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Once-Through CANDU Reactor Models for the ORIGEN2 Computer Code

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ONCE-THROUGH CANDU REACTOR MODELS FOR THE ORIGEN2 COMPUTER CODE

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GLOSSARY

GWd	Gigawatt-days = 10^9 watt-days
MWd	Megawatt-days = 10^6 watt-days
MTIHM	Metric tons (= 10^6 g) of initial heavy metal
CANDU	CANada Deuterium Uranium reactor; a D_2O-moderated, pressurized-heavy-water–cooled Canadian reactor
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 the principal constituent; also called a fuel pin or a fuel rod
Fuel assembly	A grouping of fuel elements that remains intact 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
NATU	Natural uranium (0.7115 wt % ^{235}U)
SEU	Slightly enriched uranium
MSBR	Molten-Salt Breeder Reactor

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ABSTRACT

Reactor physics calculations have led to the development of two CANDU reactor models for the ORIGEN2 computer code. The model CANDUs are based on (1) the existing once-through fuel cycle with feed comprised of natural uranium and (2) a projected slightly enriched (1.2 wt % ^{235}U) fuel cycle. The reactor models are based on cross sections taken directly from the reactor physics codes. Descriptions of the reactor models, as well as values for the ORIGEN2 flux parameters THERM, RES, and FAST, are 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 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

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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 Oak Ridge National Laboratory (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 the code had originally been intended. These applications were generally much more specific than the early ORNL generic fuel cycle studies. Many 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 numerous 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 one of several reports describing the various aspects of the ORIGEN update effort. Other reports that have been issued previously document (1) a revised version of the ORIGEN computer code, designated ORIGEN2,[4,5] (2) updated decay and photon data libraries,[6] (3) updated U-Pu cycle PWR and BWR models,[7] and (4) alternative-fuel-cycle (thorium cycles and extended burnup) PWR models.[8] Work is currently under way on LMFBR models.

1.1.2 Scope

This report is concerned with a description of model CANDUs operating on once-through, ^{235}U-enriched fuels and the methods used to generate the information for these models. The basic CANDU reactor model was based on the Gentilly 2 reactor,[9] which is typical of modern CANDU reactors. The reactor has a net electrical output of 638 MW(e) and a fuel assembly containing 37 fuel elements.

The two CANDU reactor models developed for the ORIGEN2 computer code are fueled with (1) natural (0.7115 wt % ^{235}U) UO_2 [CANDU-NATU] and (2) slightly enriched (1.2 wt % ^{235}U) UO_2 [CANDU-SEU], both operating on a once-through fuel cycle. These models are only differentiated by their fuel cycles since the basic reactor design was held constant. The natural uranium fuel cycle is currently in use, whereas the slightly enriched fuel cycle is presently being studied for a future implementation. The character strings given in brackets are the abbreviated names of the CANDU fuels that will be used in this report.

The fundamental objective of this work was to revise ORIGEN2 so that it could 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 ORIGEN2 had to be able to use cross sections that resulted from the processing of existing compilations such as ENDF/B.[10,11]

The generation of the information required for these reactor models began with the gathering and initial processing of existing raw cross section data into a library of 27 neutron energy groups that could be used by a modular system of reactor physics codes[12] (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 ORIGEN2 but having a negligible 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 two fuel types being considered (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 ORIGEN2. The cross sections in the larger 27-group library mentioned previously were then collapsed to one-group

cross sections using a typical neutron spectrum that 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 that yielded new values of the ORIGEN2 flux parameters — THERM, RES, and FAST.

The ORIGEN2 depletion results were then compared 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 ORIGEN2 reactor models for model CANDU reactors operating on once-through, ^{235}U-enriched fuels. The specific types of information developed for CANDU-NATU and CANDU-SEU are as follows:

1. 27-energy-group neutron spectra;
2. one-group, burnup-dependent cross sections for the major actinides;
3. one-group, typical cross sections for 234 nuclides (including the actinides);
4. new values for the ORIGEN2 flux parameters THERM, RES, and FAST;
5. recommended initial heavy-metal compositions of fresh fuel;
6. recommended initial metal compositions of fuel-assembly structural materials; and
7. recommended minor constituent concentrations for both the fuel material and the structural materials.

Using this information and the ORIGEN2 computer code, depletion calculations were made for each of the fuel types. The results of these depletion calculations were compared to predicted discharge compositions obtained from literature sources.

In general, the results of the ORIGEN2 depletion calculations agreed very well with the literature values for the CANDU-NATU reactor. Agreement in the case of the CANDU-SEU was acceptable, although not as good. This was attributed to the fact that the SEU fuel cycle is not actually being used and has not yet been sufficiently studied to ensure accurate depletion calculations. Based on the comparisons of ORIGEN2 vs the literature that 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.

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

This section describes the sources of the unprocessed cross-section data that were used in developing the U-Pu cycle CANDU reactor models and the initial processing of these cross sections into an AMPX[12] master cross-section library. Only a brief description of the data and its processing is given in this report since the processing methods used were nearly identical to those used for the U-Pu cycle LWRs. For a more detailed discussion, the reader is referred to the report[7] describing those models.

The cross sections in the master, multigroup libraries were put into a 27-energy-group structure, of which 13 groups were thermal. The group structure used in this study is listed in Appendix A.

2.1 Scope and Source of Cross Section Data

The cross sections that were used for the CANDU reactor models were collected into two groups. The first group consists of all of the fission product isotopes having cross section data available for processing. The nuclides in this group were obtained from ref. 10 and are listed in Table 1. The second group consists of all of the activation products, actinides, moderators, and structural materials considered in this work. This second

Table 1. ORIGEN2 update group 1 (fission-product) nuclides[a]

^{72}Ge	^{73}Ge	^{74}Ge	^{75}As	^{76}Ge	^{76}Se	^{77}Se	^{78}Se
^{79}Br	^{80}Se	^{80}Kr	^{81}Br	^{82}Se	^{82}Kr	^{83}Kr	^{84}Kr
^{85}Kr	^{85}Rb	^{86}Kr	^{86}Rb	^{86}Sr	^{87}Rb	^{87}Sr	^{88}Sr
^{89}Sr	^{89}Y	^{90}Sr	^{90}Y	^{90}Zr	^{91}Y	^{91}Zr	^{93}Zr
^{94}Zr	^{94}Nb	^{94}Mo	^{95}Zr	^{95}Nb	^{95}Mo	^{96}Zr	^{96}Mo
^{97}Mo	^{99}Mo	^{99}Tc	^{99}Ru	^{100}Mo	^{100}Ru	^{101}Ru	^{102}Ru
^{103}Ru	^{103}Rh	^{104}Ru	^{104}Pd	^{105}Ru	^{105}Rh	^{105}Pd	^{106}Ru
^{106}Pd	^{107}Pd	^{107}Ag	^{108}Pd	^{108}Cd	^{109}Ag	^{110}Pd	^{110}Cd
^{111}Ag	^{111}Cd	^{112}Cd	^{113}Cd	^{113}In	^{114}Cd	^{115m}Cd	^{115}In
^{115}Sn	^{116}Cd	^{116}Sn	^{117}Sn	^{118}Sn	^{119}Sn	^{120}Sn	^{121}Sb
^{122}Sn	^{122}Te	^{123}Sn	^{123}Sb	^{123}Te	^{124}Sn	^{124}Sb	^{124}Te
^{125}Sn	^{125}Sb	^{125}Te	^{126}Sn	^{126}Sb	^{126}Te	^{127m}Te	^{127}I
^{128}Te	^{128}Xe	^{129m}Te	^{129}I	^{129}Xe	^{130}Te	^{130}I	^{130}Xe
^{131}I	^{131}Xe	^{132}Te	^{132}Xe	^{133}Xe	^{133}Cs	^{134}Xe	^{134}Cs
^{134}Ba	^{135}I	^{135}Xe	^{135}Cs	^{135}Ba	^{136}Xe	^{136}Cs	^{136}Ba
^{137}Cs	^{137}Ba	^{138}Ba	^{139}La	^{140}Ba	^{140}La	^{140}Ce	^{141}Ce
^{141}Pr	^{142}Ce	^{142}Pr	^{142}Nd	^{143}Ce	^{143}Pr	^{143}Nd	^{144}Ce
^{144}Nd	^{145}Nd	^{146}Nd	^{147}Nd	^{147}Pm	^{147}Sm	^{148}Nd	^{148}Pm
^{148m}Pm	^{148}Sm	^{149}Pm	^{149}Sm	^{150}Nd	^{150}Sm	^{151}Pm	^{151}Sm
^{151}Eu	^{152}Sm	^{152}Eu	^{153}Sm	^{154}Sm	^{154}Eu	^{154}Gd	^{155}Eu
^{155}Gd	^{156}Eu	^{156}Gd	^{157}Eu	^{157}Gd	^{158}Gd	^{159}Tb	^{160}Gd
^{160}Tb	^{160}Dy	^{161}Dy	^{162}Dy	^{163}Dy	^{164}Dy	^{165}Ho	^{166}Er
^{167}Er							

[a]All cross sections taken from ref. 10.

group was divided into subgroups A and B. Subgroup A consists of those nuclides that are important in the reactor calculation. The membership and sources of subgroup A nuclides are given in Table 2. Subgroup B consists of those members of the second group that are present in the reactor in minor amounts but that may be important for some reason. The membership and sources of subgroup B nuclides are given in Table 3.

Certain ^{238}U cross section parameters were adjusted according to a procedure supplied by Benjamin[14] in order to obtain agreement with the experimental data. Changes in the neutron capture cross section were made between 0.00025 and 0.7067 eV, in the neutron and radiation widths for the low-lying s-wave resonances between 6.65 and 165 eV, and in the unresolved resonance parameters.

2.2 Resonance Nuclide Processing in NPTXS

The nuclides included in subgroup 2A (neutronically important) can be separated into two classes — resonance nuclides and nonresonance nuclides. Nuclides in the first class are those that contribute significantly to the resonance absorption in the system(s) of interest. For the purposes of this study (U-Pu cycle CANDU reactors), the following nuclides were put in this class: $^{235,236}U$, ^{238}U, and $^{239-242}Pu$.

The AMPX[12] module NPTXS is used to process the resonance parameters contained in the ENDF/B data for those nuclides listed above. In the resolved resonance range, each resonance is described by a few parameters (resonance energy, neutron widths, etc.) that 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 that 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 that span the unresolved range. The point cross sections for the resonance nuclides are used by NEWXLACS[12] (Sect. 2.3) to calculate the scattering matrix.

Table 2. ORIGEN2 subgroup 2A nuclides (included in depletion calculation)

Nuclide	Ref.	Nuclide	Ref.
^{1}H	13	^{238}Np	11
^{2}H	13	^{239}Np	11
^{16}O	13	^{238}Pu	13
Zr	13	^{239}Pu	11
Zr-2[a]	13	^{240}Pu	11
^{93}Nb	13	^{241}Pu	13
^{234}U	13	^{242}Pu	13
^{235}U	11	^{243}Pu	11
^{236}U	13	^{241}Am	11
^{237}U	11	^{242m}Am	11
^{238}U	11, 14	^{243}Am	11
^{237}Np	13	^{244}Cm	11

[a]Zircaloy-2.

Table 3. ORIGEN2 subgroup 2B nuclides (not included in depletion calculation)

Nuclide	Ref.	Nuclide	Ref.
^{6}Li	13	^{236}Pu	11
^{7}Li	13	^{237}Pu	11
^{10}B	13	^{244}Pu	11
^{11}B	13	^{242}Am	11
^{12}C	13	^{243}Cm	11
^{14}N	13	^{245}Cm	11
^{27}Al	13	^{246}Cm	11
Cr	13	^{247}Cm	11
Fe	13	^{248}Cm	11
^{59}Co	13	^{249}Bk	11
Ni	13	^{249}Cf	11
^{230}Th	11	^{250}Cf	11
^{232}Th	11	^{251}Cf	11
^{233}Pa	11	^{252}Cf	11
^{232}U	11	^{253}Cf	11
^{233}U	11	^{253}Es	11

The background cross sections from file 3 of ENDF/B are added to those calculated in the resonance regions and output to a file that 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_o that is indicative of the system in which these cross sections will be used. (σ_o is the total cross section of the surrounding medium.)

2.3 AMPX Master Interface Preparation by NEWXLACS

The purpose of NEWXLACS[12] is to create an AMPX master interface (i.e., a master cross section library) that 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 connected 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 are 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 cross sections 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.

3. MULTIGROUP DEPLETION CALCULATIONS

The 27-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 CANDU 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. 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 that comprise the input to and output from each computer code are contained in the curved figures. As is evident from Fig. 1, there are three principal cross section processing codes — NITAWL,[12] XSDRNPM,[12] and CITATION.[15] 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. 1. The first of these is to perform pin-cell neutron energy spectrum calculations at three different burnups using NITAWL and XSDRNPM, which are used to account for the 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 are exposed to a significantly different neutron spectrum from those on the periphery of the assembly. This difference is due to the varying characteristics of the surrounding medium. 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.

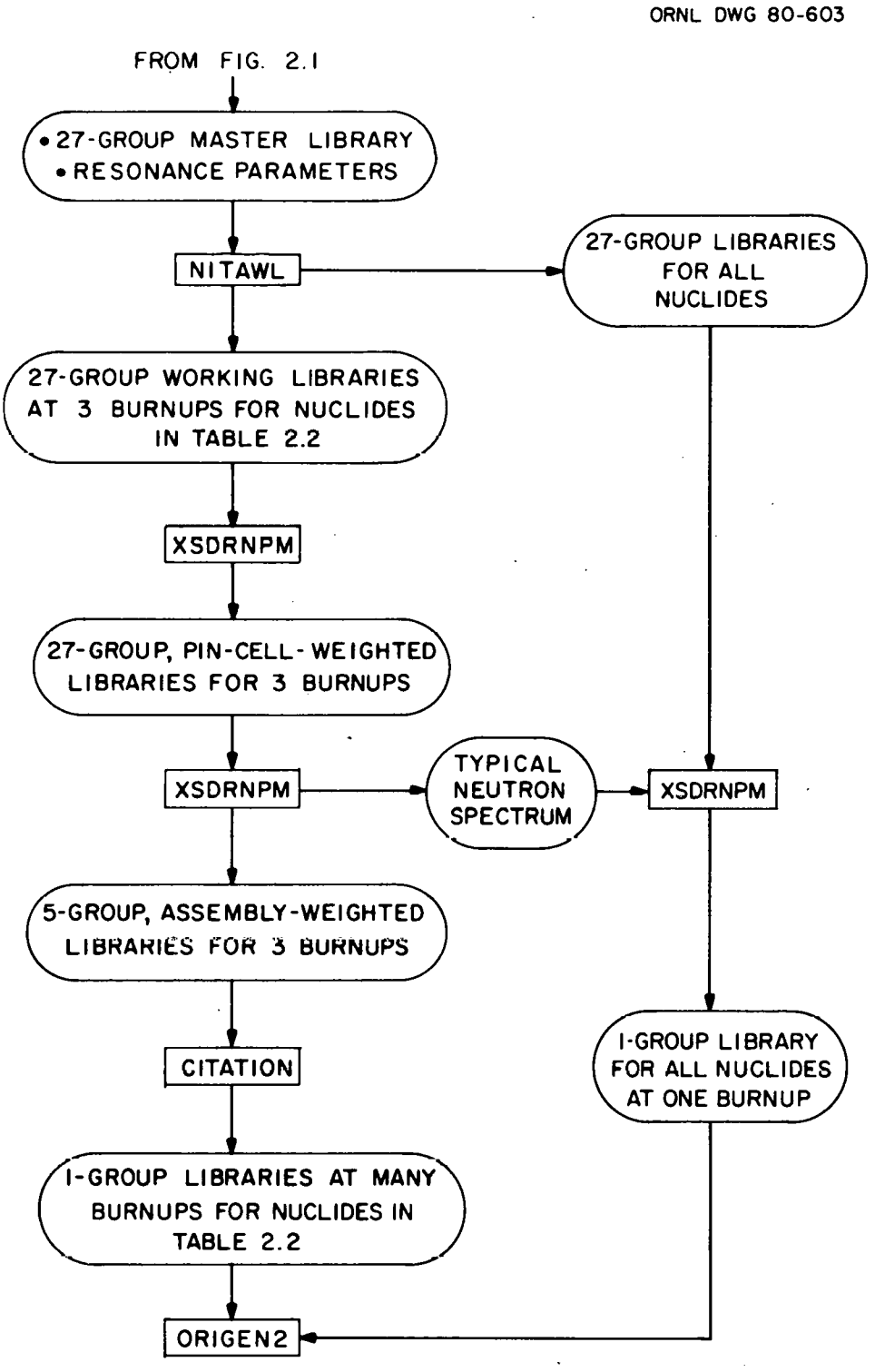

Fig. 1. Procedure for processing AMPX master interface cross sections into ORIGEN2 cross sections.

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 1 to 3 to one-group cross sections for ORIGEN2. A description of this processing sequence is given in Sects. 3.2 to 3.4.

3.1 General Description of Cross Section Processing Codes

3.1.1 NITAWL

The NITAWL computer code is used to account for resonance self-shielding effects in nuclides with resonance parameters; that is, given information about the fuel region of a pin cell, such as the Doppler temperature, moderator concentration, pin-cell dimensions, and resonance nuclide concentrations, NITAWL accounts for (1) Doppler broadening of the resonances, and (2) the fact that the effect of large resonances is diminished because 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 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, varies significantly within the fuel material and within the moderator. This library is designated as a weighted library. The principal input data required are a physical description of the fuel element or assembly

(i.e., the dimensions of each zone), the concentration of each nuclide within each zone, and a working library from NITAWL, corresponding to the nuclide concentrations.

3.1.3 CITATION

The CITATION computer code performs 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 and the 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 ORIGEN2.

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 27-energy-group master cross section library and its associated resonance parameters into a 27-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 27 energy groups, since they are to be used in subsequent XSDRNPM assembly-cell calculations (Sect. 3.3). Pin-cell calculations were made at three different fuel compositions, corresponding to burnups of 1.1, 3.4, and 5.7 GWd/MTIHM for the CANDU fueled with natural uranium and to burnups of 4.5, 12, and 20 GWd/MTIHM for the CANDU fueled with slightly enriched uranium.

3.2.2 Pin-cell description

The pin cells for both reactor models consisted of seven concentric zones: fuel, gap, clad, D_2O coolant, homogenized assembly, pressure and calandria tubes, and D_2O moderator. The D_2O was assumed to contain 0.25 wt % H_2O. A "white," or isotropic-reflecting, boundary condition was imposed on the outer boundary of the cell to simulate the presence of the cell in an infinite medium of other similar cells. The pin-cell dimensions are given in Table 4.

The nuclide number densities that are used as input to the NITAWL and XSDRNPM codes were based on linear interpolation of depletion data given as a function of burnup for CANDU reactors.[16-19] As noted above, the pin-cell calculations were made using compositions that corresponded to three different burnups. These burnup-dependent cross section libraries were developed to account for the variation in the reactor neutron energy spectrum and nuclide concentrations, and thus the effective nuclide cross sections, during the irradiation of the fuel. The calculation of the nuclide densities assumed a fuel density of 97% of the theoretical 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 result of the pin-cell calculations was 27-energy-group, pin-cell-averaged cross sections for the actinides, moderator, and principal structural materials at multiple burnups. The 27-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.

3.3 Assembly-Cell Calculations

3.3.1 General approach

The assembly-cell calculations use the XSDRNPM code and the 27-energy-group, pin-cell-weighted cross sections previously produced by XSDRNPM (Sect. 3.2.2) to determine effective cross sections and neutron energy spectra for the two CANDU reactor models. This procedure was used to account for the fact that the neutron energy spectrum on the periphery

Table 4. Pin-cell dimensions for the CANDU reactor[a]

Region	Outer radius (cm)	Volume fraction[b]
Fuel	0.5945	0.4894
Gap	0.6066	0.0201
Clad	0.6530	0.0809
Coolant	0.8498	0.4095
Smeared assembly	5.1689	
Pressure and calandria tubes	6.5875	
Moderator	16.1217	

[a]Data based on ref. 9.

[b]Fraction of pin cell only, excluding surrounding assembly, tubes, and moderator.

of a CANDU fuel assembly, which is near the large expanse of moderator between the fuel channels, is different from the neutron energy spectrum in the center of the assembly. 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 (Sect. 3.2.1).

3.3.2 Assembly-cell description

The assembly cells for all of the assembly calculations consisted of five concentric zones: homogenized fuel assembly, pressure tube, nitrogen annulus, calandria tube, and moderator. The dimensions of these zones are given in Table 5. 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.

3.3.3 Results of assembly-cell calculations

The results of the assembly-cell calculations for the CANDU reactors are as follows:

1. five-energy-group, cell-averaged cross sections for the nuclides in Table 2 at multiple burnups;
2. 27-energy-group, neutron spectra at multiple burnups;
3. one-group, cell-averaged fission product, activation product, and actinide cross sections at one burnup; and
4. cell-averaged parameters to enable THERM, RES, and FAST values to be calculated.

The five-energy-group, cell-averaged cross sections (i.e., weighted libraries) for the principal actinides are used as input to the CITATION computer code.

Typical neutron-energy spectra calculated by the XSDRNPM code for CANDU fuels comprised of natural or slightly enriched uranium are listed and graphed in Appendix A. These spectra correspond to a single burnup

Table 5. Assembly-cell dimensions for the CANDU reactor[a]

Region	Outer radius (cm)	Volume fraction
Assembly	5.1689	0.1028
Pressure tube	5.6007	0.0179
N_2 annulus	6.4478	0.0397
Calandria tube	6.5875	0.0070
Moderator	16.1217	0.8330

[a]Data based on ref. 9.

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 Tables 1 to 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 multigroup 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 are given in a separate publication.[4] 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. Hence, for the purposes of updating the ORIGEN2 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 6. This group structure consists of one thermal group and four fast groups spanning the energy range of 1.0×10^{-5} to 20.0 MeV.

The once-through, continuous refueling used for CANDU reactors results in not having to use multiple dimensions to accommodate fuel management considerations. However, the heterogeneous nature of the CANDU lattice (i.e., assemblies surrounded by large amounts of D_2O) requires the use of

Table 6. Energy-group structure for CITATION cross section

Group number	Upper boundary (eV)	Lower boundary (eV)
1	2.00000×10^{7}	1.70000×10^{4}
2	1.70000×10^{4}	5.50000×10^{2}
3	5.50000×10^{2}	3.00000×10^{1}
4	3.00000×10^{1}	3.05000×10^{0}
5	3.05000×10^{9}	1.00000×10^{-5}

a one-dimensional (radial) calculation to accommodate the radial spectrum changes. The contents and dimensions of the radial zones are the same as those given in Table 5 for the XSDRNPM assembly-cell calculations.

The initial fuel compositions were 0.71 and 1.2 wt % ^{235}U in stoichiometric ^{238}U oxide for the natural uranium and slightly enriched uranium fuels, respectively. For the purposes of the CITATION calculations, the specific power in the fuel was assumed to be 34.08 MW/MTIHM, and the fuel burnups for the natural uranium and slightly enriched uranium systems were taken to be 6800 and 23,200 MWd/MTIHM, respectively. The CITATION depletion calculation explicitly considered all of the nuclides listed in Table 2.

3.4.2 Results of CITATION depletion calculations

The results of the CITATION depletion calculations for LWRs are as follows:

1. one-energy-group, effective cross sections for the nuclides listed in Table 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 ORIGEN2 computer code as a function of burnup, as described in Sect. 4.1. The CANDU-NATU and CANDU-SEU 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 ORIGEN2 cross section library.

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 ORIGEN2 computer code. However, the full incorporation of new CANDU reactor models into ORIGEN2 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) calculation of new values of THERM, RES, and FAST, (2) specification of the initial structural material and fuel compositions, (3) comparison of ORIGEN2 fuel-depletion calculations with independent calculations to validate the new reactor models, and (4) a summary description of the new reference reactor models.

The specification of the input compositions to ORIGEN2 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 ORIGEN2 results are in good agreement with CITATION results. The results of depletion calculations were obtained from the literature, and ORIGEN2 depletion calculations were performed on a basis that was consistent with each literature reference. These two results are presented together in Sect. 4.3. In general, agreement was very good, although some discrepancies were noted in the case of CANDU-SEU fuel.

A summary description of the results of ORIGEN fuel-depletion calculations for the CANDU models is given in Sect. 4.3.

4.1 ORIGEN2 Flux Parameters THERM, RES, and FAST

The results of the assembly-cell neutron spectrum calculation described in Sect. 3.3, when cast into a two-group (thermal and fast) structure, allow the calculation of the ORIGEN2 flux parameters THERM, RES, and FAST. The equations used to calculate these parameters and their bases are described elsewhere[4] and will not be repeated here. The values of THERM, RES, and FAST that are appropriate for ORIGEN2 (on a total flux basis) are given in Table 7. Values appropriate for the older versions of ORIGEN (thermal-flux-based) are given in Table 8.

4.2 Input Compositions and Masses

There are three different composition-related aspects to be considered when specifying the CANDU fuel-input composition:

1. the characterization and composition of the CANDU fuel assembly,
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 description and composition

A CANDU fuel assembly is markedly different from that in an LWR, as is evident from Fig. 2. The dimensions and constitution of the CANDU assembly are given in Table 9. The assembly is much smaller than an LWR assembly and is comprised entirely of Zircaloy-4 and UO_2 (ignoring the small amounts of brazing materials). The composition of the Zircaloy-4 is given in Table 10, and the composition of the fuel pellets will be given in the following section.

4.2.2 Composition of LWR oxide fuels

The nonactinide elements present in fresh CANDU fuels are comprised of numerous trace elements (<100 ppm) plus the oxygen present in the actinide oxide fuel material. A typical set of nonactinide element concentrations in fresh CANDU oxide fuel, which is assumed to be the same as that for LWR fuels, is given in Table 11. The values in Table 11 generally

Table 7. Values of THERM, RES, and FAST to be used in ORIGEN2

Fuel type	THERM	RES	FAST	Total flux (neutrons/cm^2·s)
CANDU-NATU	0.2900	0.0173	0.0514	2.35×10^{14}
CANDU-SEU	0.3791	0.0215	0.0531	2.14×10^{14}

Table 8. Values of THERM, RES, and FAST to be used with older thermal-flux-based versions of ORIGEN

Fuel Type	THERM	RES	FAST	Total flux (neutrons/cm^2·s)
CANDU-NATU	0.4340	0.0259	0.0769	1.57×10^{14}
CANDU-SEU	0.5883	0.0333	0.0825	1.38×10^{14}

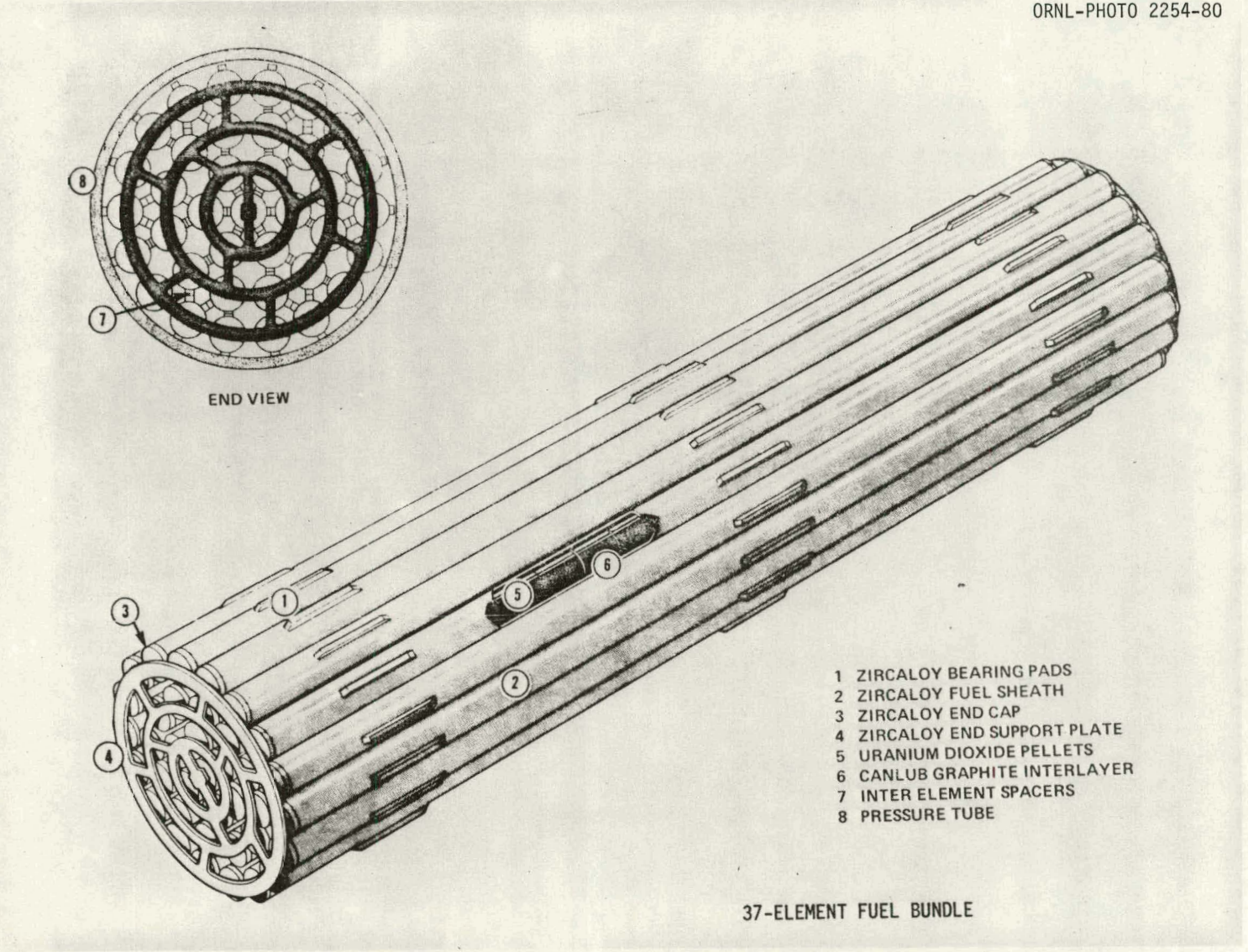

Fig. 2. CANDU fuel assembly containing 37 fuel elements.

Table 9. Physical characteristics of CANDU fuel assemblies[a]

Overall assembly length, cm	49.5
Diameter, cm	10.24
Fuel element length, cm	49.5
Active fuel height, cm	48.0
Fuel element OD, cm	1.308
Fuel element array	
Outer ring	18
Intermediate ring	12
Inner ring	6
Center	1
Fuel elements per assembly	37
Assembly total weight, kg	23.5
Uranium/assembly, kg	18.7
UO_2/assembly, kg	21.2
Zircaloy/assembly, kg	232
Nominal volume/assembly, cm^3	4077

[a]Source: ref. 9.

Table 10. Composition of Zircaloy-4[a]

Element	Atomic number	Composition (g/MTIHM)
H	1	13
B	5	0.33
C	6	120
N	7	80
O	8	950
Al	13	24
Si	14	0
P	15	0
S	16	35
Ti	22	20
V	23	20
Cr	24	1,250
Mn	25	20
Fe	26	2,250
Co	27	0.3
Ni	28	20
Cu	29	20
Zr	40	979,110
Nb	41	0
Mo	42	0
Cd	48	0.25
Sn	50	16,000
Hf	72	78
W	74	20
U	92	0.2
Density, g/cm^3		6.56

[a]Data obtained from refs. 20 to 22.

Table 11. Assumed nonactinide composition of CANDU oxide fuels[a]

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

[a]Data obtained from refs. 21, 23 tc 26.

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

[c]Stoichiometric quantity for UO_2 fuel.

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 11.

The initial heavy-metal composition of the natural-uranium CANDU fuel is as follows: ^{234}U, 55 g/MTIHM; ^{235}U, 7115 g/MTIHM; and ^{238}U, 992,830 g/MTIHM. For the slightly enriched CANDU fuel, the initial composition is ^{234}U, 102 g/MTIHM; ^{235}U, 12,000 g/MTIHM; and ^{238}U, 987,898 g/MTIHM. The reference burnups for these fuels are 7500 and 20,900 MWd/MTIHM, respectively.

4.3 Comparison of ORIGEN2 and Literature Depletion Calculations

This section serves to verify the accuracy of the calculational procedures used in developing the alternative fuel cycle PWR models described herein by comparing them with results calculated by an independent organization. The comparisons of the ORIGEN2 and literature depletion calculations are summarized in Tables 12 and 13 for the CANDU-NATU and CANDU-SEU, respectively. The agreement between ORIGEN2 and the literature values is very good for the CANDU-NATU, as shown in Table 12. The agreement is not as good for the CANDU-SEU. However, the results documented herein for the CANDU-SEU were deemed acceptable for the following reasons:

1. The CANDU-SEU literature values were calculated using the same correlations as those for the CANDU-NATU.
2. The CANDU-SEU case represents a significant extrapolation of the well-known CANDU-NATU values (burnup increased by a factor of 3).
3. The CANDU-SEU is a relatively new system that has been neither implemented nor extensively studied.
4. The methodology used gave very good results for the CANDU-NATU.

Table 12. Comparison of ORIGEN2 CANDU-NATU depletion calculation with literature values

Nuclide	Discharged CANDU-NATU fuel composition (g/MTIHM)		
	ORIGEN2 calculation	Ref. 16	Ref. 27
^{235}U	2,474	2,560	2,514
^{236}U	698	689	696
^{238}U	986,000	986,150	986,300
^{237}Np	22.6		21.9
^{238}Pu	2.45		2.36
$^{239}Pu^{a}$	2.701	2,495	2,416
^{240}Pu	915	926	936
^{241}Pu	177	169	155
^{242}Pu	38.7	44.7	40.9
Total Pu	3,834	3,635	3,550
^{241}Am	1.55		0.97
^{243}Am	1.20		1.40
^{242}Cm	0.31		0.22
^{244}Cm	0.087		0.094
Initial ^{235}U	7,115	7,100	7,116
Burnup, MWd/MTIHM	6,720	6,728	6,703

^{a}Includes ^{239}Np.

Table 13. Comparison of ORIGEN2 CANDU-SEU depletion calculation with literature values

Nuclide	Discharged CANDU-SEU fuel composition (g/MTIHM)			
	ORIGEN2 calculation	Ref. 16	ORIGEN2 calculation	Ref. 27
^{235}U	685	821	830	819
^{236}U	1,479	1,457	1,623	1,593
^{238}U	970,000	970,290	969,600	969,800
^{237}Np	94.9		98.5	103
^{238}Pu	25.1		25.2	27.3
$^{239}Pu^{a}$	3,107	2,743	3,103	2,736
^{240}Pu	2,313	2,332	2,272	2,408
^{241}Pu	584	525	568	501
^{242}Pu	446	479	413	450
Total Pu	6,475	6,079	6,381	6,122
^{241}Am	11.4		11.1	6.6
^{243}Am	45.0		40.0	42.0
^{242}Cm	5.98		5.63	4.3
^{244}Cm	10.5		8.89	9.4
Initial ^{235}U	10,891	10,891	12,000	12,000
Burnup, MWd/MTIHM	20,600	20,569	20,828	20,861

^{a}Includes ^{239}Np.

4.4 Summary Description of the ORIGEN2 CANDU Reactor Models

Fuel depletion calculations have been performed for the two CANDU fuel types considered in this report — NATU and SEU — using the input compositions given in Sect. 4.2. The results of these calculations, including irradiation conditions and measurements of the uranium and plutonium contents of the fresh and spent fuels, are presented in Table 14. Many of the values in this table are given in terms of both kilograms per refueling cycle and kilograms per year, assuming an 80% capacity factor for the latter.

Table 14. Summary of ORIGEN LWR model characteristics

Parameter	CANDU-NATU	CANDU-SEU
Electric power, MW(e)	638	638
Thermal power, MW(t)	2,180	2,180
Average specific power,[a] MW(t)/MTIHM	25.57	25.57
Average fuel burnup, MWd/MTIHM[b]	7,500	20,900
Irradiation duration, full-power days	293.3	817.4
Charge, kg/MTIHM (kg/yr at 80% capacity factor)		
^{235}U	7.115 (604)	12.00 (365)
Total U	1,000 (84,875)	1,000 (30,457)
Discharge, kg/MTIHM (kg/yr at 80% capacity factor)		
^{235}U	2.191 (186)	0.836 (25.4)
Total U	988 (83,865)	972 (29,604)
Fissile Pu[c]	3.003 (255)	3.672 (112)
Total Pu[d]	4.051 (344)	6.388 (195)
Total (U + Pu)	992 (84,209)	978 (29,799)
Total heavy metal	992 (84,221)	979 (29,805)

[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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6. APPENDIXES

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Appendix A: 27-Energy-Group Neutron Spectra Graphs and Listings

Appendix A contains graphs and listings of 27-energy-group neutron spectra for the two once-through uranium-cycle CANDU fuel types considered in this report as calculated by the XSDRNPM code. The units of the neutron fluxes are flux per unit lethargy. All spectra are averaged over the assembly cell (see Sect. 3.3).

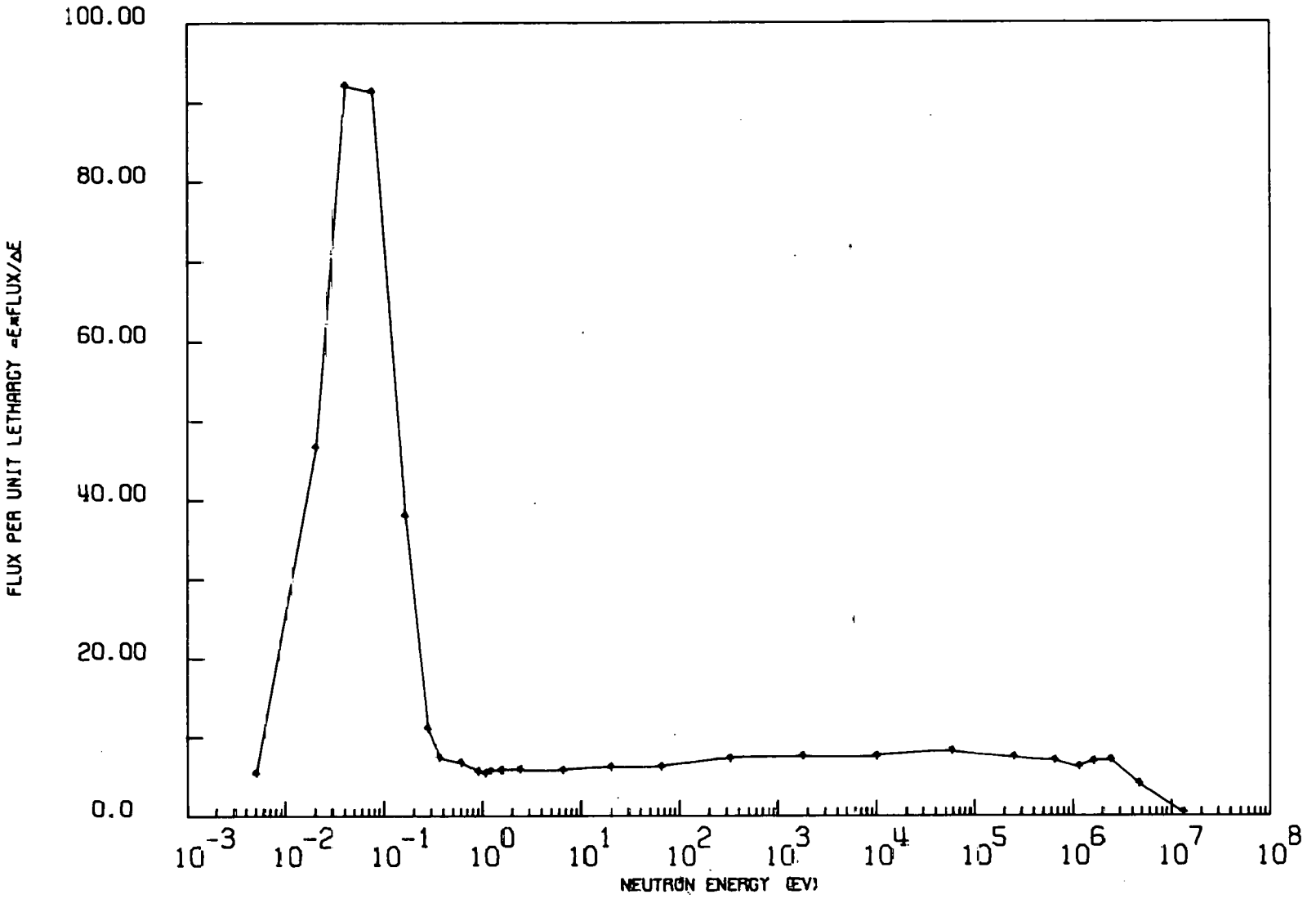

Fig. A.1. Neutron energy spectrum in a CANDU-NATU.

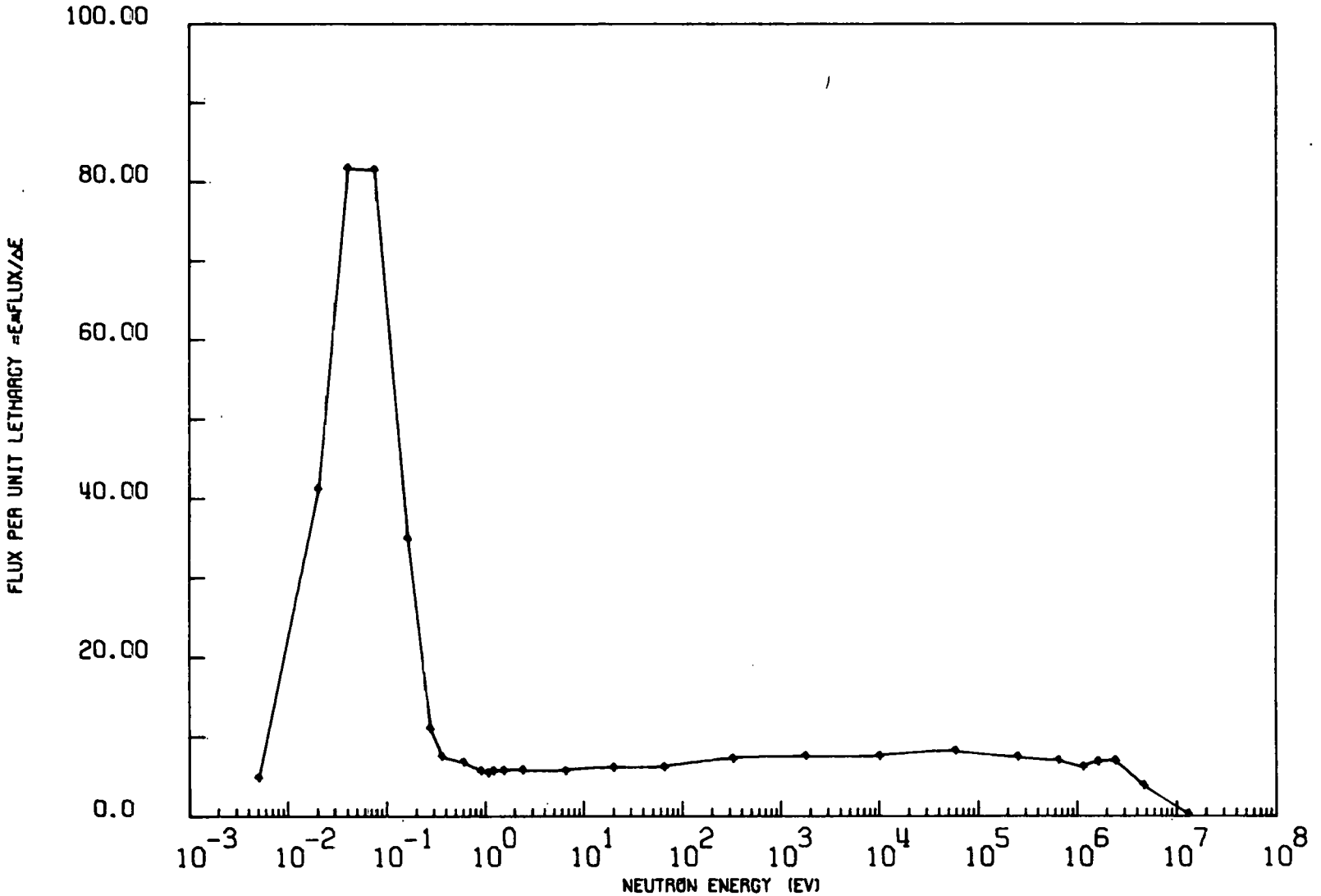

Fig. A.2. Neutron energy spectrum in a CANDU-SEU.

Table A.1. Flux per unit lethargy for once-through CANDU fuels

Energy group	Energy group boundaries (eV) Upper	Lower	CANDU-NATU	CANDU-SEU
1	2.00000E 07	6.43400E 06	3.20038E-01	2.98796E-01
2	6.43400E 06	3.00000E 06	4.01405E 00	3.86611E 00
3	3.00000E 06	1.85000E 06	7.07220E 00	6.93295E 00
4	1.85000E 06	1.40000E 06	6.88505E 00	6.81882E 00
5	1.40000E 06	9.00000E 05	6.15843E 00	6.14573E 00
6	9.00000E 05	4.00000E 05	6.96775E 00	6.97166E 00
7	4.00000E 05	1.00000E 05	7.40882E 00	7.41907E 00
8	1.00000E 05	1.70000E 04	8.18232E 00	8.18577E 00
9	1.70000E 04	3.00000E 03	7.50735E 00	7.50128E 00
10	3.00000E 03	5.50000E 02	7.51274E 00	7.49543E 00
11	5.50000E 02	1.00000E 02	7.23139E 00	7.19985E 00
12	1.00000E 02	3.00000E 01	6.15458E 00	6.11451E 00
13	3.00000E 01	1.00000E 01	6.10806E 00	6.05582E 00
14	1.00000E 01	3.04999E 00	5.70953E 00	5.65800E 00
15	3.04999E 00	1.77000E 00	5.81728E 00	5.76577E 00
16	1.77000E 00	1.29999E 00	5.71695E 00	5.66953E 00
17	1.29999E 00	1.12999E 00	5.65617E 00	5.63741E 00
18	1.12999E 00	1.00000E 00	5.34844E 00	5.40144E 00
19	1.00000E 00	8.00000E-01	5.58224E 00	5.61770E 00
20	8.00000E-01	4.00000E-01	6.74727E 00	6.73788E 00
21	4.00000E-01	3.25000E-01	7.37141E 00	7.45884E 00
22	3.25000E-01	2.25000E-01	1.09864E 01	1.09566E 01
23	2.25000E-01	1.00000E-01	3.79753E 01	3.48730E 01
24	1.00000E-01	5.00000E-02	9.13188E 01	8.14754E 01
25	5.00000E-02	3.00000E-02	9.21167E 01	8.16579E 01
26	3.00000E-02	1.00000E-02	4.65857E 01	4.11695E 01
27	1.00000E-02	1.00000E-05	5.37855E 00	4.75944E 00

Appendix B: One-Group, Spectrum-Averaged Cross Sections for Once-Through, Uranium-Cycle CANDU Fuels

Table B.1. One-group, spectrum-averaged cross sections for once-through fuel-cycle CANDU reactors

Nuclide	Cross section type[a]	Cross section, barns	
		CAND-NATU	CAND-SEU
H - 1[b]	N,G	2.02E-01	1.96E-01
H - 1[c]	N,G	1.01E-01	8.69E-02
H - 2[b]	N,G	3.17E-04	3.07E-04
H - 2[c]	N,G	1.60E-04	1.37E-04
LI- 6	N,A	5.30E 02	5.09E 02
LI- 7	N,G	2.02E-02	1.94E-02
B - 10	N,A	2.16E 03	2.08E 03
B - 11	N,G	2.82E-03	2.71E-03
C - 12	N,G	1.90E-03	1.83E-03
N - 14	N,P	1.03E 00	9.89E-01
O - 16	N,A	4.38E-04	4.49E-04
AL- 27	N,G	1.31E-01	1.26E-01
CO- 59	N,G	2.22E 01	2.14E 01
GE- 72	N,G	5.55E-01	5.35E-01
GE- 73	N,G	9.52E 00	9.29E 00
GE- 74	N,G	2.20E-01	2.13E-01
GE- 76	N,G	1.04E-01	1.03E-01
AS- 75	N,G	3.51E 00	3.50E 00
SE- 76	N,G	4.70E 01	4.52E 01
SE- 77	N,G	2.35E 01	2.26E 01
SE- 78	N,G	3.09E-01	3.07E-01
SE- 80	N,G	3.53E-01	3.41E-01
SE- 82	N,G	2.63E-02	2.55E-02
BR- 79	N,G	8.62E 00	8.55E 00
BR- 81	N,G	2.45E 00	2.46E 00
KR- 80	N,G	8.98E 00	8.75E 00
KR- 82	N,G	1.98E 01	1.93E 01
KR- 83	N,G	1.15E 02	1.11E 02
KR- 84	N,G	1.16E-01	1.19E-01
KR- 85	N,G	9.35E-01	9.00E-01
KR- 86	N,G	3.62E-02	3.51E-02
RB- 85	N,G	3.70E-01	3.69E-01
RB- 86	N,G	3.13E 00	3.06E 00
RB- 87	N,G	1.07E-01	1.07E-01
SR- 86	N,G	1.65E 00	1.59E 00
SR- 87	N,G	1.09E 01	1.07E 01
SR- 88	N,G	3.40E-03	3.28E-03
SR- 89	N,G	2.39E-01	2.30E-01
SR- 90	N,G	4.99E-01	4.79E-01
Y - 89	N,G	7.15E-01	6.88E-01
Y - 90	N,G	1.99E 00	1.92E 00
Y - 91	N,G	7.94E-01	7.65E-01
ZR- 90	N,G	6.18E-02	6.01E-02
ZR- 91	N,G	6.76E-01	6.62E-01
ZR- 92[d]	N,G	9.88E-02	9.26E-02
ZR- 92[e]	N,G	1.59E-01	1.54E-01
ZR- 93	N,G	2.12E 00	2.12E 00
ZR- 94	N,G	3.78E-02	3.71E-02

Table B.1. (cont'd)

Nuclide	Cross section type[a]	Cross section, barns	
		CAND-NATU	CAND-SEU
ZR- 95	N,G	3.74E-01	3.71E-01
ZR- 96	N,G	1.13E-01	1.20E-01
NB- 93[d]	N,G	6.67E-01	6.42E-01
NB- 93[e]	N,G	8.15E-01	8.04E-01
NB- 94	N,G	9.60E 00	9.45E 00
NB- 95	N,G	1.25E 00	1.25E 00
MO- 94	N,G	2.79E-02	2.89E-02
MO- 95	N,G	9.97E 00	9.79E 00
MO- 96	N,G	9.26E-01	9.32E-01
MO- 97	N,G	1.49E 00	1.47E 00
MO- 98	N,G	2.06E-01	2.13E-01
MO- 99	N,G	1.45E 00	1.45E 00
MO-100	N,G	1.86E-01	1.88E-01
TC- 99	N,G	1.66E 01	1.66E 01
RU- 99	N,G	5.30E 00	5.36E 00
RU-100	N,G	3.35E 00	3.23E 00
RU-101	N,G	3.48E 00	3.53E 00
RU-102	N,G	7.87E-01	7.64E-01
RU-103	N,G	5.51E 00	5.43E 00
RU-104	N,G	3.68E-01	3.68E-01
RU-105	N,G	2.57E-01	2.64E-01
RU-106	N,G	1.22E-01	1.22E-01
RH-103	N,G	1.04E 02	1.02E 02
RH-105	N,G	6.76E 03	6.49E 03
PD-104	N,G	5.67E-01	5.85E-01
PD-105	N,G	9.35E 00	9.16E 00
PD-106	N,G	2.73E-01	2.78E-01
PD-107	N,G	6.76E 00	6.63E 00
PD-108	N,G	1.08E 01	1.08E 01
PD-110	N,G	2.62E-01	2.68E-01
AG-107	N,G	2.22E 01	2.15E 01
AG-109	N,G	7.56E 01	7.52E 01
AG-111	N,G	3.64E 00	3.72E 00
CD-108	N,G	6.84E-01	6.66E-01
CD-110	N,G	6.83E 00	6.64E 00
CD-111	N,G	1.43E 01	1.38E 01
CD-112	N,G	1.46E 00	1.43E 00
CD-113	N,G	1.70E 04	1.65E 04
CD-114	N,G	5.68E-01	5.88E-01
CD-115	N,G	2.06E 01	2.02E 01
CD-116	N,G	9.24E-02	9.44E-02
IN-113	N,G	1.05E 01	1.05E 01
IN-115	N,G	1.76E 02	1.76E 02
SN-115	N,G	2.76E 01	2.65E 01
SN-116	N,G	2.98E-01	3.13E-01
SN-117	N,G	1.77E 00	1.74E 00
SN-118	N,G	1.68E-01	1.75E-01
SN-119	N,G	1.33E 00	1.28E 00
SN-120	N,G	1.04E-01	1.03E-01
SN-122	N,G	1.14E-01	1.11E-01

Table B.1. (cont'd)

Nuclide	Cross section type[a]	Cross section, barns	
		CAND-NATU	CAND-SEU
SN-123	N,G	7.29E-02	7.63E-02
SN-124	N,G	2.03E-01	2.10E-01
SN-125	N,G	6.02E-01	6.13E-01
SN-126	N,G	1.66E-01	1.60E-01
SE-121	N,G	7.14E 00	7.25E 00
SB-123	N,G	4.76E 00	4.83E 00
SB-124	N,G	4.05E 00	3.94E 00
SB-125	N,G	9.09E-01	9.13E-01
SE-126	N,G	4.03E 00	3.97E 00
TE-122	N,G	2.91E 00	2.95E 00
TE-123	N,G	3.34E 02	3.32E 02
TE-124	N,G	3.86E 00	3.72E 00
TE-125	N,G	1.31E 00	1.31E 00
TE-126	N,G	7.68E-01	7.60E-01
TE-127	N,G	5.93E 00	5.77E 00
TE-128	N,G	1.66E-01	1.65E-01
TE-129	N,G	7.19E-01	7.03E-01
TE-130	N,G	1.65E-01	1.59E-01
TE-132	N,G	1.16E-03	1.12E-03
I -127	N,G	6.29E 00	6.36E 00
I -129	N,G	1.54E 01	1.48E 01
I -130	N,G	1.32E 01	1.30E 01
I -131	N,G	5.40E-01	5.36E-01
I -135	N,G	1.11E-02	1.07E-02
XE-128	N,G	2.13E 00	2.06E 00
XE-129	N,G	1.42E 01	1.41E 01
XE-130	N,G	3.45E 00	3.32E 00
XE-131	N,G	6.54E 01	6.45E 01
XE-132	N,G	2.80E-01	2.72E-01
XE-133	N,G	1.10E 02	1.06E 02
XE-134	N,G	1.47E-01	1.43E-01
XE-135	N,G	1.65E 06	1.59E 06
XE-136	N,G	8.93E-02	8.59E-02
CS-133	N,G	2.28E 01	2.26E 01
CS-134	N,G	8.00E 01	7.72E 01
CS-135	N,G	5.88E 00	5.77E 00
CS-136	N,G	1.48E 00	1.50E 00
CS-137	N,G	6.96E-02	6.79E-02
BA-134	N,G	1.64E 00	1.63E 00
BA-135	N,G	5.07E 00	5.07E 00
BA-136	N,G	2.61E-01	2.55E-01
BA-137	N,G	2.87E 00	2.76E 00
BA-138	N,G	1.94E-01	1.86E-01
BA-140	N,G	1.12E 00	1.10E 00
LA-139	N,G	5.11E 00	4.92E 00
LA-140	N,G	2.73E 00	2.76E 00
CE-140	N,G	3.18E-01	3.06E-01
CE-141	N,G	1.62E 01	1.56E 01
CE-142	N,G	5.32E-01	5.12E-01
CE-143	N,G	4.08E 00	4.01E 00

Table B.1. (cont'd)

Nuclide	Cross section type[a]	Cross section, barns	
		CAND-NATU	CAND-SEU
CE-144	N,G	5.85E-01	5.66E-01
PR-141	N,G	6.63E 00	6.40E 00
PR-142	N,G	1.36E 01	1.33E 01
PR-143	N,G	5.19E 01	5.02E 01
ND-142	N,G	1.03E 01	9.92E 00
ND-143	N,G	1.78E 02	1.71E 02
ND-144	N,G	2.07E 00	1.99E 00
ND-145	N,G	2.70E 01	2.64E 01
ND-146	N,G	8.27E-01	8.00E-01
ND-147	N,G	3.84E 01	3.81E 01
ND-148	N,G	1.75E 00	1.73E 00
ND-150	N,G	9.76E-01	9.72E-01
PM-147	N,G	1.38E 02	1.37E 02
PM-148	N,G	1.95E 03	1.97E 03
PM-148	N,G	6.73E 03	6.58E 03
PM-149	N,G	7.76E 02	7.46E 02
PM-151	N,G	4.16E 02	4.03E 02
SM-147	N,G	4.82E 01	4.77E 01
SM-148	N,G	2.01E 00	1.99E 00
SM-149	N,G	3.88E 04	3.73E 04
SM-150	N,G	6.08E 01	5.89E 01
SM-151	N,G	7.12E 03	6.80E 03
SM-152	N,G	1.65E 02	1.63E 02
SM-153	N,G	2.30E 02	2.26E 02
SM-154	N,G	3.63E 00	3.55E 00
EU-151	N,G	4.56E 03	4.39E 03
EU-152	N,G	1.13E 03	1.09E 03
EU-153	N,G	2.65E 02	2.57E 02
EU-154	N,G	7.40E 02	7.12E 02
EU-155	N,G	2.23E 03	2.14E 03
EU-156	N,G	2.89E 02	2.80E 02
EU-157	N,G	1.27E 02	1.24E 02
GD-154	N,G	5.09E 01	4.93E 01
GD-155	N,G	2.22E 04	2.12E 04
GD-156	N,G	3.23E 00	3.37E 00
GD-157	N,G	9.42E 04	9.00E 04
GD-158	N,G	2.56E 00	2.59E 00
GD-160	N,G	5.89E-01	5.85E-01
TB-159	N,G	2.25E 01	2.25E 01
TB-160	N,G	3.06E 02	2.95E 02
DY-160	N,G	6.53E 01	6.61E 01
DY-161	N,G	3.36E 02	3.24E 02
DY-162	N,G	1.58E 02	1.56E 02
DY-163	N,G	1.02E 02	1.01E 02
DY-164	N,G	1.34E 03	1.29E 03
HO-165	N,G	5.01E 01	4.96E 01
ER-166	N,G	2.16E 01	2.11E 01
ER-167	N,G	5.28E 02	5.21E 02
TH-230	N,G	2.89E 01	2.94E 01
TH-230	N,F	9.49E-03	1.00E-02

Table B.1. (cont'd)

Nuclide	Cross section type[a]	Cross section, barns	
		CAND-NATU	CAND-SEU
TH-232	N,G	5.60E 00	5.54E 00
TH-232	N,F	3.80E-03	3.97E-03
PA-231	N,G	1.52E 02	1.49E 02
PA-231	N,F	6.81E-02	7.17E-02
PA-233	N,G	1.89E 01	1.90E 01
PA-233	N,F	2.36E-02	2.47E-02
PA-233	N,GX	1.89E 01	1.90E 01
U -232	N,G	4.31E 01	4.19E 01
U -232	N,F	4.81E 01	4.70E 01
U -233	N,G	2.86E 01	2.78E 01
U -233	N,F	3.01E 02	2.91E 02
U -234	N,G	3.42E 01	3.03E 01
U -234	N,F	2.48E-01	2.62E-01
U -235	N,G	2.86E 01	2.45E 01
U -235	N,F	1.59E 02	1.34E 02
U -236	N,G	5.66E 00	5.76E 00
U -236	N,F	1.07E-01	1.12E-01
U -237	N,G	2.59E 02	2.49E 02
U -237	N,F	1.22E 00	1.19E 00
U -238	N,G	1.16E 00	1.06E 00
U -238	N,F	5.42E-02	5.64E-02
NP-237	N,G	5.86E 01	5.20E 01
NP-237	N,F	2.91E-01	3.06E-01
NP-238	N,G	1.10E 02	1.06E 02
NP-238	N,F	1.10E 03	1.06E 03
PU-236	N,G	9.24E 01	8.92E 01
PU-236	N,F	9.26E 01	8.94E 01
PU-237	N,G	2.97E 02	2.85E 02
PU-237	N,F	1.16E 03	1.12E 03
PU-238	N,G	1.43E 02	1.18E 02
PU-238	N,F	5.09E 00	4.42E 00
PU-239	N,G	1.23E 02	1.15E 02
PU-239	N,F	2.67E 02	2.40E 02
PU-240	N,G	1.45E 02	1.56E 02
PU-240	N,F	3.33E-01	3.52E-01
PU-241	N,G	1.16E 02	1.00E 02
PU-241	N,F	3.39E 02	2.95E 02
PU-242	N,G	2.38E 01	2.41E 01
PU-242	N,F	2.52E-01	2.65E-01
PU-243	N,G	5.23E 01	5.07E 01
PU-243	N,F	1.08E 02	1.04E 02
PU-244	N,G	2.98E 00	3.08E 00
PU-244	N,F	5.70E-02	6.04E-02
AM-241	N,G	3.15E 02	3.07E 02
AM-241	N,F	2.19E 00	2.15E 00
AM-241	N,GX	7.87E 01	7.68E 01
AM-242	N,G	1.36E 02	1.30E 02
AM-242	N,F	1.22E 03	1.17E 03
AM-242	N,G	8.49E 02	8.17E 02
AM-242	N,F	4.17E 03	4.01E 03

Table B.1. (cont'd)

Nuclide	Cross section type[a]	Cross section, barns	
		CAND-NATU	CAND-SEU
AM-243	N,G	3.81E 00	3.85E 00
AM-243	N,F	6.52E-02	6.89E-02
AM-243	N,GX	7.25E 01	7.31E 01
CM-241	N,G	1.38E 02	1.33E 02
CM-241	N,F	1.43E 03	1.38E 03
CM-242	N,G	1.21E 01	1.20E 01
CM-242	N,F	1.68E 00	1.61E 00
CM-243	N,G	3.64E 01	3.54E 01
CM-243	N,F	4.16E 02	4.03E 02
CM-244	N,G	2.32E 01	2.33E 01
CM-244	N,F	1.04E 00	1.02E 00
CM-245	N,G	2.02E 02	1.93E 02
CM-245	N,F	1.14E 03	1.10E 03
CM-246	N,G	2.56E 00	2.66E 00
CM-246	N,F	1.85E-01	1.93E-01
CM-247	N,G	4.28E 01	4.22E 01
CM-247	N,F	6.25E 01	6.17E 01
CM-248	N,G	5.80E 00	6.05E 00
CM-248	N,F	2.82E-01	2.95E-01
BK-249	N,G	9.39E 02	9.10E 02
BK-249	N,F	4.77E-02	5.03E-02
CF-249	N,G	2.72E 02	2.63E 02
CF-249	N,F	9.08E 02	8.76E 02
CF-250	N,G	1.14E 03	1.13E 03
CF-250	N,F	1.73E-01	1.85E-01
CF-251	N,G	1.57E 03	1.51E 03
CF-251	N,F	2.99E 03	2.88E 03
CF-252	N,G	1.18E 01	1.14E 01
CF-252	N,F	1.94E 01	1.88E 01
CF-253	N,G	2.00E 02	1.93E 02
CF-253	N,F	6.64E 02	6.41E 02
ES-253	N,G	1.53E 02	1.57E 02
ES-253	N,GX	1.07E 02	1.09E 02

[a] 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).

[b] Cross section in moderator.

[c] Cross section in coolant.

[d] Cross section in structural materials.

[e] Cross section in fuel.

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Appendix C: Listings of Selected Burnup-Dependent ORIGEN2 Cross Sections for Once-Through, Uranium-Cycle CANDU Fuels

Table C.1. Cross sections as a function of burnup for a CANDU-NATU

NUCLIDE	CROSS SECT. TYPE	FUEL BURNUP, MEGAWATT-DAYS (THERMAL) PER GRAM-ATOM HEAVY METAL PRESENT							
		0.0	0.130	0.260	0.391	0.521	0.652	0.782	0.913
U234	N,G	3.74E 01	3.72E 01	3.68E 01	3.65E 01	3.62E 01	3.63E 01	3.62E 01	3.61E 01
U235	N,G	3.17E 01	3.14E 01	3.09E 01	3.06E 01	3.03E 01	3.03E 01	3.01E 01	3.00E 01
U235	FISS	1.76E 02	1.74E 02	1.72E 02	1.69E 02	1.68E 02	1.68E 02	1.67E 02	1.66E 02
U236	N,G	6.17E 00	6.20E 00	6.24E 00	6.27E 00	6.30E 00	6.23E 00	6.25E 00	6.26E 00
U238	N,G	1.27E 00	1.26E 00	1.25E 00	1.25E 00	1.24E 00	1.25E 00	1.24E 00	1.24E 00
NP237	N,G	6.46E 01	6.40E 01	6.31E 01	6.24E 01	6.18E 01	6.20E 01	6.17E 01	6.14E 01
PU238	N,G	1.57E 02	1.55E 02	1.53E 02	1.51E 02	1.49E 02	1.50E 02	1.49E 02	1.48E 02
PU238	FISS	5.57E 00	5.52E 00	5.46E 00	5.41E 00	5.37E 00	5.40E 00	5.38E 00	5.36E 00
PU239	N,G	1.45E 02	1.43E 02	1.41E 02	1.39E 02	1.38E 02	1.33E 02	1.33E 02	1.32E 02
PU239	FISS	3.08E 02	3.04E 02	2.99E 02	2.95E 02	2.92E 02	2.87E 02	2.85E 02	2.83E 02
PU240	N,G	2.04E 02	2.02E 02	1.98E 02	1.95E 02	1.93E 02	1.66E 02	1.65E 02	1.64E 02
PU241	N,G	1.31E 02	1.29E 02	1.27E 02	1.25E 02	1.24E 02	1.23E 02	1.22E 02	1.21E 02
PU241	FISS	3.83E 02	3.79E 02	3.73E 02	3.68E 02	3.64E 02	3.61E 02	3.59E 02	3.57E 02
PU242	N,G	2.54E 01	2.51E 01	2.47E 01	2.44E 01	2.42E 01	2.48E 01	2.46E 01	2.45E 01
AM241	N,G	2.38E 02	2.35E 02	2.31E 02	2.28E 02	2.26E 02	2.24E 02	2.23E 02	2.21E 02
AM241	NGEX	2.94E 01	2.91E 01	2.86E 01	2.82E 01	2.79E 01	2.77E 01	2.75E 01	2.74E 01
AM243	N,G	3.75E 00	3.72E 00	3.66E 00	3.62E 00	3.58E 00	3.55E 00	3.53E 00	3.51E 00
AM243	NGEX	7.13E 01	7.06E 01	6.95E 01	6.87E 01	6.80E 01	6.75E 01	6.71E 01	6.68E 01
CM242	N,G	1.39E 01	1.38E 01	1.37E 01	1.36E 01	1.35E 01	1.34E 01	1.33E 01	1.33E 01
CM244	N,G	1.68E 01	1.68E 01	1.68E 01	1.68E 01	1.68E 01	1.67E 01	1.67E 01	1.67E 01

Table C.1. (cont'd)

NUCLIDE	CROSS SECT. TYPE	FUEL BURNUP, MEGAWATT-DAYS (THERMAL) PER GRAM-ATOM HEAVY METAL PRESENT							
		1.044	1.175	1.305	1.436	1.567	1.698	1.830	1.961
U234	N,G	3.60E 01	3.59E 01	3.61E 01	3.60E 01	3.60E 01	3.59E 01	3.59E 01	3.59E 01
U235	N,G	2.98E 01	2.97E 01	2.99E 01	2.98E 01	2.98E 01	2.97E 01	2.96E 01	2.96E 01
U235	FISS	1.65E 02	1.64E 02	1.66E 02	1.65E 02	1.65E 02	1.64E 02	1.64E 02	1.64E 02
U236	N,G	6.27E 00	6.28E 00	6.20E 00	6.20E 00	6.21E 00	6.22E 00	6.22E 00	6.23E 00
U238	N,G	1.24E 00	1.24E 00	1.25E 00	1.24E 00	1.24E 00	1.24E 00	1.24E 00	1.24E 00
NP237	N,G	6.11E 01	6.09E 01	6.14E 01	6.12E 01	6.11E 01	6.10E 01	6.09E 01	6.08E 01
PU238	N,G	1.47E 02	1.47E 02	1.48E 02	1.48E 02	1.47E 02	1.47E 02	1.47E 02	1.46E 02
PU238	FISS	5.34E 00	5.33E 00	5.37E 00	5.36E 00	5.35E 00	5.35E 00	5.34E 00	5.33E 00
PU239	N,G	1.31E 02	1.31E 02	1.28E 02	1.28E 02	1.27E 02	1.27E 02	1.27E 02	1.27E 02
PU239	FISS	2.82E 02	2.81E 02	2.78E 02	2.77E 02	2.76E 02	2.76E 02	2.75E 02	2.75E 02
PU240	N,G	1.63E 02	1.63E 02	1.50E 02	1.50E 02	1.50E 02	1.49E 02	1.49E 02	1.49E 02
PU241	N,G	1.21E 02	1.20E 02	1.20E 02	1.20E 02	1.20E 02	1.19E 02	1.19E 02	1.19E 02
PU241	FISS	3.55E 02	3.54E 02	3.53E 02	3.52E 02	3.51E 02	3.51E 02	3.50E 02	3.49E 02
PU242	N,G	2.44E 01	2.43E 01	2.48E 01	2.48E 01	2.47E 01	2.47E 01	2.46E 01	2.46E 01
AM241	N,G	2.20E 02	2.20E 02	2.19E 02	2.19E 02	2.18E 02	2.17E 02	2.17E 02	2.17E 02
AM241	NGEX	2.72E 01	2.71E 01	2.71E 01	2.70E 01	2.69E 01	2.69E 01	2.68E 01	2.68E 01
AM243	N,G	3.50E 00	3.49E 00	3.48E 00	3.47E 00	3.46E 00	3.46E 00	3.45E 00	3.44E 00
AM243	NGEX	6.65E 01	6.63E 01	6.61E 01	6.59E 01	6.58E 01	6.57E 01	6.55E 01	6.54E 01
CM242	N,G	1.32E 01	1.32E 01	1.32E 01	1.31E 01	1.31E 01	1.31E 01	1.31E 01	1.31E 01
CM244	N,G	1.67E 01	1.67E 01	1.67E 01	1.67E 01	1.67E 01	1.67E 01	1.67E 01	1.67E 01

Table C.2. Cross sections as a function of burnup for a CANDU-SEU

NUCLIDE	CROSS SECT. TYPE	FUEL BURNUP, MEGAWATT-DAYS (THERMAL) PER GRAM-ATOM HEAVY METAL PRESENT							
		0.0	0.366	0.732	1.098	1.466	1.833	2.202	2.571
U234	N,G	3.28E 01	3.27E 01	3.27E 01	3.27E 01	3.28E 01	3.35E 01	3.37E 01	3.38E 01
U235	N,G	2.63E 01	2.62E 01	2.61E 01	2.62E 01	2.63E 01	2.70E 01	2.71E 01	2.73E 01
U235	FISS	1.44E 02	1.43E 02	1.43E 02	1.43E 02	1.43E 02	1.48E 02	1.49E 02	1.50E 02
U236	N,G	6.35E 00	6.38E 00	6.39E 00	6.40E 00	6.40E 00	6.23E 00	6.23E 00	6.22E 00
U238	N,G	1.16E 00	1.16E 00	1.16E 00	1.16E 00	1.16E 00	1.18E 00	1.18E 00	1.19E 00
NP237	N,G	5.61E 01	5.58E 01	5.57E 01	5.58E 01	5.59E 01	5.67E 01	5.71E 01	5.74E 01
PU238	N,G	1.27E 02	1.26E 02	1.26E 02	1.26E 02	1.27E 02	1.31E 02	1.32E 02	1.33E 02
PU238	FISS	4.80E 00	4.78E 00	4.77E 00	4.77E 00	4.78E 00	4.92E 00	4.95E 00	4.97E 00
PU239	N,G	1.23E 02	1.23E 02	1.22E 02	1.22E 02	1.23E 02	1.20E 02	1.21E 02	1.22E 02
PU239	FISS	2.57E 02	2.56E 02	2.55E 02	2.55E 02	2.56E 02	2.56E 02	2.58E 02	2.60E 02
PU240	N,G	1.67E 02	1.67E 02	1.66E 02	1.66E 02	1.67E 02	1.34E 02	1.35E 02	1.36E 02
PU241	N,G	1.08E 02	1.07E 02	1.07E 02	1.07E 02	1.07E 02	1.09E 02	1.09E 02	1.10E 02
PU241	FISS	3.17E 02	3.15E 02	3.14E 02	3.14E 02	3.15E 02	3.20E 02	3.23E 02	3.25E 02
PU242	N,G	2.60E 01	2.58E 01	2.58E 01	2.58E 01	2.59E 01	2.56E 01	2.58E 01	2.60E 01
AM241	N,G	2.03E 02	2.02E 02	2.02E 02	2.02E 02	2.02E 02	2.03E 02	2.04E 02	2.06E 02
AM241	NGEX	2.51E 01	2.50E 01	2.49E 01	2.49E 01	2.50E 01	2.51E 01	2.53E 01	2.55E 01
AM243	N,G	3.24E 00	3.23E 00	3.22E 00	3.22E 00	3.23E 00	3.24E 00	3.26E 00	3.29E 00
AM243	NGEX	6.16E 01	6.14E 01	6.12E 01	6.12E 01	6.14E 01	6.16E 01	6.20E 01	6.24E 01
CM242	N,G	1.25E 01	1.25E 01	1.24E 01	1.24E 01	1.25E 01	1.25E 01	1.26E 01	1.26E 01
CM244	N,G	1.65E 01	1.65E 01	1.65E 01	1.65E 01	1.66E 01	1.66E 01	1.66E 01	1.66E 01

Table C.2. (cont'd)

NUCLIDE	CROSS SECT. TYPE	FUEL BURNUP, MEGAWATT-DAYS(THERMAL) PER GRAM-ATOM HEAVY METAL PRESENT							
		2.940	3.310	3.681	4.052	4.424	4.796	5.169	5.543
U234	N,G	3.40E 01	3.41E 01	3.49E 01	3.49E 01	3.50E 01	3.50E 01	3.51E 01	3.50E 01
U235	N,G	2.75E 01	2.76E 01	2.84E 01	2.85E 01	2.86E 01	2.86E 01	2.86E 01	2.86E 01
U235	FISS	1.51E 02	1.52E C2	1.56E 02	1.57E 02	1.57E 02	1.58E 02	1.58E 02	1.58E 02
U236	N,G	6.21E 00	6.21E 00	6.06E 00	6.05E 00	6.05E 00	6.05E 00	6.06E 00	6.06E 00
U238	N,G	1.19E 00	1.19E 00	1.21E 00	1.22E 00	1.22E 00	1.22E 00	1.22E 00	1.22E 00
NP237	N,G	5.78E 01	5.80E 01	5.89E 01	5.91E 01	5.92E 01	5.93E 01	5.93E 01	5.93E 01
PU238	N,G	1.34E 02	1.35E 02	1.40E 02	1.40E 02	1.41E 02	1.41E 02	1.41E 02	1.41E 02
PU238	FISS	4.99E 00	5.01E 00	5.15E 00	5.17E 00	5.18E 00	5.18E 00	5.18E 00	5.18E 00
PU239	N,G	1.23E 02	1.24E C2	1.22E 02	1.22E 02	1.23E 02	1.23E 02	1.23E 02	1.23E 02
PU239	FISS	2.61E 02	2.63E 02	2.64E 02	2.65E 02	2.65E 02	2.66E 02	2.66E 02	2.66E 02
PU240	N,G	1.37E 02	1.38E C2	1.25E 02	1.26E 02	1.26E 02	1.26E 02	1.26E 02	1.26E 02
PU241	N,G	1.11E 02	1.12E 02	1.14E 02	1.14E 02	1.14E 02	1.15E 02	1.15E 02	1.14E 02
PU241	FISS	3.27E 02	3.29E C2	3.34E 02	3.36E 02	3.37E 02	3.37E 02	3.37E 02	3.37E 02
PU242	N,G	2.62E 01	2.63E 01	2.57E 01	2.58E 01	2.59E 01	2.59E 01	2.59E 01	2.59E 01
AM241	N,G	2.07E 02	2.08E 02	2.09E 02	2.10E 02	2.11E 02	2.11E 02	2.11E 02	2.11E 02
AM241	NGEX	2.56E 01	2.58E 01	2.59E 01	2.60E 01	2.60E 01	2.61E 01	2.61E 01	2.61E 01
AM243	N,G	3.31E 00	3.32E 00	3.33E 00	3.35E 00	3.35E 00	3.36E 00	3.36E 00	3.36E 00
AM243	NGEX	6.28E 01	6.31E 01	6.34E 01	6.36E 01	6.37E 01	6.38E 01	6.38E 01	6.38E 01
CM242	N,G	1.27E 01	1.27E 01	1.28E 01	1.28E 01	1.28E 01	1.28E 01	1.28E 01	1.28E 01
CM244	N,G	1.66E 01	1.67E 01	1.67E 01	1.67E 01	1.67E 01	1.67E 01	1.67E 01	1.67E 01

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