-90	N,G	2.480E-02	2.366E-02	2.425E-02	2.447E-02	2.329E-02
Zr-90	N,A	7.803E-07	8.118E-07	7.843E-07	7.702E-07	8.107E-07
Zr-90	N,P	1.943E-04	2.021E-04	1.953E-04	1.918E-04	2.019E-04
Zr-91	N,G	2.748E-01	2.624E-01	2.717E-01	2.728E-01	2.611E-01
Zr-91	N,A	3.239E-04	3.370E-04	3.256E-04	3.197E-04	3.365E-04
Zr-91	N,P	6.032E-05	6.275E-05	6.062E-05	5.953E-05	6.266E-05
Zr-92	N,G	1.106E-01	1.048E-01	1.093E-01	1.099E-01	1.045E-01
Zr-92	N,2N	1.923E-04	1.998E-04	1.658E-04	1.728E-04	1.717E-04
Zr-92	N,A	1.596E-05	1.660E-05	1.604E-05	1.575E-05	1.658E-05
Zr-92	N,P	1.025E-04	1.069E-04	9.129E-05	9.426E-05	9.478E-05
Zr-93	N,G	1.034E+00	1.010E+00	1.005E+00	1.013E+00	9.822E-01
Zr-93	N,A	1.079E-05	1.122E-05	1.084E-05	1.065E-05	1.121E-05
Zr-94	N,G	1.946E-02	1.903E-02	1.952E-02	1.937E-02	1.920E-02
Zr-94	N,A	3.019E-06	3.141E-06	3.034E-06	2.980E-06	3.137E-06
Zr-94	N,P	2.160E-06	2.247E-06	2.171E-06	2.132E-06	2.244E-06
Zr-95	N,G	2.343E-01	2.324E-01	2.387E-01	2.349E-01	2.376E-01
Zr-96	N,G	1.805E-01	1.860E-01	1.857E-01	1.808E-01	1.906E-01
Zr-96	N,A	6.480E-08	6.742E-08	6.514E-08	6.396E-08	6.733E-08
Zr-96	N,P	8.626E-08	8.973E-08	8.669E-08	8.514E-08	8.961E-08
Nb-93	N,G	4.115E-01	4.011E-01	4.131E-01	4.105E-01	4.047E-01
Nb-93	N,GX	9.395E-02	7.412E-02	8.267E-02	9.044E-02	6.524E-02
Nb-93	N,2N	3.237E-04	3.367E-04	3.253E-04	3.195E-04	3.363E-04
Nb-93	N,A	1.672E-05	1.740E-05	1.460E-05	1.516E-05	1.514E-05
Nb-93	N,P	1.587E-04	1.653E-04	1.393E-04	1.444E-04	1.446E-04
Nb-94	N,G	4.222E+00	3.981E+00	4.279E+00	4.311E+00	4.066E+00
Nb-95	N,G	8.591E-01	8.548E-01	8.666E-01	8.559E-01	8.632E-01
Mo-92	N,G	1.518E-02	1.435E-02	1.502E-02	1.534E-02	1.426E-02
Mo-92	N,GX	5.125E-04	4.043E-04	4.509E-04	4.933E-04	3.559E-04
Mo-92	N,A	6.722E-06	6.993E-06	6.756E-06	6.635E-06	6.984E-06
Mo-92	N,P	1.338E-03	1.392E-03	1.345E-03	1.320E-03	1.390E-03
Mo-94	N,G	4.015E-02	4.136E-02	4.198E-02	4.058E-02	4.326E-02

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Mo-94	N,A	1.424E-04	1.482E-04	1.432E-04	1.406E-04	1.480E-04
Mo-94	N,P	1.791E-04	1.863E-04	1.800E-04	1.768E-04	1.861E-04
Mo-95	N,G	4.204E+00	3.991E+00	4.098E+00	4.166E+00	3.912E+00
Mo-95	N,A	7.985E-04	8.306E-04	8.025E-04	7.881E-04	8.295E-04
Mo-95	N,P	2.810E-05	2.923E-05	2.824E-05	2.773E-05	2.919E-05
Mo-96	N,G	6.928E-01	6.902E-01	6.873E-01	6.839E-01	6.832E-01
Mo-96	N,A	6.472E-05	6.733E-05	6.505E-05	6.388E-05	6.724E-05
Mo-96	N,P	4.102E-05	4.268E-05	4.123E-05	4.049E-05	4.262E-05
Mo-97	N,G	6.960E-01	6.722E-01	7.019E-01	6.992E-01	6.827E-01
Mo-97	N,A	2.805E-04	2.918E-04	2.820E-04	2.769E-04	2.915E-04
Mo-97	N,P	2.808E-05	2.921E-05	2.822E-05	2.771E-05	2.917E-05
Mo-98	N,G	1.297E+00	1.270E+00	1.298E+00	1.296E+00	1.275E+00
Mo-98	N,2N	4.230E-04	4.395E-04	3.645E-04	3.800E-04	3.775E-04
Mo-98	N,A	1.424E-05	1.482E-05	1.432E-05	1.406E-05	1.480E-05
Mo-98	N,P	1.725E-06	1.794E-06	1.733E-06	1.702E-06	1.792E-06
Mo-99	N,G	1.021E+00	1.013E+00	1.038E+00	1.026E+00	1.032E+00
Mo-100	N,G	1.504E-01	1.513E-01	1.546E-01	1.512E-01	1.558E-01
Mo-100	N,A	8.205E-07	8.535E-07	8.247E-07	8.098E-07	8.524E-07
Mo-100	N,P	2.155E-07	2.241E-07	2.166E-07	2.127E-07	2.238E-07
Tc-99	N,G	9.323E+00	8.997E+00	8.898E+00	9.180E+00	8.606E+00
Tc-99	N,2N	4.316E-04	4.490E-04	4.338E-04	4.260E-04	4.484E-04
Ru-96	N,G	1.685E-01	1.637E-01	1.698E-01	1.716E-01	1.653E-01
Ru-98	N,G	6.833E-01	5.390E-01	6.012E-01	6.578E-01	4.745E-01
Ru-99	N,G	4.261E+00	4.195E+00	4.343E+00	4.320E+00	4.277E+00
Ru-100	N,G	7.579E-01	6.622E-01	7.101E-01	7.420E-01	6.264E-01
Ru-101	N,G	2.978E+00	2.963E+00	3.104E+00	3.060E+00	3.094E+00
Ru-102	N,G	2.587E-01	2.409E-01	2.517E-01	2.556E-01	2.367E-01
Ru-103	N,G	2.662E+00	2.560E+00	2.656E+00	2.666E+00	2.567E+00
Ru-104	N,G	2.647E-01	2.642E-01	2.689E-01	2.647E-01	2.689E-01
Ru-105	N,G	2.962E-01	3.026E-01	3.066E-01	2.977E-01	3.130E-01
Ru-106	N,G	9.014E-02	9.049E-02	9.425E-02	9.148E-02	9.511E-02
Rh-103	N,G	3.828E+01	3.505E+01	3.739E+01	3.873E+01	3.449E+01
Rh-103	N,GX	2.655E+00	2.454E+00	2.588E+00	2.666E+00	2.408E+00
Rh-103	N,2N	1.575E-04	1.639E-04	1.583E-04	1.555E-04	1.637E-04
Rh-104	N,G	3.416E+00	2.695E+00	3.006E+00	3.289E+00	2.372E+00
Rh-104m	N,G	6.833E+01	5.390E+01	6.012E+01	6.578E+01	4.745E+01
Rh-105	N,G	9.848E+02	8.187E+02	8.933E+02	9.582E+02	7.510E+02
Rh-105	N,GX	4.588E+02	3.739E+02	4.121E+02	4.454E+02	3.374E+02
Pd-102	N,G	4.100E-01	3.234E-01	3.607E-01	3.947E-01	2.847E-01
Pd-104	N,G	6.567E-01	6.686E-01	6.712E-01	6.566E-01	6.815E-01
Pd-105	N,G	3.785E+00	3.568E+00	3.784E+00	3.817E+00	3.594E+00
Pd-106	N,G	2.791E-01	2.837E-01	2.863E-01	2.797E-01	2.908E-01
Pd-106	N,GX	6.513E-03	6.269E-03	6.523E-03	6.617E-03	6.298E-03
Pd-107	N,G	2.794E+00	2.648E+00	2.745E+00	2.780E+00	2.617E+00
Pd-108	N,G	7.078E+00	6.916E+00	7.120E+00	7.125E+00	6.965E+00
Pd-108	N,GX	1.080E-01	1.043E-01	1.084E-01	1.098E-01	1.049E-01
Pd-110	N,G	2.640E-01	2.684E-01	2.743E-01	2.665E-01	2.790E-01

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Pd-110	N,GX	3.688E-02	3.646E-02	3.761E-02	3.777E-02	3.717E-02
Ag-107	N,G	6.114E+00	5.494E+00	5.901E+00	6.087E+00	5.357E+00
Ag-107	N,GX	3.973E-01	3.430E-01	3.703E-01	3.915E-01	3.223E-01
Ag-107	N,2N	3.237E-04	3.367E-04	3.253E-04	3.195E-04	3.363E-04
Ag-107	N,A	2.034E-05	2.114E-05	1.758E-05	1.830E-05	1.821E-05
Ag-107	N,P	1.933E-05	2.012E-05	1.691E-05	1.755E-05	1.754E-05
Ag-109	N,G	3.944E+01	3.778E+01	3.784E+01	3.912E+01	3.637E+01
Ag-109	N,GX	1.921E+00	1.838E+00	1.916E+00	1.948E+00	1.839E+00
Ag-109	N,2N	3.453E-04	3.592E-04	3.470E-04	3.408E-04	3.587E-04
Ag-109	N,A	1.864E-05	1.940E-05	1.619E-05	1.684E-05	1.679E-05
Ag-109	N,P	2.148E-05	2.237E-05	1.882E-05	1.952E-05	1.952E-05
Ag-110	N,G	7.004E+00	5.525E+00	6.163E+00	6.742E+00	4.863E+00
Ag-110	N,GX	8.541E-22	6.738E-22	7.516E-22	8.222E-22	5.931E-22
Ag-110m	N,G	7.004E+00	5.525E+00	6.163E+00	6.742E+00	4.863E+00
Ag-110m	N,GX	7.004E+00	5.525E+00	6.163E+00	6.742E+00	4.863E+00
Ag-111	N,G	3.513E+00	3.540E+00	3.624E+00	3.550E+00	3.651E+00
Cd-106	N,G	8.541E-02	6.738E-02	7.516E-02	8.222E-02	5.931E-02
Cd-108	N,G	2.619E-01	2.485E-01	2.566E-01	2.590E-01	2.453E-01
Cd-109	N,G	5.552E+01	4.380E+01	4.885E+01	5.344E+01	3.855E+01
Cd-110	N,G	2.210E+00	2.039E+00	2.206E+00	2.231E+00	2.062E+00
Cd-110	N,GX	1.578E-02	1.397E-02	1.495E-02	1.566E-02	1.334E-02
Cd-111	N,G	3.454E+00	3.044E+00	3.252E+00	3.393E+00	2.890E+00
Cd-112	N,G	6.333E-01	6.068E-01	6.342E-01	6.351E-01	6.123E-01
Cd-112	N,GX	2.971E-01	2.960E-01	3.046E-01	3.049E-01	3.031E-01
Cd-113	N,G	3.891E+03	3.015E+03	3.422E+03	3.779E+03	2.625E+03
Cd-113	N,2N	1.036E-03	1.078E-03	1.041E-03	1.022E-03	1.076E-03
Cd-113	N,P	1.812E-06	1.882E-06	1.562E-06	1.628E-06	1.618E-06
Cd-114	N,G	6.570E-01	6.702E-01	6.576E-01	6.482E-01	6.693E-01
Cd-114	N,GX	5.129E-02	5.055E-02	5.219E-02	5.247E-02	5.145E-02
Cd-115m	N,G	8.181E+00	7.742E+00	8.068E+00	8.151E+00	7.682E+00
Cd-116	N,G	1.060E-01	1.082E-01	1.096E-01	1.067E-01	1.119E-01
Cd-116	N,GX	3.700E-03	3.211E-03	3.460E-03	3.652E-03	3.028E-03
In-113	N,G	6.976E+00	6.809E+00	7.098E+00	7.119E+00	6.949E+00
In-113	N,GX	4.771E+00	4.629E+00	4.804E+00	4.858E+00	4.670E+00
In-115	N,G	9.646E+01	9.207E+01	9.726E+01	9.910E+01	9.342E+01
In-115	N,GX	6.998E+01	6.705E+01	6.987E+01	7.100E+01	6.718E+01
In-117	N,G	4.270E-01	3.369E-01	3.758E-01	4.111E-01	2.965E-01
In-117	N,GX	4.270E-01	3.369E-01	3.758E-01	4.111E-01	2.965E-01
In-117m	N,G	4.270E-01	3.369E-01	3.758E-01	4.111E-01	2.965E-01
In-117m	N,GX	4.270E-01	3.369E-01	3.758E-01	4.111E-01	2.965E-01
In-119	N,G	3.416E-02	2.695E-02	3.006E-02	3.289E-02	2.372E-02
In-119	N,GX	5.125E-02	4.043E-02	4.509E-02	4.933E-02	3.559E-02
In-119m	N,G	3.416E-02	2.695E-02	3.006E-02	3.289E-02	2.372E-02
In-119m	N,GX	5.125E-02	4.043E-02	4.509E-02	4.933E-02	3.559E-02
In-120	N,G	1.361E-03	1.358E-03	1.397E-03	1.397E-03	1.392E-03
In-120m	N,G	1.361E-03	1.358E-03	1.397E-03	1.397E-03	1.392E-03
Sn-112	N,G	4.861E-01	4.709E-01	4.889E-01	4.948E-01	4.748E-01

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Sn-112	N,GX	2.127E-01	2.060E-01	2.139E-01	2.165E-01	2.077E-01
Sn-114	N,G	3.474E-02	3.289E-02	3.440E-02	3.512E-02	3.273E-02
Sn-114	N,A	4.130E-06	4.297E-06	4.151E-06	4.077E-06	4.291E-06
Sn-114	N,P	6.472E-05	6.733E-05	6.505E-05	6.388E-05	6.724E-05
Sn-115	N,G	4.319E+00	3.419E+00	3.809E+00	4.161E+00	3.017E+00
Sn-115	N,A	6.265E-05	6.517E-05	6.297E-05	6.183E-05	6.509E-05
Sn-115	N,P	6.468E-05	6.728E-05	6.500E-05	6.384E-05	6.719E-05
Sn-116	N,G	3.523E-01	3.591E-01	3.399E-01	3.403E-01	3.448E-01
Sn-116	N,GX	2.499E-01	2.494E-01	2.565E-01	2.566E-01	2.555E-01
Sn-116	N,A	4.497E-07	4.679E-07	4.520E-07	4.439E-07	4.672E-07
Sn-116	N,P	1.079E-05	1.123E-05	1.085E-05	1.065E-05	1.121E-05
Sn-117	N,G	7.608E-01	7.271E-01	7.447E-01	7.528E-01	7.154E-01
Sn-117	N,A	1.165E-05	1.212E-05	1.171E-05	1.150E-05	1.211E-05
Sn-117	N,P	1.942E-05	2.020E-05	1.952E-05	1.917E-05	2.018E-05
Sn-118	N,G	2.305E-01	2.360E-01	2.361E-01	2.309E-01	2.413E-01
Sn-118	N,GX	1.600E-01	1.594E-01	1.640E-01	1.642E-01	1.632E-01
Sn-118	N,A	4.402E-08	4.580E-08	4.425E-08	4.345E-08	4.574E-08
Sn-118	N,P	2.157E-06	2.244E-06	2.168E-06	2.129E-06	2.241E-06
Sn-119	N,G	2.934E-01	2.550E-01	2.735E-01	2.868E-01	2.399E-01
Sn-119	N,A	1.877E-06	1.952E-06	1.886E-06	1.852E-06	1.950E-06
Sn-119	N,P	4.316E-06	4.490E-06	4.338E-06	4.260E-06	4.484E-06
Sn-120	N,G	5.897E-02	5.814E-02	5.967E-02	5.900E-02	5.910E-02
Sn-120	N,GX	3.168E-04	2.984E-04	3.127E-04	3.198E-04	2.960E-04
Sn-120	N,A	4.316E-07	4.490E-07	4.338E-07	4.260E-07	4.484E-07
Sn-122	N,G	4.182E-02	3.945E-02	4.017E-02	4.095E-02	3.810E-02
Sn-122	N,GX	1.497E-04	1.316E-04	1.412E-04	1.483E-04	1.252E-04
Sn-122	N,A	8.632E-08	8.980E-08	8.676E-08	8.520E-08	8.968E-08
Sn-123	N,G	1.104E-01	1.139E-01	1.155E-01	1.116E-01	1.190E-01
Sn-124	N,G	2.269E-01	2.294E-01	2.404E-01	2.341E-01	2.434E-01
Sn-124	N,GX	1.616E-01	1.590E-01	1.642E-01	1.652E-01	1.617E-01
Sn-124	N,A	2.158E-08	2.245E-08	2.169E-08	2.130E-08	2.242E-08
Sn-125	N,G	5.539E-01	5.620E-01	5.833E-01	5.636E-01	5.932E-01
Sb-121	N,G	5.539E+00	5.459E+00	5.536E+00	5.583E+00	5.464E+00
Sb-121	N,GX	4.406E-02	4.299E-02	4.453E-02	4.494E-02	4.352E-02
Sb-123	N,G	3.331E+00	3.278E+00	3.166E+00	3.231E+00	3.111E+00
Sb-123	N,GX	8.751E-03	8.542E-03	8.847E-03	8.927E-03	8.648E-03
Sb-124	N,G	1.359E+00	1.265E+00	1.328E+00	1.348E+00	1.247E+00
Sb-125	N,G	6.994E-01	7.002E-01	7.072E-01	6.973E-01	7.083E-01
Sb-125	N,GX	6.569E-02	5.163E-02	5.767E-02	6.318E-02	4.531E-02
Sb-126	N,G	1.871E+00	1.801E+00	1.872E+00	1.873E+00	1.812E+00
Te-120	N,G	1.708E-01	1.348E-01	1.503E-01	1.644E-01	1.186E-01
Te-120	N,GX	2.904E-02	2.291E-02	2.555E-02	2.795E-02	2.017E-02
Te-122	N,G	2.472E+00	2.474E+00	2.591E+00	2.537E+00	2.603E+00
Te-122	N,GX	7.956E-01	7.745E-01	8.028E-01	8.109E-01	7.829E-01
Te-123	N,G	1.689E+02	1.610E+02	1.693E+02	1.725E+02	1.621E+02
Te-124	N,G	8.081E-01	6.940E-01	7.474E-01	7.879E-01	6.477E-01
Te-124	N,GX	3.938E-03	3.216E-03	3.542E-03	3.824E-03	2.906E-03

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Te-125	N,G	9.065E-01	9.008E-01	9.177E-01	9.062E-01	9.130E-01
Te-126	N,G	4.285E-01	4.195E-01	4.262E-01	4.250E-01	4.178E-01
Te-126	N,GX	3.973E-02	3.724E-02	3.909E-02	4.006E-02	3.685E-02
Te-127m	N,G	2.033E+00	1.894E+00	1.990E+00	2.021E+00	1.869E+00
Te-128	N,G	1.158E-01	1.155E-01	1.167E-01	1.153E-01	1.166E-01
Te-128	N,GX	3.506E-03	3.231E-03	3.411E-03	3.518E-03	3.165E-03
Te-129m	N,G	2.892E-01	2.760E-01	2.870E-01	2.877E-01	2.758E-01
Te-130	N,G	3.462E-02	2.977E-02	3.210E-02	3.378E-02	2.788E-02
Te-130	N,GX	2.459E-03	2.097E-03	2.274E-03	2.416E-03	1.954E-03
I-125	N,G	3.786E+02	3.620E+02	3.775E+02	3.839E+02	3.622E+02
I-126	N,G	1.369E+03	1.260E+03	1.331E+03	1.373E+03	1.233E+03
I-127	N,G	4.885E+00	4.847E+00	4.920E+00	4.895E+00	4.885E+00
I-129	N,G	3.116E+00	2.653E+00	2.873E+00	3.039E+00	2.468E+00
I-129	N,GX	1.898E+00	1.573E+00	1.723E+00	1.850E+00	1.436E+00
I-130	N,G	6.658E+00	6.430E+00	6.648E+00	6.665E+00	6.446E+00
I-131	N,G	3.257E-01	3.221E-01	3.320E-01	3.268E-01	3.296E-01
Xe-124	N,G	7.558E+01	7.355E+01	7.625E+01	7.703E+01	7.434E+01
Xe-124	N,GX	1.569E+01	1.527E+01	1.583E+01	1.599E+01	1.543E+01
Xe-125	N,G	4.783E+02	3.773E+02	4.209E+02	4.604E+02	3.321E+02
Xe-126	N,G	1.087E+00	1.018E+00	1.069E+00	1.096E+00	1.007E+00
Xe-126	N,GX	7.557E-02	7.079E-02	7.432E-02	7.618E-02	7.001E-02
Xe-128	N,G	6.445E-01	5.921E-01	6.191E-01	6.341E-01	5.730E-01
Xe-128	N,GX	4.538E-02	3.886E-02	4.207E-02	4.462E-02	3.631E-02
Xe-128	N,2N	3.453E-04	3.592E-04	3.470E-04	3.408E-04	3.587E-04
Xe-128	N,A	5.797E-07	6.032E-07	5.045E-07	5.243E-07	5.231E-07
Xe-128	N,P	7.714E-07	8.033E-07	6.798E-07	7.039E-07	7.053E-07
Xe-129	N,G	8.218E+00	7.873E+00	8.401E+00	8.417E+00	8.087E+00
Xe-129	N,2N	3.021E-03	3.143E-03	3.037E-03	2.982E-03	3.139E-03
Xe-129	N,A	4.265E-06	4.444E-06	3.760E-06	3.893E-06	3.902E-06
Xe-129	N,P	2.293E-06	2.388E-06	2.018E-06	2.090E-06	2.094E-06
Xe-130	N,G	5.981E-01	4.891E-01	5.374E-01	5.789E-01	4.418E-01
Xe-130	N,GX	5.245E-02	4.485E-02	4.858E-02	5.156E-02	4.187E-02
Xe-130	N,2N	4.748E-04	4.939E-04	4.772E-04	4.686E-04	4.932E-04
Xe-130	N,A	2.127E-07	2.213E-07	1.846E-07	1.920E-07	1.913E-07
Xe-130	N,P	1.841E-06	1.917E-06	1.621E-06	1.679E-06	1.682E-06
Xe-131	N,G	3.018E+01	2.866E+01	3.086E+01	3.104E+01	2.952E+01
Xe-131	N,2N	4.532E-03	4.715E-03	4.555E-03	4.473E-03	4.708E-03
Xe-131	N,A	1.483E-06	1.544E-06	1.299E-06	1.347E-06	1.347E-06
Xe-131	N,P	2.727E-06	2.841E-06	2.444E-06	2.517E-06	2.536E-06
Xe-132	N,G	1.016E-01	9.584E-02	9.957E-02	1.006E-01	9.464E-02
Xe-132	N,GX	3.057E-03	2.605E-03	2.825E-03	3.002E-03	2.426E-03
Xe-132	N,2N	7.121E-04	7.408E-04	7.158E-04	7.029E-04	7.399E-04
Xe-132	N,A	6.108E-08	6.350E-08	5.282E-08	5.500E-08	5.473E-08
Xe-132	N,P	4.104E-07	4.272E-07	3.596E-07	3.729E-07	3.729E-07
Xe-133	N,G	2.355E+01	2.019E+01	2.178E+01	2.304E+01	1.882E+01
Xe-134	N,G	4.338E-02	3.975E-02	4.178E-02	4.275E-02	3.868E-02
Xe-134	N,GX	3.120E-04	2.578E-04	2.827E-04	3.039E-04	2.350E-04

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Xe-134	N,2N	9.927E-04	1.033E-03	9.977E-04	9.798E-04	1.031E-03
Xe-134	N,A	2.641E-08	2.744E-08	2.277E-08	2.373E-08	2.358E-08
Xe-134	N,P	4.751E-07	4.949E-07	4.178E-07	4.329E-07	4.335E-07
Xe-135	N,G	2.215E+05	1.640E+05	1.825E+05	2.057E+05	1.328E+05
Xe-136	N,G	1.593E-02	1.313E-02	1.441E-02	1.545E-02	1.196E-02
Xe-136	N,2N	1.230E-03	1.280E-03	1.236E-03	1.214E-03	1.278E-03
Xe-136	N,A	1.133E-08	1.177E-08	9.763E-09	1.018E-08	1.011E-08
Xe-136	N,P	2.078E-08	2.161E-08	1.799E-08	1.873E-08	1.865E-08
Cs-133	N,G	1.059E+01	1.010E+01	1.054E+01	1.073E+01	1.009E+01
Cs-133	N,GX	9.995E-01	9.530E-01	9.947E-01	1.013E+00	9.523E-01
Cs-133	N,2N	4.316E-03	4.490E-03	4.338E-03	4.260E-03	4.484E-03
Cs-134	N,G	1.128E+01	8.754E+00	9.823E+00	1.081E+01	7.607E+00
Cs-134	N,GX	5.787E+00	4.579E+00	5.102E+00	5.575E+00	4.040E+00
Cs-134m	N,G	1.110E-01	8.759E-02	9.770E-02	1.069E-01	7.710E-02
Cs-134m	N,GX	1.110E-01	8.759E-02	9.770E-02	1.069E-01	7.710E-02
Cs-135	N,G	2.480E+00	2.320E+00	2.437E+00	2.499E+00	2.293E+00
Cs-136	N,G	1.821E+00	1.795E+00	1.853E+00	1.863E+00	1.827E+00
Cs-137	N,G	2.546E-02	2.409E-02	2.558E-02	2.554E-02	2.450E-02
Ba-130	N,G	3.599E+00	3.396E+00	3.556E+00	3.635E+00	3.373E+00
Ba-130	N,GX	8.181E-01	7.719E-01	8.084E-01	8.263E-01	7.667E-01
Ba-132	N,G	7.260E-01	5.727E-01	6.388E-01	6.989E-01	5.041E-01
Ba-132	N,GX	5.810E-02	4.583E-02	5.112E-02	5.593E-02	4.034E-02
Ba-134	N,G	8.582E-01	8.363E-01	8.261E-01	8.365E-01	8.054E-01
Ba-134	N,GX	5.309E-02	5.017E-02	5.252E-02	5.365E-02	4.988E-02
Ba-135	N,G	3.494E+00	3.445E+00	3.575E+00	3.539E+00	3.534E+00
Ba-135	N,GX	6.490E-03	6.230E-03	6.488E-03	6.588E-03	6.248E-03
Ba-136	N,G	1.015E-01	9.643E-02	1.002E-01	1.008E-01	9.596E-02
Ba-136	N,GX	1.489E-03	1.308E-03	1.403E-03	1.474E-03	1.243E-03
Ba-137	N,G	5.291E-01	4.403E-01	4.809E-01	5.137E-01	4.030E-01
Ba-138	N,G	3.393E-02	2.780E-02	3.027E-02	3.273E-02	2.487E-02
Ba-139	N,G	5.125E-01	4.043E-01	4.509E-01	4.933E-01	3.559E-01
Ba-140	N,G	5.297E-01	5.126E-01	5.247E-01	5.252E-01	5.096E-01
La-138	N,G	2.230E+01	1.919E+01	2.074E+01	2.196E+01	1.799E+01
La-139	N,G	1.009E+00	8.525E-01	9.344E-01	9.894E-01	7.982E-01
La-140	N,G	2.243E+00	2.247E+00	2.300E+00	2.261E+00	2.305E+00
Ce-136	N,G	1.853E+00	1.737E+00	1.824E+00	1.869E+00	1.719E+00
Ce-136	N,GX	2.794E-01	2.619E-01	2.749E-01	2.817E-01	2.592E-01
Ce-138	N,G	9.395E-02	7.412E-02	8.267E-02	9.044E-02	6.524E-02
Ce-138	N,GX	1.285E-03	1.014E-03	1.131E-03	1.237E-03	8.926E-04
Ce-140	N,G	6.048E-02	5.070E-02	5.480E-02	5.858E-02	4.621E-02
Ce-141	N,G	2.843E+00	2.330E+00	2.561E+00	2.756E+00	2.111E+00
Ce-142	N,G	1.020E-01	8.569E-02	9.295E-02	9.900E-02	7.870E-02
Ce-143	N,G	1.753E+00	1.681E+00	1.718E+00	1.730E+00	1.653E+00
Ce-144	N,G	1.457E-01	1.299E-01	1.392E-01	1.437E-01	1.256E-01
Pr-141	N,G	1.462E+00	1.268E+00	1.364E+00	1.430E+00	1.194E+00
Pr-141	N,GX	3.793E-01	3.089E-01	3.406E-01	3.681E-01	2.786E-01
Pr-141	N,2N	4.100E-04	4.265E-04	4.121E-04	4.047E-04	4.260E-04

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Pr-141	N,A	3.392E-06	3.530E-06	2.959E-06	3.073E-06	3.069E-06
Pr-141	N,P	3.160E-06	3.290E-06	2.762E-06	2.867E-06	2.865E-06
Pr-142	N,G	5.538E+00	5.235E+00	5.423E+00	5.507E+00	5.153E+00
Pr-143	N,G	1.177E+01	1.022E+01	1.093E+01	1.150E+01	9.558E+00
Nd-142	N,G	1.623E+00	1.287E+00	1.432E+00	1.563E+00	1.136E+00
Nd-143	N,G	2.710E+01	2.138E+01	2.380E+01	2.602E+01	1.877E+01
Nd-143	N,2N	6.258E-03	6.510E-03	6.290E-03	6.177E-03	6.502E-03
Nd-143	N,A	4.925E-05	5.134E-05	4.409E-05	4.545E-05	4.578E-05
Nd-143	N,P	2.386E-06	2.483E-06	2.083E-06	2.163E-06	2.160E-06
Nd-144	N,G	4.575E-01	3.977E-01	4.257E-01	4.466E-01	3.733E-01
Nd-145	N,G	9.165E+00	8.463E+00	8.826E+00	9.118E+00	8.210E+00
Nd-145	N,2N	8.200E-03	8.531E-03	8.242E-03	8.094E-03	8.520E-03
Nd-145	N,A	1.747E-05	1.820E-05	1.549E-05	1.601E-05	1.608E-05
Nd-145	N,P	7.149E-07	7.438E-07	6.215E-07	6.461E-07	6.443E-07
Nd-146	N,G	2.345E-01	2.136E-01	2.242E-01	2.305E-01	2.060E-01
Nd-146	N,2N	2.374E-03	2.469E-03	2.386E-03	2.343E-03	2.466E-03
Nd-146	N,A	2.159E-06	2.248E-06	1.885E-06	1.957E-06	1.954E-06
Nd-146	N,P	5.553E-08	5.770E-08	4.789E-08	4.991E-08	4.961E-08
Nd-147	N,G	1.991E+01	1.911E+01	1.980E+01	2.004E+01	1.908E+01
Nd-148	N,G	8.669E-01	8.406E-01	8.532E-01	8.565E-01	8.293E-01
Nd-148	N,2N	2.590E-04	2.694E-04	2.603E-04	2.556E-04	2.690E-04
Nd-148	N,A	6.474E-07	6.735E-07	5.624E-07	5.848E-07	5.830E-07
Nd-148	N,P	1.289E-08	1.340E-08	1.111E-08	1.158E-08	1.151E-08
Nd-150	N,G	6.503E-01	6.441E-01	6.730E-01	6.614E-01	6.696E-01
Nd-150	N,2N	2.590E-03	2.694E-03	2.603E-03	2.556E-03	2.690E-03
Nd-150	N,A	1.570E-07	1.633E-07	1.359E-07	1.415E-07	1.409E-07
Nd-150	N,P	8.406E-09	8.733E-09	7.243E-09	7.550E-09	7.501E-09
Pm-147	N,G	6.272E+01	5.962E+01	5.964E+01	6.183E+01	5.691E+01
Pm-147	N,GX	2.966E+01	2.809E+01	2.938E+01	2.999E+01	2.796E+01
Pm-147	N,2N	2.158E-03	2.245E-03	2.169E-03	2.130E-03	2.242E-03
Pm-147	N,A	5.610E-06	5.842E-06	4.922E-06	5.103E-06	5.106E-06
Pm-147	N,P	1.492E-06	1.553E-06	1.301E-06	1.351E-06	1.349E-06
Pm-148	N,G	1.258E+03	1.201E+03	1.280E+03	1.304E+03	1.225E+03
Pm-148m	N,G	2.810E+03	2.199E+03	2.478E+03	2.730E+03	1.921E+03
Pm-149	N,G	1.259E+02	1.009E+02	1.118E+02	1.215E+02	8.982E+01
Pm-151	N,G	9.942E+01	8.664E+01	9.337E+01	9.836E+01	8.208E+01
Sm-144	N,G	5.979E-02	4.716E-02	5.261E-02	5.755E-02	4.152E-02
Sm-145	N,G	9.395E+00	7.412E+00	8.267E+00	9.044E+00	6.524E+00
Sm-147	N,G	2.400E+01	2.304E+01	2.340E+01	2.383E+01	2.252E+01
Sm-147	N,2N	5.193E-03	5.399E-03	4.494E-03	4.679E-03	4.657E-03
Sm-147	N,A	1.069E-04	1.101E-04	1.014E-04	1.023E-04	1.040E-04
Sm-147	N,P	5.774E-06	5.967E-06	5.268E-06	5.389E-06	5.424E-06
Sm-148	N,G	1.142E+00	1.119E+00	1.146E+00	1.141E+00	1.128E+00
Sm-149	N,G	6.663E+03	5.002E+03	5.532E+03	6.219E+03	4.083E+03
Sm-149	N,2N	3.237E-03	3.367E-03	3.253E-03	3.195E-03	3.363E-03
Sm-149	N,A	4.092E-05	4.261E-05	3.590E-05	3.722E-05	3.725E-05
Sm-149	N,P	4.092E-05	4.261E-05	3.590E-05	3.722E-05	3.725E-05

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Sm-150	N,G	1.480E+01	1.307E+01	1.349E+01	1.427E+01	1.195E+01
Sm-151	N,G	6.597E+02	4.999E+02	5.558E+02	6.169E+02	4.214E+02
Sm-151	N,2N	1.101E-02	1.145E-02	1.106E-02	1.086E-02	1.143E-02
Sm-151	N,A	9.048E-06	9.428E-06	8.003E-06	8.278E-06	8.306E-06
Sm-151	N,P	7.949E-07	8.272E-07	6.928E-07	7.197E-07	7.184E-07
Sm-152	N,G	7.447E+01	7.080E+01	7.201E+01	7.429E+01	6.877E+01
Sm-152	N,2N	1.316E-03	1.369E-03	1.323E-03	1.299E-03	1.368E-03
Sm-152	N,A	4.949E-07	5.147E-07	4.289E-07	4.463E-07	4.445E-07
Sm-152	N,P	6.920E-08	7.192E-08	5.976E-08	6.225E-08	6.190E-08
Sm-153	N,G	8.881E+01	8.278E+01	8.577E+01	8.855E+01	8.040E+01
Sm-154	N,G	1.513E+00	1.440E+00	1.541E+00	1.537E+00	1.480E+00
Eu-151	N,G	7.275E+02	5.971E+02	6.724E+02	7.209E+02	5.556E+02
Eu-151	N,GX	2.753E+02	2.158E+02	2.413E+02	2.646E+02	1.890E+02
Eu-151	N,2N	8.200E-04	8.531E-04	8.242E-04	8.094E-04	8.520E-04
Eu-151	N,A	3.670E-05	3.826E-05	3.356E-05	3.433E-05	3.485E-05
Eu-151	N,P	1.802E-05	1.876E-05	1.589E-05	1.645E-05	1.648E-05
Eu-152	N,G	1.879E+02	1.621E+02	1.739E+02	1.838E+02	1.518E+02
Eu-152	N,2N	3.021E-03	3.143E-03	3.037E-03	2.982E-03	3.139E-03
Eu-152	N,A	4.768E-05	4.970E-05	4.338E-05	4.444E-05	4.503E-05
Eu-152	N,P	6.136E-05	6.387E-05	5.462E-05	5.638E-05	5.663E-05
Eu-153	N,G	5.568E+01	5.121E+01	5.388E+01	5.552E+01	4.995E+01
Eu-153	N,2N	6.945E-04	7.215E-04	5.984E-04	6.238E-04	6.197E-04
Eu-153	N,A	2.461E-05	2.562E-05	2.284E-05	2.326E-05	2.369E-05
Eu-153	N,P	2.548E-06	2.654E-06	2.272E-06	2.343E-06	2.358E-06
Eu-154	N,G	1.250E+02	1.083E+02	1.162E+02	1.226E+02	1.020E+02
Eu-154	N,2N	2.029E-03	2.110E-03	2.039E-03	2.002E-03	2.107E-03
Eu-154	N,A	6.421E-05	6.694E-05	5.881E-05	6.012E-05	6.107E-05
Eu-154	N,P	9.254E-05	9.641E-05	8.223E-05	8.488E-05	8.534E-05
Eu-155	N,G	3.466E+02	2.738E+02	3.052E+02	3.338E+02	2.412E+02
Eu-155	N,2N	1.662E-03	1.729E-03	1.670E-03	1.640E-03	1.726E-03
Eu-155	N,A	3.571E-07	3.713E-07	3.094E-07	3.220E-07	3.207E-07
Eu-155	N,P	2.951E-07	3.070E-07	2.565E-07	2.667E-07	2.660E-07
Eu-156	N,G	7.154E+01	6.275E+01	6.747E+01	7.095E+01	5.971E+01
Gd-152	N,G	1.195E+01	1.181E+01	1.214E+01	1.209E+01	1.201E+01
Gd-152	N,2N	4.437E-04	4.610E-04	3.823E-04	3.986E-04	3.960E-04
Gd-152	N,A	2.710E-06	2.824E-06	2.439E-06	2.509E-06	2.532E-06
Gd-152	N,P	4.213E-06	4.391E-06	3.756E-06	3.877E-06	3.899E-06
Gd-154	N,G	1.315E+01	1.173E+01	8.619E+00	8.725E+00	8.327E+00
Gd-154	N,GX	6.536E-03	5.643E-03	6.092E-03	6.441E-03	5.302E-03
Gd-155	N,G	2.451E+03	1.785E+03	2.961E+02	3.277E+02	2.375E+02
Gd-155	N,A	7.601E-06	5.997E-06	6.689E-06	7.318E-06	5.279E-06
Gd-156	N,G	3.798E+00	3.823E+00	3.585E+00	3.562E+00	3.609E+00
Gd-157	N,G	1.041E+04	7.551E+03	1.142E+03	1.278E+03	8.890E+02
Gd-157	N,A	4.698E-05	3.706E-05	4.134E-05	4.522E-05	3.262E-05
Gd-158	N,G	1.794E+00	1.771E+00	1.538E+00	1.552E+00	1.542E+00
Gd-160	N,G	3.728E-01	3.692E-01	3.388E-01	3.314E-01	3.447E-01
Gd-161	N,G	2.648E+03	2.089E+03	2.330E+03	2.549E+03	1.839E+03

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Tb-159	N,G	1.477E+01	1.443E+01	1.503E+01	1.500E+01	1.472E+01
Tb-159	N,P	1.708E+00	1.348E+00	1.503E+00	1.644E+00	1.186E+00
Tb-160	N,G	6.617E+01	5.666E+01	6.141E+01	6.516E+01	5.303E+01
Dy-156	N,G	2.425E+01	2.362E+01	2.448E+01	2.473E+01	2.388E+01
Dy-156	N,A	7.612E-04	6.005E-04	6.698E-04	7.327E-04	5.286E-04
Dy-158	N,G	5.955E+00	5.176E+00	5.575E+00	5.880E+00	4.885E+00
Dy-158	N,A	5.143E-04	4.057E-04	4.525E-04	4.950E-04	3.571E-04
Dy-160	N,G	4.857E+01	4.747E+01	4.942E+01	4.951E+01	4.840E+01
Dy-160	N,A	2.553E-05	2.014E-05	2.246E-05	2.458E-05	1.773E-05
Dy-161	N,G	7.320E+01	6.331E+01	6.814E+01	7.189E+01	5.945E+01
Dy-161	N,A	2.548E-06	2.010E-06	2.242E-06	2.453E-06	1.769E-06
Dy-162	N,G	7.596E+01	7.234E+01	7.236E+01	7.504E+01	6.916E+01
Dy-163	N,G	4.788E+01	4.547E+01	4.752E+01	4.845E+01	4.533E+01
Dy-163	N,A	1.665E-06	1.314E-06	1.466E-06	1.603E-06	1.157E-06
Dy-164	N,G	7.837E+01	6.035E+01	6.793E+01	7.499E+01	5.211E+01
Dy-164	N,GX	1.332E+02	1.026E+02	1.155E+02	1.275E+02	8.859E+01
Dy-164	N,2N	1.662E-03	1.729E-03	1.670E-03	1.640E-03	1.726E-03
Dy-165	N,G	7.923E+02	7.211E+02	7.644E+02	7.922E+02	7.010E+02
Ho-165	N,G	1.978E+01	1.862E+01	1.951E+01	1.997E+01	1.847E+01
Ho-165	N,GX	1.098E+00	1.033E+00	1.083E+00	1.108E+00	1.025E+00
Ho-165	N,A	1.708E-06	1.348E-06	1.503E-06	1.644E-06	1.186E-06
Ho-165	N,P	1.483E+00	1.170E+00	1.305E+00	1.427E+00	1.030E+00
Er-162	N,G	1.232E+01	1.195E+01	1.240E+01	1.254E+01	1.206E+01
Er-162	N,A	9.421E-04	7.432E-04	8.290E-04	9.069E-04	6.542E-04
Er-164	N,G	3.359E+00	3.121E+00	3.285E+00	3.378E+00	3.071E+00
Er-164	N,A	1.022E-04	8.058E-05	8.989E-05	9.834E-05	7.093E-05
Er-166	N,G	2.800E+00	2.437E+00	2.624E+00	2.766E+00	2.303E+00
Er-166	N,GX	2.100E+00	1.828E+00	1.968E+00	2.074E+00	1.727E+00
Er-166	N,A	5.979E-06	4.716E-06	5.261E-06	5.755E-06	4.152E-06
Er-167	N,G	1.177E+02	1.056E+02	1.125E+02	1.172E+02	1.016E+02
Er-167	N,A	5.722E-06	4.514E-06	5.035E-06	5.509E-06	3.974E-06
Er-168	N,G	9.631E-01	9.266E-01	9.642E-01	9.783E-01	9.305E-01
Er-168	N,A	7.661E-06	6.044E-06	6.741E-06	7.375E-06	5.320E-06
Er-170	N,G	8.823E-01	7.788E-01	8.343E-01	8.747E-01	7.426E-01
Er-171	N,G	2.391E+01	1.887E+01	2.104E+01	2.302E+01	1.661E+01
Tm-169	N,G	4.356E+01	4.179E+01	4.353E+01	4.421E+01	4.190E+01
Tm-169	N,GX	3.199E+00	3.049E+00	3.183E+00	3.241E+00	3.045E+00
Tm-169	N,A	8.541E-07	6.738E-07	7.516E-07	8.222E-07	5.931E-07
Tm-170	N,G	1.735E+01	1.568E+01	1.666E+01	1.731E+01	1.517E+01
Tm-171	N,G	3.015E+00	2.929E+00	3.038E+00	3.071E+00	2.958E+00
Yb-168	N,G	9.640E+02	9.003E+02	9.461E+02	9.709E+02	8.888E+02
Yb-168	N,A	3.260E-04	2.572E-04	2.869E-04	3.138E-04	2.264E-04
Yb-170	N,G	7.556E+00	7.364E+00	7.631E+00	7.704E+00	7.449E+00
Yb-170	N,A	3.416E-06	2.695E-06	3.006E-06	3.289E-06	2.372E-06
Yb-171	N,G	1.129E+01	1.038E+01	1.096E+01	1.132E+01	1.015E+01
Yb-171	N,A	3.416E-06	2.695E-06	3.006E-06	3.289E-06	2.372E-06
Yb-172	N,G	6.648E-01	6.403E-01	6.661E-01	6.755E-01	6.435E-01

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Yb-172	N,A	2.554E-06	2.015E-06	2.247E-06	2.458E-06	1.773E-06
Yb-173	N,G	1.027E+01	9.916E+00	1.031E+01	1.045E+01	9.977E+00
Yb-174	N,G	1.648E+00	1.305E+00	1.453E+00	1.588E+00	1.152E+00
Yb-174	N,GX	3.989E+00	3.159E+00	3.519E+00	3.844E+00	2.790E+00
Yb-174	N,A	1.721E-06	1.358E-06	1.514E-06	1.657E-06	1.195E-06
Yb-176	N,G	3.166E-01	2.731E-01	2.949E-01	3.119E-01	2.565E-01
Lu-175	N,G	6.564E+00	6.427E+00	6.650E+00	6.702E+00	6.518E+00
Lu-175	N,GX	1.538E+01	1.506E+01	1.558E+01	1.570E+01	1.527E+01
Lu-175	N,A	5.196E-06	4.099E-06	4.572E-06	5.002E-06	3.608E-06
Lu-176	N,G	1.836E+02	1.459E+02	1.623E+02	1.771E+02	1.291E+02
Lu-176	N,GX	6.080E-01	4.830E-01	5.373E-01	5.863E-01	4.275E-01
Lu-176	N,A	1.704E-04	1.344E-04	1.499E-04	1.640E-04	1.183E-04
Hf-174	N,G	3.988E+01	3.283E+01	3.605E+01	3.881E+01	2.985E+01
Hf-176	N,G	1.873E+01	1.802E+01	1.875E+01	1.903E+01	1.810E+01
Hf-177	N,G	1.915E+02	1.847E+02	1.920E+02	1.947E+02	1.857E+02
Hf-177	N,GX	5.763E-01	5.557E-01	5.779E-01	5.858E-01	5.588E-01
Hf-178	N,G	1.945E+01	1.883E+01	1.955E+01	1.979E+01	1.897E+01
Hf-178	N,GX	3.125E+01	3.024E+01	3.141E+01	3.180E+01	3.048E+01
Hf-179	N,G	1.686E+01	1.603E+01	1.675E+01	1.707E+01	1.600E+01
Hf-179	N,GX	1.291E-01	1.228E-01	1.283E-01	1.307E-01	1.225E-01
Hf-180	N,G	1.923E+00	1.694E+00	1.816E+00	1.905E+00	1.613E+00
Hf-181	N,G	3.416E+00	2.695E+00	3.006E+00	3.289E+00	2.372E+00
Ta-180	N,G	6.625E+01	5.362E+01	5.924E+01	6.419E+01	4.813E+01
Ta-181	N,G	1.768E+01	1.727E+01	1.788E+01	1.804E+01	1.749E+01
Ta-181	N,GX	8.665E-03	8.466E-03	8.765E-03	8.842E-03	8.575E-03
Ta-182	N,G	6.394E+02	4.916E+02	5.537E+02	6.116E+02	4.239E+02
W-180	N,G	4.799E+00	4.728E+00	4.882E+00	4.909E+00	4.811E+00
W-182	N,G	1.494E+01	1.454E+01	1.507E+01	1.523E+01	1.470E+01
W-182	N,GX	4.243E-02	3.347E-02	3.734E-02	4.085E-02	2.947E-02
W-183	N,G	8.592E+00	8.394E+00	8.691E+00	8.767E+00	8.502E+00
W-184	N,G	4.524E-01	4.194E-01	4.419E-01	4.547E-01	4.123E-01
W-184	N,GX	4.983E-04	4.620E-04	4.867E-04	5.009E-04	4.541E-04
W-186	N,G	1.418E+01	1.348E+01	1.408E+01	1.436E+01	1.345E+01
W-187	N,G	6.741E+01	6.615E+01	6.839E+01	6.887E+01	6.716E+01
Re-185	N,G	4.766E+01	4.557E+01	4.752E+01	4.833E+01	4.561E+01
Re-187	N,G	2.647E-01	2.359E-01	2.519E-01	2.632E-01	2.264E-01
Re-187	N,GX	1.211E+01	1.079E+01	1.152E+01	1.203E+01	1.035E+01
Re-188	N,G	1.708E-01	1.348E-01	1.503E-01	1.644E-01	1.186E-01
Os-184	N,G	2.562E+02	2.021E+02	2.255E+02	2.467E+02	1.779E+02
Os-184	N,A	8.456E-04	6.670E-04	7.440E-04	8.140E-04	5.872E-04
Os-186	N,A	8.541E-06	6.738E-06	7.516E-06	8.222E-06	5.931E-06
Os-187	N,G	4.545E+01	3.937E+01	4.245E+01	4.483E+01	3.707E+01
Os-187	N,A	8.609E-06	6.792E-06	7.576E-06	8.288E-06	5.978E-06
Os-188	N,G	3.385E+00	3.302E+00	3.421E+00	3.453E+00	3.342E+00
Os-188	N,A	2.571E-06	2.028E-06	2.262E-06	2.475E-06	1.785E-06
Os-189	N,G	1.872E+01	1.828E+01	1.893E+01	1.910E+01	1.851E+01
Os-189	N,GX	2.061E-04	2.012E-04	2.084E-04	2.103E-04	2.038E-04

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Os-189	N,A	8.447E-07	6.664E-07	7.433E-07	8.132E-07	5.866E-07
Os-190	N,G	4.906E-01	4.200E-01	4.548E-01	4.824E-01	3.924E-01
Os-190	N,GX	1.145E+00	9.800E-01	1.061E+00	1.126E+00	9.157E-01
Os-190	N,A	1.665E-06	1.314E-06	1.466E-06	1.603E-06	1.157E-06
Os-192	N,G	2.720E-01	2.357E-01	2.541E-01	2.683E-01	2.221E-01
Os-192	N,GX	1.960E-03	1.706E-03	1.837E-03	1.936E-03	1.612E-03
Os-192	N,A	8.601E-07	6.785E-07	7.568E-07	8.280E-07	5.973E-07
Os-193	N,G	1.315E+02	1.038E+02	1.157E+02	1.266E+02	9.134E+01
Ir-191	N,G	1.545E+02	1.377E+02	1.470E+02	1.535E+02	1.321E+02
Ir-191	N,GX	6.335E-02	5.646E-02	6.028E-02	6.297E-02	5.417E-02
Ir-192	N,G	9.395E+01	7.412E+01	8.267E+01	9.044E+01	6.524E+01
Ir-193	N,G	3.735E+01	3.532E+01	3.696E+01	3.775E+01	3.512E+01
Ir-193	N,GX	1.969E+00	1.862E+00	1.949E+00	1.990E+00	1.852E+00
Pt-190	N,G	1.281E+01	1.011E+01	1.127E+01	1.233E+01	8.896E+00
Pt-190	N,A	6.790E-04	5.357E-04	5.975E-04	6.537E-04	4.715E-04
Pt-192	N,G	2.474E+00	2.259E+00	2.392E+00	2.476E+00	2.200E+00
Pt-192	N,GX	4.611E-01	4.210E-01	4.458E-01	4.615E-01	4.100E-01
Pt-192	N,A	1.674E-05	1.321E-05	1.473E-05	1.612E-05	1.162E-05
Pt-194	N,G	1.674E-01	1.472E-01	1.579E-01	1.658E-01	1.401E-01
Pt-194	N,GX	1.357E-02	1.194E-02	1.281E-02	1.344E-02	1.136E-02
Pt-194	N,A	4.305E-07	3.396E-07	3.788E-07	4.144E-07	2.989E-07
Pt-195	N,G	1.008E+01	9.581E+00	1.001E+01	1.020E+01	9.555E+00
Pt-195	N,A	4.382E-07	3.456E-07	3.856E-07	4.218E-07	3.043E-07
Pt-196	N,G	2.211E-01	2.083E-01	2.183E-01	2.232E-01	2.068E-01
Pt-196	N,GX	1.603E-02	1.510E-02	1.583E-02	1.618E-02	1.499E-02
Pt-198	N,G	1.537E+00	1.469E+00	1.532E+00	1.558E+00	1.469E+00
Pt-198	N,GX	1.130E-02	1.080E-02	1.126E-02	1.146E-02	1.080E-02
Au-197	N,G	4.281E+01	4.097E+01	4.271E+01	4.342E+01	4.102E+01
Au-198	N,G	2.204E+03	1.738E+03	1.939E+03	2.121E+03	1.530E+03
Au-199	N,G	2.562E+00	2.021E+00	2.255E+00	2.467E+00	1.779E+00
Hg-196	N,G	2.410E+02	1.855E+02	2.088E+02	2.306E+02	1.601E+02
Hg-196	N,GX	1.036E+01	8.196E+00	9.133E+00	9.981E+00	7.231E+00
Hg-198	N,G	1.714E+00	1.677E+00	1.736E+00	1.750E+00	1.700E+00
Hg-198	N,GX	1.644E-02	1.609E-02	1.665E-02	1.678E-02	1.631E-02
Hg-199	N,G	1.539E+02	1.178E+02	1.329E+02	1.470E+02	1.013E+02
Hg-200	N,G	5.125E+00	4.043E+00	4.509E+00	4.933E+00	3.559E+00
Hg-201	N,G	5.125E+00	4.043E+00	4.509E+00	4.933E+00	3.559E+00
Hg-202	N,G	4.796E-01	3.912E-01	4.310E-01	4.656E-01	3.531E-01
Hg-204	N,G	3.673E-02	2.897E-02	3.232E-02	3.535E-02	2.550E-02
Tl-203	N,G	1.723E+00	1.523E+00	1.631E+00	1.709E+00	1.454E+00
Tl-204	N,G	3.575E+00	3.182E+00	3.399E+00	3.553E+00	3.051E+00
Tl-205	N,G	2.340E-02	2.157E-02	2.276E-02	2.348E-02	2.113E-02
Pb-204	N,G	8.827E-02	7.629E-02	8.233E-02	8.701E-02	7.174E-02
Pb-206	N,G	2.298E+00	1.749E+00	1.977E+00	2.193E+00	1.495E+00
Pb-207	N,G	6.239E-02	4.960E-02	5.517E-02	6.018E-02	4.393E-02
Pb-208	N,G	4.159E-05	3.281E-05	3.660E-05	4.004E-05	2.888E-05
Bi-209	N,G	3.910E-03	3.563E-03	3.776E-03	3.911E-03	3.467E-03

Table A.1 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Bi-209	N,GX	2.881E-03	2.625E-03	2.782E-03	2.882E-03	2.554E-03
Bi-209	N,2N	4.087E-04	4.252E-04	4.108E-04	4.034E-04	4.246E-04
Bi-209	N,P	1.230E-06	1.280E-06	1.236E-06	1.214E-06	1.278E-06
Bi-210	N,G	4.612E-03	3.638E-03	4.058E-03	4.440E-03	3.203E-03
Po-210	N,G	2.562E-03	2.021E-03	2.255E-03	2.467E-03	1.779E-03
Po-210	N,GX	4.272E-05	3.370E-05	3.759E-05	4.113E-05	2.967E-05

[a] N,G = (n,γ) reaction to a ground state.
N,GX = (n,γ) reaction to an excited state.
N,3N = (n,3n) reaction to a ground state.
N,F = (n,fission) reaction.
N,2N = (n,2n) reaction to a ground state.
N,2NX = (n,2n) reaction to an excited state.
N,A = (n,α) reaction to a ground state.
N,P = (n,p) reaction to a ground state.

Table A.2. One-group, spectrum-averaged cross sections for uranium cycle light-water reactors - actinides and daughters

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Pb-206	N,G	2.298E+00	1.749E+00	1.977E+00	2.193E+00	1.495E+00
Pb-207	N,G	6.239E-02	4.960E-02	5.517E-02	6.018E-02	4.393E-02
Pb-208	N,G	4.159E-05	3.281E-05	3.660E-05	4.004E-05	2.888E-05
Pb-210	N,G	4.271E-02	3.369E-02	3.758E-02	4.111E-02	2.965E-02
Bi-209	N,G	3.910E-03	3.563E-03	3.776E-03	3.911E-03	3.467E-03
Bi-209	N,GX	2.881E-03	2.625E-03	2.782E-03	2.882E-03	2.554E-03
Bi-209	N,F	1.230E-06	1.280E-06	1.236E-06	1.214E-06	1.278E-06
Bi-209	N,2N	4.087E-04	4.252E-04	4.108E-04	4.034E-04	4.246E-04
Bi-210	N,G	4.612E-03	3.638E-03	4.058E-03	4.440E-03	3.203E-03
Po-210	N,G	2.562E-03	2.021E-03	2.255E-03	2.467E-03	1.779E-03
Po-210	N,GX	4.272E-05	3.370E-05	3.759E-05	4.113E-05	2.967E-05
Rn-220	N,G	1.708E-02	1.348E-02	1.503E-02	1.644E-02	1.186E-02
Rn-222	N,G	6.150E-02	4.851E-02	5.411E-02	5.920E-02	4.270E-02
Ra-223	N,G	1.110E+01	8.759E+00	9.770E+00	1.069E+01	7.710E+00
Ra-223	N,F	5.979E-02	4.716E-02	5.261E-02	5.755E-02	4.152E-02
Ra-224	N,G	1.025E+00	8.085E-01	9.019E-01	9.866E-01	7.117E-01
Ra-226	N,G	5.900E+00	5.684E+00	5.912E+00	5.995E+00	5.712E+00
Ra-226	N,F	8.541E-06	6.738E-06	7.516E-06	8.222E-06	5.931E-06
Ra-228	N,G	3.075E+00	2.426E+00	2.706E+00	2.960E+00	2.135E+00
Ra-228	N,F	1.708E-01	1.348E-01	1.503E-01	1.644E-01	1.186E-01
Ac-227	N,G	4.399E+01	3.470E+01	3.871E+01	4.234E+01	3.054E+01
Ac-227	N,F	1.708E-04	1.348E-04	1.503E-04	1.644E-04	1.186E-04
Th-227	N,F	1.708E+01	1.348E+01	1.503E+01	1.644E+01	1.186E+01
Th-228	N,G	3.222E+01	2.997E+01	3.154E+01	3.242E+01	2.951E+01
Th-228	N,F	2.562E-02	2.021E-02	2.255E-02	2.467E-02	1.779E-02
Th-229	N,G	2.674E+01	2.573E+01	2.677E+01	2.716E+01	2.584E+01
Th-229	N,F	1.282E+01	1.225E+01	1.277E+01	1.299E+01	1.226E+01
Th-230	N,G	2.318E+01	2.272E+01	2.376E+01	2.389E+01	2.336E+01
Th-230	N,F	5.791E-02	6.043E-02	5.383E-02	5.489E-02	5.594E-02
Th-230	N,2N	4.540E-03	4.721E-03	3.931E-03	4.092E-03	4.074E-03
Th-232	N,G	2.906E+00	2.814E+00	2.869E+00	2.889E+00	2.786E+00
Th-232	N,3N	1.843E-05	1.917E-05	1.852E-05	1.819E-05	1.915E-05
Th-232	N,F	2.275E-02	2.374E-02	2.076E-02	2.129E-02	2.157E-02
Th-232	N,2N	4.273E-03	4.443E-03	3.697E-03	3.849E-03	3.831E-03
Th-233	N,G	1.219E+02	9.484E+01	1.063E+02	1.169E+02	8.258E+01
Th-233	N,F	1.281E+00	1.011E+00	1.127E+00	1.233E+00	8.896E-01
Th-234	N,G	1.537E-01	1.213E-01	1.353E-01	1.480E-01	1.068E-01
Th-234	N,F	8.541E-04	6.738E-04	7.516E-04	8.222E-04	5.931E-04
Pa-231	N,G	6.630E+01	6.080E+01	6.549E+01	6.739E+01	6.026E+01
Pa-231	N,F	3.716E-01	3.869E-01	3.535E-01	3.573E-01	3.665E-01
Pa-231	N,2N	3.000E-03	3.119E-03	2.597E-03	2.703E-03	2.691E-03
Pa-232	N,G	6.491E+01	5.121E+01	5.712E+01	6.249E+01	4.508E+01
Pa-232	N,F	5.979E+01	4.716E+01	5.261E+01	5.755E+01	4.152E+01
Pa-233	N,G	1.215E+01	1.181E+01	1.230E+01	1.242E+01	1.201E+01
Pa-233	N,GX	1.215E+01	1.181E+01	1.230E+01	1.242E+01	1.201E+01
Pa-233	N,3N	4.748E-05	4.939E-05	4.772E-05	4.686E-05	4.932E-05

Table A.2 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Pa-233	N,F	1.420E-01	1.481E-01	1.300E-01	1.332E-01	1.350E-01
Pa-233	N,2N	1.222E-03	1.270E-03	1.056E-03	1.100E-03	1.094E-03
Pa-234	N,F	4.270E+02	3.369E+02	3.758E+02	4.111E+02	2.965E+02
Pa-234m	N,G	2.030E+01	2.026E+01	2.084E+01	2.084E+01	2.076E+01
Pa-234m	N,F	4.271E+01	3.369E+01	3.758E+01	4.111E+01	2.965E+01
U-230	N,F	2.135E+00	1.684E+00	1.879E+00	2.056E+00	1.483E+00
U-231	N,F	3.416E+01	2.695E+01	3.006E+01	3.289E+01	2.372E+01
U-232	N,G	1.041E+01	9.340E+00	9.809E+00	1.021E+01	8.884E+00
U-232	N,3N	6.474E-06	6.735E-06	6.507E-06	6.390E-06	6.726E-06
U-232	N,F	1.573E+01	1.455E+01	1.527E+01	1.564E+01	1.424E+01
U-232	N,2N	1.871E-03	1.945E-03	1.615E-03	1.683E-03	1.673E-03
U-233	N,G	7.210E+00	6.296E+00	6.791E+00	7.147E+00	5.982E+00
U-233	N,3N	1.360E-06	1.414E-06	1.366E-06	1.342E-06	1.412E-06
U-233	N,F	5.885E+01	4.950E+01	5.399E+01	5.762E+01	4.574E+01
U-233	N,2N	2.594E-03	2.699E-03	2.252E-03	2.342E-03	2.334E-03
U-234	N,G	2.071E+01	1.911E+01	1.940E+01	2.028E+01	1.800E+01
U-234	N,3N	9.495E-06	9.878E-06	9.544E-06	9.372E-06	9.865E-06
U-234	N,F	4.516E-01	4.711E-01	4.289E-01	4.333E-01	4.455E-01
U-234	N,2N	3.639E-04	3.782E-04	3.143E-04	3.274E-04	3.256E-04
U-235	N,G	1.068E+01	8.910E+00	9.742E+00	1.038E+01	8.120E+00
U-235	N,3N	8.049E-07	8.374E-07	8.090E-07	7.945E-07	8.363E-07
U-235	N,F	4.752E+01	3.775E+01	4.186E+01	4.558E+01	3.301E+01
U-235	N,2N	2.338E-03	2.433E-03	2.035E-03	2.114E-03	2.109E-03
U-236	N,G	8.348E+00	8.295E+00	8.132E+00	8.281E+00	8.084E+00
U-236	N,3N	1.554E-05	1.616E-05	1.562E-05	1.534E-05	1.614E-05
U-236	N,F	1.910E-01	1.993E-01	1.779E-01	1.812E-01	1.848E-01
U-236	N,2N	2.271E-03	2.361E-03	1.965E-03	2.045E-03	2.036E-03
U-237	N,G	4.077E+01	3.269E+01	3.608E+01	3.922E+01	2.901E+01
U-237	N,F	6.174E-01	6.010E-01	5.942E-01	6.029E-01	5.802E-01
U-237	N,2N	7.112E-03	7.399E-03	6.183E-03	6.427E-03	6.410E-03
U-238	N,G	8.872E-01	8.546E-01	8.025E-01	8.262E-01	7.730E-01
U-238	N,3N	3.237E-05	3.367E-05	3.253E-05	3.195E-05	3.363E-05
U-238	N,F	9.281E-02	9.685E-02	8.486E-02	8.699E-02	8.818E-02
U-238	N,2N	4.751E-03	4.940E-03	4.116E-03	4.283E-03	4.266E-03
U-239	N,G	1.879E+00	1.482E+00	1.653E+00	1.809E+00	1.305E+00
U-239	N,F	1.196E+00	9.433E-01	1.052E+00	1.151E+00	8.303E-01
U-240	N,G	8.541E-02	6.738E-02	7.516E-02	8.222E-02	5.931E-02
Np-235	N,G	1.572E+01	1.240E+01	1.383E+01	1.513E+01	1.091E+01
Np-235	N,GX	1.367E+02	1.078E+02	1.202E+02	1.316E+02	9.490E+01
Np-236	N,F	2.139E+02	1.688E+02	1.883E+02	2.059E+02	1.487E+02
Np-237	N,G	3.328E+01	3.031E+01	3.260E+01	3.364E+01	2.992E+01
Np-237	N,3N	2.050E-06	2.133E-06	2.061E-06	2.023E-06	2.130E-06
Np-237	N,F	4.953E-01	5.163E-01	4.702E-01	4.755E-01	4.881E-01
Np-237	N,2N	1.934E-04	2.012E-04	1.943E-04	1.908E-04	2.009E-04
Np-237	N,2NX	5.546E-04	5.770E-04	5.574E-04	5.474E-04	5.762E-04
Np-238	N,G	1.590E+01	1.258E+01	1.397E+01	1.527E+01	1.106E+01
Np-238	N,F	1.568E+02	1.234E+02	1.373E+02	1.504E+02	1.081E+02

Table A.2 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Np-239	N,G	1.047E+01	1.020E+01	1.057E+01	1.067E+01	1.031E+01
Np-239	N,GX	2.648E+00	2.089E+00	2.330E+00	2.549E+00	1.839E+00
Np-239	N,F	3.206E-01	3.121E-01	3.116E-01	3.144E-01	3.037E-01
Pu-236	N,G	1.935E+01	1.659E+01	1.788E+01	1.897E+01	1.546E+01
Pu-236	N,F	2.030E+01	1.757E+01	1.880E+01	1.989E+01	1.642E+01
Pu-236	N,2N	6.820E-04	7.087E-04	5.881E-04	6.129E-04	6.091E-04
Pu-237	N,G	2.050E+02	1.617E+02	1.804E+02	1.973E+02	1.423E+02
Pu-237	N,F	2.050E+02	1.617E+02	1.804E+02	1.973E+02	1.423E+02
Pu-238	N,G	3.483E+01	2.690E+01	2.978E+01	3.288E+01	2.293E+01
Pu-238	N,3N	6.474E-06	6.735E-06	6.507E-06	6.390E-06	6.726E-06
Pu-238	N,F	2.308E+00	2.119E+00	2.143E+00	2.225E+00	1.981E+00
Pu-238	N,2N	1.025E-03	1.065E-03	8.837E-04	9.209E-04	9.153E-04
Pu-239	N,G	6.909E+01	5.753E+01	6.845E+01	7.234E+01	5.697E+01
Pu-239	N,3N	5.934E-07	6.174E-07	5.965E-07	5.858E-07	6.165E-07
Pu-239	N,F	1.211E+02	9.922E+01	1.168E+02	1.245E+02	9.581E+01
Pu-239	N,2N	9.684E-04	1.007E-03	8.411E-04	8.746E-04	8.720E-04
Pu-240	N,G	2.228E+02	2.135E+02	2.227E+02	2.277E+02	2.137E+02
Pu-240	N,3N	7.553E-06	7.858E-06	7.592E-06	7.455E-06	7.847E-06
Pu-240	N,F	5.787E-01	5.992E-01	5.566E-01	5.608E-01	5.741E-01
Pu-240	N,2N	3.825E-04	3.977E-04	3.306E-04	3.443E-04	3.426E-04
Pu-241	N,G	4.202E+01	3.363E+01	3.890E+01	4.202E+01	3.113E+01
Pu-241	N,3N	1.705E-05	1.774E-05	1.714E-05	1.683E-05	1.771E-05
Pu-241	N,F	1.259E+02	1.013E+02	1.153E+02	1.247E+02	9.282E+01
Pu-241	N,2N	6.566E-03	6.835E-03	5.733E-03	5.952E-03	5.946E-03
Pu-242	N,G	3.320E+01	3.290E+01	3.402E+01	3.412E+01	3.363E+01
Pu-242	N,3N	1.597E-05	1.661E-05	1.605E-05	1.576E-05	1.659E-05
Pu-242	N,F	4.063E-01	4.237E-01	3.855E-01	3.899E-01	4.004E-01
Pu-242	N,2N	2.173E-03	2.259E-03	1.880E-03	1.957E-03	1.948E-03
Pu-243	N,G	1.312E+01	1.161E+01	1.245E+01	1.303E+01	1.114E+01
Pu-243	N,F	2.699E+01	2.390E+01	2.560E+01	2.680E+01	2.290E+01
Pu-243	N,2N	1.617E-02	1.684E-02	1.416E-02	1.469E-02	1.469E-02
Pu-244	N,G	1.103E+00	1.071E+00	1.111E+00	1.123E+00	1.081E+00
Pu-245	N,G	1.627E+01	1.356E+01	1.482E+01	1.588E+01	1.243E+01
Am-241	N,G	9.570E+01	8.303E+01	9.599E+01	1.001E+02	8.356E+01
Am-241	N,GX	2.392E+01	2.076E+01	2.400E+01	2.503E+01	2.089E+01
Am-241	N,3N	1.079E-06	1.122E-06	1.084E-06	1.065E-06	1.121E-06
Am-241	N,F	1.120E+00	1.054E+00	1.093E+00	1.128E+00	1.025E+00
Am-241	N,2N	3.085E-04	3.207E-04	2.666E-04	2.776E-04	2.762E-04
Am-242	N,G	1.929E+01	1.497E+01	1.673E+01	1.843E+01	1.295E+01
Am-242	N,3N	4.100E-05	4.265E-05	4.121E-05	4.047E-05	4.260E-05
Am-242	N,F	1.728E+02	1.339E+02	1.498E+02	1.651E+02	1.158E+02
Am-242	N,2N	3.992E-03	4.153E-03	4.013E-03	3.940E-03	4.148E-03
Am-242m	N,G	9.372E+01	7.533E+01	8.514E+01	9.264E+01	6.804E+01
Am-242m	N,3N	4.100E-05	4.265E-05	4.121E-05	4.047E-05	4.260E-05
Am-242m	N,F	4.461E+02	3.608E+02	4.064E+02	4.412E+02	3.271E+02
Am-242m	N,2N	3.992E-03	4.153E-03	4.013E-03	3.940E-03	4.148E-03
Am-243	N,G	2.518E+00	2.434E+00	2.545E+00	2.581E+00	2.470E+00

Table A.2 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Am-243	N,GX	4.784E+01	4.625E+01	4.835E+01	4.904E+01	4.694E+01
Am-243	N,F	3.965E-01	4.138E-01	3.721E-01	3.781E-01	3.866E-01
Am-243	N,2N	2.002E-04	2.082E-04	1.734E-04	1.804E-04	1.796E-04
Am-244	N,F	1.964E+02	1.550E+02	1.729E+02	1.891E+02	1.364E+02
Am-244m	N,F	1.367E+02	1.078E+02	1.202E+02	1.316E+02	9.490E+01
Cm-242	N,G	5.677E+00	5.439E+00	5.709E+00	5.736E+00	5.499E+00
Cm-242	N,F	5.425E-01	5.040E-01	4.881E-01	5.147E-01	4.538E-01
Cm-242	N,2N	5.099E-05	5.299E-05	4.403E-05	4.586E-05	4.561E-05
Cm-243	N,G	8.206E+00	7.690E+00	8.079E+00	8.280E+00	7.619E+00
Cm-243	N,3N	1.079E-05	1.122E-05	1.084E-05	1.065E-05	1.121E-05
Cm-243	N,F	6.997E+01	6.434E+01	6.791E+01	7.016E+01	6.293E+01
Cm-243	N,2N	3.574E-03	3.719E-03	3.110E-03	3.232E-03	3.225E-03
Cm-244	N,G	1.281E+01	1.271E+01	1.266E+01	1.285E+01	1.258E+01
Cm-244	N,F	8.612E-01	8.781E-01	8.279E-01	8.383E-01	8.422E-01
Cm-244	N,2N	1.009E-03	1.049E-03	8.718E-04	9.081E-04	9.032E-04
Cm-245	N,G	2.712E+01	2.104E+01	2.345E+01	2.583E+01	1.816E+01
Cm-245	N,F	1.599E+02	1.252E+02	1.391E+02	1.527E+02	1.089E+02
Cm-245	N,2N	6.268E-03	6.523E-03	5.461E-03	5.673E-03	5.663E-03
Cm-246	N,G	2.877E+00	2.880E+00	2.968E+00	2.958E+00	2.974E+00
Cm-246	N,F	5.749E-01	5.943E-01	5.494E-01	5.569E-01	5.659E-01
Cm-246	N,2N	1.168E-03	1.214E-03	1.009E-03	1.051E-03	1.045E-03
Cm-247	N,G	1.741E+01	1.604E+01	1.704E+01	1.762E+01	1.577E+01
Cm-247	N,F	2.688E+01	2.492E+01	2.635E+01	2.720E+01	2.453E+01
Cm-247	N,2N	9.748E-03	1.015E-02	8.530E-03	8.850E-03	8.849E-03
Cm-248	N,G	6.051E+00	6.057E+00	6.101E+00	6.104E+00	6.100E+00
Cm-248	N,F	7.441E-01	7.650E-01	7.190E-01	7.259E-01	7.368E-01
Cm-248	N,2N	1.760E-03	1.830E-03	1.523E-03	1.585E-03	1.578E-03
Cm-249	N,G	1.367E-01	1.078E-01	1.202E-01	1.316E-01	9.490E-02
Cm-249	N,F	4.270E+00	3.369E+00	3.758E+00	4.111E+00	2.965E+00
Cm-250	N,G	1.708E-01	1.348E-01	1.503E-01	1.644E-01	1.186E-01
Bk-249	N,G	2.240E+02	1.947E+02	2.108E+02	2.225E+02	1.846E+02
Bk-249	N,3N	2.812E+01	2.807E+01	2.887E+01	2.888E+01	2.877E+01
Bk-249	N,F	2.865E-01	2.989E-01	2.675E-01	2.723E-01	2.780E-01
Bk-249	N,2N	6.170E-03	6.418E-03	5.353E-03	5.568E-03	5.548E-03
Bk-250	N,G	8.199E+01	6.468E+01	7.215E+01	7.893E+01	5.694E+01
Bk-250	N,F	2.562E+02	2.021E+02	2.255E+02	2.467E+02	1.779E+02
Cf-249	N,G	4.779E+01	4.113E+01	4.482E+01	4.740E+01	3.892E+01
Cf-249	N,F	1.461E+02	1.232E+02	1.346E+02	1.435E+02	1.145E+02
Cf-249	N,2N	6.645E-03	6.915E-03	5.784E-03	6.011E-03	5.998E-03
Cf-250	N,G	4.516E+02	4.133E+02	4.470E+02	4.623E+02	4.104E+02
Cf-250	N,3N	2.631E+02	2.626E+02	2.700E+02	2.702E+02	2.691E+02
Cf-250	N,F	9.023E-01	9.409E-01	8.768E-01	8.760E-01	9.106E-01
Cf-250	N,2N	3.172E-03	3.299E-03	2.749E-03	2.861E-03	2.849E-03
Cf-251	N,G	2.478E+02	1.971E+02	2.195E+02	2.395E+02	1.748E+02
Cf-251	N,F	5.253E+02	4.284E+02	4.742E+02	5.125E+02	3.881E+02
Cf-251	N,2N	1.140E-02	1.187E-02	9.974E-03	1.035E-02	1.035E-02
Cf-252	N,G	2.810E+00	2.478E+00	2.665E+00	2.774E+00	2.374E+00

Table A.2 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Cf-252	N,F	6.084E+00	5.599E+00	5.870E+00	6.030E+00	5.445E+00
Cf-252	N,2N	2.572E-03	2.675E-03	2.233E-03	2.322E-03	2.315E-03
Cf-253	N,G	3.682E+01	3.129E+01	3.466E+01	3.676E+01	2.965E+01
Cf-253	N,F	1.221E+02	1.037E+02	1.149E+02	1.219E+02	9.819E+01
Cf-254	N,G	4.118E+01	4.022E+01	4.165E+01	4.202E+01	4.072E+01
Es-253	N,G	1.181E+02	1.125E+02	1.249E+02	1.262E+02	1.191E+02
Es-253	N,GX	8.204E+01	7.820E+01	8.680E+01	8.768E+01	8.278E+01
Es-254	N,G	3.416E+00	2.695E+00	3.006E+00	3.289E+00	2.372E+00
Es-254	N,F	2.678E+02	2.154E+02	2.386E+02	2.591E+02	1.925E+02
Es-254m	N,G	1.110E-01	8.759E-02	9.770E-02	1.069E-01	7.710E-02
Es-254m	N,F	1.572E+02	1.240E+02	1.383E+02	1.513E+02	1.091E+02

[a] N,G = (n,γ) reaction to a ground state.
N,GX = (n,γ) reaction to an excited state.
N,3N = (n,3n) reaction to a ground state.
N,F = (n,fission) reaction.
N,2N = (n,2n) reaction to a ground state.
N,2NX = (n,2n) reaction to an excited state.
N,A = (n,α) reaction to a ground state.
N,P = (n,p) reaction to a ground state.

Table A.3. One-group, spectrum-averaged cross sections for uranium cycle light-water reactors - fission products

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
H-3	N,G	5.125E-07	4.043E-07	4.509E-07	4.933E-07	3.559E-07
Li-6	N,G	2.391E-03	1.887E-03	2.104E-03	2.302E-03	1.661E-03
Li-6	N,A	7.704E+01	6.010E+01	6.731E+01	7.395E+01	5.243E+01
Li-6	N,P	8.925E-04	9.285E-04	8.971E-04	8.810E-04	9.273E-04
Li-7	N,G	3.095E-03	2.445E-03	2.726E-03	2.981E-03	2.154E-03
Li-7	N,2N	4.103E-06	4.263E-06	3.535E-06	3.685E-06	3.661E-06
Li-7	N,A	5.568E-03	5.792E-03	5.596E-03	5.495E-03	5.784E-03
Be-9	N,G	7.826E-04	6.167E-04	6.882E-04	7.532E-04	5.424E-04
Be-9	N,2N	4.644E-02	4.831E-02	4.668E-02	4.584E-02	4.825E-02
Be-9	N,A	2.693E-02	2.802E-02	2.707E-02	2.658E-02	2.798E-02
Be-10	N,G	8.541E-05	6.738E-05	7.516E-05	8.222E-05	5.931E-05
C-14	N,G	8.541E-08	6.738E-08	7.516E-08	8.222E-08	5.931E-08
Cu-66	N,G	1.153E+01	9.096E+00	1.015E+01	1.110E+01	8.007E+00
Zn-66	N,G	8.207E-02	6.672E-02	7.360E-02	7.961E-02	6.010E-02
Zn-66	N,A	1.742E-06	1.375E-06	1.533E-06	1.677E-06	1.210E-06
Zn-67	N,G	9.725E-01	8.474E-01	9.119E-01	9.608E-01	8.012E-01
Zn-67	N,A	5.127E-07	4.045E-07	4.512E-07	4.936E-07	3.560E-07
Zn-68	N,G	1.450E-01	1.269E-01	1.363E-01	1.434E-01	1.203E-01
Zn-68	N,GX	1.045E-02	9.142E-03	9.823E-03	1.033E-02	8.666E-03
Zn-68	N,A	1.740E-06	1.372E-06	1.531E-06	1.675E-06	1.208E-06
Zn-70	N,G	7.089E-03	5.592E-03	6.238E-03	6.824E-03	4.923E-03
Zn-70	N,GX	7.432E-04	5.863E-04	6.540E-04	7.155E-04	5.161E-04
Ga-69	N,G	4.802E-01	4.493E-01	4.718E-01	4.838E-01	4.440E-01
Ga-71	N,G	1.040E+00	9.540E-01	1.009E+00	1.042E+00	9.317E-01
Ga-71	N,GX	3.316E-02	3.042E-02	3.216E-02	3.323E-02	2.971E-02
Ge-72	N,G	1.189E-01	1.027E-01	1.101E-01	1.159E-01	9.597E-02
Ge-73	N,G	3.247E+00	3.029E+00	3.129E+00	3.194E+00	2.936E+00
Ge-74	N,G	5.366E-02	4.760E-02	5.067E-02	5.261E-02	4.543E-02
Ge-74	N,GX	1.440E-02	1.181E-02	1.299E-02	1.400E-02	1.071E-02
Ge-75	N,G	3.843E-01	3.032E-01	3.382E-01	3.700E-01	2.669E-01
Ge-76	N,G	5.684E-02	5.578E-02	5.754E-02	5.691E-02	5.671E-02
Ge-76	N,GX	3.628E-02	3.458E-02	3.609E-02	3.675E-02	3.454E-02
As-75	N,G	2.164E+00	2.126E+00	2.176E+00	2.167E+00	2.144E+00
Se-76	N,G	7.484E+00	5.959E+00	6.623E+00	7.217E+00	5.282E+00
Se-76	N,GX	1.826E+00	1.448E+00	1.612E+00	1.760E+00	1.279E+00
Se-77	N,G	4.201E+00	3.463E+00	3.796E+00	4.072E+00	3.148E+00
Se-78	N,G	1.870E-01	1.848E-01	1.881E-01	1.863E-01	1.864E-01
Se-78	N,GX	1.133E-01	1.072E-01	1.121E-01	1.145E-01	1.066E-01
Se-79	N,G	3.371E-01	2.650E-01	2.960E-01	3.242E-01	2.326E-01
Se-80	N,G	8.774E-02	7.814E-02	8.328E-02	8.620E-02	7.502E-02
Se-80	N,GX	1.101E-02	9.556E-03	1.030E-02	1.086E-02	9.014E-03
Se-82	N,G	8.457E-03	7.837E-03	8.051E-03	8.266E-03	7.511E-03
Se-82	N,GX	4.955E-04	3.909E-04	4.361E-04	4.770E-04	3.441E-04
Br-79	N,G	4.794E+00	4.653E+00	4.786E+00	4.796E+00	4.656E+00
Br-79	N,GX	9.780E-01	9.298E-01	9.713E-01	9.900E-01	9.275E-01
Br-81	N,G	1.807E+00	1.799E+00	1.844E+00	1.819E+00	1.838E+00

Table A.3 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Br-81	N,GX	1.228E+00	1.182E+00	1.230E+00	1.247E+00	1.188E+00
Kr-80	N,G	2.787E+00	2.569E+00	2.583E+00	2.684E+00	2.386E+00
Kr-80	N,GX	7.556E-01	6.729E-01	7.187E-01	7.510E-01	6.453E-01
Kr-80	N,2N	4.316E-05	4.490E-05	4.338E-05	4.260E-05	4.484E-05
Kr-80	N,A	8.126E-06	8.464E-06	7.347E-06	7.541E-06	7.624E-06
Kr-80	N,P	1.023E-04	1.065E-04	8.946E-05	9.283E-05	9.278E-05
Kr-82	N,G	7.185E+00	6.708E+00	6.924E+00	7.097E+00	6.505E+00
Kr-82	N,GX	3.520E+00	3.156E+00	3.363E+00	3.505E+00	3.039E+00
Kr-82	N,2N	8.416E-05	8.755E-05	8.459E-05	8.307E-05	8.744E-05
Kr-82	N,A	2.422E-06	2.518E-06	2.096E-06	2.182E-06	2.172E-06
Kr-82	N,P	2.089E-05	2.174E-05	1.821E-05	1.892E-05	1.888E-05
Kr-83	N,G	2.025E+01	1.664E+01	1.820E+01	1.959E+01	1.503E+01
Kr-83	N,2N	1.446E-03	1.504E-03	1.453E-03	1.427E-03	1.502E-03
Kr-83	N,A	2.762E-06	2.872E-06	2.393E-06	2.491E-06	2.481E-06
Kr-83	N,P	7.258E-05	7.560E-05	6.391E-05	6.619E-05	6.632E-05
Kr-84	N,G	1.275E-01	1.301E-01	1.330E-01	1.289E-01	1.356E-01
Kr-84	N,GX	4.852E-02	4.683E-02	4.868E-02	4.933E-02	4.711E-02
Kr-84	N,2N	1.338E-04	1.392E-04	1.345E-04	1.321E-04	1.390E-04
Kr-84	N,A	1.618E-05	1.687E-05	1.503E-05	1.529E-05	1.561E-05
Kr-84	N,P	2.154E-06	2.239E-06	1.863E-06	1.940E-06	1.930E-06
Kr-85	N,G	1.770E-01	1.484E-01	1.622E-01	1.724E-01	1.373E-01
Kr-86	N,G	9.914E-03	8.988E-03	9.610E-03	9.824E-03	8.828E-03
Kr-86	N,2N	1.834E-04	1.908E-04	1.844E-04	1.810E-04	1.906E-04
Kr-86	N,A	1.012E-05	1.056E-05	9.402E-06	9.569E-06	9.766E-06
Kr-86	N,P	4.573E-08	4.754E-08	3.953E-08	4.116E-08	4.095E-08
Kr-87	N,G	5.125E+01	4.043E+01	4.509E+01	4.933E+01	3.559E+01
Rb-85	N,G	2.545E-01	2.539E-01	2.595E-01	2.548E-01	2.596E-01
Rb-85	N,GX	2.225E-02	2.132E-02	2.221E-02	2.257E-02	2.136E-02
Rb-86	N,G	1.133E+00	1.066E+00	1.104E+00	1.119E+00	1.044E+00
Rb-87	N,G	7.784E-02	7.774E-02	7.859E-02	7.756E-02	7.850E-02
Rb-88	N,G	8.541E-02	6.738E-02	7.516E-02	8.222E-02	5.931E-02
Sr-86	N,G	3.811E-01	3.346E-01	3.587E-01	3.737E-01	3.181E-01
Sr-86	N,GX	1.539E-01	1.386E-01	1.475E-01	1.534E-01	1.339E-01
Sr-87	N,G	4.159E+00	3.857E+00	4.060E+00	4.194E+00	3.788E+00
Sr-88	N,G	1.212E-03	1.138E-03	1.143E-03	1.178E-03	1.078E-03
Sr-89	N,G	5.092E-02	4.394E-02	4.725E-02	4.968E-02	4.118E-02
Sr-90	N,G	8.324E-02	6.728E-02	7.411E-02	8.032E-02	6.007E-02
Y-89	N,G	1.275E-01	1.051E-01	1.154E-01	1.237E-01	9.581E-02
Y-89	N,GX	9.289E-05	7.485E-05	8.284E-05	8.991E-05	6.697E-05
Y-90	N,G	4.195E-01	3.607E-01	3.897E-01	4.097E-01	3.384E-01
Y-90	N,GX	1.475E-01	1.159E-01	1.295E-01	1.418E-01	1.018E-01
Y-91	N,G	1.594E-01	1.357E-01	1.474E-01	1.556E-01	1.267E-01
Zr-90	N,G	2.480E-02	2.366E-02	2.425E-02	2.447E-02	2.329E-02
Zr-90	N,A	7.803E-07	8.118E-07	7.843E-07	7.702E-07	8.107E-07
Zr-90	N,P	1.943E-04	2.021E-04	1.953E-04	1.918E-04	2.019E-04
Zr-91	N,G	2.748E-01	2.624E-01	2.717E-01	2.728E-01	2.611E-01
Zr-91	N,A	3.239E-04	3.370E-04	3.256E-04	3.197E-04	3.365E-04

Table A.3 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Zr-91	N,P	6.032E-05	6.275E-05	6.062E-05	5.953E-05	6.266E-05
Zr-92	N,G	1.106E-01	1.048E-01	1.093E-01	1.099E-01	1.045E-01
Zr-92	N,2N	1.923E-04	1.998E-04	1.658E-04	1.728E-04	1.717E-04
Zr-92	N,A	1.596E-05	1.660E-05	1.604E-05	1.575E-05	1.658E-05
Zr-92	N,P	1.025E-04	1.069E-04	9.129E-05	9.426E-05	9.478E-05
Zr-93	N,G	1.034E+00	1.010E+00	1.005E+00	1.013E+00	9.822E-01
Zr-93	N,A	1.079E-05	1.122E-05	1.084E-05	1.065E-05	1.121E-05
Zr-94	N,G	1.946E-02	1.903E-02	1.952E-02	1.937E-02	1.920E-02
Zr-94	N,A	3.019E-06	3.141E-06	3.034E-06	2.980E-06	3.137E-06
Zr-94	N,P	2.160E-06	2.247E-06	2.171E-06	2.132E-06	2.244E-06
Zr-95	N,G	2.343E-01	2.324E-01	2.387E-01	2.349E-01	2.376E-01
Zr-96	N,G	1.805E-01	1.860E-01	1.857E-01	1.808E-01	1.906E-01
Zr-96	N,A	6.480E-08	6.742E-08	6.514E-08	6.396E-08	6.733E-08
Zr-96	N,P	8.626E-08	8.973E-08	8.669E-08	8.514E-08	8.961E-08
Nb-93	N,G	4.115E-01	4.011E-01	4.131E-01	4.105E-01	4.047E-01
Nb-93	N,GX	9.395E-02	7.412E-02	8.267E-02	9.044E-02	6.524E-02
Nb-93	N,2N	3.237E-04	3.367E-04	3.253E-04	3.195E-04	3.363E-04
Nb-93	N,A	1.672E-05	1.740E-05	1.460E-05	1.516E-05	1.514E-05
Nb-93	N,P	1.587E-04	1.653E-04	1.393E-04	1.444E-04	1.446E-04
Nb-94	N,G	4.222E+00	3.981E+00	4.279E+00	4.311E+00	4.066E+00
Nb-95	N,G	8.591E-01	8.548E-01	8.666E-01	8.559E-01	8.632E-01
Mo-95	N,G	4.204E+00	3.991E+00	4.098E+00	4.166E+00	3.912E+00
Mo-95	N,A	7.985E-04	8.306E-04	8.025E-04	7.881E-04	8.295E-04
Mo-95	N,P	2.810E-05	2.923E-05	2.824E-05	2.773E-05	2.919E-05
Mo-96	N,G	6.928E-01	6.902E-01	6.873E-01	6.839E-01	6.832E-01
Mo-96	N,A	6.472E-05	6.733E-05	6.505E-05	6.388E-05	6.724E-05
Mo-96	N,P	4.102E-05	4.268E-05	4.123E-05	4.049E-05	4.262E-05
Mo-97	N,G	6.960E-01	6.722E-01	7.019E-01	6.992E-01	6.827E-01
Mo-97	N,A	2.805E-04	2.918E-04	2.820E-04	2.769E-04	2.915E-04
Mo-97	N,P	2.808E-05	2.921E-05	2.822E-05	2.771E-05	2.917E-05
Mo-98	N,G	1.297E+00	1.270E+00	1.298E+00	1.296E+00	1.275E+00
Mo-98	N,2N	4.230E-04	4.395E-04	3.645E-04	3.800E-04	3.775E-04
Mo-98	N,A	1.424E-05	1.482E-05	1.432E-05	1.406E-05	1.480E-05
Mo-98	N,P	1.725E-06	1.794E-06	1.733E-06	1.702E-06	1.792E-06
Mo-99	N,G	1.021E+00	1.013E+00	1.038E+00	1.026E+00	1.032E+00
Mo-100	N,G	1.504E-01	1.513E-01	1.546E-01	1.512E-01	1.558E-01
Mo-100	N,A	8.205E-07	8.535E-07	8.247E-07	8.098E-07	8.524E-07
Mo-100	N,P	2.155E-07	2.241E-07	2.166E-07	2.127E-07	2.238E-07
Tc-99	N,G	9.323E+00	8.997E+00	8.898E+00	9.180E+00	8.606E+00
Tc-99	N,2N	4.316E-04	4.490E-04	4.338E-04	4.260E-04	4.484E-04
Ru-99	N,G	4.261E+00	4.195E+00	4.343E+00	4.320E+00	4.277E+00
Ru-100	N,G	7.579E-01	6.622E-01	7.101E-01	7.420E-01	6.264E-01
Ru-101	N,G	2.978E+00	2.963E+00	3.104E+00	3.060E+00	3.094E+00
Ru-102	N,G	2.587E-01	2.409E-01	2.517E-01	2.556E-01	2.367E-01
Ru-103	N,G	2.662E+00	2.560E+00	2.656E+00	2.666E+00	2.567E+00
Ru-104	N,G	2.647E-01	2.642E-01	2.689E-01	2.647E-01	2.689E-01
Ru-105	N,G	2.962E-01	3.026E-01	3.066E-01	2.977E-01	3.130E-01

Table A.3 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Ru-106	N,G	9.014E-02	9.049E-02	9.425E-02	9.148E-02	9.511E-02
Rh-103	N,G	3.828E+01	3.505E+01	3.739E+01	3.873E+01	3.449E+01
Rh-103	N,GX	2.655E+00	2.454E+00	2.588E+00	2.666E+00	2.408E+00
Rh-103	N,2N	1.575E-04	1.639E-04	1.583E-04	1.555E-04	1.637E-04
Rh-104	N,G	3.416E+00	2.695E+00	3.006E+00	3.289E+00	2.372E+00
Rh-104m	N,G	6.833E+01	5.390E+01	6.012E+01	6.578E+01	4.745E+01
Rh-105	N,G	9.848E+02	8.187E+02	8.933E+02	9.582E+02	7.510E+02
Rh-105	N,GX	4.588E+02	3.739E+02	4.121E+02	4.454E+02	3.374E+02
Pd-104	N,G	6.567E-01	6.686E-01	6.712E-01	6.566E-01	6.815E-01
Pd-105	N,G	3.785E+00	3.568E+00	3.784E+00	3.817E+00	3.594E+00
Pd-106	N,G	2.791E-01	2.837E-01	2.863E-01	2.797E-01	2.908E-01
Pd-106	N,GX	6.513E-03	6.269E-03	6.523E-03	6.617E-03	6.298E-03
Pd-107	N,G	2.794E+00	2.648E+00	2.745E+00	2.780E+00	2.617E+00
Pd-108	N,G	7.078E+00	6.916E+00	7.120E+00	7.125E+00	6.965E+00
Pd-108	N,GX	1.080E-01	1.043E-01	1.084E-01	1.098E-01	1.049E-01
Pd-110	N,G	2.640E-01	2.684E-01	2.743E-01	2.665E-01	2.790E-01
Pd-110	N,GX	3.688E-02	3.646E-02	3.761E-02	3.777E-02	3.717E-02
Ag-107	N,G	6.114E+00	5.494E+00	5.901E+00	6.087E+00	5.357E+00
Ag-107	N,GX	3.973E-01	3.430E-01	3.703E-01	3.915E-01	3.223E-01
Ag-107	N,2N	3.237E-04	3.367E-04	3.253E-04	3.195E-04	3.363E-04
Ag-107	N,A	2.034E-05	2.114E-05	1.758E-05	1.830E-05	1.821E-05
Ag-107	N,P	1.933E-05	2.012E-05	1.691E-05	1.755E-05	1.754E-05
Ag-109	N,G	3.944E+01	3.778E+01	3.784E+01	3.912E+01	3.637E+01
Ag-109	N,GX	1.921E+00	1.838E+00	1.916E+00	1.948E+00	1.839E+00
Ag-109	N,2N	3.453E-04	3.592E-04	3.470E-04	3.408E-04	3.587E-04
Ag-109	N,A	1.864E-05	1.940E-05	1.619E-05	1.684E-05	1.679E-05
Ag-109	N,P	2.148E-05	2.237E-05	1.882E-05	1.952E-05	1.952E-05
Ag-110	N,G	7.004E+00	5.525E+00	6.163E+00	6.742E+00	4.863E+00
Ag-110	N,GX	8.541E-22	6.738E-22	7.516E-22	8.222E-22	5.931E-22
Ag-110m	N,G	7.004E+00	5.525E+00	6.163E+00	6.742E+00	4.863E+00
Ag-110m	N,GX	7.004E+00	5.525E+00	6.163E+00	6.742E+00	4.863E+00
Ag-111	N,G	3.513E+00	3.540E+00	3.624E+00	3.550E+00	3.651E+00
Cd-108	N,G	2.619E-01	2.485E-01	2.566E-01	2.590E-01	2.453E-01
Cd-109	N,G	5.552E+01	4.380E+01	4.885E+01	5.344E+01	3.855E+01
Cd-110	N,G	2.210E+00	2.039E+00	2.206E+00	2.231E+00	2.062E+00
Cd-110	N,GX	1.578E-02	1.397E-02	1.495E-02	1.566E-02	1.334E-02
Cd-111	N,G	3.454E+00	3.044E+00	3.252E+00	3.393E+00	2.890E+00
Cd-112	N,G	6.333E-01	6.068E-01	6.342E-01	6.351E-01	6.123E-01
Cd-112	N,GX	2.971E-01	2.960E-01	3.046E-01	3.049E-01	3.031E-01
Cd-113	N,G	3.891E+03	3.015E+03	3.422E+03	3.779E+03	2.625E+03
Cd-113	N,2N	1.036E-03	1.078E-03	1.041E-03	1.022E-03	1.076E-03
Cd-113	N,P	1.812E-06	1.882E-06	1.562E-06	1.628E-06	1.618E-06
Cd-114	N,G	6.570E-01	6.702E-01	6.576E-01	6.482E-01	6.693E-01
Cd-114	N,GX	5.129E-02	5.055E-02	5.219E-02	5.247E-02	5.145E-02
Cd-115m	N,G	8.181E+00	7.742E+00	8.068E+00	8.151E+00	7.682E+00
Cd-116	N,G	1.060E-01	1.082E-01	1.096E-01	1.067E-01	1.119E-01
Cd-116	N,GX	3.700E-03	3.211E-03	3.460E-03	3.652E-03	3.028E-03

Table A.3 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Cd-118	N,G	4.271E-02	3.369E-02	3.758E-02	4.111E-02	2.965E-02
Cd-118	N,GX	4.271E-02	3.369E-02	3.758E-02	4.111E-02	2.965E-02
In-113	N,G	6.976E+00	6.809E+00	7.098E+00	7.119E+00	6.949E+00
In-113	N,GX	4.771E+00	4.629E+00	4.804E+00	4.858E+00	4.670E+00
In-115	N,G	9.646E+01	9.207E+01	9.726E+01	9.910E+01	9.342E+01
In-115	N,GX	6.998E+01	6.705E+01	6.987E+01	7.100E+01	6.718E+01
In-117	N,G	4.270E-01	3.369E-01	3.758E-01	4.111E-01	2.965E-01
In-117	N,GX	4.270E-01	3.369E-01	3.758E-01	4.111E-01	2.965E-01
In-117m	N,G	4.270E-01	3.369E-01	3.758E-01	4.111E-01	2.965E-01
In-117m	N,GX	4.270E-01	3.369E-01	3.758E-01	4.111E-01	2.965E-01
In-119	N,G	3.416E-02	2.695E-02	3.006E-02	3.289E-02	2.372E-02
In-119	N,GX	5.125E-02	4.043E-02	4.509E-02	4.933E-02	3.559E-02
In-119m	N,G	3.416E-02	2.695E-02	3.006E-02	3.289E-02	2.372E-02
In-119m	N,GX	5.125E-02	4.043E-02	4.509E-02	4.933E-02	3.559E-02
In-120	N,G	1.361E-03	1.358E-03	1.397E-03	1.397E-03	1.392E-03
In-120m	N,G	1.361E-03	1.358E-03	1.397E-03	1.397E-03	1.392E-03
Sn-114	N,G	3.474E-02	3.289E-02	3.440E-02	3.512E-02	3.273E-02
Sn-114	N,A	4.130E-06	4.297E-06	4.151E-06	4.077E-06	4.291E-06
Sn-114	N,P	6.472E-05	6.733E-05	6.505E-05	6.388E-05	6.724E-05
Sn-115	N,G	4.319E+00	3.419E+00	3.809E+00	4.161E+00	3.017E+00
Sn-115	N,A	6.265E-05	6.517E-05	6.297E-05	6.183E-05	6.509E-05
Sn-115	N,P	6.468E-05	6.728E-05	6.500E-05	6.384E-05	6.719E-05
Sn-116	N,G	3.523E-01	3.591E-01	3.399E-01	3.403E-01	3.448E-01
Sn-116	N,GX	2.499E-01	2.494E-01	2.565E-01	2.566E-01	2.555E-01
Sn-116	N,A	4.497E-07	4.679E-07	4.520E-07	4.439E-07	4.672E-07
Sn-116	N,P	1.079E-05	1.123E-05	1.085E-05	1.065E-05	1.121E-05
Sn-117	N,G	7.608E-01	7.271E-01	7.447E-01	7.528E-01	7.154E-01
Sn-117	N,A	1.165E-05	1.212E-05	1.171E-05	1.150E-05	1.211E-05
Sn-117	N,P	1.942E-05	2.020E-05	1.952E-05	1.917E-05	2.018E-05
Sn-118	N,G	2.305E-01	2.360E-01	2.361E-01	2.309E-01	2.413E-01
Sn-118	N,GX	1.600E-01	1.594E-01	1.640E-01	1.642E-01	1.632E-01
Sn-118	N,A	4.402E-08	4.580E-08	4.425E-08	4.345E-08	4.574E-08
Sn-118	N,P	2.157E-06	2.244E-06	2.168E-06	2.129E-06	2.241E-06
Sn-119	N,G	2.934E-01	2.550E-01	2.735E-01	2.868E-01	2.399E-01
Sn-119	N,A	1.877E-06	1.952E-06	1.886E-06	1.852E-06	1.950E-06
Sn-119	N,P	4.316E-06	4.490E-06	4.338E-06	4.260E-06	4.484E-06
Sn-120	N,G	5.897E-02	5.814E-02	5.967E-02	5.900E-02	5.910E-02
Sn-120	N,GX	3.168E-04	2.984E-04	3.127E-04	3.198E-04	2.960E-04
Sn-120	N,A	4.316E-07	4.490E-07	4.338E-07	4.260E-07	4.484E-07
Sn-122	N,G	4.182E-02	3.945E-02	4.017E-02	4.095E-02	3.810E-02
Sn-122	N,GX	1.497E-04	1.316E-04	1.412E-04	1.483E-04	1.252E-04
Sn-122	N,A	8.632E-08	8.980E-08	8.676E-08	8.520E-08	8.968E-08
Sn-123	N,G	1.104E-01	1.139E-01	1.155E-01	1.116E-01	1.190E-01
Sn-124	N,G	2.269E-01	2.294E-01	2.404E-01	2.341E-01	2.434E-01
Sn-124	N,GX	1.616E-01	1.590E-01	1.642E-01	1.652E-01	1.617E-01
Sn-124	N,A	2.158E-08	2.245E-08	2.169E-08	2.130E-08	2.242E-08
Sn-125	N,G	5.539E-01	5.620E-01	5.833E-01	5.636E-01	5.932E-01

Table A.3 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Sn-126	N,G	2.943E-02	2.418E-02	2.635E-02	2.842E-02	2.174E-02
Sn-126	N,GX	1.262E-02	9.914E-03	1.108E-02	1.214E-02	8.699E-03
Sb-121	N,G	5.539E+00	5.459E+00	5.536E+00	5.583E+00	5.464E+00
Sb-121	N,GX	4.406E-02	4.299E-02	4.453E-02	4.494E-02	4.352E-02
Sb-123	N,G	3.331E+00	3.278E+00	3.166E+00	3.231E+00	3.111E+00
Sb-123	N,GX	8.751E-03	8.542E-03	8.847E-03	8.927E-03	8.648E-03
Sb-124	N,G	1.359E+00	1.265E+00	1.328E+00	1.348E+00	1.247E+00
Sb-125	N,G	6.994E-01	7.002E-01	7.072E-01	6.973E-01	7.083E-01
Sb-125	N,GX	6.569E-02	5.163E-02	5.767E-02	6.318E-02	4.531E-02
Sb-126	N,G	1.871E+00	1.801E+00	1.872E+00	1.873E+00	1.812E+00
Te-122	N,G	2.472E+00	2.474E+00	2.591E+00	2.537E+00	2.603E+00
Te-122	N,GX	7.956E-01	7.745E-01	8.028E-01	8.109E-01	7.829E-01
Te-123	N,G	1.689E+02	1.610E+02	1.693E+02	1.725E+02	1.621E+02
Te-124	N,G	8.081E-01	6.940E-01	7.474E-01	7.879E-01	6.477E-01
Te-124	N,GX	3.938E-03	3.216E-03	3.542E-03	3.824E-03	2.906E-03
Te-125	N,G	9.065E-01	9.008E-01	9.177E-01	9.062E-01	9.130E-01
Te-126	N,G	4.285E-01	4.195E-01	4.262E-01	4.250E-01	4.178E-01
Te-126	N,GX	3.973E-02	3.724E-02	3.909E-02	4.006E-02	3.685E-02
Te-127m	N,G	2.033E+00	1.894E+00	1.990E+00	2.021E+00	1.869E+00
Te-128	N,G	1.158E-01	1.155E-01	1.167E-01	1.153E-01	1.166E-01
Te-128	N,GX	3.506E-03	3.231E-03	3.411E-03	3.518E-03	3.165E-03
Te-129m	N,G	2.892E-01	2.760E-01	2.870E-01	2.877E-01	2.758E-01
Te-130	N,G	3.462E-02	2.977E-02	3.210E-02	3.378E-02	2.788E-02
Te-130	N,GX	2.459E-03	2.097E-03	2.274E-03	2.416E-03	1.954E-03
Te-132	N,G	4.757E-04	4.526E-04	4.477E-04	4.608E-04	4.274E-04
Te-134	N,G	8.541E-03	6.738E-03	7.516E-03	8.222E-03	5.931E-03
I-127	N,G	4.885E+00	4.847E+00	4.920E+00	4.895E+00	4.885E+00
I-129	N,G	3.116E+00	2.653E+00	2.873E+00	3.039E+00	2.468E+00
I-129	N,GX	1.898E+00	1.573E+00	1.723E+00	1.850E+00	1.436E+00
I-130	N,G	6.658E+00	6.430E+00	6.648E+00	6.665E+00	6.446E+00
I-131	N,G	3.257E-01	3.221E-01	3.320E-01	3.268E-01	3.296E-01
I-135	N,G	2.025E-03	1.677E-03	1.817E-03	1.954E-03	1.511E-03
Xe-128	N,G	6.445E-01	5.921E-01	6.191E-01	6.341E-01	5.730E-01
Xe-128	N,GX	4.538E-02	3.886E-02	4.207E-02	4.462E-02	3.631E-02
Xe-128	N,2N	3.453E-04	3.592E-04	3.470E-04	3.408E-04	3.587E-04
Xe-128	N,A	5.797E-07	6.032E-07	5.045E-07	5.243E-07	5.231E-07
Xe-128	N,P	7.714E-07	8.033E-07	6.798E-07	7.039E-07	7.053E-07
Xe-129	N,G	8.218E+00	7.873E+00	8.401E+00	8.417E+00	8.087E+00
Xe-129	N,2N	3.021E-03	3.143E-03	3.037E-03	2.982E-03	3.139E-03
Xe-129	N,A	4.265E-06	4.444E-06	3.760E-06	3.893E-06	3.902E-06
Xe-129	N,P	2.293E-06	2.388E-06	2.018E-06	2.090E-06	2.094E-06
Xe-130	N,G	5.981E-01	4.891E-01	5.374E-01	5.789E-01	4.418E-01
Xe-130	N,GX	5.245E-02	4.485E-02	4.858E-02	5.156E-02	4.187E-02
Xe-130	N,2N	4.748E-04	4.939E-04	4.772E-04	4.686E-04	4.932E-04
Xe-130	N,A	2.127E-07	2.213E-07	1.846E-07	1.920E-07	1.913E-07
Xe-130	N,P	1.841E-06	1.917E-06	1.621E-06	1.679E-06	1.682E-06
Xe-131	N,G	3.018E+01	2.866E+01	3.086E+01	3.104E+01	2.952E+01

Table A.3 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Xe-131	N,2N	4.532E-03	4.715E-03	4.555E-03	4.473E-03	4.708E-03
Xe-131	N,A	1.483E-06	1.544E-06	1.299E-06	1.347E-06	1.347E-06
Xe-131	N,P	2.727E-06	2.841E-06	2.444E-06	2.517E-06	2.536E-06
Xe-132	N,G	1.016E-01	9.584E-02	9.957E-02	1.006E-01	9.464E-02
Xe-132	N,GX	3.057E-03	2.605E-03	2.825E-03	3.002E-03	2.426E-03
Xe-132	N,2N	7.121E-04	7.408E-04	7.158E-04	7.029E-04	7.399E-04
Xe-132	N,A	6.108E-08	6.350E-08	5.282E-08	5.500E-08	5.473E-08
Xe-132	N,P	4.104E-07	4.272E-07	3.596E-07	3.729E-07	3.729E-07
Xe-133	N,G	2.355E+01	2.019E+01	2.178E+01	2.304E+01	1.882E+01
Xe-134	N,G	4.338E-02	3.975E-02	4.178E-02	4.275E-02	3.868E-02
Xe-134	N,GX	3.120E-04	2.578E-04	2.827E-04	3.039E-04	2.350E-04
Xe-134	N,2N	9.927E-04	1.033E-03	9.977E-04	9.798E-04	1.031E-03
Xe-134	N,A	2.641E-08	2.744E-08	2.277E-08	2.373E-08	2.358E-08
Xe-134	N,P	4.751E-07	4.949E-07	4.178E-07	4.329E-07	4.335E-07
Xe-135	N,G	2.215E+05	1.640E+05	1.825E+05	2.057E+05	1.328E+05
Xe-136	N,G	1.593E-02	1.313E-02	1.441E-02	1.545E-02	1.196E-02
Xe-136	N,2N	1.230E-03	1.280E-03	1.236E-03	1.214E-03	1.278E-03
Xe-136	N,A	1.133E-08	1.177E-08	9.763E-09	1.018E-08	1.011E-08
Xe-136	N,P	2.078E-08	2.161E-08	1.799E-08	1.873E-08	1.865E-08
Cs-133	N,G	1.059E+01	1.010E+01	1.054E+01	1.073E+01	1.009E+01
Cs-133	N,GX	9.995E-01	9.530E-01	9.947E-01	1.013E+00	9.523E-01
Cs-133	N,2N	4.316E-03	4.490E-03	4.338E-03	4.260E-03	4.484E-03
Cs-134	N,G	1.128E+01	8.754E+00	9.823E+00	1.081E+01	7.607E+00
Cs-134	N,GX	5.787E+00	4.579E+00	5.102E+00	5.575E+00	4.040E+00
Cs-134m	N,G	1.110E-01	8.759E-02	9.770E-02	1.069E-01	7.710E-02
Cs-134m	N,GX	1.110E-01	8.759E-02	9.770E-02	1.069E-01	7.710E-02
Cs-135	N,G	2.480E+00	2.320E+00	2.437E+00	2.499E+00	2.293E+00
Cs-136	N,G	1.821E+00	1.795E+00	1.853E+00	1.863E+00	1.827E+00
Cs-137	N,G	2.546E-02	2.409E-02	2.558E-02	2.554E-02	2.450E-02
Cs-141	N,G	1.409E-03	1.112E-03	1.240E-03	1.357E-03	9.786E-04
Ba-134	N,G	8.582E-01	8.363E-01	8.261E-01	8.365E-01	8.054E-01
Ba-134	N,GX	5.309E-02	5.017E-02	5.252E-02	5.365E-02	4.988E-02
Ba-135	N,G	3.494E+00	3.445E+00	3.575E+00	3.539E+00	3.534E+00
Ba-135	N,GX	6.490E-03	6.230E-03	6.488E-03	6.588E-03	6.248E-03
Ba-136	N,G	1.015E-01	9.643E-02	1.002E-01	1.008E-01	9.596E-02
Ba-136	N,GX	1.489E-03	1.308E-03	1.403E-03	1.474E-03	1.243E-03
Ba-137	N,G	5.291E-01	4.403E-01	4.809E-01	5.137E-01	4.030E-01
Ba-138	N,G	3.393E-02	2.780E-02	3.027E-02	3.273E-02	2.487E-02
Ba-139	N,G	5.125E-01	4.043E-01	4.509E-01	4.933E-01	3.559E-01
Ba-140	N,G	5.297E-01	5.126E-01	5.247E-01	5.252E-01	5.096E-01
La-138	N,G	2.230E+01	1.919E+01	2.074E+01	2.196E+01	1.799E+01
La-139	N,G	1.009E+00	8.525E-01	9.344E-01	9.894E-01	7.982E-01
La-140	N,G	2.243E+00	2.247E+00	2.300E+00	2.261E+00	2.305E+00
Ce-140	N,G	6.048E-02	5.070E-02	5.480E-02	5.858E-02	4.621E-02
Ce-141	N,G	2.843E+00	2.330E+00	2.561E+00	2.756E+00	2.111E+00
Ce-142	N,G	1.020E-01	8.569E-02	9.295E-02	9.900E-02	7.870E-02
Ce-143	N,G	1.753E+00	1.681E+00	1.718E+00	1.730E+00	1.653E+00

Table A.3 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Ce-144	N,G	1.457E-01	1.299E-01	1.392E-01	1.437E-01	1.256E-01
Pr-141	N,G	1.462E+00	1.268E+00	1.364E+00	1.430E+00	1.194E+00
Pr-141	N,GX	3.793E-01	3.089E-01	3.406E-01	3.681E-01	2.786E-01
Pr-141	N,2N	4.100E-04	4.265E-04	4.121E-04	4.047E-04	4.260E-04
Pr-141	N,A	3.392E-06	3.530E-06	2.959E-06	3.073E-06	3.069E-06
Pr-141	N,P	3.160E-06	3.290E-06	2.762E-06	2.867E-06	2.865E-06
Pr-142	N,G	5.538E+00	5.235E+00	5.423E+00	5.507E+00	5.153E+00
Pr-143	N,G	1.177E+01	1.022E+01	1.093E+01	1.150E+01	9.558E+00
Nd-142	N,G	1.623E+00	1.287E+00	1.432E+00	1.563E+00	1.136E+00
Nd-143	N,G	2.710E+01	2.138E+01	2.380E+01	2.602E+01	1.877E+01
Nd-143	N,2N	6.258E-03	6.510E-03	6.290E-03	6.177E-03	6.502E-03
Nd-143	N,A	4.925E-05	5.134E-05	4.409E-05	4.545E-05	4.578E-05
Nd-143	N,P	2.386E-06	2.483E-06	2.083E-06	2.163E-06	2.160E-06
Nd-144	N,G	4.575E-01	3.977E-01	4.257E-01	4.466E-01	3.733E-01
Nd-145	N,G	9.165E+00	8.463E+00	8.826E+00	9.118E+00	8.210E+00
Nd-145	N,2N	8.200E-03	8.531E-03	8.242E-03	8.094E-03	8.520E-03
Nd-145	N,A	1.747E-05	1.820E-05	1.549E-05	1.601E-05	1.608E-05
Nd-145	N,P	7.149E-07	7.438E-07	6.215E-07	6.461E-07	6.443E-07
Nd-146	N,G	2.345E-01	2.136E-01	2.242E-01	2.305E-01	2.060E-01
Nd-146	N,2N	2.374E-03	2.469E-03	2.386E-03	2.343E-03	2.466E-03
Nd-146	N,A	2.159E-06	2.248E-06	1.885E-06	1.957E-06	1.954E-06
Nd-146	N,P	5.553E-08	5.770E-08	4.789E-08	4.991E-08	4.961E-08
Nd-147	N,G	1.991E+01	1.911E+01	1.980E+01	2.004E+01	1.908E+01
Nd-148	N,G	8.669E-01	8.406E-01	8.532E-01	8.565E-01	8.293E-01
Nd-148	N,2N	2.590E-04	2.694E-04	2.603E-04	2.556E-04	2.690E-04
Nd-148	N,A	6.474E-07	6.735E-07	5.624E-07	5.848E-07	5.830E-07
Nd-148	N,P	1.289E-08	1.340E-08	1.111E-08	1.158E-08	1.151E-08
Nd-150	N,G	6.503E-01	6.441E-01	6.730E-01	6.614E-01	6.696E-01
Nd-150	N,2N	2.590E-03	2.694E-03	2.603E-03	2.556E-03	2.690E-03
Nd-150	N,A	1.570E-07	1.633E-07	1.359E-07	1.415E-07	1.409E-07
Nd-150	N,P	8.406E-09	8.733E-09	7.243E-09	7.550E-09	7.501E-09
Pm-147	N,G	6.272E+01	5.962E+01	5.964E+01	6.183E+01	5.691E+01
Pm-147	N,GX	2.966E+01	2.809E+01	2.938E+01	2.999E+01	2.796E+01
Pm-147	N,2N	2.158E-03	2.245E-03	2.169E-03	2.130E-03	2.242E-03
Pm-147	N,A	5.610E-06	5.842E-06	4.922E-06	5.103E-06	5.106E-06
Pm-147	N,P	1.492E-06	1.553E-06	1.301E-06	1.351E-06	1.349E-06
Pm-148	N,G	1.258E+03	1.201E+03	1.280E+03	1.304E+03	1.225E+03
Pm-148m	N,G	2.810E+03	2.199E+03	2.478E+03	2.730E+03	1.921E+03
Pm-149	N,G	1.259E+02	1.009E+02	1.118E+02	1.215E+02	8.982E+01
Pm-151	N,G	9.942E+01	8.664E+01	9.337E+01	9.836E+01	8.208E+01
Sm-147	N,G	2.400E+01	2.304E+01	2.340E+01	2.383E+01	2.252E+01
Sm-147	N,2N	5.193E-03	5.399E-03	4.494E-03	4.679E-03	4.657E-03
Sm-147	N,A	1.069E-04	1.101E-04	1.014E-04	1.023E-04	1.040E-04
Sm-147	N,P	5.774E-06	5.967E-06	5.268E-06	5.389E-06	5.424E-06
Sm-148	N,G	1.142E+00	1.119E+00	1.146E+00	1.141E+00	1.128E+00
Sm-149	N,G	6.663E+03	5.002E+03	5.532E+03	6.219E+03	4.083E+03
Sm-149	N,2N	3.237E-03	3.367E-03	3.253E-03	3.195E-03	3.363E-03

Table A.3 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-USO	BWR-UE
Sm-149	N,A	4.092E-05	4.261E-05	3.590E-05	3.722E-05	3.725E-05
Sm-149	N,P	4.092E-05	4.261E-05	3.590E-05	3.722E-05	3.725E-05
Sm-150	N,G	1.480E+01	1.307E+01	1.349E+01	1.427E+01	1.195E+01
Sm-151	N,G	6.597E+02	4.999E+02	5.558E+02	6.169E+02	4.214E+02
Sm-151	N,2N	1.101E-02	1.145E-02	1.106E-02	1.086E-02	1.143E-02
Sm-151	N,A	9.048E-06	9.428E-06	8.003E-06	8.278E-06	8.306E-06
Sm-151	N,P	7.949E-07	8.272E-07	6.928E-07	7.197E-07	7.184E-07
Sm-152	N,G	7.447E+01	7.080E+01	7.201E+01	7.429E+01	6.877E+01
Sm-152	N,2N	1.316E-03	1.369E-03	1.323E-03	1.299E-03	1.368E-03
Sm-152	N,A	4.949E-07	5.147E-07	4.289E-07	4.463E-07	4.445E-07
Sm-152	N,P	6.920E-08	7.192E-08	5.976E-08	6.225E-08	6.190E-08
Sm-153	N,G	8.881E+01	8.278E+01	8.577E+01	8.855E+01	8.040E+01
Sm-154	N,G	1.513E+00	1.440E+00	1.541E+00	1.537E+00	1.480E+00
Eu-151	N,G	7.275E+02	5.971E+02	6.724E+02	7.209E+02	5.556E+02
Eu-151	N,GX	2.753E+02	2.158E+02	2.413E+02	2.646E+02	1.890E+02
Eu-151	N,2N	8.200E-04	8.531E-04	8.242E-04	8.094E-04	8.520E-04
Eu-151	N,A	3.670E-05	3.826E-05	3.356E-05	3.433E-05	3.485E-05
Eu-151	N,P	1.802E-05	1.876E-05	1.589E-05	1.645E-05	1.648E-05
Eu-152	N,G	1.879E+02	1.621E+02	1.739E+02	1.838E+02	1.518E+02
Eu-152	N,2N	3.021E-03	3.143E-03	3.037E-03	2.982E-03	3.139E-03
Eu-152	N,A	4.768E-05	4.970E-05	4.338E-05	4.444E-05	4.503E-05
Eu-152	N,P	6.136E-05	6.387E-05	5.462E-05	5.638E-05	5.663E-05
Eu-153	N,G	5.568E+01	5.121E+01	5.388E+01	5.552E+01	4.995E+01
Eu-153	N,2N	6.945E-04	7.215E-04	5.984E-04	6.238E-04	6.197E-04
Eu-153	N,A	2.461E-05	2.562E-05	2.284E-05	2.326E-05	2.369E-05
Eu-153	N,P	2.548E-06	2.654E-06	2.272E-06	2.343E-06	2.358E-06
Eu-154	N,G	1.250E+02	1.083E+02	1.162E+02	1.226E+02	1.020E+02
Eu-154	N,2N	2.029E-03	2.110E-03	2.039E-03	2.002E-03	2.107E-03
Eu-154	N,A	6.421E-05	6.694E-05	5.881E-05	6.012E-05	6.107E-05
Eu-154	N,P	9.254E-05	9.641E-05	8.223E-05	8.488E-05	8.534E-05
Eu-155	N,G	3.466E+02	2.738E+02	3.052E+02	3.338E+02	2.412E+02
Eu-155	N,2N	1.662E-03	1.729E-03	1.670E-03	1.640E-03	1.726E-03
Eu-155	N,A	3.571E-07	3.713E-07	3.094E-07	3.220E-07	3.207E-07
Eu-155	N,P	2.951E-07	3.070E-07	2.565E-07	2.667E-07	2.660E-07
Eu-156	N,G	7.154E+01	6.275E+01	6.747E+01	7.095E+01	5.971E+01
Eu-157	N,G	4.521E+01	4.172E+01	4.411E+01	4.551E+01	4.101E+01
Gd-152	N,G	1.195E+01	1.181E+01	1.214E+01	1.209E+01	1.201E+01
Gd-152	N,2N	4.437E-04	4.610E-04	3.823E-04	3.986E-04	3.960E-04
Gd-152	N,A	2.710E-06	2.824E-06	2.439E-06	2.509E-06	2.532E-06
Gd-152	N,P	4.213E-06	4.391E-06	3.756E-06	3.877E-06	3.899E-06
Gd-154	N,G	1.315E+01	1.173E+01	8.619E+00	8.725E+00	8.327E+00
Gd-154	N,GX	6.536E-03	5.643E-03	6.092E-03	6.441E-03	5.302E-03
Gd-155	N,G	2.451E+03	1.785E+03	2.961E+02	3.277E+02	2.375E+02
Gd-155	N,A	7.601E-06	5.997E-06	6.689E-06	7.318E-06	5.279E-06
Gd-156	N,G	3.798E+00	3.823E+00	3.585E+00	3.562E+00	3.609E+00
Gd-157	N,G	1.041E+04	7.551E+03	1.142E+03	1.278E+03	8.890E+02
Gd-157	N,A	4.698E-05	3.706E-05	4.134E-05	4.522E-05	3.262E-05

Table A.3 (continued)

Nuclide	Reaction[a]	PWR-US	PWR-UE	BWR-US	BWR-US0	BWR-UE
Gd-158	N,G	1.794E+00	1.771E+00	1.538E+00	1.552E+00	1.542E+00
Gd-160	N,G	3.728E-01	3.692E-01	3.388E-01	3.314E-01	3.447E-01
Gd-161	N,G	2.648E+03	2.089E+03	2.330E+03	2.549E+03	1.839E+03
Tb-159	N,G	1.477E+01	1.443E+01	1.503E+01	1.500E+01	1.472E+01
Tb-159	N,P	1.708E+00	1.348E+00	1.503E+00	1.644E+00	1.186E+00
Tb-160	N,G	6.617E+01	5.666E+01	6.141E+01	6.516E+01	5.303E+01
Dy-160	N,G	4.857E+01	4.747E+01	4.942E+01	4.951E+01	4.840E+01
Dy-160	N,A	2.553E-05	2.014E-05	2.246E-05	2.458E-05	1.773E-05
Dy-161	N,G	7.320E+01	6.331E+01	6.814E+01	7.189E+01	5.945E+01
Dy-161	N,A	2.548E-06	2.010E-06	2.242E-06	2.453E-06	1.769E-06
Dy-162	N,G	7.596E+01	7.234E+01	7.236E+01	7.504E+01	6.916E+01
Dy-163	N,G	4.788E+01	4.547E+01	4.752E+01	4.845E+01	4.533E+01
Dy-163	N,A	1.665E-06	1.314E-06	1.466E-06	1.603E-06	1.157E-06
Dy-164	N,G	7.837E+01	6.035E+01	6.793E+01	7.499E+01	5.211E+01
Dy-164	N,GX	1.332E+02	1.026E+02	1.155E+02	1.275E+02	8.859E+01
Dy-164	N,2N	1.662E-03	1.729E-03	1.670E-03	1.640E-03	1.726E-03
Dy-165	N,G	7.923E+02	7.211E+02	7.644E+02	7.922E+02	7.010E+02
Ho-165	N,G	1.978E+01	1.862E+01	1.951E+01	1.997E+01	1.847E+01
Ho-165	N,GX	1.098E+00	1.033E+00	1.083E+00	1.108E+00	1.025E+00
Ho-165	N,A	1.708E-06	1.348E-06	1.503E-06	1.644E-06	1.186E-06
Ho-165	N,P	1.483E+00	1.170E+00	1.305E+00	1.427E+00	1.030E+00
Er-166	N,G	2.800E+00	2.437E+00	2.624E+00	2.766E+00	2.303E+00
Er-166	N,GX	2.100E+00	1.828E+00	1.968E+00	2.074E+00	1.727E+00
Er-166	N,A	5.979E-06	4.716E-06	5.261E-06	5.755E-06	4.152E-06
Er-167	N,G	1.177E+02	1.056E+02	1.125E+02	1.172E+02	1.016E+02
Er-167	N,A	5.722E-06	4.514E-06	5.035E-06	5.509E-06	3.974E-06
Er-168	N,G	9.631E-01	9.266E-01	9.642E-01	9.783E-01	9.305E-01
Er-168	N,A	7.661E-06	6.044E-06	6.741E-06	7.375E-06	5.320E-06
Er-170	N,G	8.823E-01	7.788E-01	8.343E-01	8.747E-01	7.426E-01
Er-171	N,G	2.391E+01	1.887E+01	2.104E+01	2.302E+01	1.661E+01
Tm-169	N,G	4.356E+01	4.179E+01	4.353E+01	4.421E+01	4.190E+01
Tm-169	N,GX	3.199E+00	3.049E+00	3.183E+00	3.241E+00	3.045E+00
Tm-169	N,A	8.541E-07	6.738E-07	7.516E-07	8.222E-07	5.931E-07
Tm-170	N,G	1.735E+01	1.568E+01	1.666E+01	1.731E+01	1.517E+01
Tm-171	N,G	3.015E+00	2.929E+00	3.038E+00	3.071E+00	2.958E+00
Yb-168	N,G	9.640E+02	9.003E+02	9.461E+02	9.709E+02	8.888E+02
Yb-168	N,A	3.260E-04	2.572E-04	2.869E-04	3.138E-04	2.264E-04
Yb-170	N,G	7.556E+00	7.364E+00	7.631E+00	7.704E+00	7.449E+00
Yb-170	N,A	3.416E-06	2.695E-06	3.006E-06	3.289E-06	2.372E-06
Yb-171	N,G	1.129E+01	1.038E+01	1.096E+01	1.132E+01	1.015E+01
Yb-171	N,A	3.416E-06	2.695E-06	3.006E-06	3.289E-06	2.372E-06
Yb-172	N,G	6.648E-01	6.403E-01	6.661E-01	6.755E-01	6.435E-01
Yb-172	N,A	2.554E-06	2.015E-06	2.247E-06	2.458E-06	1.773E-06

[a] N,G = (n,γ) to a ground state. N,GX = (n,γ) to an excited state.
N,3N = (n,3n) to a ground state. N,F = (n,fission) reaction.
N,2N = (n,2n) to a ground state. N,2NX = (n,2n) to an excited state.
N,A = (n,α) to a ground state. N,P = (n,p) to a ground state.

Appendix B.

LISTINGS OF BURNUP-DEPENDENT CROSS SECTIONS FOR THE ORIGEN2 COMPUTER CODE

In order to adjust certain actinide cross sections considered most significant in uranium-cycle depletion calculations, the ORIGEN2 code has a series of subroutines (one for each reactor model) that provide the variable cross-section data at points along the burnup curve. At each irradiation step (IRP command), the variable cross sections are adjusted by a simple curve fit. The variation of these actinide cross sections during fuel depletion is caused by the change in the concentration of the fissile radionuclides within the oxide fuel. For uranium-enriched fuels, the burnup of the fuel (and depletion of the fissile nuclides) is based solely on the fissioning of uranium isotopes in fresh fuel. As the burnup continues, an increasing fraction of the total burnup is caused by the fissioning of other isotopes, primarily plutonium. The neutron fission energy spectrum also changes over the burnup range. Since ORIGEN2 calculations are independent of neutron energy spectrum, the variable cross-section data are provided to represent this phenomenon. For burnups that extend beyond the maximum burnup provided in the original model, the cross section is no longer adjusted and thus remains constant. For this reason, the user should avoid using an ORIGEN2 reactor model to achieve a burnup that exceeds the original design burnup of the reactor model, because results at higher burnups would tend to be progressively less defensible.

Tables B.1 through B.5 list the variable cross sections for actinide reactions that are assumed to be most important in uranium-cycle depletion calculations. (These tables are included at the end of this appendix.) Note that only 7 of the 20 actinide cross-section reactions for the five updated reactor models (i.e., PWR-US, PWR-UE, BWR-US, BWR-US0, and BWR-UE) listed in these tables are provided with burnup-dependent values. The list of 20 reactions was developed for use in the previous ORIGEN2 LWR models[7,8] (i.e., PWR-U, PUD50, and BWR-U). During development of the updated reactor models, only the seven

actinide cross-section reactions noted in Tables B.1 through B.5 were observed to show significant variation with burnup to be included with the updated reactor models. The remaining 13 cross sections exhibited insignificant variation with burnup because they were either at constant (or nearly constant) concentration, as in the case of ^{238}U, or at infinite dilution, with number densities less than 10^{-5}, and thus did not affect the neutron flux. Therefore, these 13 reactions were not included in the variable cross-section data provided in the ORIGEN2 code.

Another trivial difference between the previous and updated LWR reactor models, at least from the standpoint of performing calculations with ORIGEN2 using variable actinide cross-section data, should be noted. The previous LWR model cross-section data libraries contain "midcycle" values for the 20 selected actinide cross sections, whereas the values in the variable cross-section subroutines are provided for burnups from zero through maximum. The actinide cross-section values in the updated LWR cross-section data libraries are provided at zero (not "midcycle") burnup. This interpretative difference (of the variable actinide cross-section data) between the previous and the updated LWR models is not important when the variable cross-section data option is employed in ORIGEN2 calculations but could result in wide variation in results when this option is not used.

Table B.1. Cross sections as a function of burnup for a PWR-US

Nuclide	Type of cross-section reaction[a]	Cross section (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of:				
		0.0	2.0	4.0	6.0	8.0
^{234}U	N,G	2.071E+1	Not included[b]			
^{235}U	N,G	1.068E+1	Not included[b]			
^{235}U	N,F	4.752E+1	Not included[b]			
^{236}U	N,G	8.348E+0	Not included[b]			
^{238}U	N,G	8.872E-1	Not included[b]			
^{237}Np	N,G	3.328E+1	Not included[b]			
^{238}Pu	N,G	3.483E+1	Not included[b]			
^{238}Pu	N,F	2.308E+0	Not included[b]			
^{239}Pu	N,G	6.909E+1	6.431E+1	5.993E+1	5.585E+1	5.205E+1
^{239}Pu	N,F	1.211E+2	1.137E+2	1.067E+2	1.002E+2	9.407E+1
^{240}Pu	N,G	2.228E+2	1.762E+2	1.394E+2	1.102E+2	8.717E+1
^{241}Pu	N,G	4.202E+1	4.023E+1	3.851E+1	3.687E+1	3.529E+1
^{241}Pu	N,F	1.259E+2	1.214E+2	1.170E+2	1.128E+2	1.087E+2
^{242}Pu	N,G	3.320E+1	3.240E+1	3.163E+1	3.087E+1	3.013E+1
^{241}Am	N,G	9.570E+1	9.028E+1	8.517E+1	8.035E+1	7.581E+1
^{241}Am	N,GX	2.392E+1	Not included[b]			
^{243}Am	N,G	2.518E+0	Not included[b]			
^{243}Am	N,GX	4.784E+1	Not included[b]			
^{242}Cm	N,G	5.677E+0	Not included[b]			
^{244}Cm	N,G	1.281E+1	Not included[b]			

[a]N,G = (n,γ) reaction to a ground state.
N,GX = (n,γ) reaction to an excited state.
N,F = (n, fission) reaction.
[b]Not included - These types of cross-section reactions did not exhibit significant variation with burnup and have not been included.

Table B.2. Cross sections as a function of burnup for a PWR-UE

Nuclide	Type of cross-section reaction[a]	Cross section (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of:			
		0.0	2.0	4.0	6.0
^{234}U	N,G	1.911E+1	Not included[b]		
^{235}U	N,G	8.910E+0	Not included[b]		
^{235}U	N,F	3.775E+1	Not included[b]		
^{236}U	N,G	8.295E+0	Not included[b]		
^{238}U	N,G	8.546E-1	Not included[b]		
^{237}Np	N,G	3.031E+1	Not included[b]		
^{238}Pu	N,G	2.690E+1	Not included[b]		
^{238}Pu	N,F	2.119E+0	Not included[b]		
^{239}Pu	N,G	5.753E+1	5.361E+1	4.996E+1	4.656E+1
^{239}Pu	N,F	9.922E+1	9.315E+1	8.745E+1	8.210E+1
^{240}Pu	N,G	2.135E+2	1.689E+2	1.335E+2	1.056E+2
^{241}Pu	N,G	3.363E+1	3.219E+1	3.082E+1	2.951E+1
^{241}Pu	N,F	1.013E+2	9.765E+1	9.412E+1	9.073E+1
^{242}Pu	N,G	3.290E+1	3.211E+1	3.134E+1	3.059E+1
^{241}Am	N,G	8.303E+1	7.833E+1	7.390E+1	6.972E+1
^{241}Am	N,GX	2.076E+1	Not included[b]		
^{243}Am	N,G	2.434E+0	Not included[b]		
^{243}Am	N,GX	4.625E+1	Not included[b]		
^{242}Cm	N,G	5.439E+0	Not included[b]		
^{244}Cm	N,G	1.271E+1	Not included[b]		

Table B.2 (continued)

Nuclide	Type of cross-section reaction[a]	Cross section (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of:			
		8.0	10.0	12.0	14.0
^{234}U	N,G		Not included[b]		
^{235}U	N,G		Not included[b]		
^{235}U	N,F		Not included[b]		
^{236}U	N,G		Not included[b]		
^{238}U	N,G		Not included[b]		
^{237}Np	N,G		Not included[b]		
^{238}Pu	N,G		Not included[b]		
^{238}Pu	N,F		Not included[b]		
^{239}Pu	N,G	4.339E+1	4.044E+1	3.768E+1	3.512E+1
^{239}Pu	N,F	7.707E+1	7.236E+1	6.793E+1	6.377E+1
^{240}Pu	N,G	8.353E+1	6.606E+1	5.225E+1	4.132E+1
^{241}Pu	N,G	2.825E+1	2.704E+1	2.589E+1	2.478E+1
^{241}Pu	N,F	8.745E+1	8.430E+1	8.126E+1	7.832E+1
^{242}Pu	N,G	2.985E+1	2.914E+1	2.844E+1	2.776E+1
^{241}Am	N,G	6.577E+1	6.205E+1	5.854E+1	5.522E+1
^{241}Am	N,GX		Not included[b]		
^{243}Am	N,G		Not included[b]		
^{243}Am	N,GX		Not included[b]		
^{242}Cm	N,G		Not included[b]		
^{244}Cm	N,G		Not included[b]		

[a]N,G = (n,γ) reaction to a ground state.
N,GX = (n,γ) reaction to an excited state.
N,F = (n, fission) reaction.
[b]Not included - These cross-section reaction types did not exhibit significant variation with burnup and have not been included.

Table B.3. Cross sections as a function of burnup for a BWR-US

Nuclide	Type of cross-section reaction[a]	Cross section (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of:				
		0.0	2.0	4.0	6.0	8.0
^{234}U	N,G	1.940E+1	Not included[b]			
^{235}U	N,G	9.742E+0	Not included[b]			
^{235}U	N,F	4.186E+1	Not included[b]			
^{236}U	N,G	8.132E+0	Not included[b]			
^{238}U	N,G	8.025E-1	Not included[b]			
^{237}Np	N,G	3.260E+1	Not included[b]			
^{238}Pu	N,G	2.978E+1	Not included[b]			
^{238}Pu	N,F	2.143E+0	Not included[b]			
^{239}Pu	N,G	6.845E+1	6.379E+1	5.944E+1	5.540E+1	5.163E+1
^{239}Pu	N,F	1.168E+2	1.097E+2	1.029E+2	9.664E+1	9.073E+1
^{240}Pu	N,G	2.227E+2	1.761E+2	1.393E+2	1.102E+2	8.713E+1
^{241}Pu	N,G	3.890E+1	3.724E+1	3.565E+1	3.413E+1	3.267E+1
^{241}Pu	N,F	1.153E+2	1.111E+2	1.071E+2	1.033E+2	9.954E+1
^{242}Pu	N,G	3.402E+1	3.320E+1	3.241E+1	3.163E+1	3.087E+1
^{241}Am	N,G	9.599E+1	9.056E+1	8.543E+1	8.060E+1	7.604E+1
^{241}Am	N,GX	2.400E+1	Not included[b]			
^{243}Am	N,G	2.545E+0	Not included[b]			
^{243}Am	N,GX	4.835E+1	Not included[b]			
^{242}Cm	N,G	5.709E+0	Not included[b]			
^{244}Cm	N,G	1.266E+1	Not included[b]			

[a]N,G = (n,γ) reaction to a ground state.
N,GX = (n,γ) reaction to an excited state.
N,F = (n, fission) reaction.
[b]Not included - These types of cross-section reactions did not exhibit significant variation with burnup and have not been included.

Table B.4. Cross sections as a function of burnup for a BWR-US0

Nuclide	Type of cross-section reaction[a]	Cross section (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 0.0	2.0	4.0	6.0	8.0
^{234}U	N,G	2.028E+1	Not included[b]			
^{235}U	N,G	1.038E+1	Not included[b]			
^{235}U	N,F	4.558E+1	Not included[b]			
^{236}U	N,G	8.281E+0	Not included[b]			
^{238}U	N,G	8.262E-1	Not included[b]			
^{237}Np	N,G	3.364E+1	Not included[b]			
^{238}Pu	N,G	3.288E+1	Not included[b]			
^{238}Pu	N,F	2.225E+0	Not included[b]			
^{239}Pu	N,G	7.234E+1	6.741E+1	6.282E+1	5.855E+1	5.456E+1
^{239}Pu	N,F	1.245E+2	1.169E+2	1.097E+2	1.030E+2	9.671E+1
^{240}Pu	N,G	2.277E+2	1.801E+2	1.424E+2	1.126E+2	8.909E+1
^{241}Pu	N,G	4.202E+1	4.023E+1	3.851E+1	3.687E+1	3.529E+1
^{241}Pu	N,F	1.247E+2	1.202E+2	1.159E+2	1.117E+2	1.077E+2
^{242}Pu	N,G	3.412E+1	3.330E+1	3.250E+1	3.172E+1	3.096E+1
^{241}Am	N,G	1.001E+2	9.443E+1	8.909E+1	8.405E+1	7.929E+1
^{241}Am	N,GX	2.503E+1	Not included[b]			
^{243}Am	N,G	2.581E+0	Not included[b]			
^{243}Am	N,GX	4.904E+1	Not included[b]			
^{242}Cm	N,G	5.736E+0	Not included[b]			
^{244}Cm	N,G	1.285E+1	Not included[b]			

[a]N,G = (n,γ) reaction to a ground state.
N,GX = (n,γ) reaction to an excited state.
N,F = (n, fission) reaction.
[b]Not included - These types of cross-section reactions did not exhibit significant variation with burnup and have not been included.

Table B.5. Cross sections as a function of burnup for a BWR-UE

Nuclide	Type of cross-section reaction[a]	Cross section (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of:			
		0.0	2.0	4.0	6.0
^{234}U	N,G	1.800E+1	Not included[b]		
^{235}U	N,G	8.120E+0	Not included[b]		
^{235}U	N,F	3.301E+1	Not included[b]		
^{236}U	N,G	8.084E+0	Not included[b]		
^{238}U	N,G	7.730E-1	Not included[b]		
^{237}Np	N,G	2.992E+1	Not included[b]		
^{238}Pu	N,G	2.293E+1	Not included[b]		
^{238}Pu	N,F	1.981E+0	Not included[b]		
^{239}Pu	N,G	5.697E+1	5.309E+1	4.948E+1	4.611E+1
^{239}Pu	N,F	9.581E+1	8.995E+1	8.444E+1	7.928E+1
^{240}Pu	N,G	2.137E+2	1.690E+2	1.337E+2	1.057E+2
^{241}Pu	N,G	3.113E+1	2.980E+1	2.853E+1	2.731E+1
^{241}Pu	N,F	9.282E+1	8.947E+1	8.624E+1	8.313E+1
^{242}Pu	N,G	3.363E+1	3.282E+1	3.204E+1	3.127E+1
^{241}Am	N,G	8.356E+1	7.883E+1	7.437E+1	7.016E+1
^{241}Am	N,GX	2.089E+1	Not included[b]		
^{243}Am	N,G	2.470E+0	Not included[b]		
^{243}Am	N,GX	4.694E+1	Not included[b]		
^{242}Cm	N,G	5.499E+0	Not included[b]		
^{244}Cm	N,G	1.258E+1	Not included[b]		

Table B.5 (continued)

Nuclide	Type of cross-section reaction[a]	Cross section (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 8.0	10.0	12.0	14.0
^{234}U	N,G	Not included[b]			
^{235}U	N,G	Not included[b]			
^{235}U	N,F	Not included[b]			
^{236}U	N,G	Not included[b]			
^{238}U	N,G	Not included[b]			
^{237}Np	N,G	Not included[b]			
^{238}Pu	N,G	Not included[b]			
^{238}Pu	N,F	Not included[b]			
^{239}Pu	N,G	4.297E+1	4.004E+1	3.732E+1	3.477E+1
^{239}Pu	N,F	7.443E+1	6.987E+1	6.560E+1	6.158E+1
^{240}Pu	N,G	8.361E+1	6.613E+1	5.230E+1	4.136E+1
^{241}Pu	N,G	2.615E+1	2.503E+1	2.396E+1	2.25E+1
^{241}Pu	N,F	8.013E+1	7.724E+1	7.445E+1	7.14E+1
^{242}Pu	N,G	3.052E+1	2.978E+1	2.907E+1	2.837E+1
^{241}Am	N,G	6.619E+1	6.244E+1	5.891E+1	5.558E+1
^{241}Am	N,GX	Not included[b]			
^{243}Am	N,G	Not included[b]			
^{243}Am	N,GX	Not included[b]			
^{242}Cm	N,G	Not included[b]			
^{244}Cm	N,G	Not included[b]			

[a]N,G = (n,γ) reaction to a ground state.
N,GX = (n,γ) reaction to an excited state.
N,F = (n, fission) reaction.
[b]Not included - These types of cross-section reactions did not exhibit significant variation with burnup and have not been included.

Appendix C.

ORIGEN2 USER INFORMATION

The following information is provided as an interim measure to help ORIGEN2 users begin to use the new cross-section libraries described in this report. Eventually, a revised version of the ORIGEN2 (called ORIGEN2 Version 2.1) code package will be made available through the Radiation Shielding Information Center at Oak Ridge National Laboratory. At such time, this information will be incorporated in the user's manual distributed with the code package.

The new ORIGEN2 reactor models are provided as individual cross-section library files and as FORTRAN subroutines (to allow variation of certain actinide cross sections with burnup). The ORIGEN2 Users' Manual[3,9] (pp. 46 and 46a) lists a number of reactor models available with the code package. Listed are the type of library paired with the identification number for the activation product [NLIB(5)], actinide and daughter [NLIB(6)] and fission product [NLIB(7)] library segments, as well as the control code [NLIB(12) value] required to select the correct variable cross-section subroutine.

The values for the updated models presented in Table C.1 should be used to select the new cross-section data libraries and variable actinide cross-section values.

Table C.1. Previous and updated ORIGEN2 reactor model identifiers

Type of library	ORIGEN2 cross-section library segment			Variable actinide control character
	Activation products	Actinides/ daughters	Fission products	
ORIGEN2 input parameter[a]	NLIB(5)	NLIB(6)	NLIB(7)	NLIB(12)
Previous models[b]				
PWR-U	204	205	206	1
PUD50	219	220	221	9
BWR-U	251	252	253	4
Updated models				
PWR-US	601	602	603	38
PWR-UE	604	605	606	39
BWR-US	651	652	653	40
BWR-US0	654	655	656	41
BWR-UE	657	658	659	42

[a]The ORIGEN2 input parameters, denoted by the NLIB(5)-NLIB(7) and NLIB(12), are the data values needed by the LIB command to provide access to each new cross-section library segment and the correct variable actinide cross-section data. The new cross-section libraries and variable cross-section subroutines will be part of an updated ORIGEN2 code package (called ORIGEN2 Version 2.1).
[b]As documented in refs. 7 and 8.

Appendix D.

SAMPLE ORIGEN2 INPUTS FOR UPDATED LWR MODELS

The ORIGEN2 results for the five updated LWR models presented in Section 5 were prepared using the input files presented in Tables D.1 through D.5.) These tables are included at the end of this appendix.) Additional calculations were performed (as a basis for comparison) using the previous LWR models, but the input files are not provided with this report. The input files used with the previous LWR models have some important differences, which are described in the following paragraphs.

First, it is important to state that the comparisons of results from the previous-vs-updated ORIGEN2 reactor models presented in Section 5 were performed using each model's default design parameters. Therefore, some of the cases used for comparison had differences in (1) initial fuel enrichment, (2) burnup history (i.e., number and length of burnup cycles), or (3) both. The cases compared in Section 5 were (1) PWR-U vs PWR-US, (2) PUD50 vs PWR-UE, (3) BWR-U vs BWR-US, (4) BWR-U vs BWR-US0, and (5) BWR-UX vs BWR-UE. Comparison of previous-vs-updated models, within each case, used the same specific power value and achieved the same final burnup.

Of the five comparison cases, only PWR-U vs PWR-US used identical values of initial enrichment and burnup history. The PUD50-vs-PWR-UE comparison used a slightly different enrichment, 4.15 vs 4.2 wt %, and five average-length vs five extended-length burnup cycles to achieve a burnup of 50,000 MWd/MTIHM.

The BWR-U model used 2.75 wt % enriched fuel, whereas BWR-US and BWR-US0 used 3.0 wt % fuel. All of the standard-burnup BWR models used identical, four-cycle burnup histories. No previous extended-burnup BWR model existed for ORIGEN2; therefore, for comparison, the BWR-UX calculation uses the BWR-U cross-section library, 3.5 wt % enriched fuel, and six (average-length) burnup cycles, in a fashion similar to the PUD50 model, to achieve a burnup of 40,000 MWd/MTIHM. The BWR-UE differs by using 3.4 wt % enriched fuel and four (extended-length) burnup cycles. Although use of the BWR-U cross-section library in

extended-burnup calculations violates the assumptions for which this model was developed, the comparison is justified because previous documents[12] have attempted to address the characteristics of extended-burnup BWR spent fuel.

The primary difference between the previous and updated implementation of extended-burnup LWR models within ORIGEN2 is the number of burnup cycles used by each model. Although the total number of irradiation days between each previous-vs-updated model is identical, the updated models achieve this total by extending the length of each burnup cycle rather than adding more "average-length" burnup cycles. This difference is more representative of the true operation of extended-burnup fuel cycles, which provide economic benefit to nuclear utilities by providing a longer power generation cycle between scheduled outages.

Table D.1. Sample ORIGEN2 input file for PWR-US irradiation and decay

```
-1
-1
-1
RDA  *  BURNUP OF PWR 3.2% UO2 FUEL & ASSY HARDWARE, 33,000 MWD/MT
RDA  **  CROSS SECTION LIBRARY = PWRUS, 3 CYCLE
RDA  ***  SCOTT B. LUDWIG, OAK RIDGE NATIONAL LABORATORY
RDA  ****  (615) 574-7916, FTS 624-7916
RDA     -1 = FRESH PWR FUEL WITH IMPURITIES (1 MT = 1000 KG)
RDA     -2 = FRESH ZIRCALOY-4 COMPOSITION (1 KG)
RDA     -3 = FRESH SS-304 COMPOSITION (1 KG)
RDA     -4 = FRESH SS-302 COMPOSITION (1 KG)
RDA     -5 = FRESH INCONEL-718 COMPOSITION (1 KG)
RDA     -6 = FRESH NICROBRAZE-50 COMPOSITION (1 KG)
RDA     WARNING:  VECTORS ARE CHANGED WITH RESPECT TO CONTENT.
RDA               THESE CHANGES WILL BE NOTED ON RDA CARDS.
CUT     5 1.0E-10 7 1.0E-10 9 1.0E-10 -1
LIP     0 0 0
RDA        DECAY LIB    XSECT LIB                        VAR. XSECT
LIB     0    1 2 3     601 602 603    9   50   0   1        38
RDA      PHOTON LIB
PHO     101 102 103  10
TIT  INITIAL COMP. OF UNIT AMOUNTS OF FUEL AND STRUCTURAL MAT'LS
RDA     READ FUEL COMPOSITION INCLUDING IMPURITIES (1000 KG)
INP     -1  1  -1  -1  1  1
RDA     READ  ZIRCALOY-4 COMPOSITION (1.0 KG)
INP     -2  1  -1  -1  1  1
RDA     READ  SS-304 COMPOSITION (1.0 KG)
INP     -3  1  -1  -1  1  1
RDA     READ  SS-302 COMPOSITION (1.0 KG)
INP     -4  1  -1  -1  1  1
RDA     READ  INCONEL-718 COMPOSITION (1.0 KG)
INP     -5  1  -1  -1  1  1
RDA     READ  NICROBRAZE-50 COMPOSITION (1.0 KG)
INP     -6  1  -1  -1  1  1
TIT     IRRADIATION OF ONE METRIC TON OF PWR FUEL
MOV     -1 1 0 1.0
PCH     1  1  1
HED     1                                                CHARGE
BUP
IRP       66.7    37.5    1   2   4 2  BURNUP=  2,500 MWD/MTIHM
IRP      133.3    37.5    2   3   4 0  BURNUP=  5,000 MWD/MTIHM
IRP      293.3    37.5    3   4   4 0  BURNUP= 11,000 MWD/MTIHM
DEC      399.3            4   5   4 0   DECAY FOR 106.0 DAYS
IRP      506.0    37.5    5   6   4 0  BURNUP= 15,000 MWD/MTIHM
IRP      639.3    37.5    6   7   4 0  BURNUP= 20,000 MWD/MTIHM
IRP      692.7    37.5    7   8   4 0  BURNUP= 22,000 MWD/MTIHM
DEC      798.7            8   9   4 0   DECAY FOR 106.0 DAYS
IRP      878.7    37.5    9  10   4 0  BURNUP= 25,000 MWD/MTIHM
IRP     1012.0    37.5   10  11   4 0  BURNUP= 30,000 MWD/MTIHM
```

Table D.1 (continued)

```
IRP     1092.0    37.5   11  12   4 0  BURNUP= 33,000 MWD/MTIHM
BUP
RDA     -10 = IRRADIATED FUEL AT DISCHARGE
MOV     12 -10  0 1.0
PCH     -10  -10  -10
RDA     IRRADIATION OF ZIRCALOY-4 AT 1.000 FLUX
TIT     IRRADIATION OF ZIRCALOY-4 AT 1.000 FLUX
MOV     -2 1 0  223.0  ZIRCALOY-4
PCH     1  1  1
HED     1                                                  CHARGE
IRF       66.7    -1.0    1   2   4 2  BURNUP=  2,500 MWD/MTIHM
IRF      133.3    -1.0    2   3   4 0  BURNUP=  5,000 MWD/MTIHM
IRF      293.3    -1.0    3   4   4 0  BURNUP= 11,000 MWD/MTIHM
DEC      399.3            4   5   4 0   DECAY FOR 106.0 DAYS
IRF      506.0    -1.0    5   6   4 0  BURNUP= 15,000 MWD/MTIHM
IRF      639.3    -1.0    6   7   4 0  BURNUP= 20,000 MWD/MTIHM
IRF      692.7    -1.0    7   8   4 0  BURNUP= 22,000 MWD/MTIHM
DEC      798.7            8   9   4 0   DECAY FOR 106.0 DAYS
IRF      878.7    -1.0    9  10   4 0  BURNUP= 25,000 MWD/MTIHM
IRF     1012.0    -1.0   10  11   4 0  BURNUP= 30,000 MWD/MTIHM
IRF     1092.0    -1.0   11  12   4 0  BURNUP= 33,000 MWD/MTIHM
RDA     -9 = IRRADIATED ZIRCALOY-4 AT DISCHARGE
MOV     12  -9  0  1.0
PCH     -9  -9  -9
RDA     IRRADIATION OF INCONEL-718 + NICROBRAZE-50 AT 1.000 FLUX
TIT     IRRADIATION OF INCONEL-718 + NICROBRAZE-50 AT 1.000 FLUX
MOV     -5 1 0  12.8   INCONEL-718
ADD     -6 1 0   2.6   NICROBRAZE-50
PCH     1  1  1
HED     1                                                  CHARGE
IRF       66.7    -1.0    1   2   4 2  BURNUP=  2,500 MWD/MTIHM
IRF      133.3    -1.0    2   3   4 0  BURNUP=  5,000 MWD/MTIHM
IRF      293.3    -1.0    3   4   4 0  BURNUP= 11,000 MWD/MTIHM
DEC      399.3            4   5   4 0   DECAY FOR 106.0 DAYS
IRF      506.0    -1.0    5   6   4 0  BURNUP= 15,000 MWD/MTIHM
IRF      639.3    -1.0    6   7   4 0  BURNUP= 20,000 MWD/MTIHM
IRF      692.7    -1.0    7   8   4 0  BURNUP= 22,000 MWD/MTIHM
DEC      798.7            8   9   4 0   DECAY FOR 106.0 DAYS
IRF      878.7    -1.0    9  10   4 0  BURNUP= 25,000 MWD/MTIHM
IRF     1012.0    -1.0   10  11   4 0  BURNUP= 30,000 MWD/MTIHM
IRF     1092.0    -1.0   11  12   4 0  BURNUP= 33,000 MWD/MTIHM
RDA    -8 = IRRADIATED INCONEL-718 AND NICROBRAZE-50 AT DISCHARGE
MOV     12  -8  0  1.0
PCH     -8  -8  -8
RDA     IRRADIATION OF SS-304 AT 1.000 FLUX
TIT     IRRADIATION OF SS-304 AT 1.000 FLUX
MOV     -3 1 0  9.9    SS-304
PCH     1  1  1
HED     1                                                  CHARGE
```

Table D.1 (continued)

```
IRF      66.7    -1.0    1   2   4 2  BURNUP=  2,500 MWD/MTIHM
IRF     133.3    -1.0    2   3   4 0  BURNUP=  5,000 MWD/MTIHM
IRF     293.3    -1.0    3   4   4 0  BURNUP= 11,000 MWD/MTIHM
DEC     399.3            4   5   4 0   DECAY FOR 106.0 DAYS
IRF     506.0    -1.0    5   6   4 0  BURNUP= 15,000 MWD/MTIHM
IRF     639.3    -1.0    6   7   4 0  BURNUP= 20,000 MWD/MTIHM
IRF     692.7    -1.0    7   8   4 0  BURNUP= 22,000 MWD/MTIHM
DEC     798.7            8   9   4 0   DECAY FOR 106.0 DAYS
IRF     878.7    -1.0    9  10   4 0  BURNUP= 25,000 MWD/MTIHM
IRF    1012.0    -1.0   10  11   4 0  BURNUP= 30,000 MWD/MTIHM
IRF    1092.0    -1.0   11  12   4 0  BURNUP= 33,000 MWD/MTIHM
RDA    -7 = IRRADIATED SS-304 AT DISCHARGE
MOV    12  -7  0  1.0
PCH    -7  -7  -7
RDA    IRRADIATION OF SS-304 END PIECES AT 0.011 FLUX
TIT    IRRADIATION OF SS-304 END PIECES AT 0.011 FLUX
MOV    -3 1 0 27.2    SS-304
PCH    1  1  1
HED    1                                                    CHARGE
IRF      66.7    -0.011  1   2   4 2  BURNUP=  2,500 MWD/MTIHM
IRF     133.3    -0.011  2   3   4 0  BURNUP=  5,000 MWD/MTIHM
IRF     293.3    -0.011  3   4   4 0  BURNUP= 11,000 MWD/MTIHM
DEC     399.3            4   5   4 0   DECAY FOR 106.0 DAYS
IRF     506.0    -0.011  5   6   4 0  BURNUP= 15,000 MWD/MTIHM
IRF     639.3    -0.011  6   7   4 0  BURNUP= 20,000 MWD/MTIHM
IRF     692.7    -0.011  7   8   4 0  BURNUP= 22,000 MWD/MTIHM
DEC     798.7            8   9   4 0   DECAY FOR 106.0 DAYS
IRF     878.7    -0.011  9  10   4 0  BURNUP= 25,000 MWD/MTIHM
IRF    1012.0    -0.011 10  11   4 0  BURNUP= 30,000 MWD/MTIHM
IRF    1092.0    -0.011 11  12   4 0  BURNUP= 33,000 MWD/MTIHM
RDA    -6 = IRRADIATED SS-304 END PIECES AT DISCHARGE
MOV    12 -6   0 1.0
PCH    -6  -6  -6
RDA    IRRADIATION OF SS-302 PLENUM SPRINGS AT 0.042 FLUX
TIT    IRRADIATION OF SS-302 PLENUM SPRINGS AT 0.042 FLUX
MOV    -4 1 0 4.2     SS-302
ADD    -2 1 0 12.0    ZIRC-4 IN PLENUM
PCH    1  1  1
HED    1                                                    CHARGE
IRF      66.7    -0.042  1   2   4 2  BURNUP=  2,500 MWD/MTIHM
IRF     133.3    -0.042  2   3   4 0  BURNUP=  5,000 MWD/MTIHM
IRF     293.3    -0.042  3   4   4 0  BURNUP= 11,000 MWD/MTIHM
DEC     399.3            4   5   4 0   DECAY FOR 106.0 DAYS
IRF     506.0    -0.042  5   6   4 0  BURNUP= 15,000 MWD/MTIHM
IRF     639.3    -0.042  6   7   4 0  BURNUP= 20,000 MWD/MTIHM
IRF     692.7    -0.042  7   8   4 0  BURNUP= 22,000 MWD/MTIHM
DEC     798.7            8   9   4 0   DECAY FOR 106.0 DAYS
IRF     878.7    -0.042  9  10   4 0  BURNUP= 25,000 MWD/MTIHM
IRF    1012.0    -0.042 10  11   4 0  BURNUP= 30,000 MWD/MTIHM
```

Table D.1 (continued)

```
IRF     1092.0    -0.042 11  12   4 0  BURNUP= 33,000 MWD/MTIHM
RDA     -5 = IRRADIATED SS-302 PLENUM SPRINGS AT DISCHARGE
MOV     12 -5   0 1.0
PCH     -5  -5  -5
RDA     ***** OUTPUT MODULE *****
TIT    TEST OF ORIGEN2 V2.1 - PWR FUEL - STANDARD BURNUP (PWRUS)
BAS      1 MTIHM 3.2% UO2;BURNUP=33,000 MWD/MTIHM, 3 CYCLE
HED      -10    FUEL DIS
HED      -1     FUEL CHG
MOV      -1    1    0   1.0
MOV     -10    2    0   1.0
RDA     ***** DECAY MODULE *****
DEC     0.5   2   4   5   2
DEC     1.0   4   3   5   0
DEC     2.0   3   4   5   0
DEC     5.0   4   5   5   0
DEC    10.0   5   6   5   0
OPTL      4*8  7  8  7  8  7  8  8*8  5*7  8
OPTA      4*8  7  8  7  8  7  8  8*8  5*7  8
OPTF      4*8  7  8  7  8  7  8  8*8  5*7  8
OUT      6    1     -1    0
TIT  TEST OF ORIGEN2 V2.1 - PWR FUEL ASSY-STANDARD BURNUP (PWRUS)
BAS  1 MTIHM 3.2% UO2 FUEL ASSY;BURNUP=33,000 MWD/MTIHM, 3 CYCLE
OPTL      6*8  7  8  7  8  14*8
OPTA      6*8  7  8  7  8  14*8
OPTF      6*8  7  8  7  8  14*8
MOV     -10    1    0   1.0
ADD      -9    1    0   1.0
ADD      -8    1    0   1.0
ADD      -7    1    0   1.0
ADD      -6    1    0   1.0
ADD      -5    1    0   1.0
HED     1    ASSY DIS
RDA     ***** DECAY MODULE *****
DEC     0.5   1   3   5   2
DEC     1.0   3   2   5   0
DEC     2.0   2   3   5   0
DEC     5.0   3   4   5   0
DEC    10.0   4   5   5   0
OUT      5    1     -1    0
END
2 922340  290.0  922350  32000. 922380   967710.     0  0.0 FUEL 3.2%
4 030000  1.0    050000  1.0    060000   89.4   070000 25.0 FUEL IMPU
4 080000 134454. 090000  10.7   110000   15.0   120000  2.0 FUEL IMPU
4 130000  16.7   140000  12.1   150000   35.0   170000  5.3 FUEL IMPU
4 200000  2.0    220000  1.0    230000   3.0    240000  4.0 FUEL IMPU
4 250000  1.7    260000  18.0   270000   1.0    280000 24.0 FUEL IMPU
4 290000  1.0    300000  40.3   420000   10.0   470000 0.1  FUEL IMPU
4 480000  25.0   490000  2.0    500000   4.0    640000 2.5  FUEL IMPU
```

Table D.1 (continued)

4	740000	2.0	820000	1.0	830000	0.4	0	0.0	FUEL IMPU
0									
4	400000	979.11	500000	16.0	260000	2.25	240000	1.25	ZIRC-4
4	280000	0.02	130000	0.024	050000	0.00033	480000	0.00025	ZIRC-4
4	060000	0.120	270000	0.010	290000	0.020	720000	0.078	ZIRC-4
4	010000	0.013	250000	0.020	070000	0.080	080000	0.950	ZIRC-4
4	160000	0.035	220000	0.020	740000	0.020	230000	0.020	ZIRC-4
5	920000	0.0002	0	0.0					ZIRC-4
0									
4	260000	688.44	240000	190.0	280000	89.2	250000	20.0	SS-304
4	060000	0.8	150000	0.45	160000	0.3	140000	10.0	SS-304
4	070000	1.3	270000	0.8	0	0.0			SS-304
0									
4	260000	697.74	240000	180.0	280000	89.2	250000	20.0	SS-302
4	060000	1.5	150000	0.45	160000	0.3	140000	10.0	SS-302
4	070000	1.3	270000	0.8	0	0.0			SS-302
0									
4	260000	179.766	240000	189.753	280000	519.625	130000	5.992	INC-718
4	060000	0.4	270000	4.694	290000	0.999	250000	1.997	INC-718
4	420000	29.961	070000	1.3	410000	55.458	160000	0.07	INC-718
4	140000	1.997	220000	7.99	0	0.0			INC-718
0									
4	260000	0.471	240000	149.709	280000	744.438	400000	0.1	NICR-50
4	130000	0.1	050000	0.05	060000	0.1	270000	0.381	NICR-50
4	250000	0.1	070000	0.066	080000	0.043	150000	103.244	NICR-50
4	160000	0.1	140000	0.511	220000	0.1	740000	0.1	NICR-50
0									

Table D.2. Sample ORIGEN2 input file for PWR-UE irradiation and decay

```
-1
-1
-1
RDA  *  BURNUP OF PWR 4.2% UO2 FUEL & ASSY HDWARE, 50,000 MWD/MT
RDA     **  CROSS SECTION LIBRARY = PWRUE, 3 CYCLE
RDA     ***  SCOTT B. LUDWIG, OAK RIDGE NATIONAL LABORATORY
RDA     ****  (615) 574-7916, FTS 624-7916
RDA     -1 = FRESH PWR FUEL WITH IMPURITIES (1 MT = 1000 KG)
RDA     -2 = FRESH ZIRCALOY COMPOSITION (1 KG)
RDA     -3 = FRESH SS 304 COMPOSITION (1 KG)
RDA     -4 = FRESH SS 302 COMPOSITION (1 KG)
RDA     -5 = FRESH INCONEL COMPOSITION (1 KG)
RDA     -6 = FRESH NICROBRAZE COMPOSITION (1 KG)
RDA     WARNING:  VECTORS ARE CHANGED WITH RESPECT TO CONTENT.
RDA               THESE CHANGES WILL BE NOTED ON RDA CARDS.
CUT     5 1.0E-10 7 1.0E-10 9 1.0E-10 -1
LIP     0 0 0
RDA        DECAY LIB    XSECT LIB                        VAR. XSECT
LIB     0    1 2 3     604 605 606    9   50   0   1        39
RDA      PHOTON LIB
PHO     101 102 103   10
TIT   INITIAL COMP. OF UNIT AMOUNTS OF FUEL AND STRUCTURAL MAT'LS
RDA     READ FUEL COMPOSITION INCLUDING IMPURITIES (1000 KG)
INP     -1   1  -1  -1  1  1
RDA     READ ZIRCALOY COMPOSITION (1.0 KG)
INP     -2   1  -1  -1  1  1
RDA     READ SS304 COMPOSITION (1.0 KG)
INP     -3   1  -1  -1  1  1
RDA     READ SS302 COMPOSITION (1.0 KG)
INP     -4   1  -1  -1  1  1
RDA     READ INCONEL 718 COMPOSITION (1.0 KG)
INP     -5   1  -1  -1  1  1
RDA     READ NICROBRAZE 50 COMPOSITION (1.0 KG)
INP     -6   1  -1  -1  1  1
TIT     IRRADIATION OF ONE METRIC TON OF PWRU FUEL
MOV     -1   1   0   1.0
PCH      1   1   1
HED      1                                                CHARGE
BUP
IRP      66.7    37.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRP     133.3    37.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRP     266.7    37.5   3   4   4  0   BURNUP=10,000 MWD/MTIHM
IRP     360.0    37.5   4   5   4  0   BURNUP=13,500 MWD/MTIHM
IRP     444.4    37.5   5   6   4  0   BURNUP=16,665 MWD/MTIHM
DEC     550.4           6   7   4  0    DECAY FOR 106.0 DAYS
IRP     650.0    37.5   7   8   4  0   BURNUP=20,400 MWD/MTIHM
IRP     780.0    37.5   8   9   4  0   BURNUP=25,275 MWD/MTIHM
IRP     890.0    37.5   9  10   4  0   BURNUP=29,400 MWD/MTIHM
IRP     994.8    37.5  10  11   4  0   BURNUP=33,330 MWD/MTIHM
```

Table D.2 (continued)

```
DEC    1100.8          11  12   4  0    DECAY FOR 106.0 DAYS
IRP    1220.0    37.5  12   1   4  0   BURNUP=37,830 MWD/MTIHM
IRP    1320.0    37.5   1   2   4  0   BURNUP=41,550 MWD/MTIHM
IRP    1430.0    37.5   2   3   4  0   BURNUP=45,675 MWD/MTIHM
IRP    1545.33   37.5   3   4   4  0   BURNUP=50,000 MWD/MTIHM
BUP
RDA    -10 = IRRADIATED U FUEL AT DISCHARGE
MOV      4   -10     0     1.0
PCH     -10   -10   -10
RDA     IRRADIATION OF ZIRCALOY AT 1.000 FLUX
TIT     IRRADIATION OF ZIRCALOY AT 1.000 FLUX
MOV      -2     1     0   223.0       ZIRCALOY
PCH       1     1     1
HED      1                                                  CHARGE
IRF      66.7    -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     133.3    -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     266.7    -1.0   3   4   4  0   BURNUP=10,000 MWD/MTIHM
IRF     360.0    -1.0   4   5   4  0   BURNUP=13,500 MWD/MTIHM
IRF     444.4    -1.0   5   6   4  0   BURNUP=16,665 MWD/MTIHM
DEC     550.4           6   7   4  0    DECAY FOR 106.0 DAYS
IRF     650.0    -1.0   7   8   4  0   BURNUP=20,400 MWD/MTIHM
IRF     780.0    -1.0   8   9   4  0   BURNUP=25,275 MWD/MTIHM
IRF     890.0    -1.0   9  10   4  0   BURNUP=29,400 MWD/MTIHM
IRF     994.8    -1.0  10  11   4  0   BURNUP=33,330 MWD/MTIHM
DEC    1100.8          11  12   4  0    DECAY FOR 106.0 DAYS
IRF    1220.0    -1.0  12   1   4  0   BURNUP=37,830 MWD/MTIHM
IRF    1320.0    -1.0   1   2   4  0   BURNUP=41,550 MWD/MTIHM
IRF    1430.0    -1.0   2   3   4  0   BURNUP=45,675 MWD/MTIHM
IRF    1545.33   -1.0   3   4   4  0   BURNUP=50,000 MWD/MTIHM
RDA    -9 = IRRADIATED ZIRCALOY
MOV      4   -9     0     1.0
PCH     -9   -9   -9
TIT     IRRADIATION OF INCONEL + NICROBRAZE 50 AT 1.000 FLUX
MOV      -5     1     0   12.8       INCONEL
ADD      -6     1     0    2.6       NICROBRAZE 50
PCH       1     1     1
HED      1                                                  CHARGE
IRF      66.7    -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     133.3    -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     266.7    -1.0   3   4   4  0   BURNUP=10,000 MWD/MTIHM
IRF     360.0    -1.0   4   5   4  0   BURNUP=13,500 MWD/MTIHM
IRF     444.4    -1.0   5   6   4  0   BURNUP=16,665 MWD/MTIHM
DEC     550.4           6   7   4  0    DECAY FOR 106.0 DAYS
IRF     650.0    -1.0   7   8   4  0   BURNUP=20,400 MWD/MTIHM
IRF     780.0    -1.0   8   9   4  0   BURNUP=25,275 MWD/MTIHM
IRF     890.0    -1.0   9  10   4  0   BURNUP=29,400 MWD/MTIHM
IRF     994.8    -1.0  10  11   4  0   BURNUP=33,330 MWD/MTIHM
DEC    1100.8          11  12   4  0    DECAY FOR 106.0 DAYS
IRF    1220.0    -1.0  12   1   4  0   BURNUP=37,830 MWD/MTIHM
```

Table D.2 (continued)

```
IRF    1320.0    -1.0   1   2   4  0   BURNUP=41,550 MWD/MTIHM
IRF    1430.0    -1.0   2   3   4  0   BURNUP=45,675 MWD/MTIHM
IRF    1545.33   -1.0   3   4   4  0   BURNUP=50,000 MWD/MTIHM
RDA    -8 = IRRADIATED INCONEL AND NICROBRAZE
MOV      4    -8     0     1.0
PCH     -8    -8    -8
TIT     IRRADIATION OF SS 304 AT 1.000 FLUX
MOV      -3     1     0    9.9      SS 304
PCH       1     1     1
HED      1                                                CHARGE
IRF      66.7    -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     133.3    -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     266.7    -1.0   3   4   4  0   BURNUP=10,000 MWD/MTIHM
IRF     360.0    -1.0   4   5   4  0   BURNUP=13,500 MWD/MTIHM
IRF     444.4    -1.0   5   6   4  0   BURNUP=16,665 MWD/MTIHM
DEC     550.4           6   7   4  0    DECAY FOR 106.0 DAYS
IRF     650.0    -1.0   7   8   4  0   BURNUP=20,400 MWD/MTIHM
IRF     780.0    -1.0   8   9   4  0   BURNUP=25,275 MWD/MTIHM
IRF     890.0    -1.0   9  10   4  0   BURNUP=29,400 MWD/MTIHM
IRF     994.8    -1.0  10  11   4  0   BURNUP=33,330 MWD/MTIHM
DEC    1100.8          11  12   4  0    DECAY FOR 106.0 DAYS
IRF    1220.0    -1.0  12   1   4  0   BURNUP=37,830 MWD/MTIHM
IRF    1320.0    -1.0   1   2   4  0   BURNUP=41,550 MWD/MTIHM
IRF    1430.0    -1.0   2   3   4  0   BURNUP=45,675 MWD/MTIHM
IRF    1545.33   -1.0   3   4   4  0   BURNUP=50,000 MWD/MTIHM
RDA    -7 = IRRADIATED SS 304
MOV      4    -7     0     1.0
PCH     -7    -7    -7
RDA     IRRADIATION OF SS 304 END PIECES AT 0.011 FLUX
TIT     IRRADIATION OF SS 304 END PIECES AT 0.011 FLUX
MOV      -3     1     0    27.2     SS 304
PCH       1     1     1
HED      1                                                CHARGE
IRF      66.7  -0.011   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     133.3  -0.011   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     266.7  -0.011   3   4   4  0   BURNUP=10,000 MWD/MTIHM
IRF     360.0  -0.011   4   5   4  0   BURNUP=13,500 MWD/MTIHM
IRF     444.4  -0.011   5   6   4  0   BURNUP=16,665 MWD/MTIHM
DEC     550.4           6   7   4  0    DECAY FOR 106.0 DAYS
IRF     650.0  -0.011   7   8   4  0   BURNUP=20,400 MWD/MTIHM
IRF     780.0  -0.011   8   9   4  0   BURNUP=25,275 MWD/MTIHM
IRF     890.0  -0.011   9  10   4  0   BURNUP=29,400 MWD/MTIHM
IRF     994.8  -0.011  10  11   4  0   BURNUP=33,330 MWD/MTIHM
DEC    1100.8          11  12   4  0    DECAY FOR 106.0 DAYS
IRF    1220.0  -0.011  12   1   4  0   BURNUP=37,830 MWD/MTIHM
IRF    1320.0  -0.011   1   2   4  0   BURNUP=41,550 MWD/MTIHM
IRF    1430.0  -0.011   2   3   4  0   BURNUP=45,675 MWD/MTIHM
IRF    1545.33 -0.011   3   4   4  0   BURNUP=50,000 MWD/MTIHM
```

Table D.2 (continued)

```
RDA    -6 = IRRADIATED SS 304 END PIECES AT DISCHARGE
MOV     4    -6     0     1.0
PCH    -6    -6    -6
TIT    IRRADIATION OF SS302 PLENUM SPRINGS AT 0.042 FLUX
MOV     -4     1     0    4.2      SS 302
ADD     -2     1     0    12.0     ZR IN PLENUM
PCH      1     1     1
HED     1                                                  CHARGE
IRF     66.7  -0.042   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF    133.3  -0.042   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF    266.7  -0.042   3   4   4  0   BURNUP=10,000 MWD/MTIHM
IRF    360.0  -0.042   4   5   4  0   BURNUP=13,500 MWD/MTIHM
IRF    444.4  -0.042   5   6   4  0   BURNUP=16,665 MWD/MTIHM
DEC    550.4           6   7   4  0    DECAY FOR 106.0 DAYS
IRF    650.0  -0.042   7   8   4  0   BURNUP=20,400 MWD/MTIHM
IRF    780.0  -0.042   8   9   4  0   BURNUP=25,275 MWD/MTIHM
IRF    890.0  -0.042   9  10   4  0   BURNUP=29,400 MWD/MTIHM
IRF    994.8  -0.042  10  11   4  0   BURNUP=33,330 MWD/MTIHM
DEC   1100.8          11  12   4  0    DECAY FOR 106.0 DAYS
IRF   1220.0  -0.042  12   1   4  0   BURNUP=37,830 MWD/MTIHM
IRF   1320.0  -0.042   1   2   4  0   BURNUP=41,550 MWD/MTIHM
IRF   1430.0  -0.042   2   3   4  0   BURNUP=45,675 MWD/MTIHM
IRF   1545.33 -0.042   3   4   4  0   BURNUP=50,000 MWD/MTIHM
RDA    -5 = IRRADIATED SS 302 IN PLENUM SPRINGS AT DISCHARGE
MOV      4   -5     0     1.0
PCH     -5   -5    -5
RDA    ***** OUTPUT MODULE *****
TIT   TEST OF ORIGEN2 V2.1 - PWR FUEL - EXTENDED BURNUP (PWRUE)
BAS     1 MTIHM 4.2% UO2;BURNUP=50,000 MWD/MTIHM, 3 CYCLE
HED     -10    FUEL DIS
HED     -1     FUEL CHG
MOV     -1    1    0   1.0
MOV    -10    2    0   1.0
RDA    ***** DECAY MODULE *****
DEC    0.5   2   4   5   2
DEC    1.0   4   3   5   0
DEC    2.0   3   4   5   0
DEC    5.0   4   5   5   0
DEC   10.0   5   6   5   0
OPTL     4*8  7  8  7  8  7  8  8*8  5*7  8
OPTA     4*8  7  8  7  8  7  8  8*8  5*7  8
OPTF     4*8  7  8  7  8  7  8  8*8  5*7  8
OUT     6    1     -1    0
TIT  TEST OF ORIGEN2 V2.1 - PWR FUEL ASSY-EXTENDED BURNUP (PWRUE)
BAS   1 MTIHM 4.2% UO2 FUEL ASSY;BURNUP=50,000 MWD/MTIHM, 3 CYCLE
OPTL     6*8  7  8  7  8  14*8
OPTA     6*8  7  8  7  8  14*8
OPTF     6*8  7  8  7  8  14*8
MOV    -10    1    0   1.0
```

Table D.2 (continued)

```
  ADD      -9    1    0   1.0
  ADD      -8    1    0   1.0
  ADD      -7    1    0   1.0
  ADD      -6    1    0   1.0
  ADD      -5    1    0   1.0
  HED     1    ASSY DIS
  RDA     ***** DECAY MODULE *****
  DEC     0.5   1   3   5   2
  DEC     1.0   3   2   5   0
  DEC     2.0   2   3   5   0
  DEC     5.0   3   4   5   0
  DEC    10.0   4   5   5   0
  OUT      5    1     -1    0
  END
2 922340  379.0 922350 42000. 922380 957621.   0     0.0  FUEL 4.2%
4 030000  1.0    050000 1.0    060000   89.4 070000 25.0 FUEL IMPU
4 080000 134454. 090000 10.7   110000   15.0 120000  2.0 FUEL IMPU
4 130000  16.7   140000 12.1   150000   35.0 170000  5.3 FUEL IMPU
4 200000  2.0    220000 1.0    230000   3.0  240000  4.0 FUEL IMPU
4 250000  1.7    260000 18.0   270000   1.0  280000 24.0 FUEL IMPU
4 290000  1.0    300000 40.3   420000   10.0 470000 0.1  FUEL IMPU
4 480000  25.0   490000 2.0    500000   4.0  640000 2.5  FUEL IMPU
4 740000  2.0    820000 1.0    830000   0.4  0      0.0  FUEL IMPU
0
4 400000  979.11 500000 16.0   260000   2.25   240000 1.25  ZIRC-4
4 280000  0.02   130000 0.024 050000 0.00033 480000 0.00025 ZIRC-4
4 060000  0.120  270000 0.010  290000   0.020  720000 0.078 ZIRC-4
4 010000  0.013  250000 0.020  070000   0.080  080000 0.950 ZIRC-4
4 160000  0.035  220000 0.020  740000   0.020  230000 0.020 ZIRC-4
5 920000  0.0002      0 0.0                                 ZIRC-4
0
4 260000  688.44 240000 190.0  280000   89.2   250000 20.0  SS-304
4 060000  0.8    150000 0.45   160000   0.3    140000 10.0  SS-304
4 070000  1.3    270000 0.8         0   0.0                 SS-304
0
4 260000  697.74 240000 180.0  280000   89.2   250000 20.0  SS-302
4 060000  1.5    150000 0.45   160000   0.3    140000 10.0  SS-302
4 070000  1.3    270000 0.8         0   0.0                 SS-302
0
4 260000 179.766 240000 189.753 280000 519.625 130000 5.992 INC-718
4 060000  0.4    270000 4.694  290000   0.999  250000 1.997 INC-718
4 420000  29.961 070000 1.3    410000   55.458 160000 0.07  INC-718
4 140000  1.997  220000 7.99         0   0.0                 INC-718
0
4 260000  0.471 240000 149.709 280000  744.438 400000 0.1   NICR-50
4 130000  0.1    050000 0.05   060000   0.1    270000 0.381 NICR-50
4 250000  0.1    070000 0.066  080000 0.043 150000 103.244 NICR-50
4 160000  0.1    140000 0.511  220000   0.1    740000 0.1   NICR-50
0
```

Table D.3. Sample ORIGEN2 input file for BWR-US irradiation and decay

```
-1
-1
-1
RDA *  BURNUP OF BWR 3.0% UO2 FUEL & ASSY HDWARE, 27,500 MWD/MT
RDA     **  CROSS SECTION LIBRARY = BWRUS, 4 CYCLE
RDA     ***  SCOTT B. LUDWIG, OAK RIDGE NATIONAL LABORATORY
RDA     ****  (615) 574-7916, FTS 624-7916
RDA     -1 = FRESH BWR FUEL WITH IMPURITIES (1 MT = 1000 KG)
RDA     -2 = FRESH ZIRCALOY-4 COMPOSITION (1 KG)
RDA     -3 = FRESH ZIRCALOY-2 COMPOSITION (1 KG)
RDA     -4 = FRESH SS 304 COMPOSITION (1 KG)
RDA     -5 = FRESH SS 302 COMPOSITION (1 KG)
RDA     -6 = FRESH INCONEL X-750 COMPOSITION (1 KG)
RDA     WARNING:  VECTORS ARE CHANGED WITH RESPECT TO CONTENT.
RDA               THESE CHANGES WILL BE NOTED ON RDA CARDS.
CUT     5 1.0E-10 7 1.0E-10 9 1.0E-10 -1
LIP     0 0 0
RDA        DECAY LIB    XSECT LIB                        VAR. XSECT
LIB     0    1 2 3     651 652 653    9   50   0   1       40
RDA      PHOTON LIB
PHO     101 102 103  10
TIT  INITIAL COMP. OF UNIT AMOUNTS OF FUEL AND STRUCTURAL MAT'LS
RDA     READ FUEL COMPOSITION INCLUDING IMPURITIES (1000 KG)
INP     -1   1  -1  -1  1  1
RDA     READ ZIRCALOY-4 COMPOSITION (1.0 KG)
INP     -2   1  -1  -1  1  1
RDA     READ ZIRCALOY-2 COMPOSITION (1.0 KG)
INP     -3   1  -1  -1  1  1
RDA     READ SS304 COMPOSITION (1.0 KG)
INP     -4   1  -1  -1  1  1
RDA     READ SS302 COMPOSITION (1.0 KG)
INP     -5   1  -1  -1  1  1
RDA     READ INCONEL X-750 COMPOSITION (1.0 KG)
INP     -6   1  -1  -1  1  1
TIT     IRRADIATION OF ONE METRIC TON OF BWRU FUEL
MOV     -1   1   0   1.0
PCH      1   1   1
HED      1                                                CHARGE
BUP
IRP      96.50   25.90   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRP     193.05   25.90   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRP     265.45   25.90   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45           4   5   4  0    DECAY FOR 106.0 DAYS
IRP     414.90   25.90   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRP     492.10   25.90   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRP     588.65   25.90   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRP     636.90   25.90   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90           9  10   4  0    DECAY FOR 106.0 DAYS
IRP     791.15   25.90  10  11   4  0   BURNUP=15,000 MWD/MTIHM
```

Table D.3 (continued)

```
IRP     887.70   25.90  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRP     984.20   25.90  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRP    1008.35   25.90   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35           2   3   4  0    DECAY FOR 106.0 DAYS
IRP    1186.75   25.90   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRP    1283.25   25.90   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRP    1379.80   25.90   5   6   4  0   BURNUP=27,500 MWD/MTIHM
BUP
RDA    -10 = IRRADIATED U FUEL AT DISCHARGE
MOV      6    -10     0     1.0
PCH     -10   -10   -10
RDA     IRRADIATION OF ZIRCALOY-2 CLADDING AT 1.000 FLUX
TIT     IRRADIATION OF ZIRCALOY-2 CLADDING AT 1.000 FLUX
MOV      -3     1     0   279.50       ZIRCALOY-2 CLAD
HED      1                                              CHARGE
IRF      96.50   -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -1.0   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -1.0   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -1.0   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -1.0   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -1.0   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -1.0  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -1.0  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -1.0  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -1.0   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -1.0   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -1.0   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -1.0   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA      -9 = IRRADIATED ZIRCALOY-2 CLADDING AT DISCHARGE
MOV       6     -9      0      1.0
RDA      IRRADIATION OF ZIRCALOY CLADDING AT 0.500 FLUX
TIT      IRRADIATION OF ZIRCALOY CLADDING AT 0.500 FLUX
MOV      -3      1      0     25.4      ZIRCALOY-2 CLAD
MOV      -3      2      0    304.90     TOTAL ZIRC-2 CLAD
PCH       2      2      2
HED      1                                              CHARGE
IRF      96.50   -0.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -0.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -0.5   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -0.5   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -0.5   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -0.5   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -0.5   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
```

Table D.3 (continued)

```
IRF     791.15   -0.5  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -0.5  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -0.5  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -0.5   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -0.5   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -0.5   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -0.5   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA      -9 = IRRADIATED ZIRCALOY CLADDING AT DISCHARGE
ADD       6     -9      0      1.0    TOTAL IRRAD ZIRC-2 CLAD
PCH      -9     -9     -9
RDA       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 1.000 FLUX
TIT       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 1.000 FLUX
MOV      -2      1      0    227.50     ZIRCALOY-4 CHANNEL
HED      1                                                CHARGE
IRF      96.50   -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -1.0   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -1.0   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -1.0   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -1.0   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -1.0   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -1.0  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -1.0  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -1.0  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -1.0   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -1.0   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -1.0   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -1.0   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA      -8 = IRRADIATED ZIRCALOY CHANNEL AT DISCHARGE
MOV       6     -8      0      1.0
RDA       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 0.500 FLUX
TIT       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 0.500 FLUX
MOV      -2      1     0     20.70     ZIRCALOY-4 CHANNEL
MOV      -2      2     0    248.20     TOTAL ZIRCALOY-4 CHANNEL
PCH       2      2     2
HED      1                                                CHARGE
IRF      96.50   -0.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -0.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -0.5   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -0.5   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -0.5   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -0.5   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -0.5   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
```

Table D.3 (continued)

```
IRF     791.15   -0.5  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -0.5  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -0.5  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -0.5   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -0.5   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -0.5   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -0.5   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA      -8 = IRRADIATED ZIRCALOY CHANNEL AT DISCHARGE
ADD       6     -8      0      1.0
PCH      -8     -8     -8
RDA    IRRADIATION OF ZIRCALOY GRID SPACERS AND INCONEL SPRINGS
TIT    IRRADIATION OF ZIRCALOY GRID SPACERS AND INCONEL SPRINGS
MOV      -2      1      0     10.60     ZIRCALOY GRIDS
ADD      -6      1      0      1.80     INCONEL SPRINGS
PCH      1      1      1
HED      1                                                CHARGE
IRF      96.50   -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -1.0   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -1.0   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -1.0   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -1.0   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -1.0   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -1.0  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -1.0  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -1.0  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -1.0   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -1.0   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -1.0   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -1.0   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA    -7 = IRRADIATED ZIRCALOY GRID SPACERS AND INCONEL SPRINGS
RDA           AT DISCHARGE
MOV       6     -7     0      1.0
PCH      -7     -7     -7
RDA  IRRAD. OF SS 304 END PIECES & INCONEL SPRINGS AT 0.130 FLUX
TIT  IRRAD. OF SS 304 END PIECES & INCONEL SPRINGS AT 0.130 FLUX
MOV      -4      1     0    37.00      SS 304 IN END PIECES
ADD      -6      1     0     2.10  INCONEL X-750 EXPANSION SPRINGS
PCH       1      1     1
HED      1                                                CHARGE
IRF      96.50   -0.13   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -0.13   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -0.13   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -0.13   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
```

Table D.3 (continued)

```
IRF     492.10   -0.13   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -0.13   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -0.13   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90           9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -0.13  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -0.13  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -0.13  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -0.13   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35           2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -0.13   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -0.13   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -0.13   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA -6 = IRRADIATED SS 304 END PIECES & INCONEL SPRINGS @ DISCH.
MOV       6    -6    0    1.0
PCH      -6    -6    -6
RDA      IRRADIATION OF SS 302 IN PLENUM SPRINGS AT 0.500 FLUX
TIT      IRRADIATION OF SS 302 IN PLENUM SPRINGS AT 0.500 FLUX
MOV      -5     1     0     6.00       SS302
PCH       1     1     1
HED      1                                                CHARGE
IRF      96.50   -0.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -0.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -0.5   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -0.5   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -0.5   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -0.5   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -0.5   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -0.5  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -0.5  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -0.5  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -0.5   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -0.5   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -0.5   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -0.5   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA     -5 = IRRADIATED SS 302 IN PLENUM SPRINGS AT DISCHARGE
MOV      6    -5   0   1.0
PCH     -5    -5   -5
RDA     ***** OUTPUT MODULE *****
TIT   TEST OF ORIGEN2 V2.1 - BWR FUEL - STANDARD BURNUP (BWRUS)
BAS      1 MTIHM 3.0% UO2;BURNUP=27,500 MWD/MTIHM, 4 CYCLE
HED      -10    FUEL DIS
HED      -1     FUEL CHG
MOV      -1    1    0   1.0
MOV     -10    2    0   1.0
RDA     ***** DECAY MODULE *****
DEC     0.5   2   4   5   2
```

Table D.3 (continued)

```
  DEC      1.0   4   3   5   0
  DEC      2.0   3   4   5   0
  DEC      5.0   4   5   5   0
  DEC     10.0   5   6   5   0
  OPTL      4*8  7  8  7  8  7  8  8*8  5*7  8
  OPTA      4*8  7  8  7  8  7  8  8*8  5*7  8
  OPTF      4*8  7  8  7  8  7  8  8*8  5*7  8
  OUT       6    1     -1    0
  RDA      ***** OUTPUT MODULE *****
  TIT TEST OF ORIGEN2 V2.1 - BWR FUEL ASSY-STANDARD BURNUP (BWRUS)
  BAS  1 MTIHM 3.0% UO2 FUEL ASSY;BURNUP=27,500 MWD/MTIHM, 4 CYCLE
  OPTL      6*8  7  8  7  8  14*8
  OPTA      6*8  7  8  7  8  14*8
  OPTF      6*8  7  8  7  8  14*8
  MOV      -10    1    0   1.0
  ADD       -9    1    0   1.0
  ADD       -8    1    0   1.0
  ADD       -7    1    0   1.0
  ADD       -6    1    0   1.0
  ADD       -5    1    0   1.0
  HED      1    ASSY DIS
  RDA      ***** DECAY MODULE *****
  DEC      0.5   1   3   5   2
  DEC      1.0   3   2   5   0
  DEC      2.0   2   3   5   0
  DEC      5.0   3   4   5   0
  DEC     10.0   4   5   5   0
  OUT       5    1     -1    0
  END
2 922340  269.0  922350 30000. 922380 969731.    0   0.0 FUEL 3.0%
4 030000  1.0    050000 1.0    060000 89.4  070000  25.0 FUEL IMPU
4 080000 134454. 090000 10.7   110000 15.0  120000   2.0 FUEL IMPU
4 130000  16.7   140000 12.1   150000 35.0  170000   5.3 FUEL IMPU
4 200000  2.0    220000 1.0    230000 3.0   240000   4.0 FUEL IMPU
4 250000  1.7    260000 18.0   270000 1.0   280000  24.0 FUEL IMPU
4 290000  1.0    300000 40.3   420000 10.0  470000  0.1  FUEL IMPU
4 480000  25.0   490000 2.0    500000 4.0   640000 1573. FUEL IMPU
4 740000  2.0    820000 1.0    830000 0.4   0       0.0  FUEL IMPU
0
4 400000 979.11 500000  16.0   260000   2.25   240000 1.25 ZIRC-4
4 280000 0.02 130000  0.024  050000 0.00033 480000 0.00025 ZIRC-4
4 060000 0.120  270000  0.010  290000  0.020  720000 0.078 ZIRC-4
4 010000 0.013  250000  0.020  070000  0.080  080000 0.950 ZIRC-4
4 160000 0.035  220000  0.020  740000  0.020  230000 0.020 ZIRC-4
5 920000 0.0002      0  0.0                                ZIRC-4
0
4 400000 979.63 500000  16.0   260000   1.5    240000 1.00 ZIRC-2
4 280000 0.5    130000 0.024 050000 0.00033 480000 0.00025 ZIRC-2
4 060000 0.120  270000  0.010  290000  0.020  720000 0.078 ZIRC-2
```

Table D.3 (continued)

```
4 010000 0.013  250000  0.020  070000  0.080  080000 0.950 ZIRC-2
4 160000 0.035  220000  0.020  740000  0.020  230000 0.020 ZIRC-2
5 920000 0.0002      0  0.0                                ZIRC-2
0
4 260000 688.44 240000  190.0  280000  89.2   250000 20.0  SS-304
4 060000 0.8    150000  0.45   160000  0.3    140000 10.0  SS-304
4 070000 1.3    270000  0.8         0  0.0                 SS-304
0
4 260000 697.74 240000  180.0  280000  89.2   250000 20.0  SS-302
4 060000 1.5    150000  0.45   160000  0.3    140000 10.0  SS-302
4 070000 1.3    270000  0.8         0  0.0                 SS-302
0
4 260000 67.846 240000 149.66 280000 721.861 130000 7.982 INC-X-750
4 060000 0.399  270000 6.485 290000  0.499  250000 6.984 INC-X-750
4 070000 1.3    410000 8.98  160000  0.07   140000 2.993 INC-X-750
4 220000 24.943      0 0.0                               INC-X-750
0
```

Table D.4. Sample ORIGEN2 input file for BWR-US0 irradiation and decay

```
-1
-1
-1
RDA *  BURNUP OF BWR 3.0% UO2 FUEL & ASSY HDWARE, 27,500 MWD/MT
RDA     **  CROSS SECTION LIBRARY = BWRUS0, 4 CYCLE
RDA     ***  SCOTT B. LUDWIG, OAK RIDGE NATIONAL LABORATORY
RDA     ****  (615) 574-7916, FTS 624-7916
RDA     -1 = FRESH BWR FUEL WITH IMPURITIES (1 MT = 1000 KG)
RDA     -2 = FRESH ZIRCALOY-4 COMPOSITION (1 KG)
RDA     -3 = FRESH ZIRCALOY-2 COMPOSITION (1 KG)
RDA     -4 = FRESH SS 304 COMPOSITION (1 KG)
RDA     -5 = FRESH SS 302 COMPOSITION (1 KG)
RDA     -6 = FRESH INCONEL X-750 COMPOSITION (1 KG)
RDA     WARNING:  VECTORS ARE CHANGED WITH RESPECT TO CONTENT.
RDA               THESE CHANGES WILL BE NOTED ON RDA CARDS.
CUT     5 1.0E-10 7 1.0E-10 9 1.0E-10 -1
LIP     0 0 0
RDA        DECAY LIB    XSECT LIB                        VAR. XSECT
LIB     0    1 2 3     654 655 656    9   50   0   1       41
RDA      PHOTON LIB
PHO     101 102 103  10
TIT  INITIAL COMP. OF UNIT AMOUNTS OF FUEL AND STRUCTURAL MAT'LS
RDA     READ FUEL COMPOSITION INCLUDING IMPURITIES (1000 KG)
INP     -1   1  -1  -1  1  1
RDA     READ ZIRCALOY-4 COMPOSITION (1.0 KG)
INP     -2   1  -1  -1  1  1
RDA     READ ZIRCALOY-2 COMPOSITION (1.0 KG)
INP     -3   1  -1  -1  1  1
RDA     READ SS304 COMPOSITION (1.0 KG)
INP     -4   1  -1  -1  1  1
RDA     READ SS302 COMPOSITION (1.0 KG)
INP     -5   1  -1  -1  1  1
RDA     READ INCONEL X-750 COMPOSITION (1.0 KG)
INP     -6   1  -1  -1  1  1
TIT     IRRADIATION OF ONE METRIC TON OF BWRU FUEL
MOV     -1   1   0   1.0
PCH      1   1   1
HED      1                                                CHARGE
BUP
IRP      96.50   25.90   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRP     193.05   25.90   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRP     265.45   25.90   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45           4   5   4  0    DECAY FOR 106.0 DAYS
IRP     414.90   25.90   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRP     492.10   25.90   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRP     588.65   25.90   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRP     636.90   25.90   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90           9  10   4  0    DECAY FOR 106.0 DAYS
IRP     791.15   25.90  10  11   4  0   BURNUP=15,000 MWD/MTIHM
```

Table D.4 (continued)

```
IRP     887.70   25.90  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRP     984.20   25.90  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRP    1008.35   25.90   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35           2   3   4  0    DECAY FOR 106.0 DAYS
IRP    1186.75   25.90   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRP    1283.25   25.90   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRP    1379.80   25.90   5   6   4  0   BURNUP=27,500 MWD/MTIHM
BUP
RDA    -10 = IRRADIATED U FUEL AT DISCHARGE
MOV      6    -10     0     1.0
PCH     -10   -10   -10
RDA     IRRADIATION OF ZIRCALOY-2 CLADDING AT 1.000 FLUX
TIT     IRRADIATION OF ZIRCALOY-2 CLADDING AT 1.000 FLUX
MOV      -3     1     0   279.50       ZIRCALOY-2 CLAD
HED      1                                                CHARGE
IRF      96.50   -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -1.0   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -1.0   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -1.0   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -1.0   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -1.0   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -1.0  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -1.0  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -1.0  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -1.0   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -1.0   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -1.0   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -1.0   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA      -9 = IRRADIATED ZIRCALOY-2 CLADDING AT DISCHARGE
MOV       6     -9      0      1.0
RDA      IRRADIATION OF ZIRCALOY CLADDING AT 0.500 FLUX
TIT      IRRADIATION OF ZIRCALOY CLADDING AT 0.500 FLUX
MOV      -3      1      0     25.4      ZIRCALOY-2 CLAD
MOV      -3      2      0    304.90     TOTAL ZIRC-2 CLAD
PCH       2      2      2
HED      1                                                CHARGE
IRF      96.50   -0.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -0.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -0.5   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -0.5   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -0.5   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -0.5   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -0.5   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
```

Table D.4 (continued)

```
IRF     791.15   -0.5  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -0.5  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -0.5  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -0.5   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -0.5   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -0.5   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -0.5   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA      -9 = IRRADIATED ZIRCALOY CLADDING AT DISCHARGE
ADD       6     -9      0      1.0    TOTAL IRRAD ZIRC-2 CLAD
PCH      -9     -9     -9
RDA       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 1.000 FLUX
TIT       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 1.000 FLUX
MOV      -2      1      0    227.50     ZIRCALOY-4 CHANNEL
HED      1                                                  CHARGE
IRF      96.50   -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -1.0   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -1.0   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -1.0   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -1.0   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -1.0   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -1.0  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -1.0  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -1.0  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -1.0   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -1.0   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -1.0   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -1.0   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA      -8 = IRRADIATED ZIRCALOY CHANNEL AT DISCHARGE
MOV       6     -8      0      1.0
RDA       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 0.500 FLUX
TIT       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 0.500 FLUX
MOV      -2      1      0     20.70     ZIRCALOY-4 CHANNEL
MOV      -2      2      0    248.20     TOTAL ZIRCALOY-4 CHANNEL
PCH       2      2      2
HED      1                                                  CHARGE
IRF      96.50   -0.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -0.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -0.5   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -0.5   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -0.5   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -0.5   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -0.5   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
```

Table D.4 (continued)

```
IRF     791.15   -0.5  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -0.5  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -0.5  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -0.5   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -0.5   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -0.5   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -0.5   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA      -8 = IRRADIATED ZIRCALOY CHANNEL AT DISCHARGE
ADD       6     -8      0      1.0
PCH      -8     -8     -8
RDA  IRRADIATION OF ZIRCALOY GRID SPACERS AND INCONEL SPRINGS
TIT  IRRADIATION OF ZIRCALOY GRID SPACERS AND INCONEL SPRINGS
MOV      -2      1      0     10.60     ZIRCALOY GRIDS
ADD      -6      1      0      1.80     INCONEL SPRINGS
PCH      1      1      1
HED      1                                                CHARGE
IRF      96.50   -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -1.0   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45          4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -1.0   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -1.0   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -1.0   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -1.0   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90          9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -1.0  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -1.0  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -1.0  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -1.0   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35          2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -1.0   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -1.0   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -1.0   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA  -7 = IRRADIATED ZIRCALOY GRID SPACERS AND INCONEL SPRINGS
RDA           AT DISCHARGE
MOV       6     -7     0     1.0
PCH      -7     -7     -7
RDA IRRAD. OF SS 304 END PIECES & INCONEL SPRINGS AT 0.130 FLUX
TIT IRRAD. OF SS 304 END PIECES & INCONEL SPRINGS AT 0.130 FLUX
MOV      -4     1     0    37.00      SS 304 IN END PIECES
ADD      -6     1     0     2.10 INCONEL X-750 EXPANSION SPRINGS
PCH       1     1     1
HED      1                                                CHARGE
IRF      96.50   -0.13   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -0.13   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -0.13   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45           4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -0.13   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
```

Table D.4 (continued)

```
IRF     492.10   -0.13   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -0.13   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -0.13   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90           9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -0.13  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -0.13  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -0.13  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -0.13   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35           2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -0.13   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -0.13   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -0.13   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA -6 = IRRAD. SS 304 END PIECES & INCONEL SPRINGS AT DISCHARGE
MOV      6    -6    0    1.0
PCH     -6    -6    -6
RDA     IRRADIATION OF SS 302 IN PLENUM SPRINGS AT 0.500 FLUX
TIT     IRRADIATION OF SS 302 IN PLENUM SPRINGS AT 0.500 FLUX
MOV     -5     1     0     6.00       SS302
PCH      1     1     1
HED     1                                                  CHARGE
IRF      96.50   -0.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.05   -0.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     265.45   -0.5   3   4   4  0   BURNUP= 6,875 MWD/MTIHM
DEC     371.45           4   5   4  0    DECAY FOR 106.0 DAYS
IRF     414.90   -0.5   5   6   4  0   BURNUP= 8,000 MWD/MTIHM
IRF     492.10   -0.5   6   7   4  0   BURNUP=10,000 MWD/MTIHM
IRF     588.65   -0.5   7   8   4  0   BURNUP=12,500 MWD/MTIHM
IRF     636.90   -0.5   8   9   4  0   BURNUP=13,750 MWD/MTIHM
DEC     742.90           9  10   4  0    DECAY FOR 106.0 DAYS
IRF     791.15   -0.5  10  11   4  0   BURNUP=15,000 MWD/MTIHM
IRF     887.70   -0.5  11  12   4  0   BURNUP=17,500 MWD/MTIHM
IRF     984.20   -0.5  12   1   4  0   BURNUP=20,000 MWD/MTIHM
IRF    1008.35   -0.5   1   2   4  0   BURNUP=20,625 MWD/MTIHM
DEC    1114.35           2   3   4  0    DECAY FOR 106.0 DAYS
IRF    1186.75   -0.5   3   4   4  0   BURNUP=22,500 MWD/MTIHM
IRF    1283.25   -0.5   4   5   4  0   BURNUP=25,000 MWD/MTIHM
IRF    1379.80   -0.5   5   6   4  0   BURNUP=27,500 MWD/MTIHM
RDA     -5 = IRRADIATED SS 302 IN PLENUM SPRINGS AT DISCHARGE
MOV      6    -5   0   1.0
PCH     -5    -5   -5
RDA     ***** OUTPUT MODULE *****
TIT  TEST OF ORIGEN2 V2.1 - BWR FUEL - STANDARD BURNUP (BWRUS0)
BAS      1 MTIHM 3.0% UO2;BURNUP=27,500 MWD/MTIHM, 4 CYCLE
HED     -10    FUEL DIS
HED     -1     FUEL CHG
MOV     -1    1    0   1.0
MOV    -10    2    0   1.0
RDA     ***** DECAY MODULE *****
DEC     0.5   2   4   5   2
```

Table D.4 (continued)

```
  DEC      1.0   4   3   5   0
  DEC      2.0   3   4   5   0
  DEC      5.0   4   5   5   0
  DEC     10.0   5   6   5   0
  OPTL      4*8  7  8  7  8  7  8  8*8  5*7  8
  OPTA      4*8  7  8  7  8  7  8  8*8  5*7  8
  OPTF      4*8  7  8  7  8  7  8  8*8  5*7  8
  OUT      6    1     -1    0
  RDA     ***** OUTPUT MODULE *****
  TIT TEST OF ORIGEN2 V2.1 - BWR FUEL ASSY-STANDARD BURNUP (BWRUS0)
  BAS 1 MTIHM 3.0% UO2 FUEL ASSY;BURNUP=27,500 MWD/MTIHM, 4 CYCLE
  OPTL      6*8  7  8  7  8  14*8
  OPTA      6*8  7  8  7  8  14*8
  OPTF      6*8  7  8  7  8  14*8
  MOV     -10    1    0   1.0
  ADD      -9    1    0   1.0
  ADD      -8    1    0   1.0
  ADD      -7    1    0   1.0
  ADD      -6    1    0   1.0
  ADD      -5    1    0   1.0
  HED     1    ASSY DIS
  RDA     ***** DECAY MODULE *****
  DEC      0.5   1   3   5   2
  DEC      1.0   3   2   5   0
  DEC      2.0   2   3   5   0
  DEC      5.0   3   4   5   0
  DEC     10.0   4   5   5   0
  OUT      5    1     -1    0
  END
2 922340  269.0 922350 30000. 922380   969731.  0  0.0   FUEL 3.0%
4 030000  1.0    050000 1.0    060000 89.4 070000   25.0 FUEL IMPU
4 080000 134454. 090000 10.7   110000 15.0 120000    2.0 FUEL IMPU
4 130000  16.7   140000 12.1   150000 35.0 170000    5.3 FUEL IMPU
4 200000  2.0    220000 1.0    230000 3.0  240000    4.0 FUEL IMPU
4 250000  1.7    260000 18.0   270000 1.0  280000   24.0 FUEL IMPU
4 290000  1.0    300000 40.3   420000 10.0 470000   0.1  FUEL IMPU
4 480000  25.0   490000 2.0    500000 4.0  640000 1573.  FUEL IMPU
4 740000  2.0    820000 1.0    830000 0.4  0        0.0  FUEL IMPU
0
4 400000 979.11 500000  16.0   260000   2.25   240000 1.25  ZIRC-4
4 280000 0.02   130000  0.024 050000 0.00033 480000 0.00025 ZIRC-4
4 060000 0.120  270000  0.010  290000   0.020  720000 0.078 ZIRC-4
4 010000 0.013  250000  0.020  070000   0.080  080000 0.950 ZIRC-4
4 160000 0.035  220000  0.020  740000   0.020  230000 0.020 ZIRC-4
5 920000 0.0002      0  0.0                                 ZIRC-4
0
4 400000 979.63 500000  16.0   260000 1.5    240000   1.00  ZIRC-2
4 280000 0.5  130000 0.024 050000 0.00033 480000  0.00025   ZIRC-2
4 060000 0.120  270000  0.010  290000 0.020  720000   0.078 ZIRC-2
```

Table D.4 (continued)

```
4 010000 0.013  250000  0.020  070000 0.080  080000   0.950 ZIRC-2
4 160000 0.035  220000  0.020  740000 0.020  230000   0.020 ZIRC-2
5 920000 0.0002      0  0.0                                  ZIRC-2
0
4 260000 688.44 240000  190.0  280000 89.2   250000   20.0  SS-304
4 060000 0.8    150000  0.45   160000 0.3    140000   10.0  SS-304
4 070000 1.3    270000  0.8         0 0.0                   SS-304
0
4 260000 697.74 240000  180.0  280000 89.2   250000   20.0  SS-302
4 060000 1.5    150000  0.45   160000 0.3    140000   10.0  SS-302
4 070000 1.3    270000  0.8         0 0.0                   SS-302
0
4 260000 67.846 240000 149.66 280000 721.861 130000 7.982 INC-X-750
4 060000 0.399  270000  6.485  290000 0.499  250000 6.984 INC-X-750
4 070000 1.3    410000   8.98  160000 0.07   140000 2.993 INC-X-750
4 220000 24.943      0   0.0                              INC-X-750
0
```

Table D.5. Sample ORIGEN2 input file for BWR-UE irradiation and decay

```
-1
-1
-1
RDA  *  BURNUP OF BWR 3.4% UO2 FUEL & ASSY HWARE, 40,000 MWD/MT
RDA     **  CROSS SECTION LIBRARY = BWRUE, 4 CYCLE
RDA     ***  SCOTT B. LUDWIG, OAK RIDGE NATIONAL LABORATORY
RDA     ****  (615) 574-7916, FTS 624-7916
RDA     -1 = FRESH BWR FUEL WITH IMPURITIES (1 MT = 1000 KG)
RDA     -2 = FRESH ZIRCALOY-4 COMPOSITION (1 KG)
RDA     -3 = FRESH ZIRCALOY-2 COMPOSITION (1 KG)
RDA     -4 = FRESH SS 304 COMPOSITION (1 KG)
RDA     -5 = FRESH SS 302 COMPOSITION (1 KG)
RDA     -6 = FRESH INCONEL X-750 COMPOSITION (1 KG)
RDA     WARNING:  VECTORS ARE CHANGED WITH RESPECT TO CONTENT.
RDA               THESE CHANGES WILL BE NOTED ON RDA CARDS.
CUT     5 1.0E-10 7 1.0E-10 9 1.0E-10 -1
LIP     0 0 0
RDA        DECAY LIB    XSECT LIB                        VAR. XSECT
LIB     0    1 2 3     657 658 659    9   50   0   1       42
RDA      PHOTON LIB
PHO     101 102 103  10
TIT   INITIAL COMP. OF UNIT AMOUNTS OF FUEL AND STRUCTURAL MAT'LS
RDA     READ FUEL COMPOSITION INCLUDING IMPURITIES (1000 KG)
INP     -1   1  -1  -1  1  1
RDA     READ ZIRCALOY-4 COMPOSITION (1.0 KG)
INP     -2   1  -1  -1  1  1
RDA     READ ZIRCALOY-2 COMPOSITION (1.0 KG)
INP     -3   1  -1  -1  1  1
RDA     READ SS304 COMPOSITION (1.0 KG)
INP     -4   1  -1  -1  1  1
RDA     READ SS302 COMPOSITION (1.0 KG)
INP     -5   1  -1  -1  1  1
RDA     READ INCONEL X-750 COMPOSITION (1.0 KG)
INP     -6   1  -1  -1  1  1
TIT     IRRADIATION OF ONE METRIC TON OF BWRU FUEL
MOV     -1   1   0   1.0
PCH      1   1   1
HED      1                                                CHARGE
BUP
IRP      96.5   25.9   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRP     193.0   25.9   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRP     289.6   25.9   3   4   4  0   BURNUP= 7,500 MWD/MTIHM
IRP     386.1   25.9   4   5   4  0   BURNUP=10,000 MWD/MTIHM
DEC     492.1          5   6   4  0    DECAY FOR 106.0 DAYS
IRP     620.8   25.9   6   7   4  0   BURNUP=13,333 MWD/MTIHM
IRP     749.5   25.9   7   8   4  0   BURNUP=16,667 MWD/MTIHM
IRP     878.2   25.9   8   9   4  0   BURNUP=20,000 MWD/MTIHM
DEC     984.2          9  10   4  0    DECAY FOR 106.0 DAYS
IRP    1112.9   25.9  10  11   4  0   BURNUP=23,333 MWD/MTIHM
```

Table D.5 (continued)

```
IRP    1241.6   25.9  11  12   4  0   BURNUP=26,667 MWD/MTIHM
IRP    1370.3   25.9  12   1   4  0   BURNUP=30,000 MWD/MTIHM
DEC    1476.3          1   2   4  0    DECAY FOR 106.0 DAYS
IRP    1605.0   25.9   2   3   4  0   BURNUP=33,333 MWD/MTIHM
IRP    1733.7   25.9   3   4   4  0   BURNUP=36,667 MWD/MTIHM
IRP    1862.4   25.9   4   5   4  0   BURNUP=40,000 MWD/MTIHM
BUP
RDA    -10 = IRRADIATED U FUEL AT DISCHARGE
MOV      5    -10     0     1.0
PCH     -10   -10   -10
RDA     IRRADIATION OF ZIRCALOY-2 CLADDING AT 1.000 FLUX
TIT     IRRADIATION OF ZIRCALOY-2 CLADDING AT 1.000 FLUX
MOV      -3     1     0   279.50       ZIRCALOY-2 CLAD
HED      1                                            CHARGE
IRF      96.5   -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.0   -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     289.6   -1.0   3   4   4  0   BURNUP= 7,500 MWD/MTIHM
IRF     386.1   -1.0   4   5   4  0   BURNUP=10,000 MWD/MTIHM
DEC     492.1          5   6   4  0    DECAY FOR 106.0 DAYS
IRF     620.8   -1.0   6   7   4  0   BURNUP=13,333 MWD/MTIHM
IRF     749.5   -1.0   7   8   4  0   BURNUP=16,667 MWD/MTIHM
IRF     878.2   -1.0   8   9   4  0   BURNUP=20,000 MWD/MTIHM
DEC     984.2          9  10   4  0    DECAY FOR 106.0 DAYS
IRF    1112.9   -1.0  10  11   4  0   BURNUP=23,333 MWD/MTIHM
IRF    1241.6   -1.0  11  12   4  0   BURNUP=26,667 MWD/MTIHM
IRF    1370.3   -1.0  12   1   4  0   BURNUP=30,000 MWD/MTIHM
DEC    1476.3          1   2   4  0    DECAY FOR 106.0 DAYS
IRF    1605.0   -1.0   2   3   4  0   BURNUP=33,333 MWD/MTIHM
IRF    1733.7   -1.0   3   4   4  0   BURNUP=36,667 MWD/MTIHM
IRF    1862.4   -1.0   4   5   4  0   BURNUP=40,000 MWD/MTIHM
RDA      -9 = IRRADIATED ZIRCALOY-2 CLADDING AT DISCHARGE
MOV       5     -9      0      1.0
RDA      IRRADIATION OF ZIRCALOY CLADDING AT 0.500 FLUX
TIT      IRRADIATION OF ZIRCALOY CLADDING AT 0.500 FLUX
MOV      -3      1      0     25.4     ZIRCALOY-2 CLAD
MOV      -3      2      0    304.90    TOTAL ZIRC-2 CLAD
PCH       2      2      2
HED      1                                            CHARGE
IRF      96.5   -0.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.0   -0.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     289.6   -0.5   3   4   4  0   BURNUP= 7,500 MWD/MTIHM
IRF     386.1   -0.5   4   5   4  0   BURNUP=10,000 MWD/MTIHM
DEC     492.1          5   6   4  0    DECAY FOR 106.0 DAYS
IRF     620.8   -0.5   6   7   4  0   BURNUP=13,333 MWD/MTIHM
IRF     749.5   -0.5   7   8   4  0   BURNUP=16,667 MWD/MTIHM
IRF     878.2   -0.5   8   9   4  0   BURNUP=20,000 MWD/MTIHM
DEC     984.2          9  10   4  0    DECAY FOR 106.0 DAYS
IRF    1112.9   -0.5  10  11   4  0   BURNUP=23,333 MWD/MTIHM
IRF    1241.6   -0.5  11  12   4  0   BURNUP=26,667 MWD/MTIHM
```

Table D.5 (continued)

```
IRF    1370.3   -0.5  12   1   4  0   BURNUP=30,000 MWD/MTIHM
DEC    1476.3         1   2   4  0    DECAY FOR 106.0 DAYS
IRF    1605.0   -0.5   2   3   4  0   BURNUP=33,333 MWD/MTIHM
IRF    1733.7   -0.5   3   4   4  0   BURNUP=36,667 MWD/MTIHM
IRF    1862.4   -0.5   4   5   4  0   BURNUP=40,000 MWD/MTIHM
RDA      -9 = IRRADIATED ZIRCALOY CLADDING AT DISCHARGE
ADD       5     -9      0      1.0    TOTAL IRRAD ZIRC-2 CLAD
PCH      -9     -9     -9
RDA       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 1.000 FLUX
TIT       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 1.000 FLUX
MOV      -2      1      0    227.50     ZIRCALOY-4 CHANNEL
HED      1                                                CHARGE
IRF      96.5   -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.0   -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     289.6   -1.0   3   4   4  0   BURNUP= 7,500 MWD/MTIHM
IRF     386.1   -1.0   4   5   4  0   BURNUP=10,000 MWD/MTIHM
DEC     492.1          5   6   4  0    DECAY FOR 106.0 DAYS
IRF     620.8   -1.0   6   7   4  0   BURNUP=13,333 MWD/MTIHM
IRF     749.5   -1.0   7   8   4  0   BURNUP=16,667 MWD/MTIHM
IRF     878.2   -1.0   8   9   4  0   BURNUP=20,000 MWD/MTIHM
DEC     984.2          9  10   4  0    DECAY FOR 106.0 DAYS
IRF    1112.9   -1.0  10  11   4  0   BURNUP=23,333 MWD/MTIHM
IRF    1241.6   -1.0  11  12   4  0   BURNUP=26,667 MWD/MTIHM
IRF    1370.3   -1.0  12   1   4  0   BURNUP=30,000 MWD/MTIHM
DEC    1476.3         1   2   4  0    DECAY FOR 106.0 DAYS
IRF    1605.0   -1.0   2   3   4  0   BURNUP=33,333 MWD/MTIHM
IRF    1733.7   -1.0   3   4   4  0   BURNUP=36,667 MWD/MTIHM
IRF    1862.4   -1.0   4   5   4  0   BURNUP=40,000 MWD/MTIHM
RDA      -8 = IRRADIATED ZIRCALOY CHANNEL AT DISCHARGE
MOV       5     -8      0      1.0
RDA       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 0.500 FLUX
TIT       IRRADIATION OF ZIRCALOY-4 CHANNEL AT 0.500 FLUX
MOV      -2      1      0     20.70     ZIRCALOY-4 CHANNEL
MOV      -2      2      0    248.20     TOTAL ZIRCALOY-4 CHANNEL
PCH       2      2      2
HED      1                                                CHARGE
IRF      96.5   -0.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.0   -0.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     289.6   -0.5   3   4   4  0   BURNUP= 7,500 MWD/MTIHM
IRF     386.1   -0.5   4   5   4  0   BURNUP=10,000 MWD/MTIHM
DEC     492.1          5   6   4  0    DECAY FOR 106.0 DAYS
IRF     620.8   -0.5   6   7   4  0   BURNUP=13,333 MWD/MTIHM
IRF     749.5   -0.5   7   8   4  0   BURNUP=16,667 MWD/MTIHM
IRF     878.2   -0.5   8   9   4  0   BURNUP=20,000 MWD/MTIHM
DEC     984.2          9  10   4  0    DECAY FOR 106.0 DAYS
IRF    1112.9   -0.5  10  11   4  0   BURNUP=23,333 MWD/MTIHM
IRF    1241.6   -0.5  11  12   4  0   BURNUP=26,667 MWD/MTIHM
IRF    1370.3   -0.5  12   1   4  0   BURNUP=30,000 MWD/MTIHM
DEC    1476.3         1   2   4  0    DECAY FOR 106.0 DAYS
```

Table D.5 (continued)

```
IRF    1605.0   -0.5   2   3   4  0   BURNUP=33,333 MWD/MTIHM
IRF    1733.7   -0.5   3   4   4  0   BURNUP=36,667 MWD/MTIHM
IRF    1862.4   -0.5   4   5   4  0   BURNUP=40,000 MWD/MTIHM
RDA      -8 = IRRADIATED ZIRCALOY CHANNEL AT DISCHARGE
ADD       5     -8      0      1.0
PCH      -8     -8     -8
RDA  IRRADIATION OF ZIRCALOY GRID SPACERS AND INCONEL SPRINGS
TIT  IRRADIATION OF ZIRCALOY GRID SPACERS AND INCONEL SPRINGS
MOV      -2      1      0     10.60     ZIRCALOY GRIDS
ADD      -6      1      0      1.80     INCONEL SPRINGS
PCH      1      1      1
HED      1                                                CHARGE
IRF      96.5   -1.0   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.0   -1.0   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     289.6   -1.0   3   4   4  0   BURNUP= 7,500 MWD/MTIHM
IRF     386.1   -1.0   4   5   4  0   BURNUP=10,000 MWD/MTIHM
DEC     492.1          5   6   4  0    DECAY FOR 106.0 DAYS
IRF     620.8   -1.0   6   7   4  0   BURNUP=13,333 MWD/MTIHM
IRF     749.5   -1.0   7   8   4  0   BURNUP=16,667 MWD/MTIHM
IRF     878.2   -1.0   8   9   4  0   BURNUP=20,000 MWD/MTIHM
DEC     984.2          9  10   4  0    DECAY FOR 106.0 DAYS
IRF    1112.9   -1.0  10  11   4  0   BURNUP=23,333 MWD/MTIHM
IRF    1241.6   -1.0  11  12   4  0   BURNUP=26,667 MWD/MTIHM
IRF    1370.3   -1.0  12   1   4  0   BURNUP=30,000 MWD/MTIHM
DEC    1476.3          1   2   4  0    DECAY FOR 106.0 DAYS
IRF    1605.0   -1.0   2   3   4  0   BURNUP=33,333 MWD/MTIHM
IRF    1733.7   -1.0   3   4   4  0   BURNUP=36,667 MWD/MTIHM
IRF    1862.4   -1.0   4   5   4  0   BURNUP=40,000 MWD/MTIHM
RDA      -7 = IRRADIATED ZIRCALOY GRID SPACERS AND INCONEL SPRINGS
RDA           AT DISCHARGE
MOV       6     -7     0     1.0
PCH      -7     -7     -7
RDA  IRRAD. OF SS 304 END PIECES AND INCONEL SPRINGS AT 0.130 FLUX
TIT  IRRAD. OF SS 304 END PIECES AND INCONEL SPRINGS AT 0.130 FLUX
MOV      -4      1     0    37.00      SS 304 IN END PIECES
ADD   -6     1     0     2.10      INCONEL X-750 EXPANSION SPRINGS
PCH       1      1     1
HED      1                                                CHARGE
IRF      96.5   -0.13   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.0   -0.13   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     289.6   -0.13   3   4   4  0   BURNUP= 7,500 MWD/MTIHM
IRF     386.1   -0.13   4   5   4  0   BURNUP=10,000 MWD/MTIHM
DEC     492.1           5   6   4  0    DECAY FOR 106.0 DAYS
IRF     620.8   -0.13   6   7   4  0   BURNUP=13,333 MWD/MTIHM
IRF     749.5   -0.13   7   8   4  0   BURNUP=16,667 MWD/MTIHM
IRF     878.2   -0.13   8   9   4  0   BURNUP=20,000 MWD/MTIHM
DEC     984.2           9  10   4  0    DECAY FOR 106.0 DAYS
IRF    1112.9   -0.13  10  11   4  0   BURNUP=23,333 MWD/MTIHM
IRF    1241.6   -0.13  11  12   4  0   BURNUP=26,667 MWD/MTIHM
```

Table D.5 (continued)

```
IRF    1370.3   -0.13  12   1   4  0   BURNUP=30,000 MWD/MTIHM
DEC    1476.3           1   2   4  0    DECAY FOR 106.0 DAYS
IRF    1605.0   -0.13   2   3   4  0   BURNUP=33,333 MWD/MTIHM
IRF    1733.7   -0.13   3   4   4  0   BURNUP=36,667 MWD/MTIHM
IRF    1862.4   -0.13   4   5   4  0   BURNUP=40,000 MWD/MTIHM
RDA  -6 - IRRAD. SS 304 END PIECES AND INCONEL SPRINGS AT DISCH
MOV       6    -6    0    1.0
PCH      -6    -6    -6
RDA      IRRADIATION OF SS 302 IN PLENUM SPRINGS AT 0.500 FLUX
TIT      IRRADIATION OF SS 302 IN PLENUM SPRINGS AT 0.500 FLUX
MOV      -5     1     0     6.00       SS302
PCH       1     1     1
HED      1                                                CHARGE
IRF      96.5   -0.5   1   2   4  2   BURNUP= 2,500 MWD/MTIHM
IRF     193.0   -0.5   2   3   4  0   BURNUP= 5,000 MWD/MTIHM
IRF     289.6   -0.5   3   4   4  0   BURNUP= 7,500 MWD/MTIHM
IRF     386.1   -0.5   4   5   4  0   BURNUP=10,000 MWD/MTIHM
DEC     492.1          5   6   4  0    DECAY FOR 106.0 DAYS
IRF     620.8   -0.5   6   7   4  0   BURNUP=13,333 MWD/MTIHM
IRF     749.5   -0.5   7   8   4  0   BURNUP=16,667 MWD/MTIHM
IRF     878.2   -0.5   8   9   4  0   BURNUP=20,000 MWD/MTIHM
DEC     984.2          9  10   4  0    DECAY FOR 106.0 DAYS
IRF    1112.9   -0.5  10  11   4  0   BURNUP=23,333 MWD/MTIHM
IRF    1241.6   -0.5  11  12   4  0   BURNUP=26,667 MWD/MTIHM
IRF    1370.3   -0.5  12   1   4  0   BURNUP=30,000 MWD/MTIHM
DEC    1476.3          1   2   4  0    DECAY FOR 106.0 DAYS
IRF    1605.0   -0.5   2   3   4  0   BURNUP=33,333 MWD/MTIHM
IRF    1733.7   -0.5   3   4   4  0   BURNUP=36,667 MWD/MTIHM
IRF    1862.4   -0.5   4   5   4  0   BURNUP=40,000 MWD/MTIHM
RDA     -5 - IRRADIATED SS 302 IN PLENUM SPRINGS AT DISCHARGE
MOV      5    -5   0   1.0
PCH     -5    -5   -5
RDA     ***** OUTPUT MODULE *****
TIT  TEST OF ORIGEN2 V2.1 - BWR FUEL - EXTENDED BURNUP (BWRUE)
BAS      1 MTIHM 3.4% UO2;BURNUP=40,000 MWD/MTIHM (4 CYCLE)
HED      -10    FUEL DIS
HED      -1     FUEL CHG
MOV      -1    1    0   1.0
MOV     -10    2    0   1.0
RDA     ***** DECAY MODULE *****
DEC     0.5   2   4   5   2
DEC     1.0   4   3   5   0
DEC     2.0   3   4   5   0
DEC     5.0   4   5   5   0
DEC    10.0   5   6   5   0
OPTL      4*8  7  8  7  8  7  8  8*8  5*7  8
OPTA      4*8  7  8  7  8  7  8  8*8  5*7  8
OPTF      4*8  7  8  7  8  7  8  8*8  5*7  8
OUT      6    1     -1    0
```

Table D.5 (continued)

```
TIT  TEST OF ORIGEN2 V2.1 - BWR FUEL ASSY-EXTENDED BURNUP (BWRUE)
BAS   1 MTIHM 3.4% UO2 FUEL ASSY;BURNUP=40,000 MWD/MTIHM (4 CYCLE)
OPTL      6*8  7  8  7  8  14*8
OPTA      6*8  7  8  7  8  14*8
OPTF      6*8  7  8  7  8  14*8
MOV     -10    1    0   1.0
ADD      -9    1    0   1.0
ADD      -8    1    0   1.0
ADD      -7    1    0   1.0
ADD      -6    1    0   1.0
ADD      -5    1    0   1.0
HED     1    ASSY DIS
RDA     ***** DECAY MODULE *****
DEC     0.5   1   3   5   2
DEC     1.0   3   2   5   0
DEC     2.0   2   3   5   0
DEC     5.0   3   4   5   0
DEC    10.0   4   5   5   0
OUT      5    1     -1    0
END
2 922340  305.0  922350  34000. 922380   965695. 0   0.0  FUEL 3.4%
4 030000  1.0    050000 1.0    060000 89.4 070000   25.0  FUEL IMPU
4 080000 134454. 090000 10.7   110000 15.0 120000    2.0  FUEL IMPU
4 130000  16.7   140000 12.1   150000 35.0 170000    5.3  FUEL IMPU
4 200000  2.0    220000 1.0    230000 3.0  240000    4.0  FUEL IMPU
4 250000  1.7    260000 18.0   270000 1.0  280000   24.0  FUEL IMPU
4 290000  1.0    300000 40.3   420000 10.0 470000   0.1   FUEL IMPU
4 480000  25.0   490000 2.0    500000 4.0  640000   1573. FUEL IMPU
4 740000  2.0    820000 1.0    830000 0.4  0        0.0   FUEL IMPU
0
4 400000  979.11 500000  16.0   260000   2.25   240000 1.25  ZIRC-4
4 280000 0.02 130000  0.024 050000  0.00033 480000 0.00025   ZIRC-4
4 060000  0.120  270000  0.010  290000   0.020  720000 0.078 ZIRC-4
4 010000  0.013  250000  0.020  070000   0.080  080000 0.950 ZIRC-4
4 160000  0.035  220000  0.020  740000   0.020  230000 0.020 ZIRC-4
5 920000  0.0002      0  0.0                                 ZIRC-4
0
4 400000  979.63 500000  16.0   260000   1.5    240000 1.00  ZIRC-2
4 280000  0.5  130000  0.024 050000 0.00033 480000 0.00025   ZIRC-2
4 060000  0.120  270000  0.010  290000   0.020  720000 0.078 ZIRC-2
4 010000  0.013  250000  0.020  070000   0.080  080000 0.950 ZIRC-2
4 160000  0.035  220000  0.020  740000   0.020  230000 0.020 ZIRC-2
5 920000  0.0002      0  0.0                                 ZIRC-2
0
4 260000  688.44 240000  190.0  280000   89.2   250000 20.0  SS-304
4 060000  0.8    150000  0.45   160000   0.3    140000 10.0  SS-304
4 070000  1.3    270000  0.8         0   0.0                 SS-304
0
4 260000  697.74 240000  180.0  280000   89.2   250000 20.0  SS-302
```

Table D.5 (continued)

```
4 060000  1.5    150000  0.45   160000   0.3    140000 10.0  SS-302
4 070000  1.3    270000  0.8         0   0.0                 SS-302
0
4 260000 67.846 240000 149.66 280000 721.861 130000 7.982 INC-X-750
4 060000  0.399 270000 6.485  290000  0.499  250000 6.984 INC-X-750
4 070000  1.3    410000 8.98  160000  0.07   140000 2.993 INC-X-750
4 220000  24.943      0   0.0                             INC-X-750
0
```

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