1. CASMO-4

1.1 INTRODUCTION

1.1.1 General

The CASMO-4 code is a multi-group two-dimensional transport code developed by Studsvik, which is entirely written in FORTRAN 77. It is used for burnup calculations on Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) assemblies or pin cells. The code can deal with geometries consisting of cylindrical fuel rods of varying composition in a square pitch. It accommodates various rods such as those containing gadolinium, burnable absorber rods, cluster control rods etc. Typical fuel storage rack geometries can also be handled. This laboratory manual supplements the detailed code user's manual provided by Studsvik[1].

1.1.2 Library

Version 1.10 of the state-of-the-art, licensing quality CASMO-4 code is widely used. The neutron data libraries J2/E6 (revised January 22, 1998) based on the evaluated neutron data files JEF-2.2 developed at the Organization for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) Data Bank, and ENDF/B-6, the evaluated neutron data file developed at the Brookheaven National Nuclear Data Center, contain cross sections, decay constants and fission yields for 305 materials, most of which are individual nuclides. Microscopic cross sections are divided into 70 energy groups whose structure satisfies the following three requirements: 1) The 14 fast groups (10.0 MeV to 9118.0 eV) give enough detail in the fast energy region to account for leakage and fast fission accurately; 2) The 13 resonance groups (9118.0 eV to 4.00 eV) give correct flux levels for the calculation of resonance absorption; 3) The 43 thermal groups (below 4.00 eV) make the thermal cross sections independent of the weighting spectrum used for their generation. To be specific, J2/E6 contains absorption, fission, nufission, transport and scattering cross sections.

However, J2/E6 contains no gamma data since there is no gamma data included in the JEF-2.2 or ENDF/B-6 data files. Gamma data has been taken from the CLOSEUP library, which is based on ENDF/B-4 and was processed from the SCALE code, at Renssellaer Polytechnic Institute. Although the original library contains data for 18 gamma groups, a condensed library with 10 energy groups is adequate for all benchmark calculations and is recommended.

1.1.3 Benchmarks

In the development of CASMO-4 code, numerous benchmarks, such as single-assembly CASMO-4 versus Monte Carlo calculations, CASMO-4 versus critical experiments, etc., have been performed by Studsvik of America, Inc. In summary, fission distribution root-mean-squares between CASMO-4 and MCNP fall between 0.5% and 1% for BWR assemblies and below 0.5% for PWR assemblies, which is outstanding agreement.[3] Here three benchmarks will be presented: a CASMO-4 versus CASMO-3 benchmark[3], a CASMO-4 versus MCNP-3A benchmark (Doppler Coefficients Calculations)[4], and a CASMO-4 versus MOCUP (MCNP/ORIGEN Coupling Utility Programs) benchmark[6].

- CASMO-4 versus CASMO-3 Benchmark[3]

There are two major differences between CASMO-4 and CASMO-3: 1) In the CASMO-4 code gadolinium (Gd) depletion and other absorber depletions are done automatically without the need for auxiliary codes, whereas CASMO-3 uses MICBURN-3 to model the depletion of Gd externally. 2) In the CASMO-4 code the two-dimensional calculation is performed in the true heterogeneous geometry using the KRAM characteristics module whereas CASMO-3 performs a transmission probability solution to the transport equation using homogenized pin cells. Therefore, there should be no difference in the accuracy of the models for cores without burnable absorbers. The result in [3] shows strong agreement between CASMO-4 and CASMO-3 without Gd. When the burnable absorbers are present, the reactivity calculated by CASMO-4 is slightly higher than CASMO-3, which means that CASMO-4 depletes the Gd faster than CASMO-3. In this case CASMO-4 results have better accuracy due to the above two differences. Indeed the use of heavy burnable absorber loading has been a driving force behind the development of CASMO-4. Figure 1-1 illustrates the difference in assembly eigenvalues compared with exposure for a modern BWR assembly design with a high gadolinium concentration. The CASMO-4 code depletes the gadolinium faster than CASMO-3.

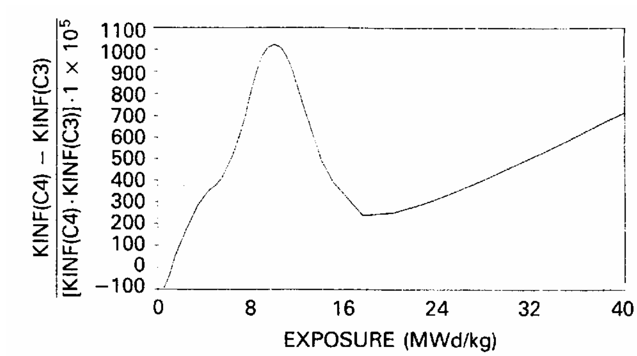

Figure 1-1. The BWR assembly depletion: CASMO-4 versus CASMO-3.[3]

Furthermore, CASMO-3 has been used extensively in past fuel management analyses.[5] The results of CASMO-4 and CASMO-3 agree well for an all-uranium fuel assembly.

- CASMO-4 versus MCNP-3A Benchmark[4]

The studied cases are pin cells representative of 17×17 PWR lattices at hot zero power (600K) and hot full power (900K) with 1400 ppm boron in the moderator. Five enrichments, from natural uranium to 3.9 wt% were studied. Geometry and compositions were taken from "Benchmark Calculations For The Doppler Coefficient of Reactivity," *Nuclear Science & Engineering*, **107**, 265, 1991 by R. D. Mosteller et al. The temperature for cladding and moderator is 600K in all cases and there are no thermal expansions. The results are shown in Table 1-1 and Table 1-2.

The results show good agreement of Doppler coefficients calculated by CASMO-4 and MCNP-3A. k_∞ versus enrichment is also in good agreement between CASMO-4 and MCNP-3A, especially considering that the cross-section library used with CASMO has a different origin than the MCNP-3A library (ENDF/B-V). This demonstrates the high accuracy of the resonance calculation and data in CASMO.

Table 1-1. Calculated k_∞.

Enrichment (%)	Temperature (K)	k_∞	$k_\infty^{CASMO} - k_\infty^{MCNP}$ (pcm)
		MCNP-3A	CASMO-4
0.7	600	0.6638 ± 0.0006	370
	900	0.6567 ± 0.0008	350
1.6	600	0.9581 ± 0.0006	380
	900	0.9484 ± 0.0006	410
2.4	600	1.0961 ± 0.0007	330
	900	1.0864 ± 0.0007	300
3.1	600	1.1747 ± 0.0007	240
	900	1.1641 ± 0.0006	270
3.9	600	1.2379 ± 0.0006	210
	900	1.2271 ± 0.0006	200

Table 1-2. Calculated Doppler Coefficients (pcm/K).

Enrichment (%)	Doppler Coefficient* (percentage difference from MCNP-3A)	
	MCNP-3A	CASMO-4
0.7	−5.4 ± 0.8	−5.5 (+1.6%)
1.6	−3.6 ± 0.3	−3.4 (−5.5%)
2.4	−2.7 ± 0.3	−2.8 (+2.9%)
3.1	−2.6 ± 0.2	−2.5 (−4.3%)
3.9	−2.4 ± 0.2	−2.4 (+0.3%)

∗ Doppler coefficient = $(k_{900} - k_{600})/(k_{900}\, k_{600} \times 300)$.

- CASMO-4 versus MOCUP Benchmark

A single pin cell model of a Westinghouse 17×17 assembly is employed to benchmark the CASMO-4 code involving the use of thorium. As to the description of MOCUP, users can refer to the latter part of this manual. The fuel contains a mixture of 75% thorium and 25% uranium with the 19.5% enrichment of U-235. In particular, the dimensions being used come from the hot dimensions generated by CASMO-4 as shown in Table 1-3. The initial fuel loading is shown in Table 1-4.

Table 1-3. Pin cell model parameters.[6]

Parameter	Cold	Hot
Fuel Temperature (K)	300	900
Power Density (kw/kg HM)	0.0	38.1347
Power Density (kW/liter)	0.0	107.284
Fuel Density (g/cm^3)	9.614	9.424
Cladding Temperature (K)	300.0	621.1
Cladding Density (g/cm^3)	6.550	6.505
Coolant Pressure (bar)	155.13	155.13
Coolant Temperature (K)	300.0	583.1
Coolant Density (g/cm^3)	1.003	0.705
Fuel Pellet Radius (mm)	4.096	4.1274
Cladding Inner Radius (mm)	4.178	4.1896
Cladding Outer Radius (mm)	4.750	4.7609
Pin Pitch (mm)	12.6	12.626

Table 1-4. Initial Compositions.[6]

	Nuclide	Weight Percent (wt %)	Number Density (1/cm^3)
Fuel	Th-232	65.909	1.61215×10^{22}
	U-234	0.034	8.24518×10^{18}
	U-235	4.291	1.03615×10^{21}
	U-238	17.740	4.22957×10^{21}
	O-16	12.026	4.26835×10^{22}
Cladding	Zircaloy-4	100	4.31438×10^{22}
Coolant	H-1	11.19	4.71053×10^{22}
	O-16	88.81	2.35662×10^{22}

The thorium cross section library for the MOCUP calculation was obtained from the University of Texas, Austin.[8] The results show very good agreement between CASMO-4 and MOCUP as shown in the following figure.

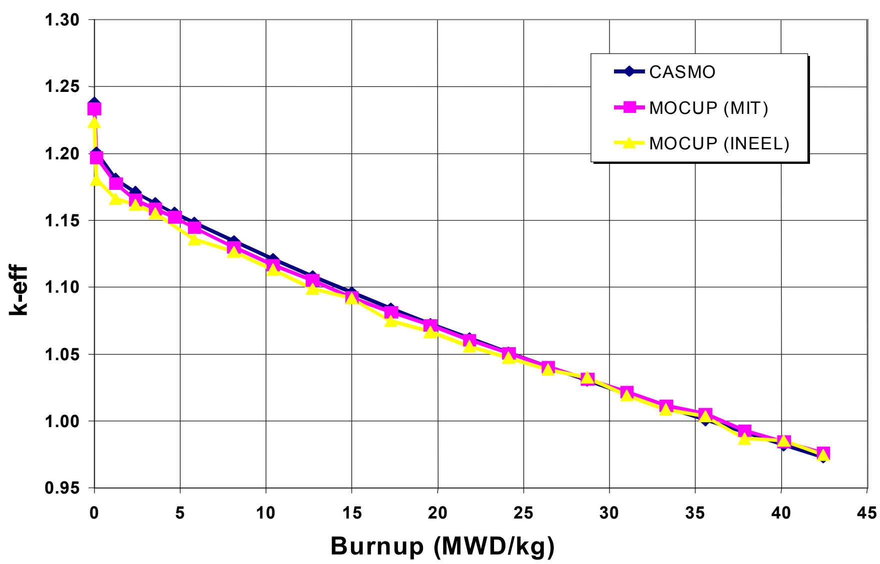

Figure 1-2. Comparison of CASMO-4 with MOCUP for pin cell benchmark calculation.[6]

1.2 METHODOLOGY

The CASMO-4 calculation process is shown in Figure 1-3. First, macroscopic cross sections are prepared for the following micro group calculations. The macroscopic group cross sections are calculated for the fuel assembly using the user's input data, the densities, geometries, compositions, and other operation parameters, along with the integrated nuclear data libraries.

The effective cross sections in the resonance energy region (4 eV to 9118 eV) for important resonance absorbers (U-235, U-236, U-238, Pu-239) are calculated using an equivalence theorem, which relates tabulated effective resonance integrals for each resonance absorber in each resonance group to the particular heterogeneous problem under consideration. The resonance integrals are used to calculate effective absorption and fission cross sections for these absorbers. The screening effect between different

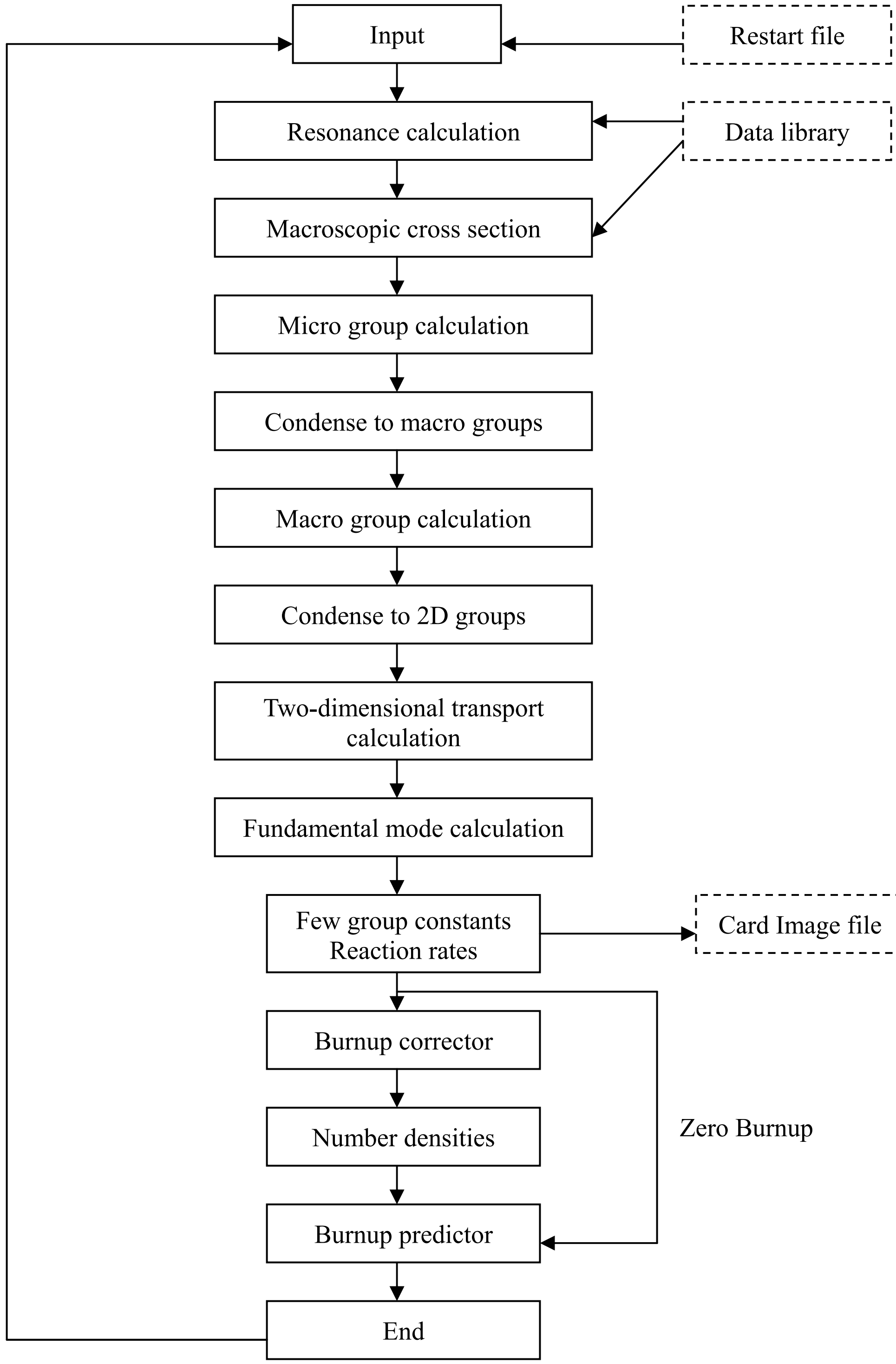

Figure 1-3. Flow diagram of CASMO-4.[2]

pins is accounted for by the use of Dancoff factors. The 1 eV resonance in Pu-240 and 0.3 eV resonance in Pu-239 are adequately covered by the concentration of thermal groups around these resonances and are consequently excluded from the special resonance treatment.

Then the cross sections prepared above are used in a series of micro group calculations to obtain detailed neutron energy spectra for energy condensation of the pin cells. Normally, a micro group calculation is performed for each pin type in the assembly. Collision probabilities are determined in a simplified geometry consisting of the different material regions of the pin type. Generally the fuel pins are modeled using either three or four regions (i.e. fuel, air, canning and coolant). If the lattice being modeled contains water gaps outside the fuel assembly, an additional region representing the water gap and box wall if necessary will be added to the cell.

The two-dimensional macro group calculation using an approximate and fast response matrix solution follows the micro group calculations. It provides flux spectra for energy condensation to the group structure used in KRAM.

One of the new features of CASMO-4 is that two-dimensional characteristic routine KRAM, which gives the eigenvalue and the flux distribution in an assembly, is employed to do the true heterogeneous geometry calculation. The previously generated cross sections constitute the input to the routine KRAM. The flux distribution throughout the lattice is determined by solving the Boltzmann transport equation using the method of characteristics.

In single bundle calculations a fundamental buckling mode is used for modifying the infinite lattice results obtained from the transport calculation to include leakage effects. It is normally made in diffusion theory and should be bypassed in calculations of two by two segments, reflectors and fuel storage racks.

The isotopic depletion as a function of irradiation is calculated for each fuel pin and for each region containing a burnable absorber. The burnup calculation is carried out with the predictor-corrector approach. For each burnup step from t_{n-1} to t_n, a "predictor" step is first taken using the fluxes obtaining from the neutron calculation at t_{n-1} to predict the number densities at t_n. The cross sections are then updated and the new spectrum calculation gives fluxes to be used in a "corrector" step after which final number densities at t_n are given by the average value of the results from the predictor and corrector steps.

The output of CASMO-4 is quite flexible; it prints the eigenvalue, the power distribution, reaction rates, and few group parameters for any region of the assembly for use in core calculations. The discontinuity factors for bundle interfaces and reflector regions included in the output can be used by advanced codes such as SIMULATE-3 in order to preserve the net currents calculated by the CASMO multi-group transport solution in two-group diffusion theory.

CASMO contains a module which calculates prompt and delayed gamma sources from neutron capture, fission and inelastic scattering and solves the gamma transport problem using the same two-dimensional model that is used for the neutron transport calculation, from which the gamma detector response is determined. The gamma transport calculation can be carried out either separately for prompt and for delayed gamma sources, or using the total gamma source. The so-called gamma smearing means that the contribution of gamma irradiation to the power distribution is assumed to be flat over the assembly.

1.3 SAMPLE PROBLEMS

In this part, some sample inputs will be illustrated. But before starting, the actual running procedure and the file organization of CASMO-4 will be discussed.

First the user has to compile the input file using the text editor. The input file, which consists of a sequence of "cards" identified by an alpha-numeric string of three characters, defines the entirety of the problem. Some tips for writing input files are as follows:

- The input of one card may be given on one or several lines and several cards are allowed on the same line. A card name or any other input parameter must not be divided on two lines.
- Input parameters are given in a way similar to standard format free FORTRAN. Parameters are separated by a separator: blank, comma, equal sign, semicolon, or the end of a line. A slash (`/`) is used as a separator between arrays of variable length. A separator may be surrounded by any number of blanks.
- The three character card identifier must be surrounded by separators. A slash is not allowed in the first position after the card identifier.
- A repetition factor can be used to specify the same parameter value to several consecutive positions. For example, `4*1` is identical to `1,1,1,1`.
- A position for a parameter value can be skipped in input by inputting two consecutive commas. The user can use default values instead of defining everything himself/herself, thus simplify the user oriented input. Default values are available for many input quantities and nuclear data are automatically read from the library.
- If a card of the same type is given more than once in the input, then the parameters of the last card overwrites earlier data.
- Comments can be written on any input card after an asterisk, which must be preceded by a separator.
- CASMO-4 uses metric units if nothing else is clearly marked in the description of the input card. The default units are shown in the Table 1-5:

Table 1-5. Default units in CASMO-4.

Quantities	Units
Material densities	gram/cm^3
Temperatures	K (Kelvin)
Dimensions	cm
Burnup (Exposure)	MWd/kg (= GWd/metric ton)

- CASMO-4 makes an automatic thermal expansion of dimensions and densities. By using the card `THE`, the expansion can be bypassed.

When the input file is ready, the user can run CASMO-4. Here the installed CASMO-4 in the machine FUELCYLE is used as our platform. The prompt is shown as

```
fuelcycle>
```

Suppose that the input file's name is `test.inp` and this file is in the current directory. The following command will be typed to launch the CASMO-4 run.

```
fuelcycle> cas4 test.inp
```

After finishing the CASMO-4 run, there will be four files in your directory:

`test.inp` — the original input file, which defines the desired problem;

`test.log` — the log file, which records the screen display when running;

`test.cax` — the card image file containing data to be used by other programs;

`test.out` — the output file, which contains the results of the calculation.

1.3.1 Pin Cell Example

Our first example is a unit cell calculation on 9.75% enrichment uranium, all uranium fuel, representative of Westinghouse PWR pin cell. The input file `pin.inp` is listed as following:

```
line 1    TTL * All-U, with 9.75% enrichment of U-235
line 2    TFU=900.0 TMO=583.1
line 3    FUE 1 10.3024/8.5934 92234=0.0687 92238=79.5443 8000=11.7935
line 4    CAN 6.55/304=100
line 5    PIC 0.4096 0.4178 0.4750 0.7109/'1' 'AIR' 'CAN' 'COO'
line 6    PDE 38.147742
line 7    DEP 0 0.1 0.5 1 5 10 20 30 40 50 60 70 80 90 100
line 8    STA
line 9    END
```

In this input file the whole problem is defined. Next we shall explain the input file line by line.

line 1 `TTL * All-U, with 9.75% enrichment of U-235`

The card `TTL` defines the beginning of our case, explaining briefly what our case is. The asterisk `*` must be preceded by a blank and the total length of the title should not exceed 70 characters.

line 2 `TFU=900.0 TMO=583.1`

The card `TFU` defines the fuel temperature and the card `TMO` defines the moderator temperature. In our case, the values are 900.0 K and 583.1 K respectively.

line 3 `FUE 1 10.3024/8.5934 92234=0.0687 92238=79.5443 8000=11.7935`

This is the fuel composition card. The fuel composition number is defined to be `1`. The range of the number is between 1 and 99 (including 1 and 99). Immediately after that, the fuel (UO_2) density is defined to be 10.3024 g/cm^3. In this example, the fuel density is of 94% theoretical density (10.96g/cm^3). Then a slash is used to indicate the following detailed fuel composition in weight percent.

Table 1-6. Fuel compositions.

Nuclides	Weight percentage
U-235	8.5934%
U-234	0.0687%
U-238	79.5443%
O	11.7935%

In this case, these numbers are based on the fact that the enrichment of U-235 in heavy metal is 9.75% and the fuel is in the form of uranium dioxide. Except for the weight percentage of U-235, all the other weight percentages are preceded by their nuclide identification, which is a four or five digit ID chosen so that the first digit(s) are equal to the atomic number of the nuclide and the last three digits show the isotope number. Three zeros (000) are used in the place of the isotope number for elements of natural composition. For instance, `92234` stands for U-234 while `8000` stands for natural oxygen.

The weight percentage of U-234 (W_{92234}) in the fuel is calculated by the user using the method given in CASMO-4 manual which can be expressed as

$$W_{92234} = 0.008\ W_{92235}.$$

Also, the number densities can be specified instead of weight percents for each composition. Any number greater than 10^3 will be interpreted as a number density. Number densities must then be entered for all nuclides of the composition whereas they can not be used for IDs that will be spit into individual isotopes by CASMO-4, e.g. `8000`. Then the density of the composition is calculated from the number density input. Thermal expansion will be bypassed if the number density is specified.

line 4 `CAN 6.55/304=100`

Canning composition is defined here. The density of the canning material is 6.55 g/cm^3 and the material is 100% Zircaloy-4 whose nuclide identification number is 304.

line 5 `PIC 0.4096 0.4178 0.4750 0.7109/'1' 'AIR' 'CAN' 'COO'`

This card defines our unit pin cell model. The radii of the fuel pin from innermost to outermost are 0.4096 cm, 0.4178 cm, 0.4750 cm. The composition for each successive region is fuel 1 defined in line 3, air gap, cladding material (Zircaloy-4) defined in line 4. Outside the fuel pin, the coolant (water) properties will be used.

The last value 0.7109cm is the equivalent radius (r) of the unit cell. In our problem, the pitch (p) is set to be 1.26 cm. Thus,

$$\pi \cdot r^2 = p^2,$$

$$r = \frac{p}{\sqrt{\pi}} = \frac{1.26\,\mathrm{cm}}{\sqrt{\pi}} = 0.7109\,\mathrm{cm}.$$

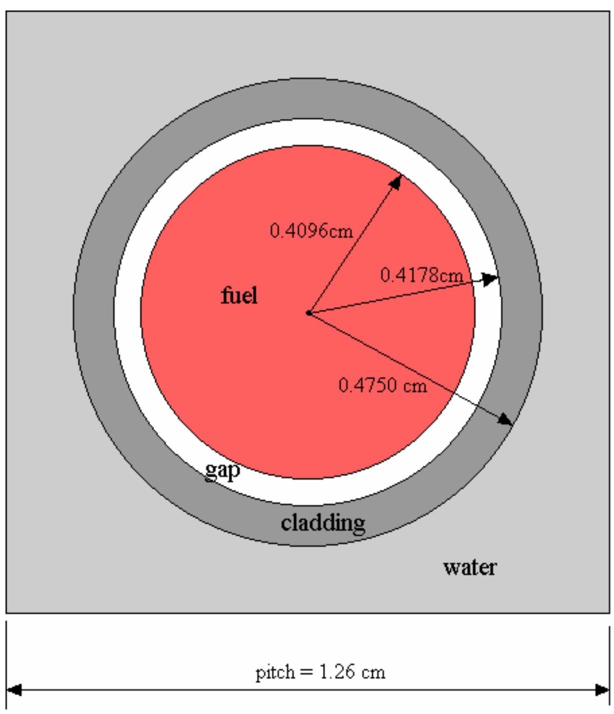

Figure 1-4. Unit cell model illustration.

line 6 `PDE 38.147742`

Here the power density is set to be 38.147742 MW/kg HM (HM refers to initial heavy metal), which is the usual power level of the Westinghouse assembly being modeled.

line 7 `DEP 0 0.1 0.5 1 5 10 20 30 40 50 60 70 80 90 100`

This card gives the depletion (burnup) points. The units are MWd/kg HM. It can be seen that the initial depletion step is very small, from 0 MWd/kg HM to 0.1 MWd/kg HM. This is to achieve the accurate calculation of the xenon equilibrium reactivity penalty. Thereafter, the time steps are larger and larger, since the reactivity versus burnup will be very linear.

line 8 `STA`

This card starts the computation. In our case here, the calculation will proceed until the burnup steps given on card `DEP` are completed.

line 9 `END`

By this card, the program execution is terminated. The card must be given on a new line as the last input card.

At this point the input file has been fully explained. Next the user can type the command to run this case and obtain the output file, `pin.out`, which contains the results of this run. The summary of the depletion calculation is given at the end of the file, i.e., k_∞ versus burnup, which is shown in Figure 1-5. It can be seen that initially there is a sudden drop of reactivity which is due to the xenon poison effect. After that the curve looks quite linear. From this curve, we can obtain the value of B_1 which is defined as

$$B_1 = \text{Burnup}(k_\infty = 1.03).$$

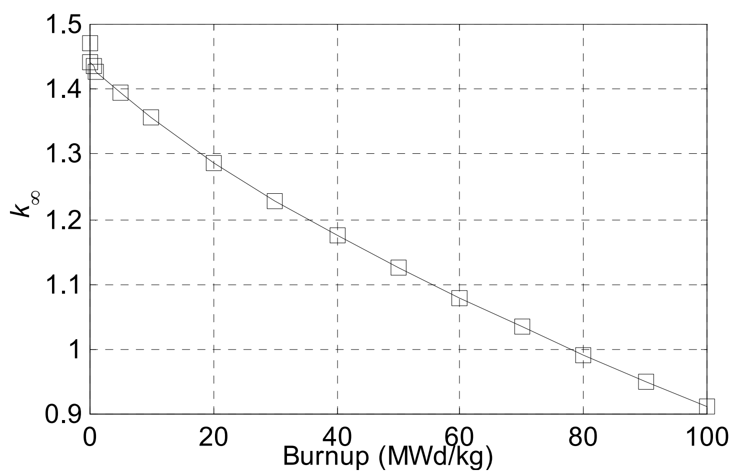

Figure 1-5. Burnup history curve.

Note here 1.03 is used instead of 1.0 to allow for the leakage effect typical of large PWR reactors. This is in fact, the maximum burnup which could be achieved if a single-batch whole-core reload scheme were employed. And it is also the equilibrium-core-average burnup at end of cycle for a multi-batch loading scheme. Multiple-staggered-batch discharge burnups are a constant multiple of B_1; for n-batch management ($1/n^{\text{th}}$ of the core refueled each cycle) one can show that[7]:

$$B_d = \left(\frac{2n}{n+1}\right) B_1 ,$$

where B_d is the discharge burnup. The estimates are first-order without consideration of the effects of soluble and burnable control poison.

Also from the output file, the isotopic amount of each element is obtained at each depletion step. In the following we extract the nuclide density for Pu-238, Pu-239, Pu-240 at each depletion step and plot out the plutonium composition versus the burnup history.

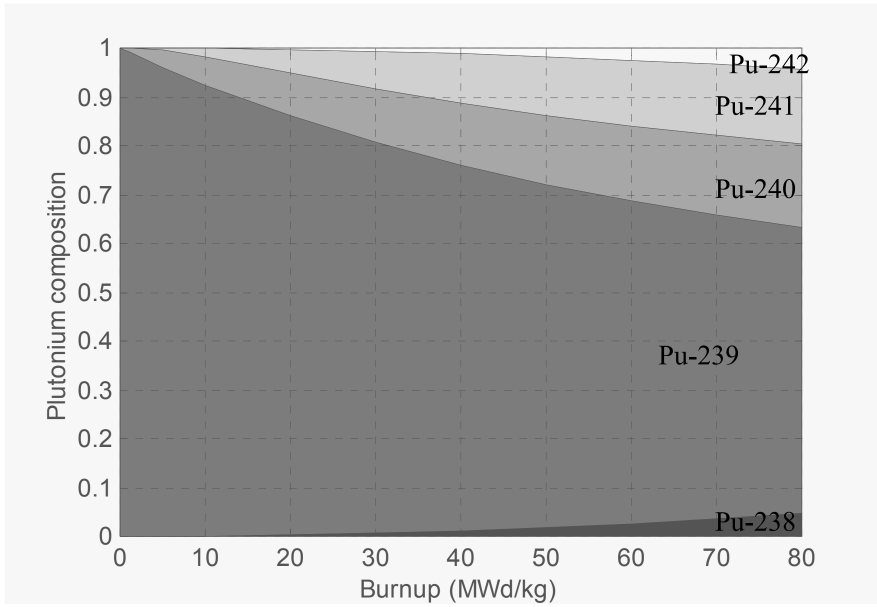

Figure 1-6. Plutonium composition versus burnup.

In the beginning, the concentration of Pu-239 is very high and the plutonium is super grade. As the burnup increases, the concentration of Pu-239 decreases and the other isotopes build up.

1.3.2 PWR Assembly Example

This example shows PWR assembly depletion calculations. In this case, the entire assembly with water rods and burnable poison is studied.

The input file `assembly.inp` is listed as following:

```
*       CASMO-4 Input
*       17x17 Westinghouse PWR Assembly
*       All-U Case
*       12 April 1999
*       Xianfeng Zhao
*       Massachusetts Institute of Technology
TTL *All-U Case
TFU=900.0,TMO=583.1,BOR=0.0,VOI=0.0,IDE='ALLU'
BOX,6.550,.7000E-05/304=100.0                           *ZR4
FUE,1,10.302/6.015,92234=0.048,92238=82.075,8000=11.861
FUE,2,10.302/6.015,92234=0.048,92238=76.075,8000=12.661,7300=6.0
PIN,1,.4096,.4178,.4750/"1","AIR","CAN"                  *FUEL
PIN,2,.5690,.6147/"COO","BOX"                            *GuideTubeT
PIN,3,.4096,.4178,.4750/"2","AIR","CAN"                  *FUEL
```

```
line 15 PRE,155.1296
line 16 PDE,38.13470
line 17 PWR,17,1.260,21.50,,,,,8
line 18 DEP,0.,5.,10.,20.,30.,40.,50.,60.,70.
line 19 LPI                                                          *44 BA
line 20    2
line 21    1  3
line 22    1  1  1
line 23    2  1  1  2
line 24    1  1  1  1  3
line 25    1  3  1  1  1  2
line 26    2  1  1  2  1  1  3
line 27    1  1  1  3  1  1  1  1
line 28    1  1  1  1  1  1  1  1  1
line 29 *LST,1,0,0,0
line 30 STA
line 31 END
```

Then let's explain this input file. Only some additional features will be explained in detail. Line 1-6 introduces this CASMO-4 run. Line 7 is the title of the run. And in this case, two kinds of fuel with different compositions are defined in line 10 and 11 respectively. Note that the fuel defined in line 11 contains burnable poison Gd. The nuclide identification `7300` refers to Gd_2O_3. Correspondingly, line 12 and 14 define the fuel pins. Line 13 defines the guide tube. Line 15 gives the pressure and line 16 defines the power density. Line 18 gives the depletion steps. And line 30, 31 are the same as in pin cell example.

The new cards in this example are in line 9, 17, 19 which will be described in detail in the following:

line 9 `BOX,6.550,.7000E-05/304=100.0                    *ZR4`

This card (`BOX`) defines the box wall (channel) composition. And this material is actually Zircolay-4.

line 17 `PWR,17,1.260,21.50,,,,,8`

This card (`PWR`) defines the PWR geometry. The first number 17 is the number of pins along one side of the assembly. In Westinghouse, the standard assembly is 17×17. The second number gives the pin pitch, i.e., 1.260 cm. And the third number is the assembly pitch (21.50 cm), which is the distance between the centers of two adjacent assemblies. The last number 8 indicates the symmetry in assembly. The octant of the assembly is defined as shown in Figure 1-7:

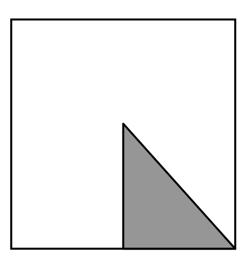

Figure 1-7. Octant of the whole assembly.

line 19 `LPI` `*44 BA`

The card defines the layout of pins. Lines 20-28 specify the one eighth of the whole assembly. The numbers indicate the pin type defined in lines 12-14.

After execution, the burnup curve can be obtained as in Figure 1-8. The burnable poison Gd lowers the initial reactivity and as burnup increase, it is depleted more and more until the burnup curve merges into the free-poison case. Therefore the burnable poison will make the control of the reactor easier due to less excessive reactivity.

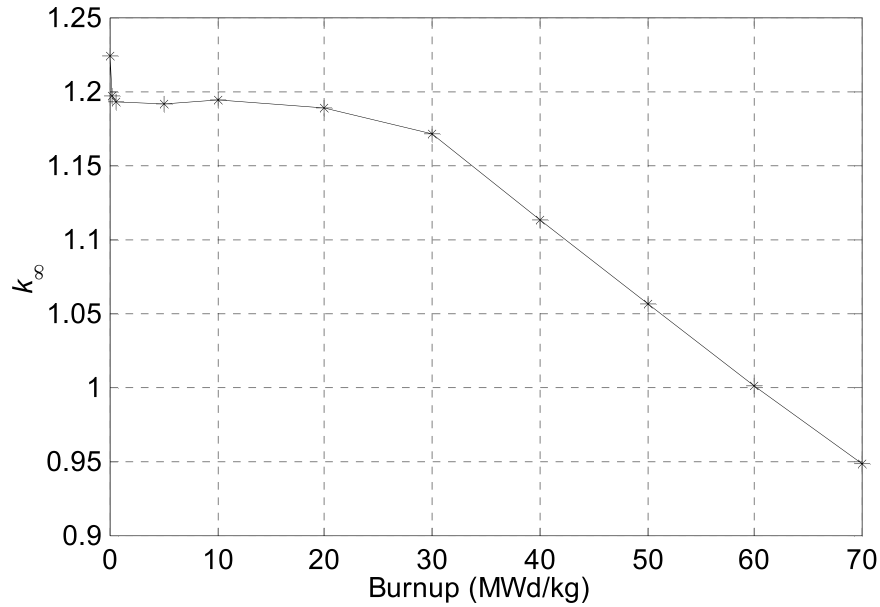

Figure 1-8. Burnup history curve for whole assembly case.

1.4 REFERENCES

[1] Malte Edenius, Kim Ekberg, Bengt H. Forssén, Dave Knott, “CASMO-4, A Fuel Assembly Burnup Program, User’s Manual,” STUDSVIK/SOA-95/1, Studsvik of America, Inc., 1995.

[2] Dave Knott, Bengt H. Forssén, Malte Edenius, “CASMO-4, A Fuel Assembly Burnup Program, Methodology,” STUDSVIK/SOA-95/2, Studsvik of America, Inc., 1995.

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