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THE LARGE CORE CODE EVALUATION WORKING GROUP
BENCHMARK PROBLEM FOUR:
NEUTRONICS AND BURNUP ANALYSIS OF A LARGE
HETEROGENEOUS FAST REACTOR

Part I: Analysis of Benchmark Results

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LARGE HETEROGENEOUS FAST REACTOR

Part 1: Analysis of Benchmark Results

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ABSTRACT

The Large Core Code Evaluation Working Group Benchmark Problem Four was specified to provide a stringent test of the current methods which are used in the nuclear design and analyses process. The benchmark specifications provided a base for performing detailed burnup calculations over the first two irradiation cycles for a large heterogeneous fast reactor. Particular emphasis was placed on the techniques for modeling the three-dimensional benchmark geometry, and sensitivity studies were carried out to determine the performance parameter sensitivities to changes in the neutronics and burnup specifications.

The results of the Benchmark Four calculations indicated that a linked RZ-XY (Hex) two-dimensional representation of the benchmark model geometry can be used to predict mass balance data, power distributions, regionwise fuel exposure data and burnup reactivities with good accuracy when compared with the results of direct three-dimensional computations. Most of the small differences in the results of the benchmark analyses by the different participants were attributed to ambiguities in carrying out the regionwise flux renormalization calculations throughout the burnup step. The Benchmark Four performance parameters were found to be moderately sensitive to the coarse mesh specifications. In particular, the results confirm that some caution must be exercised in computing the peak power densities for the inner and radial blankets when carrying out mesh-centered flux solution calculations. Parameter sensitivities to the lumped fission product representation, energy groups and burnup zone specifications were generally found to be small. A tabulation of the most important of the benchmark results, along with the analysis of these results is given in Part I of this report. Individual contributions, giving the results of all the benchmark calculations, are provided in Part II.

FOREWORD

The Large Core Code Evaluation Working Group Benchmark Problem Four Report is divided into two parts. Part I of the report includes the correlation, tabulation and analysis of the most important of the results of the benchmark calculations. The results of all of the calculations, as provided by the individual contributors, are included in Part II of the benchmark report. The principal contributors to Benchmark Problem Four are listed as follows:

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I. INTRODUCTION

Benchmark studies have been carried out by the members of the Large Core Code Evaluation Working Group (LCCEWG) as part of a broad effort to systematically evaluate the important steps in the reactor design and analysis process for large fast breeder reactors. The specific objectives of the LCCEWG benchmark studies have been:

1. To quantify the accuracy and efficiency of current neutronics methods for large cores;
2. To identify neutronic design problems unique to large breeder reactors;
3. To identify computer code development requirements;
4. To provide support for large core critical benchmark experiments.

A total of three benchmark problems have been analyzed by the LCCEWG since November 1975. These problems have focused on the calculations of key design parameters for both the large homogeneous and large heterogeneous fast reactor concepts with particular emphasis on the calculation of the sodium void worth. The early benchmark studies were limited to the analysis of the results of two-dimensional neutron transport calculations, although a simple fuel depletion computation was also included for the first time as part of LCCEWG Benchmark Problem Three. In assessing the results of the benchmark studies, it has been recognized that any conclusions must take into account the problem dependent nature of the benchmark specifications. Thus, great care must be exercised in extrapolating the results of benchmark calculations for a given configuration to another configuration with a different material composition; a different pin, assembly or reactor geometry; or a different driver fuel, blanket fuel

and control layout. In spite of the above limitation, the results of the benchmark studies have been useful in providing guidelines for assessing the sensitivities of key design parameters to design variables such as the energy group structure, mesh spacing and neutron transport approximations. The benchmark studies have also been particularly important in providing a comparison of the major design codes, and the early benchmark studies have indicated that the neutronics codes which used the same computational models and identical cross sections gave results which were in excellent agreement.

A fourth benchmark problem was specified to provide a stringent test of the current methods and procedures which are used in the design and analysis process. The results of this benchmark study have been documented to provide guidelines so that the designer can choose from a wide variety of design tools and methods for different applications, varying from scoping studies to a final design verification. The fourth benchmark problem was specified as a three-dimensional extension of LCCEWG Benchmark Problem Three, and detailed burnup calculations for the first two irradiation cycles were incorporated. Because of the complexity of the three-dimensional burnup problem, several "rigorous" benchmark solutions were sought to provide a base for testing the less accurate design approaches including the two-dimensional modeling of the three-dimensional geometry.

II. SPECIFICATIONS FOR LCCEWG BENCHMARK PROBLEM FOUR

2.1 Computational Model

The specifications for LCCEWG Benchmark Problem Four were based upon the results of an earlier preconceptual design study. These provide fully realistic computational challenges, but do not correspond specifically to those of any current conceptual or design work. The driver fuel, blanket fuel, shield and control assembly layout for the heterogeneous reactor system is shown in Figure 2.1. It is noted that this design layout provides a 30 degree azimuthal symmetry for modeling the core. The lattice pitch shown in Figure 2.1 is 16.33 cm and the driver fuel subassemblies contain 271 pins with an outer pin diameter of 7.87 mm. All internal and radial blanket subassemblies contain 127 pins with an outer pin diameter of 12.48 mm.

The axial geometry for Benchmark Problem Four is shown in an RZ format in Figure 2.2. A vacuum boundary is assumed above the fission gas plenum, below the lower axial shield and outside of the radial shield. The information given in Figures 2.1 and 2.2, along with the dimensions for the lattice pitch, can be used to set up either the two-dimensional or three-dimensional geometry for the benchmark analysis. It was noted that there are perhaps better approaches for modeling the control channels in the RZ geometry. However, the layout shown in Figure 2.2 was recommended to provide consistency in the results of the RZ calculations.

All primary and secondary control rods, with the exception of the control rods labelled C in Figure 2.1 were assumed to be parked at the upper axial blanket/core interface (i.e., control rods fully withdrawn from the core so that the lower edge is at the blanket/core interface). For purposes of the benchmark problem it was also assumed that the withdrawn rods do not extend into the upper fission gas plenum. To further simplify the computational requirements for the benchmark

FIGURE 2.1- SIXTY DEGREE SECTOR OF CORE LAYOUT FOR BENCHMARK PROBLEM FOUR (ASSEMBLIES GIVEN FOR FULL CORE LAYOUT)

FIGURE 2.2-CORE DIAGRAM IN RZ GEOMETRY FOR BENCHMARK
PROBLEM FOUR (NOT TO SCALE)

problem, the positions of the burnup control rods (i.e., control rods labelled C in Figure 2.1) were specified at the beginning and end of cycles 1 and 2 as follows:

	<u>Beginning</u> <u>of Cycle 1</u>	<u>End of</u> <u>Cycle 1</u>	<u>Beginning</u> <u>of Cycle 2</u>	<u>End of</u> <u>Cycle 2</u>
Position of burnup control rods above the lower surface of the lower axial shield (cm)	172.72	193.04	182.88	198.12

The control locations given above correspond to the rod insertion depths for which the respective effective multiplication constants have been precomputed to be approximately 1.0, based upon ENDF/B-V data. The corresponding burnup control rods in the RZ model were represented by the third control annulus (i.e., third control ring from the center of the core), as shown in Figure 2.2. The control insertion depths for the RZ model were assumed to be the same as those for the 3D configuration.

The material atom densities for the startup core are listed in Table 2.1. The isotopic compositions and the reactor geometry were given for the cold shutdown system. However, it was assumed that the change in the reactor dimensions and material compositions in going from the cold shutdown to the hot operating conditions was not significant.

2.2 Fuel Management Plan

The fuel management plan for Benchmark Problem Four was based upon a reactor power of 2740 MW_t, and a refueling interval of 1.5 years. Fuel burnup calculations were specified for the first two irradiation cycles of the startup core in accordance with the fuel

TABLE 2.1
INITIAL ATOM DENSITIES FOR BENCHMARK PROBLEM FOUR
(atoms/barn-cm)

	<u>Driver</u>	<u>Internal or Radial Blanket</u>	<u>Radial Shield Structure</u>	<u>Control</u> ⁽¹⁾	<u>Na Channel</u> ⁽²⁾	<u>Axial Blanket</u>	<u>Fission Gas Plenum</u>	<u>Lower Axial Shield</u>
Th-232	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
U-233	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
U-235	1.9168-5	2.9907-5	0.0	0.0	0.0	2.7953-5	0.0	0.0
U-238	8.6912-3	1.3558-2	0.0	0.0	0.0	1.2673-2	0.0	0.0
Pu-239	1.3591-3	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-240	4.0789-4	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-241	2.0392-4	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Pu-242	4.8481-5	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Cr	2.8111-3	2.3756-3	1.3180-2	3.1030-3	1.3900-3	2.8111-3	2.5335-3	1.2280-2
Fe	9.6814-3	8.1994-3	4.5410-2	1.0690-2	4.8170-3	9.6814-3	8.7337-3	4.2290-2
Ni	1.9207-3	1.6266-3	9.0060-3	2.1200-3	9.5520-4	1.9207-3	1.7326-3	8.3900-3
Mo	2.1764-4	1.8431-4	1.0210-3	2.4030-4	1.0820-4	2.1764-4	1.9633-4	9.5070-4
Mn	2.6609-4	2.2528-4	1.2480-3	2.9370-4	1.3240-4	2.6609-4	2.4001-4	1.1620-3
Na	8.2217-3	6.2401-3	2.2370-3	1.0490-3	1.8420-2	8.2217-3	7.8509-3	3.9930-3
O	2.1460-2	2.7176-2	0.0	0.0	0.0	2.1460-2	0.0	0.0
F.P.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
B-10	0.0	0.0	0.0	2.9110-2	0.0	0.0	0.0	0.0
B-11	0.0	0.0	0.0	2.5310-3	0.0	0.0	0.0	0.0
C	0.0	0.0	0.0	8.2700-3	0.0	0.0	0.0	0.0

1) Control assembly inserted

2) Control assembly withdrawn

FIGURE 2.3 REFUELING SCHEME FOR BENCHMARK
PROBLEM FOUR

management scheme shown in Figure 2.3. It is noted that the reference fuel management scheme differs from that for LCCEWG Benchmark Problem Three in that the cycle length was increased from 1.0 to 1.5 years and the average fraction of the fuel assemblies replaced after a given irradiation cycle was increased from 0.333 to 0.5 for the driver fuel, from 0.366 to 0.5 for the inner blanket fuel and from 0.167 to 0.25 for the radial blanket fuel. The change in the refueling interval was included to provide a more meaningful test of the methods for analyzing the impact of control effects, and for predicting the local power peaking and peak fuel burnup. The changes in the batch loading scheme were incorporated to maintain the 30 degree symmetry (rotational, periodic) in the computational model.

To avoid any confusion in the interpretation of the plant capacity factor (i.e., confusion in the calculation of isotopic decay such as that for Pu-241), the burnup calculations were performed on the basis of a refueling interval of 383.5 effective full power days (EFPD). Thus, the reactor was assumed to operate continuously at full power (i.e., with no down time for refueling, inspection, maintenance, etc.).

It was recommended for Benchmark Problem Four that the energy deposition include the in-situ contributions from fission and neutron capture. The recommended values of the in-situ energy conversion factors for the major fuel, structure and coolant isotopes are given in Table 2.2 in Section 2.3.

It was proposed that the fresh fuel to be loaded at the beginning of irradiation cycle 2 have the same fissile enrichment as that for the startup core. This fuel management plan provided a reasonable approach to the equilibrium cycle (for a slightly overreactive system with the control rods still inserted at the end of cycle 1), but it was not considered a critical step in the benchmark specifications. Any number of other fuel management approaches would have served equally well. The principal objective in performing burnup

calculations for two irradiation cycles was to assess the methods which were used in the analysis of a mix of fresh and exposed fuel.

2.3 Fuel Burnup Chain, Lumped Fission Products and Cross Sections

Cross section data were provided for the principal fuel isotopes and for many "non-fuel" actinides so that the fuel burnup calculations could be performed in accordance with the linked burnup chains shown in Figure 2.4. The recommended decay constants for the important actinides are also listed here. However, it is noted that most of the long lived alpha decay constants have been omitted from this burnup chain representation.

The extent of the burnup calculations were to be determined by the individual laboratories. However, an abbreviated version of the chain shown in Figure 4 was recommended for the reference burnup analysis. This chain consisted of the following isotopes:

U-235, U-236, U-238, Pu-239, Pu-240, Pu-241, and Pu-242.

The decay of Pu-241 was also included in all the fuel cycle calculations.

Several different representations for the lumped fission products were tested as part of the benchmark four analysis. The different representations included:

1. Two lumped fission products for isotopes with:
 - a. Odd atomic weight
 - b. Even atomic weight
2. One lumped fission product representation for all isotopes.

FIGURE 2.4- EXTENDED FUEL BURNUP CHAIN
FOR BENCHMARK PROBLEM 4

The above options provided a simple treatment of the fission product representations, with no distinction made between the fission products from different sources (i.e., from different uranium and plutonium isotopes). The failure to identify the source was not expected to have a significant impact on the Benchmark Four results.

Preliminary testing of the odd-even approach had indicated that it was satisfactory for a broad range of design applications. However, one of the objectives of the benchmark analysis was to determine the sensitivity of the important performance parameters to the fission product treatment and to provide guidelines for this treatment in the design process.

The cross sections for Benchmark Problem Four were provided by General Electric in the Computer Code Coordinating Committee (CCCC) ISOTXS format.⁽¹⁾ The cross section library had an 8-group energy structure and contained transport-corrected, P_0 microscopic data based upon a "heterogeneous" modeling of the cell in the data processing step (i.e., explicit modeling of the pin and assembly geometry). The cross sections for the major fuel isotopes (U-238, Pu-239) were provided for each of the driver fuel and blanket fuel regions indicated in Figures 2.1 and 2.2. However, many of the non-fuel and the less important fuel isotopes were consolidated for several regions in order to reduce the overall data manipulation requirements. In generating the cross section data, the average fuel temperatures for the driver, inner blanket, radial blanket and the axial blanket extension of the driver assemblies were assumed to be 1500°K, 1500°K, 900°K and 900°K respectively.

A listing of all the materials provided for Benchmark Problem Four is given in Table 2.2. For convenience the recommended energy conversion factors for those materials were also included.

TABLE 2.2

CROSS SECTION DATA PROVIDED FOR BENCHMARK PROBLEM FOUR

<u>Isotope</u>	<u>Mass</u>	<u>w.sec/fiss</u>	<u>w.sec/capt</u>
B10	10.013	0.0	1.8354-12
B11	11.009	0.0	1.5858-13
C	12.000	0.0	7.9247-13
O16	15.995	0.0	6.6365-13
NA23	22.990	0.0	1.8666-12
MG	24.305	0.0	1.3270-12
AL27	26.982	0.0	1.9010-12
SI	28.086	0.0	1.3810-12
CR	51.996	0.0	1.4801-12
MN55	54.938	0.0	1.1650-12
FE	55.847	0.0	1.2514-12
CO59	58.933	0.0	1.1999-12
NI	58.700	0.0	1.4881-12
MO	95.940	0.0	1.3631-12
U234	234.04	3.0488-11	8.4860-13
U235	235.04	3.1084-11	1.0470-12
U236	236.05	3.0888-11	8.7730-13
U237	237.05	2.8836-11	9.8363-13
U238	238.05	3.1741-11	9.2220-13
NP237	237.05	3.1257-11	1.0720-12
PU238	238.05	3.1690-11	9.0190-13
PU239	239.05	3.2050-11	1.0470-12
PU240	240.05	3.2009-11	8.3960-13
PU241	241.06	3.2398-11	1.0090-12
PU242	242.06	3.2140-11	8.8400-13
PU244	244.06	3.2500-11	8.8400-13
AM241	241.06	3.2410-11	8.8560-13
AM242M	242.06	3.2407-11	1.0295-12
AM242	242.06	3.2040-11	1.0216-12
AM243	243.06	3.2376-11	8.5915-13
CM242	242.06	3.2449-11	9.1362-13
CM243	243.06	3.2676-11	1.0892-12
CM244	244.06	3.3097-11	1.0335-12
FP		0.0	1.3631-12

2.4 Mesh Spacing and Region Description

Linked burnup and flux solution calculations are somewhat sensitive to the choice of the region and mesh descriptions for the computational model. Thus, it was recommended that the reactor models shown in Figures 2.1 and 2.2 be subdivided into regions and mesh intervals as follows:

<u>XY Plane</u>	<u>Region Description</u>	<u>Mesh Interval Description</u>
By Zone	Inner Blanket I	6 triangles/hex
	Driver I	6 triangles/hex
	Inner Blanket II	6 triangles/hex
	Driver II	6 triangles/hex
	Inner Blanket III	6 triangles/hex
	Driver III	6 triangles/hex
	Radial BLankets	6 triangles/hex
	Radial Shield	6 triangles/hex
<u>Z Dimension</u>		
Driver Fuel	8 axial regions	3 mesh intervals/ region
Lower Axial Blanket	2 axial regions	3 mesh intervals/ region
Upper Axial Blanket	2 axial regions	3 mesh intervals/ region
Lower Axial Shield	1 region	2 mesh intervals/ region
Upper Gas Plenum	1 region	4 mesh intervals/ region
<u>R Dimension (RZ Calculations Only)</u>		
<u>Region</u>	<u>Radius</u>	<u>Mesh Intervals</u>
Blanket	37.37	6
Driver	63.59	8

R Dimension (RZ Calculations Only) Cont'd

<u>Region</u>	<u>Radius</u>	<u>Mesh Intervals</u>
Control	66.96	2
Blanket	79.05	3
Control	81.79	2
Driver	109.46	8
Driver*	111.46	2
Control	113.42	2
Blanket	136.38	6
Control	137.98	2
Driver*	174.66	8
Control	175.92	2
Blanket	202.71	6
Shield	234.02	4

The principal reason for specifying the region width was to provide consistency in the 2D and 3D reactor burnup calculations. It is noted that the number of regions was doubled for irradiation cycle two because of the requirements for treating both the fresh and exposed fuel (i.e., distinction should be made between the fuel types in carrying out the burnup calculations for irradiation cycle two). It is also noted that an extra driver fuel region was inserted between 109.46 and 111.46 cm in the RZ problem. This driver region was included to ensure that the RZ representation of the radial power would be relatively flat, while maintaining the correct volume of the driver fuel assemblies. This special region was treated as part of the outer driver fuel region*.

*The extra driver has been provided to maintain a consistency between the 2D and 3D region descriptions. That is, the sum of the volumes for the two driver fuel regions marked by the asterisks is equivalent to the volume of the outer driver fuel region for the 3D problem. In reporting results, the two regions marked by the asterisks were combined.

III. BENCHMARK CALCULATIONS AND REPORTING OF RESULTS

3.1 Design Parameters of Interest For Benchmark Problem Four

The material which was reported for Benchmark Problem Four included the results of calculations which described the power distribution, burnup distribution, fuel inventory and neutron balances. A summary of all the quantities reported for the first three benchmarks and those which were required for the fourth benchmark is given in Table 3.1. The benchmark calculations were carried out for the first two irradiation cycles of the startup core, and the results were reported for four different cases as follows:

- Case 1. Beginning-of-irradiation Cycle 1.
- Case 2. End-of-Irradiation Cycle 1.
- Case 3. Beginning-of-Irradiation Cycle 2.
- Case 4. End-of-Irradiation Cycle 2.

3.2 Reference Solution

Reference computational procedures were specified to provide a base for the intercomparison of design codes, and for the determination of the sensitivities of the performance parameters to the methods. The reference procedures are briefly described as follows:

1. General - All computations were to be performed using:
 - a. The mesh spacing and region descriptions from Section 2.4.
 - b. The eight energy group cross section file provided by GE.
 - c. One lumped fission product.
2. Burnup - The reference burnup computations were to be

TABLE 3.1
SUMMARY OF LCCEWG RESULTS

	<u>Benchmark</u>			
	<u>1st</u>	<u>2nd</u>	<u>3rd</u>	<u>4th</u>
k_{eff}	X	X	X	X
Breeding Ratio (Gain)	X	0	0	X
Control Rod k(w & w/o Na)	Many	Central Rod	Many	X
Na Void k				
Rods in	X	X	X	0
Rods out	X	X	X	0
Region detail	0	X	X	0
Regional Fission Source Rate	0	X	X	0
Fission Rate (Power)				
Core Average	X	0	0	0
Peak to Average	X	0	0	X
Peak Positions	X	0	0	X
By Regions	0	X	X	X
Traverse	0	0	X	0
Fuel Burnup				
By Region	0	0	0	X
Peak Average	0	0	0	0
Core Leakage	0	w & w/o rod	Detailed	0
Isotopic Reaction Rates by Region	X	X	X	0
Neutron Balance	0	X	X	X
Inventories	0	0	X	X
Reactor α , η	0	0	X	0

Note: A non-zero entry signifies the calculation was or is to be done,
a zero indicates it was not done or is not requested.

carried out by region based upon:

- a. Continuous reactor operation at 2740 MW_t for 383.5 full power days.
 - b. An abbreviated burnup chain consisting of U-235, U-236, U-238, Pu-239, Pu-240, Pu-241 and Pu-242. The Pu-241 decay was also to be explicitly treated in the burnup calculations based upon the half-life given in Figure 2.4.
 - c. Two flux solutions for each irradiation cycle. Burnup calculations were to be carried out for the first half of the irradiation cycle based upon the flux solution at the beginning-of-cycle (BOC) with the control inserted to the BOC position as given in Section 2.1. Burnup calculations for the last half of the irradiation cycle were to be carried out based upon the flux solution at the middle-of-cycle (MOC) utilizing the isotopic compositions from the first burn step, but with the control inserted to the end-of-cycle position (again see Section 2.1). The same procedure was to be utilized for the second irradiation cycle.
3. Energy Deposition - The reference neutronics calculations were to be normalized based upon the energy deposition for in-situ capture and fission as listed for the different materials (isotopes) in Table 2.2. Should the codes not distinguish between capture and fission or if it was not possible to specify the energy deposition by isotope, the participants were encouraged to improvise so that the specified deposition rates were consistent with those calculated using the data in Table 2.2.
4. Convergence - The recommended convergence criteria for the neutronics calculations were 10^{-6} for K_{eff} and 10^{-5} for the

pointwise fluxes.

5. Linked RZ and XY Calculations - Those participants who performed the linked R-Z and X-Y calculations were requested to use regionwise and groupwise axial bucklings to establish the model links.

The results of the Benchmark Problem Three analysis indicated several areas which contribute to differences. Thus, it was recommended that the neutronics codes be modified, as necessary, to eliminate the (n,2n) induced discrepancy. In addition, it was recommended that those codes which required extra regions at the outer boundaries utilize a lower density filler. The recommended filler was radial shield material with one hundredth (0.01) of the indicated density.

3.3 Data Processing

An eight group cross section library was provided by GE for common use by all participants. The data base for this library was the LANL LIB-V library⁽²⁾, containing 94 materials in 70 groups and processed from ENDF/B-V data. Three additional materials, representing total, odd and even components of the fission product pairs for Pu-239 fission were added to the data base; these cross sections were obtained from HEDL.

The eight group cross section library generated by GE contained two files, written in ISOTXS⁽¹⁾ and ANISN⁽³⁾ formats, respectively. It contained eight sets of transport-corrected P_0 microscopic cross sections. These sets included (1) driver, (2) internal blanket, (3) radial blanket, (4) axial blanket above driver, (5) radial shield, (6) fission gas plenum and axial shield, (7) control in, and (8) control out cross sections. Tables A-1 through A-8 in the Appendix contain the cross section library designators for all materials contained on the library.

Computations were carried out with the data processing code TDOWN-V⁽⁴⁾ to generate selfshielded cross sections, to correct the elastic-removal cross sections for flux deviation from the LIB-V reference fluxes, and to collapse the cross sections to eight groups. Selfshielding of the cross sections included correction for the heterogeneous pin and subassembly configuration. Axial and radial bucklings in 20 groups were obtained from the GE analysis of LCCEWG Benchmark 3, expanded to 70 groups assuming constant bucklings for each fine group within each coarse group, and then were used in the TDOWN-V flux solutions.

The cell descriptions used for the heterogeneous selfshielding of the cross sections are shown in Figures 3.1 and 3.2. The fuel cross sections were selfshielded for the composition of the fuel pin at the center of the cell; the fuel in the Fuel-Clad-Na mix region produced a Dancoff correction to the selfshielding of the fuel pin. A similar Dancoff correction to the selfshielding of the clad and sodium cross sections was produced by the outer regions of the cell.

FIGURE 3.1 INTERNAL AND RADIAL BLANKET CELL FOR
LCCEWG BENCHMARK PROBLEM 4
CROSS SECTION SELF SHIELDING

FIGURE 3.2 DRIVER CELL FOR LCCEWG BENCHMARK
PROBLEM 4 CROSS SECTION SELF SHIELDING

IV. RESULTS OF BENCHMARK FOUR ANALYSES

4.1 Computer Codes, Methods and Nomenclature

The computer codes and methods which were utilized by the LCCEWG participants in the Benchmark Four analysis are listed in Table 4.1. The analysis was divided into three broad categories which included 1) direct three-dimensional triangular-Z computations, 2) linked two-dimensional RZ and XY (triangular) computations, and 3) two-dimensional RZ computations. The specifications for the neutronics and burnup calculations were described in considerable detail in Sections II and III in order to minimize the differences in the results (for each of the separate categories) because of differences in the analysis approximations. In spite of this attempt to standardize the analysis approach, some small differences in the results were noted due to differences in code modeling techniques (e.g., boundary conditions) and to differences in the neutronics and burnup procedures not stipulated in Sections II and III (e.g., power normalization during the burnup step). These differences are discussed in Section 4.2.

A listing of the terms which are utilized in describing the results of the Benchmark Four analysis is given in Table 4.2.

4.2 Reference Neutronics and Burnup Benchmark Results

The results of the neutronics and burnup calculations for the first two irradiation cycles of Benchmark Problem Four are given in Tables 4.3 - 4.9. The material in those tables includes the effective multiplication constants, regionwise power distributions, regionwise peak power densities, regionwise burnup distributions, regionwise fuel inventories and neutron balance data at the beginning and end of the two startup fuel cycles. The components of the burnup reactivity

TABLE 4.1
BENCHMARK PROBLEM IV
COMPUTER CODES AND METHODS

<u>Organization</u>	<u>Flux Solution</u>		<u>Irradiation</u>		<u>Reactor Model</u>	
	<u>Code</u>	<u>Theory</u>	<u>Code</u>	<u>Dimensions</u>	<u>Geometry</u>	
ANL	DIF3D ⁽⁵⁾	Diffusion	REBUS-3 ⁽⁶⁾	3-D	Triangular-Z	
CE	DIFPB ⁺	Diffusion	DIFPB	2-D	XY Triangular and RZ	
GE	DIF3D	Diffusion	FUMBLE ⁽⁷⁾	2-D	Linked RZ and XY Triangular; RZ	
HEDL	2DB ⁽⁸⁾	Diffusion	2DB	3-D	Triangular-Z	
MIT	2DB	Diffusion	2DB	2-D	RZ	
ORNL	VENTURE ⁽⁹⁾ /VALE ⁽¹⁰⁾	Diffusion	BURNER ⁽¹¹⁾	3-D/2-D	Triangular-Z/RZ	
W-ARD	W-2DB ⁺	Diffusion	W-2DB	2-D	Linked RZ and XY Triangular	

+) Proprietary Codes

TABLE 4.2
BENCHMARK PROBLEM IV
NOMENCLATURE

IRRADIATION CYCLE

BOC1	—	Beginning of Irradiation Cycle 1
MOC1	—	Middle of Irradiation Cycle 1
EOC1	—	End of Irradiation Cycle 1
BOC2	—	Beginning of Irradiation Cycle 2
MOC2	—	Middle of Irradiation Cycle 2
EOC2	—	End of Irradiation Cycle 2
Cycle 1	—	Irradiation Cycle 1 (Change)
Cycle 2	—	Irradiation Cycle 2 (Change)

REGION DESCRIPTION⁽¹⁾

D-I	—	Inner Driver Fuel Region
D-II	—	Middle Driver Fuel Region
D-III	—	Outer Driver Fuel Region
IB-I	—	Inner Blanket Region near Center of Core
IB-II	—	Inner Blanket Region near Middle of Core
IB-III	—	Inner Blanket Region near the Outer Core
RB	—	Radial Blanket Regions
AB	—	All Axial Blankets Extensions of the Driver Fuel
S	—	Radial Shield, Fission Gap Plenum, Lower Axial Shield & Orifice
C	—	All Control

1) See Figures 2.1 and 2.2

TABLE 4.3
BENCHMARK PROBLEM IV EFFECTIVE MULTIPLICATION
CONSTANTS FOR THE REFERENCE CONFIGURATION

EIGENVALUES ⁽¹⁾	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY (HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEDL	ORNL	CE	GE	WARD	CE	GE	MIT	ORNL
<u>k_{eff}</u>										
BOC1	1.00377	1.00376	1.00377	1.00165	1.00036	1.00329	1.00376	1.00249	1.00239	1.00251
EOC1	1.00153	0.98988	1.00125	0.99429	0.99786	1.00038	0.99801	1.00103	-	1.00133
BOC2	1.00439	0.99982	1.00409	0.99998	1.00167	1.00345	1.00371	1.00474	-	1.00485
EOC2	1.00320	0.99584	1.00276	0.99448	1.00103	1.00188	0.99972	1.00411	-	1.00435
<u>ΔK</u>										
CYCLE-1 (EOC1-BOC1)	-0.00224	-0.01388	-0.00252	-0.00736	-0.00250	-0.00291	-0.00575	-0.00146	-	-0.00118
CYCLE-2 (EOC2-BOC2)	-0.00119	-0.00398	-0.00133	-0.00550	-0.00064	-0.00157	-0.00399	-0.00063	-	-0.00050

1) See Table 4.2 for the Nomenclature

TABLE 4.4

BENCHMARK PROBLEM IV POWER DISTRIBUTIONS
FOR THE REFERENCE CONFIGURATION

POWER DISTRIBUTION ⁽¹⁾		THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
		ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
POWER FRACTION-BOCI											
(F-FISSION; C-CAPTURE)											
D-I	F	0.1174	0.1174	0.1175	0.1159	0.1152	0.1190	0.1110	0.1119	0.1117	0.1114
	C	0.0037	0.0037	0.0037	0.0036	0.0036	0.0036	0.0035	0.0035	0.0035	0.0035
D-II	F	0.2330	0.2330	0.2331	0.2223	0.2214	0.2345	0.2240	0.2252	0.2257	0.2246
	C	0.0072	0.0072	0.0072	0.0068	0.0067	0.0072	0.0068	0.0068	0.0069	0.0068
D-III	F	0.5313	0.5314	0.5312	0.5067	0.5446	0.5284	0.5474	0.5451	0.5424	0.5462
	C	0.0159	0.0159	0.0159	0.0166	0.0162	0.0154	0.0163	0.0162	0.0162	0.0163
IB-I	F	0.0038	0.0038	0.0038	0.0035	0.0037	0.0039	0.0035	0.0035	0.0027	0.0036
	C	0.0016	0.0016	0.0016	0.0014	0.0016	0.0015	0.0015	0.0015	0.0013	0.0015
IB-II	F	0.0104	0.0104	0.0104	0.0096	0.0101	0.0106	0.0099	0.0100	0.0104	0.0099
	C	0.0032	0.0032	0.0032	0.0028	0.0031	0.0031	0.0031	0.0031	0.0033	0.0031
IB-III	F	0.0227	0.0228	0.0228	0.0216	0.0226	0.0230	0.0225	0.0225	0.0230	0.0225
	C	0.0086	0.0087	0.0086	0.0076	0.0084	0.0082	0.0084	0.0084	0.0087	0.0084
RB	F	0.0172	0.0172	0.0172	0.0172	0.0175	0.0172	0.0170	0.0169	0.0181	0.0169
	C	0.0071	0.0071	0.0071	0.0067	0.0075	0.0068	0.0073	0.0072	0.0075	0.0073
AB	F	0.0080	0.0080	0.0080	-	0.0079	0.0078	0.0078	0.0078	0.0074	0.0078
	C	0.0036	0.0035	0.0035	-	0.0033	0.0033	0.0032	0.0033	0.0033	0.0033
S	F	-	-	-	-	-	-	-	-	-	-
	C	0.0017	0.0017	0.0018	0.0010	0.0017	0.0016	0.0016	0.0016	0.0025	0.0017
C	F	-	-	-	-	-	-	-	-	-	-
	C	0.0034	0.0034	0.0034	0.0027	0.0051	0.0049	0.0052	0.0052	0.0054	0.0052
		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.4

BENCHMARK PROBLEM IV POWER DISTRIBUTIONS
FOR THE REFERENCE CONFIGURATION (CONTINUED)

POWER DISTRIBUTION ⁽¹⁾		THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS		
		ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	ORNL
POWER FRACTION-EOC1										
(F-FISSIONS; C-CAPTURES)										
D-I	F	0.1321	0.1244	0.1332	0.1383	0.1358	0.1328	0.1322	0.1333	0.1332
	C	0.0049	0.0043	0.0049	0.0047	0.0051	0.0045	0.0046	0.0050	0.0049
D-II	F	0.2540	0.2473	0.2557	0.2567	0.2530	0.2562	0.2590	0.2605	0.2604
	C	0.0094	0.0085	0.0095	0.0087	0.0092	0.0085	0.0088	0.0095	0.0095
D-III	F	0.4113	0.4452	0.4091	0.4330	0.4134	0.4165	0.4134	0.4049	0.4052
	C	0.0145	0.0148	0.0145	0.0143	0.0147	0.0135	0.0138	0.0144	0.0144
IB-I	F	0.0119	0.0091	0.0117	0.0111	0.0115	0.0113	0.0109	0.0114	0.0114
	C	0.0021	0.0019	0.0021	0.0019	0.0021	0.0019	0.0020	0.0021	0.0021
IB-II	F	0.0328	0.0262	0.0325	0.0302	0.0317	0.0314	0.0318	0.0333	0.0332
	C	0.0041	0.0038	0.0041	0.0036	0.0040	0.0037	0.0040	0.0042	0.0041
IB-III	F	0.0610	0.0525	0.0608	0.0558	0.0577	0.0586	0.0582	0.0608	0.0608
	C	0.0093	0.0093	0.0093	0.0081	0.0086	0.0085	0.0091	0.0091	0.0091
RB	F	0.0251	0.0249	0.0253	0.0254	0.0252	0.0252	0.0244	0.0241	0.0241
	C	0.0058	0.0064	0.0058	0.0054	0.0059	0.0055	0.0058	0.0056	0.0057
AB	F	0.0135	0.0131	0.0134	-	0.0128	0.0131	0.0127	0.0128	0.0128
	C	0.0036	0.0037	0.0036	-	0.0034	0.0033	0.0034	0.0034	0.0034
S	F	-	-	-	-	-	-	-	-	-
	C	0.0016	0.0017	0.0017	0.0009	0.0015	0.0014	0.0015	0.0015	0.0015
C	F	-	-	-	-	-	-	-	-	-
	C	0.0028	0.0028	0.0028	0.0019	0.0044	0.0041	0.0044	0.0042	0.0042
		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.4
BENCHMARK PROBLEM IV POWER DISTRIBUTIONS
FOR THE REFERENCE CONFIGURATION (CONTINUED)

		THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS		
POWER DISTRIBUTION ⁽¹⁾		ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	ORNL
POWER FRACTION-BOC2										
(F-FISSION; C-CAPTURE)										
D-I	F	0.1291	0.1311	0.1293	0.1306	0.1347	0.1289	0.1311	0.1327	0.1322
	C	0.0043	0.0042	0.0042	0.0040	0.0045	0.0041	0.0042	0.0044	0.0044
D-II	F	0.2387	0.2455	0.2397	0.2343	0.2406	0.2406	0.2462	0.2482	0.2478
	C	0.0082	0.0080	0.0081	0.0076	0.0080	0.0070	0.0079	0.0083	0.0083
D-III	F	0.4691	0.4698	0.4682	0.4952	0.4626	0.4720	0.4626	0.4559	0.4567
	C	0.0152	0.0148	0.0152	0.0154	0.0150	0.0145	0.0146	0.0148	0.0148
IB-I	F	0.0087	0.0074	0.0084	0.0078	0.0086	0.0083	0.0081	0.0084	0.0085
	C	0.0019	0.0019	0.0019	0.0017	0.0019	0.0018	0.0019	0.0019	0.0019
IB-II	F	0.0203	0.0186	0.0200	0.0184	0.0207	0.0197	0.0206	0.0214	0.0213
	C	0.0036	0.0036	0.0036	0.0032	0.0036	0.0034	0.0037	0.0038	0.0037
IB-III	F	0.0424	0.0378	0.0423	0.0396	0.0405	0.0414	0.0406	0.0420	0.0420
	C	0.0089	0.0088	0.0089	0.0078	0.0084	0.0083	0.0086	0.0087	0.0087
RB	F	0.0244	0.0231	0.0248	0.0249	0.0247	0.0246	0.0239	0.0238	0.0239
	C	0.0065	0.0065	0.0066	0.0062	0.0066	0.0062	0.0064	0.0063	0.0062
AB	F	0.0104	0.0103	0.0104	-	0.0100	0.0098	0.0100	0.0100	0.0100
	C	0.0036	0.0036	0.0036	-	0.0033	0.0035	0.0033	0.0033	0.0033
S	F	-	-	-	-	-	-	-	-	-
	C	0.0017	0.0017	0.0017	0.0010	0.0016	0.0016	0.0015	0.0016	0.0016
C	F	-	-	-	-	-	-	-	-	-
	C	<u>0.0031</u>	<u>0.0031</u>	<u>0.0031</u>	<u>0.0037</u>	<u>0.0047</u>	<u>0.0043</u>	<u>0.0048</u>	<u>0.0047</u>	<u>0.0047</u>
		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.4

BENCHMARK PROBLEM IV POWER DISTRIBUTIONS
FOR THE REFERENCE CONFIGURATION (CONTINUED)

POWER DISTRIBUTION ⁽¹⁾		THREE-DIMENSIONAL HEX-Z CALCULATION			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS		
		ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	ORNL
POWER FRACTION-EOC2										
(F-FISSION; C-CAPTURE)										
D-I	F	0.1328	0.1302	0.1337	0.1402	0.1382	0.1344	0.1337	0.1344	0.1343
	C	0.0053	0.0043	0.0052	0.0050	0.0055	0.0047	0.0048	0.0053	0.0053
D-II	F	0.2437	0.2505	0.2454	0.2501	0.2474	0.2472	0.2508	0.2512	0.2512
	C	0.0098	0.0084	0.0098	0.0089	0.0098	0.0087	0.0090	0.0100	0.0099
D-III	F	0.3831	0.4585	0.3812	0.4039	0.3791	0.3877	0.3833	0.3749	0.3752
	C	0.0143	0.0148	0.0144	0.0138	0.0142	0.0131	0.0133	0.0140	0.0140
IB-I	F	0.0181	0.0080	0.0175	0.0166	0.0178	0.0169	0.0165	0.0174	0.0175
	C	0.0023	0.0019	0.0023	0.0021	0.0023	0.0021	0.0022	0.0023	0.0023
IB-II	F	0.0411	0.0205	0.0407	0.0385	0.0413	0.0397	0.0412	0.0428	0.0427
	C	0.0042	0.0037	0.0043	0.0037	0.0043	0.0038	0.0042	0.0044	0.0043
IB-III	F	0.0766	0.0414	0.0765	0.0710	0.0722	0.0741	0.0734	0.0762	0.0762
	C	0.0095	0.0091	0.0095	0.0083	0.0088	0.0086	0.0092	0.0093	0.0094
RB	F	0.0296	0.0235	0.0302	0.0302	0.0295	0.0297	0.0290	0.0285	0.0285
	C	0.0056	0.0064	0.0056	0.0053	0.0056	0.0053	0.0056	0.0054	0.0054
AB	F	0.0157	0.0108	0.0156	-	0.0149	0.0151	0.0147	0.0149	0.0149
	C	0.0037	0.0037	0.0037	-	0.0035	0.0033	0.0035	0.0035	0.0034
S	F	-	-	-	-	-	-	-	-	-
	C	0.0016	0.0017	0.0017	0.0009	0.0015	0.0016	0.0015	0.0015	0.0016
C	F	-	-	-	-	-	-	-	-	-
	C	0.0026	0.0026	0.0027	0.0016	0.0041	0.0040	0.0041	0.0039	0.0039
		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.5

BENCHMARK PROBLEM IV PEAK POWER DENSITIES
FOR THE REFERENCE CONFIGURATION

POWER PEAKING ⁽¹⁾	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY (HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>PEAK POWER-DRIVER FUEL (KW/CM³)</u>										
BOC1	5.28-1	5.28-1	5.28-1	3.96-1	5.26-1	5.33-1	5.11-1	5.10-1	5.25-1	5.09-1
EOC1	5.36-1	5.34-1	5.40-1	4.05-1	5.46-1	5.45-1	5.04-1	5.14-1	-	5.13-1
BOC2	5.23-1	5.38-1	5.25-1	3.80-1	5.34-1	5.22-1	4.99-1	5.08-1	-	5.06-1
EOC2	5.10-1	5.45-1	5.14-1	4.03-1	5.23-1	5.15-1	4.91-1	5.03-1	-	5.02-1
<u>PEAK POWER-INNER BLANKETS (KW/CM³)</u>										
BOC1	8.05-2	8.04-2	8.05-2	5.91-2	8.13-2	8.18-2	6.97-2	7.02-2	5.90-2	7.04-2
EOC1	2.23-1	1.59-1	2.22-1	1.44-1	2.15-1	2.15-1	1.84-1	2.14-1	-	2.14-1
BOC2	1.86-1	1.22-1	1.83-1	1.18-1	1.45-1	1.73-1	1.26-1	1.41-1	-	1.41-1
EOC2	2.93-1	1.32-1	2.90-1	1.93-1	2.69-1	2.82-1	2.28-1	2.66-1	-	2.66-1
<u>PEAK POWER-RADIAL BLANKETS (KW/CM³)</u>										
BOC1	6.27-2	6.26-2	6.27-2	4.96-2	6.48-2	6.22-2	4.74-2	4.73-2	4.03-2	4.67-2
EOC1	7.29-2	6.84-2	7.31-2	5.48-2	7.35-2	7.33-2	5.03-2	5.34-2	-	5.28-2
BOC2	7.62-2	6.73-2	8.10-2	5.61-2	7.58-2	7.62-2	5.26-2	5.53-2	-	5.48-2
EOC2	8.06-2	6.72-2	8.67-2	5.81-2	7.97-2	8.11-2	5.48-2	5.91-2	-	5.86-2
<u>DRIVER FUEL PEAK-TO-AVERAGE RATIO POWER</u>										
BOC1	1.61	1.61	1.61	1.21	1.61	1.63	1.56	1.56	1.60	1.56
EOC2	1.80	1.75	1.81	1.35	1.82	1.82	1.68	1.72	-	1.72
BOC2	1.68	1.71	1.68	1.22	1.71	1.67	1.60	1.63	-	1.63
EOC2	1.79	1.74	1.80	1.40	1.83	1.80	1.71	1.77	-	1.76
<u>DRIVER FUEL AXIAL PEAK-TO-AVERAGE POWER (AT POSITION OF PEAK POWER)</u>										
BOC1	1.35	1.35	-	-	1.35	1.40	1.33	1.35	1.34	-
EOC1	1.31	1.34	-	-	1.31	1.32	1.30	1.31	-	-
BOC2	1.35	1.36	-	-	1.34	1.34	1.31	1.34	-	-
EOC2	1.29	1.34	-	-	1.30	1.29	1.27	1.30	-	-

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.6

BENCHMARK PROBLEM IV FUEL EXPOSURES
FOR THE REFERENCE CONFIGURATION

BURNUP (MWD/MT)	THREE-DIMENSIONAL HEX-Z CALCULATIONS		TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS	
	ANL ⁽³⁾	HEDL ⁽⁴⁾	CE ⁽²⁾	GE ⁽⁴⁾	WARD ⁽⁴⁾	CE ⁽⁴⁾	GE ⁽⁴⁾
<u>AVERAGE DISCHARGE BURNUP - EOC1⁽¹⁾</u>							
D-I	3.154+4	3.106+4	3.127+4	2.958+4	3.146+4	3.067+4	2.910+4
D-II	3.063+4	3.088+4	2.946+4	2.836+4	3.030+4	3.044+4	2.927+4
D-III	2.554+4	2.819+4	2.715+4	2.679+4	2.651+4	2.688+4	2.634+4
IB-I	1.822+3	1.583+3	3.182+3	1.552+3	2.214+3	2.062+3	1.521+3
IB-II	4.033+3	3.612+3	6.505+3	3.518+3	4.720+3	4.465+3	3.675+3
IB-III	2.417+3	2.204+3	4.049+3	2.199+3	2.826+3	2.763+3	2.288+3
RB	6.996+2	8.191+2	1.346+3	7.298+2	1.007+3	8.607+2	6.968+2
AB	5.185+2	6.067+2	-	4.773+2	6.460+2	6.397+2	4.773+2
<u>AVERAGE DISCHARGE BURNUP - EOC2⁽¹⁾</u>							
D-I	6.476+4	-	-	6.233+4	6.464+4	-	6.126+4
D-II	6.128+4	-	-	5.789+4	6.189+4	-	5.962+4
D-III	4.904+4	-	-	5.020+4	4.959+4	-	4.923+4
IB-I	4.945+3	-	-	4.376+3	5.886+3	-	4.315+3
IB-II	9.809+3	-	-	9.001+3	1.146+4	-	9.418+3
IB-III	5.837+3	-	-	5.305+3	7.151+3	-	5.561+3
RB	1.579+3	-	-	1.627+3	1.905+3	-	1.556+3
AB	1.149+3	-	-	1.055+3	1.376+3	-	1.055+3
<u>PEAK BURNUP - EOC1⁽¹⁾</u>							
D-I	4.272+4	4.331+4	3.271+4	4.094+4	4.380+4	4.164+4	3.974+4
D-II	4.610+4	4.881+4	3.418+4	4.467+4	4.771+4	4.365+4	4.219+4
D-III	4.308+4	4.632+4	3.340+4	4.258+4	4.234+4	4.124+4	3.905+4
<u>PEAK BURNUP - EOC2⁽¹⁾</u>							
D-I	8.827+4	-	-	8.561+4	8.718+4	8.636+4	8.319+4
D-II	9.121+4	-	-	9.038+4	9.332+4	8.765+4	8.514+4
D-III	8.264+4	-	-	8.015+4	8.099+4	7.792+4	7.271+4

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

3) The Peak Burnup is Defined as the Average Burnup Times the EOC Peak-to-Average Power

4) The Peak Burnup is Defined as the Average of the Peak Burnups at BOC and EOC

TABLE 4.7

BENCHMARK PROBLEM IV FUEL INVENTORIES
FOR THE REFERENCE CONFIGURATION

FUEL INVENTORY (KG)	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	AWL	HEOL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>DRIVER FUEL - BOC⁽¹⁾</u>										
U-235	5.687+1	5.686+1	5.687+1	5.687+1	5.684+1	5.688+1	5.688+1	5.684+1	5.685+1	5.687+1
U-236	-	-	-	-	-	-	-	-	-	-
U-238	2.612+4	2.611+4	2.612+4	2.612+4	2.610+4	2.612+4	2.612+4	2.610+4	2.611+4	2.612+4
Pu-239	4.101+3	4.101+3	4.101+3	4.101+3	4.099+3	4.100+3	4.102+3	4.099+3	4.100+3	4.101+3
Pu-240	1.236+3	1.236+3	1.236+3	1.236+3	1.235+3	1.236+3	1.236+3	1.235+3	1.236+3	1.236+3
Pu-241	6.206+2	6.204+2	6.205+2	6.205+2	6.201+2	6.203+2	6.206+2	6.201+2	6.203+2	6.205+2
Pu-242	1.481+2	1.481+2	1.481+2	1.481+2	1.480+2	1.481+2	1.482+2	1.480+2	1.481+2	1.481+2
FP	-	-	-	-	-	-	-	-	-	-
<u>INNER BLANKET - BOC⁽¹⁾</u>										
U-235	6.297+1	6.294+1	6.296+1	3.977+1	6.294+1	6.296+1	6.298+1	6.294+1	6.296+1	6.298+1
U-236	-	-	-	-	-	-	-	-	-	-
U-238	2.891+4	2.891+4	2.891+4	1.826+4	2.890+4	2.891+4	2.892+4	2.891+4	2.891+4	2.892+4
Pu-239	-	-	-	-	-	-	-	-	-	-
Pu-240	-	-	-	-	-	-	-	-	-	-
Pu-241	-	-	-	-	-	-	-	-	-	-
Pu-242	-	-	-	-	-	-	-	-	-	-
FP	-	-	-	-	-	-	-	-	-	-

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.7

BENCHMARK PROBLEM IV FUEL INVENTORIES
FOR THE REFERENCE CONFIGURATION (CONTINUED)

FUEL INVENTORY (KG)	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY (HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>RADIAL BLANKETS - BOC1⁽¹⁾</u>										
U-235	7.181+1	7.182+1	7.181+1	4.535+1	7.176+1	7.180+1	7.180+1	7.176+1	7.178+1	7.180+1
U-236	-	-	-	-	-	-	-	-	-	-
U-238	3.297+4	3.296+4	3.297+4	2.082+4	3.295+4	3.297+4	3.297+4	3.295+4	3.296+4	3.297+4
Pu-239	-	-	-	-	-	-	-	-	-	-
Pu-240	-	-	-	-	-	-	-	-	-	-
Pu-241	-	-	-	-	-	-	-	-	-	-
Pu-242	-	-	-	-	-	-	-	-	-	-
<u>AXIAL BLANKETS - BOC1⁽¹⁾</u>										
U-235	4.838+1	4.837+1	4.838+1	-	4.835+1	4.837+1	4.838+1	4.835+1	4.837+1	4.838+1
U-236	-	-	-	-	-	-	-	-	-	-
U-238	2.222+4	2.221+4	2.221+4	-	2.220+4	2.221+4	2.222+4	2.220+4	2.221+4	2.221+4
Pu-239	-	-	-	-	-	-	-	-	-	-
Pu-240	-	-	-	-	-	-	-	-	-	-
Pu-241	-	-	-	-	-	-	-	-	-	-
Pu-242	-	-	-	-	-	-	-	-	-	-
FP	-	-	-	-	-	-	-	-	-	-

1) See Table 4.2 for the Nomenclature

2) Results for the XY Plane Only

TABLE 4 7

BENCHMARK PROBLEM IV FUEL INVENTORIES
FOR THE REFERENCE CONFIGURATION (CONTINUED)

FUEL INVENTORY (KG)	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINED RZ-XY (HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
DRIVER FUEL - EOC1 ⁽¹⁾										
U-235	4.602+1	4.598+1	4.598+1	4.596+1	4.589+1	4.594+1	4.601+1	4.601+1		4.604+1
U-236	2.378+0	2.425+0	2.382+0	2.381+0	2.395+0	2.387+0	2.369+0	2.361+0		2.361+0
U-238	2.537+4	2.535+4	2.536+4	2.537+4	2.535+4	2.537+4	2.537+4	2.535+4		2.537+4
Pu-239	3.920+3	3.914+3	3.917+3	3.915+3	3.914+3	3.914+3	3.917+3	3.914+3		3.917+3
Pu-240	1.299+3	1.300+3	1.299+3	1.297+3	1.297+3	1.298+3	1.297+3	1.296+3		1.297+3
Pu-241	5.075+2	4.950+2	5.071+2	5.063+2	5.064+2	5.066+2	5.070+2	5.074+2		5.072+2
Pu-242	1.602+2	1.601+2	1.601+2	1.600+2	1.601+2	1.601+2	1.600+2	1.599+2		1.599+2
FP	-	9.516+2	9.411+2	-	9.471+2	9.181+2	-	9.773+2		9.419+2
INTERNAL BLANKET - EOC1 ¹⁾										
U-235	5.413+1	5.401+1	5.429+1	5.230+1	5.441+1	5.438+1	5.439+1	5.442+1		5.446+1
U-236	2.183+0	2.242+0	2.154+0	1.834+0	2.115+0	2.129+0	2.133+0	2.113+0		2.110+0
U-238	2.838+4	2.838+4	2.839+4	1.781+4	2.839+4	2.840+4	2.840+4	2.839+4		2.840+4
Pu-239	4.441+2	4.536+2	4.394+2	3.808+2	4.331+2	4.348+2	4.368+2	4.320+2		4.317+2
Pu-240	1.244+1	9.486+0	1.217+1	1.045+1	1.121+1	1.135+1	1.139+1	1.223+1		1.222+1
Pu-241	2.536-1	1.316-1	2.471-1	1.908-1	2.031-1	2.076-1	2.100-1	2.559-1		2.552-1
Pu-242	3.498-3	1.130-3	3.363-3	2.200-3	2.354-3	2.432-3	2.492-3	3.593-3		3.578-3
FP	-	6.780-1	7.613+1	-	7.327+1	7.199+1	-	7.592+1		7.563+1

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4 7
BENCHMARK PROBLEM IV FUEL INVENTORIES
FOR THE REFERENCE CONFIGURATION (CONTINUED)

FUEL INVENTORY (KG)	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINED RZ-XY (HFX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEDT	ORNL	CE ⁽¹⁾	GE	WAPD	CE	GE	MIT	ORNL
<u>RADIAL BLANKETS - EOC⁽¹⁾</u>										
U-235	6 704+1	6 666+1	6 683+1	4 081+1	6 670+1	6 689+1	6 685+1	6 683+1		6 687+
U-236	1.265+0	1 379+0	1 322+0	1 196+0	1 342+0	1 304+0	1 312+0	1 313+0		1 311+0
U-238	3 274+4	3 272+4	3 272+4	2 060+4	3 271+4	3 273+4	3 272+4	3 271+4		3 272+4
Pu-239	2.111+2	2 276+2	2 194+2	2 025+2	2 237+2	2 166+2	2 246+2	2 177+2		2 172+2
Pu-240	3.102+0	2 743+0	3 431+0	3 358+0	3 402+0	3 185+0	3 236+0	3 433+0		3 434+0
Pu-241	3.297-2	2 158-2	3 912-2	3 660-2	3 587-2	3 255-2	3 273-2	3 924-2		3 921-2
Pu-242	1.915-4	8 622-5	2 441-4	2 066-4	1 981-4	1 740-4	1 762-4	2 431-4		2 431-4
FP	-	2 414+1	2 539+1	-	2 588+1	2 437+1	-	2 472+1		2 460+1
<u>AXIAL BLANKETS EOC⁽¹⁾</u>										
U-235	4 570+1	4 567+1	4 574+1	-	4 590+1	4 591+1	4 606+1	4 590+1		4 590+1
U-236	7.183-1	7 278-1	7 082-1	-	6 538-1	6 563-1	6 068-1	6 538-1		6 479-1
U-238	2 209+4	2 208+4	2 209+4	-	2 208+4	2 209+4	2 210+4	2 208+4		2 209+4
Pu-239	1.182+2	1 189+2	1 167+2	-	1 057+2	1 099+2	1 087+2	1 097+2		1 095+2
Pu-240	1.349+0	1 193+0	1 320+0	-	1 183+0	1 194+0	1 005+0	1 183+0		1 179+0
Pu-241	1.148-2	8 358-3	1 112-2	-	1 000-2	1 020-2	7 158-3	1 001-2		9 922-3
Pu-242	5.373-5	3 012-5	5 087-5	-	4 691-5	4 840-5	2 986-5	4 691-5		4 691-5
FP	-	1 160+1	1 183+1	-	1 143+1	1 134+1	-	1 143+1		1 135+1

1) See Table 4 2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.7
BENCHMARK PROBLEM IV FUEL INVENTORIES
FOR THE REFERENCE CONFIGURATION (CONTINUED)

FUEL INVENTORY (KG)	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEOL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>DRIVER FUEL - BOC2⁽¹⁾</u>										
U-235	5.180+1	5.171+1	5.178+1	5.177+1	5.179+1	5.176+1	5.181+1	5.186+1		5.183+1
U-236	1.111+0	1.129+0	1.112+0	1.113+0	1.114+0	1.115+0	1.102+0	1.098+0		1.099+0
U-238	2.577+4	2.576+4	2.576+4	2.577+4	2.575+4	2.577+4	2.577+4	2.576+4		2.577+4
Pu-239	4.016+3	4.014+3	4.015+3	4.014+3	4.011+3	4.012+3	4.015+3	4.013+3		4.015+3
Pu-240	1.265+3	1.265+3	1.265+3	1.264+3	1.264+3	1.265+3	1.264+3	1.263+3		1.264+3
Pu-241	5.678+2	5.620+2	5.675+2	5.671+2	5.673+2	5.673+2	5.676+2	5.677+2		5.677+2
Pu-242	1.537+2	1.537+2	1.537+2	1.537+2	1.536+2	1.537+2	1.537+2	1.536+2		1.536+2
FP	-	4.432+2	4.402+2	-	4.408+2	4.292+2	-	4.552+2		4.388+2
<u>INNER BLANKET - BOC2⁽¹⁾</u>										
U-235	5.822+1	5.812+1	5.831+1	5.572+1	5.831+1	5.835+1	5.834+1	5.832+1		5.837+1
U-236	1.177+0	1.213+0	1.161+0	9.987-1	1.145+0	1.151+0	1.154+0	1.143+0		1.142+0
U-238	2.862+4	2.862+4	2.863+4	1.802+4	2.862+4	2.863+4	2.864+4	2.862+4		2.864+4
Pu-239	2.412+2	2.452+2	2.359+2	2.061+2	2.344+2	2.339+2	2.362+2	2.335+2		2.334+2
Pu-240	6.597+0	5.089+0	6.451+0	5.675+0	6.016+0	6.061+0	6.104+0	6.556+0		6.548+0
Pu-241	1.324-1	7.014-2	1.292-1	1.035-1	1.083-1	1.099-1	1.117-1	1.361-1		1.357-1
Pu-242	1.789-3	5.979-4	1.724-3	1.186-3	1.245-3	1.268-3	1.314-3	1.890-3		1.888-3
FP	-	3.638+1	3.952+1	-	3.933+1	3.746+1	-	4.070+1		4.056+1

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.7
BENCHMARK PROBLEM IV FUEL INVENTORIES
FOR THE REFERENCE CONFIGURATION (CONTINUED)

FUEL INVENTORY (KG)	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY (HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>RADIAL BLANKETS - BOC2⁽¹⁾</u>										
U-235	6.838+1	6.798+1	6.822+1	4.208+1	6.802+1	6.824+1	6.814+1	6.811+1		6.816+1
U-236	9.147-1	1.019+0	9.558-1	4.653-1	9.921-1	9.454-1	9.697-1	9.701-1		9.691-1
U-238	3.280+4	3.278+4	3.279+4	2.066+4	3.277+4	3.280+4	3.279+4	3.277+4		3.279+4
Pu-239	1.155+2	1.682+2	1.571+2	1.451+2	1.654+2	1.556+2	1.660+2	1.609+2		1.610+2
Pu-240	2.180+0	2.028+0	2.419+0	2.365+0	2.515+0	2.242+0	2.392+0	2.537+0		2.538+0
Pu-241	2.270-2	1.595-2	2.709-2	2.527-2	2.652-2	2.242-2	2.419-2	2.901-2		2.901-2
Pu-242	1.279-4	6.372-5	1.654-4	1.386-4	1.463-4	1.162-4	1.302-4	1.705-4		1.797-4
FP	-	1.784+1	1.762+1	-	1.913+1	1.695+1	-	1.827+1		1.823+1
<u>AXIAL BLANKETS - BOC2⁽¹⁾</u>										
U-235	4.713+1	4.712+1	4.714+1	-	4.720+1	4.723+1	4.730+1	4.720+1		4.725+1
U-236	3.353-1	3.386-1	3.308-1	-	3.035-1	3.046-1	2.832-1	3.035-1		3.008-1
U-238	2.216+4	2.215+4	2.215+4	-	2.215+4	2.216+4	2.216+4	2.215+4		2.216+4
Pu-239	5.519+1	5.532+1	5.455+1	-	5.094+1	5.103+1	5.074+1	5.094+1		5.089+1
Pu-240	6.304-1	5.530-1	6.177-1	-	5.490-1	5.539-1	4.691-1	5.490-1		5.454-1
Pu-241	5.382-3	3.866-3	5.218-3	-	4.609-3	4.696-3	3.340-3	4.609-3		4.574-3
Pu-242	2.534-5	1.392-5	2.399-5	-	1.881-5	2.224-5	1.393-5	1.881-5		2.142-5
FP	-	5.395+0	5.528+0	-	5.315+0	5.269+0	-	5.315+0		5.293+0

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.7

BENCHMARK PROBLEM IV FUEL INVENTORIES
FOR THE REFERENCE CONFIGURATION (CONTINUED)

FUEL INVENTORY (KG)	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY (HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>DRIVER - EOC2⁽¹⁾</u>										
U-235	4.216+1	5.061+1	4.210+1	4.199+1	4.199+1	4.195+1	4.212+1	4.222+1		4.219+1
U-236	3.162+0	1.371+0	3.170+0	3.189+0	3.202+0	3.205+0	3.156+0	3.142+0		3.141+0
U-238	2.505+4	2.569+4	2.504+4	2.504+4	2.502+4	2.504+4	2.504+4	2.504+4		2.504+4
Pu-239	3.849+3	3.995+3	3.846+3	3.842+3	3.841+3	3.837+3	3.845+3	3.843+3		3.846+3
Pu-240	1.322+3	1.272+3	1.322+3	1.319+3	1.320+3	1.322+3	1.320+3	1.319+3		1.320+3
Pu-241	4.716+2	5.146+2	4.710+2	4.692+2	4.698+2	4.696+2	4.707+2	4.712+2		4.712+2
Pu-242	1.635+2	1.549+2	1.635+2	1.635+2	1.635+2	1.635+2	1.633+2	1.633+2		1.633+2
FP	-	5.375+2	1.335+3	-	1.343+3	1.291+3	-	1.367+3		1.334+3
<u>INNER BLANKET - EOC2⁽¹⁾</u>										
U-235	5.014+1	5.722+1	5.033+1	2.887+1	5.036+1	5.038+1	5.031+1	5.042+1		5.048+1
U-236	3.093+0	1.438+0	3.063+0	2.608+0	3.036+0	3.049+0	3.067+0	3.015+0		3.013+0
U-238	2.808+4	2.857+4	2.810+4	1.755+4	2.810+4	2.811+4	2.811+4	2.809+4		2.811+4
Pu-239	6.519+2	2.913+2	6.407+2	5.561+2	6.387+2	6.376+2	6.447+2	6.339+2		6.337+2
Pu-240	2.926+1	6.180+0	2.875+1	2.540+1	2.760+1	2.737+1	2.803+1	2.967+1		2.964+1
Pu-241	9.828-1	8.718-2	9.631-1	7.903-1	8.615-1	8.428-1	8.875-1	1.048+0		1.046+0
Pu-242	2.510-2	8.658-4	2.433-2	1.734-2	1.934-2	1.853-2	2.033-2	2.822-2		2.812-2
	-	4.388+1	1.515+2	-	1.488+2	1.409+2	-	1.551+2		1.545+2

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.7

BENCHMARK PROBLEM IV FUEL INVENTORIES
FOR THE REFERENCE CONFIGURATION (CONTINUED)

FUEL INVENTORY (KC)	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEOL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>RADIAL BLANKETS - EJC2⁽¹⁾</u>										
U-235	6.426+1	6.750+1	6.399+1	3.826+1	6.387+1	6.402+1	6.405+1	6.410+1		6.416+1
U-236	1.979+0	1.145+0	2.052+0	1.840+0	2.061+0	2.037+0	2.023+0	2.005+0		2.003+0
U-238	3.259+4	3.276+4	3.257+4	2.045+4	3.255+4	3.257+4	3.256+4	3.256+4		3.257+4
Pu-239	3.364+2	1.894+2	3.469+2	3.182+2	3.517+2	3.446+2	3.544+2	3.406+2		3.408+2
Pu-240	8.273+0	2.419+0	9.037+0	8.819+0	9.015+0	8.525+0	8.603+0	8.966+0		8.969+0
Pu-241	1.489-1	1.943-2	1.732-1	1.625-1	1.632-1	1.483-1	1.493-1	1.739-1		1.738-1
Pu-242	1.567-3	8.988-5	1.959-3	1.657-3	1.662-3	1.439-3	1.483-3	1.963-3		1.972-3
FP	-	2.047+1	4.972+1	-	5.090+1	4.732+1	-	4.871+1		4.860+1
<u>AXIAL BLANKETS - ECC2⁽¹⁾</u>										
U-235	4.457+1	4.685+1	4.461+1	-	4.485+1	4.487+1	4.503+1	4.485+1		4.491+1
U-236	1.014+0	4.112-1	1.000+0	-	9.222-1	9.264-1	8.677-1	9.222-1		9.139-
U-238	2.203+4	2.214+4	2.203+4	-	2.203+4	2.204+4	2.204+4	2.203+4		2.204+4
Pu-239	1.692+2	1.189+2	1.673+2	-	1.569+2	1.573+2	1.574+2	1.569+2		1.568+2
Pu-240	3.077+0	1.193+0	3.016+0	-	2.725+0	2.753+0	2.337+0	2.726+0		2.707+0
Pu-241	4.454-2	8.358-3	4.326-2	-	3.953-2	4.030-2	2.840-2	3.953-2		3.922-2
Pu-242	3.930-4	3.012-5	3.736-4	-	3.517-4	3.648-4	2.210-4	3.517-4		3.518-4
FP	-	1.160+1	1.994+1	-	1.913+1	1.888+1	-	1.913+1		1.904+

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

TABLE 4.8
BENCHMARK PROBLEM IV
NEUTRON BALANCE RESULTS
FOR THE REFERENCE CONFIGURATION

	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
NEUTRON BALANCE - 30C1 ⁽¹⁾	ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>SOURCE</u>										
FISSION	1.00000	1.00000	1.00000	1.00000	-	1.00000	1.00000	1.00000	1.00000	1.00000
n,2n	0.00254	0.00254	0.00254	0.00246	-	0.00000	0.00257	0.00254	-	0.00255
TOTAL SOURCE	1.00254	1.00254	1.00254	1.00246	-	1.00000	1.00257	1.00254	1.00000	1.00255
<u>ABSORPTIONS (F-FISSION; C-CAPTURE)</u>										
U-238 F	0.05952	0.05952	0.05952	0.05754	-	0.05989	0.05966	0.05958	0.05958	0.05958
C	0.48336	0.48336	0.48318	0.41792	-	0.47545	0.47702	0.47547	0.47304	0.47540
Pu-239 F	0.21647	0.21647	0.21647	0.21796	-	0.21691	0.21641	0.21610	0.21611	0.21610
C	0.05638	0.05638	0.05638	0.05569	-	0.05591	0.05540	0.05546	0.05535	0.05546
Pu-240 F	0.01532	0.01532	0.01532	0.01562	-	0.01541	0.01547	0.01545	0.01545	0.01545
C	0.01848	0.01848	0.01848	0.01834	-	0.01838	0.01824	0.01824	0.01824	0.01825
Pu-241 F	0.04394	0.04394	0.04395	0.04404	-	0.04394	0.04376	0.04371	0.04370	0.04371
C	0.00791	0.00791	0.00791	0.00786	-	0.00782	0.00782	0.00782	0.00778	0.00782
OTHER FUEL F	0.00880	0.00880	0.00878	0.00774	-	0.00869	0.00864	0.00868	0.00868	0.00868
C	0.00425	0.00425	0.00425	0.00382	-	0.00414	0.00414	0.00418	0.00394	0.00417
FISSION PRODUCT C	-	-	-	-	-	-	-	-	-	-
STRUCTURE C	0.06010 ⁽³⁾	0.05718	0.05775	0.04630	-	0.05740 ⁽³⁾	0.05477	0.09070 ⁽⁴⁾	0.05598	0.05648
COOLANT C	0.00254	0.00254	0.00249 ⁽³⁾	0.00216 ⁽³⁾	-	0.00243	0.00248 ⁽³⁾	0.00245	0.00245 ⁽³⁾	0.00242 ⁽³⁾
CONTROL	0.02030	0.02323 ⁽³⁾	0.02321 ⁽³⁾	0.01857 ⁽³⁾	-	0.02893	0.03470 ⁽³⁾	-	0.03486 ⁽³⁾	0.03458 ⁽³⁾
TOTAL F	0.34405	0.34404	0.34404	0.34290	-	0.34484	0.34396	0.34352	0.34352	0.34352
C	0.65332	0.65334	0.65364	0.57066	-	0.65046	0.65459	0.65432	0.65164	0.65457
<u>LEAKAGE</u>										
AXIAL	0.00039	0.00039	0.00013	0.08476	-	0.00036	0.00035	0.00036	0.00036	0.00012
RADIAL	0.00476	0.00477	0.00473	0.00414	-	0.00434	0.00367	0.00435	0.00435	0.00434
TOTAL	0.00515	0.00516	0.00486	0.08890	-	0.00470	0.00402	0.00471	0.00471	0.00446
TOTAL LOSSES	1.00252	1.00254	1.00254	1.00246	-	1.00000	1.00257	1.00254	0.99987	1.00255

- 1) See Table 4.2 for the Nomenclature
- 2) Results for the XY Plane Only
- 3) Captures Include Those for O-16
- 4) Captures Include Those for O-16 and Control Materials

TABLE 4.8
BENCHMARK PROBLEM IV
NEUTRON BALANCE RESULTS
FOR THE REFERENCE CONFIGURATION (CONTINUED)

NEUTRON BALANCE-EOC ⁽¹⁾	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>SOURCE</u>										
FISSION	1.00000	1.00000	1.00000	1.00000	-	1.00000	1.00000	1.00000		1.00000
n,2n	0.00256	0.00256	0.00257	0.00250	-	0.00000	0.00257	0.00257		0.00256
TOTAL SOURCE	1.00256	1.00256	1.00257	1.00250		1.00000	1.00257	1.00257		1.00256
<u>ABSORPTIONS (F-FISSION, C-CAPTURE)</u>										
U-238 F	0.05886	0.05890	0.05887	0.05683	-	0.05930	0.05894	0.05883		0.05886
C	0.46740	0.47264	0.46740	0.40411	-	0.46323	0.46402	0.46115		0.46128
Pu-239 F	0.22722	0.22300	0.22709	0.22622	-	0.22752	0.22579	0.22704		0.22713
C	0.06334	0.06181	0.06330	0.06129	-	0.06293	0.06202	0.06273		0.06275
Pu-240 F	0.01545	0.01555	0.01545	0.01578	-	0.01554	0.01558	0.01555		0.01555
C	0.01947	0.01946	0.01946	0.01928	-	0.01944	0.01926	0.01931		0.01933
Pu-241 F	0.03462	0.03450	0.03465	0.03528	-	0.03525	0.03463	0.03460		0.03459
C	0.00627	0.00625	0.00628	0.00634	-	0.00634	0.00624	0.00624		0.00624
OTHER FUEL F	0.00739	0.00765	0.00740	0.00654	-	0.00743	0.00736	0.00728		0.00730
C	0.00398	0.00406	0.00396	0.00361	-	0.00396	0.00392	0.00395		0.00391
FISSION PRODUCT C	0.01608	0.01464	0.01598	0.01472	-	0.01451	0.01575	0.01644		0.01597
STRUCTURE C	0.05260 ⁽³⁾	0.05699	0.05628	0.04437	-	0.05622 ⁽³⁾	0.05355	0.08254 ⁽⁴⁾		0.05474
COOLANT C	0.00253	0.00256	0.00248	0.00215	-	0.00246	0.00250 ⁽³⁾	0.00245		0.00241
OTHER CAPTURES C	0.01628	0.01918 ⁽³⁾	0.01922 ⁽³⁾	0.01323 ⁽³⁾	-	0.02141	0.02915 ⁽³⁾	-		0.02820 ⁽³⁾
TOTAL F	0.34354	0.33961	0.34347	0.34065	-	0.34504	0.34230	0.34333		0.34344
C	0.65395	0.65757	0.65438	0.56910	-	0.65050	0.65641	0.65641		0.65495
<u>LEAKAGE</u>										
AXIAL	0.00047	0.00053	0.00016	0.08880	-	0.00043	0.00043	0.00042		0.00015
RADIAL	0.00458	0.00484	0.00454	0.00395	-	0.00403	0.00343	0.00403		0.00402
TOTAL	0.00506	0.00537	0.00470	0.09275	-	0.00446	0.00386	0.00445		0.00417
TOTAL LOSSES	1.00255	1.00256	1.00255	1.00250	-	1.00000	1.00257	1.00257		1.00256

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

3) Captures Include Those for 0-16

4) Captures Include Those for 0-16 and Control Materials

TABLE 4.8

BENCHMARK PROBLEM IV
NEUTRON BALANCE RESULTS
FOR THE REFERENCE CONFIGURATION (CONTINUED)

NEUTRON BALANCE - $\text{BOC}_2^{(1)}$	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEOL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>SOURCE</u>										
FISSION	1.00000	1.00000	1.00000	1.00000	-	1.00000	1.00000	1.00000		1.00000
n,2n	0.00255	0.00255	0.00255	0.00249	-	0.00000	0.00254	0.00255		0.00255
TOTAL SOURCE	1.00255	1.00255	1.00255	1.00249	-		1.00254	1.00255		1.00255
<u>ABSORPTIONS (F-FISSION; C-CAPTURE)</u>										
U-238 F	0.05927	0.05927	0.05927	0.05726	-	0.05972	0.05936	0.05927		0.05929
C	0.47471	0.47712	0.47460	0.41052	-	0.46820	0.46941	0.46711		0.46715
Pu-239 F	0.22239	0.22064	0.22230	0.22230	-	0.22257	0.22203	0.22255		0.22263
C	0.06038	0.05972	0.06036	0.05905	-	0.05984	0.05941	0.05987		0.05982
Pu-241 F	0.01530	0.01535	0.01530	0.01562	-	0.01540	0.01543	0.01540		0.01542
C	0.01888	0.01890	0.01889	0.01872	-	0.01879	0.01868	0.01871		0.01871
Pu-241 F	0.03940	0.03941	0.03940	0.03975	-	0.03966	0.03934	0.03930		0.03923
C	0.00711	0.00712	0.00712	0.00712	-	0.00709	0.00706	0.00706		0.00705
OTHER FUEL F	0.00807	0.00818	0.00806	0.00711	-	0.00804	0.00795	0.00795		0.00754
C	0.00404	0.00413	0.00409	0.00362	-	0.00404	0.00401	0.00400		0.00402
FISSION PRODUCT C	0.00760	0.00689	0.00758	0.00699	-	0.00684 ⁽³⁾	0.00744	0.00777		0.00755
STRUCTURE C	0.05945 ⁽³⁾	0.05681	0.05713	0.04548	-	0.05670	0.05402	0.08640 ⁽⁴⁾		0.05545
COOLANT C	0.00253	0.00255	0.00248	0.00215 ⁽³⁾	-	0.00246	0.00249 ⁽³⁾	0.00243		0.00241
OTHER CAPTURES C	0.01798	0.02116 ⁽³⁾	0.02091 ⁽³⁾	0.01593 ⁽³⁾	-	0.02596	0.03188 ⁽³⁾	-		0.03137 ⁽³⁾
TOTAL F	0.34443	0.34286	0.34433	0.34249	-	0.34539	0.34411	0.34447		0.34451
C	0.65273	0.65439	0.65315	0.56965	-	0.64992	0.65440	0.65337		0.65367
<u>LEAKAGE</u>										
AXIAL	0.00043	0.00046	0.00014	0.08603	-	0.00039	0.00038	0.00039		0.00013
RADIAL	0.00496	0.00484	0.00494	0.00432	-	0.00430	0.00365	0.00432		0.00431
TOTAL	0.00539	0.00530	0.00508	0.09035	-	0.00469	0.00403	0.00471		0.00444
TOTAL LOSSES	1.00255	1.00255	1.00256	1.00249	-	1.00000	1.00254	1.00255		1.00256

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

3) Captures Include Those for 0-16

4) Captures Include Those for 0-16 and Control Materials

TABLE 4.8

BENCHMARK PROBLEM IV
NEUTRON BALANCE RESULTS
FOR THE REFERENCE CONFIGURATION (CONTINUED)

NEUTRON BALANCE - EOC2 ⁽¹⁾	THREE-DIMENSIONAL HEX-Z CALCULATIONS			TWO-DIMENSIONAL LINKED RZ-XY(HEX) CALCULATIONS			TWO-DIMENSIONAL RZ-CALCULATIONS			
	ANL	HEDL	ORNL	CE ⁽²⁾	GE	WARD	CE	GE	MIT	ORNL
<u>SOURCE</u>										
FISSION	1.00000	1.00000	1.00000	1.00000	-	1.00000	1.00000	1.00000		1.00000
n,2n	0.00257	0.00255	0.00257	0.00251	-	0.00000	0.00257	0.00257		0.00256
TOTAL	1.00257	1.00255	1.00257	1.00251	-		1.00257	1.00257		1.00256
<u>ABSORPTIONS (F-FISSION; C-CAPTURE)</u>										
U-238 F	0.05852	0.05926	0.05852	0.05647	-	0.05903	0.05858	0.05845		0.05846
C	0.45839	0.47893	0.45843	0.39626	-	0.45462	0.45543	0.45208		0.45221
Pu-239 F	0.23181	0.22255	0.23164	0.23030	-	0.23198	0.23043	0.23211		0.23220
C	0.06572	0.06086	0.06568	0.06338	-	0.06561	0.06449	0.06533		0.06538
Pu-240 F	0.01551	0.01541	0.01550	0.01585	-	0.01559	0.01562	0.01562		0.01562
C	0.01991	0.01981	0.01991	0.01971	-	0.01986	0.01971	0.01981		0.01984
Pu-241 F	0.03152	0.03628	0.03154	0.03222	-	0.03212	0.03153	0.03154		0.03152
C	0.00572	0.00657	0.00573	0.00580	-	0.00578	0.00569	0.00570		0.00569
OTHER FUEL F	0.00682	0.00811	0.00684	0.00603	-	0.00687	0.00680	0.00674		0.00674
C	0.00371	0.00414	0.00382	0.00349	-	0.00382	0.00378	0.00377		0.00377
FISSION PRODUCT C	0.02348	0.00844	0.02335	0.02165	-	0.02106	0.02306	0.02395		0.02352
STRUCTURE C	0.05830 ⁽³⁾	0.05698	0.05603	0.04389	-	0.05569 ⁽³⁾	0.05335	0.08040 ⁽⁴⁾		0.05453
COOLANT C	0.00252	0.00257 ⁽³⁾	0.00246	0.00214 ⁽³⁾	-	0.00245	0.00249 ⁽³⁾	0.00243		0.00240 ⁽³⁾
OTHER CAPTURES (CONTROL)	0.01520	0.01800 ⁽³⁾	0.01809 ⁽³⁾	0.01137 ⁽³⁾	-	0.02088	0.02759 ⁽³⁾	-		0.02637 ⁽³⁾
TOTAL F	0.34418	0.34160	0.34404	0.34077	-	0.34559	0.34296	0.34445		0.34453
C	0.65307	0.65568	0.65337	0.56769	-	0.64977	0.65559	0.65349		0.65371
<u>LEAKAGE</u>										
AXIAL	0.00051	0.00048	0.00017	0.08992	-	0.00046	0.00046	0.00046		0.00016
RADIAL	0.00480	0.00480	0.00477	0.00413	-	0.00418	0.00356	0.00418		0.00417
TOTAL	0.00531	0.00528	0.00494	0.09405	-	0.00464	0.00402	0.00464		0.00433
TOTAL LOSSES	1.00256	1.00255	1.00255	1.00251	-	1.00000	1.00257	1.00257		1.00257

1) See Table 4.2 for the Nomenclature

2) Results for XY Plane Only

3) Captures Include Those for 0-16

4) Captures Include Those for 0-16 and Control Materials

TABLE 4.9
BENCHMARK PROBLEM IV
COMPONENTS OF THE BURNUP REACTIVITY FOR
IRRADIATION CYCLE 1 OF THE REFERENCE CONFIGURATION

	(ANL TRIANGULAR-Z)	W-ARD (RZ-XY)	CE (RZ)
Global Reactivity Loss ⁽¹⁾	-0.00720	-0.00832	-0.01270
Reactivity Gain Due to Pu Build-up in the Blankets ⁽²⁾	0.04063	0.03977	0.03514
Reactivity Loss Due to Fission Product Build-up ⁽³⁾	-0.01270	-0.01235	-0.01245
Reactivity Loss Due to Burnup of Pu and U Isotopes ⁽⁴⁾	-0.03618	-0.03731	-0.04872

1) $(k_{\text{eff}} (2) - k_{\text{eff}} (1))/k_{\text{eff}} (1)$, See also Footnote 5

2) $(k_{\text{eff}} (3) - k_{\text{eff}} (1))/k_{\text{eff}} (1)$, See also Footnote 5

3) $(k_{\text{eff}} (4) - k_{\text{eff}} (1))/k_{\text{eff}} (1)$, See also Footnote 5

4) $(k_{\text{eff}} (5) - k_{\text{eff}} (1))/k_{\text{eff}} (1)$, See also Footnote 5

5) $k_{\text{eff}} (1) = k_{\text{eff}}$ corresponding to the beginning-of-cycle composition

$k_{\text{eff}} (2) = k_{\text{eff}}$ corresponding to the end-of-cycle composition with
the beginning of cycle control

$k_{\text{eff}} (3) = k_{\text{eff}}$ corresponding to the beginning-of-cycle compositions
for the driver zones and the end-of-cycle compositions for
the blankets

$k_{\text{eff}} (4) = k_{\text{eff}}$ corresponding to the beginning-of-cycle compositions
but including the end-of-cycle fission product densities

$k_{\text{eff}} (5) = k_{\text{eff}}$ corresponding to the end-of-cycle plutonium and uranium
densities for the driver fuel and the beginning-of-cycle
compositions for all other zones

swing for the first irradiation cycle were also calculated by several of the LCCEWG participants and reported in this section.

Differences in computer modeling techniques and code limitations accounted for many of the differences in the results of the Benchmark Four analysis. Thus, limited computer storage in the CE flux solution code DIFPB made it necessary to utilize alternate microscopic cross section data for a number of the less important fuel and structural materials in several regions. The substitution of the alternative data was found to have a noticeable impact on the results of the Benchmark Four computations.

Several of the participants used the supplied cross sections in the ANISN format. For this library, the absorption cross section, σ_a , was defined to be:

$$\sigma_a = \sigma_f + \sigma_c - \sigma_{n,2n}$$

where σ_f is the fission cross section, σ_c is the capture cross section, and $\sigma_{n,2n}$ is the (n,2n) cross section. The versions of 2DB used by MIT and W-ARD do not use the σ_c provided, but instead use $\sigma_a - \sigma_f$ for the capture cross section. This is equivalent to treating the (n,2n) reaction as negative absorption and thus effects the irradiation calculation, leading to an increase in the burnup reactivity. In addition, the neutron balance tables for these participants do not explicitly give the (n,2n) component; rather, it is included in the capture reaction.

Several alternative approaches were utilized to establish the outer boundary conditions in the XY dimensions for those codes which required extra regions to conform to a rectilinear model layout. In the version of 2DB used by MIT and W-ARD the extra triangular regions were assumed to be filled with a low density filler material. The use of the filler material was specified on the basis of the Benchmark Three results which indicated that this approach provided a

reasonable approximation to the leakage in the XY plane. Extra regions were also utilized at ORNL in the VENTURE code. However, a black absorber zone (non-return boundary condition) was assigned in these regions instead of the recommended low density filler material, since the latter specification resulted in a slow pointwise convergence in the outer regions of the model. The use of the black absorber appeared to have only a negligible effect on the results of the Benchmark Four calculations.

Fuel mass data at the beginning of the first irradiation cycle (BOC1) were computed from the initial atom densities as specified in Table 2.1. Small differences in the BOC1 mass data were noted (see Table 4.7), and these are primarily explained by differences in Avagadro's number.

Some small differences in the exposure dependent parameters (e.g., fissile inventory, burnup, reactivity change, etc.) were attributed to differences in the normalization of the regionwise neutron fluxes during the burnup step. That is, in some analyses the regionwise fluxes were normalized to the total reactor power based upon a net isotopic energy release per fission. In other analyses the fluxes were normalized to only the fission power (i.e., not including the approximate 6 percent of the total power due to capture gammas) using the supplied isotopic fission energy data. Here again the differences in the flux normalization approach had only a small effect on the Benchmark Four results.

Ambiguities in the frequency of the flux normalization procedure during a burnup interval has contributed to small differences in the results given in Tables 4.3 - 4.9. In general, the burnup calculations were performed using the normalized fluxes at the beginning of the burnup interval (e.g., from BOC to MOC or from MOC to EOC). However, the ORNL calculations were carried out based upon the renormalization of the fluxes over six sub-steps of the burnup interval. This, in effect, led to a small increase in the mean flux level which

was used for the burnup analysis (i.e., the relative flux distribution was not changed). The renormalization of the neutron flux level, along with the differences in the radial boundary conditions, contributed to the differences in the 3D results between ORNL and ANL. However, additional analyses are required to determine if these were the principal factors leading to a small radial skewing of the burnup sensitive data (i.e., the ANL results showed an increase in the burnup for the inner blankets and a decrease in the burnup for the radial blankets when compared to the ORNL results).

The results in Tables 4.3 - 4.9 provide some guidelines for the core design and analysis of a large heterogeneous fast reactor. In general, it was noted that the linked RZ-XY triangular model gave a reasonable representation of the three-dimensional configuration. The differences between the GE and W-ARD results for the linked RZ-XY triangular model are primarily due to differences in specifying the control atom density requirements in two-dimensional plane geometry. For a consistent modeling of the control in the triangular-Z and linked RZ-XY triangular representations (i.e., consistency in the prediction of the effective multiplication constants), the agreements in the respective mass balance data, power distributions, regionwise fuel burnup data and burnup reactivity swings were very good. The results in Tables 4.3 - 4.9 also indicate that the RZ model representation can be used to predict the mass balance data and fuel inventories with sufficient accuracy for most scoping type calculations. However, considerable caution must be used in interpreting the power distribution results from calculations in RZ geometry. The prediction of power peaking, particularly in the inner and radial blankets, is the major limitation of the RZ model representation.

It is of some interest to compare the relative control worth in the triangular-Z and RZ configurations. Thus, the ORNL results for control capture at BOC1 in Table 4.8 indicate that the annular ring modeling of the control assemblies tends to increase the control worth

by approximately 50%. This is attributed to the increased surface to volume representation in the RZ model (compared to the triangular-Z model), and this value is somewhat sensitive to the radial position of the control ring.

The components of the burnup reactivity for cycle 1 of the reference configuration are listed in Table 4.9; these include the reactivity gain due to the Pu buildup in the blanket regions, the reactivity loss due to fission product build-up, and the reactivity loss due to the burnup of plutonium and uranium in the driver fuel. Of particular significance in Table 4.9 is the contribution of the Pu build-up in the blanket regions of the large heterogeneous core. The build-up in the inner blankets, in particular, was found to have a significant impact on the overall global reactivity loss.

4.3 Sensitivity Studies

The sensitivities of the reference configuration results to variations in the neutronics and/or burnup model (e.g., mesh spacing, lumped fission product treatment, energy groups, zonewise burnup approach) were investigated by a number of the LCCEWG participants. The results of the sensitivity studies are reported in the remainder of this section. It is noted that only a partial listing of the results is included. For a complete listing of the results of the sensitivity studies by each of the contributing laboratories, the reader is directed to Volume II of this report.

4.3.1 Lumped Fission Product Representation

An odd-even lumped fission product representation for benchmark problem four was investigated by both ANL and HEDL. The results of these investigations are given in Table 4.10. Based upon the results, it was concluded that there are only small changes in the Benchmark Four performance parameters in going to two lumped fission products.

TABLE 4.10
EFFECT OF LUMPED FISSION
PRODUCT REPRESENTATION ON K_{eff}
FOR BENCHMARK PROBLEM IV

	<u>ONE-LUMPED FISSION PRODUCT</u>	<u>ODD-EVEN LUMPED FISSION PRODUCTS</u>	<u>DIFFERENCE</u>
<u>ANL RESULTS</u> ⁽¹⁾			
K_{eff} - BOC1	1.00377	1.00377	-
K_{eff} - EOC1	1.00153	1.00141	1.2-4
K_{eff} - BOC2	1.00439	1.00434	0.5-4
K_{eff} - EOC2	1.00320	1.00316	0.4-4
ΔK - CYCLE 1	-2.24-3	-2.36-3	1.2-4
ΔK - CYCLE 2	-1.19-3	-1.18-3	-0.1-4
<u>HEDL RESULTS</u> ⁽¹⁾			
K_{eff} - BOC1	1.00376	1.00376	-
K_{eff} - EOC1	0.98988	0.98976	1.2-4
K_{eff} - BOC2	0.99982	0.99978	0.4-4
K_{eff} - EOC2	0.99584	0.99579	0.5-4
ΔK - CYCLE 1	-1.388-2	-1.400-2	1.2-4
ΔK - CYCLE 2	-3.98 -3	-3.99 -3	0.1-4

1) See Table 4.2 for the nomenclature

4.3.2 Mesh Sensitivities

The sensitivities of the effective multiplication constant, breeding gain and power distribution to the mesh spacing were investigated by ORNL through the use of the mesh-edge, diffusion theory code VALE. The results of the mesh sensitivity analysis are shown in Table 4.11. The VENTURE results for the Benchmark Four reference configuration are also shown in Figure 4.11 to provide a comparison of the results of mesh-edge and mesh-center approaches. The results are given for only the first irradiation cycle of the benchmark problem since the second irradiation cycle results are not appreciably different.

The bias in the effective multiplication factor in going from coarse mesh to the extrapolated infinite mesh using the VALE code was found to be approximately -0.25 percent. This bias factor was identical to that which was computed for the Benchmark Problem Three analysis (Reference 2). Since the major change in going from Benchmark Problem Three to Benchmark Four was the incorporation of the third dimension (i.e., the axial dimension), it was concluded that the mesh sensitivity of k_{eff} was primarily that due to the sensitivity to the triangular mesh specifications in the XY plane (this was later confirmed by several independent analyses). The differences between the results of the coarse mesh-edge and coarse mesh-center computations were found to be approximately the same as reported for Benchmark Problem Three.

The results of the mesh sensitivity studies have indicated two areas of concern. First, the extrapolated results shown in Table 4.11 give evidence that the burnup reactivity is overpredicted when using the coarse mesh-center approach, and underpredicted when using the coarse mesh-edge approach. As a consequence, a coarse mesh analysis may lead to an incorrect assessment of the control requirements. It was interesting to note that for Benchmark Problem Four, the coarse mesh-edge calculations gave somewhat better results than the coarse

TABLE 4.11
BENCHMARK PROBLEM IV
PARAMETER SENSITIVITY TO MESH SPACING

<u>PARAMETER</u> ⁽¹⁾	<u>VENTURE</u>	<u>VALE</u>		
	<u>REFERENCE</u> ⁽²⁾	<u>REFERENCE</u> ⁽³⁾	<u>FINE-MESH</u> ⁽⁴⁾	<u>EXTRAPOLATED</u> ⁽⁵⁾
<u>K_{eff}</u>				
BOC1	1.003772	0.996033	0.997883	0.998500
EOC1	1.001252	0.996742	0.997614	0.997905
<u>ΔK</u>				
CYCLE 1	-2.52-3	0.709-3	-0.269-3	-0.595-3
<u>BREEDING GAIN</u> (Kg)	4.787+2	4.788+2	4.793+2	4.795+2
<u>PEAK POWER (Kw/cm³)</u>				
DRIVER FUEL - BOC1	5.280-1	5.308-1	5.270-1	5.294-1
DRIVER FUEL - EOC1	5.400-1	5.469-1	5.445-1	5.437-1
INNER BLANKET - BOC1	8.051-2	1.058-1	1.041-1	1.035-1
INNER BLANKET - EOC1	2.215-1	2.447-1	2.432-1	2.427-1
RADIAL BLANKET - BOC1	6.271-2	8.345-2	8.544-2	8.570-2
RADIAL BLANKET - EOC1	7.307-2	8.974-2	9.116-2	9.163-2
<u>DRIVER FUEL PEAK/AVERAGE POWER DENSITY</u>				
BOC1	1.612	1.634	1.618	1.623
EOC1	1.812	1.849	1.836	1.832

- 1) See Table 2.4 for the Nomenclature
- 2) 6 Points per hex and 42 mesh points in axial dimension
- 3) 3 points per hex and 42 mesh points in axial dimension
- 4) 12 points per hex and 84 mesh points in axial dimension
- 5) Extrapolated VALE results were obtained from the reference (X_1) and Fine-mesh (X_2) VALE solutions using the following extrapolation formula

$$X_{\infty} = (4X_2 - X_1)/3$$

mesh-center calculations, when compared with the extrapolated values.

The mesh sensitivity study also demonstrated the well-known problems inherent in using cell average power densities to predict peak power densities in regions with large flux gradients. In particular it was found that peak power densities in the blankets were underpredicted by from 10 to 40 percent (depending upon the blanket exposures) by the codes based upon mesh-center differencing. The coarse mesh-edge approach provided a better prediction of the peak at the edge of the cell. It is, of course, possible to derive cell edge values of the flux directly from the mesh-center results by using the assumed flux shape implicit in the differencing algorithm, and options do exist in mesh-centered codes (e.g., DIF3D) which perform this calculation. When such options are used, accurate peak power densities are predicted by codes using mesh-centered differences equations (Reference Toppel, et al., ANL Kiamesha Lake paper).

Similar mesh sensitivity results to those discussed above had been reported earlier in an ANL study of the benchmark configuration (Toppel, et al.). In that study it was shown that a nodal solution with just one mesh point per hexagon gave results in close agreement with the fine mesh finite difference solutions. In particular, the nodal eigenvalues were found to be approximately 0.3% lower than the 6 mesh point per hexagon finite-difference results while blanket peaking and burnup were significantly higher. In effect the coarse mesh nodal calculations showed the same effects as the current fine mesh sensitivity study.

4.3.3 Sensitivity to the Number of Energy Groups

The results of 20-group calculations by GE for Benchmark Problem Four are compared with the 8-group results in Table 4.12. The results in Table 4.12 indicate little significant difference between the two sets of results. The differences between the results of the 20-group and 8-group calculations of k_{eff} varied from 0.00154 at BOC1 to

TABLE 4.12
PARAMETER SENSITIVITY TO
THE NUMBER OF ENERGY GROUPS
(TWO-DIMENSIONAL RZ MODEL)

<u>PARAMETER</u> ⁽¹⁾	<u>8-GROUPS</u>	<u>20-GROUPS</u>
<u>K_{eff}</u>		
BOC1	1.00249	1.00095
EOC1	1.00103	0.99964
BOC2	1.00474	1.00334
EOC2	1.00411	1.00298
<u>ΔK</u>		
CYCLE 1	1.46-3	1.31-3
CYCLE 2	0.63-3	0.36-3
<u>BREEDING GAIN (KG)</u>		
CYCLE 1	4.625+2	4.655+2
CYCLE 2	4.221+2	4.248+2
<u>PEAK POWER (W/cm³)</u>		
DRIVER FUEL - BOC1	5.096+2	5.103+2
EOC1	5.137+2	5.145+2
INNER BLKT - BOC1	0.702+2	0.703+2
EOC1	2.142+2	2.144+2
RADIAL BLKT - BOC1	0.473+2	0.477+2
EOC1	0.534+2	0.530+2

1. See Table 4.2 for the nomenclature.

0.00113 at EOC2. As a consequence the burnup reactivity in going to 20-groups was reduced from 0.00146 to 0.00131 for cycle 1 and from 0.00064 to 0.00036 in cycle 2. These results are consistent with the results obtained for LCCEWG Benchmark Three.

4.3.4 Zone Versus Assembly Burnup

The Benchmark Four calculations at W-ARD were carried out using two different models in order to investigate the sensitivity of the performance parameters to the burnup zone specifications. The base case utilized the reference hexagonal model with two burnup zones per ring, (one zone for each refueling batch, based upon the rings shown in Figure 2.1), while the second case used a hexagonal model with burnup computations performed by individual assembly. The same RZ model was linked with each of the hex models to produce the two sets of results given in Table 4.13.

The results in Table 4.13 indicate that the differences between the base case and the assemblywise hex burn case were minor. One difference which should be noted is the increase in radial blanket peak power in the assemblywise burnup case. This is due to the better representation of the Pu buildup in the radial blankets nearest the core, resulting in higher power in these assemblies than when they were lumped with the rest of the radial blankets.

4.3.5 Alternative Burnup Approach

An alternative burnup approach based upon a redepletion calculation was performed by GE to assess the sensitivity of the Benchmark Four performance parameters to the burnup model. In this approach the burnup was carried out over the entire irradiation cycle based upon the flux solutions at BOC and EOC (with control inserted to the specified BOC and EOC positions). A linear interpolation procedure was utilized to determine the regionwise fluxes for several intermediate burnup steps over the operations cycle. Since the EOC

TABLE 4.13
BENCHMARK PROBLEM IV
ZONE VERSUS ASSEMBLY BURNUP RESULTS
(LINKED RZ-XY REACTOR MODEL)

<u>PARAMETER</u> (1)	<u>ZONE BURNUP</u> (2)	<u>ASSEMBLY BURNUP</u> (3)
<u>K_{eff}</u>		
BOC1	1.00329	1.00329
EOC1	1.00038	1.00101
BOC2	1.00345	1.00387
EOC2	1.00188	1.00271
<u>ΔK</u>		
CYCLE 1	-2.91-3	-2.28-3
CYCLE 2	-1.57-3	-1.16-3
<u>BREEDING GAIN (KG fissile PU)</u>		
CYCLE 1	461.9	461.1
CYCLE 2	427.2	425.7
<u>PEAK POWER (KW/cm³)</u>		
DRIVER FUEL - EOC1	5.446-1	5.356-1
- EOC2	5.150-1	5.078-1
INNER BLANKET - EOC1	2.154-1	2.125-1
EOC2	2.822-1	2.790-1
RADIAL BLANKET - EOC1	7.333-2	8.482-2
EOC2	8.107-2	1.435-1

-
- 1) See Table 4.2 for the nomenclature.
 - 2) Burnup calculations performed for fuel zones as shown in Figures 2.1 and 2.3
 - 3) Burnup calculations performed for each fuel assembly.

densities were not initially known, this required an iterative computational procedure. However, essentially only one iteration step was required to converge on the EOC fluxes. The results of the alternative burnup approach are compared with the results of the reference burnup approach in Table 4.14 (for the first operating cycle only). In interpreting the results it is noted that there is a substantial shift in the power (and flux) distribution from the outer core regions towards the inner core regions over the irradiation cycle (see Figure 4.4). This was due, primarily, to the build up of Pu in the inner blankets. In the reference burnup procedure, the flux which was used for the fuel burnup remained constant at the BOC condition until the MOC (middle-of-cycle) condition was reached; the flux then remained constant at the MOC condition (but with the EOC control position) for the second burn step. Thus, the effect of the radial shift in the flux was delayed for one-half cycle using the reference procedure. In the alternative burnup approach the flux during the burnup calculations was obtained by interpolating between the BOC and EOC endpoints. Therefore, the effect of the radial shift in the flux was accounted for at an earlier time. This resulted in a small reduction in the power in the radial blankets, a small increase in the breeding gain and a slightly higher EOC k_{eff} (i.e., the burnup reactivity was reduced). The significance of such results may depend on the design criteria and margins relevant to specific concepts. The results do indicate that the burnup calculations are not particularly sensitive to the modeling of the control rods during the irradiation cycle.

4.3.6 Extended Burnup Chain Investigation

The nuclide inventories for the extended burnup chain shown in Figure 2.4 were calculated by ORNL using VALE flux solutions for the reference Benchmark Four configuration. The effects of the higher actinides on the Benchmark Four performance parameters were determined by comparing the results of the abbreviated burnup chain and the extended burnup chain calculations as shown in Figure 4.15. These

TABLE 4.14
BENCHMARK PROBLEM IV
PARAMETER SENSITIVITY TO ALTERNATE BURNUP APPROACH
(RZ REACTOR MODEL)

<u>PARAMETER</u> (1)	<u>REFERENCE BURNUP MODEL</u> (2)	<u>ALTERNATE BURNUP APPROACH</u> (3)
<u>K_{eff}</u>		
BOC1	1.00249	1.00249
EOC1	1.00103	1.00191
<u>ΔK</u>		
CYCLE 1	1.46-3	0.58-3
<u>BREEDING GAIN (Kg Fissile PU)</u>		
CYCLE 1	4.625+2	4.688+2
<u>PEAK POWER (KW/cm³)</u>		
DRIVER FUEL - BOC1	5.096-1	5.096-1
- EOC1	5.137-1	5.106-1
INNER BLANKET - BOC1	0.702-1	0.702-1
- EOC1	2.142-1	2.158-1
RADIAL BLANKET - BOC1	4.73-2	4.73-2
- EOC1	5.34-2	5.31-2

-
- 1) See Table 4.2 for the nomenclature.
 - 2) See Sections II and III.
 - 3) Alternate burnup model based upon a redepletion calculation using the BOC and EOC neutron fluxes.

TABLE 4.15

THE IMPACT OF HIGHER ACTINIDES ON THE
BENCHMARK FOUR PERFORMANCE PARAMETERS

<u>Parameter⁽¹⁾</u>	<u>Abbreviated Burnup Chain⁽²⁾</u>	<u>Extended Burnup Chain⁽³⁾</u>
<u>K_{eff}</u>		
BOC1	0.996033	0.996033
EOC1	0.996742	0.996421
BOC2	0.998225	(0.998065) ⁽⁴⁾
EOC2	0.999273	0.998985
<u>ΔK</u>		
Cycle 1	0.709-3	0.388-3
Cycle 2	1.048-3	(0.920-3) ⁽⁴⁾
<u>Breeding Gain (kg Fissile Pu)</u>		
Cycle 1	4.788+2	4.786+2
Cycle 2	4.409+2	4.402+2
<u>Peak Power (w/cm³)</u>		
Driver Fuel - BOC1	5.308-1	5.308-1
- EOC1	5.469-1	5.472-1
Inner Blanket - BOC1	1.058-1	1.058-1
EOC1	2.447-1	2.446-1
Radial Blanket - BOC1	8.345-2	8.345-2
EOC1	8.974-2	8.970-2

1) See Table 4.2 for the Nomenclature

2) Burnup chain includes only U-235, U-236, U-238, Pu-239, Pu-240, Pu-241 and Pu-242

3) See Figure 2.4

4) Estimated value only

results indicated that a higher actinide accountability had only a negligible effect on the breeding gain and power distribution. The inclusion of the higher actinides had a somewhat larger effect on the burnup reactivity swing, but the total impact of this change was still small.

Mass balance data for the actinides (including fuel materials) were computed for the first irradiation cycle of benchmark problem four and are listed in Table 4.16 (the mass balance data correspond to 383.5 Full Power Days at 2740 MW). This information can be of general use in planning fuel handling, shielding and waste storage operations.

TABLE 4.16

ACTINIDE MASS BALANCE DATA
FOR IRRADIATION CYCLE 1

ISOTOPE ⁽¹⁾	ACTINIDE INVENTORY AT BOC1 (KG)		ACTINIDE INVENTORY AT EOC1 (KG)				Mass Increase (Decrease) During Irradiation Cycle 1 (KG)
	Driver	All Blkt	Driver	Inner Blkt	Radial Blkt	Axial Blkt	
U-234	-		4.436-3	1.515-3	6.005-4	2.576-4	6.809-3
U-235	5.687+1	1.832+2	4.596+1	5.431+1	6.688+1	4.583+1	(2.709+1)
U-236			2.403+0	2.126+0	1.302+0	6.810-1	6.512+0
U-237			1.199-1	4.597-2	1.491-2	6.762-3	1.875-1
U-238	2.612+4	8.409+4	2.536+4	2.838+4	3.273+4	2.209+4	(1.650+3)
Np-237			4.373+0	1.539+0	5.858-1	2.487-1	6.747+0
Pu-238			1.003+0	1.262-1	2.374-2	7.312-3	1.160+0
Pu-239	4.101+3		3.922+3	4.400+2	2.175+2	1.136+2	(5.921+2)
Pu-240	1.236+3		1.302+3	1.189+1	3.324+0	1.267+0	8.243+1
Pu-241	6.205+2		5.076+2	2.370-1	3.697-2	1.083-2	(1.126+2)
Pu-242	1.481+2		1.607+2	3.209-3	2.274-4	5.155-5	1.260+1
Am-241			2.588+1	2.934-3	5.069-4	1.392-4	2.588+1
Am-242			8.426-3	1.287-6	9.233-8	2.140-8	8.427-3
Am-242m			2.149-1	1.327-5	1.063-6	2.320-7	2.149-1
Am-243			5.583+0	3.520-5	1.327-3	2.521-7	5.584+0
Cm-242			9.037-1	6.706-5	5.083-6	1.117-6	9.038-1
Cm-243			9.741-3	5.141-7	2.230-8	4.221-9	9.742-3
Cm-244			2.654-1	8.338-7	1.542-8	2.461-9	2.654-1
FPT			9.331+2	8.244+1	2.686+1	1.156+1	1.054+3

1) See Figure 2.4 for extended burnup chain

V. SUMMARY AND CONCLUSIONS

An analysis of a large heterogeneous fast reactor has been carried out by the members of the LCCEWG in accordance with the specifications for LCCEWG Benchmark Problem Four. This study has included an investigation of different approaches for modeling the three-dimensional benchmark geometry based upon Triangular-Z model representations, linked RZ-XY(Triangular) model representations, and RZ model representations. In general, the results have indicated that the linked RZ-XY (Triangular) two-dimensional description of the Benchmark Four geometry can be used to predict mass balance data, power distributions, regionwise fuel exposure data and burnup reactivities with good accuracy when compared with the Triangular-Z benchmark results. Similarly, the results of the RZ calculations have indicated that this modeling approach can be utilized to predict mass balance data and fuel inventories with an accuracy which is probably sufficient for most scoping type calculations. However, the results of the power distribution calculations in RZ geometry were found to be in poor agreement with the Triangular-Z results. In particular, the RZ model was deemed to be inadequate for predicting the peak power densities for the inner and radial blanket assemblies. Further ambiguities in predicting control characteristics in the RZ model were noted. These deficiencies, along with the increasing efficiency of solution algorithms and the availability of faster computers may encourage a trend toward increasing reliance on direct 3D calculations for detailed design work.

The Benchmark Four specifications have also included an extensive burnup analysis of a typical LMFBR fuel cycle. As a consequence, increased emphasis was placed on the design and analysis procedures for the burnup calculations. Most of the small differences which were noted in the results of the benchmark analysis were attributed to ambiguities in carrying out the burnup calculations. These included

differences in the renormalization of the regionwise neutron fluxes throughout the burnup interval, and differences in the treatment of the capture gamma contribution to the total power during the renormalization step. Differences in the treatment of the $n,2n$ reaction also led to small differences in the burnup results. Alternative burnup approaches based upon redepletion calculations using the beginning-of-cycle and end-of-cycle fluxes were investigated for Benchmark Problem Four. The results of the redepletion calculations indicated a slight skewing of the burnup sensitive data towards the center of the core when compared to the results of the reference calculations (i.e., slightly greater burnup in the inner blankets and slightly less burnup in the radial blankets for the redepletion case). However, the changes were small and it was concluded that the prediction of the fuel cycle parameters was only slightly sensitive to the treatment of the control during the burnup calculations.

Studies were carried out to determine the sensitivities of the Benchmark Four performance parameters to changes in the neutronics and/or burnup model. Of particular interest were the effects of changes in the mesh spacing, lumped fission product treatment, energy groups and zonewise burnup model. The performance parameters were found to be most sensitive to the changes in the mesh specifications, and based upon the results presented in Section 4.3.2, it was concluded that the mesh sensitivity was primarily that due to the coarse mesh specifications in the XY plane. The coarse mesh bias factors (i.e., differences between the extrapolated fine mesh and coarse mesh results) were found to be significantly different for the mesh-center and mesh-edge flux solutions, and were moderately dependent upon the fuel exposure. Thus, the mesh-edge bias factors for the effective multiplication constant varied from between -0.25 to $-0.12 \Delta k$ during irradiation cycle 1. In contrast, the mesh-center bias factors varied from about 0.52 to $0.33 \Delta k$ during the first cycle. The results of the mesh sensitivity study also point out (what has long been recognized as) the difficulty in predicting peak power

densities in blanket regions when using the average flux derived from a coarse mesh-center approach. That is, the results of the Benchmark Four analysis indicated differences for the peak power densities for the inner and radial blankets of between 10 to 40% for the results of the coarse mesh-center and extrapolated fine mesh computations. It has, however, been shown that edge power extrapolation can be performed using the coarse mesh-center computations to obtain accurate power densities in regions of large flux gradients such as blankets. It is noted that this ability to predict the peak densities is important in specifying design margins for fuel failure.

An odd-even lumped fission product representation was investigated as part of the Benchmark Four analysis. The results of this investigation indicated that the performance parameters were only slightly sensitive to the fission product model. Thus, it was concluded that the use of one lumped fission product was adequate for most design and analysis applications for burnups of less than approximately 200,000 MWd/T.

Increasing the number of energy groups from 8 to 20 was found to decrease k_{eff} by about 0.02-0.03 percent. This change was found to be consistent with the energy group sensitivity which was determined for Benchmark Three, and it was concluded that the use of an 8 energy group structure was most adequate for the analysis of the large heterogeneous system (i.e., for calculating parameters other than the Doppler coefficient and sodium void reactivity). The sensitivities of the benchmark results to the number of fuel assemblies which are included in the composite burnup zone specifications were also found to be small. However, it was noted that there was a significant increase in the predicted peak radial blanket power density when going from the zonewise to the assembly-wise burnup step. This was attributed to the better representation of those blanket assemblies adjacent to the core and could have been achieved by considering the first row of radial blanket as a distinct burnup zone.

An abbreviated and extended actinide burnup chain comparison was made. This did not show strong sensitivities of the basic properties of the neutronic solution, such as power distribution and reactivity, to the inclusion of minor actinides in the coupled neutronic depletion calculations. Thus, it is possible that good approximations to inventories of minor actinides might be obtained by cost-effective use of flux edits, from the abbreviated chain analysis, in combination with depletion routines which contain more complete chain models.

The results of the LCCEWG Benchmark Four analysis of a large heterogeneous fast reactor have provided guidelines for carrying out neutronics and burnup calculations for a wide range of design and analysis applications, and for testing the accuracies and capabilities of different codes and methods. As a consequence, excellent progress has continued to be made in the implementation of the methods and procedures which are required for the successful design of a large LMFBFR.

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APPENDIX A
DESCRIPTION OF THE CROSS SECTION LIBRARY
FOR THE LARGE CORE CODE EVALUATION
WORKING GROUP BENCHMARK PROBLEM FOUR

Table A-1

Inner Blanket Zones I, II and III

<u>Isotope</u>	<u>HABSID</u>	<u>HMAT</u>	<u>ANISN^(a)</u>	<u>Density</u>	<u>Temperature</u>
O16	O16	A0816	108	2.7176-2	1500.
Na	NA23	A1123	111	6.2401-3	700.
Cr	Cr	A2400	124	2.3756-3	700.
Mn	MN55	A2555	125	2.2528-4	700.
Fe	FE	A2600	126	8.1994-3	700.
Ni	NI	A2800	128	1.6266-3	700.
Mo	MO	A4200	130	1.8431-4	700.
U-235	U235	A9235	135	2.9907-5	1500.
U-236	U236	A9236	136	0.0	1500.
U-238	U238	A9238	138	1.3558-2	1500.
Pu-239	PU239	A9439	139	0.0	1500.
Pu-240	PU240	A9440	140	0.0	1500.
Pu-241	PU241	A9441	141	0.0	1500.
Pu-242	PU242	A9442	142	0.0	1500.
FP Total	FPTOT	A9990	146	0.0	1500.
FP Odd	FPODD	A9991	147	0.0	1500.
FP Even	FPEVEN	A9992	148	0.0	1500.
U-238	U238	A9238D	4138	1.3558-2	900.

(a) On the ANISN library, the cross section table positions are defined for group g as follows:

<u>Position</u>	<u>Contents</u>	<u>Description</u>
1	$(\sigma_h)_g$	Heating cross section (w.sec.b)
2	$(\sigma_c)_g$	Capture cross section
3	$(\sigma_f)_g$	Fission cross section
4	$(\sigma_a)_g$	Absorption cross section = $\sigma_c + \sigma_f - \sigma_{n,2n}$
5	$(\nu\sigma_f)_g$	Fission production cross section
6	$\sigma_{g' \rightarrow g}$	Scattering matrix; $g' = g + 7 - p$ for position p

Table A-2
Driver Zones I, II and III

<u>Isotope</u>	<u>HABSID</u>	<u>HMAT</u>	<u>ANISN</u>	<u>Density</u>	<u>Temperature</u>
O16	O16	B0816	208	2.1460-2	1500.
Na	NA23	B1123	211	8.2217-3	700.
Cr	CR	B2400	224	2.8111-3	700.
Mn	MN55	B2555	225	2.6609-4	700.
Fe	FE	B2600	226	9.6814-3	700.
Ni	NI	B2800	228	1.9207-3	700.
Mo	MO	B4200	230	2.1764-4	700.
U-235	U235	B9235	235	1.9168-5	1500.
U-236	U236	B9236	236	0.0	1500.
U-238	U238	B9238	238	8.6912-3	1500.
Pu-239	PU239	B9439	239	1.3591-3	1500.
Pu-240	PU240	B9440	240	4.0789-4	1500.
Pu-241	PU241	B9441	241	2.0392-4	1500.
Pu-242	PU242	B9442	242	4.8481-5	1500.
FP Total	FPTOT	B9990	246	0.0	1500.
FP Odd	FPODD	B9991	247	0.0	1500.
FP Even	FPEVEN	B9992	248	0.0	1500.
U-238	U238	B9238D	4238	8.6912-3	900.
Pu-239	PU239	B9439D	4239	1.3591-3	900.
Pu-240	PU240	B9440D	4240	4.0789-4	900.

Table A-3
Radial Blanket Zone

<u>Isotope</u>	<u>HABSID</u>	<u>HMAT</u>	<u>ANISN</u>	<u>Density</u>	<u>Temperature</u>
O16	O16	G0816	708	2.7176-2	900.
Na	NA23	G1123	711	6.2401-3	700.
Cr	CR	G2400	724	2.3756-3	700.
Mn	MN55	G2555	725	2.2528-4	700.
Fe	FE	G2600	726	8.1994-3	700.
Ni	NI	G2800	728	1.6266-3	700.
Mo	MO	G4200	730	1.8431-4	700.
U-235	U235	G9235	735	2.9907-5	900.
U-236	U236	G9236	736	0.0	900.
U-238	U238	G9238	738	1.3558-2	900.
Pu-239	PU239	G9439	739	0.0	900.
Pu-240	PU240	G9440	740	0.0	900.
Pu-241	PU241	G9441	741	0.0	900.
Pu-242	PU242	G9442	742	0.0	900.
FP Total	FPTOT	G9990	746	0.0	900.
FP Odd	FPODD	G9991	747	0.0	900.
FP EVEN	FPEVEN	G9992	748	0.0	900.

Table A-4

Axial Blanket Above Driver Zones I, II and III

<u>Isotope</u>	<u>HABSID</u>	<u>HMAT</u>	<u>ANISN</u>	<u>Density</u>	<u>Temperature</u>
O16	O16	P0816	1008	2.1460-2	900.
Na	NA23	P1123	1011	8.2217-3	700.
Cr	CR	P2400	1024	2.8111-3	700.
Mn	MN55	P2555	1025	2.6609-4	700.
Fe	FE	P2600	1026	9.6814-3	700.
Ni	NI	P2800	1028	1.9207-3	700.
Mo	MO	P4200	1030	2.1764-4	700.
U-235	U235	P9235	1035	2.7953-5	900.
U-236	U236	P9236	1036	0.0	900.
U-238	U238	P9238	1038	1.2673-2	900.
Pu-239	PU239	P9439	1039	0.0	900.
Pu-240	PU240	P9440	1040	0.0	900.
Pu-241	PU241	P9441	1041	0.0	900.
Pu-242	PU242	P9442	1042	0.0	900.
FP Total	FPTOT	P9990	1046	0.0	900.
FP Odd	FPODD	P9991	1047	0.0	900.
FP Even	FPEVEN	P9992	1048	0.0	900.

Table A-5
Radial Shield Zone

<u>Isotope</u>	<u>HABSID</u>	<u>HMAT</u>	<u>ANISN</u>	<u>Density</u>	<u>Temperature</u>
Na	NA23	H1123	811	2.2370-3	700.
Cr	CR	H2400	824	1.3180-2	700.
Mn	MN55	H2555	825	1.2480-3	700.
Fe	FE	H2600	826	4.5410-2	700.
Ni	NI	H2800	828	9.0060-3	700.
Mo	MO	H4000	830	1.0210-3	700.

Table A-6
Fission Gas Plenum & Axial Shield Zones

<u>Isotope</u>	<u>HABSID</u>	<u>HMAT</u>	<u>ANISN</u>	<u>Density(FGP/AS)</u>	<u>Temperature</u>
Na	NA23	Q1123	1111	8.2217-3/3.9930-3	700.
Cr	CR	Q2400	1124	2.8111-3/1.2280-2	700.
Mn	MN55	Q2555	1125	2.6609-4/1.1620-3	700.
Fe	FE	Q2600	1126	9.6814-3/4.2290-2	700.
Ni	NI	Q2800	1128	1.9207-3/8.3900-3	700.
Mo	MO	Q4200	1130	2.1764-4/9.5070-4	700.

Table A-7
Control in Zone

<u>Isotope</u>	<u>HABSID</u>	<u>HMAT</u>	<u>ANISN</u>	<u>Density</u>	<u>Temperature</u>
B-10	B10	I0510I	904	2.9110-2	700.
B-11	B11	I0511I	905	2.5310-3	700.
C	C	I0600I	906	8.2700-3	700.
Na	NA23	I1123I	911	1.0490-3	700.
Cr	CR	I2400I	924	3.1030-3	700.
Mn	MN55	I2555I	925	2.9370-4	700.
Fe	FE	I2600I	926	1.0690-2	700.
Ni	NI	I2800I	928	2.1200-3	700.
Mo	MO	I4200I	930	2.4030-4	700.

Table A-8
Control Out Zone

<u>Isotope</u>	<u>HABSID</u>	<u>HMAT</u>	<u>ANISN</u>	<u>Density</u>	<u>Temperature</u>
Na	NA23	I11230	961	1.8420-2	700.
Cr	CR	I24000	974	1.3980-3	700.
Mn	MN55	I25550	975	1.3240-4	700.
Fe	FE	I25000	976	4.8170-3	700.
Ni	NI	I28000	978	9.5520-4	700.
Mo	MO	I42000	980	1.0820-4	700.