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SPECIFICATIONS FOR THE LARGE CORE CODE
EVALUATION WORKING GROUP BENCHMARK PROBLEM FOUR

C. L. Cowan and R. Protsik

September, 1981

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GENERAL ELECTRIC COMPANY
ADVANCED REACTOR SYSTEMS DEPARTMENT

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ADVANCED REACTOR SYSTEMS DEPARTMENT
XL-589-10074

September 29, 1981

TO: Distribution

SUBJECT: Forwarding of LCCEWG Benchmark Four Specifications

The specifications for the Large Core Code Evaluation Working Group (LCCEWG) Benchmark Problem Four are forwarded as an attachment to this letter. Many of the comments which have been transmitted by the working group members have been incorporated in this final report. In particular, there has been a reduction in the number of axial mesh points to provide a less expensive base for the 3-D calculations, and the description of the reference solution has been (hopefully) clarified. I am very pleased with the present description of the benchmark problem and believe that the results of the analysis will have an important impact on the LMFBR design process.

In carrying out the calculations for the reference benchmark problem, the LCCEWG members are asked to review the material in Section 3.2 of the attached specifications. In order to proceed with the calculations, an eight-group cross section file will be forwarded by GE-ARSD in about one month. Most of the LCCEWG participants will also be asked to perform certain alternate calculations (e.g., fine mesh, hex versus regionwise burnup calculations, extended burnup chain, odd-even lumped fission product representation, 3-D flux synthesis calculations, etc.). However, individual laboratories will be contacted to determine specific interests and the assignment of alternate calculations will be specifically spelled out in a separate letter.

At the April, 1981 meeting of the LCCEWG it was agreed that the "preliminary" results of the Benchmark Four analysis would be reported by January 1, 1982. However, this schedule now seems to be somewhat tight. Let me instead ask that the preliminary results be reported by March 15, 1982. If you have any questions concerning the schedule, or the reference calculations please give me a call. Thank you for your cooperation. I look forward to seeing you at an LCCEWG meeting sometime in 1982.

Sincerely yours,

C.L. Cowan

Attachment

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SPECIFICATIONS FOR THE LARGE CORE CODE
EVALUATION WORKING GROUP BENCHMARK PROBLEM FOUR

C.L. Cowan and R. Protsik

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I. INTRODUCTION AND REVIEW

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.^(1,2) 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.⁽³⁾ 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 has been 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 will be helpful to the designer in choosing from a wide variety of design tools and methods for applications varying from scoping studies to a final design verification. The proposed benchmark problem was specified as an extension of LCCEWG Benchmark Problem Three (Reference 3) to three-dimensions, and the incorporation of detailed burnup calculations for the first two irradiation cycles. Because of the complexity of the three-dimensional burnup problem, a "rigorous" benchmark solution will be sought to provide a base for testing the less accurate design approaches including the two-dimensional modelling of the three-dimensional geometry, the utilization of three-dimensional synthesis techniques, and the utilization of a variety of procedures for approximating the change in the neutron flux spectrum and shape throughout the irradiation cycle due to the fuel redistribution during burnup and due to changes in the control position. The proposed benchmark analysis will also consider the use of different gamma transport approximations in computing the energy deposition rates, and the incorporation of abbreviated fuel burnup chains and lumped fission product representations.

II SPECIFICATIONS FOR LCCEWG BENCHMARK PROBLEM FOUR

2.1 Computational Model

The specifications for LCCEWG Benchmark Problem Four were (as for LCCEWG Benchmark Problem Three) based upon the Westinghouse - ARD contribution to the Proliferation Resistant Preconceptual Core Design Study.⁽⁴⁾ The driver fuel, blanket fuel, shield and control assembly layout for the heterogeneous reactor system is shown in Figure 1. It is noted that this design layout provides a 30 degree azimuthal symmetry for modelling the core. The lattice pitch shown in Figure 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 R-Z format in Figure 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 1 and 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 problem. It is noted that there are perhaps better approaches for modelling the control channels in the R-Z geometry. However, the layout shown in Figure 2 is recommended to provide consistency in the results of the R-Z calculations. A summary of all the design data specifications for Benchmark Problem Four is included in Appendix A to this report.

All primary and secondary control rods, with the exception of the control rods labelled C in Figure 1, are 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 is also assumed that the withdrawn rods do not extend into the upper fission gas plenum.

To further simplify the computational requirements for the benchmark problem, the positions of the burnup control rods (i.e., control rods labelled C in Figure 1) are specified at the beginning and end of cycles 1 and 2 as follows:

FIGURE 1 - SIXTY DEGREE SECTOR OF CORE LAYOUT FOR BENCHMARK PROBLEM FOUR (ASSEMBLIES GIVEN FOR FULL CORE LAYOUT)

FIGURE 2 CORE DIAGRAM IN R-Z GEOMETRY FOR BENCHMARK PROBLEM FOUR (NOT TO SCALE)

	<u>Beginning of Cycle 1</u>	<u>End of Cycle 1</u>	<u>Beginning of Cycle 2</u>	<u>End of 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 R-Z model are represented by the third control annulus (i.e., third control ring from the center of the core), as shown in Figure 2. The control insertion depths for the R-Z model are assumed to be the same as those for the 3D configuration.

The material atom densities for the startup core are listed in Table 1. These isotopic compositions and the reactor geometry are given for the cold shutdown system. However, it will be assumed that the reactor dimensions and material compositions do not change in going from the cold shutdown to the hot operating conditions.

2.2 Fuel Management Plan

The fuel management plan for Benchmark Problem Four is based upon a reactor power of 2740 MW_{th}, and a refueling interval of 1.5 years. Fuel burnup calculations are to be carried out for the first two irradiation cycles of the startup core in accordance with the fuel management scheme shown in Figure 3. It is noted that the reference fuel management scheme differs from the scheme given in Reference 4 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.

TABLE I
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>	<u>Na Channel[†]</u>	<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	1.3980-3	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

* Control assembly inserted

† Control assembly withdrawn

†† per cent as 1.9168×10^{-5}

FIGURE 3 REFUELING SCHEME FOR BENCHMARK
PROBLEM FOUR

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) it is recommended that all burnup calculations be performed on the basis of a refueling interval of 383.5 effective full power days (EFPD). Thus, the reactor is assumed to operate continuously at full power (i.e., with no down time for refueling, inspection, maintenance, etc).

Special care should be taken in predicting the local energy deposition rates. It is recommended for the proposed benchmark problem that the energy deposition include the in-situ contributions from fission and neutron capture. Other approaches for predicting the heating rates are acceptable but must be described in the summary write up. Recommended values of the in-situ energy conversion factors for the major fuel, structure and coolant isotopes are given in Table 2 in Section 2.3.

It is 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 provides 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 is not considered a critical step in the specifications of the benchmark problem. Any number of other fuel management approaches would probably serve equally well. The principal objective in performing burnup calculations for two irradiation cycles is to assess the methods which are 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 will be provided for the principal fuel isotopes and for many "non-fuel" actinides so that the fuel burnup calculations may be performed in accordance with the linked burnup chains shown in Figure 4. Several of the LCCEWG participants will be asked to carry out the extended burnup calculations. However, the extent of these calculations will be determined by the individual laboratories. Recommended decay constants for the important actinides are also listed in Figure 4. It is noted that most of the long lived alpha decay constants have been omitted from this figure.

FIGURE 4 EXTENDED FUEL BURNUP CHAIN
FOR BENCHMARK PROBLEM 4

The majority of the benchmark burnup calculations will be carried out using an abbreviated version of the chain shown in Figure 4. It is recommended that, at a minimum, the burnup calculations be performed using the following isotopes:

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

The decay of Pu-241 should also be considered in all fuel cycle calculations.

It is proposed that several different representations for the lumped fission products be tested as part of the Benchmark Four analysis. The different representations include:

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.

The above options provide a simple treatment of the fission product representation, 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 is not expected to have a significant impact on the Benchmark Four analysis.

Preliminary testing of the odd-even approach indicates that it will be satisfactory for a broad range of design applications. However, one of the objectives of the benchmark analysis will be 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 will be provided by General Electric in the Computer Code Coordinating Committee (CCCC) ISOTXS format.⁽⁵⁾ The cross section library will have an 8-group energy structure and contain transport-corrected, P_0 microscopic data based upon a "heterogeneous" modelling

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) will be provided for each of the driver fuel and blanket fuel regions indicated in Figures 1 and 2. However, many of the non-fuel and the less important fuel isotopes will be 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 will be assumed to be 1500°, 1500°, 900°K and 900°K respectively.

A listing of all the materials to be provided for Benchmark Problem Four is given in Table 2. For convenience the recommended energy conversion factors for those materials are also included.

2.4 Mesh Spacing and Region Description

The linked burnup and flux solution calculations are somewhat sensitive to the choice of the region and mesh descriptions for the computational model. Thus, it is recommended that the reactor models shown in Figures 1 and 2 be subdivided into regions and mesh intervals as follows:

<u>X-Y 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
<u>Z Dimension</u>	Radial Blankets	6 triangles/hex
	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

TABLE 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
C059	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

R Dimension (R-Z Calculations Only)

<u>Region</u>	<u>Radius</u>	<u>Mesh Intervals</u>
Blanket	37.37	6
Driver	63.59	8
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 will be 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 has been inserted between 109.46 and 111.46 cm in the R-Z problem. This driver region was inserted to ensure that the R-Z representation of the radial power would be relatively flat, while maintaining the correct volume of the driver fuel assemblies. This special region should be treated as part of the outer driver fuel region*.

*The extra driver region 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 should be combined.

III. BENCHMARK CALCULATIONS AND REPORTING OF RESULTS

3.1 Design Parameters of Interest

The material to be reported for Benchmark Problem Four includes the results of calculations which describe the power distribution, burnup distribution, fuel inventory and neutron balances. A summary of all the quantities reported for the first three benchmarks and those proposed for the fourth benchmark are given in Table 3. The detailed reporting forms for Benchmark Problem Four are found in Appendix B.

The benchmark calculations will be carried out for the first two irradiation cycles of the startup core, and the results will be 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 have been 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 will be performed using:
 - a. The mesh spacing and region descriptions from Section 2.4.
 - b. The eight energy group cross section file to be provided by G.E.
 - c. One lumped fission product to be provided.
2. Burnup - The reference burnup computations will be 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 will also be explicitly treated in the burnup calculations based upon the half-life given in Figure 4.
 - c. Two flux solutions for each irradiation cycle. Burnup calculations will 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 will be carried out based upon the flux solution at the middle-of-cycle 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 will be utilized for the second irradiation cycle.
3. Energy Deposition - The reference neutronics calculations will be normalized based upon the energy deposition for in-situ capture and fission as listed for the different materials (isotopes) in Table 2. Should the codes not distinguish between capture and fission or if it is not possible to specify the energy deposition by isotope, the participants are encouraged to improvise so that the specified deposition rates are consistent with those calculated using the data in Table 2.
4. Convergence - The recommended convergence criteria for the neutronics calculations will be 10^{-6} for K_{eff} and 10^{-5} for the pointwise fluxes.
5. Linked R-Z and X-Y Calculations - Those participants who will perform the linked R-Z and X-Y calculations should use region-wise and groupwise axial bucklings to establish the model links.

The results of the Benchmark Problem Three analysis have indicated several areas which contribute to differences. Thus, it is recommended that the neutronics codes be modified, as necessary, to eliminate the (n,2n) induced discrepancy. In addition, it is

recommended that those codes which require extra regions at the outer boundaries utilize a low density filler. The recommended filler is radial shield material with one hundredth (0.01) of the indicated density.

3.3 Alternate Computations

The LCCEWG participants will be asked to carry out specific assignments, subject to laboratory interest and concurrence. These assignments will include the reference solution plus several alternate computations as follows:

1. Direct Three-Dimensional Calculations - Alternate Control Treatment
2. Direct Three-Dimensional Calculations - Burnup by Hex
3. Three-Dimensional Synthesis Calculations
4. Two-Dimensional Calculations (R-Z Geometry)
5. Two-Dimensional Calculations (X-Y Geometry)
6. Linked X-Y and R-Z calculations
7. Fine Mesh Calculations in 2D and 3D
8. Fine Group Calculations
9. Assessment of Extended Burnup Chain and Lumped Fission Products
10. Two-Dimensional Transport Calculations Including Gamma Transport Calculations
11. Assessment of Change in Cross Section Self-Shielding During the Burnup Cycle
12. Assessments of Different Approaches for Approximating the Changes in the Flux Spectrum Due to Changes in the Fuel Distribution and Control Position During Burnup.
13. Breakdown of Reactivity Change During Burnup Due to Fission Product Buildup, Actinide Buildup, Blanket Fissile Buildup, and Driver Fuel Redistribution.

TABLE 3
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 , n	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.

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1. E. Kujawski and H. S. Bailey, Nucl. Sci. Eng. 64, 90 (1977)
2. R. M. Wu (Hanford Engineering Development Laboratory), et. al., The Large Core Code Evaluation Working Group Benchmark Analysis of a Heterogeneous Fast Reactor, Department of Energy, to be published.
3. M. B. Parker, Hanford Engineering Development Laboratory, Letter to Large Core Code Evaluation Working Group Distribution, "Large Core Code Evaluation Working Group (LCCEWG) Third Benchmark Problem," 8050990, March 12, 1980.
4. Westinghouse Electric Corporation - Advanced Reactors Division, "Preconceptual Study of Proliferation Resistant Heterogeneous Oxide Fueled LMFBR Core, Final Report," August 1978.
5. B. M. Carmichael et. al., "Standard Interface Files and Procedures for Reactor Physics Codes, Version III," LA-5486-MS, Los Alamos Scientific Laboratory (1974).

APPENDIX A

DESIGN DATA SPECIFICATIONS FOR THE LARGE CORE CODE EVALUATION WORKING GROUP BENCHMARK PROBLEM FOUR

Benchmark Specifications

The specifications for the Large Core Code Evaluation Working Group (LCCEWG) Benchmark Problem Four were based upon the Westinghouse - ARD contribution to the Proliferation Resistant Preconceptual Core Design Study.^(A.1) The only change from the reference design study was the modification of the fuel management plan to provide a more stringent test of the fast reactor design and analysis procedures. A partial listing of the Benchmark Problem Four specifications is given in Table A-I.

References

- A.1 Westinghouse Electric Corporation - Advanced Reactors Division, "Preconceptual Study of Proliferation Resistant Heterogeneous Oxide Fueled LMFBR Core, Final Report," August 1978.

TABLE A.1
DESIGN DATA SPECIFICATIONS FOR THE LARGE CORE CODE
EVALUATION WORKING GROUP BENCHMARK PROBLEM FOUR

1.0 CORE AND REACTOR DATA

1.1 Power Information

1.1.1 Plant Thermal Power, MWt 2740

1.2 Geometric Information

1.2.1 Core Height, cm 121.92

1.2.2 Axial Blanket Height, cm 35.56

1.2.3 Radial Blanket Height, cm 193.04

1.2.4 Lower Axial Shield Height, cm 60.96

1.2.5 Fission Gas Plenum Height, cm 134.62

1.3 Number of Assemblies

1.3.1 Driver Fuel 270

1.3.2 Inner Blanket 121

1.3.3 Radial Blanket 138

1.3.4 Control 30

1.3.5 Radial Shield 186

1.4 Fuel Management

1.4.1 Refueling Interval, Full Power Days 383.5

1.4.2 Fraction of Assemblies Replaced at
Each Refueling (Average)

1.4.2.1 Driver Fuel 0.5

1.4.2.2 Inner Blanket 0.5

1.4.2.3 Radial Blanket 0.25

2.0 DRIVER FUEL ASSEMBLY DATA

2.1	Pins per Assembly	271
2.2	Pin Pitch-to-Diameter Ratio	1.15
2.3	Spacer Description	
2.3.1	Wire Wrap Diameter (or grid spacer thickness & height), mm	1.18
2.3.2	Spacer Pitch cm	30.48
2.3.3	Edge Ratio	1.0
2.4	Overall Assembly Length, cm	465
2.5	Lattice Pitch, cm	16.33
2.6	Duct Inside Flat-to-Flat, cm	15.22
2.7	Duct Wall Thickness, mm	3.81
2.9	Interduct Gap, mm	0.76
2.10	Duct Material	
2.10.1	Material Type	SS316

3.0 FUEL PIN DATA

3.1	Fuel Parameters	
3.1.1	Fuel Form (powder or pellet)	Pellet
3.1.1.1	Pellet Diameter, mm	7.06
3.1.1.2	Pellet Density, g/cm ³	10.45
3.2	Cladding Parameters	
3.2.1	Cladding Outside Diameter, mm	7.87
3.2.2	Cladding Wall Thickness, mm	0.33
3.2.3	Diametral Gap, mm	0.152
3.2.4	Cladding Material	SS316
3.3	Equivalent Plenum Volume, cc	
3.3.1	Top Plenum	55.8
3.3.2	Bottom Plenum	0.
3.4	Bond Type (sodium or helium)	He

4.0	BLANKET ASSEMBLY DATA	
4.1	Pins per Assembly	127
4.2	Pin Pitch-to-Diameter Ratio	1.07
4.3	Spacer Description	
4.3.1	Wire Wrap Diameter (or grid spacer thickness & height), mm	0.874
4.3.2	Spacer Pitch, cm	10.16
4.4	Overall Assembly Length, cm	465
4.5	Lattice Pitch, cm	16.33
4.6	Duct Inside Flat-to-Flat, cm	15.34
4.7	Duct Wall Thickness, mm	3.048
4.10	Duct Material	SS316
5.0	BLANKET PIN DATA	
5.1	Fuel Parameters	
5.1.1	Fuel Form (powder or pellet)	Pellet
5.1.1.1	Pellet Diameter, mm	11.59
5.1.1.2	Pellet Density, g/cm ³	10.45
5.2	Cladding Parameters	
5.2.1	Cladding Outside Diameter mm	12.48
5.2.2	Cladding Wall Thickness, mm	0.381
5.2.3	Diametral Gap, mm	0.127
5.2.4	Cladding Material	SS316
5.3	Equivalent Plenum Volume, cc	
5.3.1	Top Plenum	145
5.3.2	Bottom Plenum	0.
5.4	Bond Type (sodium or helium)	He

6.0 DRIVER FUEL INITIAL CELL ATOM DENSITIES

6.1 Smeared Fuel Pellet (atoms/barn·cm)

6.1.1	U-235	4.0006-5
6.1.2	U-238	1.8140-2
6.1.3	Pu-239	2.8366-3
6.1.4	Pu-240	8.5132-4
6.1.5	Pu-241	4.2561-4
6.1.6	Pu-242	1.0119-4
6.1.7	FP	0.0
6.1.8	O	4.4790-2

6.2 Structure (atoms/barn·cm)

6.2.1	Cr	1.4514-2
6.2.2	Mn	1.3738-3
6.2.3	Fe	4.9985-2
6.2.4	Ni	9.9165-3
6.2.5	Mo	1.1237-3

6.3 Sodium (atoms/barn·cm)

6.3.1	Na	2.5129-2
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7.0 AXIAL BLANKET EXTENSION OF DRIVER FUEL INITIAL CELL ATOM DENSITIES

7.1 Smeared Fuel Pellet (atoms/barn·cm)

7.1.1	U-235	4.9313-5
7.1.2	U-238	2.2356-2
7.1.3	O	4.4790-2

7.2 Structure (atoms/barn·cm)

See 6.2

7.3 Sodium (atoms/barn·cm)

See 6.3

8.0 INTERNAL AND RADIAL BLANKET INITIAL CELL ATOM DENSITIES

8.1 Smeared Fuel Pellet (atoms/barn·cm)

8.1.1	U-235	5.0425-5
8.1.2	U-238	2.2860-2
8.1.3	Pu-239	0.0
8.1.4	Pu-240	0.0

8.1.5	Pu-241	0.0
8.1.6	Pu-242	0.0
8.1.7	FP	0.0
8.1.8	0	4.5820-2
8.2	Structure (atoms/barn·cm)	
8.2.1	Cr	1.5380-2
8.2.2	Mn	1.4585-3
8.2.3	Fe	5.3086-2
8.2.4	Ni	1.0531-2
8.2.5	Mo	1.1933-3
8.3	Sodium (atoms/barn·cm)	
8.3.1	Na	2.4719-2

9.0 AVERAGE CELL TEMPERATURES

9.1	Driver Fuel (°K)	
9.1.1	Fuel	1500
9.1.2	Structure	700
9.1.3	Sodium	700
9.2	Inner Blanket (°K)	
9.2.1	Fuel	1500
9.2.2	Structure	700
9.2.3	Sodium	700
9.3	Radial Blanket (°K)	
9.3.1	Fuel	900
9.3.2	Structure	700
9.3.3	Sodium	700
9.4	Axial Blanket Extension of Driver Fuel (°K)	
9.4.1	Fuel	900
9.4.2	Structure	700
9.4.3	Sodium	700

APPENDIX B

REPORTING OF RESULTS FOR BENCHMARK PROBLEM FOUR

The results of calculations for Benchmark Problem Four are to be reported in the format given in this Appendix. In describing the positions of key parameters (e.g., peak power) the participants are requested to be as explicit as possible. For convenience a position indicator guide is shown in Figure B.1.

All results will be 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

FIGURE B.1 - SIXTY DEGREE SECTOR OF CORE LAYOUT FOR BENCHMARK PROBLEM FOUR SHOWING DRIVER, BLANKET AND SHIELD ASSEMBLY IDENTIFICATION.

GENERAL COMMENTS

- A. Codes Used
- B. Convergence Criteria
- C. Method of Burnup Calculations (continue on following pages)
- D. Analysis Procedures

Geometry

- 1. HEX-Z Direct
- 2. HEX-Z Synthesis
- 3. HEX X-Y
- 4. R-Z
- 5. Linked R-Z and HEX X-Y
- 6. Other (describe)

Cross Sections

- 1. GE-8 Group
- 2. Other (describe)

Fuel Management Plan

- 1. Recommended two cycle reload plan
- 2. Other (describe)

Fuel Burnup Chain

- 1. Abbreviated burnup chain with uranium and plutonium isotopes only (including Pu-241 decay)
- 2. Extended burnup chain
- 3. Other (describe)

Lumped Fission Products

1. Recommended - 2 Fission Product Pairs (odd, even)
2. Recommended - 1 Fission Product Pair
3. Other (describe)

Energy Deposition

1. In-Situ fission plus capture gamma's
2. In-Situ fission only
3. Other (describe)

Treatment of Control Rods

1. Control positioned at the recommended BOC location during the first half of the irradiation cycle, and at the recommended EOC location during the last half of the irradiation cycle
2. Positioned as recommended at the beginning-of-cycle and left unchanged during the burnup calculations
3. All control out
4. Other (describe)

Calculation of Control Rod Worth

1. Direct 3-D calculation
2. Synthesis 3-D calculations
3. R-Z calculations
4. R-Z calculations (Modified B_{10} Density)
5. X-Y HEX
6. Other (describe)

Core Layout

1. Recommended
2. Other (describe)

Mesh Spacings

1. Recommended
2. Other (describe)

Flux Solutions During Irradiation Cycle

1. Recommended
2. Other (describe)

GENERAL COMMENTS (CONTINUED)

ADDITIONAL DESCRIPTION OF BENCHMARK SPECIFICATIONS

RESULTS FOR CASE _____

A. General

1. k_{eff}	X.XXXXX	
2. Control Rod Worth (Δk) ^a	X.XXXXX	
3. Sodium Void Worth (Δk) ^a	X.XXXXX	
4. Breeding Gain (kg of fissile plutonium) ^b	XXX.X	
5. Peak Power (kw/cm ³)-Position		
Driver Fuel	X.XXEXX	- Position
Inner Blankets	X.XXEXX	- Position
Radial Blankets	X.XXEXX	- Position
6. Driver Fuel Peak-to-Average Power	X.XXEXX	
7. Driver Fuel Axial Peak-to-Average Power (at position of peak power)	X.XXEXX	

a. Optional calculations

b. Required for Cases 2 and 4 only

RESULTS FOR CASE _____

B. Power

<u>ZONE^a</u>	<u>POWER FRACTION</u>			<u>PEAK POWER DENSITY (KW/CM³)</u>
	<u>Fission</u>	<u>Capture</u>	<u>Total</u>	
D-I	X.XXXX	X.XXXX	X.XXXX	X.XXXEXX ↓
D-II	X.XXXX	X.XXXX	X.XXXX	
D-III	X.XXXX	X.XXXX	X.XXXX	
IB-I	X.XXXX	X.XXXX	X.XXXX	
IB-II	X.XXXX	X.XXXX	X.XXXX	
IB-III	X.XXXX	X.XXXX	X.XXXX	
RB	X.XXXX	X.XXXX	X.XXXX	
AB	X.XXXX	X.XXXX	X.XXXX	
S	X.XXXX	X.XXXX	X.XXXX	
C	<u>X.XXXX</u>	<u>X.XXXX</u>	<u>X.XXXX</u>	
			1.0	

- a) A partial description of the zone designators is given for Figure B.1.
Other terms are:

AB - Axial blanket extensions to the driver fuel
S - All axial and radial shields
C - All control channels

RESULTS FOR CASE _____

C. Burnup

<u>ZONE^b</u>	<u>AVERAGE BURNUP (MWD/MT)</u>	<u>PEAK BURNUP (MWD/MT)^c</u>
D-I	X.XXXEXX	X.XXXEXX
D-II	↓	↓
D-III		
IB-I		
IB-II		
IB-III		
RB		
AB		

a) Required for Cases 2 and 4 only

b) A partial description of the zone designations is given for Figure B.1.
Other terms are:

AB-Axial blanket extensions to the driver fuel

c) Optional calculation-if this calculation is performed please describe procedure.

RESULTS FOR CASE _____

D. Fuel Inventories

	<u>Inventory (kg)</u>			
	<u>Driver</u>	<u>Inner</u>	<u>Radial</u>	<u>Axial</u>
	<u>Fuel</u>	<u>Blankets</u>	<u>Blankets</u>	<u>Blanket Extension</u>
U-234	X.XXXEXX	X.XXXEXX	X.XXXEXX	X.XXXEXX
U-235	↓	↓	↓	↓
U-236	↓	↓	↓	↓
U-237	↓	↓	↓	↓
U-238	↓	↓	↓	↓
U-239	↓	↓	↓	↓
N _p -237	↓	↓	↓	↓
N _p -238	↓	↓	↓	↓
N _p -239	↓	↓	↓	↓
P _u -238	↓	↓	↓	↓
P _u -239	↓	↓	↓	↓
P _u -240	↓	↓	↓	↓
P _u -241	↓	↓	↓	↓
P _u -242	↓	↓	↓	↓
P _u -243	↓	↓	↓	↓
P _u -244	↓	↓	↓	↓
Am-241	↓	↓	↓	↓
Am-242m	↓	↓	↓	↓
Am-242	↓	↓	↓	↓
Am-243	↓	↓	↓	↓
Am-244	↓	↓	↓	↓
Cm-242	↓	↓	↓	↓
Cm-243	↓	↓	↓	↓
Cm-244	↓	↓	↓	↓

RESULTS FOR CASE _____

E. Neutron Balance for Reactor

Source

Fission	1.00000 n/sec	
n,2n	X.XXXXX	
total source		X.XXXXX

Absorptions

Captures

Fissions

U-235	X.XXXXX	X.XXXXX
U-236		
U-238		
Pu-238		
Pu-239		
Pu-240		
Pu-241		
Pu-242		
Other Actinides		
Fission Products		
Structure		
Coolant		
Total		

Total Absorptions

Leakage

Axial	X.XXXXX	
Radial	X.XXXXX	
Total		X.XXXXX
Total Losses		X.XXXXX

All values are normalized to the unit fission source

RESULTS FOR CASE _____

F. Reactivity Effects^a

Global reactivity loss per cycle^b

$$\frac{(k_{eff}(2) - k_{eff}(1))}{k_{eff}(1)} \quad X.XXXXX$$

Reactivity gain due to Pu build-up in the blankets^b

$$\frac{(k_{eff}(3) - k_{eff}(1))}{k_{eff}(1)} \quad X.XXXXX$$

Reactivity loss due to fission product build-up^b

$$\frac{(k_{eff}(4) - k_{eff}(1))}{k_{eff}(1)} \quad X.XXXXX$$

Reactivity loss due to burnup of Pu and U Isotopes^b

$$\frac{(k_{eff}(5) - k_{eff}(1))}{k_{eff}(1)} \quad X.XXXXX$$

Reactivity change due to build-up of heavy actinides^b

$$\frac{(k_{eff}(6) - k_{eff}(1))}{k_{eff}(1)} \quad X.XXXXX$$

a) Assigned task for Case 2 only.

b) The k_{eff} terms are defined as follows:

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

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

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

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

$k_{eff}(5)$ = k_{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

$k_{eff}(6)$ = k_{eff} corresponding to the beginning-of-cycle compositions but including the end-of-cycle actinide densities for other than the plutonium and uranium isotopes.