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Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code

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CHEMICAL TECHNOLOGY DIVISION

REVISED URANIUM-PLUTONIUM CYCLE PWR AND BWR MODELS FOR THE ORIGEN COMPUTER CODE

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GLOSSARY

BWR	Boiling-water reactor
BWR-PuPu	Self-generated mixed-oxide fuel in a BWR in which the plutonium is being recycled
BWR-PuU	Uranium-enriched fuel in a BWR in which the plutonium is being recycled
BWR-U	Uranium-enriched fuel in a wholly uranium-enriched BWR
GWd	Gigawatt-days = 10^9 watt-days
LWR	Light water (moderated) reactor; i.e., BWRs and PWRs
MWd	Megawatt-days = 10^6 watt-days
MTIHM	Metric tonnes (= 10^6 g) of initial heavy metal
PWR	Pressurized water reactor
PWR-PuPu	Self-generated mixed-oxide fuel in a PWR in which the plutonium is being recycled
PWR-PuU	Uranium-enriched fuel in a PWR in which the plutonium is being recycled
PWR-U	Uranium-enriched fuel in a wholly uranium-enriched PWR
Depletion calculation	Calculational irradiation of fresh reactor fuel resulting in the prediction of the discharged fuel composition
Fuel element	The smallest structurally discrete part of a fuel assembly which has nuclear fuel as its principal constituent; also called a fuel pin or a fuel rod
Fuel assembly	A grouping of fuel elements that is not taken apart during the charging and discharging of a reactor core
Pin cell	A cylindrical model of a fuel element used in a reactor physics calculation
Assembly cell	A cylindrical model of a fuel assembly used in a reactor physics calculation

REVISED URANIUM-PLUTONIUM CYCLE PWR AND BWR MODELS FOR THE ORIGEN COMPUTER CODE

A. G. Croff, M. A. Bjerke,* G. W. Morrison,* and L. M. Petrie*

ABSTRACT

Reactor physics calculations and literature searches have been conducted, leading to the creation of revised enriched-uranium and enriched-uranium/mixed-oxide-fueled PWR and BWR reactor models for the ORIGEN computer code. These ORIGEN reactor models are based on cross sections that have been taken directly from the reactor physics codes and eliminate the need to make adjustments in uncorrected cross sections in order to obtain correct depletion results. Revised values of the ORIGEN flux parameters THERM, RES, and FAST were calculated along with new parameters related to the activation of fuel-assembly structural materials not located in the active fuel zone. Recommended fuel and structural material masses and compositions are presented. A summary of the new ORIGEN reactor models is given.

1. INTRODUCTION AND SUMMARY

1.1 Introduction

1.1.1 Background

The ORIGEN[1] computer code is a versatile tool used for calculating the buildup and depletion of isotopes in nuclear materials. This computer code was written in the late 1960s and early 1970s by the ORNL Chemical Technology Division. At that time, the required nuclear data libraries (half-lives, cross sections, fission product yields, etc.) and reactor models (PWR-U, PWR-Pu, LMFBR, HTGR, and MSBR) were also developed. The code was principally intended for use in generating spent fuel and waste

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characteristics (composition, thermal power, etc.) that would form the basis for the study and design of fuel reprocessing plants, spent fuel shipping casks, waste treatment and disposal facilities, and waste shipping casks. Since fuel cycle operations were being examined generically, and thus were expected to accommodate a wide range of fuel characteristics, it was only necessary that the ORIGEN results be somewhat representative of this range. A satisfactory result was obtained by simply adjusting the resonance integrals of the major fissile and fertile species to obtain agreement with a spent fuel composition from an exogenous source.

Soon after the ORIGEN computer code was documented, it 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 resulted in its being acquired by many organizations. Many of these organizations began using ORIGEN for applications that required greater precision in the calculated results than those for which it had originally been intended. These applications were generally much more specific than the early ORNL generic fuel cycle studies, and many of these were environmental impact studies that required relatively precise calculations of minor isotopes such as ^{3}H, ^{14}C, ^{232}U, and $^{242,244}Cm$. The initial responses to these requirements were attempts to update specific aspects of ORIGEN and its data bases.[2,3] However, inconsistencies and a large number of different data bases soon resulted from these efforts.

In an effort to remedy these problems, a concerted program was initiated in 1975 to update the ORIGEN computer code and its associated

data bases and reactor models. This report is the first of several reports describing the various aspects of the ORIGEN update effort. Other reports to be issued in the future will document a revised version of the ORIGEN computer code, an updated decay data library, an updated photon library, additional updated libraries containing cross sections and fission product yields, and additional reactor models.

1.1.2 Scope

This report is concerned with a description of a revised set of reactor models for PWRs and BWRs operating on uranium or uranium-plutonium fuels and the methods used to generate the information for these models. The PWR model was based on a Westinghouse design,[4] and the BWR was based on a General Electric BWR/6 design.[5] The specific reactor types considered in this report are as follows (see the Glossary for the definition of these and other terms):

1. PWR-U,
2. PWR-Pu (both PWR-PuU and PWR-PuPu fuel types),
3. BWR-U,
4. BWR-Pu (both BWR-PuU and BWR-PuPu fuel types).

The fundamental objective of this work was that ORIGEN would predict the correct spent fuel compositions without having to resort to the adjustment of cross sections, which had typified previous ORIGEN reactor models. This meant that ORIGEN had to be able to use cross sections which resulted from the processing of existing compilations such as ENDF/B.[6]

The generation of the information required for these reactor models began with 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[7] (see Sect. 2). Two separate libraries were created: (1) a smaller library containing those nuclides whose presence in the reactor would have the greatest effects on the neutron spectrum and depletion characteristics and (2) a larger library containing many nuclides of interest in ORIGEN but having a minor effect on the spectrum and depletion. 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 of the six fuel types being considered (see Sect. 3). The libraries resulting from this procedure contained five neutron energy groups. The burnup-dependent, five-group 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 ORIGEN. The cross sections in the larger 84-group library mentioned previously were then collapsed to one-group cross sections using a "typical" neutron spectrum which was calculated while the burnup-dependent cross sections were being generated. Fission product yields were obtained by flux-weighting energy-dependent yields using the same neutron spectrum. Additional calculations were then performed which yielded 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.

A modified version of the ORIGEN code was then developed and used to predict the composition of the spent fuel (see Sect. 4) using the cross sections generated by the more sophisticated reactor physics codes.

The ORIGEN results were compared to both the results predicted by the multigroup fuel-depletion code and to spent fuel compositions obtained from the open literature. Finally, an investigation was undertaken to determine appropriate input parameters for the reactor models. The parameters investigated included the actinide composition of the fresh fuel, the impurity composition of the fresh fuel, and the structural material type and composition of a fuel assembly.

1.2 Summary

This project involved the gathering and processing of a large amount of diverse data which led to the generation of revised ORIGEN reactor models for uranium- and uranium-plutonium-fueled PWRs and BWRs. The specific types of information developed for PWR-U, PWR-PuU, PWR-PuPu, BWR-U, BWR-PuU, and BWR-PuPu fuels 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 233 nuclides (including the actinides);
4. new values for the ORIGEN flux parameters THERM, RES, and FAST;
5. parameters related to the activation of fuel-assembly structural materials outside the active fuel region;
6. recommended initial heavy-metal compositions of fresh fuel;
7. recommended initial metal compositions of fuel-assembly structural materials; and
8. recommended minor constituent concentrations for both the fuel material and the structural materials.

Using this information and a modified version of the ORIGEN computer code, depletion calculations were made for each of the six fuel types. The results of these depletion calculations were compared to the results of diffusion-theory depletion calculations which were performed as a part of this program and to predicted discharge compositions obtained from literature sources.

In general, the results of the ORIGEN depletion calculations agreed very well with the literature values for all three of the PWR fuels. Adequate agreement was obtained for the BWR-U fuel and fair agreement for the BWR-PuPu fuel. Comparative literature values were not available for the BWR-PuU fuel. It should be noted that the moderator density in the BWR calculations was increased to a value 20% greater than the volume-averaged moderator density (i.e., a value nearer the power-averaged moderator density) to obtain the agreement indicated. The results of the PWR-U and BWR-U fuel depletions from ORIGEN and the diffusion-theory code were found to be in excellent agreement. This agreement is expected since ORIGEN uses cross sections that had been generated by the diffusion-theory code. Based on the ORIGEN vs literature comparisons which were made as a part of this project, it appears that the cross section information is adequate for performing depletion calculations for fuel enrichments within a few tenths of a percentage point of the enrichments used in generating the cross sections for the reactor models. However, the depletion calculations will become progressively less accurate as the fuel composition deviates from the reference conditions.

1.3 References for Sect. 1

1. M. J. Bell, ORIGEN — The ORNL Isotope Generation and Depletion Code, ORNL-4628 (May 1973).
2. C. W. Kee, A Revised Light Element Library for the ORIGEN Code, ORNL/TM-4896 (May 1975).
3. C. W. Kee, C. R. Weisbin, and R. E. Schenter, Processing and Testing of ENDF/B-IV Fission Product and Transmutation Data, Trans. Am. Nucl. Soc. 19, 398-399 (1974).
4. Westinghouse Nuclear Energy Systems, RESAR-3, Reference Safety Analysis Report, DOCKET STN 50-480 (1972).
5. General Electric Standard Safety Analysis Report, BWR/6, DOCKET STN 50-447 (1973).
6. ENDF/B-IV Library Tapes 401-411 and 414-419, available from the National Neutron Cross Section Center, Brookhaven National Laboratory (December 1974).
7. N. M. Greene, J. L. Lucius, L. M. Petrie, W. E. Ford III, J. E. White, and R. Q. Wright, AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B, ORNL/TM-3706 (March 1976).

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

This section describes the processing of raw cross section data into a multiple energy group (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-plutonium (U-Pu) cycle LWRs.

A schematic information flow diagram for the processing of the raw cross section data is given in Fig. 2.1. In this diagram, the principal computer codes used in processing the cross sections are contained in rectangular figures. Descriptions of the cross section parameters which comprise the input to and output from each computer code are contained in the oval figures. The sources of the raw cross section data are discussed in Sect. 2.1. The two principal computer codes required to process this raw data into the multigroup cross section library are NPTXS[1] and NEWXLACS.[1]

There are two principal steps involved in processing the raw cross section data into a master library, as depicted in Fig. 2.1. The first of these generates 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 U-Pu cycle reactors using NPTXS. 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. 2.2 and 2.3,

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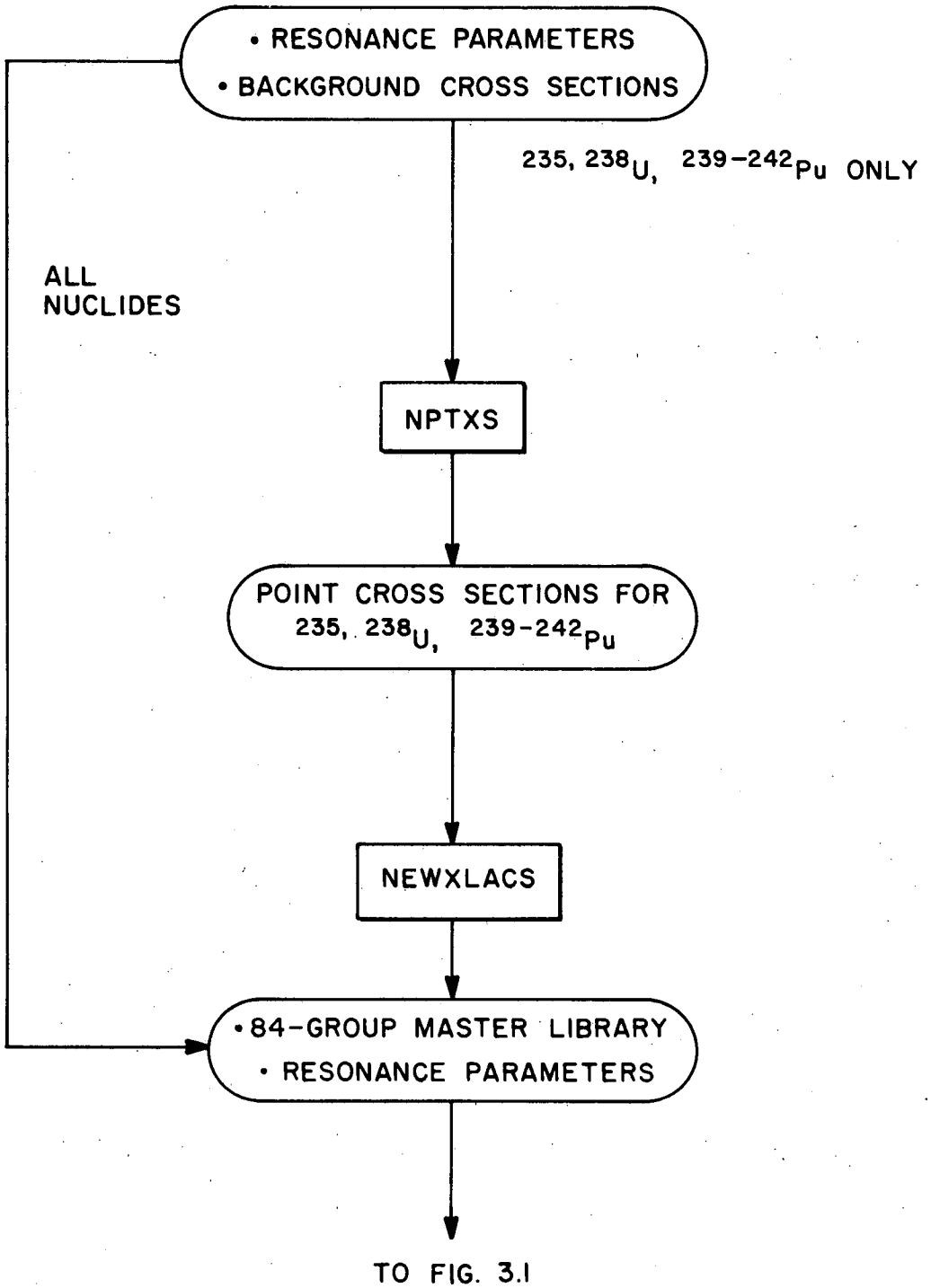

Fig. 2.1. Procedure for processing raw cross section data into an AMPX master interface.

is the master cross section library. A summary description of the processing sequence is given in Table 2.1.

2.1 Scope and Source of Cross Section Data

The cross section data base used in the multigroup reactor physics calculations described in Sect. 3 contains all cross sections that could be cast into a multigroup library (as of spring 1977) and that were machine-readable. These cross sections were obtained from three principal sources: ENDF/B-IV,[2] LENDL (Livermore Evaluated Nuclear Data Library),[3] and Benjamin.[4] The cross section data obtained from these sources 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. The first group of nuclides is neutronically important in the reactor, and all of these nuclides were represented explicitly in the depletion calculations. The cross sections of nuclides in the second group were obtained by collapsing the multigroup cross sections using a neutron spectrum selected after the depletion calculation had been performed. A list of the nuclides in the first and second groups and the sources of the data are given in Tables 2.2 and 2.3.

Certain ^{238}U cross section parameters were adjusted according to a recipe supplied by Benjamin[5] in order to obtain agreement with the experimental data. Changes in the neutron capture cross section were made between 0.00025 eV and 0.7067 eV, in the neutron and radiation widths for the low-lying s-wave resonances between 6.65 eV and 165 eV, and in the unresolved resonance parameters.

Table 2.1. Description of the processing of ENDF/B-formatted data into a master cross section library

Computer code	Input	Function	Output	Applicability of result
NPTXS	Resonance parameters and background cross sections in ENDF/B format	Generate Doppler-broadened point cross sections for resonance nuclides ($^{235,238}U$, $^{239-242}Pu$)	Point cross sections, background cross sections, and resonance parameters for resonance nuclides	All U-Pu LWRs
NEWXLACS	Resonance parameters and background cross sections in ENDF/B format plus point cross sections for resonance nuclides from NPTXS	Generate multigroup cross sections from point cross sections using Maxwellian-1/E-fission-spectrum weighting function	84-group master library, including resonance parameters	All U-Pu LWRs

Table 2.2. ORIGEN update nuclides included in fuel-depletion calculations

Nuclide	Ref.	Nuclide	Ref.	Nuclide	Ref.	Nuclide	Ref.	Nuclide	Ref.
H-1	2	B-10	2	B-11	2	C-12	2	N-14	2
O-16	2	Na-23	2	Cr-52	2	Mn-55	2	Fe-56	2
Ni-58	2	Zr-92[a]	2	Nb-93	2	Mo-98	2	Sn-119	3
Th-232	2	Pa-233	2	U-233	2	U-234	2	U-235	2
U-236	2	U-237	3	U-238[b]	2	U-239	3	U-240	3
Np-237	2	Np-238	4	Pu-236	4	Pu-238	2	Pu-239	2
Pu-240	2	Pu-241	2	Pu-242	2	Pu-243	4	Am-241	2
Am-242M	4	Am-242	4	Am-243	4	Cm-242	4	Cm-243	4
Cm-244	4	Cm-245	4	Cm-246	4	Cm-247	4	Cm-248	4
Bk-249	4	Cf-249	4	Cf-250	4	Cf-251	4	Cf-252	4
Cf-253	4	Cf-254	4	Es-253	4				

[a]Zircaloy-2.

[b]Some resonance parameters adjusted from ref. 2 values according to recipe given in ref. 5.

Table 2.3. ORIGEN update nuclides not included in fuel depletion calculations[a]

GE- 72	GE- 73	GE- 74	AS- 75	GE- 76	SE- 76	SE- 77	SE- 78
BR- 79	SE- 80	KR- 80	BR- 81	SE- 82	KR- 82	KR- 83	KR- 84
KR- 85	RB- 85	KR- 86	RB- 86	SR- 86	RB- 87	SR- 87	SR- 88
SR- 89	Y - 89	SR- 90	Y - 90	ZR- 90	Y - 91	ZR- 91	ZR- 93
ZR- 94	NB- 94	MO- 94	ZR- 95	NB- 95	MO- 95	ZR- 96	MO- 96
MO- 97	MO- 99	TC- 99	RU- 99	MO-100	RU-100	RU-101	RU-102
RU-103	RH-103	RU-104	PD-104	RU-105	RH-105	PD-105	RU-106
PD-106	PD-107	AG-107	PD-108	CD-108	AG-109	PD-110	CD-110
AG-111	CD-111	CD-112	CD-113	IN-113	CD-114	CD-115M	IN-115
SN-115	CD-116	SN-116	SN-117	SN-118	SN-119	SN-120	SB-121
SN-122	TE-122	SN-123	SB-123	TE-123	SN-124	SB-124	TE-124
SN-125	SB-125	TE-125	SN-126	SB-126	TE-126	TE-127M	I -127
TE-128	XE-128	TE-129M	I -129	XE-129	TE-130	I -130	XE-130
I -131	XE-131	TE-132	XE-132	XE-133	CS-133	XE-134	CS-134
BA-134	I -135	XE-135	CS-135	BA-135	XE-136	CS-136	BA-136
CS-137	BA-137	BA-138	LA-139	BA-140	LA-140	CE-140	CE-141
PR-141	CE-142	PR-142	ND-142	CE-143	PR-143	ND-143	CE-144
ND-144	ND-145	ND-146	ND-147	PM-147	SM-147	ND-148	PM-148
PM-148M	SM-148	PM-149	SM-149	ND-150	SM-150	PM-151	SM-151
EU-151	SM-152	EU-152	SM-153	SM-154	EU-154	GD-154	EU-155
GD-155	EU-156	GD-156	EU-157	GD-157	GD-158	TB-159	GD-160
TB-160	DY-160	DY-161	DY-162	DY-163	DY-164	HO-165	ER-166
ER-167							

[a]All cross sections taken from ref. 2.

2.2 Resonance Nuclide Processing in NPTXS

The nuclides included in the first (neutronically important) group can be separated into two classes: resonance nuclides and nonresonance nuclides. Nuclides in the first class are those which contribute significantly to the resonance absorption in the system(s) of interest. For the purposes of this study (U-Pu cycle thermal reactors), the following nuclides fall in this class: ^{235}U, ^{238}U, and $^{239-242}Pu$.

The AMPX[1] module NPTXS is used to process the resonance parameters contained in the ENDF/B data[2] for those nuclides listed above. In the resolved resonance range, each resonance is described by a few parameters (resonance energy, neutron widths, etc.) which are used as variables in a mathematical representation of the resonance shape. NPTXS uses these mathematical functions to reconstruct each of the resonances, giving the equivalent cross section at a number of energy points which span the resolved range. In the unresolved range, only average resonance quantities are given because the resonances are too closely spaced to be distinguished. The distribution of the average quantities can be integrated (flux-averaged) to yield average cross sections at energy points which span the unresolved range. The point cross sections for the resonance nuclides are used by NEWXLACS[1] (see Sect. 2.3) to calculate the scattering matrix.

The background cross sections from file 3 of ENDF/B are added to those calculated in the resonance regions and output to a file which is similar to an ENDF/B file 3. Only the total, elastic, fission, and capture cross sections are put on this file, and the data are Doppler-broadened to a specified temperature (1000°K in our case). The data in

the unresolved resonance range are evaluated at an input value of σ_0 that is indicative of the system in which these cross sections will be used. (σ_0 is the total cross section of the surrounding medium.)

2.3 AMPX Master Interface Preparation by NEWXLACS

The purpose of NEWXLACS[1] is to create an AMPX master interface (i.e., a master cross section library) which contains all of the s-wave resonance parameters, the flux-averaged group cross sections, and the scattering matrix required in subsequent steps. NEWXLACS requires as input the ENDF/B-format tape for the desired nuclide and, if it is a resonance nuclide, the NPTXS-created file of point cross sections.

If the nuclide is not a resonance nuclide, NEWXLACS group-averages the point cross sections from file 3 of the ENDF/B-formatted data with an input weighting function. This weighting function was a Maxwellian thermal spectrum coupled to a 1/E spectrum in the resonance range which was coupled to a fission spectrum in the fast region. These data are Doppler-broadened to a specified temperature. If it is a resonance nuclide, NEWXLACS puts both the resonance parameters and the group background cross sections on the interface so that the NITAWL module can be used to analyze systems with various resonance nuclide concentrations. These group background cross sections are evaluated for the total, elastic, fission, and capture reactions and taken from the file 3 background in ENDF/B and averaged over the weight function mentioned above.

All the other reactions in the ENDF/B file are group-averaged using the weight function and placed on the master interface. This includes the values for total, elastic, fission, and capture that are taken from

the NPTXS-created file. The elastic scattering point data from this latter file are also used in conjunction with the secondary energy/angle distributions in the ENDF/B file to generate group-to-group transfer arrays (i.e., the scattering matrix) for the master interface.

2.4 References for Sect. 2

1. N. M. Greene, J. L. Lucius, L. M. Petrie, W. E. Ford III, J. E. White, and R. Q. Wright, AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B, ORNL/TM-3706 (March 1976).
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3. R. J. Howerton, D. E. MacGregor, S. T. Perkins, and E. F. Plechaty, The LLL Evaluated Nuclear Data Library (ENDL): Evaluation Techniques, Reaction Index, and Descriptions of Individual Evaluations, UCRL-50400, Vol. 15, Part A (September 1975).
4. R. W. Benjamin, F. J. McCrosson, and P. L. Roggenkamp, Conversion of ^{238}Pu and ^{252}Cf Production Chain Cross Section Data to ENDF/B-IV Format, EPRI-NP-161 (December 1975).
5. R. W. Benjamin, Savannah River Laboratory, personal communication to A. G. Croff, Oak Ridge National Laboratory, January 1977.

3. MULTIGROUP DEPLETION CALCULATIONS

The 84-energy-group AMPX master cross section library described in Sect. 2 of this report contains cross section information of a general nature for solving U-Pu 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 ORIGEN.

A schematic information flow diagram for the processing of the master cross section library into the problem-dependent, one-group libraries is given in Fig. 3.1. In this diagram, the principal computer codes used in processing the cross sections are contained in rectangular figures. Descriptions of the cross section parameters which comprise the input to and output from each computer code are contained in the curved figures. As is evident from Fig. 3.1, there are three principal cross section processing codes: NITAWL,[1] XSDRNPM,[1] and CITATION.[2] A general description of these three computer codes is given in Sect. 3.1.

There are four principal steps involved in processing the master cross section library into ORIGEN libraries, as depicted in Fig. 3.1. The first of these is to perform "pin-cell" neutron energy spectrum calculations at three (for a PWR) or four (for a BWR) different burnups using NITAWL and XSDRNPM which are used to account for self-shielding effects for the nuclides in Table 2.2. The second processing step is to perform "assembly-cell" spectrum calculations using these same computer codes to account for the fact that the fuel elements at the center of a fuel assembly see a significantly different neutron spectrum than those on the periphery of the assembly because of the varying characteristics

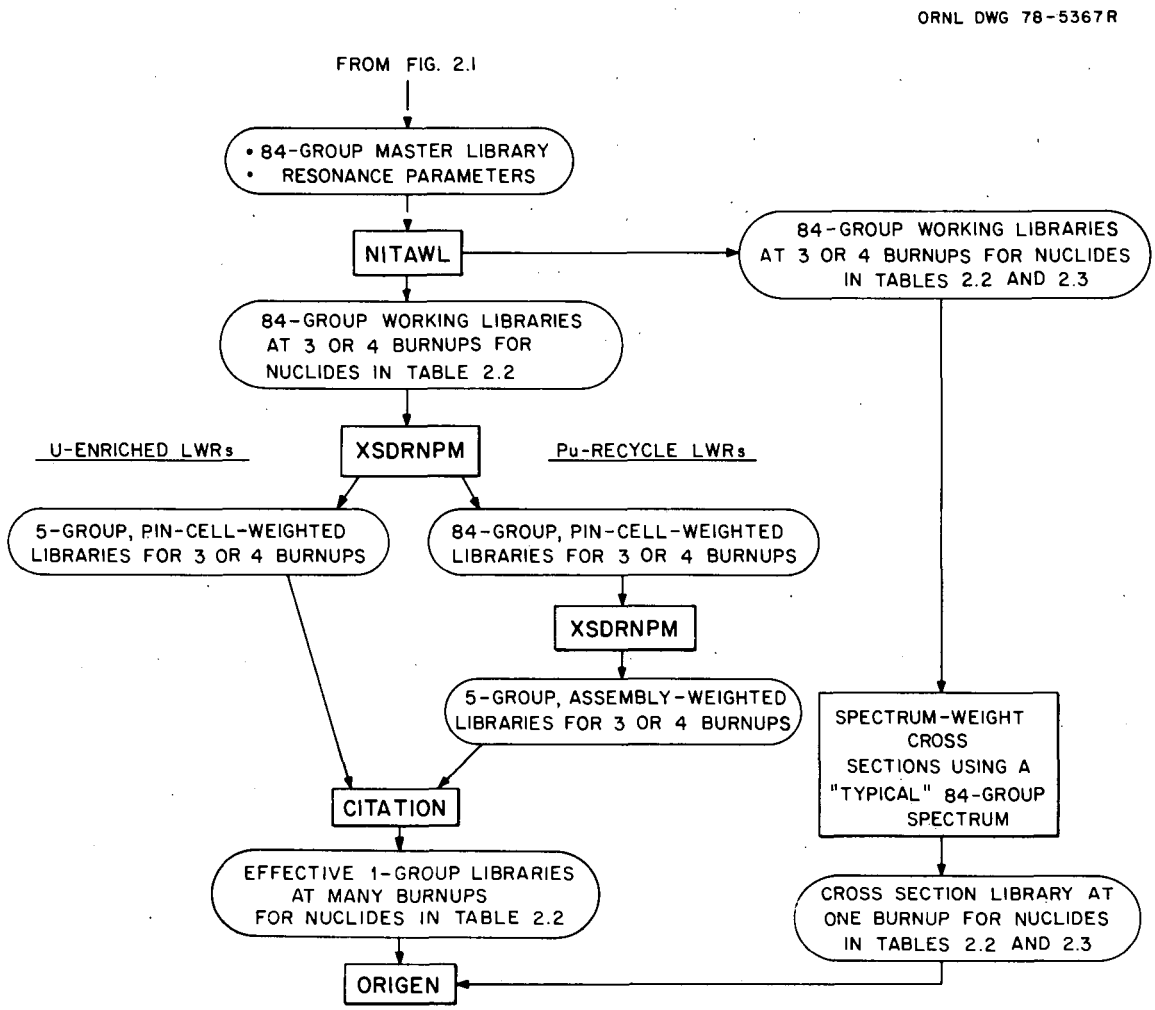


Fig. 3.1. Procedure for processing AMPX master interface cross sections into ORIGEN cross sections.

of the surrounding assemblies. This step is only performed for LWRs containing both plutonium- and uranium-enriched fuel. The third step is to perform fuel-depletion calculations with CITATION using few-group, self-shielded cross sections from XSDRNPM. The principal result of the CITATION calculation is one-group, burnup-dependent cross sections for a few of the most neutronically important actinides. The final step is to use a single neutron energy spectrum that is "typical" of each fuel type to collapse the multigroup cross sections for the nuclides in Tables 2.2 and 2.3 to one-group cross sections for ORIGEN. A summary description of this four-step processing sequence is given in Table 3.1. A detailed description of each of the four steps is given in Sects. 3.2 to 3.4.

A final, somewhat separate calculation that was performed involved using the XSDRNPM computer code to calculate the axial neutron flux shape and energy spectrum for uranium-enriched BWR and PWR fuel assemblies. These calculations, which are described in Sect. 3.5, result in a prediction of the activation of fuel-assembly end pieces and fuel-element plenum springs.

3.1 General Description of Cross Section Processing Codes

3.1.1 NITAWL

The NITAWL[1] 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

Table 3.1. Description of the processing of the master cross section library into ORIGEN cross sections for LWRs

Computer code	Input	Function	Output	Applicability of step
NITAWL	84-group master library Resonance parameters Doppler temperature Estimated actinide composition at three or four burnups	Calculates resonance self-shielding and Doppler-broadening effects using Nordheim integral treatment for a pin cell	84-group working libraries at three or four burnups	All U-Pu LWRs
XSDRNPM	84-group working libraries at three or four burnups	Calculates spatial and energy self-shielding effects and neutron energy spectrum using a 1-D discrete ordinates code for a pin cell	Five-group, pin-cell-weighted libraries at three or four burnups	Uranium-enriched LWRs only
XSDRNPM	84-group working libraries at three or four burnups	Calculates spatial self-shielding effects and neutron energy spectrum using a 1-D discrete ordinates code for a pin cell	84-group, pin-cell-weighted libraries at three or four burnups	Plutonium-recycle LWRs only
XSDRNPM	84-group, pin-cell-weighted libraries at three or four burnups	Calculates spatial self-shielding effects and neutron energy spectrum using a 1-D discrete ordinates code for an assembly	Five-group, assembly-weighted libraries at three or four burnups	Plutonium-recycle LWRs only
CITATION	Five-group, pin-cell- or assembly-weighted libraries at three or four burnups	Five-group diffusion-theory depletion and fuel management	One-group effective libraries at many burnups	All U-Pu LWRs
ORIGEN	One-group, effective libraries at many burnups	Zero-dimensional, one-group depletion calculation for 1300 isotopes	-	All U-Pu LWRs

resonances and (2) the fact that the effect of large resonances is diminished because there are relatively few neutrons at the resonance energy. The latter effect results from the resonance itself depleting 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 library (working library) in which the resonance parameters have been incorporated into the group-averaged cross sections.

3.1.2 XSDRNPM

The XSDRNPM[1] 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 does a one-dimensional, static, S_8P_3, discrete-ordinates flux calculation and then uses this flux to weight the input cross section library (the working library) in space, energy, or both. This results in a cross section library which accounts for the fact that the neutron energy spectrum, and thus the effective nuclide cross sections, vary 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, corresponding to the nuclide concentrations.

3.1.3 CITATION

The CITATION[2] computer code is used to perform the reactor fuel-depletion and fuel-management calculations using multigroup diffusion theory. The code can be used with a variety of geometries in one, two, or three dimensions. Output from CITATION includes the discharge composition of the fuel specific power and neutron flux in a particular unit of fuel as a function of burnup. Minor modifications to CITATION also allow effective cross sections for each nuclide as a function of burnup to be output for subsequent use in ORIGEN.

3.2 Pin-Cell Calculations

3.2.1 General approach

The so-called pin-cell calculation involves two distinct substeps. The first substep is to process the 84-energy-group master cross section library and its associated resonance parameters into an 84-group working library using the NITAWL computer code. This step accounts for Doppler broadening of the resonances and energy self-shielding. The second substep is to process the working library into a weighted library using the XSDRNPM computer code. This step accounts for the spatial self-shielding effects in the fuel element. The spatial weighting is over the entire pin cell because the cross sections will subsequently be used with nuclide densities that have been averaged over the entire pin cell. The results of this XSDRNPM pin-cell calculation are in either five energy groups, if they are to be used directly in CITATION calculations (see Sect. 3.4), or 84 energy groups, if they are to be used in subsequent XSDRNPM assembly-cell calculations (see Sect. 3.3). Pin-cell calculations

were made at three different fuel compositions for the PWRs, corresponding to burnups of 5, 16, and 27 GWd/MTIHM and at four different fuel compositions for the BWRs, corresponding to burnups of 3, 10, 17, and 25 GWd/MTIHM.

3.2.2 Pin-cell description

The pin cells for all four reactor types consisted of five concentric zones: fuel, gap, clad, moderator, and "extra." The extra zone consists of moderator, structural materials, and soluble neutron poisons not specifically associated with a particular fuel element. Examples of the types of material included in the extra zone include grid spacers, the material in water holes, fuel channels, and the moderator between fuel assemblies. A "white," or isotopic-reflecting, boundary condition was imposed on the outer boundary of the pin cells to simulate the presence of the cell in an infinite medium of other pin cells of its own type. The pin-cell dimensions for the PWR and BWR are given in Tables 3.2 and 3.3 respectively.

The nuclide number densities that are used as input to the NITAWL and XSDRNPM codes were based on depletion data[3] given as a function of burnup for uranium- and plutonium-enriched PWRs and BWRs. The pin-cell calculations were made using compositions which corresponded to burnups of 5, 16, and 27 GWd/MTIHM for the PWR and 3, 10, 17, and 25 GWd/MTIHM for the BWR. These burnups were selected because they approximate the average, middle-of-cycle burnups for a PWR on a three-cycle refueling scheme with a discharge burnup of 33 GWd/MTIHM and a BWR on a four-cycle refueling scheme with a discharge burnup of 27.5 GWd/MTIHM. These burnup-dependent cross section libraries were developed to account for the variation in the reactor neutron energy spectrum, and thus the effective nuclide cross sections, during the irradiation of the fuel. The calculation of the nuclide densities assumed a fuel density of 95% of the theoretical

Table 3.2. Pin-cell dimensions for the PWR[a]

Region	Outer radius (cm)	Volume fraction
Fuel	0.41212	0.30659
Gap	0.41738	0.0078762
Clad	0.47572	0.094055
Moderator	0.71187	0.50626
Extra	0.74429	0.085219

[a]Data obtained from Westinghouse Nuclear Energy Systems, RESAR-3, Reference Safety Analysis Report, DOCKET STN 50-480 (1972).

Table 3.3. Pin-cell dimensions for the BWR[a]

Region	Outer radius (cm)	Volume fraction
Fuel	0.53165	0.24015
Gap	0.53975	0.0073744
Clad	0.62705	0.085799
Moderator	0.91845	0.38272
Extra	1.08481	0.28321

[a]Data obtained from General Electric Standard Safety Analysis Report, BWR/6, DOCKET STN 50-447 (1973).

and a stoichiometric amount of oxygen. For the Doppler-broadening of the resonances calculated by NITAWL, a Doppler temperature of 1000°K was assumed.

The concentration of the soluble boron poison in the moderator and extra zones of the PWR pin-cells was assumed to be a constant 550 ppm for all burnups. In reality, the boron concentration varies from ∿1100 ppm to ∿0 ppm during each cycle. The choice of an average boron concentration for all burnups was made because later CITATION and ORIGEN calculations would not be able to account for this continuous variation.

The saturated steam fraction of the water in the moderator zone for the BWR calculations was taken to be 0.392 at a pressure of 1050 psia, a value significantly lower than the volume-averaged steam fraction in the moderator zone of a modern BWR (∿0.50).[4] Similarly, the saturated steam fraction of the water in the extra zone was assumed to be 0.213 as compared to an estimated volume-averaged value of 0.353.[5] These lower steam fractions (i.e., higher water densities) were used to account for the fact that the subsequent fuel-depletion calculations were designed to simulate the "average" depletion characteristics of the reactor. Since the axial power shape in a BWR is skewed toward the bottom of the reactor because of the higher water density at the bottom. Volume-averaging will not yield the appropriate nuclide densities. This is because the nuclide densities in low-power regions are weighted equally with those in high-power regions. Thus, a more appropriate weighting method would be to power-average the nuclide densities. Although this weighting method is made difficult because of the presence of control rods in the core, the actual power-averaged saturated steam fractions of the water in the moderator and extra zones together were estimated to be 0.365.

This value is comparable to the value of 0.316, which was adjudged to give the "best" depletion results in this study, and to a volume-averaged saturated steam fraction of 0.429. Power-averaged moderator densities are not necessary in a PWR because the axial power shape is nearly symmetrical with respect to the reactor mid-plane, thus making power-averaging and volume-averaging nearly equivalent. PWR nuclide densities in the moderator and extra zones were based on the properties of water at a temperature of 590°F and a pressure of 2250 psia.

3.2.3 Results of pin-cell calculations

The results of the pin-cell calculations are as follows:

1. 84- or 5-energy-group, pin-cell-averaged cross sections for the actinides, moderator, principal structural materials, and poisons at multiple burnups;
2. 84-energy-group, pin-cell-averaged neutron energy spectra, at multiple burnups;
3. one-group, pin-cell-averaged fission and activation product cross sections; and
4. pin-cell-averaged parameters to enable THERM, RES, and FAST values to be calculated.

The 84-energy-group cross sections are used in subsequent XSDRNPM assembly-cell calculations for the plutonium-recycle LWRs and will be discussed in Sect. 3.3. The five-energy-group cross sections are for the uranium-enriched LWRs and are used in subsequent CITATION depletion calculations described in Sect. 3.4. The nuclides in these cross section libraries are listed in Table 2.2.

XSDRNPM also calculates a cell-averaged, 84-energy-group neutron spectrum for each of the pin-cells at each burnup. "Typical" spectra for the uranium-enriched PWR and BWR pin cells are given in Appendix A in both tabular and graphical form. The spectra for the plutonium-recycle LWRs result from the assembly-cell calculations discussed in Sect. 3.3.

After the uranium-enriched LWR depletion calculations have been performed with CITATION, one of the pin cells is selected for each reactor as being representative of that reactor, based on the similarity of its neutron spectrum to that calculated by CITATION for the entire reactor. The neutron spectrum from this pin cell is used to collapse the multigroup cross sections for all nuclides listed in Tables 2.2 and 2.3 to one-group, pin-cell-averaged cross sections that are incorporated directly into the ORIGEN cross section library. These cross sections are listed in Appendix B. The collapse of these cross sections for the plutonium-recycle reactors is discussed in Sect. 3.3.

Finally, the pin cells selected for collapsing the cross sections to one group are also used to generate two sets of two neutron-energy-group cross sections and neutron spectra. These cross sections and neutron spectra are used to calculate values of THERM, RES, and FAST for the reactor being considered. The details of the methods used to calculate these values will be given in a separate publication. The results of the calculations are presented in Sect. 4.1.

3.3 Assembly-Cell Calculations

3.3.1 General approach

The assembly-cell calculations use the XSDRNPM code and the 84-energy-group, pin-cell-weighted cross sections previously produced by XSDRNPM (see Sect. 3.2.3) to determine effective cross sections and neutron energy spectra for uranium- and plutonium-enriched assemblies in plutonium-recycle LWRs. This procedure was used to account for the fact that the neutron energy spectrum on the periphery of a plutonium-enriched fuel assembly that is next to a uranium-enriched fuel assembly is different from the neutron energy spectrum in the center of either a plutonium- or uranium-enriched assembly. The assembly-cell calculations were not used for the uranium-enriched LWRs because of the relative similarity of adjacent fuel assemblies. The output cross sections from the assembly calculations are cast into a five-energy-group format that is suitable for input to CITATION after reformatting. A five-energy-group cross section set is produced for each of the burnups used in the pin-cell calculations (see Sect. 3.2.1) for both the uranium- and plutonium-enriched assemblies.

3.3.2 Assembly-cell description

The assembly cells for all of the assembly calculations consisted of two concentric zones: an inner, homogenized mixture representing the fuel assembly of interest and an outer, homogenized mixture representing the composition of a mixture of uranium- and plutonium-enriched fuel assemblies. The inner zone is the size of one cylindricized fuel assembly, whereas the outer zone is simply made large enough to establish

a neutron energy spectrum. The resulting radii of these two zones were 12.08 and 22.08 cm, respectively, for the PWR and 7.564 and 17.564 cm, respectively, for the BWR. An isotropic-reflecting boundary condition was used at the outer boundary. The composition of the inner zone was equivalent to that obtained by homogenizing (volume-weighting) the pin-cell compositions described in Sect. 3.2.2. The composition of the outer zone corresponded to that of a homogenized, middle-of-cycle pin cell which was 30% plutonium-enriched and 70% uranium-enriched.

3.3.3 Results of assembly-cell calculations

The results of the assembly-cell calculations for the plutonium-recycle LWRs are as follows:

1. five-energy-group, inner-zone-averaged cross sections for the actinides, principal structural materials, and poisons at multiple burnups;
2. 84-energy-group, inner-zone-averaged neutron spectra at multiple burnups;
3. one-group, inner-zone-averaged fission product, activation product, and actinide cross sections at one burnup; and
4. inner-zone-averaged parameters to enable THERM, RES, and FAST values to be calculated.

The five-energy-group, assembly-cell-averaged cross sections (i.e., weighted libraries) are used as input to the CITATION computer code. There are six weighted libraries for the plutonium-recycle PWR — three for the uranium-enriched assembly, corresponding to burnups of 5, 16, and 27 GWd/MTIHM, and three for the plutonium-enriched assembly, corresponding to the same burnups. There are eight weighted libraries

for the plutonium-recycle BWR — four each for the uranium- and plutonium-enriched assemblies, corresponding to burnups of 3, 10, 17, and 25 GWd/MTIHM. The nuclides in these weighted libraries are listed in Table 2.2.

Typical neutron-energy spectra calculated by the XSDRNPM code for uranium- and plutonium-enriched fuel assemblies in both the PWR and BWR are listed in Appendix A. These spectra correspond to a single burnup for each reactor that was selected because of the similarity of its neutron spectrum to that calculated by CITATION during the depletion calculations described in Sect. 3.4.

These same neutron spectra are used to collapse cross sections for the actinides listed in Table 2.2 and the nuclides listed in Table 2.3 to one-group, assembly-cell-averaged cross sections that are incorporated directly into the ORIGEN cross section library. These cross sections are listed in Appendix B.

Finally, the same assembly cells selected for collapsing the multi-group cross sections to one group are rerun to generate two sets of two neutron-energy-group cross sections and neutron spectra that are used to calculate values of THERM, RES, and FAST for each fuel type being considered. The details of the methods used to calculate these values will be given in a separate publication. The results of the calculations are presented in Sect. 4.1.

3.4 Fuel-Depletion Calculations

3.4.1 General approach

The purpose of performing a multigroup fuel-depletion calculation with the CITATION computer code was to model the depletion characteristics of an entire batch of reactor fuel at steady state. Hence for the purposes of updating the ORIGEN computer code, spatial details of the depletion are not important. Therefore, the basic approach taken was to use the simplest neutronic and fuel management model possible while still obtaining adequate depletion results. As is evident from the preceding sections, five-energy-group cross section sets were employed. The energy boundaries of this five-group structure are given in Table 3.4. This group structure consists of one thermal group and four fast groups spanning the energy range of 1.0×10^{-5} to 10.0 MeV. Early in the planning stages of the ORIGEN update effort, discussions[6] indicated that the neutronic interaction of the fuel assemblies with differing burnups would have an important effect on the results of the depletion calculation. Therefore, 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, respectively, and two-dimensional, six- and eight-region models for the plutonium-recycle PWR and BWR, respectively.

Table 3.4. CITATION cross section energy-group structure

Group number	Upper boundary (eV)	Lower boundary (eV)
1	1.00000×10^{7}	5.53084×10^{3}
2	5.53084×10^{3}	2.90232×10^{1}
3	2.90232×10^{1}	1.85539×10^{0}
4	1.85539×10^{0}	6.32500×10^{-1}
5	6.32500×10^{-1}	1.00000×10^{-5}

3.4.2 CITATION depletion model description

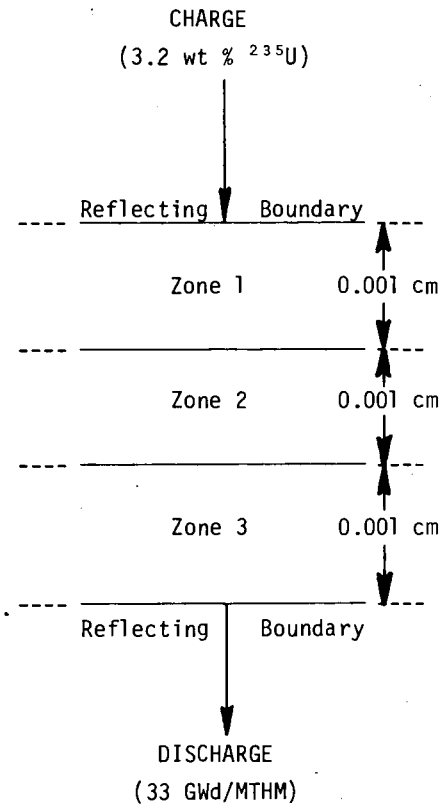
3.4.2.1 Uranium-enriched LWRs. The relative neutronic homogeneity of the uranium-erniched LWRs allowed the use of spatial models that were effectively zero-dimensional. More specifically, the spatial model used was a one-dimensional, three-region (PWR) or four-region (BWR) slab with reflecting boundary 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 three zones. In the PWR model, the first, second, and third regions were assigned cross sections corresponding to fuel burnups of 5, 16, and 27 GWd/MTIHM, respectively. Similarly, the four regions in the BWR model are 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 which correspond 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 then moved to the region that has been assigned cross sections which correspond 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. Schematic depictions of the steady-state fuel management models for the uranium-enriched PWR and BWR are given in Figs. 3.2 and 3.3, respectively.

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 were altered from the assignments described previously in that all regions were assigned cross sections which corresponded to the lowest burnup for the reactor.

The CITATION depletion calculations consider all nuclides listed in Table 2.2 explicitly. All nuclide densities input to CITATION are pin-cell or assembly-cell-averaged quantities.

The power-level input to CITATION is equivalent ot 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 models varies throughout its life, depending on its enrichment and the enrichment of the other fuel in the model at that time.



PWR-U Fuel Management Description

Zone	Fuel burnup at which cross sections were determined (GWd/MTHM)	Status of fuel at beginning of cycle	Status of fuel at end of cycle	Disposition of zone at end of cycle
1	5	Fresh	Irradiated one cycle	Move to Zone 2
2	16	Irradiated one cycle	Irradiated two cycles	Move to Zone 3
3	27	Irradiated two cycles	Irradiated three cycles	Discharge

NOTES: 1. Cycle length is 293.3 full-power days.

2. One-dimensional slab geometry used; the small thicknesses of the slabs (= 0.001 cm) effectively make this a zero-dimensional calculation.

Fig. 3.2. CITATION fuel management model for the uranium-enriched PWR (PWR-U) after the first cycle.

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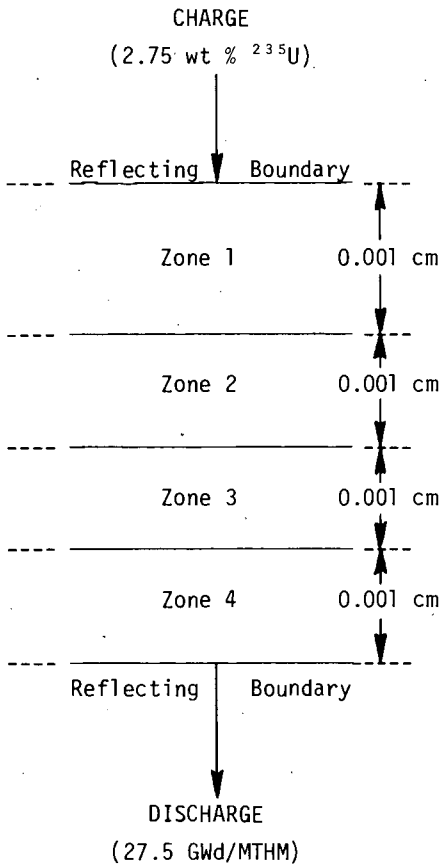

BWR-U Fuel Management Description

Zone	Fuel burnup at which cross sections were determined (GWd/MTHM)	Status of fuel at beginning of cycle	Status of fuel at end of cycle	Disposition of zone at end of cycle
1	3	Fresh	Irradiated one cycle	Move to Zone 2
2	10	Irradiated one cycle	Irradiated two cycles	Move to Zone 3
3	17	Irradiated two cycles	Irradiated three cycles	Move to Zone 4
4	25	Irradiated three cycles	Irradiated four cycles	Discharge

NOTES: 1. Cycle length is 265.4 full-power days.

2. One-dimensional slab geometry used; the small thicknesses of the slabs (= 0.001 cm) effectively make this a zero-dimensional calculation.

Fig. 3.3. CITATION fuel management model for the uranium-enriched BWR (BWR-U) after the first cycle.

3.4.2.2 Plutonium-recycle LWRs. The significant neutronic differences between ^{235}U and the fissile plutonium isotopes (^{239}Pu and ^{241}Pu) indicate that the cross sections used for uranium-enriched fuel must be different from those of plutonium-enriched fuel if the depletion calculations are to be accurate. Furthermore, the fact that LWR fuel assemblies are large when compared to the neutron mean free path in the reactor means that significant changes in the neutron spectrum can be expected between the periphery and the center of a plutonium-enriched assembly located next to a uranium-enriched assembly. These considerations require that that CITATION model used for depleting fuel in a plutonium-recycle LWR be greater than zero-dimensional to account for the spatial variation in neutron spectrum and flux. The model used was a two-dimensional, six-region (PWR) or eight-region (BWR) slab with a periodic boundary condition on the right and left and a reflecting boundary condition on the top and bottom. Half of the regions in each reactor were assumed to be plutonium-enriched, and half were assumed to be uranium-enriched. The fuel management and cross section assignment models used are analogous to those described in Sect. 3.4.2.1 and are depicted in Figs. 3.4 and 3.5. As noted above and described in Sect. 3.2 and 3.3, a different cross section set is used for each region of the model. The fuel management scheme is countercurrent in order to reduce power-peaking during the depletion calculation. The volume of each pair of plutonium- plus uranium-enriched regions was considered to be identical to that of a fuel assembly for the reactor being considered. The volume of the uranium-enriched regions was assumed to be larger than the volume of the plutonium-enriched regions by the ratio of the amount of uranium-enriched fuel to plutonium-enriched fuel

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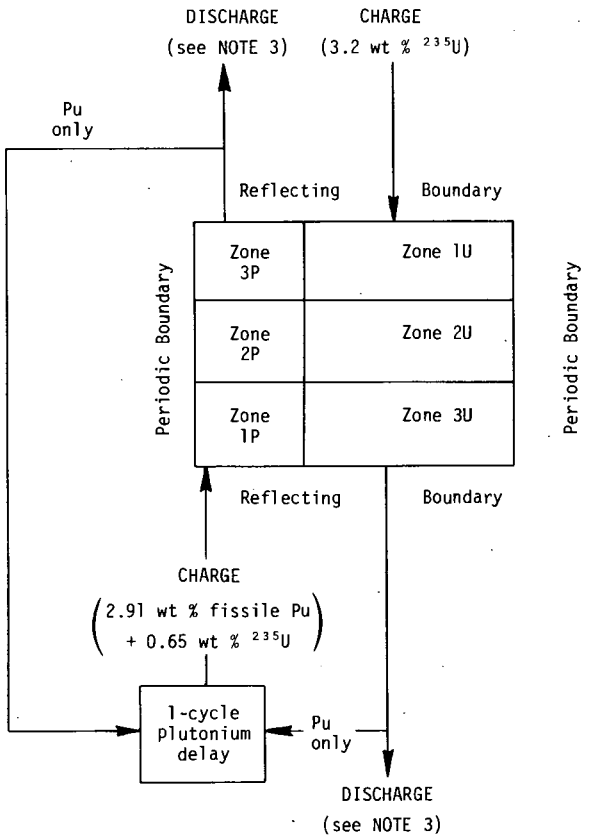
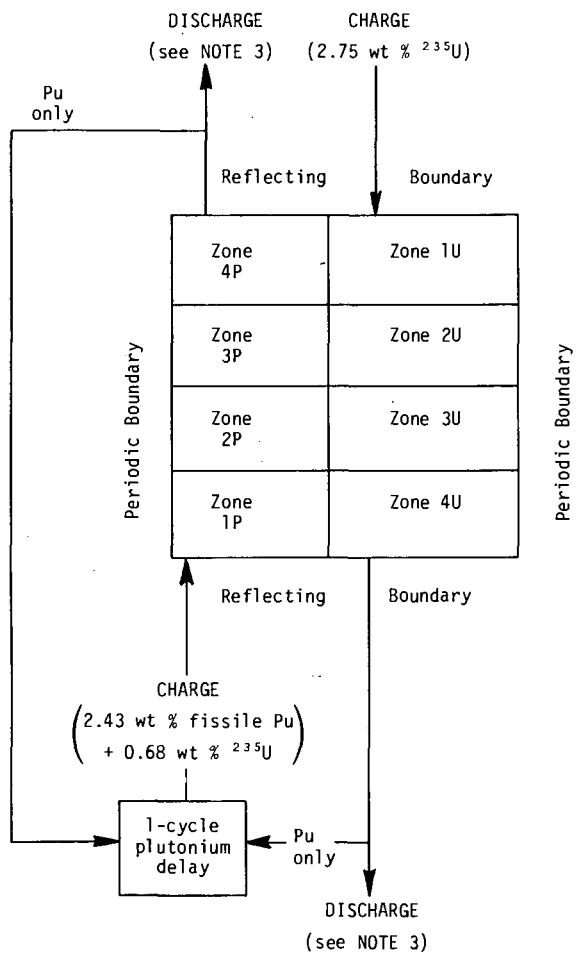


PWR-Pu Fuel Management Description

Zone	Fuel burnup at which cross sections were determined (GWd/MTHM)	Status of fuel at beginning of cycle	Status of fuel at end of cycle	Disposition of zone at end of cycle
1U (1P)	5	Fresh	Irradiated one cycle	Move to Zone 2U (2P)
2U (2P)	16	Irradiated one cycle	Irradiated two cycles	Move to Zone 3U (3P)
3U (3P)	27	Irradiated two cycles	Irradiated three cycles	Discharge; Pu recycled

NOTES:
1. Cycle length is 293.3 full-power days.
2. Two-dimensional X-Y geometry used; "U" zones are 15.12 x 21.0 cm, and "P" zones are 5.88 x 21.0 cm.
3. Weighted-average burnup of "U" and "P" discharged fuel is 33 GWd/MTHM.

Fig. 3.4. CITATION fuel management model for the plutonium-recycle PWR (PWR-Pu) after the first cycle.



BWR-Pu Fuel Management Description

Zone	Fuel burnup at which cross sections were determined (GWd/MTHM)	Status of fuel at beginning of cycle	Status of fuel at end of cycle	Disposition of zone at end of cycle
1U (1P)	3	Fresh	Irradiated one cycle	Move to Zone 2U (2P)
2U (2P)	10	Irradiated one cycle	Irradiated two cycles	Move to Zone 3U (3P)
3U (3P)	17	Irradiated two cycles	Irradiated three cycles	Move to Zone 4U (4P)
4U (4P)	25	Irradiated three cycles	Irradiated four cycles	Discharge; Pu recycled

NOTES: 1. Cycle length is 265.4 full-power days.

2. Two-dimensional X-Y geometry used; "U" zones are 9.12 x 13.41 cm, and "P" zones are 4.29 x 13.41 cm.

3. Weighted-average burnup of "U" and "P" discharged fuel is 27.5 GWd/MTHM.

Fig. 3.5. CITATION fuel management model for the plutonium-recycle BWR (BWR-Pu) after the first cycle.

in the reactor after the second-plutonium recycle. This ratio is 72/28 for the PWR and 68/32 for the BWR. The entire reactor is assumed to be uranium-enriched initially with the same initial compositions as those used for the uranium-enriched LWRs (see Sect. 3.4.2.1). The plutonium from both the uranium- and plutonium-enriched regions was assumed to be recycled to the plutonium-enriched regions after a one-cycle, out-of-reactor delay. The CITATION depletion calculation was terminated after the second complete plutonium recycle, which corresponded to 11 cycles for the PWR and 14 cycles for the BWR. As with the uranium-enriched reactors, the reactor-average specific powers were assumed to be 37.5 MW/MTIHM for the PWR and 25.9 MW/MTIHM for the BWR.

3.4.3 Results of CITATION depletion calculations

The results of the CITATION depletion calculations for uranium- and plutonium-recycle LWRs are as follows:

1. one-energy-group, effective cross sections for the nuclides listed in Table 2.2, 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 ORIGEN computer code as a function of burnup as described in Sect. 4.1. The PWR-U and PWR-PuPu cross sections incorporated into ORIGEN in this manner are listed in Appendix C.

The discharged composition of the fuels is used as a basis of comparison to ensure that ORIGEN depletion calculations are correct. This consideration will be discussed further in Sect. 4.

The five-energy-group neutron spectra are used as a basis for selecting one of the cell calculations described in Sects. 3.2 and 3.3 to provide collapsed, one-energy-group cross sections for the fission products, activation products, and actinides for incorporation into the ORIGEN cross section library.

It should be noted that two sets of the results listed above are produced for each plutonium-recycle LWR (one for the uranium-enriched fuel and one for the plutonium-enriched fuel).

3.5 Axial Neutron Energy Spectrum Calculations

3.5.1 General approach

Axial spectrum calculations were performed for uranium-enriched PWR and BWR fuel assemblies with the XSDRNPM computer code to provide information concerning the neutron flux level and spectrum outside the active fuel region of the core. These calculations are useful in determining the degree to which the fuel plenum springs and fuel-assembly end pieces become activated.

3.5.2 Axial assembly model

The axial assembly model for the PWR consisted of 12 one-dimensional zones, each with a nuclide composition corresponding to one vertical segment of the fuel assembly. The fuel composition corresponded to that at middle-of-cycle and was assumed to be uniform throughout the active fuel region. The moderator density was assumed to be 0.7283 g/cm^3 throughout the entire assembly and to contain 550 ppm of boron. The dimensions and material makeup of each of the 12 PWR axial assembly zones are given in Table 3.5.

Table 3.5. Dimensions and material makeup of the PWR axial assembly model

Zone number	Description	Zone length (cm)	Material makeup (vol %)
1	Lower water reflector	25.4	Water[a] (100)
2	Lower core plate	5.08	Water (25); SS-304[b] (75)
3	Lower end piece	6.86	Water (77.5); SS-304 (22.5)
4	Space	2.54	Water (99.2); Zr-2[c] (0.8)
5	Lower end plug	1.65	Water (59.6); Zr-2 (40.4)
6	Fuel	365.8	Water (64.3); fuel[d] + Zr-2 (35.7)
7	Gas plenum and spring	16.0	Water (64.3); Zr-2 (9.2); SS-302[e] (3.3); void (23.3)
8	Upper end plug	1.65	Water (59.6); Zr-2 (40.4)
9	Space	2.54	Water (99.2); Zr-2 (0.8)
10	Upper end piece	8.64	Water (78.7); SS-304 (21.3)
11	Upper core plate	5.08	Water (25); SS-304 (75)
12	Upper water reflector	25.4	Water (100)

[a]Density = 0.7283 g/cm^3; contains 550 ppm of boron.

[b]Stainless steel 304.

[c]Zircaloy-2.

[d]Reactor-average, middle-of-cycle fuel composition from CITATION.

[e]Stainless steel 302.

The axial assembly model for the BWR consisted of 16 one-dimensional zones. The fuel composition corresponded to that of middle-of-cycle and was assumed to be uniform throughout the active fuel region. The moderator density was assumed to vary in the active fuel region, corresponding to the axial change in void fraction as given in ref. 4. The dimensions and material makeup of each of the 16 BWR axial assembly zones are given in Table 3.6.

3.5.3 Results of the axial spectrum calculations

The desired result of the axial spectrum calculations is the ratio of the activation of selected structural materials in the end pieces and plenum springs to the activation of the same materials in the active fuel zone (i.e., an activation ratio). When this ratio is multiplied by the average neutron flux in the active fuel zone, the resulting value is the correct flux to use when irradiating the structural materials using the one-group cross sections obtained from XSDRNPM (Sects. 3.2.3 and 3.3.3). The activation ratios resulting from the uranium-enriched PWR and BWR spectrum calculations are summarized in Table 3.7. Activation ratios for the fuels in plutonium-recycle PWRs and BWRs were calculated by multiplying the corresponding uranium-enriched reactor activation ratio by the ratio of FAST for the new fuel type to FAST for the uranium-enriched reactor (see Sect. 4.1 for a discussion of FAST). This procedure was used because it is the fast (high-energy) neutrons that have the ability to reach the structural materials outside the core before being captured. The large difference in the activation ratios of the PWRs and the BWRs is a result of the soluble boron neutron poison assumed to be present in the PWR moderator, which effectively reduces the neutron flux.

Table 3.6. Dimensions and material makeup of the BWR axial assembly model

Zone No.	Description	Zone length (cm)	Material makeup (vol %)
1	Lower water reflector	25.4	Water, D = 0.86[a] (100)
2	Lower core plate	5.08	Water, D = 0.86 (25); SS-304[b] (75)
3	Orifice	12.7	Water, D = 0.86 (75); SS-304 (25)
4	Lower tie plate	11.8	Water, D = 0.86 (81.9); SS-304 (18.1)
5	Lower end plug	1.6	Water, D = 0.86 (63.5); Zr-2[c] (36.5)
6-11	Active fuel zones	Zone 8 = 71.1 Others = 61.0	All zones: fuel[d] + Zr-2 (40.4) Zone 6: water, D = 0.72 (59.6) Zone 7: water, D = 0.57 (59.6) Zone 8: water, D = 0.41 (59.6) Zone 9: water, D = 0.34 (59.6) Zone 10: water, D = 0.30 (59.6) Zone 11: water, D = 0.28 (59.6)
12	Gas plenum and spring	30.5	Water, D = 0.63 (59.7) Zr-2 (7.8) SS-302[e] (2.0)
13	Upper end plug and space	2.1	Water, D = 0.63 (59.7) Zr-2 (40.5) IX-750[f] (15.4)
14	Upper tie plate	8.5	Water, D = 0.63 (81.0) SS-304 (19.0)
15	Handle	8.4	Water, D = 0.63 (93.6) SS-304 (6.4)
16	Upper water reflector	25.4	Water, D = 0.63 (100)

[a]Density of moderator in g/cm^3.

[b]Stainless steel alloy 304.

[c]Zircaloy-2.

[d]Reactor-average, middle-of-cycle fuel composition from CITATION.

[e]Stainless steel alloy 302.

[f]Inconel alloy X-750.

Table 3.7. Summary of the results of the axial spectrum calculations

	PWR-U	PWR-PuU[a]	PWR-PuPu[a]	BWR-U	BWR-PuU[b]	BWR-PuPu[b]
Activation ratios[c]						
End pieces[d]	0.011	0.011	0.013	0.130	0.131	0.142
Plenum springs	0.042	0.043	0.048	0.500	0.505	0.545
Fraction of amount actually present to be used in ORIGEN						
Fe, Ni, Cr	1.0	1.0	1.0	1.0	1.0	1.0
Mn	0.80	0.80	0.80	0.80	0.80	0.80
Co	0.67	0.67	0.67	0.67	0.67	0.67
Zr	0.40	0.40	0.40	0.32	0.32	0.32

[a]Activation ratios based on PWR-U axial spectrum calculation and ratio of FAST parameter (see Sect. 4.1); the fraction of the elements to be used in ORIGEN is assumed to be constant for all PWRs.

[b]Activation ratio based on BWR-U axial spectrum calculation and ratio of FAST parameter (see Sect. 4.1); the fraction of the elements to be used in ORIGEN is assumed to be constant for all BWRs.

[c]Ratio of element activation in the specified material and location to the element activation for the same material in the active fuel zone.

[d]Average of upper and lower end pieces.

A second important result of the axial spectrum calculations is that, because of the changes in the neutron spectrum outside the active fuel zone, the effective cross sections of the structural materials change, to varying degrees, from their values in the active fuel zone. The effect of these cross section changes can be accounted for in ORIGEN by altering the amount of the element assumed to be initially present to a value other than the amount that would actually be present. Fortuitously, the effective crsos sections of the three major constituents of the stainless steel end pieces and plenum springs — iron, nickel, and chromium — all change to the same extent. Thus, these elements can be assumed to be present initially in their actual quantities. This fact is reflected in the portion of Table 3.7 labelled "fraction of amount actually present to be used in ORIGEN." However, the effective cross sections of three other constituents of the structural materials (manganese, cobalt, and zirconium) decrease to a greater extent than the cross sections of iron, nickel, and chromium in the neutron spectrum outside the active fuel region, thus requiring that the amount assumed to be present initially in ORIGEN be less than that actually present if the activation calculation is to be correct. The fraction of these three elements that should be input to ORIGEN to yield the correct activation is also indicated in Table 3.7. This fraction appears to be strongly correlated with the ratio of the thermal cross section to the resonance integral for the element of interest.

3.6 References for Sect. 3

1. N. M. Greene, J. L. Lucius, L. M. Petrie, W. E. Ford III, J. E. White, and R. Q. Wright, AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B, ORNL/TM-3706 (March 1976).

2. T. B. Fowler, D. R. Vondy, and G. W. Cunningham, Nuclear Reactor Core Analysis Code: CITATION, ORNL/TM-2496, Rev. 2 (July 1971).

3. Pacific Northwest Laboratories, Theoretical Analysis of Plutonium Buildup and Uranium Depletion in Pressurized Water and Boiling Water Power Reactors, Report Y49024 to IAEA Division of Nuclear Power Reactors (July 1968).

4. General Electric Standard Safety Analysis Report, BWR/6, DOCKET STN 50-447 (1973).

5. W. B. Arthur, Union Carbide Nuclear Division, personal communication to A. G. Croff, Oak Ridge National Laboratory, January 1977.

6. D. R. Vondy, Oak Ridge National Laboratory, personal communication to A. G. Croff, Oak Ridge National Laboratory, September 1976.

4. DESCRIPTION OF REACTOR MODELS

The results of the reactor physics calculations described in Sect. 3 were specifically developed to provide the neutronic and cross section data required by the ORIGEN computer code. However, the full incorporation of new U-Pu cycle LWR models into ORIGEN requires the results of work not directly involved with the reactor physics calculations. The additional work can be broken down into four major categories: (1) modification of the ORIGEN code to eliminate assumptions that are incompatible with the new cross sections, (2) specification of the initial structural material and fuel compositions, (3) comparison of ORIGEN fuel-depletion calculations with independent calculations to validate the new reactor models, and (4) a summary description of the new reference reactor models.

The required ORIGEN computer code modifications involve (1) allowing the recoverable energy per fission to be dependent on the fissile species involved, (2) redefining the ORIGEN neutron spectrum parameters THERM, RES, and FAST on a total flux basis instead of a thermal flux basis, and (3) allowing the cross sections of the principal actinides (i.e., $^{234-236,238}U$, $^{238-242}Pu$, $^{241,243}Am$, and $^{242,244}Cm$) to be a function of fuel burnup. These modifications are discussed further in Sect. 4.1.

The specification of the input compositions to ORIGEN involves a literature search to determine (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. These compositions are presented and discussed in Sect. 4.2.

A comparison of the spent fuel compositions predicted by the revised reactor models with independent calculations is required since the relative simplicity of the multigroup depletion (i.e., CITATION) models does not necessarily indicate an accurate result even when ORIGEN results agree well with CITATION results. The results of depletion calculations were obtained from the literature, and ORIGEN depletion calculations were performed on a basis consistent with each literature reference. These two results are presented together in a series of tables in Sects. 4.3.1 and 4.3.2. In general, agreement was very good, although significant discrepancies were noted in the case of BWR-PuPu fuel.

A summary description of the results of ORIGEN fuel-depletion calculations for the U-Pu cycle LWRs is given in Sect. 4.3.3.

4.1 Discussion of ORIGEN Modifications

The incorporation of cross sections derived from sophisticated reactor physics codes into ORIGEN necessitates modifications to ORIGEN if its depletion calculations are to agree with the more sophisticated calculations. Although recent ORIGEN modifications (to be described in later publications) have been relatively extensive, only three basic changes are required in the existing version of ORIGEN[1] so that the cross sections from more sophisticated reactor physics codes may be used:

1. The present assumption that the recoverable energy per fission is a constant 200 MeV per fission must be relaxed to account for the variation in this parameter for various actinides.
2. The values of THERM, RES, and FAST, used to calculate effective, one-group cross sections from 2200-m/sec cross

sections, resonance integrals, and threshold cross sections, respectively, must be redefined and recalculated.

3. The assumption that the cross sections of the actinides present in relatively high concentrations in the reactor fuels are constant must be relaxed to account for the significant variation of these cross sections during fuel irradiation.

The conceptual details of these modifications, which have been incorporated into a code designated ORIGEN2, will be described in a later publication. Additionally, since the recoverable-energy-per-fission values for the various actinides are nearly constant for all reactor types, the values used in ORIGEN2 will be discussed generically in this same publication.

The values of THERM, RES, and FAST that have been derived from XSDRNPM[2] neutron-energy-spectrum calculations are given in Table 4.1. These values are not compatible with previous versions of ORIGEN since they are based on total flux instead of thermal flux. A relatively straightforward (but approximate) conversion process yields the THERM, RES, and FAST values given in Table 4.2, which are compatible with previous versions of ORIGEN.

The variable actinide cross sections used in ORIGEN2 for the reactor models result from the CITATION calculations discussed in Sect. 3 and are given in Appendix C for the PWR-U and PWR-PuPu fuels. ORIGEN2 has provisions for calculating and incorporating new values for those cross sections listed in the tables in Appendix C before each irradiation step. The fission product yields are also adjusted for those actinides that have variable fission cross sections and are designated as fission product producers (e.g., ^{235}U, ^{239}Pu, and ^{241}Pu).

Table 4.1. Values of THERM, RES, and FAST to be used with LWR total fluxes[a]

Fuel type	THERM	RES	FAST	Total flux (neutrons cm^{-2} sec^{-1})
PWR-U	0.0748	0.0251	0.3113	3.25×10^{14}
PWR-Pu				
PWR-PuU	0.0698	0.0255	0.3129	3.18×10^{14}
PWR-PuPu	0.0322	0.0235	0.3510	3.26×10^{14}
BWR-U	0.0834	0.0256	0.2345	2.35×10^{14}
BWR-Pu				
BWR-PuU	0.0739	0.0262	0.2370	2.55×10^{14}
BWR-PuPu	0.0505	0.0246	0.2563	2.08×10^{14}

[a]Based on XSDRNPM calculations discussed in Sects. 3.1 and 3.2.

Table 4.2. Values of THERM, RES, and FAST to be used with LWR thermal fluxes

Fuel type	THERM		RES		FAST	
	old	new[a]	old	new[a]	old	new[a]
PWR-U	0.632	0.701	0.333	0.304	2.000	2.010
PWR-Pu						
PWR-PuU	0.632	0.592	0.333	0.366	2.000	2.390
PWR-PuPu	0.500	0.509	0.550	0.730	4.000	5.490
BWR-U	0.632	0.676	0.333	0.307	2.000	1.514
BWR-Pu						
BWR-PuU	0.632	b	0.333	0.355	2.000	1.694
BWR-PuPu	0.500	b	0.550	0.487	4.000	2.599

[a]Based on XSDRNPM calculations discussed in Sects. 3.1 and 3.2; calculation is approximate due to the nature of the old ORIGEN fluxes.

[b]Not calculable.

4.2 Input Compositions and Masses

There are three different composition-related aspects to 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.

4.2.1 Fuel-assembly structural material masses and compositions

4.2.1.1 Composition. The seven structural materials used in the revised ORIGEN LWR models are Zircaloy-2, Zircaloy-4, Inconel 718, Inconel X-750, stainless steel 302, stainless steel 304, and Nicrobraze 50. The assumed elemental composition of each of these materials is given in Table 4.3.

A parameter of particular interest in Table 4.3 is the cobalt content of these structural materials. This is because naturally occurring ^{59}Co produces ^{60}Co which emits high-energy gamma rays that inhibit 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 $\sim$0.08 wt % for stainless steels.[7]

Table 4.3. Assumed elemental compositions of LWR fuel-assembly structural materials

Element	Atomic number	Structural material composition, grams per tonne of 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, grams/cm^3	--	6.56	6.56	8.19	8.30	8.02	8.02	--
References		3, 4	3, 4	4-7	4-7	4, 6-8	4, 6-8	9

[a]Value used in ORIGEN should be less than this (actual) value if the materials are not in the active fuel zone; see Sect. 3.5 and Table 3.7 for details.

4.2.1.2 Structural material mass distribution. The PWR and BWR fuel-assembly structural material distributions recommended for use in reactor depletion calculations are given in Table 4.4. The various materials have been grouped into three categories (i.e., fuel zone, fuel-gas plenum zone, and end fitting zone) according to their proximity to the active fuel region. These categories correspond to those for which effective neutron fluxes were determined using axial neutron transport calculations (see Sect. 3.5). The elemental compositions of all materials listed in Table 4.4 are given in Table 4.3. It should be noted that the BWR fuel channel, which encloses the square array of fuel elements to prevent cross-flow between adjacent assemblies, is assumed to be used once before being discarded, even though it can be used more than once.[10]

4.2.1.3 Fuel-assembly dimensions. The physical parameters of spent PWR and BWR fuel assemblies are given in Table 4.5, even though they are not directly related to ORIGEN calculations. These parameters are presented because of the current interest in this information as it relates to the possible storage and disposal of unreprocessed spent fuel.

4.2.2 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 fuel material. A typical set of nonactinide element concentrations in fresh LWR oxide fuel is given in Table 4.6. The values in Table 4.6 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 4.6.

Table 4.4. Assumed fuel-assembly structural material mass distribution

	PWR[11,12]			BWR[13,14]		
	Material	Mass kg/MTHM	Mass kg/assembly	Material	Mass kg/MTHM	Mass kg/assembly
Fuel zone						
Cladding	Zircaloy-4	223.0	102.9	Zircaloy-2	279.5	51.2
Fuel channel[a]	--	--	--	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[b]	9.9	4.6	--	--	--
Fuel-gas plenum zone						
Cladding	Zircaloy-4	12.0	5.5	Zircaloy-2	25.4	4.7
Fuel channel[a]	--	--	--	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]Assumed to be discarded after one cycle (ref. 10).

[b]Distributed throughout the PWR core in sleeves and so forth.

Table 4.5. Physical characteristics of LWR fuel assemblies

	BWR[15]	PWR[16]
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[a]	108.4[b]
Hardware/assembly, kg	12.4[c]	26.1[d]
Total metal/assembly, kg	111.9	134.5
Nominal volume/assembly, m^3	0.0864[e]	0.186[e]

[a]Includes Zircaloy fuel-element spacers and fuel channel.

[b]Includes Zircaloy control-rod guide thimbles.

[c]Includes stainless steel tie-plates, Inconel springs, and plenum springs.

[d]Includes stainless steel nozzles and Inconel-718 grids.

[e]Based on overall outside dimension.

Table 4.6. Assumed nonactinide composition of LWR oxide fuels

Element	Atomic number	Concentration (ppm)[a]	Reference	Element	Atomic number	Concentration (ppm)[a]	Reference
Li	3	1.0	17	Mn	25	1.7	18
B	5	1.0	18	Fe	26	18.0	18
C	6	89.4	18	Co	27	1.0	18
N	7	25.0	4	Ni	28	24.0	18
O	8	134,454	b	Cu	29	1.0	18
F	9	10.7	18	Zn	30	40.3	18
Na	11	15.0	18	Mo	42	10.0	18
Mg	12	2.0	18	Ag	47	0.1	18
Al	13	16.7	18	Cd	48	25.0	18
Si	14	12.1	18	In	49	2.0	18
P	15	35.0	19	Sn	50	4.0	18
Cl	17	5.3	18	Gd	64	2.5[c]	18
Ca	20	2.0	18	W	74	2.0	18
Ti	22	1.0	20	Pb	82	1.0	18
V	23	3.0	18	Bi	83	0.4	20
Cr	24	4.0	18				

[a]Parts of element per million parts of heavy metal.

[b]Stoichiometric quantity for MO_2 fuel.

[c]Average of 1573 ppm of gadolinium in BWR fuel rods as a burnable poison (ref. 10).

The 2.5-ppm gadolinium concentration given in Table 4.6 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.[10]

4.2.3 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 four LWRs is given in Table 4.7. The average fuel composition for each of the plutonium recycle reactors (e.g., the PWR-Pu) is comprised of two parts: a uranium-enriched part (e.g., PWR-PuU) that accounts for ∿70% of the fresh fuel and a plutonium-enriched part (e.g., PWR-PuPu) that accounts for ∿30% of the fresh fuel. The plutonium compositions in Table 4.7 are typical of those expected at the beginning of the second plutonium recycle and include the plutonium produced during the first plutonium recycle in both the uranium- and plutonium-enriched fuel. Thus, the model represents self-generated plutonium recycle (see Sect. 3.3), but not steady-state plutonium recycle, which would require that the amount of plutonium discharged after recycle N, less processing losses, be equal to the amount charged to the plutonium-enriched fuel in recycle N+1.

4.3 ORIGEN Depletion Calculations

This section describes and compares the results of ORIGEN depletion calculations for the four reactor (e.g., PWR-U, PWR-Pu, BWR-U, and BWR-Pu) fuel-cycle combinations considered in the previously described CITATION[21] calculations (Sect. 3.4). These ORIGEN calculations require (1) the implementation of the modifications described in Sect. 4.1, (2) cross

Table 4.7. Assumed initial composition of one average metric ton of heavy metal in LWRs

	PWR-U	PWR-Pu			BWR-U	BWR-Pu		
		PWR-PuU	PWR-PuPu	Total		BWR-PuU	BWR-PuPu	Total
^{234}U, g	290	203	16	219	247	173	16	189
^{235}U, g	32,000	22,400	2,038	24,438	27,500	19,250	2,049	21,301
^{238}U, g	967,710	677,397	284,317	961,714	972,253	680,577	286,159	966,717
Total U, g	1,000,000	700,000	286,371	986,371	1,000,000	700,000	288,224	988,207
^{238}Pu, g	0	0	304	304	0	0	311	312
^{239}Pu, g	0	0	6,722	6,722	0	0	5,723	5,730
^{240}Pu, g	0	0	3,472	3,472	0	0	3,229	3,240
^{241}Pu, g	0	0	1,995	1,995	0	0	1,564	1,563
^{242}Pu, g	0	0	1,136	1,136	0	0	949	948
Total Pu, g	0	0	13,629	13,629	0	0	11,776	11,793
Total heavy metal, g	1,000,000	700,000	300,000	1,000,000	1,000,000	700,000	300,000	1,000,000

sections resulting from the reactor physics calculations described in Sect. 3, and (3) the input masses and compositions described in Sect. 4.2. A brief comparison of the PWR-U and BWR-U discharged-fuel composition predicted by CITATION and ORIGEN is given in Sect. 4.3.1. Section 4.3.2 contains comparisons of the discharged-fuel composition predicted by ORIGEN for all four reactor/fuel cycle combinations with available literature values. Section 4.3.3 contains a summary description of the ORIGEN U-Pu cycle LWR models.

4.3.1 Comparison of ORIGEN and CITATION depletion calculations

Actinide depletion calculations were performed for the PWR-U and BWR-U reactors with both the ORIGEN and CITATION computer codes to verify the accuracy of ORIGEN (or CITATION) as a depletion module and to test the effectiveness of the ORIGEN modifications described in Sect. 4.1. A summary of the results of these calculations is given in Table 4.8.

In general, the discharged compositions of the spent fuels predicted by the two depletion codes agree very well, with most differences being <1%. Any differences >1% can be attributed to the exclusion of certain decay chains from the CITATION calculations in order to eliminate the presence of feedback loops that cannot be handled by CITATION. The decay loops that were eliminated are the alpha-decay (100% of all decay events) of ^{242}Cm to ^{238}Pu and the electron-capture-decay (18% of all decay events) of ^{242}Am to ^{242}Pu. If the ^{242}Cm alpha-decays were included in CITATION, the effect would be an increase in the amounts of ^{238}Pu and total plutonium at discharge. The inclusion of the ^{242}Am electron-capture-decays would decrease the amount of ^{242}Cm present at discharge somewhat and increase the amounts of ^{242}Pu, ^{243}Am, and ^{244}Cm present at discharge.

Table 4.8. Comparison of CITATION and ORIGEN depletion calculations at discharge for uranium-enriched LWRs

Nuclide	PWR-U[a]			BWR-U[d]		
	CITATION[b] (g/MTIHM)	ORIGEN (g/MTIHM)	Difference[c] (%)	CITATION[b] (g/MTIHM)	ORIGEN (g/MTIHM)	Difference[c] (%)
^{234}U	179	180	-0.9[e]	157	157	0[e]
^{235}U	8,068	7,936	+0.6[e]	7,567	7,522	+0.2[e]
^{236}U	3,948	3,963	+0.4	3,307	3,314	+0.2
^{238}U	944,100	944,100	0[e]	951,700	951,700	0[e]
Σ		956,179				
^{237}Np	434	441	-1.6	334	338	+1.2
^{238}Pu	112[f]	127	+13.4[g]	80.6[f]	93.2	+15.6[g]
^{239}Pu	4,972	5,033	+1.2	4,787	4,831	+0.9
^{240}Pu	2,295	2,318	+1.0	2,086	2,099	+0.6
^{241}Pu	1,205	1,223	+1.5	1,086	1,094	+0.7
^{242}Pu	444[h]	461	-3.8[i]	370[h]	379	+2.4[i]
Total Pu	9,029[f,h]	9,162	+1.5[g,i]	8,410[f,h]	8,496	+1.0[g,i]
^{241}Am	32.4	31.7	-2.2	36.6	36.1	-1.4
^{243}Am	81.4[h]	85.9	+5.5[i]	61.6[h]	63.7	+3.4[i]
^{242}Cm	13.2[j]	11.4	-13.6[g]	12.0[j]	10.2	-15.0[g]
^{244}Cm	17.8[h]	19.0	+6.7[i]	12.3[h]	12.8	+4.1[i]

[a]Initial composition: 290 g ^{234}U; 32,000 g ^{235}U; 967,710 g ^{238}U.

[b]Ref. 21.

[c]% difference = 100 (ORIGEN-CITATION)/ORIGEN.

[d]Initial composition: 247 g ^{234}U; 27,500 g ^{235}U; 972,253 g ^{238}U.

[e]% difference based on depletion from initial amount.

[f]Would be increased by accounting for ^{242}Cm decays neglected in CITATION.

[g]Magnitude of difference would be reduced by considering ^{242}Cm decays to ^{238}Pu in CITATION.

[h]Would be increased by accounting for ^{242}Am electron-capture decays neglected in CITATION.

[i]Magnitude of difference would be reduced by considering ^{242}Am electron-capture decays in CITATION.

[j]Would be decreased by accounting for ^{242}Am electron-capture decays neglected in CITATION.

In all cases, these changes serve to decrease the magnitude of the difference between the CITATION and ORIGEN depletion calculations.

4.3.2 Comparison of ORIGEN and literature depletion calculations

Since the reactor models used in the CITATION calculations were relatively simple, there is no assurance that the results of the ORIGEN calculations are correct, even if they are in good agreement with the CITATION calculations. Thus, it is necessary to compare the ORIGEN results with similar calculations from an exogenous source. Unfortunately, there is no single "reference" set of calculations which can serve as a basis for comparison, and large-scale dissolution and analysis of spent fuels from modern LWRs have not been performed yet. The net result is that the only exogenous basis of comparison is the multitude of spent fuel compositions that have been published in the literature. The appraoch taken in the following discussion is to compare the results of ORIGEN depletion calculations with the published discharged-fuel compositions. This procedure generally necessitates multiple ORIGEN runs for a given reactor since the different sources assumed different burnups and/or initial enrichments. It should be noted that the appearance of a spent fuel composition from the literature does not necessarily imply that it is more accurate than the ORIGEN calculation, since the methods and assumptions used in many of the literature calculations are unknown.

4.3.2.1 PWR-U fuel. The comparison of the PWR-U spent fuel compositions calculated by ORIGEN with those given in the literature is shown in Table 4.9. Each column entitled "ORIGEN" is followed by one or more columns which give the comparable literature values. As is evident by inspection, the agreement between ORIGEN and the literature values for

Table 4.9. Comparison of ORIGEN PWR-U fuel depletion calculations with literature values

Nuclide	Discharge PWR-U fuel composition (g/MTIHM)										
	ORIGEN	Ref. 22	Ref. 23	Ref. 24[a]	Ref. 25	Ref. 26	Ref. 27[a]	ORIGEN	Ref. 28	ORIGEN	Ref. 29
^{234}U	180 (D = 110)[b]	158 (D = 132)	- (-)	159 (D = 131)	- (-)	- (-)	116 (D = 174)	- (-)	- (-)	- (-)	- (-)
^{235}U	7936 (D = 24,064)[b]	8151 (D = 23,849)	8500 (D = 23,500)	- (-)	8006 (D = 23,994)	7924 (D = 24,076)	7460 (D = 24,440)	9018 (D = 25,182)	9370 (D = 24,830)	8084 (D = 22,016)	8148 (D = 21,952)
^{236}U	3963	4002	4005	3983	4500[c]	4033	4420	-	-	3653	3588
^{238}U	944,100 (D = 23,610)[b]	943,430 (D = 24,280)	942,788 (D = 25,212)	- (-)	933,585[d] (D = 34,125)[d]	943,742 (D = 23,967)	944,000 (D = 23,810)	- (-)	- (-)	947,300 (D = 22,310)	946,764 (D = 22,846)
^{237}Np	441	449	-	453	-	-	482	-	-	-	-
^{238}Pu	127 (1.4%)[e]	131 (1.4%)	- (-)	135 (1.4%)	178 (1.9%)	- (-)	167 (1.8%)	120 (1.3%)	- (-)	107 (1.2%)	- (-)
^{239}Pu	5033 (54.9%)[e]	5313 (56.2%)	5570 (57.3%)	5490 (56.9%)	5250 (57.0%)	5036 (54.8%)	5300 (58.4%)	5007 (55.8%)	5370 (57.6%)	5008 (56.1%)	5321 (58.1%)
^{240}Pu	2318 (25.3%)[e]	2309 (24.4%)	2238 (23.0%)	2322 (24.1%)	2180 (23.7%)	2219 (24.1%)	2200 (24.2%)	2263 (25.2%)	2220 (23.8%)	2219 (24.9%)	2098 (22.9%)
^{241}Pu	1223 (13.3%)[e]	1252 (13.3%)	1370 (14.1%)	1245 (12.9%)	1150 (12.5%)	1243 (13.5%)	1044 (11.5%)	1171 (13.0%)	1310 (14.0%)	1168 (13.1%)	1277 (14.0%)
^{242}Pu	461 (5.0%)[e]	446 (4.7%)	542 (5.6%)	451 (4.7%)	452 (4.9%)	540 (5.9%)	372 (4.1%)	418 (4.7%)	430 (4.6%)	419 (4.7%)	456 (5.0%)
Fissile Pu	6256 (68.3%)	6565 (69.5%)	6940 (71.5%)	6735 (69.8%)	6400 (69.5%)	6279 (69.5%)	6344 (69.9%)	6178 (68.8%)	6680 (71.6%)	6176 (69.2%)	6598 (72.1%)
Total Pu	9162	9451	9712	9643	9210	9038	9080	8978	9330	8921	9151
^{241}Am	31.7	44.2	-	128	-	-	40.8	30.6	-	-	-
^{243}Am	85.9	88.0	-	89	-	-	102	74.0	-	-	-
^{242}Cm	11.4	17.0	-	5.0	-	-	5.1	10.4	-	-	-
^{244}Cm	19.0	24.3	-	23.8	-	-	20.2	15.5	-	-	-
Initial enrichment, wt % ^{235}U	3.20	3.20	3.20	?	3.20	3.20	3.19	3.42	3.42	3.01	3.01
Burnup, MWd(t)/tonne heavy metal	33,000	33,000	32,854	?	33,000	33,100	33,000	33,000	33,000	30,390	30,390

[a] Spent fuel composition after a decay time of 150 days. This will reduce the amount of ^{242}Cm present by a factor of 2 and increase ^{241}Am by ~19 g/MTIHM.

[b] Indicates depletion of isotope in g/MTIHM.

[c] Initial composition contained 600 g of ^{236}U/MTIHM.

[d] Probable typographical error; numbers should be 943,585 and D = 24,125.

[e] Isotopic composition of plutonium as a weight percentage.

the PWR-U is generally quite good, even for the americium and curium isotopes. It should be noted that two of the cases assume decay times of 150 days, thus affecting the amounts of ^{241}Am and ^{242}Cm present. These cases have been designated with a footnote.

A second comparison of the PWR-U discharged-fuel composition is given in Table 4.10, with the basis for comparison being the discharged-fuel composition predicted by the original ORIGEN cross section libraries. The agreement for ^{235,238}U and the principal plutonium isotopes is reasonably good; however, the new cross sections result in significant changes in the amounts of ^{238}Pu, ^{243}Am, and ^{244}Cm present at discharge.

4.3.2.2 BWR-U fuel. The comparison of the ORIGEN and literature spent fuel compositions for BWR-U fuels is given in Table 4.11. In general, the agreement is quite good, although significant differences do exist between refs. 25, 30, and 31. It should be noted however that this agreement was obtained at the expense of having to adjust the moderator density somewhat arbitrarily, as described in Sect. 3.2.3.

4.3.2.3 PWR-PuU fuel. The comparison of the ORIGEN and the single literature spent fuel compositions for the uranium-enriched fuel in a self-generated plutonium recycle reactor is given in Table 4.12. The agreement between ORIGEN and the literature values is quite good in this case, although it is difficult to generalize here because of the limited comparison basis.

4.3.2.4 PWR-PuPu fuel. The ORIGEN and literature spent fuel composition predictions for the mixed-oxide fuel in a self-generated plutonium recycle reactor are given in Table 4.13. The different assumptions used in the literature calculations make the comparison of

Table 4.10. Comparison of PWR-U discharged-fuel composition predicted by the old and new ORIGENs

	Fuel composition (g/MTIHM)		
		Discharge	
	Charge	Old ORIGEN[a]	New ORIGEN
^{235}U	33,000	7,990	8,264
^{236}U	0	4,555	4,058
^{238}U	967,000	942,480	943,700
^{237}Np	0	472	437
^{238}Pu	0	161	120
^{239}Pu	0	5,266	5,175
^{240}Pu	0	2,165	2,415
^{241}Pu	0	1,034	1,248
^{242}Pu	0	353	404
Total Pu	0	8,979	9,362
^{241}Am	0	25.1	30.6
^{243}Am	0	94.1	68.1
^{242}Cm	0	10.1	9.77
^{244}Cm	0	30.3	13.3

[a]Data obtained from ref. 1.

Table 4.11. Comparison of ORIGEN BWR-U fuel-depletion calculations with literature values

Nuclide	Discharge BWR-U fuel composition, g/MTIHM										
	ORIGEN	Ref. 22	Ref. 24[a]	Ref. 27[a]	Ref. 32	ORIGEN	Ref. 25	Ref. 30	Ref. 31	ORIGEN	Ref. 33
^{234}U	157 (D = 90)[b]	140 (D = 107)	140 (-)	127 (-)	- (-)	154 (D = 93)	- (-)	- (-)	- (-)	- (-)	- (-)
^{235}U	7522 (D = 19,978)[b]	7657 (D = 19,843)	- (-)	7480 (D = 19,820)	7301 (D = 20,099)	6608 (D = 18,994)	6400 (D = 19,300)	6209 (D = 19,391)	7002 (D = 18,597)	10,020 (D = 19,980)	9478 (D = 20,522)
^{236}U	3314	3344	3342	3660	-	3134	3700[c]	3223	3207	-	-
^{238}U	951,700 (D = 20,553)[b]	951,426 (D = 20,827)	- (-)	950,000 (D = 22,453)	- (-)	952,600 (D = 21,553)	953,000 (D = 20,700)	954,520 (D = 19,033)	952,096 (D = 22,051)	- (-)	- (-)
^{237}Np	338	360	362	368	-	-	-	-	-	-	-
^{238}Pu	93.2 (1.1%)[d]	109 (1.3%)	109 (1.3%)	120 (1.4%)	- (-)	97.4 (1.1%)	155 (1.9%)	- (-)	- (-)	- (-)	- (-)
^{239}Pu	4831 (56.9%)[d]	4905 (58.1%)	5070 (60.2%)	5260 (60.6%)	- (-)	4856 (56.0%)	4630 (56.9%)	4212 (57.0%)	4266 (56.3%)	- (-)	- (-)
^{240}Pu	2099 (24.7%)[d]	2067 (24.5%)	2064 (24.0%)	2080 (24.0%)	- (-)	2149 (24.8%)	2040 (25.1%)	1753 (23.7%)	1813 (23.9%)	- (-)	- (-)
^{241}Pu	1094 (12.9%)[d]	1025 (12.1%)	1020 (11.9%)	930 (10.7%)	- (-)	1143 (13.2%)	935 (11.5%)	991 (13.4%)	1023 (13.5%)	- (-)	- (-)
^{242}Pu	379 (4.5%)[d]	335 (4.0%)	338 (3.9%)	294 (3.4%)	- (-)	418 (4.8%)	378 (4.6%)	428 (5.8%)	470 (6.2%)	- (-)	- (-)
Fissile Pu	5925 (69.7%)	5930 (70.3%)	6090 (70.8%)	6190 (71.3%)	5648 (-)	5999 (69.2%)	5565 (68.4%)	5203 (70.5%)	5289 (69.8%)	5636 (72.0%)	5638 (69.3%)
Total Pu	8496	8441	8601	8684	-	8664	8138	7385	7578	7832	8130
^{241}Am	36.1	53.8	34.0	42.1	-	37.6	-	-	-	-	-
^{243}Am	63.7	62.7	62.8	71.5	-	73.9	-	-	-	-	-
^{242}Cm	10.2	14.1	4.3	4.5	-	11.1	-	-	-	-	-
^{244}Cm	12.8	16.0	15.6	20.2	-	15.6	-	-	-	-	-
Initial enrichment, wt % ^{235}U	2.75	2.75	?	2.73	2.74	2.56	2.57	2.60	2.56	3.0	3.0
Burnup, MWd(t)/tonne heavy metal	27,500	27,500	?	27,500	29,048	27,500	27,500	27,037	27,500	25,172	25,000

[a] Spent fuel composition after a decay time of 150 days. This will reduce the amount of ^{242}Cm present by ~19 g/MTIHM.

[b] Indicates depletion of isotope in g/MTIHM.

[c] Initial composition contained 600 g ^{236}U/MTIHM.

[d] Isotopic composition of plutonium as a weight percentage.

Table 4.12. Comparison of ORIGEN PWR-PuU fuel-depletion calculations with literature values

Nuclide	Discharged PWR-PuU fuel composition (g/MTIHM)	
	ORIGEN	Ref. 29
^{234}U	- (-)	- (-)
^{235}U	8991 (D = 22,009)[a]	8674 (D = 22,331)
^{236}U	3669	-
^{238}U	946,200 (D = 20,800)	946,174 (D = 22,821)
^{237}Np	413	-
^{238}Pu	- (-)	- (-)
^{239}Pu	5524 (59.3%)[b]	5338 (58.5%)
^{240}Pu	2101 (22.5%)	2046 (22.4%)
^{241}Pu	1277 (13.7%)	1290 (14.2%)
^{242}Pu	421 (4.5%)	445 (4.9%)
Fissile Pu	6801 (72.9%)	6628 (72.7%)
Total Pu	9323	9119
^{241}Am	32.2	-
^{243}Am	75.3	-
^{242}Cm	10.7	-
^{244}Cm	15.4	-
Initial enrichment, wt % ^{235}U	3.10	3.10
Burnup, MWd(t)/MTIHM	30,197	30,390

[a]Indicates depletion of isotope in g/MTIHM.

[b]Isotopic composition of plutonium as a weight percentage.

Table 4.13. Comparison of ORIGEN PWR-PuPu fuel-depletion calculations with literature values

Nuclide	Discharged PWR-PuPu fuel composition (g/MTIHM)								
	ORIGEN	Ref. 22	ORIGEN	Ref. 25	ORIGEN	Ref. 28	ORIGEN	Ref. 29	Ref. 24[a]
^{234}U	36 (D = 18)[b]	34 (D = 20)	- (-)	- (-)	- (-)	- (-)	- (-)	- (-)	36 (-)
^{235}U	3280 (D = 3528)	3071 (D = 3738)	3214 (D = 3578)	2996 (D = 3790)	3427 (D = 3343)	3600 (D = 3170)	3452 (D = 3368)	3458 (D = 3362)	- (-)
^{236}U	699	722	706	735	-	-	-	-	692
^{238}U	928,300 (D = 22,456)	930,714 (D = 20,087)	924,900 (D = 22,829)	925,613 (D = 21,800)	931,100 (D = 20,900)	928,520 (D = 23,460)	930,100 (D = 21,009)	929,786 (D = 21,323)	- (-)
^{237}Np	168	160	-	-	-	-	-	-	154
^{238}Pu	169[c] (0.5%)[d]	306[c] (1.0%)	832 (2.5%)	1135 (3.4%)	- (-)	- (-)	- (-)	- (-)	338[c] (1.0%)
^{239}Pu	12,251 (38.2%)	10,565 (35.9%)	11,991 (35.7%)	11,807 (35.6%)	12,467 (35.9%)	13,900 (39.2%)	12,242 (37.9%)	12,199 (37.9%)	12,012 (37.1%)
^{240}Pu	10,510 (32.8%)	9271 (31.5%)	10,320 (30.7%)	8684 (26.2%)	10,750 (30.9%)	8570 (24.2%)	10,110 (31.3%)	9989 (31.0%)	10,092 (31.1%)
^{241}Pu	5935 (18.5%)	5826 (19.8%)	6070 (18.1%)	6420 (19.3%)	6673 (19.2%)	7780 (21.9%)	6024 (18.6%)	5955 (18.5%)	6364 (19.6%)
^{242}Pu	3216 (10.0%)	3461 (11.8%)	4386 (13.1%)	5135 (15.5%)	4866 (14.0%)	5210 (14.7%)	3928 (12.2%)	4034 (12.5%)	3606 (11.1%)
Fissile Pu	18,186 (56.6%)	16,391 (55.7%)	18,061 (53.8%)	18,227 (54.9%)	19,140 (55.1%)	21,680 (61.1%)	18,266 (56.5%)	18,154 (56.4%)	18,376 (56.7%)
Total Pu	32,081	29,429	33,599	33,181	34,756	35,460	32,304	32,177	32,412

Table 4.13 (continued)

Nuclide	Discharged PWR-PuPu fuel composition (g/MTIHM)								
	ORIGEN	Ref. 22	ORIGEN	Ref. 25	ORIGEN	Ref. 28	ORIGEN	Ref. 29	Ref. 24[a]
^{241}Am	323	421	-	-	-	-	-	-	792
^{243}Am	1068	972	-	-	-	-	-	-	1005
^{242}Cm	91	133	-	-	-	-	-	-	44
^{244}Cm	371	440	-	-	-	-	-	-	431
Initial enrichment, wt % ^{235}U	0.681	0.681	0.679	0.679	0.677	0.677	0.682	0.682	?
wt % $^{239}Pu + ^{241}Pu$	2.882	2.882	2.905	2.905	3.201	3.201	2.776	2.776	?
Total	3.563	3.563	3.584	3.584	3.878	3.878	3.458	3.458	?
Final enrichment, wt % $^{235}U + ^{239}Pu + ^{241}Pu$	2.147	1.996	2.128	2.122	2.257	2.528	2.172	2.161	?
Burnup, MWd(t)/MTIHM	33,000	33,000	34,039	34,037	33,000	33,000	30,390	30,390	?

[a]Spent fuel composition after a decay time of 150 days. This will reduce the amount of ^{242}Cm present by about a factor of 2 and increase the amount of ^{241}Am by ~9 g/MTIHM.

[b]Indicates depletion of isotope in g/MTIHM.

[c]No ^{238}Pu present in plutonium charged to reactor.

[d]Isotopic composition of plutonium as a weight percentage.

these numbers with each other to determine their internal consistency very difficult. However, despite this variation, the ORIGEN results agree well with nearly all of the literature values. It should be noted that the ORIGEN spent fuel compositions represent the discharge from the second plutonium recycle. When possible, second-recycle values were also selected from the literature. However, in some cases the recycle number associated with the literature spent fuel compositions was unknown, and they were used on this basis.

4.3.2.5 BWR-PuU fuel. The results of the ORIGEN depletion of the uranium-enriched portion of a self-generated plutonium-recycle BWR are given in Table 4.14. No literature comparisons were available for this fuel.

4.3.2.6 BWR-PuPu fuel. The comparison of the ORIGEN and literature spent fuel compositions for the mixed-oxide fuel in a self-generated plutonium-recycle BWR is given in Table 4.15. In an overall sense, the agreement appears to be good and similar to that obtained for the BWR-U fuel (see Table 4.11). However, close inspection shows significant differences between ORIGEN and the literature values concerning the amounts of ^{239}Pu and ^{240}Pu in the spent fuel. The source of these differences is unknown, but the use of an adjusted moderator density identical to the one used for the BWR-U fuel may be a factor.

4.3.3 Summary description of the ORIGEN U-Pu cycle LWR models

Fuel-depletion calculations have been performed for the six U-Pu cycle LWR fuel types considered in this report — PWR-U, PWR-PuU, PWR-PuPu, BWR-U, BWR-PuU, and BWR-PuPu — using input compositions given in Table 4.7. The results of these calculations, including irradiation conditions and

Table 4.14. Results of ORIGEN BWR-PuU fuel depletion

Nuclide	Discharged BWR-PuU fuel composition (g/MTIHM)
^{234}U	154 (D = 93)[a]
^{235}U	7987 (D = 19,513)
^{236}U	3275
^{238}U	950,300 (D = 21,953)
^{237}Np	370.3
^{238}Pu	103 (1.1%)[b]
^{239}Pu	5608 (59.4%)
^{240}Pu	2148 (22.7%)
^{241}Pu	1211 (12.8%)
^{242}Pu	374 (4.0%)
Fissile Pu	6819 (72.2%)
Total Pu	9444
^{241}Am	37.9
^{243}Am	70.2
^{242}Cm	10.9
^{244}Cm	14.9
Initial enrichment, wt % ^{235}U	2.75
Burnup, MWd(t)/MTIHM	27,500

[a]Indicates depletion of isotope in g/MTIHM.

[b]Isotopic composition of plutonium as a weight percentage.

Table 4.15. Comparison of ORIGEN BWR-PuPu fuel-depletion calculations with literature values

Nuclide	Discharged BWR-PuPu fuel composition (g/MTIHM)				
	ORIGEN	Ref. 25	Ref. 27[a]	ORIGEN	Ref. 28
^{234}U	39 (D = 15)[b]	37 (D = 17)	37 (-)	- (-)	- (-)
^{235}U	3295 (D = 3585)	3199 (D = 3657)	- (-)	3111 (D = 3718)	2875 (D = 954)
^{236}U	678	707	687	700	749
^{238}U	940,100 (D = 17,311)	939,947 (D = 17,464)	- (-)	935,600 (D = 18,314)	934,805 (D = 19,109)
^{237}Np	113	149	146	-	-
^{238}Pu	155[c] (0.6%)[d]	369[c] (1.4%)	304[c] (1.1%)	821 (2.9%)	1196 (4.2%)
^{239}Pu	9349 (35.4%)	9770 (37.6%)	10,658 (38.6%)	8995 (31.8%)	9848 (34.4%)
^{240}Pu	9516 (36.1%)	8290 (31.9%)	8706 (31.5%)	9683 (34.3%)	8200 (28.7%)
^{241}Pu	4537 (17.2%)	4724 (18.2%)	5027 (18.2%)	4788 (17.0%)	5002 (17.5%)
^{242}Pu	2833 (10.7%)	2831 (10.9%)	2920 (10.6%)	3956 (14.0%)	4356 (15.2%)
Fissile Pu	13,886 (52.6%)	14,494 (55.8%)	15,685 (56.8%)	13,783 (48.8%)	14,850 (51.9%)
Total Pu	26,390	25,984	27,615	28,243	28,602
^{241}Am	304	508	639	-	-
^{243}Am	814	769	839	-	-
^{242}Cm	75.6	109	36.3	-	-
^{244}Cm	244	329	32.5	-	-
Initial enrichment, wt % ^{235}U	0.686	0.686	?	0.683	0.683
wt % (^{239}Pu + ^{241}Pu)	2.427	2.427	?	2.429	2.429
Total	3.112	3.112	?	3.112	3.112
Final enrichment, wt % (^{235}U + ^{239}Pu + ^{241}Pu)	1.717	1.769	?	1.689	1.773
Burnup, MWd(t)/MTIHM	27,500	27,500	?	29,490	29,490

[a]Spent fuel composition after a decay time of 150 days. This will reduce the amount of ^{242}Cm present by about a factor of 2 and increase the amount of ^{241}Am by ∿9 g/MTIHM.

[b]Indicates depletion of isotope in g/MTIHM.

[c]No ^{238}Pu present in plutonium charged to reactor.

[d]Isotopic composition of plutonium as a weight percentage.

measures of the uranium and plutonium contents of the fresh and spent fuels, are given in Table 4.16. Many of the values in this table are given in terms of both kilograms per cycle and kilograms per year, assuming an 80% capacity factor for the latter.

Table 4.16. Summary of ORIGEN LWR model characteristics

Parameter	PWR-U	PWR-Pu			BWR-U	BWR-Pu		
		PWR-PuU	PWR-PuPu	Total		BWR-PuU	BWR-PuPu	Total
Electric power, MW(e)	1250	875	375	1250	1250	875	375	1250
Thermal power, MW(t)	3800	2660	1140	3800	3800	2660	1140	3800
Average specific power,[a] MW(t)/MTIHM[b]	37.5	37.5	37.5	37.5	25.9	25.9	25.9	25.9
Average fuel burnup, MWd/MTIHM[b]	33,000	33,000	33,000	33,000	27,500	27,500	27,500	27,500
Irradiation duration, full-power days	880	880	880	880	1062	1062	1062	1062
Refueling cycle length, full-power days	293.3	293.3	293.3	293.3	265.5	265.5	265.5	265.5
days at 80% capacity factor	367	367	367	367	332	332	332	332
Charge, kg/refueling cycle (kg/year at 80% capacity factor)								
^{235}U	1081 (1077)	757 (754)	68.8 (68.6)	826 (823)	1009 (1110)	706 (777)	75.2 (82.7)	781 (860)
Total U	33,778 (33,647)	23,644 (23,553)	9674 (9636)	33,318 (33,189)	36,680 (40,369)	25,676 (28,258)	10,572 (11,636)	36,248 (39,894)
Fissile Pu[c]	0 (0)	0 (0)	294 (293)	294 (293)	0 (0)	0 (0)	267 (294)	267 (294)
Total Pu[d]	0 (0)	0 (0)	460 (459)	460 (459)	0 (0)	0 (0)	432 (475)	432 (475)
Total (U + Pu)	33,778 (33,647)	23,644 (23,553)	10,134 (10,095)	33,778 (33,648)	36,680 (40,369)	25,676 (28,258)	11,004 (12,111)	36,680 (40,369)

Table 4.16 (continued)

Parameter	PWR-U	PWR-Pu			BWR-U	BWR-Pu		
		PWR-PuU	PWR-PuPu	Total		BWR-PuU	BWR-PuPu	Total
Discharge, kg/refueling cycle (kg/year at 80% capacity factor)								
^{235}U	267 (266)	199 (198)	33.5 (33.4)	233 (231)	276 (304)	205 (226)	36.7 (40.4)	242 (266)
Total U	32,173 (32,048)	22,592 (22,504)	9422 (9385)	32,014 (31,889)	35,319 (38,872)	24,693 (27,177)	10,356 (11,398)	35,049 (38,575)
Fissile Pu[c]	210 (209)	164 (163)	186 (185)	350 (348)	217 (239)	175 (193)	157 (173)	332 (366)
Total Pu[d]	308 (307)	232 (231)	344 (342)	576 (573)	309 (340)	242 (267)	318 (350)	560 (617)
Total (U + Pu)	32,481 (32,354)	22,824 (22,735)	9766 (9727)	32,590 (32,462)	35,628 (39,212)	24,935 (27,444)	10,674 (11,748)	35,609 (39,192)
Total heavy metal	32,503 (32,376)	22,840 (22,751)	9795 (9756)	32,635 (32,507)	35,647 (39,233)	24,949 (27,459)	10,697 (11,773)	35,646 (39,232)

[a]Based on full power and fuel charged.

[b]MTIHM = metric ton of initial heavy metal.

[c]$^{239}Pu + ^{241}Pu + ^{239}Np$.

[d]$^{238}Pu + ^{239}Pu + ^{240}Pu + ^{241}Pu + ^{242}Pu + ^{239}Np$.

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APPENDIX A:

84-ENERGY-GROUP NEUTRON SPECTRA

GRAPHS AND LISTINGS

APPENDIX A: 84-ENERGY-GROUP NEUTRON SPECTRA GRAPHS AND LISTINGS

This appendix contains graphs and listings of 84-energy-group neutron spectra for the six U-Pu cycle LWR fuel types considered in this report as calculated by the XSDRNPM code. The units of the neutron fluxes in this appendix are flux per unit lethargy. The PWR-U and BWR-U spectra are pin-cell-averaged; the PWR-PuU, PWR-PuPu, BWR-PuU, and BWR-PuPu spectra are averaged over the zone in the assembly-cell model containing the fuel type of interest (see Sect. 3.3.2).

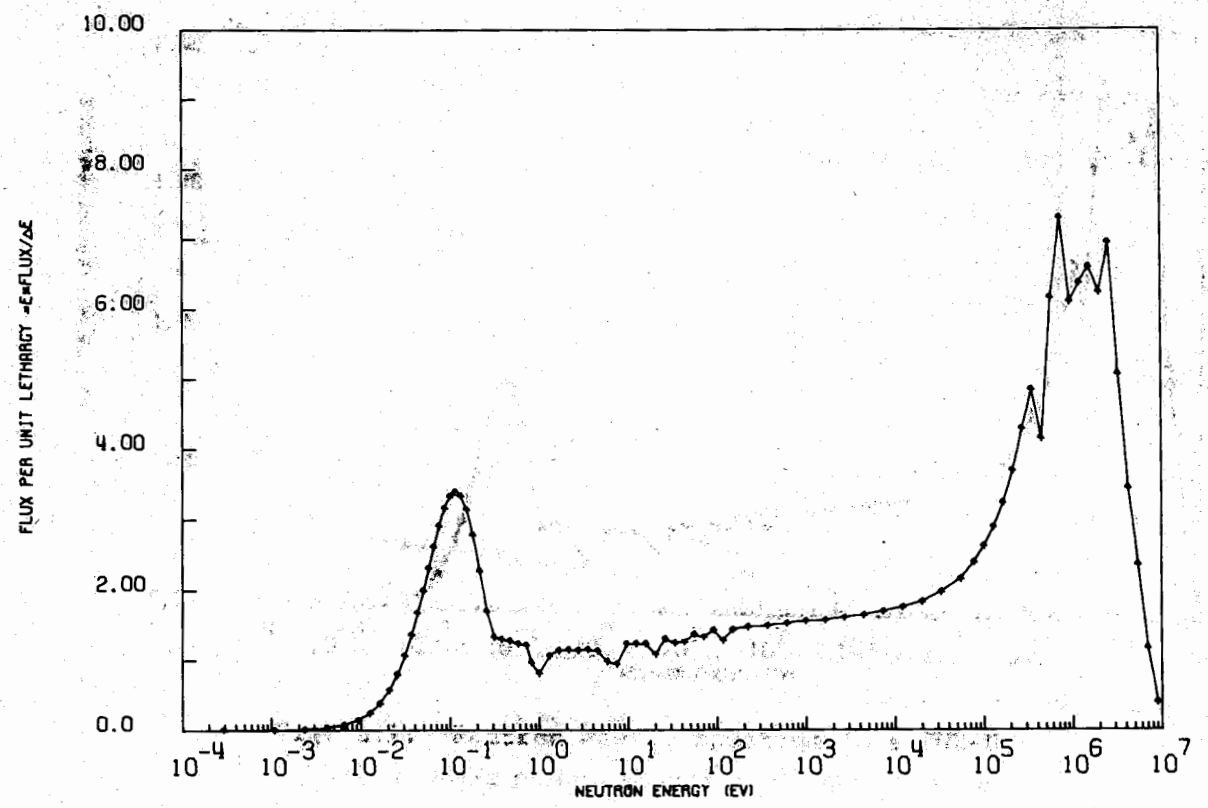

Fig. A.1. Neutron energy spectrum in a PWR-U.

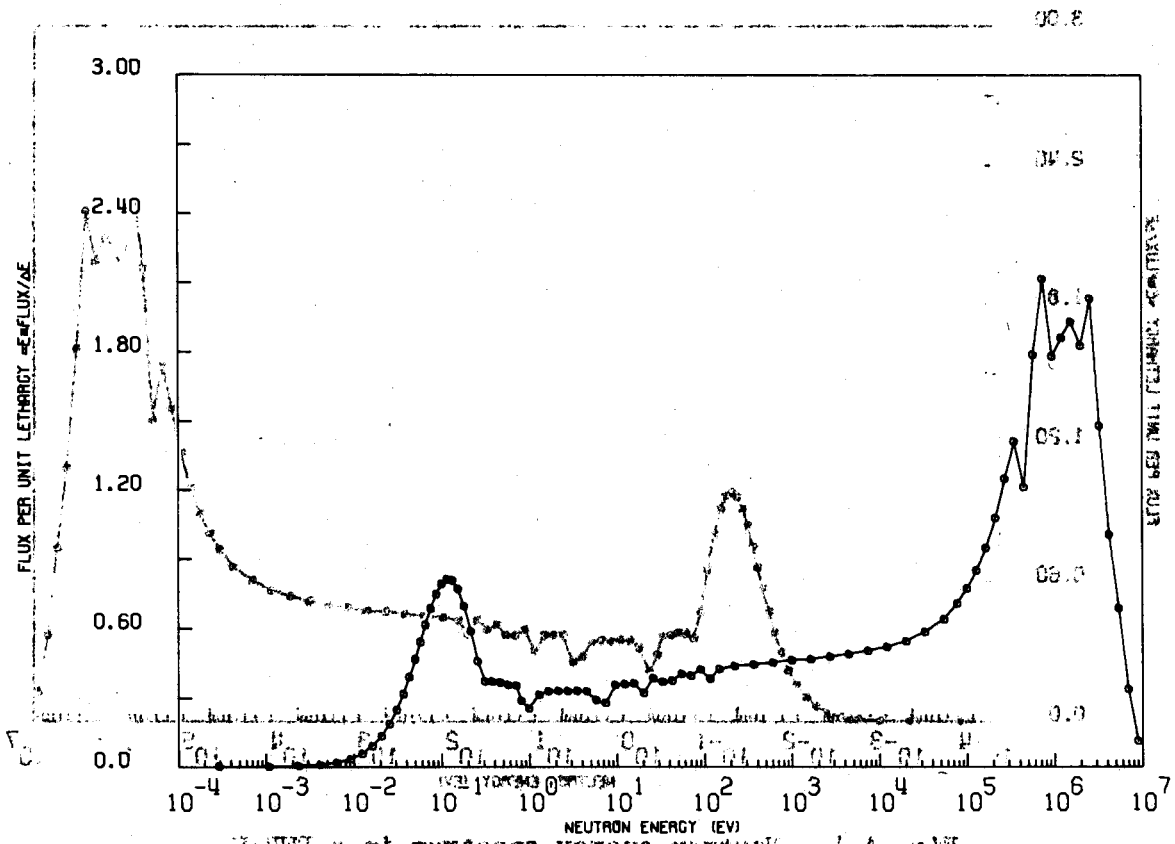

Fig. A.2. Neutron energy spectrum in a PWR-PuU.

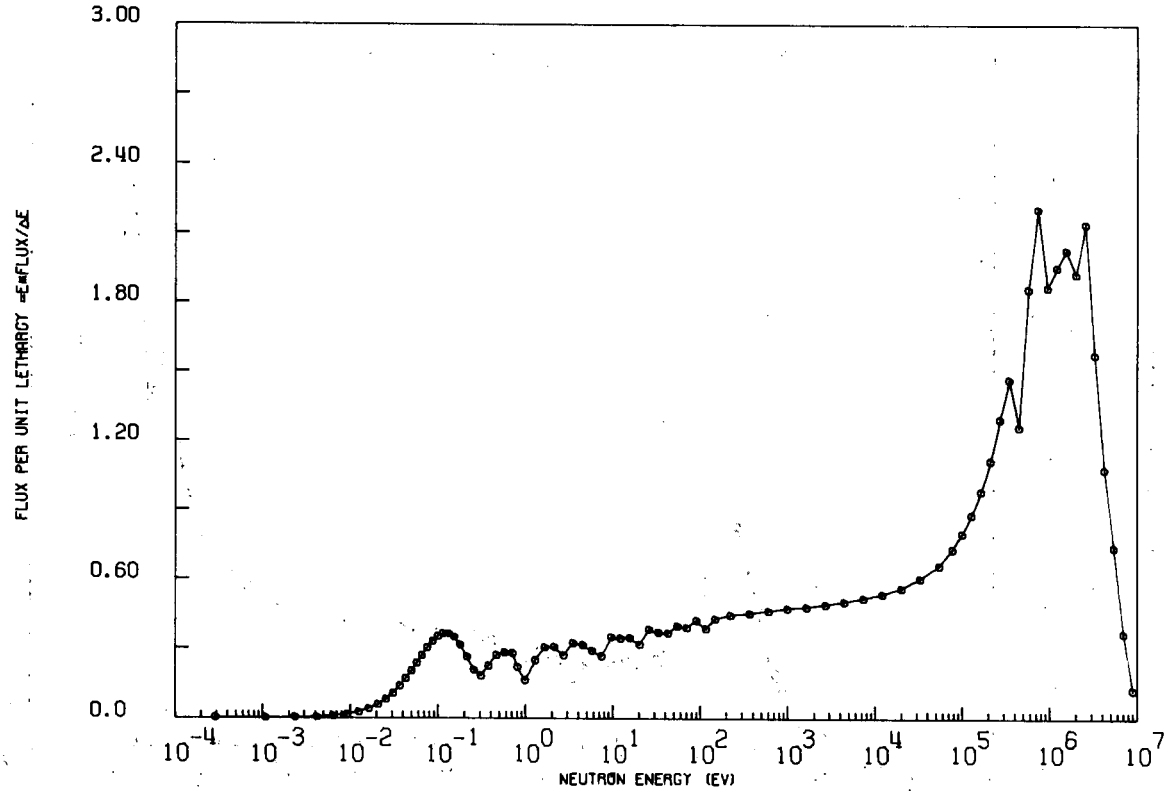

Fig. A.3. Neutron energy spectrum in a PWR-PuPu.

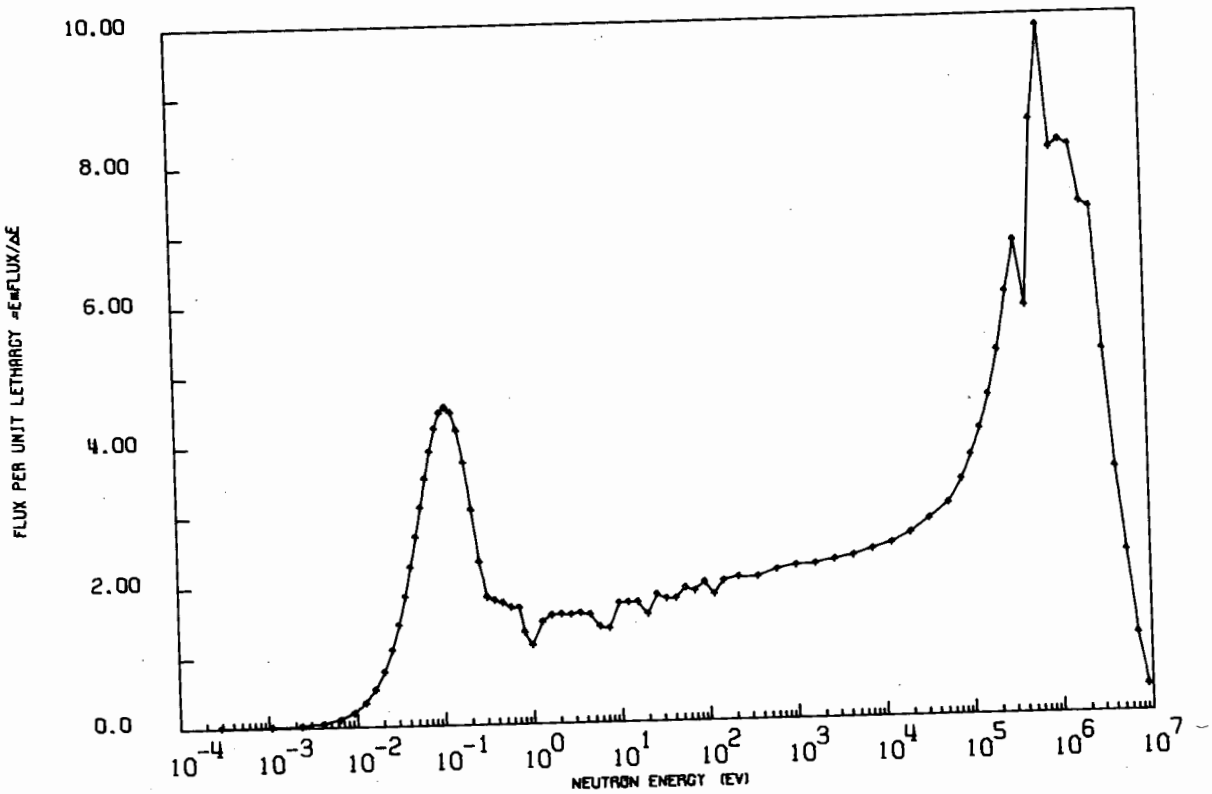

Fig. A.4. Neutron energy spectrum in a BWR-U.

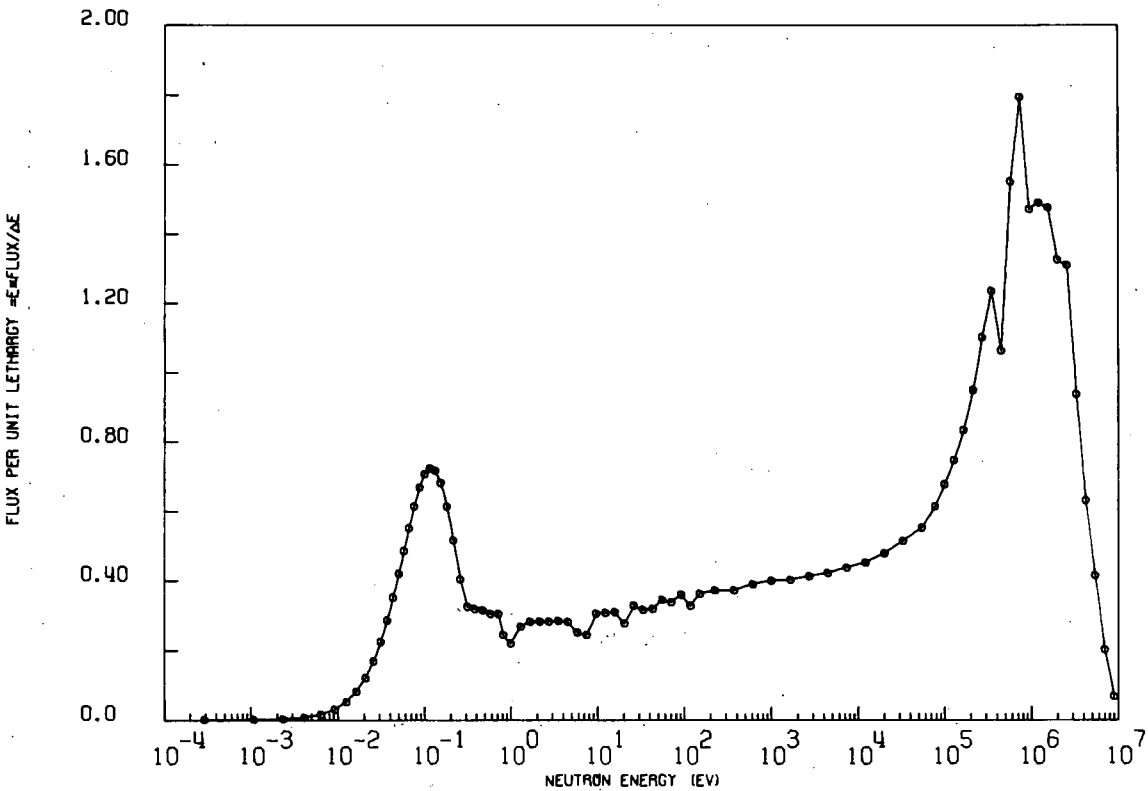

Fig. A.5. Neutron energy spectrum in a BWR-PuU.

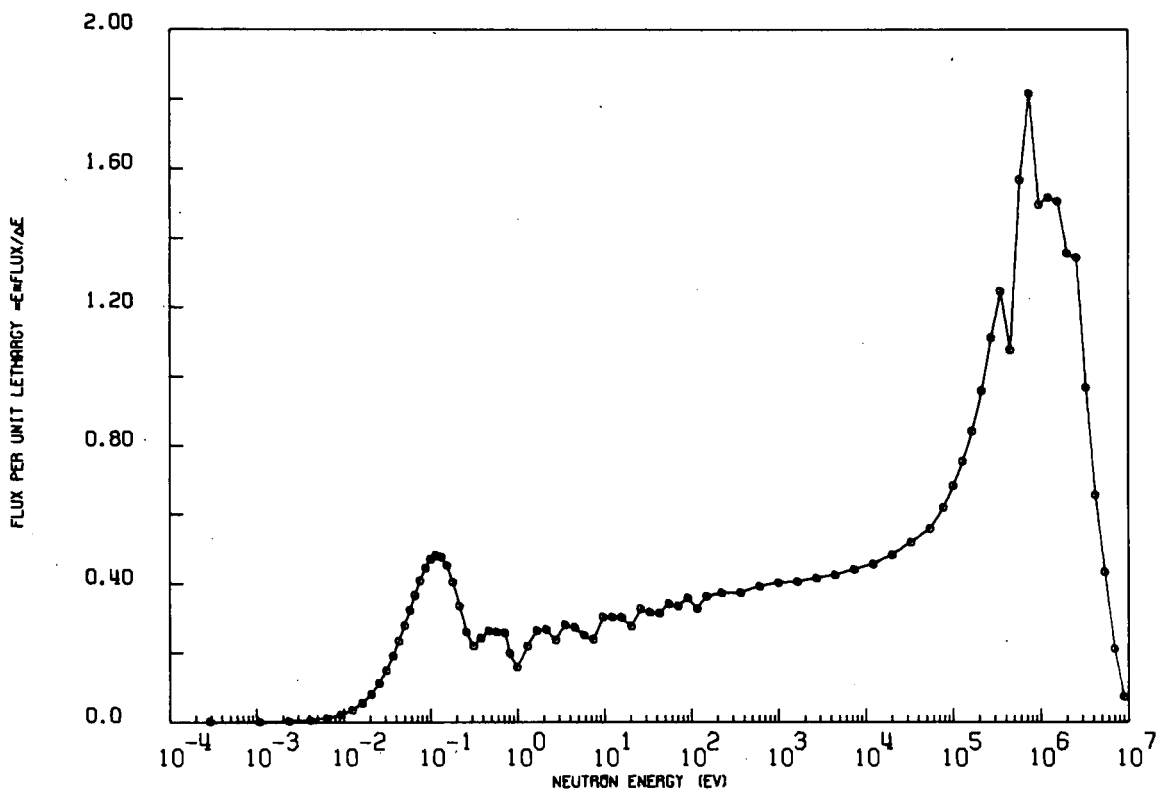

Fig. A.6. Neutron energy spectrum in a BWR-PuPu.

Table A.1. Flux per unit lethargy for U-Pu cycle PWR and BWR fuels

Energy group	Group energy boundaries, eV Upper	Lower	PWR-U	PWR-PuU	PWR-PuPu	BWR-U	BWR-PuU	BWR-PuPu
1	1.00000E 07	7.78801E 06	3.82796E-01	1.10943E-01	1.22495E-01	3.70459E-01	6.71323E-02	7.11294E-02
2	7.78801E 06	6.06531E 06	1.15302E 00	3.35652E-01	3.64866E-01	1.10690E 00	2.00858E-01	2.10620E-01
3	6.06531E 06	4.72367E 06	2.34353E 00	6.83820E-01	7.36517E-01	2.28445E 00	4.14578E-01	4.31699E-01
4	4.72367E 06	3.67879E 06	3.43144E 00	1.00238E 00	1.07581E 00	3.47874E 00	6.30739E-01	6.53983E-01
5	3.67879E 06	2.86505E 06	5.05037E 00	1.47652E 00	1.57216E 00	5.16047E 00	9.36651E-01	9.65488E-01
6	2.86505E 06	2.23130E 06	6.93200E 00	2.02802E 00	2.13877E 00	7.20083E 00	1.30980E 00	1.34210E 00
7	2.23130E 06	1.73774E 06	6.22255E 00	1.82269E 00	1.92207E 00	7.28377E 00	1.32482E 00	1.35425E 00
8	1.73774E 06	1.35335E 06	6.58550E 00	1.92728E 00	2.02675E 00	8.10350E 00	1.47541E 00	1.50398E 00
9	1.35335E 06	1.05399E 06	6.35457E 00	1.85706E 00	1.94947E 00	8.17531E 00	1.48963E 00	1.51523E 00
10	1.05399E 06	8.20850E 05	6.08772E 00	1.77512E 00	1.86187E 00	8.06481E 00	1.47009E 00	1.49309E 00
11	8.20850E 05	6.39279E 05	7.28693E 00	2.11399E 00	2.20503E 00	9.81418E 00	1.79288E 00	1.81497E 00
12	6.39279E 05	4.97871E 05	6.14762E 00	1.78378E 00	1.85579E 00	8.47161E 00	1.54942E 00	1.56584E 00
13	4.97871E 05	3.87742E 05	4.13543E 00	1.20784E 00	1.25886E 00	5.80563E 00	1.06128E 00	1.07302E 00
14	3.87742E 05	3.01974E 05	4.83536E 00	1.40783E 00	1.46398E 00	6.73930E 00	1.23310E 00	1.24528E 00
15	3.01974E 05	2.35177E 05	4.27934E 00	1.24566E 00	1.29328E 00	6.00064E 00	1.09870E 00	1.10871E 00
16	2.35177E 05	1.83156E 05	3.67692E 00	1.07368E 00	1.11347E 00	5.17088E 00	9.47254E-01	9.55416E-01
17	1.83156E 05	1.42642E 05	3.22150E 00	9.43893E-01	9.78005E-01	4.53961E 00	8.31941E-01	8.38833E-01
18	1.42642E 05	1.11090E 05	2.88100E 00	8.46394E-01	8.76241E-01	4.06617E 00	7.45422E-01	7.51353E-01
19	1.11090E 05	8.65169E 04	2.60960E 00	7.68309E-01	7.94802E-01	3.68632E 00	6.75996E-01	6.81158E-01
20	8.65169E 04	6.73794E 04	2.38376E 00	7.03218E-01	7.26969E-01	3.34013E 00	6.12670E-01	6.17205E-01
21	6.73794E 04	4.08677E 04	2.14863E 00	6.35490E-01	6.56368E-01	3.01229E 00	5.52680E-01	5.56629E-01
22	4.08677E 04	2.47875E 04	1.96486E 00	5.81470E-01	5.99825E-01	2.79939E 00	5.13864E-01	5.17198E-01
23	2.47875E 04	1.50344E 04	1.82373E 00	5.40513E-01	5.56988E-01	2.60262E 00	4.77858E-01	4.80791E-01
24	1.50344E 04	9.11882E 03	1.74240E 00	5.17009E-01	5.32219E-01	2.46052E 00	4.51896E-01	4.54509E-01
25	9.11882E 03	5.53084E 03	1.68302E 00	4.99755E-01	5.13933E-01	2.38167E 00	4.37443E-01	4.39864E-01
26	5.53084E 03	3.35463E 03	1.63157E 00	4.85273E-01	4.98489E-01	2.29575E 00	4.21680E-01	4.23934E-01
27	3.35463E 03	2.03468E 03	1.59410E 00	4.74571E-01	4.86806E-01	2.24320E 00	4.12016E-01	4.14059E-01
28	2.03468E 03	1.23410E 03	1.55846E 00	4.64330E-01	4.75740E-01	2.18903E 00	4.02013E-01	4.03928E-01
29	1.23410E 03	7.48518E 02	1.54704E 00	4.60463E-01	4.70758E-01	2.17577E 00	3.99488E-01	4.01066E-01
30	7.48518E 02	4.53999E 02	1.51053E 00	4.50002E-01	4.59076E-01	2.11954E 00	3.88941E-01	3.90352E-01
31	4.53999E 02	2.75364E 02	1.48090E 00	4.41200E-01	4.48331E-01	2.02592E 00	3.71659E-01	3.72247E-01
32	2.75364E 02	1.67017E 02	1.46626E 00	4.36053E-01	4.41618E-01	2.02705E 00	3.71398E-01	3.71778E-01
33	1.67017E 02	1.30073E 02	1.42304E 00	4.23174E-01	4.26664E-01	1.97487E 00	3.61619E-01	3.61279E-01
34	1.30073E 02	1.01301E 02	1.26999E 00	3.81249E-01	3.83537E-01	1.78573E 00	3.26925E-01	3.26430E-01
35	1.01301E 02	7.88932E 01	1.41935E 00	4.21447E-01	4.20192E-01	1.96555E 00	3.59797E-01	3.57467E-01
36	7.88932E 01	6.14421E 01	1.31845E 00	3.93753E-01	3.88502E-01	1.84309E 00	3.37645E-01	3.33539E-01
37	6.14421E 01	4.78512E 01	1.36093E 00	4.02485E-01	3.95806E-01	1.88825E 00	3.44197E-01	3.40324E-01
38	4.78512E 01	3.72665E 01	1.24571E 00	3.72209E-01	3.63927E-01	1.73952E 00	3.17935E-01	3.12933E-01
39	3.72665E 01	2.90232E 01	1.23605E 00	3.67445E-01	3.68375E-01	1.73804E 00	3.16307E-01	3.16287E-01
40	2.90232E 01	2.26033E 01	1.29933E 00	3.83031E-01	3.80909E-01	1.80008E 00	3.27447E-01	3.26083E-01
41	2.26033E 01	1.76035E 01	1.07174E 00	3.19085E-01	3.14481E-01	1.52212E 00	2.76511E-01	2.74492E-01
42	1.76035E 01	1.37096E 01	1.22767E 00	3.61973E-01	3.44570E-01	1.69862E 00	3.09337E-01	3.00993E-01
43	1.37096E 01	1.06770E 01	1.22565E 00	3.58675E-01	3.41767E-01	1.69733E 00	3.07498E-01	3.01528E-01
44	1.06770E 01	8.31529E 00	1.22118E 00	3.55327E-01	3.47201E-01	1.69039E 00	3.04769E-01	3.02415E-01
45	8.31529E 00	6.47595E 00	9.32777E-01	2.76342E-01	2.64812E-01	1.33635E 00	2.42581E-01	2.37365E-01
46	6.47595E 00	5.04348E 00	9.71215E-01	2.89749E-01	2.87674E-01	1.36359E 00	2.49926E-01	2.49009E-01
47	5.04348E 00	3.92786E 00	1.12411E 00	3.27781E-01	3.13159E-01	1.54050E 00	2.80656E-01	2.72379E-01
48	3.92786E 00	3.05902E 00	1.14280E 00	3.30246E-01	3.21761E-01	1.56312E 00	2.83531E-01	2.79253E-01
49	3.05902E 00	2.38237E 00	1.12788E 00	3.29197E-01	2.68008E-01	1.54338E 00	2.81807E-01	2.34982E-01
50	2.38237E 00	1.85539E 00	1.13741E 00	3.28861E-01	3.05389E-01	1.55455E 00	2.82057E-01	2.65635E-01
51	1.85539E 00	1.43950E 00	1.12893E 00	3.27094E-01	3.02173E-01	1.54402E 00	2.80604E-01	2.62978E-01
52	1.43950E 00	1.12535E 00	1.05086E 00	3.11473E-01	2.46480E-01	1.44788E 00	2.66876E-01	2.17789E-01
53	1.12535E 00	8.33691E-01	7.95817E-01	2.51527E-01	1.61142E-01	1.12383E 00	2.18256E-01	1.56456E-01
54	8.33691E-01	7.84930E-01	9.58259E-01	2.85595E-01	2.17545E-01	1.30843E 00	2.43213E-01	1.97988E-01
55	7.84930E-01	6.32500E-01	1.20722E 00	3.52729E-01	2.77838E-01	1.65750E 00	3.04282E-01	2.56567E-01
56	6.32500E-01	5.11187E-01	1.22111E 00	3.54315E-01	2.79585E-01	1.66831E 00	3.04099E-01	2.58350E-01
57	5.11187E-01	4.14983E-01	1.26796E 00	3.66043E-01	2.69410E-01	1.73519E 00	3.14558E-01	2.61545E-01
58	4.14983E-01	3.38908E-01	1.28950E 00	3.68577E-01	2.23081E-01	1.77226E 00	3.18467E-01	2.41321E-01
59	3.38908E-01	2.78866E-01	1.31944E 00	3.69979E-01	1.80422E-01	1.82323E 00	3.24900E-01	2.18095E-01
60	2.78866E-01	2.31511E-01	1.69604E 00	4.57546E-01	2.04469E-01	2.32186E 00	4.02996E-01	2.58583E-01
61	2.31511E-01	1.94124E-01	2.26694E 00	5.85799E-01	2.62009E-01	3.06692E 00	5.16075E-01	3.34619E-01
62	1.94124E-01	1.64513E-01	2.77332E 00	6.94546E-01	3.13720E-01	3.72949E 00	6.13907E-01	4.04385E-01
63	1.64513E-01	1.40911E-01	3.13325E 00	7.68639E-01	3.47081E-01	4.20329E 00	6.81742E-01	4.52453E-01
64	1.40911E-01	1.21896E-01	3.33073E 00	8.05793E-01	3.61916E-01	4.46468E 00	7.16792E-01	4.76993E-01
65	1.21896E-01	1.06323E-01	3.38412E 00	8.10738E-01	3.61970E-01	4.53633E 00	7.22961E-01	4.81460E-01
66	1.06323E-01	9.22968E-02	3.32074E 00	7.89690E-01	3.50404E-01	4.45566E 00	7.06102E-01	4.70065E-01
67	9.22968E-02	8.01604E-02	3.14962E 00	7.44691E-01	3.28524E-01	4.23039E 00	6.67329E-01	4.43930E-01
68	8.01604E-02	6.97166E-02	2.90174E 00	6.83057E-01	2.99720E-01	3.90225E 00	6.13300E-01	4.07560E-01
69	6.97166E-02	6.07832E-02	2.60873E 00	6.12009E-01	2.67260E-01	3.51312E 00	5.50491E-01	3.65407E-01
70	6.07832E-02	5.31932E-02	2.30039E 00	5.38301E-01	2.34079E-01	3.10256E 00	4.84984E-01	3.21534E-01
71	5.31932E-02	4.61091E-02	1.98643E 00	4.63911E-01	2.00932E-01	2.68424E 00	4.18752E-01	2.77251E-01
72	4.61091E-02	3.95312E-02	1.66503E 00	3.88271E-01	1.67550E-01	2.25493E 00	3.51163E-01	2.32173E-01
73	3.95312E-02	3.34592E-02	1.35007E 00	3.14541E-01	1.35274E-01	1.83315E 00	2.85060E-01	1.88196E-01
74	3.34592E-02	2.78932E-02	1.05486E 00	2.45694E-01	1.05365E-01	1.43670E 00	2.23145E-01	1.47104E-01
75	2.78932E-02	2.28332E-02	7.89757E-01	1.84036E-01	7.87502E-02	1.07961E 00	1.67526E-01	1.10296E-01
76	2.28332E-02	1.82792E-02	5.62542E-01	1.31272E-01	5.60948E-02	7.72487E-01	1.19790E-01	7.87793E-02
77	1.82792E-02	1.42312E-02	3.77569E-01	8.83337E-02	3.77242E-02	5.21375E-01	8.08241E-02	5.31044E-02
78	1.42312E-02	1.06892E-02	2.35558E-01	5.53244E-02	2.36402E-02	3.27552E-01	5.07706E-02	3.33428E-02
79	1.06892E-02	7.65324E-03	1.33883E-01	3.16248E-02	1.35408E-02	1.87843E-01	2.91255E-02	1.91290E-02
80	7.65324E-03	5.12325E-03	6.72343E-02	1.60117E-02	6.88518E-03	9.54525E-02	1.48098E-02	9.73606E-03
81	5.12325E-03	3.09925E-03	2.83372E-02	6.82827E-03	2.95939E-03	4.08775E-02	6.35111E-03	4.18502E-03
82	3.09925E-03	1.58125E-03	9.11039E-03	2.23372E-03	9.82468E-04	1.34444E-02	2.09392E-03	1.38653E-03
83	1.58125E-03	5.69249E-04	1.81027E-03	4.55339E-04	2.06436E-04	2.76043E-03	4.32011E-04	2.89148E-04
84	5.69249E-04	1.00000E-05	1.46351E-04	3.84344E-05	1.85729E-05	2.34397E-04	3.71398E-05	2.54548E-05

APPENDIX B:

ONE-GROUP, SPECTRUM-AVERAGED CROSS SECTIONS FOR U-Pu CYCLE PWR AND BWR FUELS

Table B.1. One-group, spectrum-averaged cross sections for U-Pu cycle light-water reactors

Isotope	Cross section type	Cross section, barns					
		PWR-U	PWR-PuU	PWR-PuPu	BWR-U	BWR-PuU	BWR-PuPu
H - 1	N,G	3.47E-02	2.89E-02	1.61E-02	3.42E-02	3.09E-02	2.31E-02
B - 10	N,A	4.02E 02	3.35E 02	1.86E 02	3.64E 02	3.28E 02	2.36E 02
B - 11	N,G	5.35E-04	4.48E-04	2.56E-04	4.87E-04	4.41E-04	3.21E-04
C - 12	N,G	3.11E-04	2.74E-04	1.44E-04	3.21E-04	2.89E-04	2.07E-04
N - 14	N,P	1.81E-01	1.62E-01	9.34E-02	1.85E-01	1.69E-01	1.26E-01
O - 16	N,A	2.72E-03	2.75E-03	3.13E-03	2.04E-03	2.06E-03	2.25E-03
NA- 23	N,G	5.28E-02	4.73E-02	2.70E-02	5.47E-02	4.98E-02	3.71E-02
CR- 52	N,G	3.35E-01	2.81E-01	1.62E-01	3.06E-01	2.77E-01	2.03E-01
MN- 55	N,G	1.52E 00	1.39E 00	9.04E-01	1.57E 00	1.46E 00	1.15E 00
FE- 56	N,G	2.68E-01	2.29E-01	1.31E-01	2.73E-01	2.48E-01	1.88E-01
NI- 58	N,G	4.87E-01	4.07E-01	2.31E-01	4.43E-01	4.01E-01	2.91E-01
GE- 72	N,G	1.23E-01	1.17E-01	8.23E-02	1.23E-01	1.24E-01	1.02E-01
GE- 73	N,G	3.27E 00	3.30E 00	2.85E 00	3.33E 00	3.44E 00	3.18E 00
GE- 74	N,G	5.51E-02	5.30E-02	4.00E-02	5.56E-02	5.59E-02	4.79E-02
GE- 76	N,G	5.65E-02	5.76E-02	5.57E-02	5.86E-02	6.02E-02	5.92E-02
AS- 75	N,G	2.15E 00	2.24E 00	2.16E 00	2.21E 00	2.31E 00	2.29E 00
SE- 76	N,G	7.88E 00	7.33E 00	4.05E 00	7.81E 00	7.84E 00	5.81E 00
SE- 77	N,G	4.38E 00	4.14E 00	2.55E 00	4.38E 00	4.41E 00	3.43E 00
SE- 78	N,G	1.85E-01	1.90E-01	1.86E-01	1.90E-01	1.95E-01	1.93E-01
SE- 80	N,G	8.99E-02	8.69E-02	6.63E-02	9.10E-02	9.18E-02	7.91E-02
SE- 82	N,G	8.62E-03	8.35E-03	7.04E-03	8.71E-03	8.69E-03	7.88E-03
BR- 79	N,G	4.80E 00	4.97E 00	4.78E 00	4.96E 00	5.16E 00	5.09E 00
BR- 81	N,G	1.79E 00	1.85E 00	1.85E 00	1.85E 00	1.91E 00	1.93E 00
KR- 80	N,G	2.82E 00	2.87E 00	2.42E 00	2.86E 00	3.00E 00	2.74E 00
KR- 82	N,G	7.25E 00	7.44E 00	6.43E 00	7.36E 00	7.75E 00	7.18E 00
KR- 83	N,G	2.13E 01	2.01E 01	1.25E 01	2.13E 01	2.14E 01	1.68E 01
KR- 84	N,G	1.25E-01	1.30E-01	1.37E-01	1.31E-01	1.35E-01	1.40E-01
KR- 85	N,G	1.84E-01	1.74E-01	1.13E-01	1.84E-01	1.86E-01	1.48E-01
KR- 86	N,G	1.01E-02	9.85E-03	7.87E-03	1.03E-02	1.04E-02	9.18E-03
RB- 85	N,G	2.52E-01	2.57E-01	2.57E-01	2.62E-01	2.68E-01	2.69E-01
RB- 86	N,G	1.14E 00	1.14E 00	1.01E 00	1.16E 00	1.19E 00	1.11E 00
RB- 87	N,G	7.69E-02	7.94E-02	7.97E-02	7.88E-02	8.12E-02	8.18E-02
SR- 86	N,G	3.91E-01	3.79E-01	2.80E-01	3.96E-01	4.01E-01	3.40E-01
SR- 87	N,G	4.30E 00	4.25E 00	3.65E 00	4.37E 00	4.44E 00	4.10E 00
SR- 88	N,G	1.24E-03	1.20E-03	1.05E-03	1.23E-03	1.21E-03	1.11E-03
SR- 89	N,G	5.26E-02	5.03E-02	3.53E-02	5.28E-02	5.31E-02	4.39E-02
SR- 90	N,G	8.74E-02	8.15E-02	4.72E-02	8.68E-02	8.71E-02	6.58E-02
Y - 89	N,G	1.33E-01	1.25E-01	7.72E-02	1.33E-01	1.33E-01	1.04E-01
Y - 90	N,G	4.33E-01	4.15E-01	2.89E-01	4.36E-01	4.40E-01	3.63E-01
Y - 91	N,G	1.65E-01	1.57E-01	1.06E-01	1.66E-01	1.67E-01	1.36E-01
ZR- 90	N,G	2.51E-02	2.47E-02	2.23E-02	2.54E-02	2.55E-02	2.40E-02
ZR- 91	N,G	2.75E-01	2.77E-01	2.52E-01	2.81E-01	2.88E-01	2.73E-01
ZR- 92	N,G	5.43E-02	5.36E-02	4.62E-02	5.56E-02	5.62E-02	5.18E-02
ZR- 92	N,G	5.80E-02	5.68E-02	5.19E-02	6.10E-02	5.98E-02	5.70E-02
ZR- 93	N,G	1.03E 00	1.07E 00	1.03E 00	1.06E 00	1.11E 00	1.09E 00
ZR- 94	N,G	1.94E-02	1.94E-02	1.85E-02	2.01E-02	2.03E-02	1.98E-02
ZR- 95	N,G	2.32E-01	2.37E-01	2.33E-01	2.40E-01	2.46E-01	2.45E-01
ZR- 96	N,G	1.77E-01	1.85E-01	1.98E-01	1.81E-01	1.87E-01	1.96E-01
NB- 93	N,G	4.10E-01	4.17E-01	3.96E-01	4.23E-01	4.35E-01	4.24E-01

Table B.1 (continued)

Isotope	Cross section type	Cross section, barns					
		PWR-U	PWR-PuU	PWR-PuPu	BWR-U	BWR-PuU	BWR-PuPu
NB- 93	N,G	4.32E-01	4.23E-01	4.03E-01	4.32E-01	4.28E-01	4.17E-01
NB- 94	N,G	4.28E 00	4.33E 00	3.86E 00	4.39E 00	4.51E 00	4.28E 00
NB- 95	N,G	8.51E-01	8.75E-01	8.72E-01	8.79E-01	9.06E-01	9.08E-01
MO- 94	N,G	3.95E-02	4.06E-02	4.33E-02	4.11E-02	4.18E-02	4.36E-02
MO- 95	N,G	4.22E 00	4.36E 00	3.91E 00	4.30E 00	4.53E 00	4.29E 00
MO- 96	N,G	6.87E-01	7.12E-01	7.13E-01	7.10E-01	7.35E-01	7.41E-01
MO- 97	N,G	6.93E-01	7.05E-01	6.55E-01	7.14E-01	7.35E-01	7.08E-01
MO- 98	N,G	2.37E-01	2.46E-01	2.56E-01	2.47E-01	2.54E-01	2.62E-01
MO- 98	N,G	1.14E 00	1.12E 00	1.06E 00	1.12E 00	1.11E 00	1.08E 00
MO- 99	N,G	1.01E 00	1.05E 00	1.04E 00	1.05E 00	1.08E 00	1.09E 00
MO-100	N,G	1.48E-01	1.53E-01	1.55E-01	1.53E-01	1.57E-01	1.60E-01
TC- 99	N,G	9.14E 00	9.83E 00	9.45E 00	9.30E 00	1.03E 01	1.01E 01
RU- 99	N,G	4.31E 00	4.45E 00	4.44E 00	4.46E 00	4.60E 00	4.65E 00
RU-100	N,G	7.80E-01	7.53E-01	5.48E-01	7.87E-01	7.96E-01	6.71E-01
RU-101	N,G	2.95E 00	3.08E 00	3.04E 00	3.05E 00	3.18E 00	3.19E 00
RU-102	N,G	2.62E-01	2.58E-01	2.20E-01	2.66E-01	2.69E-01	2.46E-01
RU-103	N,G	2.67E 00	2.74E 00	2.55E 00	2.74E 00	2.85E 00	2.75E 00
RU-104	N,G	2.62E-01	2.69E-01	2.69E-01	2.71E-01	2.78E-01	2.80E-01
RU-105	N,G	2.91E-01	3.00E-01	3.15E-01	3.05E-01	3.12E-01	3.22E-01
RU-106	N,G	8.90E-02	9.10E-02	9.19E-02	9.29E-02	9.51E-02	9.59E-02
RH-103	N,G	3.68E 01	3.75E 01	2.69E 01	3.71E 01	3.93E 01	3.17E 01
RH-105	N,G	1.02E 03	9.80E 02	5.97E 02	1.01E 03	1.04E 03	7.98E 02
PD-104	N,G	6.46E-01	6.71E-01	7.02E-01	6.70E-01	6.89E-01	7.12E-01
PD-105	N,G	3.83E 00	3.85E 00	3.42E 00	3.92E 00	4.02E 00	3.79E 00
PD-106	N,G	2.75E-01	2.84E-01	2.95E-01	2.83E-01	2.91E-01	2.99E-01
PD-107	N,G	2.82E 00	2.87E 00	2.59E 00	2.88E 00	2.99E 00	2.84E 00
PD-108	N,G	7.09E 00	7.42E 00	7.29E 00	7.35E 00	7.70E 00	7.70E 00
PD-110	N,G	2.59E-01	2.69E-01	2.80E-01	2.70E-01	2.78E-01	2.86E-01
AG-107	N,G	6.25E 00	6.16E 00	4.83E 00	6.33E 00	6.47E 00	5.68E 00
AG-109	N,G	3.89E 01	4.12E 01	3.88E 01	3.95E 01	4.29E 01	4.19E 01
AG-111	N,G	3.46E 00	3.61E 00	3.70E 00	3.59E 00	3.73E 00	3.81E 00
CD-108	N,G	2.64E-01	2.62E-01	2.34E-01	2.69E-01	2.71E-01	2.54E-01
CD-110	N,G	2.24E 00	2.21E 00	1.85E 00	2.28E 00	2.31E 00	2.10E 00
CD-111	N,G	3.55E 00	3.46E 00	2.60E 00	3.59E 00	3.64E 00	3.13E 00
CD-112	N,G	6.33E-01	6.41E-01	5.84E-01	6.49E-01	6.66E-01	6.34E-01
CD-113	N,G	4.04E 03	3.67E 03	1.73E 03	3.99E 03	3.93E 03	2.73E 03
CD-114	N,G	6.45E-01	6.83E-01	7.16E-01	6.68E-01	7.03E-01	7.28E-01
CD-115	N,G	8.22E 00	8.35E 00	7.46E 00	8.42E 00	8.70E 00	8.21E 00
CD-116	N,G	1.05E-01	1.07E-01	1.13E-01	1.09E-01	1.11E-01	1.14E-01
IN-113	N,G	7.03E 00	7.22E 00	6.86E 00	7.22E 00	7.47E 00	7.31E 00
IN-115	N,G	9.61E 01	9.78E 01	8.21E 01	9.79E 01	1.02E 02	9.02E 01
SN-115	N,G	4.55E 00	4.23E 00	2.29E 00	4.51E 00	4.53E 00	3.33E 00
SN-116	N,G	3.45E-01	3.74E-01	3.92E-01	3.57E-01	3.85E-01	4.00E-01
SN-117	N,G	7.62E-01	7.80E-01	7.09E-01	7.80E-01	8.12E-01	7.73E-01
SN-118	N,G	2.27E-01	2.36E-01	2.49E-01	2.33E-01	2.40E-01	2.50E-01
SN-119	N,G	3.02E-01	2.92E-01	2.10E-01	3.04E-01	3.09E-01	2.59E-01
SN-119	N,G	1.09E-01	1.03E-01	8.26E-02	1.15E-01	1.10E-01	9.78E-02
SN-120	N,G	5.87E-02	5.95E-02	5.79E-02	6.04E-02	6.14E-02	6.06E-02
SN-122	N,G	4.22E-02	4.22E-02	3.73E-02	4.29E-02	4.39E-02	4.10E-02
SN-123	N,G	1.08E-01	1.12E-01	1.20E-01	1.13E-01	1.15E-01	1.20E-01

Table B.1 (continued)

Isotope	Cross section type	Cross section, barns					
		PWR-U	PWR-PuU	PWR-PuPu	BWR-U	BWR-PuU	BWR-PuPu
SN-124	N,G	2.22E-01	2.35E-01	2.39E-01	2.30E-01	2.42E-01	2.47E-01
SN-125	N,G	5.44E-01	5.65E-01	5.83E-01	5.71E-01	5.88E-01	6.03E-01
SN-126	N,G	3.08E-02	2.88E-02	1.76E-02	3.06E-02	3.06E-02	2.36E-02
SB-121	N,G	5.44E 00	5.81E 00	5.75E 00	5.58E 00	6.02E 00	6.07E 00
SB-123	N,G	3.34E 00	3.64E 00	3.62E 00	3.43E 00	3.80E 00	3.83E 00
SB-124	N,G	1.37E 00	1.37E 00	1.17E 00	1.41E 00	1.44E 00	1.32E 00
SB-125	N,G	6.91E-01	7.15E-01	7.21E-01	7.15E-01	7.38E-01	7.46E-01
SB-126	N,G	1.87E 00	1.91E 00	1.76E 00	1.92E 00	1.98E 00	1.91E 00
TE-122	N,G	2.42E 00	2.55E 00	2.55E 00	2.51E 00	2.63E 00	2.65E 00
TE-123	N,G	1.73E 02	1.73E 02	1.49E 02	1.76E 02	1.80E 02	1.63E 02
TE-124	N,G	8.36E-01	7.98E-01	5.54E-01	8.39E-01	8.44E-01	6.94E-01
TE-125	N,G	9.00E-01	9.24E-01	9.20E-01	9.30E-01	9.56E-01	9.60E-01
TE-126	N,G	4.27E-01	4.35E-01	4.19E-01	4.38E-01	4.49E-01	4.42E-01
TE-127	N,G	2.05E 00	2.06E 00	1.77E 00	2.09E 00	2.15E 00	1.98E 00
TE-128	N,G	1.15E-01	1.17E-01	1.16E-01	1.18E-01	1.19E-01	1.20E-01
TE-129	N,G	2.90E-01	2.91E-01	2.64E-01	2.98E-01	3.04E-01	2.89E-01
TE-130	N,G	3.58E-02	3.41E-02	2.38E-02	3.60E-02	3.62E-02	2.98E-02
TE-132	N,G	4.89E-04	4.71E-04	4.26E-04	4.75E-04	4.65E-04	4.36E-04
I -127	N,G	4.85E 00	5.11E 00	5.10E 00	5.01E 00	5.29E 00	5.33E 00
I -129	N,G	3.22E 00	3.09E 00	2.10E 00	3.24E 00	3.27E 00	2.67E 00
I -130	N,G	6.66E 00	6.85E 00	6.44E 00	6.85E 00	7.11E 00	6.92E 00
I -131	N,G	3.23E-01	3.31E-01	3.24E-01	3.34E-01	3.43E-01	3.41E-01
I -135	N,G	2.12E-03	1.99E-03	1.24E-03	2.09E-03	2.10E-03	1.63E-03
XE-128	N,G	6.54E-01	6.46E-01	5.36E-01	6.64E-01	6.76E-01	6.10E-01
XE-129	N,G	8.45E 00	8.58E 00	8.14E 00	8.69E 00	8.90E 00	8.76E 00
XE-130	N,G	6.26E-01	5.87E-01	3.52E-01	6.24E-01	6.26E-01	4.81E-01
XE-131	N,G	3.05E 01	3.09E 01	2.75E 01	3.12E 01	3.21E 01	3.04E 01
XE-132	N,G	1.02E-01	1.02E-01	8.94E-02	1.05E-01	1.06E-01	9.92E-02
XE-133	N,G	2.44E 01	2.36E 01	1.66E 01	2.45E 01	2.50E 01	2.08E 01
XE-134	N,G	4.42E-02	4.30E-02	3.52E-02	4.49E-02	4.52E-02	4.05E-02
XE-135	N,G	2.45E 05	2.17E 05	1.00E 05	2.41E 05	2.35E 05	1.63E 05
XE-136	N,G	1.66E-02	1.56E-02	9.64E-03	1.66E-02	1.67E-02	1.30E-02
CS-133	N,G	1.07E 01	1.14E 01	1.06E 01	1.09E 01	1.19E 01	1.15E 01
CS-134	N,G	1.68E 01	1.61E 01	1.08E 01	1.68E 01	1.70E 01	1.38E 01
CS-135	N,G	2.39E 00	2.43E 00	2.16E 00	2.45E 00	2.53E 00	2.39E 00
CS-136	N,G	1.34E 00	1.40E 00	1.42E 00	1.39E 00	1.44E 00	1.47E 00
CS-137	N,G	2.56E-02	2.55E-02	2.26E-02	2.63E-02	2.68E-02	2.51E-02
BA-134	N,G	8.53E-01	8.94E-01	8.54E-01	8.76E-01	9.28E-01	9.10E-01
BA-135	N,G	3.48E 00	3.59E 00	3.54E 00	3.60E 00	3.71E 00	3.71E 00
BA-136	N,G	1.02E-01	1.02E-01	9.15E-02	1.05E-01	1.07E-01	1.01E-01
BA-137	N,G	5.51E-01	5.22E-01	3.31E-01	5.50E-01	5.55E-01	4.37E-01
BA-138	N,G	3.56E-02	3.32E-02	2.01E-02	3.52E-02	3.52E-02	2.70E-02
BA-140	N,G	5.28E-01	5.38E-01	5.04E-01	5.39E-01	5.54E-01	5.36E-01
LA-139	N,G	1.04E 00	1.00E 00	6.63E-01	1.05E 00	1.06E 00	8.55E-01
LA-140	N,G	2.21E 00	2.31E 00	2.33E 00	2.29E 00	2.38E 00	2.41E 00
CE-140	N,G	6.31E-02	5.94E-02	3.84E-02	6.27E-02	6.28E-02	4.98E-02
CE-141	N,G	2.97E 00	2.80E 00	1.70E 00	2.96E 00	2.98E 00	2.30E 00
CE-142	N,G	1.06E-01	1.00E-01	6.53E-02	1.06E-01	1.06E-01	8.47E-02
CE-143	N,G	1.75E 00	1.79E 00	1.64E 00	1.80E 00	1.86E 00	1.78E 00
CE-144	N,G	1.49E-01	1.45E-01	1.11E-01	1.51E-01	1.53E-01	1.32E-01

Table B.1 (continued)

Isotope	Cross section type	Cross section, barns					
		PWR-U	PWR-PuU	PWR-PuPu	BWR-U	BWR-PuU	BWR-PuPu
PR-141	N,G	1.51E 00	1.45E 00	1.04E 00	1.51E 00	1.53E 00	1.28E 00
PR-142	N,G	5.59E 00	5.71E 00	5.13E 00	5.72E 00	5.96E 00	5.65E 00
PR-143	N,G	1.21E 01	1.18E 01	8.61E 00	1.22E 01	1.25E 01	1.06E 01
ND-142	N,G	1.71E 00	1.59E 00	8.65E-01	1.69E 00	1.70E 00	1.25E 00
ND-143	N,G	2.87E 01	2.65E 01	1.43E 01	2.84E 01	2.84E 01	2.09E 01
ND-144	N,G	4.71E-01	4.54E-01	3.26E-01	4.73E-01	4.78E-01	3.99E-01
ND-145	N,G	9.35E 00	9.35E 00	7.86E 00	9.48E 00	9.77E 00	8.91E 00
ND-146	N,G	2.39E-01	2.33E-01	1.89E-01	2.41E-01	2.44E-01	2.16E-01
ND-147	N,G	2.01E 01	2.09E 01	1.96E 01	2.05E 01	2.18E 01	2.12E 01
ND-148	N,G	8.67E-01	8.80E-01	8.30E-01	8.86E-01	9.08E-01	8.81E-01
ND-150	N,G	6.43E-01	6.62E-01	6.50E-01	6.65E-01	6.85E-01	6.82E-01
PM-147	N,G	6.19E 01	6.61E 01	6.13E 01	6.28E 01	6.90E 01	6.69E 01
PM-148	N,G	1.17E 03	1.22E 03	9.65E 02	1.19E 03	1.27E 03	1.09E 03
PM-148	N,G	2.92E 03	2.65E 03	1.29E 03	2.89E 03	2.84E 03	1.99E 03
PM-149	N,G	1.32E 02	1.24E 02	7.00E 01	1.32E 02	1.32E 02	9.90E 01
PM-151	N,G	1.03E 02	1.01E 02	7.41E 01	1.04E 02	1.07E 02	9.07E 01
SM-147	N,G	2.39E 01	2.48E 01	2.32E 01	2.45E 01	2.58E 01	2.51E 01
SM-148	N,G	1.14E 00	1.16E 00	1.12E 00	1.17E 00	1.20E 00	1.18E 00
SM-149	N,G	7.28E 03	6.44E 03	3.02E 03	7.19E 03	6.94E 03	4.84E 03
SM-150	N,G	1.53E 01	1.54E 01	1.19E 01	1.54E 01	1.63E 01	1.42E 01
SM-151	N,G	7.26E 02	6.75E 02	3.49E 02	7.16E 02	7.30E 02	5.27E 02
SM-152	N,G	7.56E 01	8.22E 01	7.49E 01	7.67E 01	8.67E 01	8.34E 01
SM-153	N,G	8.98E 01	9.34E 01	8.24E 01	9.11E 01	9.78E 01	9.23E 01
SM-154	N,G	1.52E 00	1.52E 00	1.37E 00	1.56E 00	1.59E 00	1.50E 00
EU-151	N,G	7.41E 02	7.11E 02	4.09E 02	7.36E 02	7.58E 02	5.71E 02
EU-152	N,G	1.91E 02	1.89E 02	1.28E 02	1.92E 02	2.00E 02	1.61E 02
EU-154	N,G	1.29E 02	1.27E 02	8.85E 01	1.29E 02	1.34E 02	1.10E 02
EU-155	N,G	3.66E 02	3.39E 02	1.82E 02	3.62E 02	3.63E 02	2.66E 02
EU-156	N,G	7.38E 01	7.23E 01	5.28E 01	7.42E 01	7.62E 01	6.43E 01
EU-157	N,G	4.61E 01	4.64E 01	3.84E 01	4.67E 01	4.85E 01	4.37E 01
GD-154	N,G	1.35E 01	1.33E 01	1.04E 01	1.36E 01	1.41E 01	1.23E 01
GD-155	N,G	2.76E 03	2.48E 03	1.15E 03	2.71E 03	2.69E 03	1.88E 03
GD-156	N,G	3.77E 00	4.00E 00	4.18E 00	3.93E 00	4.13E 00	4.29E 00
GD-157	N,G	1.17E 04	1.05E 04	4.82E 03	1.15E 04	1.14E 04	7.91E 03
GD-158	N,G	1.79E 00	1.93E 00	1.92E 00	1.85E 00	2.00E 00	2.02E 00
GD-160	N,G	3.70E-01	3.77E-01	3.72E-01	3.82E-01	3.90E-01	3.88E-01
TB-159	N,G	1.48E 01	1.53E 01	1.47E 01	1.53E 01	1.58E 01	1.57E 01
TB-160	N,G	6.86E 01	6.63E 01	4.56E 01	6.88E 01	7.02E 01	5.74E 01
DY-160	N,G	4.94E 01	5.10E 01	4.95E 01	5.08E 01	5.28E 01	5.24E 01
DY-161	N,G	7.62E 01	7.35E 01	5.23E 01	7.66E 01	7.77E 01	6.47E 01
DY-162	N,G	7.45E 01	7.95E 01	7.41E 01	7.56E 01	8.29E 01	8.05E 01
DY-163	N,G	4.82E 01	4.83E 01	4.22E 01	4.91E 01	5.03E 01	4.64E 01
DY-164	N,G	1.94E 02	1.77E 02	8.64E 01	1.91E 02	1.91E 02	1.35E 02
HO-165	N,G	2.55E 01	2.59E 01	2.39E 01	2.61E 01	2.70E 01	2.60E 01
ER-166	N,G	6.74E 00	6.70E 00	5.45E 00	6.85E 00	7.01E 00	6.28E 00
ER-167	N,G	2.13E 02	2.15E 02	1.54E 02	2.15E 02	2.25E 02	1.87E 02
TH-232	N,G	3.05E 00	3.08E 00	2.93E 00	3.20E 00	3.17E 00	3.11E 00
TH-232	N,F	2.22E-02	2.24E-02	2.51E-02	1.79E-02	1.81E-02	1.96E-02
PA-233	N,G	1.23E 01	1.24E 01	1.13E 01	1.28E 01	1.28E 01	1.20E 01
PA-233	N,F	1.46E-01	1.47E-01	1.65E-01	1.19E-01	1.20E-01	1.30E-01

Table B.1 (continued)

Isotope	Cross section type	Cross section, barns					
		PWR-U	PWR-PuU	PWR-PuPu	BWR-U	BWR-PuU	BWR-PuPu
PA-233	N,GX	1.23E 01	1.24E 01	1.13E 01	1.28E 01	1.28E 01	1.20E 01
U -233	N,G	7.58E 00	7.10E 00	5.08E 00	7.84E 00	7.41E 00	6.13E 00
U -233	N,F	6.23E 01	5.71E 01	3.68E 01	6.44E 01	5.98E 01	4.70E 01
U -234	N,G	1.92E 01	1.99E 01	1.72E 01	2.09E 01	2.04E 01	1.88E 01
U -234	N,F	4.50E-01	4.42E-01	4.91E-01	3.97E-01	4.02E-01	4.30E-01
U -235	N,G	1.05E 01	9.96E 00	6.40E 00	1.12E 01	1.03E 01	8.13E 00
U -235	N,F	4.67E 01	4.32E 01	2.27E 01	5.00E 01	4.49E 01	3.20E 01
U -236	N,G	7.54E 00	8.28E 00	8.49E 00	8.25E 00	8.38E 00	8.62E 00
U -236	N,F	1.97E-01	1.94E-01	2.17E-01	1.65E-01	1.67E-01	1.79E-01
U -237	N,G	4.42E 01	3.95E 01	2.25E 01	4.58E 01	4.15E 01	3.10E 01
U -237	N,F	2.32E-01	2.14E-01	1.44E-01	2.41E-01	2.24E-01	1.81E-01
U -238	N,G	9.02E-01	9.37E-01	8.72E-01	9.19E-01	9.03E-01	8.72E-01
U -238	N,F	1.00E-01	9.88E-02	1.10E-01	8.08E-02	8.18E-02	8.81E-02
U -239	N,G	3.72E 00	3.53E 00	2.65E 00	3.87E 00	3.68E 00	3.13E 00
U -239	N,F	1.54E 00	1.39E 00	8.56E-01	1.57E 00	1.43E 00	1.10E 00
U -240	N,G	5.45E-01	5.41E-01	4.97E-01	5.68E-01	5.60E-01	5.34E-01
U -240	N,F	7.29E-02	7.36E-02	8.25E-02	5.91E-02	5.98E-02	6.45E-02
NP-237	N,G	3.21E 01	3.20E 01	2.42E 01	3.41E 01	3.30E 01	2.80E 01
NP-237	N,F	5.24E-01	5.15E-01	5.71E-01	4.61E-01	4.68E-01	4.99E-01
NP-238	N,G	1.79E 01	1.58E 01	8.42E 00	1.85E 01	1.67E 01	1.20E 01
NP-238	N,F	1.78E 02	1.57E 02	8.29E 01	1.84E 02	1.65E 02	1.19E 02
PU-236	N,G	2.04E 01	1.89E 01	1.31E 01	2.11E 01	1.97E 01	1.62E 01
PU-236	N,F	2.06E 01	1.91E 01	1.34E 01	2.14E 01	2.00E 01	1.65E 01
PU-238	N,G	3.47E 01	3.15E 01	1.53E 01	3.73E 01	3.29E 01	2.26E 01
PU-238	N,F	2.46E 00	2.38E 00	2.03E 00	2.49E 00	2.38E 00	2.16E 00
PU-239	N,G	5.86E 01	5.73E 01	2.60E 01	6.31E 01	5.92E 01	3.85E 01
PU-239	N,F	1.06E 02	1.02E 02	4.64E 01	1.14E 02	1.06E 02	6.92E 01
PU-240	N,G	1.04E 02	1.37E 02	4.39E 01	1.11E 02	1.36E 02	5.26E 01
PU-240	N,F	5.84E-01	5.84E-01	6.22E-01	5.27E-01	5.39E-01	5.54E-01
PU-241	N,G	3.87E 01	3.63E 01	1.67E 01	4.15E 01	3.78E 01	2.50E 01
PU-241	N,F	1.18E 02	1.11E 02	5.45E 01	1.26E 02	1.15E 02	7.88E 01
PU-242	N,G	2.94E 01	3.02E 01	2.39E 01	3.05E 01	3.10E 01	2.45E 01
PU-242	N,F	4.58E-01	4.50E-01	5.00E-01	3.97E-01	4.02E-01	4.31E-01
PU-243	N,G	2.59E 01	2.31E 01	1.31E 01	2.68E 01	2.43E 01	1.81E 01
PU-243	N,F	2.09E 01	1.91E 01	1.24E 01	2.17E 01	2.00E 01	1.59E 01
AM-241	N,G	9.92E 01	9.96E 01	5.76E 01	1.07E 02	1.03E 02	7.36E 01
AM-241	N,F	1.32E 00	1.33E 00	1.08E 00	1.31E 00	1.30E 00	1.12E 00
AM-241	N,GX	1.23E 01	1.23E 01	7.12E 00	1.32E 01	1.27E 01	9.10E 00
AM-242	N,G	1.89E 01	1.72E 01	8.07E 00	2.02E 01	1.80E 01	1.22E 01
AM-242	N,F	1.71E 02	1.55E 02	7.32E 01	1.83E 02	1.62E 02	1.10E 02
AM-242	N,G	1.55E 02	1.34E 02	6.41E 01	1.59E 02	1.42E 02	9.80E 01
AM-242	N,F	7.70E 02	6.71E 02	3.31E 02	7.92E 02	7.09E 02	4.96E 02
AM-243	N,G	1.90E 00	1.97E 00	1.52E 00	2.02E 00	2.02E 00	1.64E 00
AM-243	N,F	3.57E-01	3.51E-01	3.91E-01	3.04E-01	3.08E-01	3.30E-01
AM-243	N,GX	3.61E 01	3.74E 01	2.90E 01	3.84E 01	3.84E 01	3.12E 01
CM-242	N,G	5.48E 00	5.58E 00	5.05E 00	5.81E 00	5.72E 00	5.44E 00
CM-242	N,F	2.20E-01	1.99E-01	8.82E-02	2.36E-01	2.09E-01	1.38E-01
CM-243	N,G	9.92E 00	9.71E 00	7.16E 00	1.05E 01	1.00E 01	8.34E 00
CM-243	N,F	9.76E 01	9.45E 01	6.43E 01	1.04E 02	9.78E 01	7.78E 01
CM-244	N,G	3.44E 00	3.49E 00	3.17E 00	3.65E 00	3.59E 00	3.41E 00

Table B.1 (continued)

Isotope	Cross section type	Cross section, barns					
		PWR-U	PWR-PuU	PWR-PuPu	BWR-U	BWR-PuU	BWR-PuPu
CM-244	N,F	8.53E-01	8.43E-01	9.02E-01	7.79E-01	7.84E-01	8.18E-01
CM-245	N,G	1.84E 02	1.59E 02	7.48E 01	1.90E 02	1.68E 02	1.16E 02
CM-245	N,F	1.72E 02	1.51E 02	7.89E 01	1.77E 02	1.59E 02	1.14E 02
CM-246	N,G	1.32E 00	1.34E 00	1.34E 00	1.37E 00	1.38E 00	1.39E 00
CM-246	N,F	4.90E-01	4.94E-01	5.46E-01	4.27E-01	4.31E-01	4.60E-01
CM-247	N,G	1.33E 01	1.19E 01	7.03E 00	1.37E 01	1.25E 01	9.49E 00
CM-247	N,F	9.36E 00	8.66E 00	5.94E 00	9.69E 00	9.05E 00	7.38E 00
CM-248	N,G	4.02E 00	4.13E 00	4.21E 00	4.21E 00	4.24E 00	4.34E 00
CM-248	N,F	5.70E-01	5.75E-01	6.28E-01	5.26E-01	5.32E-01	5.62E-01
BK-249	N,G	1.75E 02	1.59E 02	9.75E 01	1.81E 02	1.67E 02	1.28E 02
CF-249	N,G	2.17E 02	1.98E 02	1.15E 02	2.25E 02	2.07E 02	1.55E 02
CF-249	N,F	1.74E 02	1.59E 02	9.56E 01	1.80E 02	1.67E 02	1.26E 02
CF-250	N,G	1.48E 02	1.30E 02	6.48E 01	1.53E 02	1.37E 02	9.64E 01
CF-251	N,G	6.32E 02	5.48E 02	2.58E 02	6.53E 02	5.79E 02	3.99E 02
CF-251	N,F	4.06E 02	3.54E 02	1.71E 02	4.19E 02	3.74E 02	2.60E 02
CF-252	N,G	5.11E 00	4.56E 00	2.62E 00	5.29E 00	4.80E 00	3.59E 00
CF-252	N,F	4.34E 00	4.06E 00	2.93E 00	4.51E 00	4.24E 00	3.55E 00
CF-253	N,G	9.19E 01	7.96E 01	3.77E 01	9.49E 01	8.42E 01	5.81E 01
CF-253	N,F	1.13E 02	1.01E 02	5.92E 01	1.17E 02	1.06E 02	8.01E 01
CF-254	N,G	9.20E 00	8.13E 00	4.38E 00	9.51E 00	8.57E 00	6.23E 00
ES-253	N,G	2.02E 01	1.96E 01	1.61E 01	2.10E 01	2.04E 01	1.83E 01
ES-253	N,GX	1.40E 01	1.37E 01	1.12E 01	1.46E 01	1.41E 01	1.27E 01
1/V	N,G	9.19E-02	8.12E-02	4.28E-02	9.50E-02	8.56E-02	6.15E-02

N,G = (N,GAMMA) TO A GROUND STATE
N,F = (N,FISSION)
N,GX = (N,GAMMA) TO AN EXCITED STATE
N,A = (N,ALPHA)
N,P = (N,PROTON)

APPENDIX C:

PLOTS AND LISTINGS OF SELECTED BURNUP-DEPENDENT CROSS SECTIONS FOR ORIGEN

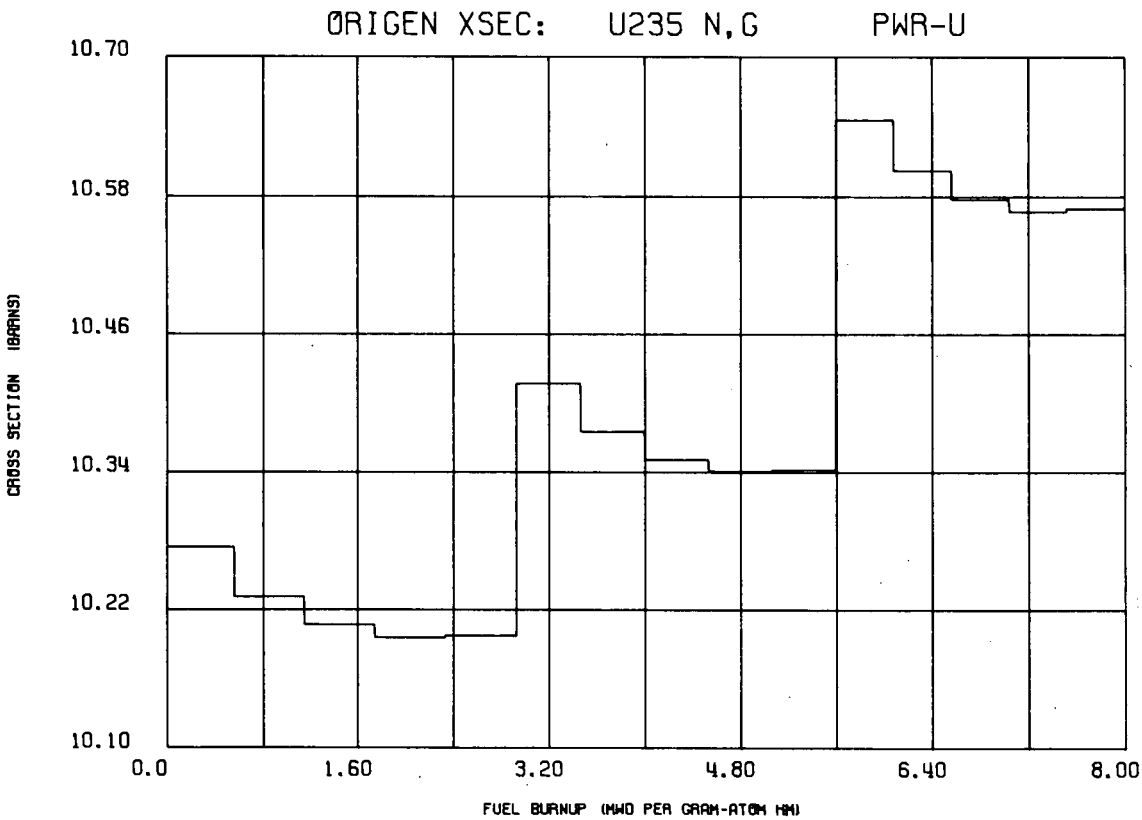

Fig. C.1. Burnup-dependent cross section of ^{235}U (n,γ) reaction for PWR-U fuel.

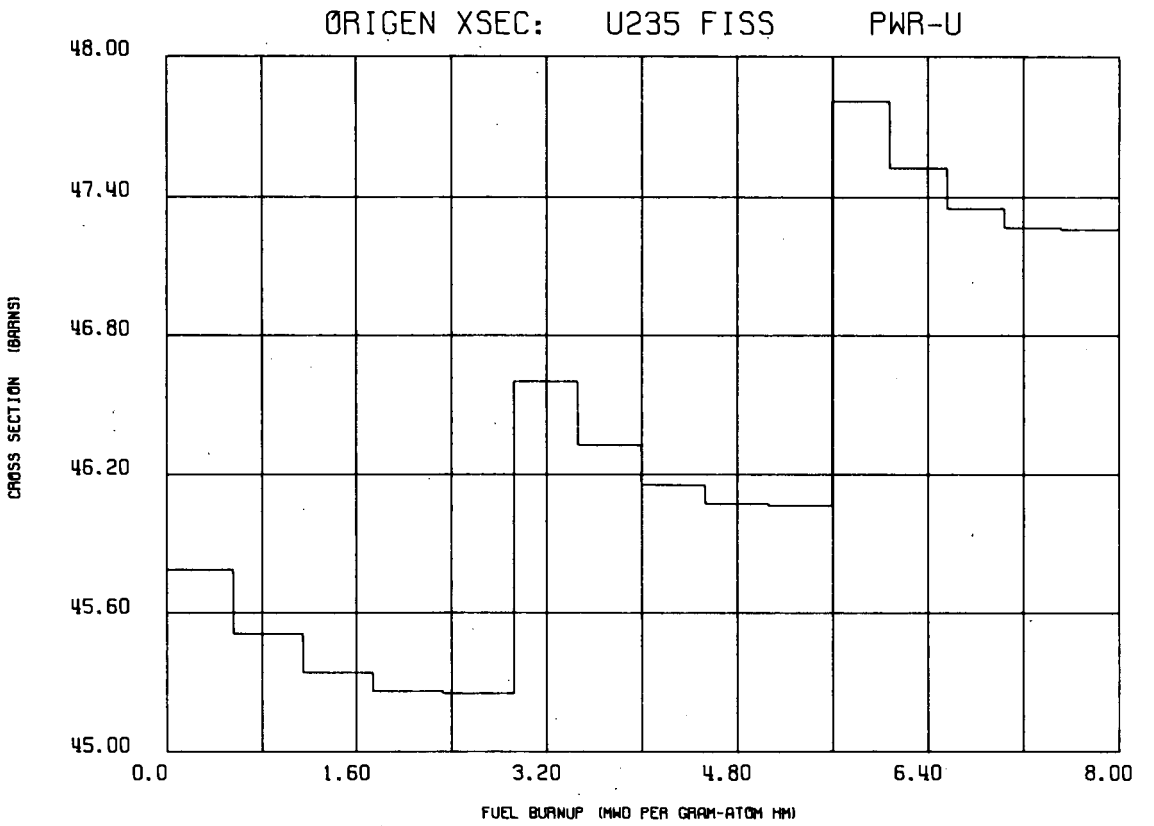

Fig. C.2. Burnup-dependent cross section of ^{235}U (n,fission) reaction for PWR-U fuel.

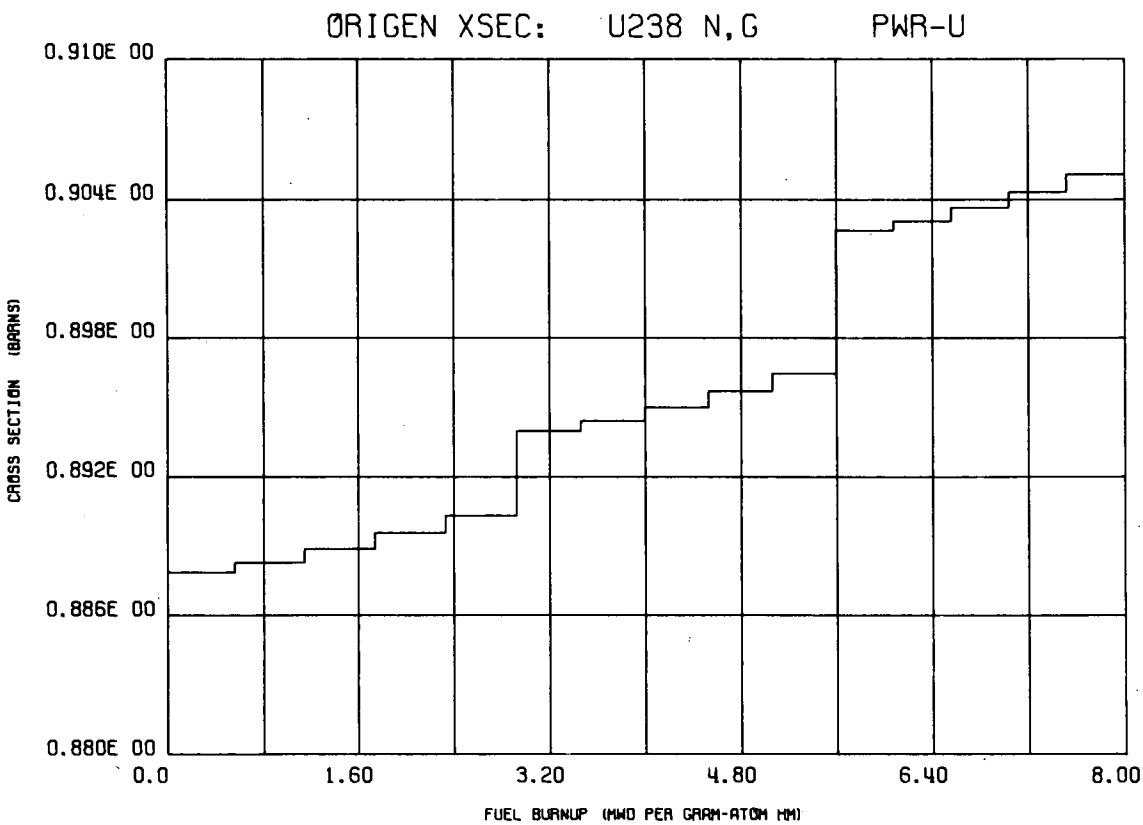

Fig. C.3. Burnup-dependent cross section of ^{238}U (n,γ) reaction for PWR-U fuel.

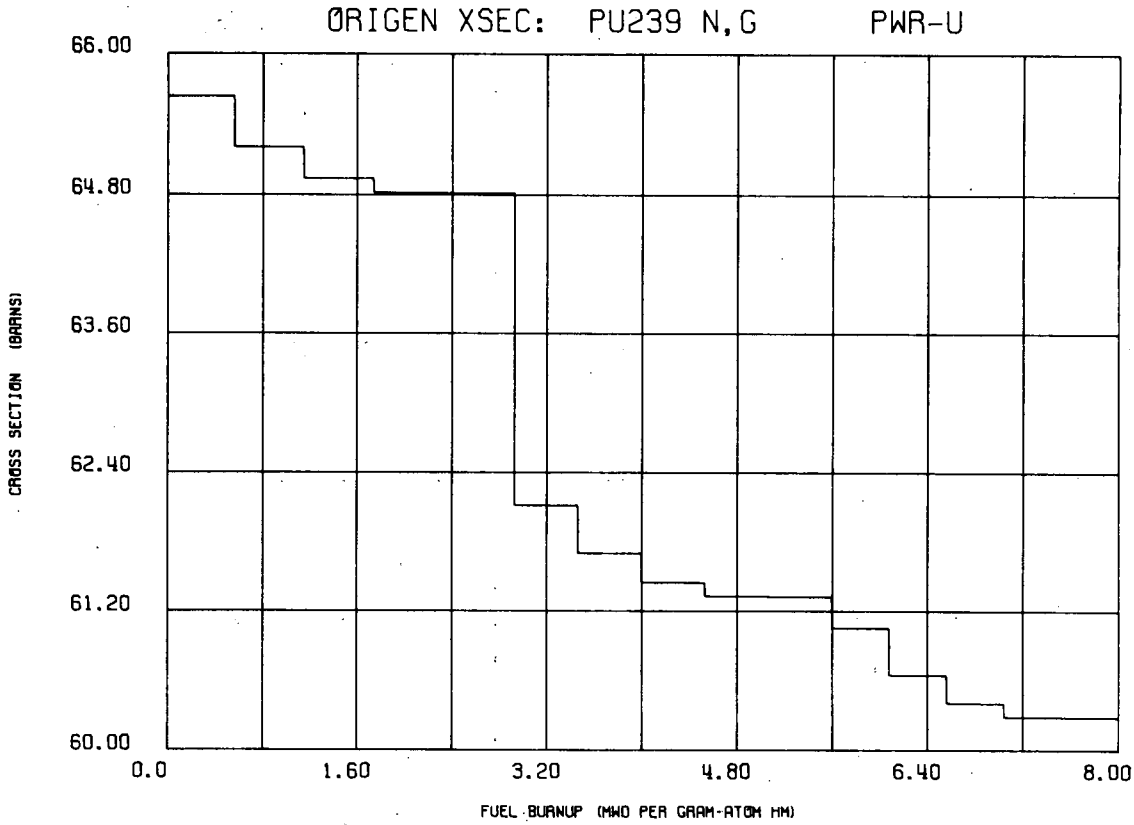

Fig. C.4. Burnup-dependent cross section of ^{239}Pu (n,γ) reaction for PWR-U fuel.

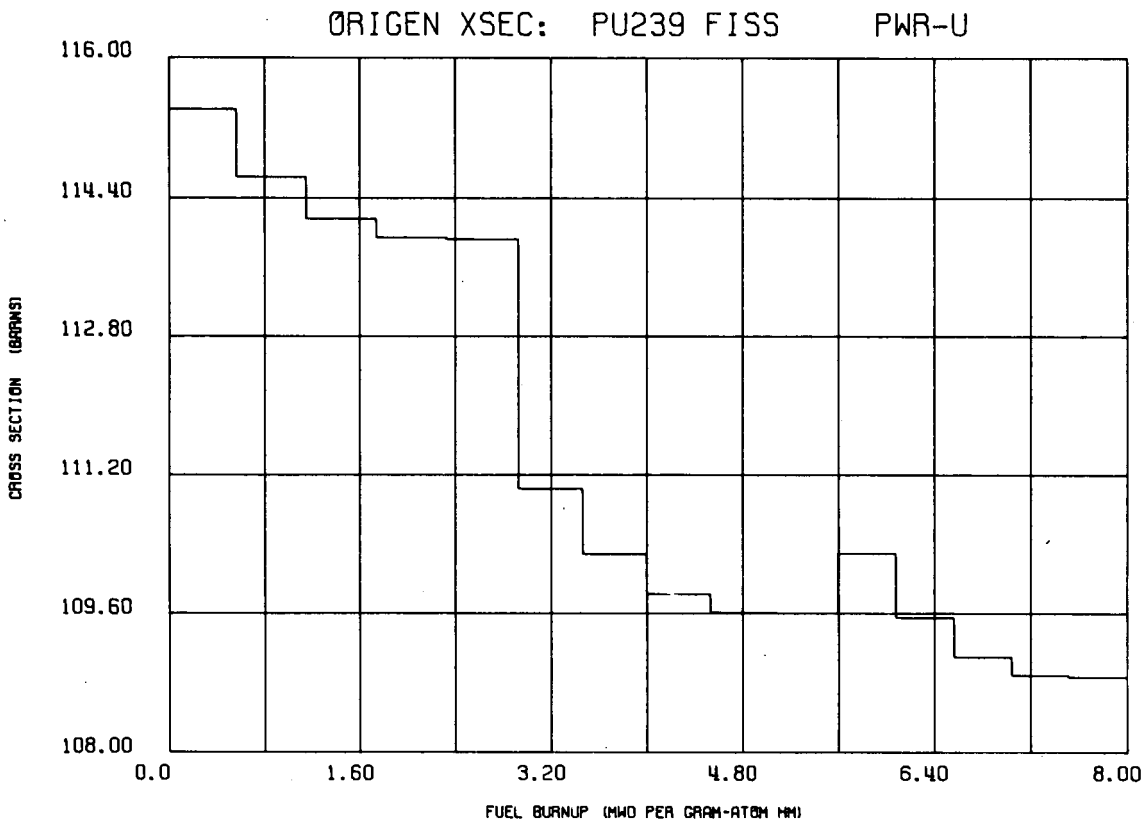

Fig. C.5. Burnup-dependent cross section of ^{239}Pu (n,fission) reaction for PWR-U fuel.

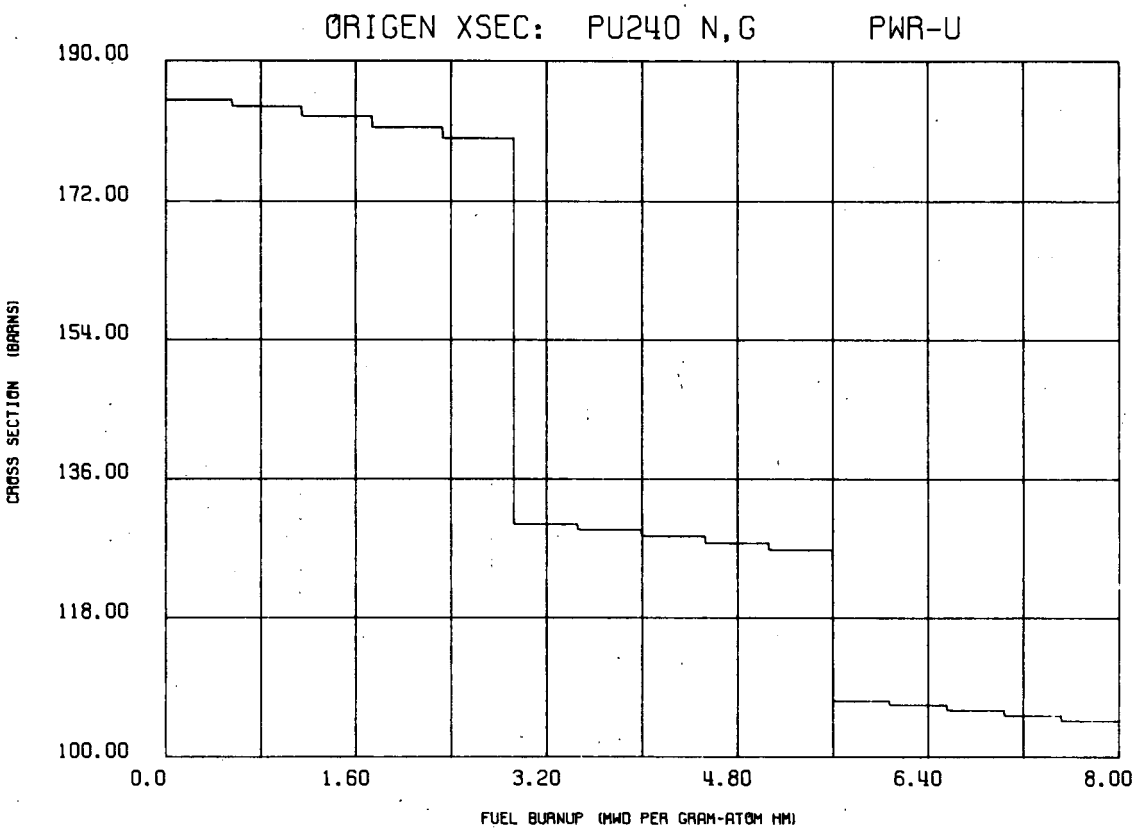

Fig. C.6. Burnup-dependent cross section of ^{240}Pu (n,γ) reaction for PWR-U fuel.

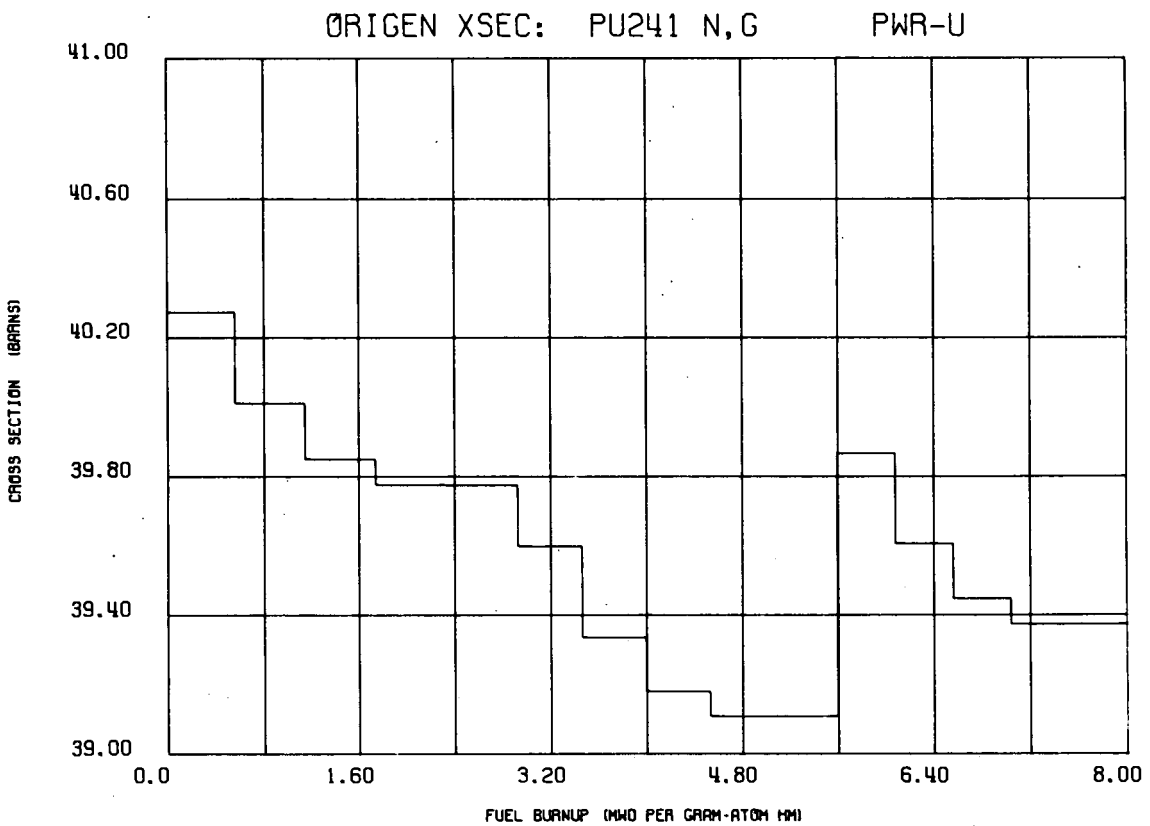

Fig. C.7. Burnup-dependent cross section of ^{241}Pu (n,γ) reaction for PWR-U fuel.

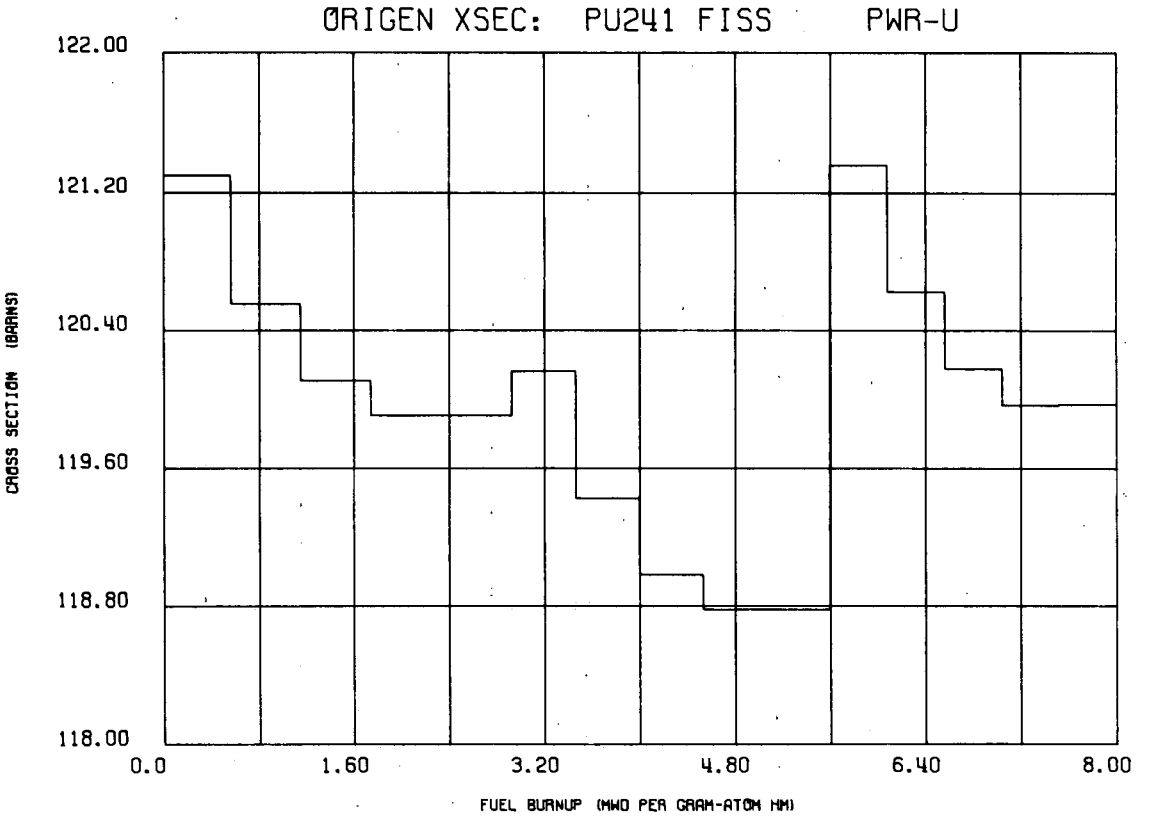

Fig. C.8. Burnup-dependent cross section of ^{241}Pu (n,fission) reaction for PWR-U fuel.

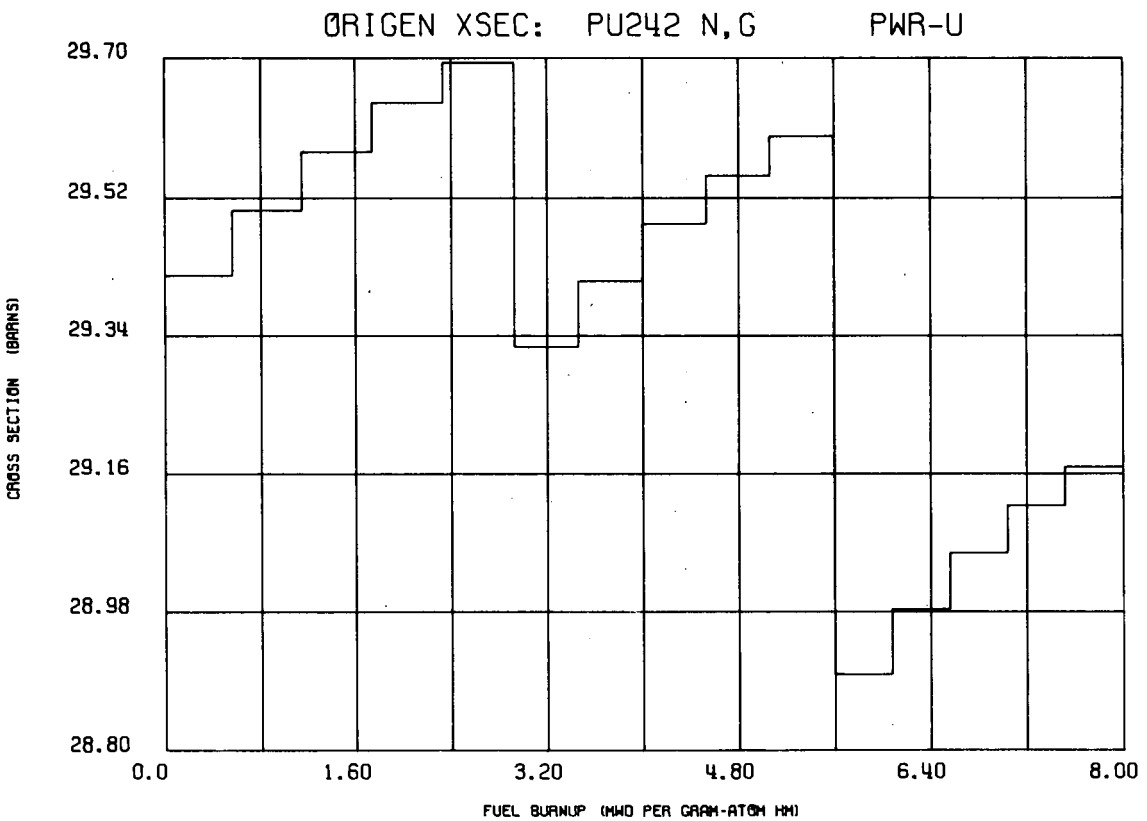

Fig. C.9. Burnup-dependent cross section of ^{242}Pu (n,γ) reaction for PWR-U fuel.

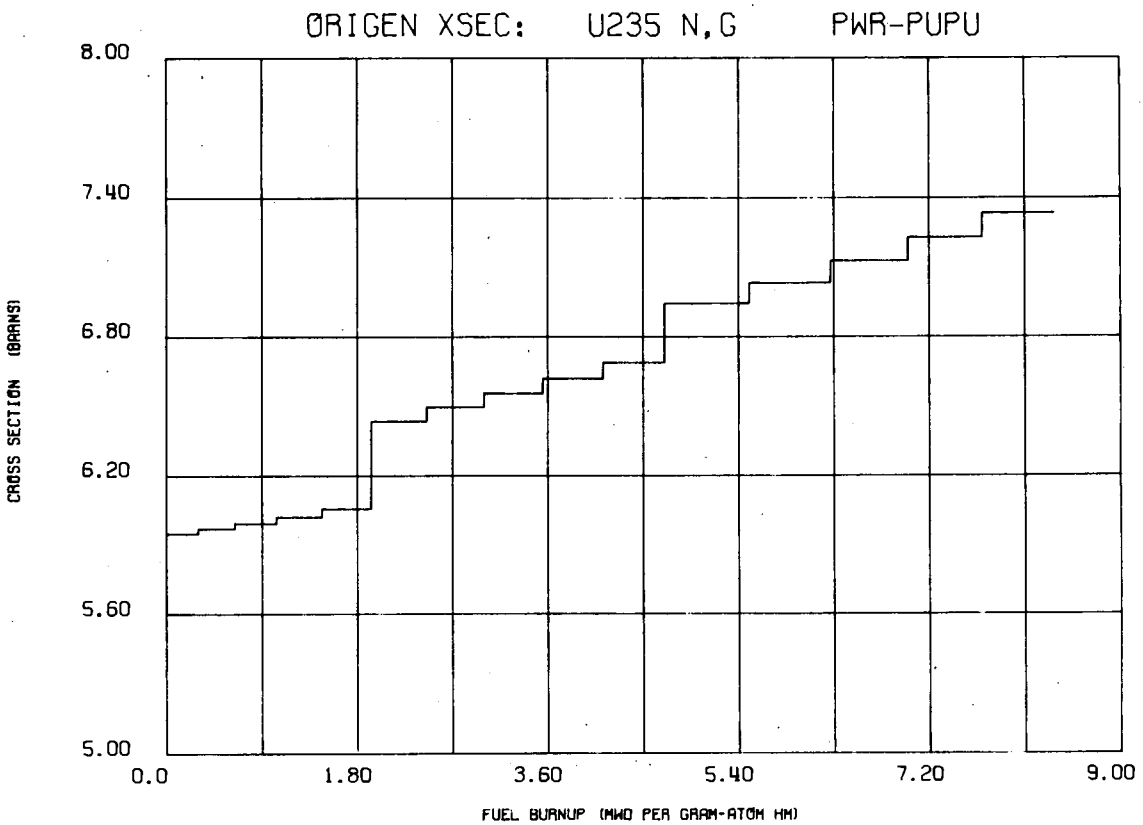

Fig. C.10. Burnup-dependent cross section of ^{235}U (n,γ) reaction for PWR-PuPu fuel.

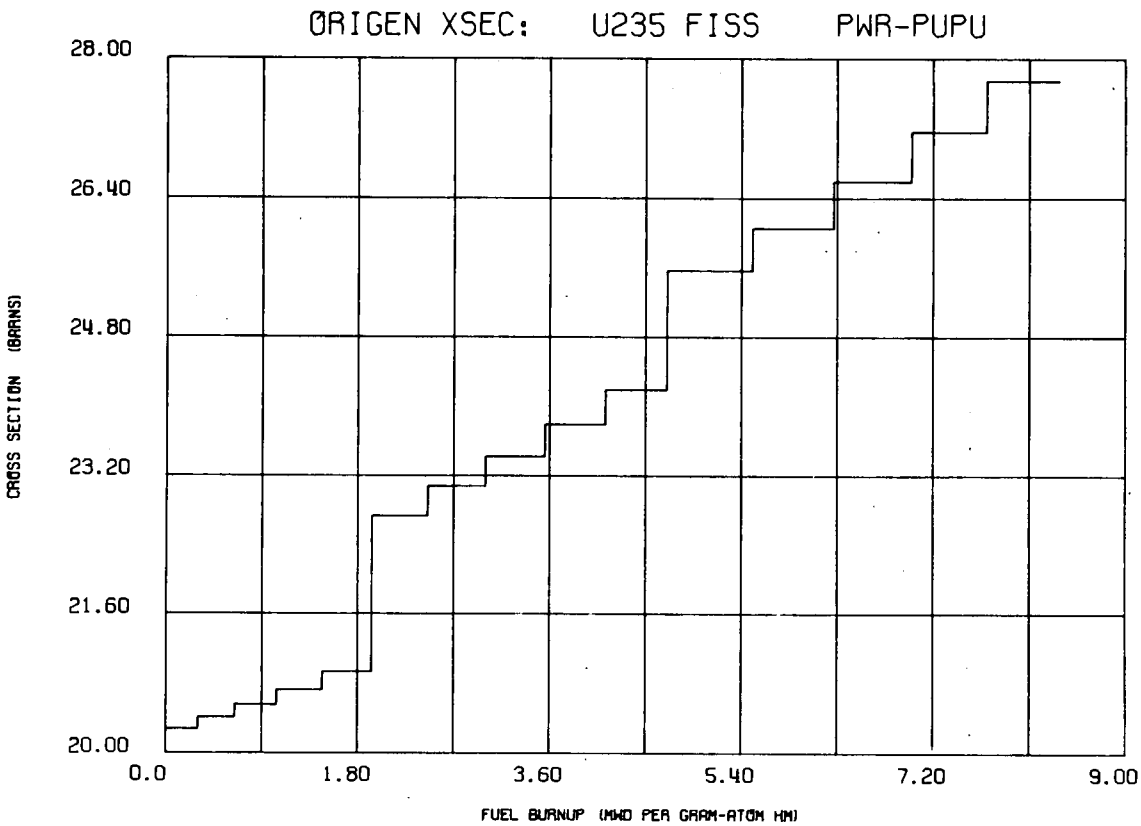

Fig. C.11. Burnup-dependent cross section of ^{235}U (n,fission) reaction for PWR-PuPu fuel.

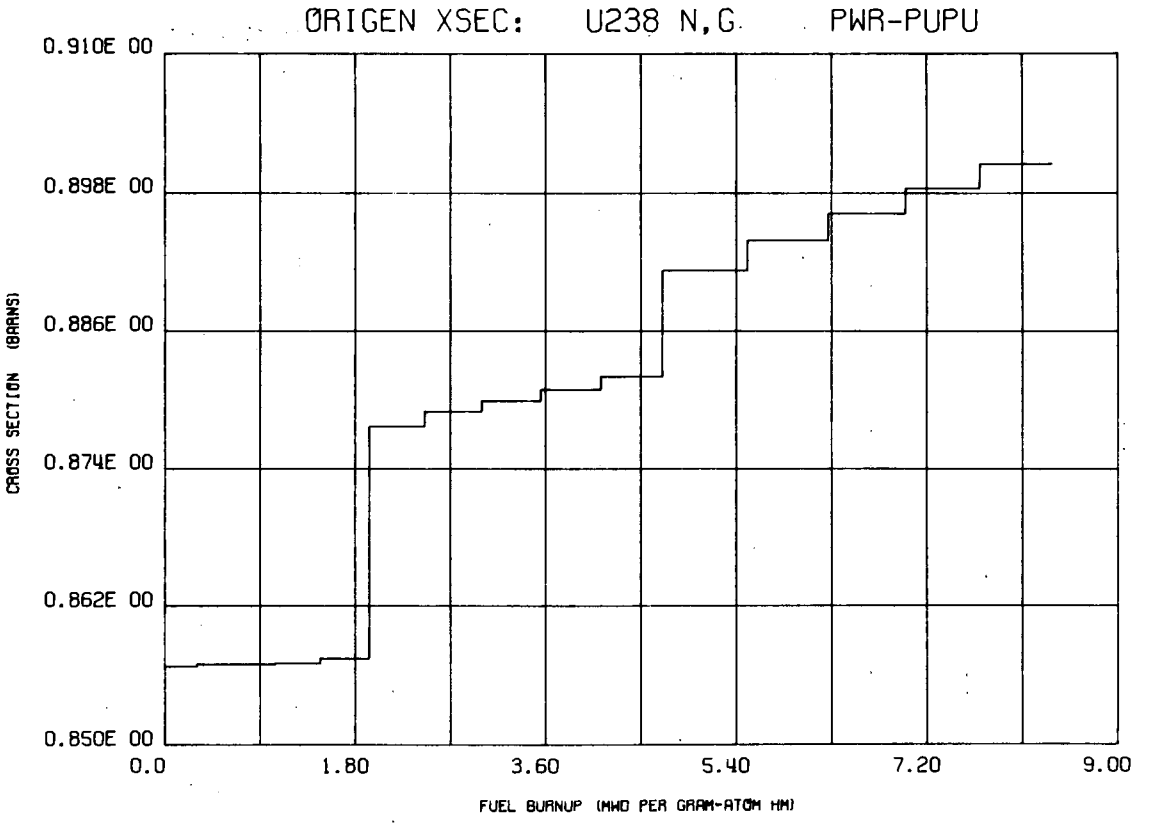

Fig. C.12. Burnup-dependent cross section of ^{238}U (n,γ) reaction for PWR-PuPu fuel.

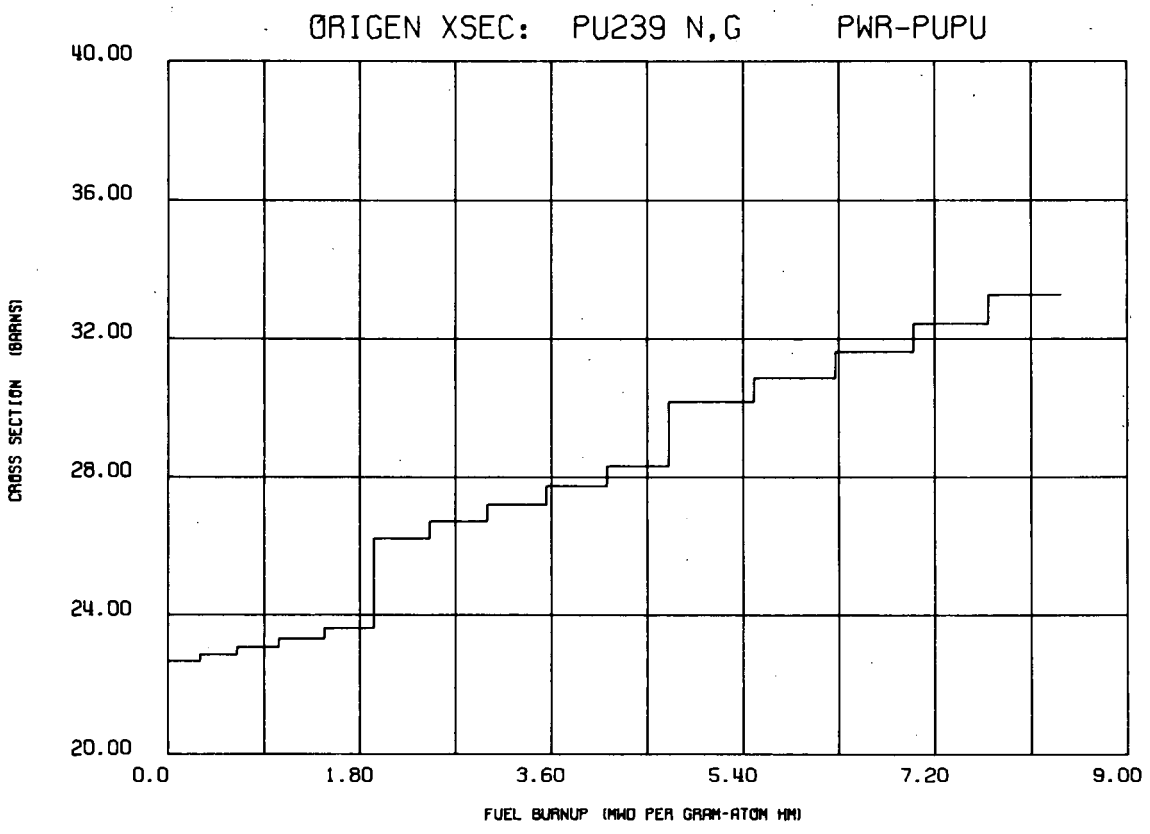

Fig. C.13. Burnup-dependent cross section of ^{239}Pu (n,γ) reaction for PWR-PuPu fuel.

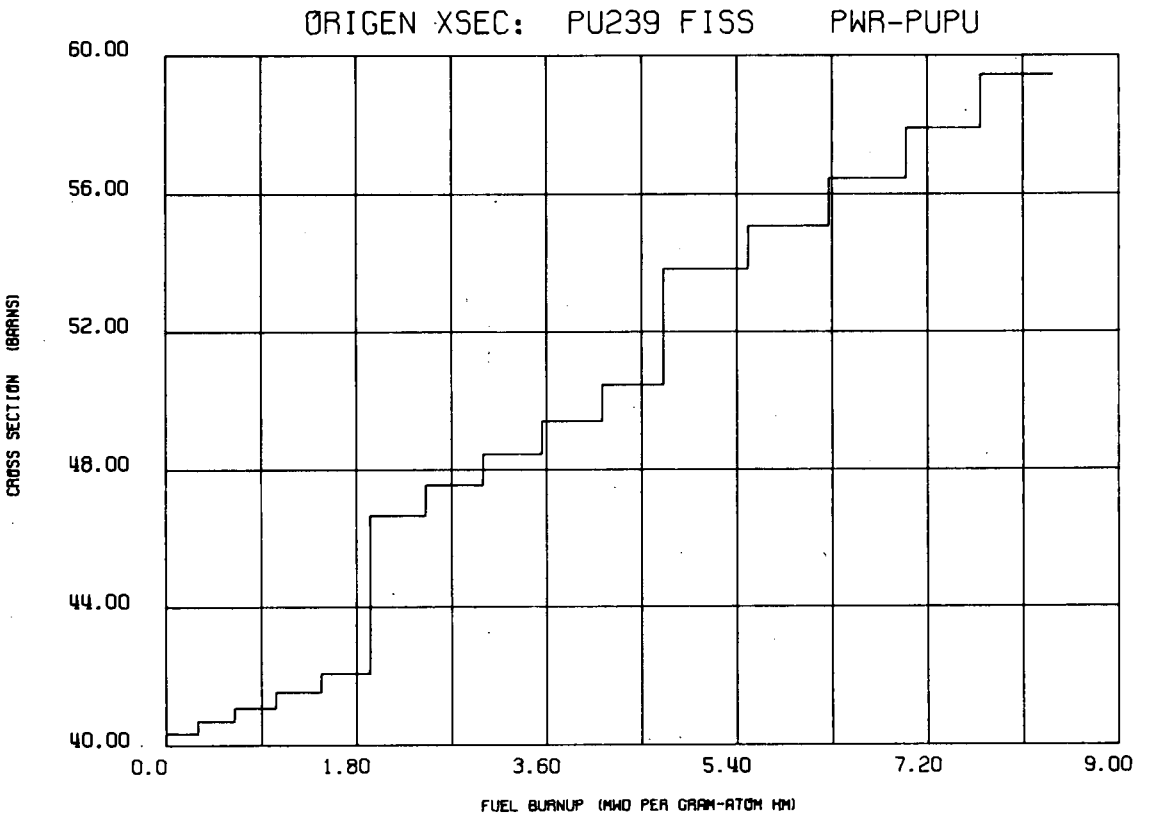

Fig. C.14. Burnup-dependent cross section of ^{239}Pu (n,fission) reaction for PWR-PuPu fuel.

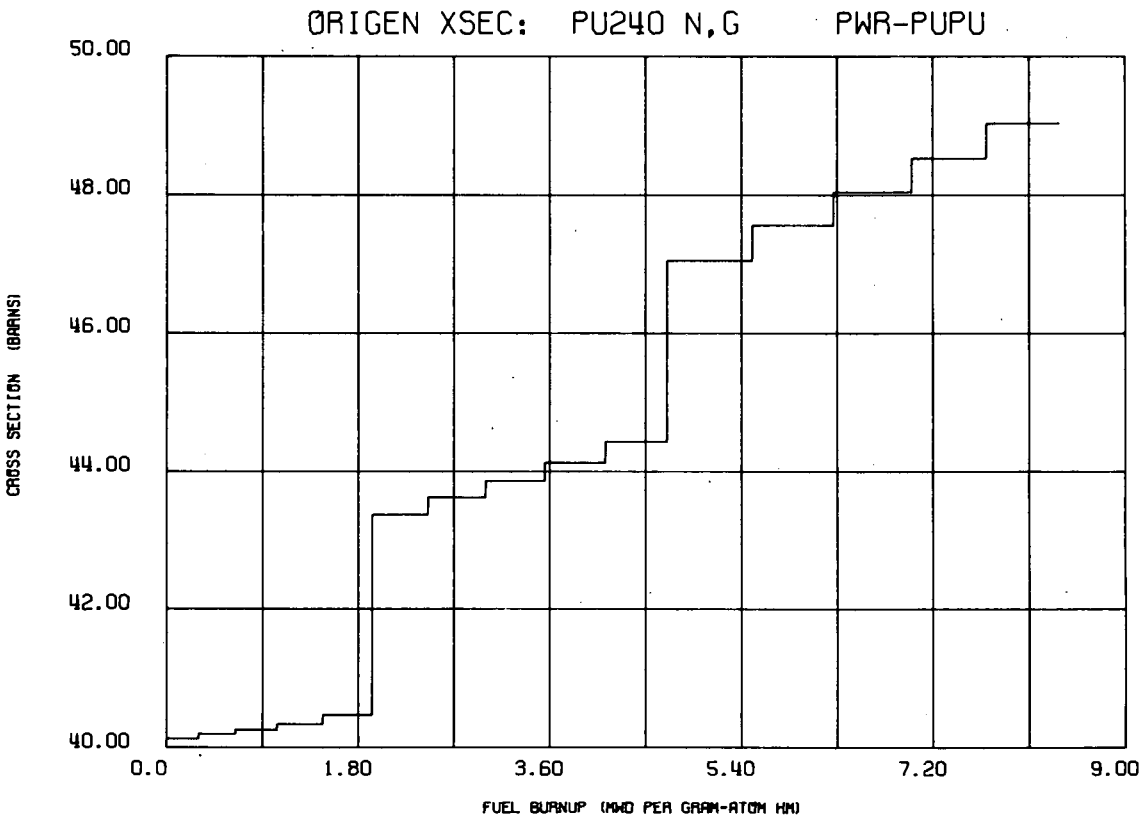

Fig. C.15. Burnup-dependent cross section of ^{240}Pu (n,γ) reaction for PWR-PuPu fuel.

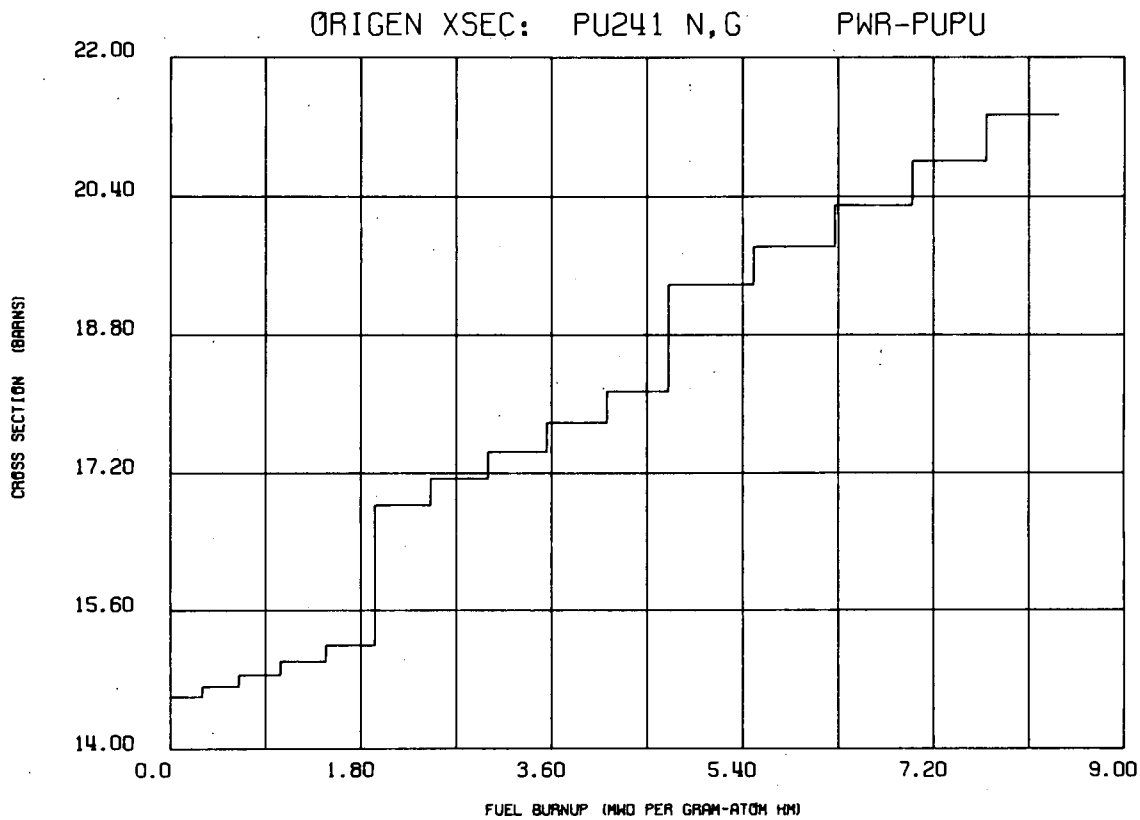

Fig. C.16. Burnup-dependent cross section of ^{241}Pu (n,γ) reaction for PWR-PuPu fuel.

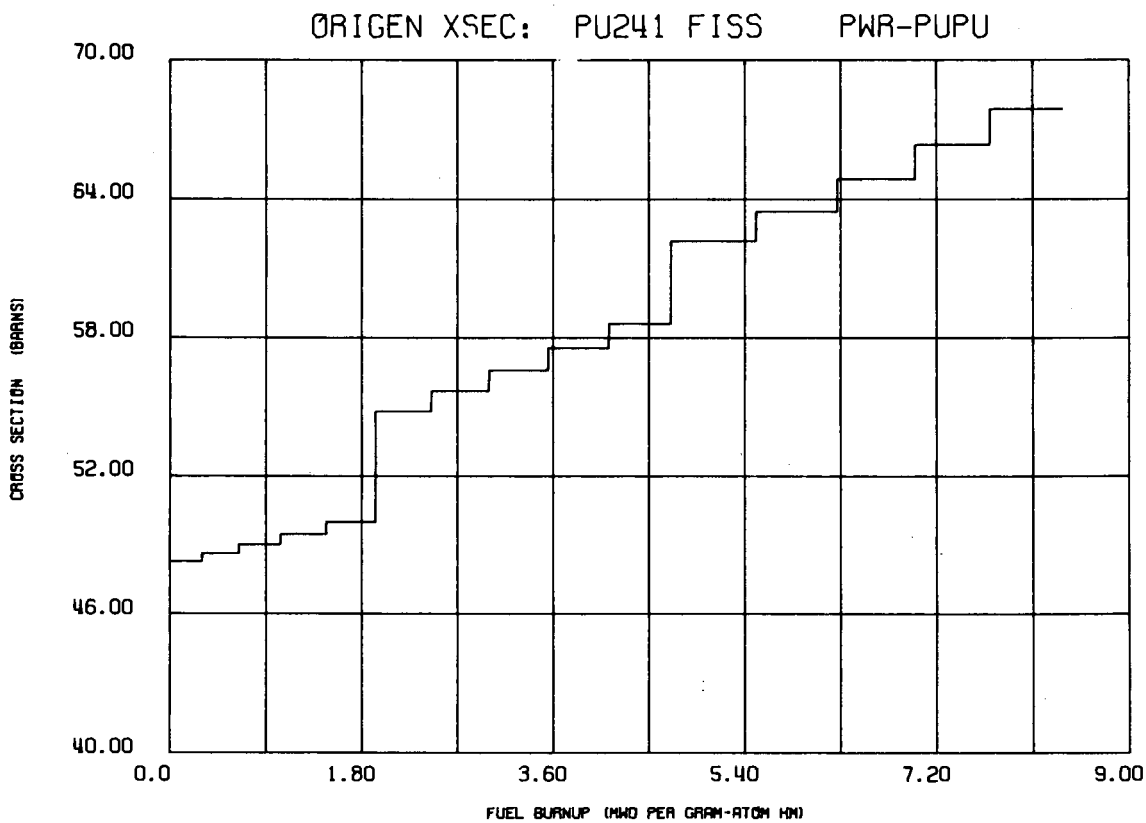

Fig. C.17. Burnup-dependent cross section of ^{241}Pu (n,fission) reaction for PWR-PuPu fuel.

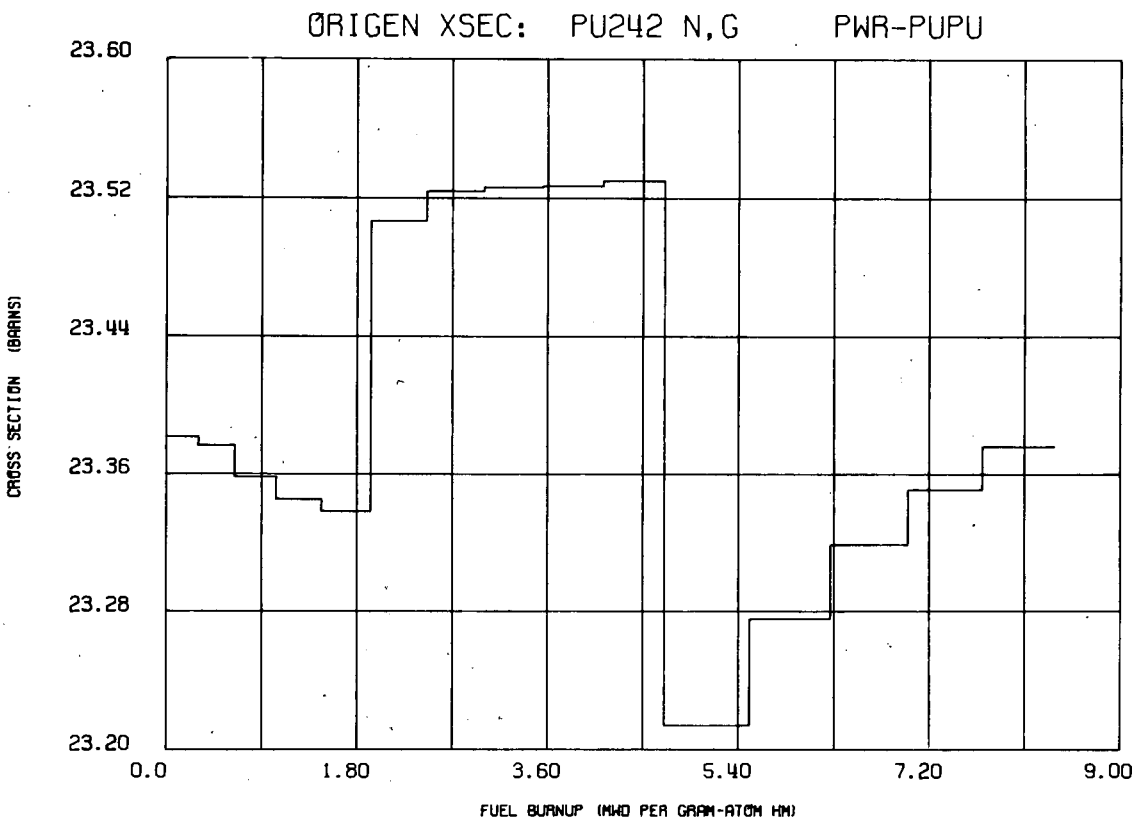

Fig. C.18. Burnup-dependent cross section of ^{242}Pu (n,γ) reaction for PWR-PuPu fuel.

Table C.1. Cross sections as a function of burnup for a PWR-U

Nuclide	Cross section[a] type	Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 0.0	0.571	1.151	1.738	2.329	2.923	2.923	3.461	3.995
U234	N,G	1.94E 01	1.93E 01	1.94E 01	1.94E 01	1.94E 01	1.94E 01	1.93E 01	1.93E 01	1.93E 01
U235	N,G	1.03E 01	1.02E 01	1.02E 01	1.02E 01	1.02E 01	1.02E 01	1.04E 01	1.04E 01	1.03E 01
U235	FISS	4.58E 01	4.55E 01	4.53E 01	4.53E 01	4.53E 01	4.53E 01	4.66E 01	4.63E 01	4.61E 01
U236	N,G	7.69E 00	7.71E 00	7.73E 00	7.75E 00	7.76E 00	7.77E 00	7.57E 00	7.59E 00	7.61E 00
U238	N,G	8.88E-01	8.88E-01	8.89E-01	8.90E-01	8.90E-01	8.91E-01	8.94E-01	8.94E-01	8.95E-01
NP237	N,G	3.29E 01	3.27E 01	3.26E 01	3.26E 01	3.25E 01	3.26E 01	3.32E 01	3.31E 01	3.30E 01
PU238	N,G	3.31E 01	3.29E 01	3.28E 01	3.27E 01	3.28E 01	3.28E 01	3.42E 01	3.40E 01	3.38E 01
PU238	FISS	2.40E 00	2.40E 00	2.39E 00	2.39E 00	2.39E 00	2.40E 00	2.43E 00	2.43E 00	2.43E 00
PU239	N,G	6.56E 01	6.52E 01	6.49E 01	6.48E 01	6.48E 01	6.49E 01	6.21E 01	6.17E 01	6.14E 01
PU239	FISS	1.15E 02	1.15E 02	1.14E 02	1.14E 02	1.14E 02	1.14E 02	1.11E 02	1.10E 02	1.10E 02
PU240	N,G	1.85E 02	1.84E 02	1.83E 02	1.81E 02	1.80E 02	1.79E 02	1.30E 02	1.29E 02	1.29E 02
PU241	N,G	4.03E 01	4.00E 01	3.98E 01	3.98E 01	3.98E 01	3.98E 01	3.96E 01	3.93E 01	3.92E 01
PU241	FISS	1.21E 02	1.21E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02	1.19E 02	1.19E 02
PU242	N,G	2.94E 01	2.95E 01	2.96E 01	2.96E 01	2.97E 01	2.97E 01	2.93E 01	2.94E 01	2.95E 01
AM241	N,G	1.09E 02	1.08E 02	1.08E 02	1.08E 02	1.08E 02	1.08E 02	1.05E 02	1.05E 02	1.04E 02
AM241	NGEX	1.35E 01	1.34E 01	1.33E 01	1.33E 01	1.33E 01	1.33E 01	1.30E 01	1.29E 01	1.29E 01
AM243	N,G	1.91E 00	1.91E 00	1.90E 00	1.89E 00	1.88E 00	1.87E 00	1.95E 00	1.94E 00	1.93E 00
AM243	NGEX	3.64E 01	3.62E 01	3.61E 01	3.59E 01	3.57E 01	3.55E 01	3.70E 01	3.68E 01	3.66E 01
CM242	N,G	5.42E 00	5.43E 00	5.43E 00	5.44E 00	5.44E 00	5.45E 00	5.45E 00	5.45E 00	5.46E 00
CM244	N,G	3.40E 00	3.40E 00	3.41E 00	3.41E 00	3.41E 00	3.42E 00	3.42E 00	3.42E 00	3.43E 00

		Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 4.528	5.061	5.593	5.593	6.077	6.558	7.038	7.518	8.000
U234	N,G	1.93E 01	1.93E 01	1.93E 01	1.93E 01	1.93E 01	1.93E 01	1.93E 01	1.94E 01	1.94E 01
U235	N,G	1.03E 01	1.03E 01	1.04E 01	1.06E 01	1.06E 01	1.06E 01	1.06E 01	1.06E 01	1.06E 01
U235	FISS	4.61E 01	4.61E 01	4.61E 01	4.78E 01	4.75E 01	4.73E 01	4.73E 01	4.73E 01	4.73E 01
U236	N,G	7.63E 00	7.64E 00	7.65E 00	7.51E 00	7.53E 00	7.55E 00	7.56E 00	7.58E 00	7.59E 00
U238	N,G	8.96E-01	8.96E-01	8.97E-01	9.03E-01	9.03E-01	9.04E-01	9.04E-01	9.05E-01	9.06E-01
NP237	N,G	3.30E 01	3.29E 01	3.29E 01	3.33E 01	3.32E 01	3.31E 01	3.30E 01	3.30E 01	3.30E 01
PU238	N,G	3.38E 01	3.38E 01	3.38E 01	3.54E 01	3.52E 01	3.50E 01	3.50E 01	3.50E 01	3.50E 01
PU238	FISS	2.43E 00	2.43E 00	2.43E 00	2.47E 00	2.47E 00	2.47E 00	2.46E 00	2.46E 00	2.47E 00
PU239	N,G	6.13E 01	6.13E 01	6.14E 01	6.11E 01	6.06E 01	6.04E 01	6.03E 01	6.03E 01	6.04E 01
PU239	FISS	1.10E 02	1.10E 02	1.10E 02	1.10E 02	1.10E 02	1.09E 02	1.09E 02	1.09E 02	1.09E 02
PU240	N,G	1.28E 02	1.27E 02	1.26E 02	1.07E 02	1.07E 02	1.06E 02	1.05E 02	1.05E 02	1.04E 02
PU241	N,G	3.91E 01	3.91E 01	3.92E 01	3.99E 01	3.96E 01	3.94E 01	3.94E 01	3.94E 01	3.94E 01
PU241	FISS	1.19E 02	1.19E 02	1.19E 02	1.21E 02	1.21E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02
PU242	N,G	2.95E 01	2.96E 01	2.96E 01	2.89E 01	2.90E 01	2.91E 01	2.91E 01	2.92E 01	2.92E 01
AM241	N,G	1.04E 02	1.04E 02	1.04E 02	1.04E 02	1.03E 02	1.03E 02	1.02E 02	1.02E 02	1.02E 02
AM241	NGEX	1.28E 01	1.28E 01	1.28E 01	1.28E 01	1.27E 01	1.27E 01	1.27E 01	1.26E 01	1.26E 01
AM243	N,G	1.92E 00	1.91E 00	1.90E 00	1.96E 00	1.95E 00	1.94E 00	1.93E 00	1.92E 00	1.91E 00
AM243	NGEX	3.64E 01	3.63E 01	3.61E 01	3.72E 01	3.71E 01	3.69E 01	3.67E 01	3.65E 01	3.63E 01
CM242	N,G	5.47E 00	5.47E 00	5.48E 00	5.49E 00	5.50E 00	5.50E 00	5.51E 00	5.52E 00	5.52E 00
CM244	N,G	3.43E 00	3.43E 00	3.44E 00	3.45E 00	3.45E 00	3.46E 00	3.46E 00	3.46E 00	3.46E 00

[a]N,G = (n,gamma) to a ground state; NGEX = (n,gamma) to an excited state; FISS = (n,fission).

Table C.2. Cross sections as a function of burnup for a PWR-PuU

Nuclide	Cross section[a] type	Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 0.0	0.854	1.662	2.420	3.136	3.821	3.821	4.299	4.786
U234	N,G	2.03E 01	2.01E 01	2.01E 01	2.01E 01	2.01E 01	2.01E 01	1.99E 01	2.00E 01	2.00E 01
U235	N,G	1.05E 01	1.02E 01	1.01E 01	9.99E 00	9.94E 00	9.93E 00	9.99E 00	9.98E 00	9.98E 00
U235	FISS	4.66E 01	4.50E 01	4.40E 01	4.35E 01	4.32E 01	4.30E 01	4.33E 01	4.32E 01	4.32E 01
U236	N,G	8.16E 00	8.23E 00	8.28E 00	8.33E 00	8.36E 00	8.38E 00	8.28E 00	8.30E 00	8.31E 00
U238	N,G	9.32E-01	9.30E-01	9.31E-01	9.32E-01	9.33E-01	9.34E-01	9.38E-01	9.39E-01	9.39E-01
NP237	N,G	3.35E 01	3.28E 01	3.23E 01	3.21E 01	3.19E 01	3.18E 01	3.21E 01	3.20E 01	3.20E 01
PU238	N,G	3.39E 01	3.26E 01	3.19E 01	3.15E 01	3.12E 01	3.12E 01	3.16E 01	3.16E 01	3.15E 01
PU238	FISS	2.42E 00	2.40E 00	2.38E 00	2.37E 00	2.37E 00	2.37E 00	2.39E 00	2.38E 00	2.38E 00
PU239	N,G	6.61E 01	6.36E 01	6.20E 01	6.12E 01	6.07E 01	6.05E 01	5.75E 01	5.74E 01	5.74E 01
PU239	FISS	1.16E 02	1.12E 02	1.09E 02	1.08E 02	1.07E 02	1.06E 02	1.02E 02	1.02E 02	1.02E 02
PU240	N,G	1.95E 02	1.94E 02	1.91E 02	1.87E 02	1.85E 02	1.83E 02	1.37E 02	1.36E 02	1.35E 02
PU241	N,G	4.07E 01	3.91E 01	3.82E 01	3.76E 01	3.74E 01	3.73E 01	3.64E 01	3.64E 01	3.64E 01
PU241	FISS	1.22E 02	1.18E 02	1.15E 02	1.14E 02	1.13E 02	1.13E 02	1.11E 02	1.11E 02	1.11E 02
PU242	N,G	2.94E 01	2.97E 01	2.99E 01	3.00E 01	3.02E 01	3.03E 01	3.02E 01	3.02E 01	3.03E 01
AM241	N,G	1.11E 02	1.08E 02	1.05E 02	1.04E 02	1.03E 02	1.03E 02	9.99E 01	9.97E 01	9.95E 01
AM241	NGEX	1.37E 01	1.33E 01	1.30E 01	1.29E 01	1.28E 01	1.27E 01	1.23E 01	1.23E 01	1.23E 01
AM243	N,G	2.00E 00	1.99E 00	1.97E 00	1.94E 00	1.92E 00	1.91E 00	1.97E 00	1.96E 00	1.95E 00
AM243	NGEX	3.80E 01	3.78E 01	3.74E 01	3.69E 01	3.65E 01	3.62E 01	3.75E 01	3.73E 01	3.71E 01
CM242	N,G	5.56E 00	5.55E 00	5.56E 00	5.56E 00	5.57E 00	5.58E 00	5.59E 00	5.60E 00	5.60E 00
CM244	N,G	3.48E 00	3.48E 00	3.47E 00	3.48E 00	3.48E 00	3.48E 00	3.49E 00	3.50E 00	3.50E 00

		Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 5.281	5.784	6.290	6.290	6.556	6.855	7.189	7.554	7.943
U234	N,G	2.00E 01	2.00E 01	2.01E 01	2.02E 01	2.02E 01	2.02E 01	2.02E 01	2.02E 01	2.02E 01
U235	N,G	9.99E 00	1.00E 01	1.00E 01	1.02E 01	1.02E 01	1.02E 01	1.02E 01	1.03E 01	1.03E 01
U235	FISS	4.32E 01	4.33E 01	4.34E 01	4.44E 01	4.44E 01	4.45E 01	4.45E 01	4.46E 01	4.48E 01
U236	N,G	8.32E 00	8.33E 00	8.33E 00	8.31E 00	8.31E 00	8.31E 00	8.31E 00	8.31E 00	8.31E 00
U238	N,G	9.40E-01	9.41E-01	9.41E-01	9.51E-01	9.51E-01	9.51E-01	9.51E-01	9.52E-01	9.52E-01
NP237	N,G	3.19E 01	3.20E 01	3.20E 01	3.23E 01	3.23E 01	3.23E 01	3.23E 01	3.23E 01	3.24E 01
PU238	N,G	3.16E 01	3.16E 01	3.17E 01	3.26E 01	3.26E 01	3.27E 01	3.27E 01	3.28E 01	3.29E 01
PU238	FISS	2.39E 00	2.39E 00	2.39E 00	2.41E 00	2.42E 00	2.42E 00	2.42E 00	2.42E 00	2.42E 00
PU239	N,G	5.74E 01	5.75E 01	5.77E 01	5.71E 01	5.72E 01	5.72E 01	5.73E 01	5.75E 01	5.77E 01
PU239	FISS	1.02E 02	1.02E 02	1.03E 02	1.02E 02	1.03E 02	1.03E 02	1.03E 02	1.03E 02	1.04E 02
PU240	N,G	1.34E 02	1.34E 02	1.33E 02	1.14E 02	1.14E 02	1.13E 02	1.13E 02	1.13E 02	1.12E 02
PU241	N,G	3.64E 01	3.65E 01	3.66E 01	3.68E 01	3.69E 01	3.69E 01	3.70E 01	3.71E 01	3.72E 01
PU241	FISS	1.11E 02	1.11E 02	1.11E 02	1.12E 02	1.13E 02	1.13E 02	1.13E 02	1.13E 02	1.13E 02
PU242	N,G	3.03E 01	3.03E 01	3.04E 01	3.02E 01	3.02E 01	3.02E 01	3.01E 01	3.01E 01	3.01E 01
AM241	N,G	9.95E 01	9.96E 01	9.97E 01	9.92E 01	9.93E 01	9.93E 01	9.94E 01	9.95E 01	9.97E 01
AM241	NGEX	1.23E 01	1.23E 01	1.23E 01	1.23E 01	1.23E 01	1.23E 01	1.23E 01	1.23E 01	1.23E 01
AM243	N,G	1.94E 00	1.93E 00	1.92E 00	1.97E 00	1.97E 00	1.96E 00	1.96E 00	1.95E 00	1.95E 00
AM243	NGEX	3.69E 01	3.67E 01	3.66E 01	3.75E 01	3.74E 01	3.73E 01	3.72E 01	3.71E 01	3.70E 01
CM242	N,G	5.61E 00	5.61E 00	5.62E 00	5.67E 00	5.67E 00	5.67E 00	5.67E 00	5.67E 00	5.67E 00
CM244	N,G	3.50E 00	3.51E 00	3.51E 00	3.54E 00	3.54E 00	3.54E 00	3.54E 00	3.54E 00	3.54E 00

[a]N,G = (n,gamma) to a ground state; NGEX = (n,gamma) to an excited state; FISS = (n,fission).

Table C.3. Cross sections as a function of burnup for a PWR-PuPu

Nuclide	Cross section[a] type	Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 0.0	0.306	0.653	1.042	1.469	1.929	1.929	2.455	2.997
U234	N,G	1.66E 01	1.66E 01	1.66E 01	1.66E 01	1.66E 01	1.67E 01	1.72E 01	1.73E 01	1.73E 01
U235	N,G	5.95E 00	5.97E 00	5.99E 00	6.02E 00	6.06E 00	6.10E 00	6.44E 00	6.50E 00	6.55E 00
U235	FISS	2.03E 01	2.04E 01	2.06E 01	2.07E 01	2.09E 01	2.12E 01	2.27E 01	2.31E 01	2.34E 01
U236	N,G	8.35E 00	8.35E 00	8.34E 00	8.34E 00	8.33E 00	8.33E 00	8.50E 00	8.50E 00	8.50E 00
U238	N,G	8.57E-01	8.57E-01	8.57E-01	8.57E-01	8.57E-01	8.58E-01	8.78E-01	8.79E-01	8.80E-01
NP237	N,G	2.24E 01	2.25E 01	2.26E 01	2.26E 01	2.28E 01	2.29E 01	2.35E 01	2.37E 01	2.39E 01
PU238	N,G	1.34E 01	1.35E 01	1.37E 01	1.38E 01	1.39E 01	1.41E 01	1.53E 01	1.56E 01	1.59E 01
PU238	FISS	2.00E 00	2.00E 00	2.00E 00	2.01E 00	2.01E 00	2.01E 00	2.04E 00	2.05E 00	2.05E 00
PU239	N,G	2.27E 01	2.29E 01	2.31E 01	2.33E 01	2.36E 01	2.40E 01	2.62E 01	2.67E 01	2.72E 01
PU239	FISS	4.03E 01	4.07E 01	4.11E 01	4.16E 01	4.21E 01	4.27E 01	4.67E 01	4.76E 01	4.85E 01
PU240	N,G	4.01E 01	4.02E 01	4.03E 01	4.03E 01	4.05E 01	4.06E 01	4.34E 01	4.36E 01	4.39E 01
PU241	N,G	1.46E 01	1.47E 01	1.49E 01	1.50E 01	1.52E 01	1.54E 01	1.68E 01	1.71E 01	1.74E 01
PU241	FISS	4.83E 01	4.87E 01	4.90E 01	4.95E 01	5.00E 01	5.07E 01	5.48E 01	5.57E 01	5.66E 01
PU242	N,G	2.34E 01	2.34E 01	2.34E 01	2.33E 01	2.33E 01	2.33E 01	2.35E 01	2.35E 01	2.35E 01
AM241	N,G	5.16E 01	5.19E 01	5.22E 01	5.26E 01	5.30E 01	5.36E 01	5.62E 01	5.70E 01	5.77E 01
AM241	NGEX	6.38E 00	6.41E 00	6.45E 00	6.50E 00	6.55E 00	6.62E 00	6.95E 00	7.04E 00	7.13E 00
AM243	N,G	1.43E 00	1.43E 00	1.43E 00	1.43E 00	1.43E 00	1.43E 00	1.48E 00	1.48E 00	1.48E 00
AM243	NGEX	2.72E 01	2.72E 01	2.71E 01	2.71E 01	2.72E 01	2.72E 01	2.81E 01	2.82E 01	2.82E 01
CM242	N,G	4.93E 00	4.93E 00	4.93E 00	4.94E 00	4.94E 00	4.94E 00	5.08E 00	5.09E 00	5.10E 00
CM244	N,G	3.11E 00	3.11E 00	3.11E 00	3.11E 00	3.11E 00	3.11E 00	3.19E 00	3.19E 00	3.20E 00

Nuclide	Cross section type	Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 3.554	4.123	4.701	4.701	5.501	6.267	6.999	7.700	8.377
U234	N,G	1.74E 01	1.74E 01	1.75E 01	1.76E 01	1.77E 01	1.78E 01	1.79E 01	1.80E 01	1.81E 01
U235	N,G	6.62E 00	6.69E 00	6.76E 00	6.94E 00	7.03E 00	7.13E 00	7.23E 00	7.33E 00	7.44E 00
U235	FISS	2.38E 01	2.42E 01	2.46E 01	2.56E 01	2.61E 01	2.66E 01	2.72E 01	2.77E 01	2.84E 01
U236	N,G	8.50E 00	8.50E 00	8.49E 00	8.51E 00	8.53E 00	8.54E 00	8.54E 00	8.55E 00	8.55E 00
U238	N,G	8.81E-01	8.82E-01	8.83E-01	8.91E-01	8.94E-01	8.96E-01	8.98E-01	9.00E-01	9.03E-01
NP237	N,G	2.41E 01	2.43E 01	2.46E 01	2.46E 01	2.49E 01	2.52E 01	2.55E 01	2.59E 01	2.62E 01
PU238	N,G	1.62E 01	1.65E 01	1.68E 01	1.76E 01	1.79E 01	1.84E 01	1.88E 01	1.93E 01	1.97E 01
PU238	FISS	2.06E 00	2.06E 00	2.07E 00	2.09E 00	2.10E 00	2.11E 00	2.12E 00	2.12E 00	2.13E 00
PU239	N,G	2.77E 01	2.83E 01	2.89E 01	3.02E 01	3.09E 01	3.16E 01	3.24E 01	3.33E 01	3.41E 01
PU239	FISS	4.94E 01	5.05E 01	5.16E 01	5.38E 01	5.51E 01	5.64E 01	5.79E 01	5.94E 01	6.10E 01
PU240	N,G	4.41E 01	4.44E 01	4.48E 01	4.70E 01	4.76E 01	4.80E 01	4.85E 01	4.90E 01	4.96E 01
PU241	N,G	1.78E 01	1.81E 01	1.85E 01	1.94E 01	1.98E 01	2.03E 01	2.08E 01	2.13E 01	2.19E 01
PU241	FISS	5.76E 01	5.86E 01	5.97E 01	6.22E 01	6.35E 01	6.49E 01	6.63E 01	6.79E 01	6.95E 01
PU242	N,G	2.35E 01	2.35E 01	2.35E 01	2.32E 01	2.33E 01	2.33E 01	2.34E 01	2.34E 01	2.34E 01
AM241	N,G	5.85E 01	5.94E 01	6.03E 01	6.13E 01	6.23E 01	6.35E 01	6.47E 01	6.60E 01	6.73E 01
AM241	NGEX	7.23E 00	7.34E 00	7.45E 00	7.57E 00	7.71E 00	7.85E 00	8.00E 00	8.15E 00	8.32E 00
AM243	N,G	1.49E 00	1.49E 00	1.50E 00	1.50E 00	1.51E 00	1.52E 00	1.53E 00	1.54E 00	1.54E 00
AM243	NGEX	2.83E 01	2.83E 01	2.84E 01	2.85E 01	2.87E 01	2.89E 01	2.90E 01	2.92E 01	2.93E 01
CM242	N,G	5.11E 00	5.11E 00	5.12E 00	5.17E 00	5.19E 00	5.20E 00	5.22E 00	5.24E 00	5.25E 00
CM244	N,G	3.20E 00	3.21E 00	3.21E 00	3.24E 00	3.25E 00	3.26E 00	3.27E 00	3.28E 00	3.29E 00

[a]N,G = (n,gamma) to a ground state; NGEX = (n,gamma) to an excited state; FISS = (n,fission).

Table C.4. Cross sections as a function of burnup for a BWR-U

Nuclide	Cross section type[a]	Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 0.0	0.348	0.702	1.060	1.422	1.787	1.787	2.132	2.477	2.821	3.166	3.510
U234	N,G	2.09E 01	2.09E 01	2.09E 01	2.09E 01	2.09E 01	2.09E 01	2.08E 01	2.08E 01	2.08E 01	2.08E 01	2.08E 01	2.08E 01
U235	N,G	1.10E 01	1.09E 01	1.09E 01	1.09E 01	1.08E 01	1.08E 01	1.11E 01	1.10E 01	1.10E 01	1.09E 01	1.09E 01	1.09E 01
U235	FISS	4.91E 01	4.88E 01	4.85E 01	4.83E 01	4.82E 01	4.81E 01	4.94E 01	4.91E 01	4.89E 01	4.87E 01	4.85E 01	4.84E 01
U236	N,G	8.33E 00	8.35E 00	8.37E 00	8.38E 00	8.40E 00	8.41E 00	8.25E 00	8.27E 00	8.29E 00	8.30E 00	8.31E 00	8.32E 00
U238	N,G	9.01E-01	9.01E-01	9.01E-01	9.01E-01	9.01E-01	9.01E-01	9.03E-01	9.03E-01	9.03E-01	9.04E-01	9.04E-01	9.04E-01
NP237	N,G	3.40E 01	3.38E 01	3.37E 01	3.36E 01	3.35E 01	3.35E 01	3.44E 01	3.42E 01	3.41E 01	3.40E 01	3.39E 01	3.39E 01
PU238	N,G	3.59E 01	3.57E 01	3.55E 01	3.54E 01	3.52E 01	3.52E 01	3.65E 01	3.62E 01	3.60E 01	3.59E 01	3.58E 01	3.57E 01
PU238	FISS	2.44E 00	2.43E 00	2.43E 00	2.43E 00	2.42E 00	2.42E 00	2.45E 00	2.45E 00	2.45E 00	2.44E 00	2.44E 00	2.44E 00
PU239	N,G	6.91E 01	6.86E 01	6.82E 01	6.79E 01	6.77E 01	6.76E 01	6.62E 01	6.58E 01	6.54E 01	6.51E 01	6.49E 01	6.47E 01
PU239	FISS	1.22E 02	1.21E 02	1.20E 02	1.20E 02	1.19E 02	1.19E 02	1.18E 02	1.17E 02	1.17E 02	1.16E 02	1.16E 02	1.16E 02
PU240	N,G	2.06E 02	2.05E 02	2.04E 02	2.03E 02	2.01E 02	2.00E 02	1.58E 02	1.57E 02	1.56E 02	1.55E 02	1.54E 02	1.53E 02
PU241	N,G	4.29E 01	4.26E 01	4.24E 01	4.22E 01	4.21E 01	4.20E 01	4.22E 01	4.19E 01	4.17E 01	4.15E 01	4.14E 01	4.13E 01
PU241	FISS	1.29E 02	1.28E 02	1.28E 02	1.27E 02	1.27E 02	1.26E 02	1.28E 02	1.27E 02	1.26E 02	1.26E 02	1.25E 02	1.25E 02
PU242	N,G	3.06E 01	3.07E 01	3.08E 01	3.08E 01	3.09E 01	3.09E 01	3.06E 01	3.07E 01	3.08E 01	3.08E 01	3.09E 01	3.09E 01
AM241	N,G	1.14E 02	1.13E 02	1.13E 02	1.12E 02	1.12E 02	1.12E 02	1.11E 02	1.10E 02	1.10E 02	1.09E 02	1.09E 02	1.09E 02
AM241	NGEX	1.41E 01	1.40E 01	1.39E 01	1.39E 01	1.38E 01	1.38E 01	1.37E 01	1.36E 01	1.36E 01	1.35E 01	1.35E 01	1.34E 01
AM243	N,G	2.02E 00	2.02E 00	2.01E 00	2.00E 00	1.99E 00	1.98E 00	2.05E 00	2.05E 00	2.04E 00	2.03E 00	2.02E 00	2.01E 00
AM243	NGEX	3.84E 01	3.83E 01	3.81E 01	3.80E 01	3.78E 01	3.76E 01	3.90E 01	3.89E 01	3.87E 01	3.85E 01	3.84E 01	3.82E 01
CM242	N,G	5.73E 00	5.73E 00	5.74E 00	5.74E 00	5.74E 00	5.75E 00	5.74E 00	5.75E 00	5.75E 00	5.75E 00	5.75E 00	5.76E 00
CM244	N,G	3.60E 00	3.60E 00	3.60E 00	3.60E 00	3.60E 00	3.60E 00	3.61E 00	3.61E 00	3.61E 00	3.61E 00	3.61E 00	3.61E 00

		Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 3.510	3.836	4.160	4.483	4.805	5.127	5.127	5.432	5.735	6.037	6.339	6.640
U234	N,G	2.08E 01	2.08E 01	2.08E 01	2.08E 01	2.08E 01	2.08E 01	2.09E 01	2.09E 01	2.08E 01	2.08E 01	2.08E 01	2.08E 01
U235	N,G	1.12E 01	1.11E 01	1.11E 01	1.11E 01	1.10E 01	1.10E 01	1.13E 01	1.13E 01	1.12E 01	1.12E 01	1.12E 01	1.12E 01
U235	FISS	5.01E 01	4.97E 01	4.95E 01	4.93E 01	4.91E 01	4.90E 01	5.08E 01	5.05E 01	5.03E 01	5.01E 01	4.99E 01	4.98E 01
U236	N,G	8.20E 00	8.22E 00	8.23E 00	8.25E 00	8.26E 00	8.27E 00	8.16E 00	8.18E 00	8.20E 00	8.21E 00	8.22E 00	8.23E 00
U238	N,G	9.08E-01	9.08E-01	9.08E-01	9.08E-01	9.08E-01	9.08E-01	9.13E-01	9.13E-01	9.13E-01	9.13E-01	9.14E-01	9.14E-01
NP237	N,G	3.45E 01	3.44E 01	3.43E 01	3.41E 01	3.41E 01	3.40E 01	3.45E 01	3.44E 01	3.43E 01	3.42E 01	3.41E 01	3.40E 01
PU238	N,G	3.71E 01	3.69E 01	3.67E 01	3.65E 01	3.64E 01	3.63E 01	3.79E 01	3.76E 01	3.74E 01	3.73E 01	3.72E 01	3.71E 01
PU238	FISS	2.48E 00	2.47E 00	2.47E 00	2.46E 00	2.46E 00	2.46E 00	2.50E 00	2.49E 00	2.49E 00	2.49E 00	2.49E 00	2.48E 00
PU239	N,G	6.49E 01	6.45E 01	6.41E 01	6.38E 01	6.36E 01	6.35E 01	6.42E 01	6.38E 01	6.34E 01	6.31E 01	6.29E 01	6.28E 01
PU239	FISS	1.17E 02	1.16E 02	1.15E 02	1.15E 02	1.14E 02	1.14E 02	1.16E 02	1.15E 02	1.15E 02	1.14E 02	1.14E 02	1.14E 02
PU240	N,G	1.31E 02	1.30E 02	1.30E 02	1.29E 02	1.28E 02	1.27E 02	1.15E 02	1.14E 02	1.14E 02	1.13E 02	1.12E 02	1.12E 02
PU241	N,G	4.21E 01	4.18E 01	4.16E 01	4.14E 01	4.13E 01	4.12E 01	4.22E 01	4.19E 01	4.17E 01	4.15E 01	4.14E 01	4.13E 01
PU241	FISS	1.28E 02	1.27E 02	1.26E 02	1.26E 02	1.25E 02	1.25E 02	1.28E 02	1.28E 02	1.27E 02	1.26E 02	1.26E 02	1.26E 02
PU242	N,G	3.05E 01	3.06E 01	3.06E 01	3.07E 01	3.07E 01	3.08E 01	3.02E 01	3.02E 01	3.03E 01	3.03E 01	3.04E 01	3.04E 01
AM241	N,G	1.10E 02	1.09E 02	1.08E 02	1.08E 02	1.08E 02	1.07E 02	1.08E 02	1.08E 02	1.07E 02	1.07E 02	1.07E 02	1.06E 02
AM241	NGEX	1.35E 01	1.35E 01	1.34E 01	1.33E 01	1.33E 01	1.33E 01	1.34E 01	1.33E 01	1.33E 01	1.32E 01	1.32E 01	1.31E 01
AM243	N,G	2.07E 00	2.06E 00	2.05E 00	2.04E 00	2.03E 00	2.03E 00	2.08E 00	2.07E 00	2.06E 00	2.05E 00	2.04E 00	2.03E 00
AM243	NGEX	3.93E 01	3.92E 01	3.90E 01	3.88E 01	3.87E 01	3.85E 01	3.95E 01	3.93E 01	3.92E 01	3.90E 01	3.88E 01	3.86E 01
CM242	N,G	5.76E 00	5.77E 00	5.77E 00	5.77E 00	5.77E 00	5.78E 00	5.79E 00	5.79E 00	5.80E 00	5.80E 00	5.80E 00	5.80E 00
CM244	N,G	3.62E 00	3.62E 00	3.62E 00	3.62E 00	3.62E 00	3.63E 00	3.64E 00	3.64E 00	3.64E 00	3.64E 00	3.64E 00	3.64E 00

[a]N,G = (n,gamma) to a ground state; NGEX = (n,gamma) to an excited state; FISS = (n,fission).

Table C.5. Cross sections as a function of burnup for a BWR-PuU

Nuclide	Cross section[a] type	Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 0.0	0.407	0.812	1.214	1.610	2.001	2.001	2.343	2.683	3.023	3.362	3.700
U234	N,G	2.07E 01	2.06E 01	2.05E 01	2.05E 01	2.05E 01	2.05E 01	2.02E 01	2.02E 01	2.02E 01	2.02E 01	2.03E 01	2.03E 01
U235	N,G	1.07E 01	1.05E 01	1.04E 01	1.03E 01	1.03E 01	1.03E 01	1.01E 01	1.01E 01	1.01E 01	1.01E 01	1.01E 01	1.01E 01
U235	FISS	4.74E 01	4.65E 01	4.58E 01	4.53E 01	4.49E 01	4.47E 01	4.39E 01	4.37E 01	4.36E 01	4.35E 01	4.35E 01	4.35E 01
U236	N,G	8.34E 00	8.37E 00	8.40E 00	8.42E 00	8.45E 00	8.46E 00	8.40E 00	8.42E 00	8.43E 00	8.44E 00	8.45E 00	8.46E 00
U238	N,G	9.00E-01	8.98E-01	8.97E-01	8.97E-01	8.97E-01	8.98E-01	8.98E-01	8.98E-01	8.99E-01	8.99E-01	8.99E-01	9.00E-01
NP237	N,G	3.40E 01	3.36E 01	3.33E 01	3.30E 01	3.28E 01	3.27E 01	3.25E 01	3.24E 01	3.24E 01	3.23E 01	3.23E 01	3.22E 01
PU238	N,G	3.45E 01	3.38E 01	3.33E 01	3.29E 01	3.26E 01	3.24E 01	3.20E 01	3.19E 01	3.18E 01	3.17E 01	3.17E 01	3.17E 01
PU238	FISS	2.41E 00	2.39E 00	2.38E 00	2.37E 00	2.37E 00	2.36E 00	2.36E 00	2.36E 00	2.36E 00	2.35E 00	2.35E 00	2.35E 00
PU239	N,G	6.73E 01	6.58E 01	6.47E 01	6.40E 01	6.34E 01	6.30E 01	5.87E 01	5.85E 01	5.83E 01	5.82E 01	5.81E 01	5.81E 01
PU239	FISS	1.18E 02	1.16E 02	1.14E 02	1.13E 02	1.11E 02	1.11E 02	1.04E 02	1.04E 02	1.04E 02	1.03E 02	1.03E 02	1.03E 02
PU240	N,G	2.05E 02	2.05E 02	2.04E 02	2.02E 02	2.00E 02	1.98E 02	1.53E 02	1.52E 02	1.52E 02	1.51E 02	1.50E 02	1.49E 02
PU241	N,G	4.14E 01	4.05E 01	3.99E 01	3.94E 01	3.90E 01	3.88E 01	3.71E 01	3.70E 01	3.68E 01	3.68E 01	3.67E 01	3.67E 01
PU241	FISS	1.25E 02	1.22E 02	1.20E 02	1.19E 02	1.18E 02	1.17E 02	1.13E 02	1.13E 02	1.12E 02	1.12E 02	1.12E 02	1.12E 02
PU242	N,G	3.04E 01	3.06E 01	3.07E 01	3.07E 01	3.08E 01	3.09E 01	3.09E 01	3.10E 01	3.10E 01	3.11E 01	3.11E 01	3.12E 01
AM241	N,G	1.13E 02	1.11E 02	1.09E 02	1.08E 02	1.07E 02	1.07E 02	1.02E 02	1.01E 02	1.01E 02	1.01E 02	1.01E 02	1.01E 02
AM241	NGEX	1.39E 01	1.37E 01	1.35E 01	1.34E 01	1.33E 01	1.32E 01	1.26E 01	1.25E 01	1.25E 01	1.25E 01	1.24E 01	1.24E 01
AM243	N,G	2.04E 00	2.04E 00	2.03E 00	2.01E 00	2.00E 00	1.98E 00	2.02E 00	2.01E 00	2.00E 00	1.99E 00	1.98E 00	1.97E 00
AM243	NGEX	3.88E 01	3.87E 01	3.85E 01	3.82E 01	3.80E 01	3.77E 01	3.84E 01	3.82E 01	3.80E 01	3.78E 01	3.76E 01	3.75E 01
CM242	N,G	5.70E 00	5.69E 00	5.69E 00	5.69E 00	5.69E 00	5.70E 00	5.69E 00	5.69E 00	5.70E 00	5.70E 00	5.70E 00	5.71E 00
CM244	N,G	3.58E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00

		Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 3.700	3.972	4.248	4.527	4.809	5.096	5.096	5.330	5.568	5.812	6.062	6.319
U234	N,G	2.00E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01	2.00E 01
U235	N,G	9.86E 00	9.86E 00	9.86E 00	9.86E 00	9.87E 00	9.88E 00	9.87E 00	9.88E 00	9.88E 00	9.89E 00	9.90E 00	9.91E 00
U235	FISS	4.21E 01	4.21E 01	4.21E 01	4.21E 01	4.22E 01	4.22E 01	4.20E 01	4.20E 01	4.20E 01	4.21E 01	4.21E 01	4.22E 01
U236	N,G	8.42E 00	8.43E 00	8.44E 00	8.44E 00	8.44E 00	8.44E 00	8.44E 00	8.44E 00	8.44E 00	8.44E 00	8.44E 00	8.44E 00
U238	N,G	8.99E-01	9.00E-01	9.00E-01	9.00E-01	9.01E-01	9.01E-01	9.05E-01	9.06E-01	9.06E-01	9.06E-01	9.06E-01	9.06E-01
NP237	N,G	3.16E 01	3.16E 01	3.16E 01	3.16E 01	3.16E 01	3.16E 01	3.14E 01	3.14E 01	3.14E 01	3.14E 01	3.14E 01	3.14E 01
PU238	N,G	3.07E 01	3.07E 01	3.07E 01	3.08E 01	3.08E 01	3.08E 01	3.07E 01	3.08E 01	3.08E 01	3.08E 01	3.09E 01	3.09E 01
PU238	FISS	2.34E 00	2.34E 00	2.34E 00	2.34E 00	2.34E 00	2.34E 00	2.34E 00	2.34E 00	2.35E 00	2.35E 00	2.35E 00	2.35E 00
PU239	N,G	5.43E 01	5.43E 01	5.44E 01	5.44E 01	5.44E 01	5.45E 01	5.29E 01	5.29E 01	5.30E 01	5.31E 01	5.32E 01	5.33E 01
PU239	FISS	9.72E 01	9.72E 01	9.72E 01	9.73E 01	9.74E 01	9.75E 01	9.50E 01	9.52E 01	9.53E 01	9.54E 01	9.55E 01	9.57E 01
PU240	N,G	1.24E 02	1.23E 02	1.23E 02	1.22E 02	1.22E 02	1.21E 02	1.07E 02	1.07E 02	1.07E 02	1.06E 02	1.06E 02	1.06E 02
PU241	N,G	3.49E 01	3.49E 01	3.49E 01	3.49E 01	3.50E 01	3.50E 01	3.44E 01	3.44E 01	3.45E 01	3.45E 01	3.46E 01	3.46E 01
PU241	FISS	1.07E 02	1.07E 02	1.07E 02	1.07E 02	1.07E 02	1.07E 02	1.06E 02	1.06E 02	1.06E 02	1.06E 02	1.06E 02	1.06E 02
PU242	N,G	3.11E 01	3.11E 01	3.11E 01	3.11E 01	3.11E 01	3.11E 01	3.09E 01	3.09E 01	3.09E 01	3.09E 01	3.09E 01	3.09E 01
AM241	N,G	9.59E 01	9.59E 01	9.58E 01	9.58E 01	9.58E 01	9.59E 01	9.38E 01	9.39E 01	9.39E 01	9.39E 01	9.40E 01	9.41E 01
AM241	NGEX	1.19E 01	1.18E 01	1.18E 01	1.18E 01	1.18E 01	1.19E 01	1.16E 01	1.16E 01	1.16E 01	1.16E 01	1.16E 01	1.16E 01
AM243	N,G	1.99E 00	1.98E 00	1.98E 00	1.97E 00	1.96E 00	1.96E 00	1.98E 00	1.97E 00	1.97E 00	1.96E 00	1.96E 00	1.95E 00
AM243	NGEX	3.78E 01	3.76E 01	3.75E 01	3.74E 01	3.73E 01	3.72E 01	3.75E 01	3.74E 01	3.74E 01	3.73E 01	3.72E 01	3.71E 01
CM242	N,G	5.69E 00	5.70E 00	5.70E 00	5.70E 00	5.70E 00	5.71E 00	5.72E 00	5.73E 00	5.73E 00	5.73E 00	5.73E 00	5.73E 00
CM244	N,G	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.57E 00	3.59E 00	3.59E 00	3.59E 00	3.59E 00	3.59E 00	3.59E 00

[a]N,G = (n,gamma) to a ground state; NGEX = (n,gamma) to an excited state; FISS = (n,fission).

Table C.6. Cross sections as a function of burnup for a BWR-PuPu

Nuclide	Cross section[a] type	Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 0.0	0.317	0.638	0.966	1.301	1.642	1.642	1.989	2.336	2.687	3.040	3.397
U234	N,G	1.75E 01	1.76E 01	1.76E 01	1.77E 01	1.77E 01	1.77E 01	1.80E 01	1.80E 01	1.81E 01	1.81E 01	1.82E 01	1.82E 01
U235	N,G	6.88E 00	6.92E 00	6.97E 00	7.01E 00	7.05E 00	7.10E 00	7.33E 00	7.38E 00	7.43E 00	7.48E 00	7.54E 00	7.59E 00
U235	FISS	2.54E 01	2.57E 01	2.59E 01	2.62E 01	2.64E 01	2.67E 01	2.78E 01	2.81E 01	2.84E 01	2.87E 01	2.90E 01	2.93E 01
U236	N,G	8.45E 00	8.45E 00	8.45E 00	8.45E 00	8.45E 00	8.45E 00	8.49E 00	8.50E 00	8.50E 00	8.50E 00	8.50E 00	8.50E 00
U238	N,G	8.30E-01	8.31E-01	8.31E-01	8.32E-01	8.33E-01	8.34E-01	8.44E-01	8.45E-01	8.46E-01	8.47E-01	8.48E-01	8.49E-01
NP237	N,G	2.52E 01	2.53E 01	2.55E 01	2.56E 01	2.58E 01	2.59E 01	2.63E 01	2.64E 01	2.66E 01	2.67E 01	2.69E 01	2.71E 01
PU238	N,G	1.75E 01	1.77E 01	1.79E 01	1.81E 01	1.83E 01	1.85E 01	1.94E 01	1.96E 01	1.98E 01	2.01E 01	2.03E 01	2.05E 01
PU238	FISS	2.04E 00	2.05E 00	2.05E 00	2.06E 00	2.06E 00	2.06E 00	2.08E 00	2.09E 00	2.09E 00	2.10E 00	2.10E 00	2.11E 00
PU239	N,G	2.89E 01	2.93E 01	2.96E 01	3.00E 01	3.03E 01	3.07E 01	3.25E 01	3.29E 01	3.33E 01	3.37E 01	3.42E 01	3.46E 01
PU239	FISS	5.21E 01	5.27E 01	5.33E 01	5.39E 01	5.46E 01	5.53E 01	5.84E 01	5.92E 01	5.99E 01	6.07E 01	6.15E 01	6.23E 01
PU240	N,G	4.42E 01	4.43E 01	4.45E 01	4.46E 01	4.47E 01	4.49E 01	4.65E 01	4.67E 01	4.69E 01	4.71E 01	4.72E 01	4.74E 01
PU241	N,G	1.90E 01	1.92E 01	1.94E 01	1.96E 01	1.99E 01	2.01E 01	2.12E 01	2.15E 01	2.17E 01	2.20E 01	2.23E 01	2.25E 01
PU241	FISS	6.14E 01	6.20E 01	6.26E 01	6.33E 01	6.39E 01	6.46E 01	6.77E 01	6.85E 01	6.92E 01	7.00E 01	7.08E 01	7.16E 01
PU242	N,G	2.46E 01	2.46E 01	2.46E 01	2.46E 01	2.46E 01	2.46E 01	2.45E 01	2.46E 01	2.46E 01	2.46E 01	2.46E 01	2.46E 01
AM241	N,G	6.13E 01	6.18E 01	6.23E 01	6.28E 01	6.33E 01	6.39E 01	6.59E 01	6.65E 01	6.71E 01	6.77E 01	6.83E 01	6.90E 01
AM241	NGEX	7.58E 00	7.64E 00	7.70E 00	7.76E 00	7.83E 00	7.89E 00	8.15E 00	8.22E 00	8.29E 00	8.37E 00	8.44E 00	8.52E 00
AM243	N,G	1.55E 00	1.56E 00	1.56E 00	1.56E 00	1.56E 00	1.56E 00	1.58E 00	1.58E 00	1.59E 00	1.59E 00	1.59E 00	1.59E 00
AM243	NGEX	2.95E 01	2.96E 01	2.96E 01	2.96E 01	2.96E 01	2.96E 01	3.01E 01	3.01E 01	3.01E 01	3.02E 01	3.02E 01	3.02E 01
CM242	N,G	5.13E 00	5.14E 00	5.15E 00	5.15E 00	5.16E 00	5.16E 00	5.24E 00	5.25E 00	5.25E 00	5.26E 00	5.27E 00	5.28E 00
CM244	N,G	3.24E 00	3.24E 00	3.25E 00	3.25E 00	3.25E 00	3.26E 00	3.30E 00	3.30E 00	3.31E 00	3.31E 00	3.32E 00	3.32E 00

Nuclide	Cross section[a] type	Cross sections (barns) for fuel burnup [MWd(t)/g-atom heavy metal present] of: 3.397	3.794	4.187	4.576	4.962	5.345	5.345	5.774	6.194	6.604	7.005	7.396
U234	N,G	1.86E 01	1.87E 01	1.87E 01	1.88E 01	1.88E 01	1.89E 01	1.91E 01	1.91E 01	1.92E 01	1.92E 01	1.93E 01	1.94E 01
U235	N,G	8.00E 00	8.06E 00	8.12E 00	8.18E 00	8.24E 00	8.31E 00	8.63E 00	8.67E 00	8.71E 00	8.76E 00	8.81E 00	8.87E 00
U235	FISS	3.15E 01	3.18E 01	3.22E 01	3.25E 01	3.29E 01	3.33E 01	3.52E 01	3.53E 01	3.56E 01	3.58E 01	3.61E 01	3.64E 01
U236	N,G	8.53E 00	8.53E 00	8.54E 00	8.54E 00	8.54E 00	8.54E 00	8.48E 00	8.49E 00	8.51E 00	8.52E 00	8.53E 00	8.53E 00
U238	N,G	8.60E-01	8.61E-01	8.63E-01	8.64E-01	8.66E-01	8.67E-01	8.73E-01	8.74E-01	8.76E-01	8.77E-01	8.79E-01	8.80E-01
NP237	N,G	2.80E 01	2.82E 01	2.84E 01	2.85E 01	2.87E 01	2.89E 01	2.94E 01	2.96E 01	2.97E 01	2.98E 01	3.00E 01	3.02E 01
PU238	N,G	2.23E 01	2.25E 01	2.28E 01	2.31E 01	2.34E 01	2.36E 01	2.51E 01	2.53E 01	2.55E 01	2.57E 01	2.59E 01	2.61E 01
PU238	FISS	2.15E 00	2.15E 00	2.16E 00	2.16E 00	2.17E 00	2.17E 00	2.21E 00	2.21E 00	2.21E 00	2.22E 00	2.22E 00	2.23E 00
PU239	N,G	3.78E 01	3.83E 01	3.88E 01	3.93E 01	3.98E 01	4.03E 01	4.30E 01	4.32E 01	4.36E 01	4.39E 01	4.43E 01	4.47E 01
PU239	FISS	6.80E 01	6.89E 01	6.97E 01	7.06E 01	7.15E 01	7.25E 01	7.74E 01	7.78E 01	7.84E 01	7.91E 01	7.98E 01	8.05E 01
PU240	N,G	5.06E 01	5.08E 01	5.10E 01	5.13E 01	5.15E 01	5.17E 01	5.47E 01	5.49E 01	5.51E 01	5.53E 01	5.55E 01	5.57E 01
PU241	N,G	2.45E 01	2.49E 01	2.52E 01	2.55E 01	2.58E 01	2.61E 01	2.79E 01	2.80E 01	2.82E 01	2.85E 01	2.87E 01	2.90E 01
PU241	FISS	7.74E 01	7.83E 01	7.92E 01	8.01E 01	8.11E 01	8.20E 01	8.70E 01	8.75E 01	8.81E 01	8.87E 01	8.95E 01	9.03E 01
PU242	N,G	2.45E 01	2.45E 01	2.45E 01	2.46E 01	2.46E 01	2.46E 01	2.41E 01	2.42E 01	2.42E 01	2.43E 01	2.43E 01	2.43E 01
AM241	N,G	7.31E 01	7.38E 01	7.45E 01	7.53E 01	7.60E 01	7.67E 01	7.99E 01	8.03E 01	8.08E 01	8.13E 01	8.19E 01	8.25E 01
AM241	NGEX	9.04E 00	9.13E 00	9.21E 00	9.30E 00	9.39E 00	9.48E 00	9.87E 00	9.92E 00	9.98E 00	1.00E 01	1.01E 01	1.02E 01
AM243	N,G	1.63E 00	1.63E 00	1.64E 00	1.64E 00	1.64E 00	1.65E 00	1.66E 00	1.66E 00	1.67E 00	1.67E 00	1.67E 00	1.68E 00
AM243	NGEX	3.10E 01	3.11E 01	3.11E 01	3.12E 01	3.12E 01	3.13E 01	3.15E 01	3.16E 01	3.17E 01	3.17E 01	3.18E 01	3.19E 01
CM242	N,G	5.36E 00	5.37E 00	5.38E 00	5.39E 00	5.40E 00	5.41E 00	5.43E 00	5.45E 00	5.46E 00	5.47E 00	5.48E 00	5.49E 00
CM244	N,G	3.37E 00	3.37E 00	3.38E 00	3.39E 00	3.39E 00	3.40E 00	3.42E 00	3.42E 00	3.43E 00	3.44E 00	3.44E 00	3.45E 00

[a]N,G = (n,gamma) to a ground state; NGEX = (n,gamma) to an excited state; FISS = (n,fission).

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