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Proceedings of the
NEACRP/IAEA Specialists Meeting

on the
INTERNATIONAL COMPARISON CALCULATION OF A
LARGE SODIUM-COOLED FAST BREEDER REACTOR
at Argonne National Laboratory
on February 7-9, 1978

by

MASTER

L. G. LeSage, R. D. McKnight, D. C. Wade,
K. E. Freese, and P. J. Collins

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Printed in the United States of America
Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

NTIS price codes
Printed copy: A13
Microfiche copy: A01

101. 1785

Distribution Category:
LMFBR—Physics (UC-79d)

ANL-80-78
NEA-CRP-L-243

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

Master

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Applied Physics Division

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ABSTRACT

The results of an international comparison calculation of a large (1250 MWe) LMFBR benchmark model are presented and discussed. Eight reactor configurations were calculated. Parameters included with the comparison were: eigenvalue, k_{∞} , neutron balance data, breeding reaction rate ratios, reactivity worths, central control rod worth, regional sodium void reactivity, core Doppler and effective delayed neutron fraction. Ten countries participated in the comparison, and sixteen solutions were contributed. The discussion focuses on the variation in parameter values, the degree of consistency among the various parameters and solutions, and the identification of unexpected results. The results are displayed and discussed both by individual participant and by groupings of participants (e.g., results from adjusted data sets versus non-adjusted data sets). Unexpected large variations among results were observed for radial reaction rate and worth distributions and for the central control rod worth.

I. INTRODUCTION

A. Background and Objectives

Results of an international comparison calculation of a large (1250 MWe) LMFBR benchmark model were presented and discussed at a specialist's meeting at Argonne National Laboratory in February 1978. The meeting was sponsored by the Nuclear Energy Agency Committee on Reactor Physics (NEACRP) in collaboration with IAEA. The purpose of the analytical exercise was to evaluate and document the agreement and differences in the calculation of key LMFBR physics and safety parameters as a function of the different data sets and processing codes used by the participating countries.

This comparison calculation was the first such international comparison exercise since the so-called "Baker Model" comparison¹ of 1970 which focused on breeding and neutron balance. It also was the first comparison for a large "commercial sized" LMFBR system, the first comprehensive comparison between the current adjusted data sets (e.g., FGL5 and CARNAVAL-III and -IV) and the unadjusted sets (e.g., ENDF/B-IV), and is the most comprehensive of such comparisons -- including a number of parameters not included in previous comparisons (e.g., control rods and certain safety parameters).

Plans for the comparison calculation were initiated at the annual NEACRP meeting in 1975 in Bologna. ANL prepared the problem specifications which were reviewed by Winfrith prior to distribution to the NEACRP and IAEA membership. Solutions were submitted to ANL in the fall of 1977. These were compiled into figures and tables and copies were sent to each of the participants in January 1978. The specialist's meeting was held February 7-9, 1978, at Argonne National Laboratory.

B. Participants and Nuclear Data Sets Used

Ten countries participated in the comparison, and sixteen solutions were contributed. A list of the participating countries, meeting attendees, solution authors, and data sets used in the solutions is included in Table I. The success of the comparison calculation was due in large part to the excellent cooperation of the solution authors in following the problem specifications and meeting the schedules. This report should be considered as a compilation of the work of the individuals listed in Table I.

TABLE I. List of Solutions and Participants

Solution Label	Country	Data Set	Adjusted	Solution Authors and Meeting Attendees
ANL	USA (ANL)	ENDF/B-IV	No	L. G. LeSage, D. C. Wade, and R. D. McKnight
BELGIUM	Belgium	KEDAK-2	No	G. Minsart, R. deWouters
CADARACHE-1	France	CARNAVAL-III	Yes	Y. Y. Bouget, P. Hammer
CADARACHE-2	France	CARNAVAL-IV	Yes	
CNEN	Italy	ENDF/B-IV	No	R. Conversano, A. Livrieri, F. Raponi, D. Rinaldis, and S. Zero
EIR-1	Switzerland	ENDF/B-IV	No	P. Wydler, J. Hadermann, W. Heer, and H. U. Wenger
EIR-2	Switzerland	ENDF/B-III	No	
HEDL	USA (HEDL)	ENDF/B-IV	No	D. B. Marr, R. W. Hardie
JAERI-1	Japan	JENDL	No	H. Yoshida, H. Nishimura, S. Iijima, T. Osugi, Y. Kikuchi, T. Nakagawa, K. Ikawa, and M. Hirata
JAERI-2	Japan	JAERI-FAST-2	Yes	M. Nakagawa, A. Zukeran
JAERI-3	Japan	GJAERI-FAST-2(25)	Yes	A. Zukeran, M. Nakagawa
KARLSRUHE-1	Germany	KEDAK-3	No	C. Broeders
KARLSRUHE-2	Germany	KFKINR	Yes	
SWEDEN	Sweden	ENDF/B-III	No	Klas Jirflow
UKAEA	England	FGL-5	Yes	J. L. Rowlands, C. J. Dean, C. G. Campbell
USSR	USSR	BNAB-70	No	M. Troyanov

C. Model Description

The benchmark reactor specification was based on a 3260 MWt conventional mixed oxide design with a 0.300 inch pin size developed at ANL in 1975.* The benchmark model was set up for R,Z modeling with specified homogeneous compositions for each region of the reactor. The model contained no control rods or control rod positions. The core height was 40 inches (101.6 cm) and the radii of the inner and outer core regions were 136.85 cm and 176.53 cm, respectively, giving a total core volume of 9950 liters. The core volume fractions were 41% for fuel, 38% for total sodium and 21% for total structural.

Eight configurations of the benchmark reactor were specified. These included:

1. Reference
2. Na-Voided Inner Core
3. Na-Voided Inner Core, Outer Core and Axial Blanket
4. Reference with a Central Na-Filled Control Rod Position
5. Reference with a Central Boron Control Rod
6. Na-Voided Inner Core, Outer Core and Axial Blanket with a Central Boron Control Rod
7. Reference with Hot Fuel
8. Na-Voided Inner Core, Outer Core and Axial Blanket with Hot Fuel

The parameters requested by the problem specification included:

- Eigenvalues for the eight configurations

*It may be noted that the reactor design model for these benchmark calculations has significantly lower enrichments (~10% and 13% fissile Pu in the two core zones) than present commercial conventional core designs (~14% and 18% for SUPER-PHENIX 1) which are based on smaller fuel pins and lower fuel inventories and fuel volume fractions. Integral experiments in a higher enrichment range (from ~12% to 30%) have been used to create and validate the current adjusted libraries. In this respect, integral experiments corresponding to this lower enrichment range (<12%) could provide useful complementary information.

- k_{∞} for the inner core composition
- Neutron balance data
- Breeding and conversion ratios (global and regional)
- Central flux and adjoint spectra, reaction rate ratios and reactivity worths
- Central control rod worths (with and without sodium)
- Fission rate and reactivity worth distributions axially and radially
- Effective delayed neutron fraction
- Regional sodium void reactivity
- Isothermal core fuel Doppler reactivities

The problem specifications for these comparison calculations are given in Appendix A of this report.

D. Summary of Results and Conclusions

The sixteen solutions (see Table I) which make up this exercise represent a large amount of data which has been processed and compiled into many tables and figures. The complete solutions are included in Appendices B and C (tabulated and plotted results, respectively). A survey of these results will be made in the following sections.

Some of the more important conclusions resulting from this exercise are summarized below.

- There are large differences in the radial fission rate and worth distributions among the solutions. The radial distributions appear to be sensitive to small variations in the inner core to outer core k_{∞} increment produced by the specified inner core to outer core enrichment change. Participants whose solutions disagree on the 1250 MWe size comparison calculation have each obtained good agreement with experiments on 300 MWe size critical experiments. Thus, there is a

strong interest in the results of larger critical experiment reaction rate distribution measurements.

- There are very large differences (over 45%) in the calculated central control rod worth. These differences are not well-correlated with the central boron worths. They are, however, related to (but not fully explained by) the differences in radial distributions. As above, a strong interest exists in control rod measurements on larger core criticals.
- Generally, strong correlations were observed among the parameters k_{eff} , k_{∞} , breeding ratio, $\langle \sigma_c^{28} \rangle$, and $^{28}\text{C}/^{49}\text{f}$ (e.g., solutions having high $\langle \sigma_c^{28} \rangle$ and $^{28}\text{C}/^{49}\text{f}$, displayed high breeding ratio and low k_{eff}). On the other hand, the correlations among the central worths, control rod worth, leakage, and safety parameters were found to be weak.
- There are large variations in the structural cross sections and capture rates. These appear to be due as much to processing as to basic data files.
- Significant differences are observed between the one-group effective cross sections from the adjusted data sets compared to those from the non-adjusted data sets.
- The scatter in the values of the adjusted cross sections is almost as great as the scatter in the values of the non-adjusted cross sections. The scatter due to different processing of the same data set is as large as the scatter between data sets.
- The scatter in the integral parameters calculated from adjusted data sets is less than that for parameters calculated from non-adjusted sets for k_{eff} , k_{∞} , breeding ratio and $^{28}\text{C}/^{49}\text{f}$. On the other hand, the

scatter in $^{28}\text{f}/^{49}\text{f}$, central worths, and the safety parameters is as large among the users of adjusted data sets as it is among users of non-adjusted data sets.

- The fact that the users of ENDF data have traditionally observed a central worth discrepancy of 15 to 20% in fissile reactivity worths while users of FGL5 and CARNAVAL-III and -IV data have not observed this discrepancy appears inconsistent with the results of this comparison exercise.
- The variation in the results observed in the NEACRP comparison exercise is a measure of the differences in the nuclear data files and/or the data processing codes in the participating countries. It is related to, but not a measure of, the uncertainty in the calculation of LMFBR parameters. The actual design uncertainty in most of the parameters is less than is indicated by the spread in the calculated results because reactor designers routinely apply bias factors (usually derived from critical experiments) to their calculated results. The bias factors, which would have the effect of reducing the spread among the results, have not been applied in this exercise.
- The NEACRP comparison exercise has clearly pointed out an uncertainty concerning the ability to correctly compute the radial power distribution in large fast reactors. This uncertainty impacts most reactor performance and safety parameters.

II. REVIEW OF BASIC DATA SETS AND PROCESSING

A. Comparison of Basic Data Sets

1. Data Sets; Adjusted and Non-Adjusted: Twelve basic data sets are represented. These are listed in Table I. Six of these sets are evaluated sets; six are adjusted sets. Two of the adjusted sets (JAERI-FAST-2 and GJAERI-FAST-2) are different number energy group versions of the same adjustment. One might expect that the adjusted data set results would be more closely clumped together than would the evaluated data set results. This comparison calculation provides a test of this hypothesis.

2. Alternate Processing of the Same Data Set: Information available about the basic data files and the processing used for each contributed solution is summarized in Table II. For two of the basic data files (ENDF/B-III and ENDF/B-IV) there are multiple solutions from different participants using different processing codes. A comparison of these solutions illustrates how much variability is introduced by processing alone, i.e., given the same basic nuclear data file. Therefore, it will often be useful to lump solutions into the following categories or subsets:

All solutions,
Adjusted solutions,
Non-adjusted solutions, and
ENDF/B-IV solutions.

3. Broad Group Energy Structures: Four basic energy group structures were used. These are compared in Table III and Fig. 1.

a. 27 Energy Group Structure

$$\Delta u = 0.5$$

$$E_{\max} = 10.000 \text{ MeV}$$

$$E_{\min} = 13.710 \text{ eV}$$

ANL, EIR-1, EIR-2, JAERI-1, UKAEA

$$\Delta u = 0.5, \text{ except } \Delta u^{12} = 0.47 \text{ and } \Delta u^{13} = 0.53$$

HEDL

b. 25 and 26 Group Bondarenko Structures

These are the same except for the thermal group.

$$E_{\max} = 10.57 \text{ MeV}$$

$$E_{\min} = 0.2164 \text{ eV} \quad 25 \text{ group}$$

thermal 26 group

25 group JAERI-2, JAERI-3

26 group Belgium, KARLSRUHE-1, KARLSRUHE-2, USSR

c. 25 Group CARNAVAL Structure

$$E_{\max} = 14.5 \text{ MeV}$$

$$E_{\min} = 0.0253 \text{ eV}$$

Cadarache-1, Cadarache-2, CNEN

d. 10 Energy Group Structure

$$E_{\max} = 10.00 \text{ MeV}$$

$$E_{\min} = 13.71 \text{ eV}$$

Sweden

The 10 energy group structure is a subset of the 27 energy group structure. The CARNAVAL and 27 energy group structures share common boundaries except for the top of groups 1, 24, and 25. The Bondarenko 26 group structure is the 25 group structure with a thermal group added. The Bondarenko group boundaries are not the same as the boundaries of the 27 group/CARNAVAL/10 group energy structures.

TABLE II. Summary of Basic Data Files and Cross Section Processing

Solution Label (Country)	Basic Data File		Processing			Delayed Neutron		Broad Group Structure			Broad Group Solution
	Name	Revisions	# groups in Fine Group Soln	Fine group σ generation	Spectrum Soln and Collapse	χ	# groups	E_{top} E_{bottom}	Comments		
ANL (USA)	ENDF/B-IV		1620	MC ² -2 . $\Delta u = 1/120$	MC ² -2 . P1 fundamental mode with B ² search to critical	ENDF/B IV	27	10 MeV 13.7 eV	As per specs	Diff. Theory ARC Edits ARC Pert ARC	
BELGIUM	KEDAK-2 (1973)		208	GRUCAL code . top 196 gr. sub- divided in 5 each for 980 gr. de- pendence of light and medium A materials . ABBN library method used to self shield heavy isotopes	MULCOS-DTF-4 . as per specs except - collapsed to 26 gr. set	ABBN spectrum for $\bar{\nu} = 2.8$ ($\bar{\nu} u^{25}$ fast fission)	26	10.57 MeV thermal	Bondarenko Structure . Mn lumped with Fe		
CADRACHE-1 (France)	CARNAVAL-III (Adjusted)		660 (25 gr. some isotopes)		HETAIRE . Collapse as per specs except - in blankets, the elastic scattering is weighted over "power plant" ave flux spectrum with B ² set to 0	T=1.044 MeV for ²³⁹ Pu T=0.965 MeV for ²³⁵ U	25	14.5 MeV thermal	"CARNAVAL Structure" $\Delta u = 0.5$ except; $\Delta u^1 = 1.37, \Delta u^{21} = 1.0$ $\Delta u^{22} = 1.5, \Delta u^{23,24} = 2.0$ $\Delta u^{25} = 2.795$		
CADRACHE-2 (France)	CARNAVAL-IV (Adjusted)						25	14.5 thermal	"CARNAVAL Structure"		
CHEN (Italy)	ENDF/B-IV				MC ² -1 . as per specs except - collapsed to 25 gr. CARNAVAL structure		25		"CARNAVAL Structure" . Separate σ 's generated for rods and control rod positions		
EIR-1 (Switzerland)	ENDF/B-IV		99 15 MeV to 0.414 eV	GAM library SUPERTOG calc. of resonance contribution to smooth background GCC4 generation of 99 gr. cross sections	Collision Prob. Code . as per specs	None	27	10 MeV 13.7 eV	As per specs	RSYST-DIFF2D RSYST-REAC RSYST-PPRT	
EIR-2 (Switzerland)	ENDF/B-III	to Jan '72	68 10 MeV to 0.414 eV	RIGEL, ETOX, MERMC2 . 1/E weighting of σ 's for each of the 68 fine groups	MC ² -1 as per specs except ϕ weighting of σ_{tr}	Maxwellian T=1.40 MeV	None 27	10 MeV 13.7 eV	As per specs		
HEDL	ENDF/B-IV		42	ETOX	IDX	Maxwellian T=1.41 MeV	ENDF/B -IV	27	10 MeV 13.7 eV	$\Delta u = 0.5$ except $\Delta u^{12} = 0.47$ $\Delta u^{13} = 0.53$	Diff. Theory - 2DB
JAERI-1 (Japan)	JENDL-1				MC ² -1 as per specs	(E) = $-\frac{E}{T}$ $\sqrt{\frac{E}{C}}$ * e T=1.393 MeV	(DNDPC) code See summary for data	27	10 MeV 13.7 eV	As per specs	

TABLE II. (Cont'd.)

Solution Label (Country)	Basic Data File		Processing				Delayed Neutron	Broad Group Structure			Broad Group Solution
	Name	Revisions	# groups in Fine Group Soln	Fine Group σ generation	Spectrum Soln and Collapse	χ		# groups	E_{top} E_{bottom}	Comments	
JAERI-2 (Japan)	JAERI-2 + (Adjustment) (of heavy) (isotopes by LSQ)		$\Delta u = 0.0084$		ESELEM5 σ/E_{tr} weighted σ_{tr}	Maxwellian $T = \text{from}$ ENDF/B -IV	ENDF/B IV	25	10.57 MeV (thermal)	Bondarenko Structure Otherwise as per specs except: . refl. uses σ 's for axial blk.	
JAERI-3 (Japan)	GJAERI-Fast-2 (1976 version) (adjusted)		Adjusted 25 gr. set			Maxwellian $T = 1.40$ MeV for Pu^{239}	ENDF/B IV	25	10.57 MeV 0.21 eV	Bondarenko Structure Otherwise as per specs except: . refl. uses σ 's for refl. . B rod in Na Void uses cross section Set A	
KARLSRUHE-1 (Germany)	KEDAK-3 ($\text{Mn}, ^{10}\text{B}, ^{11}\text{B}$ are identical to ENDF/B-IV)		208	FIDAS/FITRA/MIGROS/MITRA REMINT/GRUMA (See summary) . top 196 gr. sub- divided into 5 each	KAPROS-BUCITO/SIGMNC As per specs Current weighted σ_{tr}	$\chi(E) =$ $0.48394 * e^{-E} / \sinh \sqrt{2E}$	See summary Tuttle χ	26	10.57 MeV (thermal)	Bondarenko Structure	KAPROS-DIXCON KAPROS-DXEVA KAPROS-DXPRT
KARLSRUHE-2 (Germany)	KFKINR		Adjusted 26 gr. set		KAPROS-BUCITO/SIGMNC As per specs Current weighted σ_{tr}	Maxwellian $T = 1.41$ MeV	Tuttle	26	10.57 MeV (thermal)	Bondarenko Structure	KAPROS-DIXCON KAPROS-DXEVA KAPROS-DXPRT
SWEDEN	ENDF/B-III		~330		. SPENG code; B1 approx; geometric B^2 . Collapse to recommended 27 gr. over an "ave core" composition . smooth and temp and σ_0 dependent resonance σ_0 's . Collapse from 27 to 10 gr. using regionally local spectra			10	10 MeV 13.7 MeV	1 gr. in 10 gr. set = 2 groups in 27 gr. set except: 1 = 1 + 2 + 3 8 = 16 + 17 + 18 9 = 19 + 20 + 21 + 22 10 = 23 - 27	
UKAEA (England)	FGL5 (1971) (Adjusted)				As per specs except: . data for other than fuel isotopes are at 900°K			27	10 MeV 13.7 eV	As per specs	diff: GOG mesh integral finite diff Edits: COSMOS/RATES Pert: COSMOS/PERT
USSR	MIAB-70	49,28,25 eval. in 1970		. Bondarenko σ_0 method for resonance self shielding . b_j factors used in elastic removal				26	10.57 MeV (thermal)	Bondarenko . Mn lumped with Cr . All one set?	diff. 2D, RZ Edits:

TABLE III. Broad Group Energy Boundaries

Specification 27 gr.				Cadarache 25 gr.				Sweden 10 gr.				Bondarenko 25 & 26 gr.			
Group	E_{top}	Δu		Group	E_{top}	Δu		Group	E_{top}	Δu		Group	E_{top}	Δu	
1	10.000 MeV	0.5		1	14.5 MeV	1.37		1	10 MeV	1.5		1	10.57 MeV	0.48	
2	6.065											2	6.550	0.48	
3	3.679			2	3.679	0.5						3	4.024	0.48	
4	2.231			3	2.231			2	2.231	1.0		4	2.514	0.57	
5	1.353			4	1.353							5	1.408	0.57	
6	820.85 keV			5	820.85 keV			3	820.85 keV			6	802.1 keV	0.69	
7	497.87			6	497.87							7	402.8	0.69	
8	301.97			7	301.97			4	301.97			8	200.5	0.69	
9	183.16			8	183.16							9	100.7	0.77	
10	111.09			9	111.09			5	111.09			10	46.51		
11	67.380			10	67.380							11	21.65		
12	40.868			11	40.868			6	40.868			12	10.08		
13	24.788			12	24.788							13	4.655		
14	15.034			13	15.034			7	15.034			14	2.167		
15	9.119			14	9.119							15	1000. eV		
16	5.531			15	5.531			8	5.531	1.5		16	466.0		
17	3.355			16	3.355							17	215.1		
18	2.035			17	2.035							18	100.1		
19	1.234			18	1.234			9	1.234	2.0		19	46.64		
20	748.52 eV			19	748.52 eV							20	21.53		
21	454.00			20	454.00							21	10.02		
22	275.36			21	275.36	1.0						22	4.669		
23	167.02							10	167.02	2.5		23	2.156		
24	101.30			22	101.30	1.5						24	1.003		
25	61.442											25	0.4674		
26	37.267											26	0.2164		
27	22.603			23	22.603	2.0									
	13.717			24	3.059	2.0									
				25	0.414	2.795									
					0.0253										

ANL, EIR-1, EIR-2, HEDL,
JAERI-1, UKAEA

CADARACHE-1, CADARACHE-2
CNEN

SWEDEN

JAERI-2
JAERI-3
BELGIUM
KARLSRUHE-1
KARLSRUHE-2
USSR

} 25 gr.
} 26 gr.

Fig. 1. Broad Group Energy Boundaries.

B. Broad Group Data Comparisons: Groupings and Range of Variations

1. χ Spectra: Table IV summarizes what is known about the fission neutron emission spectra (χ spectra) used by the participants. Figures 2-4 compare the χ spectra provided in Table C4.C-3 of the problem specifications.

Karlsruhe studied the effect on the k_{∞} (inner core) and k_{eff} (configuration 1) values of a variation in the χ spectrum. Application of a Watt-Cranberg fission spectrum of ^{239}Pu

$$\chi(E) = 0.48394e^{-E} \sinh \sqrt{2E}$$

instead of the Bondarenko spectrum of ^{235}U reduced k_{∞} (208 group solution) of the inner core composition by 0.003461 and reduced k_{eff} (26 group solution) for configuration 1 by 0.003069.

2. Flux Spectra: Participants reported central flux spectra for Configuration 1 in Table D8 of the problem specifications. These results are shown in Figs. 5-7. The hardness of the spectrum may be characterized by its centroid

$$\langle E \rangle_k = (10^7 \text{ eV}) e^{-\langle u \rangle_k} \quad (1)$$

where

$$\langle u \rangle_k = \sum_{g=1}^{G_k} \left(\frac{\phi_g}{\Delta u_g} \right) \left(\frac{u_{\text{top},k}^g + \Delta u_k^g}{2} \right) \Delta u_k^g \quad (2)$$

and $u = 0$ at 10^7 eV. (3)

The central flux centroids are compared in Table V. Very little variation is seen about the average value of 110.562 keV. The standard deviation is 4.213 keV, and the range is from 101.988 keV (for JAERI-3) to 117.996 keV (for CADARACHE-1) and 118.106 keV (for USSR). From the plotted spectra in Figs. 5-7, the JAERI-3 solution clearly appears softer than the others. The softness of the JAERI-3 solution is evident in Fig. 6, particularly in the 20-40 keV and 400-800 keV energy ranges. There is a good deal of variation among all participants in the 10-300 keV energy range.

TABLE IV. χ Spectrum SummaryANL $\chi(E)$ Maxwellian

$$\begin{aligned}
 T &= 1.39679 \text{ MeV for } ^{239}\text{Pu}, \\
 &= 1.36847 \text{ MeV for } ^{240}\text{Pu}, \\
 &= 1.36435 \text{ MeV for } ^{241}\text{Pu}, \\
 &= 1.34748 \text{ MeV for } ^{238}\text{U}.
 \end{aligned}$$

These are blended according to fission fraction to form a set χ_i .

BELGIUM

ABBN Spectrum for $\bar{\nu} = 2.8$
 ($\sim \text{U}^{235}$ fast fission)

CADARACHE-1

$$\chi(E) = \frac{\exp \frac{E_f}{T}}{\sqrt{\pi E_f T}} \exp \left(\frac{-E}{T} \right) \text{Sinh} \left(\frac{2}{T} \sqrt{E E_f} \right)$$

$$\begin{aligned}
 T &= .965 \text{ MeV for } \text{U}^{235} \\
 &1.044 \text{ MeV for } \text{Pu}^{239}
 \end{aligned}$$

Displays χ^{49}

CADARACHE-2

No discussion

CNEN

No discussion

EIR-1

No discussion

EIR-2

No discussion

HEDL

Maxwellian form, with $T = 1.41 \text{ MeV}$

TABLE IV. (Cont'd.)

JAERI-1

$$\chi(E) = \sqrt{\frac{E}{c}} \exp\left(\frac{-E}{T}\right)$$

$T = 1.393 \text{ MeV}$

JAERI-2

$$\chi(E) = \frac{2}{\pi} \sqrt{\frac{E}{T}} \frac{1}{T} \exp\left(\frac{E}{T}\right)$$

$T = 1.40 \text{ MeV}$

JAERI-3

$\chi(E)$ Maxwellian $T = 1.40 \text{ MeV}$ (for Pu^{239})

?! "contributions from other fissile were ignored"

KARLSRUHE-1

Supplied in with group cross sections

$$\chi(E) = 0.48394 e^{-E} \sinh \sqrt{2E}$$

Studied sensitivity of k_{∞} and k_{eff} (ref.) to this was Watt/Cranberg change to above Bondarenko χ for U^{235}

$$\delta k_{\infty} (\text{core A}) |_{25 \text{ gr.}} = 1.11471 - 1.117932$$

$$k_{\text{eff}} (\text{ref.}) |_{26 \text{ gr.}} = 0.99651 - 1.000020$$

$$\chi(E) = .45270 \exp\left(\frac{-E}{.965}\right) \sinh\left(\sqrt{2.29E}\right)$$

KARLSRUHE-2

Maxwellian form, with $T = 1.41 \text{ MeV}$

UKAEA

Watt-Cranberg formula: $A \exp \frac{-E}{T} \sinh \frac{\sqrt{WE}}{T}$

$$\bar{E} = \frac{3T}{2} + W$$

Uranium isotopes: $T = 0.965 \text{ MeV}$, $W = 0.5331 \text{ MeV}$, $\bar{E} = 1.98 \text{ MeV}$

Plutonium isotopes: $T = 1.0036 \text{ MeV}$, $W = 0.5544 \text{ MeV}$, $\bar{E} = 2.06 \text{ MeV}$

USSR

No discussion

BROAD GROUP FISSION SOURCE SPECTRUM

Fig. 2.

BROAD GROUP FISSION SOURCE SPECTRUM

Fig. 3.

BROAD GROUP FISSION SOURCE SPECTRUM

Fig. 4.

REAL BROAD GROUP SPECTRA AT CORE CENTER

Fig. 5.

REAL BROAD GROUP SPECTRA AT CORE CENTER

Fig. 6.

REAL BROAD GROUP SPECTRA AT CORE CENTER

Fig. 7.

TABLE V. Centroids of Central Flux Spectra
for Configuration 1

Participant (Solution)	$\langle E \rangle$ (ϕ) (keV)
ANL	107.720
BELGIUM	108.005
CADARACHE-1	117.996
CADARACHE-2	---
CNEN	113.793
EIR-1	109.117
EIR-2	111.779
HEDL	---
JAERI-1	108.758
JAERI-2	108.916
JAERI-3	101.988
KARLSRUHE-1	108.797
KARLSRUHE-2	112.058
SWEDEN	108.845
UKAEA	111.991
USSR	118.106
Mean	110.562
Std. Dev.	4.213

C. Broad Group Cross Sections and Generation of Effective One-Group Values

Broad group cross sections were supplied by participants* in Tables C4.c-1 and C4.c-2 of the problem specifications. These cross sections are plotted (versus energy) for comparison in:

Figures 8 - 13 for heavy isotope capture;

Figures 14 - 19 for heavy isotope fission;

Figures 20 - 31 for light isotope transport.

Several general observations may be made from an examination of the cross section plots:

1. For the capture and fission cross sections of ^{239}Pu and ^{238}U , the variations appear relatively small in the important energy range (though σ_c^{49} displays some variation above 700 keV, the cross section is very low and the variation is of little consequence). However, integral parameters depend on these cross sections in a very sensitive way, and even a few percent variation gives rise to significant differences in integral parameters. This aspect will be discussed below when effective one-group cross sections are discussed.

2. The transport cross sections of iron, sodium, and oxygen display very substantial variations in the important energy ranges. Of particular interest is the fact that large variations occur even among the participants using the same (ENDF/B-IV) basic data set. These variations must be attributed to cross section processing alone.

3. For the transport cross section of oxygen, the CNEN (ENDF/B-IV) and SWEDEN (ENDF/B-III) cross sections show fine structure not present in the other participants results.

*Three participants elected not to supply broad group cross sections. UKAEA supplied values "Not For Publication." Cadarache and HEDL supplied no values.

BROAD GROUP CAPTURE CROSS SECTION FOR U-238

Fig. 8.

BROAD GROUP CAPTURE CROSS SECTION FOR U-238

Fig. 9.

BROAD GROUP CAPTURE CROSS SECTION FOR U-238

Fig. 10.

BROAD GROUP CAPTURE CROSS SECTION FOR PU-239

Fig. 11.

BROAD GROUP CAPTURE CROSS SECTION FOR PU-239

Fig. 12.

BROAD GROUP CAPTURE CROSS SECTION FOR PU-239

Fig. 13.

BROAD GROUP FISSION CROSS SECTION FOR U-238

Fig. 14.

BROAD GROUP FISSION CROSS SECTION FOR U-238

Fig. 15.

BROAD GROUP FISSION CROSS SECTION FOR U-238

Fig. 16.

BROAD GROUP FISSION CROSS SECTION FOR PU-239

Fig. 17.

BROAD GROUP FISSION CROSS SECTION FOR PU-239

Fig. 18.

BROAD GROUP FISSION CROSS SECTION FOR PU-239

Fig. 19.

BROAD GROUP TRANSPORT CROSS SECTION FOR B-10

Fig. 20.

BROAD GROUP TRANSPORT CROSS SECTION FOR B-10

Fig. 21.

BROAD GROUP TRANSPORT CROSS SECTION FOR B-10

Fig. 22.

BROAD GROUP TRANSPORT CROSS SECTION FOR FE

Fig. 23.

BROAD GROUP TRANSPORT CROSS SECTION FOR FE

Fig. 24.

BROAD GROUP TRANSPORT CROSS SECTION FOR FE

Fig. 25.

BROAD GROUP TRANSPORT CROSS SECTION FOR NA

Fig. 26.

BROAD GROUP TRANSPORT CROSS SECTION FOR NA

Fig. 27.

BROAD GROUP TRANSPORT CROSS SECTION FOR NA

Fig. 28.

BROAD GROUP TRANSPORT CROSS SECTION FOR O

Fig. 29.

BROAD GROUP TRANSPORT CROSS SECTION FOR O

Fig. 30.

BROAD GROUP TRANSPORT CROSS SECTION FOR O

Fig. 31.

4. The transport cross section of ^{10}B appears rather consistent among participants. Again, however, the control rod worth and the boron isotopic worth are sensitive functions of the boron cross section and can be expected to vary with even small variations in $\sigma_{\text{tr}}^{10\text{B}}$.

The cross sections may be characterized by their effective, one-group values:

$$\langle \sigma \rangle_k = \frac{\sum_{g=1}^{G_k} \sigma_k^g \phi_k^g}{\sum_{g=1}^{G_k} \phi_k^g}, \quad (4)$$

where the subscript k denotes participant. The group fluxes, ϕ_k^g , used in the flux weighting are for the core center. These values were provided by participants in Table D8 of the problem specifications and plotted in Figs. 5-7. Note that because of the different energy ranges spanned by the different group structures, the $\langle \sigma \rangle_k$'s are averaged over slightly different energy ranges. However, since the flux is almost zero below 100 eV and above 10 MeV, the average cross sections are basically for the common energy range $100 \text{ eV} \leq E \leq 10 \text{ MeV}$.

Table VI displays the effective one-group cross sections of Equation 4* for the most important isotopes of the model. The effective one-group cross sections are also tabulated in the legends of the cross section plots. Also shown in Table VI is the capture-to-fission ratio, α , for ^{239}Pu :

$$\langle \alpha \rangle_k = \frac{\langle \sigma_c \rangle_k}{\langle \sigma_f \rangle_k}. \quad (5)$$

*Values for UKAEA are shown since it is impossible to reconstruct the "Not For Publication" group-wise cross sections from the one-group values. The one-group values for CADARACHE-1 and -2 were provided by Cadarache.

TABLE VI. Summary of One-Group-Averaged Cross Sections

Organization	k-infinity Inner Core	E(Phi) in keV	²³⁸ U Capture	²³⁸ U Fission	²³⁹ Pu Capture	²³⁹ Pu Fission	²³⁹ Pu Alpha	¹⁰ B Transport	Fe Transport	Na Transport	O Transport
Cross Sections											
ANL	1.10850	107.720	0.30687	0.04031	0.56781	1.84297	0.30810	5.44836	4.01792	4.15189	3.53076
BELGIUM	1.09500	108.005	0.33170	0.04004	0.45194	1.82496	0.24764	---	4.71243	4.78848	3.74735
CADARACHE-1	1.15242	117.996	0.29234	0.03867	0.56750	1.81550	0.31259	---	---	---	---
CADARACHE-2	---	---	0.29550	0.03919	0.57910	1.81625	0.31884	---	---	---	---
CNEN	1.14558	113.793	0.30250	0.04518	0.57407	1.87984	0.30538	5.40132	4.07896	4.10724	3.53955
EIR-1	1.10442	109.117	0.30448	0.03995	0.57734	1.87503	0.30791	5.46198	5.36215	4.54541	3.58767
EIR-2	1.11735	111.779	0.32230	0.04158	0.58030	1.92367	0.30166	5.48159	4.54193	4.36690	3.50487
HEDL	1.11870	---	---	---	---	---	---	---	---	---	---
JAERI-1	1.11710	108.758	0.30991	0.04457	0.60820	1.88247	0.32309	5.52344	4.79261	4.45146	3.51690
JAERI-2	1.13564	108.916	0.29624	0.04048	0.52881	1.84266	0.28698	5.51229	3.93776	4.14213	3.49352
JAERI-3	1.13189	101.988	0.30905	0.04381	0.58704	1.89176	0.31031	5.79390	3.97266	4.38019	3.50360
KARLSRUHE-1	1.11840	108.797	0.30858	0.04097	0.57101	1.87191	0.30504	---	3.12493	4.06499	3.34958
KARLSRUHE-2	1.14843	112.058	0.30406	0.04159	0.54818	1.86915	0.29328	---	3.85887	4.55270	3.54034
SWEDEN	1.08680	108.845	0.30856	0.03868	0.54087	1.88136	0.28749	5.47947	3.76949	4.18664	3.48434
UKAEA	1.15123	111.991	0.29137	0.04302	0.54539	1.83265	0.29760	5.41029	3.53615	4.09901	3.31038
USSR	1.13270	118.106	0.30915	0.04009	0.49816	1.76992	0.28146	5.35558	4.06790	4.29630	3.44045
Mean	1.12428	110.562	0.30617	0.04121	0.55505	1.85467	0.29916	5.48682	4.13644	4.31794	3.50379
Std. Dev.	0.02054	4.213	0.01069	0.00207	0.03909	0.03889	0.01850	0.11979	0.58546	0.22086	0.10643

The heavy element effective one-group cross sections display the following variations (as expressed in percent standard deviation):

$\langle \sigma_c^{28} \rangle$	$\pm 3.49\%$
$\langle \sigma_f^{28} \rangle$	$\pm 5.02\%$
$\langle \sigma_c^{49} \rangle$	$\pm 7.04\%$
$\langle \sigma_f^{49} \rangle$	$\pm 2.10\%$
$\langle \alpha^{49} \rangle$	$\pm 6.18\%$

For ^{238}U capture, the maximum (BELGIUM)-to-minimum (UKAEA) spread in one-group-averaged cross sections is 13.17% (relative to the mean). For ^{238}U fission, the maximum (CNEN)-to-minimum (CADARACHE-1) spread is 15.80%. For ^{239}Pu capture, the maximum (JAERI-1)-to-minimum (BELGIUM) spread is 28.15%. For ^{239}Pu fission, the maximum (EIR-2)-to-minimum (USSR) spread is 8.29%. For ^{239}Pu alpha, the maximum (JAERI-1)-to-minimum (BELGIUM) spread is 25.22%.

The light element effective one-group transport cross sections displays the following (1σ) variations:

$\langle \sigma_{tr}^{\text{Fe}} \rangle$	$\pm 14.15\%$
$\langle \sigma_{tr}^{\text{Na}} \rangle$	$\pm 5.11\%$
$\langle \sigma_{tr}^{16\text{O}} \rangle$	$\pm 3.04\%$
$\langle \sigma_{tr}^{10\text{B}} \rangle$	$\pm 2.18\%$

The variation in the light element capture cross sections is reflected in the following (1σ) variation in capture rates:

<u>Material</u>	<u>Relative Capture Rate</u>	<u>1σ Variation</u>
Fe	1.00	25%
Cr	0.39	39%
Ni	0.50	15%
Na	0.16	31%

Approximate values for $\langle \bar{\nu} \rangle$ for the inner core composition were obtained from the inner core neutron balance reported in Tables D2.a-1 and D2.a-3 of the problem specifications.

$$\langle \bar{\nu} \rangle = \frac{(\text{Inner Core Fission Production})}{(\text{Inner Core Fission})}$$

There is some approximation introduced by the influence of spectral contamination near the inner core/outer core interface, but is expected to be slight.

Table VII displays the $\langle \bar{\nu} \rangle$ values. There is little variation of $\langle \bar{\nu} \rangle$ about the mean value of 2.9145. The maximum value (KARLSRUHE-2) is +0.38% above the mean; the minimum value (EIR-1) is -0.25% below the mean. The three solutions using ENDF/B-IV data (ANL, CNEN, and EIR-1) all have the smallest total $\langle \bar{\nu} \rangle$ values. Though these variations are small, it is noted that the variations in $\langle \bar{\nu} \rangle$ alone could account for $\sim 0.63\%$ δk (.38% + .25%) swing in the eigenvalue.

TABLE VII. $\langle \bar{\nu} \rangle$ for Inner Core

Participant	From Tables D2.a-1 and D2.a-3		
	^{239}Pu	^{238}U	Total
ANL	2.9261	2.7636	2.9079
BELGIUM	2.9426	2.7999	2.9250
CADARACHE-1	2.9210	2.8690	2.9163
CADARACHE-2	-----	-----	-----
CNEN	2.9282	2.7685	2.9097
EIR-1	2.9276	2.7512	2.9072
EIR-2	2.9320	2.7712	2.9115
HEDL	-----	-----	-----
JAERI-1	2.9315	2.7712	2.9113
JAERI-2	2.9381	2.7585	2.9124
JAERI-3	2.9365	2.7567	2.9126
KARLSRUHE-1	2.9426	2.7873	2.9197
KARLSRUHE-2	2.9414	2.8285	2.9256
SWEDEN	2.9296	2.7560	2.9081
UKAEA	2.9380	2.7660	2.9140
USSR	2.9350	2.8141	2.9213
Mean	2.9336	2.7830	2.9145
Std. Dev.	0.0066	0.0339	0.0062

D. Discussion of Data Set Comparisons

In order to examine the broad group cross section data for trends, the results in Tables V, VI, and VII have been organized into the following categories:

All solutions

Adjusted Data Set Solutions Only

CADARACHE-1, CADARACHE-2, JAERI-2, JAERI-3,

KARLSRUHE-2, UKAEA

Non-Adjusted Data Set Solutions Only

ANL, BELGIUM, CNEN, EIR-1, EIR-2, HEDL,

JAERI-1, KARLSRUHE-1, SWEDEN, USSR

ENDF/B-IV Data Set Solutions Only

ANL, CNEN, EIR-1, HEDL

For each of the categories, the mean value and the standard deviation are computed. The results of organizing the data in this manner are displayed in Table VIII.

With regard to the sizes of variations, inspection of Table VIII leads to the following conclusions:

1. For participants using the same (ENDF/B-IV) basic data set, but different processing codes,
 - a. the variation in heavy element $\langle \sigma \rangle$'s is small ($\sim 1\%$) except for $\langle \sigma_f^{28} \rangle$, where a standard deviation of 7% is observed, while
 - b. the variation in light element $\langle \sigma_{tr} \rangle$ is large, particularly for Fe ($\pm 16.92\%$).
 - c. It appears then, that even with a given basic data set, cross section processing differences lead to non-trivial differences in cross sections.

TABLE VIII. Comparison of Effective One-Group Cross Sections

	$\langle E \rangle$	$\langle \sigma_c^{28} \rangle$	$\langle \sigma_f^{28} \rangle$	$\langle \sigma_c^{49} \rangle$
<u>All Sets</u>				
Mean	110.562	0.30617	0.04121	0.55505
Std. Dev. *100%/Mean	3.81	3.49	5.02	7.04
(Max-Mean)*100%/Mean	+6.82 USSR	+8.34 BELGIUM	+9.64 CNEN	+9.58 JAERI-1
(Min-Mean)*100%/Mean	-7.76 JAERI-3	-4.83 UKAEA	-6.16 CADARACHE-1	-18.58 BELGIUM
(Max-Min) *100%/Mean	14.58	13.17	15.80	28.15
<u>Adjusted Sets</u>				
Mean	110.590	0.29809	0.04113	0.55934
Std. Dev. *100%/Mean	5.27	2.34	5.01	3.98
(Max-Mean)*100%/Mean	+6.70 CADARACHE-1	+3.68 JAERI-3	+6.53 JAERI-3	+4.95 JAERI-3
(Min-Mean)*100%/Mean	-7.78 JAERI-3	-2.25 UKAEA	-5.97 CADARACHE-1	-5.46 JAERI-2
(Max-Min) *100%/Mean	14.48	5.93	12.50	10.41
<u>Non-Adjusted Sets</u>				
Mean	110.547	0.31156	0.04126	0.55219
Std. Dev. *100%/Mean	3.12	3.01	5.33	8.76
(Max-Mean)*100%/Mean	+6.84 USSR	+6.46 BELGIUM	+9.49 CNEN	+10.14 JAERI-1
(Min-Mean)*100%/Mean	-2.56 ANL	-2.91 CNEN	-6.26 SWEDEN	-18.16 BELGIUM
(Max-Min) *100%/Mean	9.40	9.37	15.75	28.30
<u>ENDF/B-IV Sets</u>				
Mean	110.210	0.30462	0.04181	0.57307
Std. Dev. *100%/Mean	2.89	0.72	6.98	0.86
(Max-Mean)*100%/Mean	+3.25 CNEN	+0.74 ANL	+8.05 CNEN	+0.75 EIR-1
(Min-Mean)*100%/Mean	-2.26 ANL	-0.70 CNEN	-4.46 EIR-1	-0.92 ANL
(Max-Min) *100%/Mean	5.51	1.43	12.51	1.66

Percent Differences Relative to the Mean of the Adjusted Sets

Adjusted Sets	---	---	---	---
Non-Adjusted Sets	-0.04	+4.52	+0.32	-1.28
ENDF/B-IV Sets	-0.34	+2.19	+1.65	+2.45

TABLE VIII. (Cont'd.)

	$\langle \sigma_f^{49} \rangle$	$\langle \alpha^{49} \rangle$	$\langle v \rangle$	$\langle \sigma_{tr}^{Fe} \rangle$
<u>All Sets</u>				
Mean	1.85467	0.29916	2.9145	4.13644
Std. Dev. *100%/Mean	2.10	6.18	0.21	14.16
(Max-Mean)*100%/Mean	+3.72 EIR-2	+8.00 JAERI-1	+0.38 KARLSRUHE-2	+29.63 EIR-1
(Min-Mean)*100%/Mean	-4.57 USSR	-17.22 BELGIUM	-0.25 EIR-1	-24.45 KARLSRUHE-1
(Max-Min) *100%/Mean	8.29	25.22	0.63	54.08
<u>Adjusted Sets</u>				
Mean	1.84466	0.30327	2.9162	3.82636
Std. Dev. *100%/Mean	1.65	4.11	0.19	5.21
(Max-Mean)*100%/Mean	+2.55 JAERI-3	+5.14 CADARACHE-2	+0.32 KARLSRUHE-2	+3.82 JAERI-3
(Min-Mean)*100%/Mean	-1.58 CADARACHE-1	-5.37 JAERI-2	-0.13 JAERI-2	-7.58 UKAEA
(Max-Min) *100%/Mean	4.13	10.51	0.45	11.40
<u>Non-Adjusted Sets</u>				
Mean	1.86135	0.29642	2.9135	4.27425
Std. Dev. *100%/Mean	2.36	7.39	0.23	15.34
(Max-Mean)*100%/Mean	+3.35 EIR-2	+9.00 JAERI-1	+0.39 BELGIUM	+25.45 EIR-1
(Min-Mean)*100%/Mean	-4.91 USSR	-16.46 BELGIUM	-0.22 EIR-1	-11.81 SWEDEN
(Max-Min) *100%/Mean	8.26	25.45	0.61	37.26
<u>ENDF/B-IV Sets</u>				
Mean	1.86595	0.30713	2.9083	4.48634
Std. Dev. *100%/Mean	1.07	0.49	0.04	16.92
(Max-Mean)*100%/Mean	+0.74 CNEN	+0.32 ANL	+0.05 CNEN	+19.52 EIR-1
(Min-Mean)*100%/Mean	-1.23 ANL	-0.57 CNEN	-0.04 EIR-1	-10.44 ANL
(Max-Min)**100%/Mean	1.98	0.89	0.09	29.96
Percent Differences Relative to the Mean of the Adjusted Sets				
Adjusted Sets	---	---	---	---
Non-Adjusted Sets	+0.90	-2.26	-0.09	+11.71
ENDF/B-IV	+1.15	+1.27	-0.27	+17.25

TABLE VIII. (Cont'd.)

	$\langle \sigma_{tr}^{Na} \rangle$	$\langle \sigma_{tr}^{16O} \rangle$	$\langle \sigma_{tr}^{10B} \rangle$
<u>All Sets</u>			
Mean	4.31794	3.50379	5.48682
Std. Dev. *100%/Mean	5.11	3.04	2.18
(Max-Mean)*100%/Mean	+10.90 BELGIUM	+6.95 BELGIUM	+5.60 JAERI-3
(Min-Mean)*100%/Mean	-5.86 KARLSRUHE-1	-5.52 UKAEA	-2.39 USSR
(Max-Min) *100%/Mean	16.76	12.47	7.98
<u>Adjusted Sets</u>			
Mean	4.29351	3.46196	5.57216
Std. Dev. *100%/Mean	4.95	2.98	3.57
(Max-Mean)*100%/Mean	+6.04 KARLSRUHE-2	+2.26 KARLSRUHE-2	+3.98 JAERI-3
(Min-Mean)*100%/Mean	-4.53 UKAEA	-4.38 UKAEA	-2.90 UKAEA
(Max-Min) *100%/Mean	10.57	6.64	6.88
<u>Non-Adjusted Sets</u>			
Mean	4.32881	3.52238	5.45024
Std. Dev. *100%/Mean	5.46	3.07	1.03
(Max-Mean)*100%/Mean	+10.62 BEGIUM	+6.39 BELGIUM	+1.34 JAERI-1
(Min-Mean)*100%/Mean	-6.09 KARLSRUHE-1	-4.91 KARLSRUHE-1	-0.90 CNEN
(Max-Min) *100%/Mean	16.71	11.29	2.24
<u>ENDF/B-IV Sets</u>			
Mean	4.26818	3.55266	5.43721
Std. Dev. *100%/Mean	5.65	0.86	0.61
(Max-Mean)*100%/Mean	+6.50 EIR-1	+0.99 EIR-1	+0.46 EIR-1
(Min-Mean)*100%/Mean	-3.77 CNEN	-0.62 ANL	-0.66 CNEN
(Max-Min) *100%/Mean	10.27	1.60	1.12

Percent Differences Relative to the Mean of the Adjusted Sets			
Adjusted Sets	---	---	---
Non-Adjusted Sets	+0.82	+1.75	-2.19
ENDF/B-IV Sets	-0.59	+2.62	-2.42

2. For participants using various adjusted basic data sets,
 - a. the variations in both heavy element and light element $\langle\sigma\rangle$'s, though slightly smaller, are of the same characteristic size as the variations observed among the participants using non-adjusted data sets. The standard deviations of effective one-group $\langle\sigma\rangle$'s for the adjusted data sets are 6% and 22% smaller (relative to the standard deviations for the non-adjusted data sets) for the capture and fission cross sections of ^{238}U , 55% and 30% smaller for the capture and fission cross sections of ^{239}Pu , and 17% smaller for $\langle\bar{\nu}\rangle$. For the light elements there is little trend. For the adjusted data sets the standard deviation in $\langle\sigma_{\text{tr}}\rangle$ for iron is 66% smaller (relative to the standard deviation of the non-adjusted data sets), virtually the same variations (adjusted vs. non-adjusted) in $\langle\sigma_{\text{tr}}\rangle$ for sodium and oxygen, and much greater variations (by $\sim 250\%$) in $\langle\sigma_{\text{tr}}\rangle$ for boron. The transport cross sections of sodium and oxygen are generally not adjusted in the adjusted data sets and therefore should be similar to the non-adjusted data sets. The adjustments to the iron transport cross section appear to be large. Not only do the adjusted data sets have a larger spread in values of $\langle\sigma_{\text{tr}}\rangle$ for iron, but also the mean of these values is $\sim 12\%$ smaller than the non-adjusted sets (and $\sim 17\%$ smaller than the ENDF/B-IV sets).
 - b. Independent adjustment processes then do not lead to a common set of cross sections - though (as will be discussed in later sections of this report) they may lead to a common set of calculated integral parameters.

There are perhaps a few additional comments in regard to the "Adjusted Sets" category in Table VIII. It is noted that the solution with the hardest spectra (CADARACHE-1), as characterized by the largest $\langle E \rangle$, also has the smallest effective one-group ^{238}U fission cross section, $\langle \sigma_f^{28} \rangle$. On the other hand, the solution with the softest spectra (JAERI-3) has the largest $\langle \sigma_f^{28} \rangle$. Therefore, the difference between these two adjusted ^{238}U fission cross sections is actually somewhat larger than the 12.5% difference (shown in Table VIII) between their $\langle \sigma_f^{28} \rangle$ values. It may also be noted that among the adjusted data sets, the range of $\langle \sigma_c^{49} \rangle$ values is 10.41%. This difference is between the JAERI-3 (maximum) and JAERI-2 (minimum) values which are derived from similar adjustment procedures.

The large discrepancies observed for the structural cross-sections confirm the necessity of complementary experiments (both microscopic measurements and integral experiments) specifically devoted to the structural material neutronic characteristics.

Further comparisons of these effective one-group cross sections will be made later in this report (see Section V) with the discussion of the results of calculated integral parameters.

III. REVIEW OF OVERALL RESULTS

The principal results of the sixteen solutions submitted for comparison will now be reviewed. (The complete solutions are included in Appendices B and C.) Some of these results were reviewed and presented in preliminary form at the topical meeting² on "Advances in Reactor Physics" in Gatlinburg, Tenn., April 10-12, 1978. Some participants have submitted slight additions and/or corrections to their original solutions. Where changes have been made subsequent to the preliminary results published in Ref. 2 specific mention will be made. The results contained in this report may be considered final for the present study.

The most important and interesting results of this comparison exercise are summarized in the tables and figures of Sections III and IV of this report. In each table the mean value and the standard deviation of the results are presented in the top two lines. In the remainder of the table the percent difference from the mean value for each parameter and solution are listed. The results are presented in this way to facilitate an intercomparison. There is one exception in Table IX. For the parameters k_{eff} and k_{∞} the Δk differences ($\times 10^2$) are listed rather than percent differences. Whenever a number is missing from one of the tables it can be assumed that this value was not supplied (not all participants submitted complete solutions).

A. Performance Parameters

Some of the key parameters are discussed in the following paragraphs. Here, the discussions will focus on:

- the relative scatter in the values,
- individual values that are particularly interesting or possible discrepant, and
- the correlations among the results for different parameters.

TABLE IX. Eigenvalue, k_{∞} , Leakage Fraction, and Breeding Ratio for Configuration 1

Solution	k_{eff}	$k_{\infty}(\text{IC})$	Leakage Fraction		Breeding Ratio		
			Core	Reactor	Core	Blanket	Total
Mean	1.00509	1.12428	0.1621	0.0164	0.9898	0.4023	1.3921
Std. Deviation	0.01292	0.02054	0.0040	0.0025	0.0421	0.0097	0.0482
	$\Delta k \times 10^2$		Percent Difference Relative to the Mean				
ANL	-1.205	-1.578	1.79	-2.14	-0.03	0.72	0.19
BELGIUM	-2.096	-2.928	0.16	0.54	11.74	2.92	9.19
CADARACHE-1	1.615	2.814	0.07	15.17	-4.09	0.98	-2.62
CADARACHE-2	0.765	---	1.40	10.36	-1.10	2.82	0.04
CNEN	-0.345	2.130	4.75	-10.37	-0.76	1.04	-0.24
EIR-1	-1.050	-1.986	-1.91	-15.07	-2.45	-5.01	-3.19
EIR-2	-0.019	-0.693	-2.27	-14.21	0.16	-2.08	-0.49
HEDL	-0.298	-0.558	---	-8.97	-0.56	-0.37	-0.50
JAERI-1	0.151	-0.718	-4.46	-24.27	-1.14	-1.83	-1.34
JAERI-2	0.690	1.136	1.18	5.30	-3.26	-1.29	-2.69
JAERI-3	0.933	0.761	-2.88	5.30	-2.63	-3.55	-2.89
KARLSRUHE-1	-0.632	-0.588	0.47	13.04	0.31	0.33	0.32
KARLSRUHE-2	1.890	2.415	-3.23	6.33	-3.14	-1.31	-2.61
SWEDEN	-2.458	-3.748	1.37	-24.52	3.50	1.32	2.87
UKAEA	1.723	2.695	1.89	19.75	-3.81	1.34	-2.33
USSR	0.341	0.842	1.64	23.77	7.25	3.98	6.31

In the following discussion effective one-group cross sections are indicated by $\langle \sigma \rangle$, central reactivity worths by ρ and per atom reaction rate ratios by ratios of f and c . Superscripts refer to the isotopes and subscripts, f , c and tr refer to fission, capture and transport reactions. A 1σ standard deviation in the results is indicated by 1σ .

1. Eigenvalue and k_{∞} : Relatively large 1σ variations of 1.3% and 1.8% are observed for the eigenvalue and k_{∞} (see Table IX)*. These results are comparable to those obtained in the earlier "Baker Model" comparison.¹ The maximum spread in the calculated eigenvalues for Configuration 1 is 4.35% δk . The solutions with the largest eigenvalues (Karlsruhe-2, UKAEA, and Cadarache-1) also have:

- the largest k_{∞} ,
- the smallest $^{28}\text{C}/^{49}\text{f}$ (UKAEA and Cadarache-1 only),
- the smallest $\langle \sigma_c^{28} \rangle$ (UKAEA and Cadarache-1 only),
- the largest ρ^{49} (with the Cadarache-2 and Karlsruhe-1 solutions) and,
- the smallest core breeding ratio (with the JAERI-2 solution).

The solutions with the smallest eigenvalues (Sweden and Belgium) also have:

- the smallest k_{∞} .
- the largest $^{28}\text{C}/^{49}\text{f}$ (with the USSR solution),
- the largest $\langle \sigma_c^{28} \rangle$ (Belgium only),
- the smallest ρ^{28} (Belgium only), and
- the largest core breeding ratios (with the USSR solution).

There appears to be no significant correlation between k_{eff} and $\langle \sigma_f^{49} \rangle$, ρ^{28} , or core or reactor leakage. The k_{∞} values generally correlate with k_{eff} values. The Swedish solution was reported earlier with a high k_{∞} ,

*Note that the more meaningful comparison would be between k (adjusted σ 's) and the bias-factor-corrected k (non-adjusted σ 's). Such a comparison was not made within the confines of this exercise.

the smallest k_{eff} and the smallest reactor leakage. Their value of k_{∞} has been revised (reduced by $0.0581 \Delta k$) and is now consistent. That the k_{eff} for the USSR solution is 0.34% above the mean is surprising since it has the second largest value of $^{28}\text{C}/^{49}\text{F}$ (5.31% above the mean value) and the smallest $\langle \sigma_f^{49} \rangle$ (4.57% below the mean); however, it does have a small value of $^{49}\text{C}/^{49}\text{F}$ (5.92% below the mean). Using calculated sensitivity coefficients, Collins³ has shown that the k_{eff} differences between ANL and UKAEA and ANL and Cadarache can be largely explained by the cross section differences. The results of that sensitivity analyses are discussed in Section VII of this report.

Cadarache has indicated the deviations associated with the eigenvalue (for the reference configuration) appear to be dependent on the enrichment values of the core, as illustrated by the following table of $\Delta k/k$ values between CARNAVAL Versions III and IV for different enrichments:

Configurations	BENCHMARK	MASURCA PRE-RACINE	MASURCA/R2
Fissile Enrichment	11%	$\approx 11\%$	$\approx 25\%$
$\Delta k/k \times 10^3$ (CARNAVAL-III - CARNAVAL-IV)	850	500	300

2. Core and Reactor Leakage: The 1σ variation of the core leakage (see Table IX) is only 2.5% while that for the reactor leakage is 15.2%. This is not surprising since there is a relatively small fraction (1.64%) of the neutrons that leak from the reactor. The values of the core and reactor leakage fractions for the ANL solution have been revised. The preliminary values were in error (due to incorrect normalization of the fission production term by k_{eff}) and indicated much larger values of the leakage fractions. The corrected values reported in Table IX are in good agreement with the other solutions.

In order to check the diffusion theory solution, ANL ran a VIM Monte Carlo calculation and a DIF3D transport theory calculation using the same ENDF/B-IV data base as used for the ANL diffusion theory solution. The reactor leakage, as well as most of the other neutron balance parameters, was calculated by VIM to be nearly identical to the values from the ANL diffusion solution. The comparison of the diffusion theory solution with the Monte Carlo and transport theory solutions is discussed in Section VIII of this report.

3. Breeding Ratio: The 1σ variation for the core breeding ratio (4.3%) is about the same as that for $^{28}\text{C}/^{49}\text{F}$ (3.7%), as expected, but is about double the value for the blanket breeding ratio (2.4%). The maximum spread of the values for core, blanket, and total breeding ratio are 15.8%, 9.0%, and 12.4%, respectively. The solutions with the smallest core breeding ratios (UKAEA and Cadarache-1) also have:

- the smallest $^{28}\text{C}/^{49}\text{F}$,
- the smallest $\langle\sigma_{\text{C}}^{28}\rangle$ (with the JAERI-2 solution),
- the largest k_{eff} and k_{∞} , and
- the largest ρ^{49} (with the Cadarache-2 and Karlsruhe-1 solutions).

The solutions with the largest core breeding ratios (Belgium and USSR) have:

- the largest $^{28}\text{C}/^{49}\text{F}$,
- the smallest $\langle\sigma_{\text{F}}^{49}\rangle$,
- the smallest k_{eff} and k_{∞} (Belgium only),
- the largest ρ^{28} (with CNEN and Cadarache-1 and -2 solutions), and
- the smallest sodium-void worth.

Most of these trends are consistent and easily understandable. The correlation with sodium void is not apparent and may be only accidental. The strong correlation between $^{28}\text{C}/^{49}\text{F}$ and breeding ratio, k_{eff} and k_{∞} is very apparent.

B. Safety Parameters

Overall safety parameters, including central control rod worth, sodium-void reactivity, and Doppler reactivity, are compared in Table X.

1. Central Control Rod Worth: The large inconsistency in these results (see Table X) was one of the real surprises associated with this exercise. The 1σ variations for the sodium-in and sodium-voided cases of 13% and 10% are unusually large compared to normal accuracy requirements for control rod calculations. The maximum spread of these values are 45% and 32%, respectively. The solutions with the largest central control rod worths (USSR, Cadarache-1 and UKAEA) are not well-correlated with the solutions with the largest central ρ^{B10} or $\langle \sigma_{tr}^{B10} \rangle$, although the Cadarache and UKAEA solutions did have relatively large values of ρ^{B10} . The 1σ variation for the central control rod results is two to three times larger than that for the central ρ^{B10} results indicating that the large variation in the control rod results is more than simply a normalization problem. Differences in the calculated radial power shapes (to be more fully discussed below) contribute significantly to the central control rod problem. The more highly buckled shapes (see Figs. 41 and 43) of the UKAEA and Cadarache-1 solutions tend to enhance the worth of the central rod; however, these shapes have a similar affect on the central ρ^{B10} and the other central worths. Thus, the buckling effect contributes but does not explain the control rod problem. Figure 44 shows ANL calculations of the radial distributions with the rod inserted indicating large spatial perturbations, and differing spatial flux perturbations resulting from rod insertion may be contributing to the control rod problem. It may also be noted that the control rod worth was obtained from direct flux calculations, whereas ρ^{B10} also involved the calculated adjoint flux.

These two parameters - direct and adjoint fluxes - are not sensitive to the

TABLE X. Central Control Rod, Sodium Void and Fuel Doppler Reactivity Worths

Solution	Central B ₄ C Control Rod Worth ($k_{rod} - k_{ref}$)/ k_{ref}			Na-Void Worth ($k_{void} - k_{ref}$)/ k_{ref}		Isothermal Fuel Doppler Worth ($k_{2200} - k_{1100}$)/ k_{1100}	
	Na In Rel. to Fuel	Na In Rel. to Na	Na Void Rel. to Fuel	Inner Core	Core Plus Axial Blanket	Config. 1 Na In	Config. 3 Na Void
Mean	-0.00355	-0.00298	-0.00460	0.02116	0.02098	-0.00729	-0.00433
Std. Deviation	0.00047	0.00043	0.00047	0.00255	0.00352	0.00079	0.00062
Percent Difference Relative to the Mean							
ANL	-5.61	-7.09	-2.63	11.54	16.77	-1.46	3.91
BELGIUM	-3.23	-1.99	-2.63	-32.57	-44.50	-3.55	0.03
CADARACHE-1	20.03	21.09	19.33	6.58	6.23	3.48	12.22
CADARACHE-2	7.91	19.74	17.59	2.14	1.37	-4.07	6.45
CNEN	-5.05	-6.42	-12.62	7.86	5.04	1.70	0.45
EIR-1	-12.37	-13.13	-7.63	14.76	22.96	---	---
EIR-2	-10.68	-11.45	-6.76	-7.46	-7.21	---	---
HEDL	-5.61	-5.75	-4.37	-7.22	-6.64	0.32	12.22
JAERI-1	-10.68	-11.79	-7.19	3.98	9.24	8.15	17.07
JAERI-2	-11.81	-14.13	-8.50	11.50	15.62	-9.42	-4.63
JAERI-3	-12.52	-15.64	-10.02	11.26	17.22	5.26	18.69
KARLSRUHE-1	-1.32	-2.58	-1.22	0.50	-1.51	-6.24	-3.69
KARLSRUHE-2	8.93	8.98	7.33	-8.78	-12.93	-12.56	-6.80
SWEDEN	---	---	---	2.66	3.61	30.65	-31.42
UKAEA	9.60	9.35	8.46	-3.25	-4.39	-0.09	-0.71
USSR	32.42	30.81	10.85	-13.51	-20.89	-12.17	-23.80

same cross-sections and to the same energy ranges. Further analyses are needed on these points.

Another low energy indicator is the Doppler effect, but there appears to be no correlation with the control rod worths. The UKAEA and Cadarache-1 solutions had average Doppler values and the USSR solution actually had a low Doppler value. In another apparent inconsistency, the JAERI-3 solution has essentially the smallest central control rod results but has the largest $\langle \sigma_{tr}^{B10} \rangle$ and ρ^{B10} .

2. Sodium-Void Reactivity: Considering the traditional difficulty in sodium-void calculations, the 1 σ variations of 12% and 17% in the results (see Table X) are surprisingly good. The Belgium values seem to be somewhat inconsistent and contribute significantly to the 1 σ variation and the maximum spread (47% and 67%) in the results. The sodium-void results do not correlate well with $\langle \sigma_{tr}^{NA} \rangle$ or ρ^{NA} . The solutions with the smallest sodium-void results are Belgium and USSR, and the solution with the largest result is EIR-1. The Belgium and EIR-1 solutions have the smallest ρ^{NA} and the Belgium solution has the largest $\langle \sigma_{tr}^{NA} \rangle$ which appears to be inconsistent.

3. Doppler Reactivity: The 1 σ variations of the Doppler values (11% and 14%) are about the same as for the sodium-void values. The maximum spread of these values are 43% and 50%, respectively. This is a somewhat unexpected result which could be due to either the temperature dependence of the ^{238}U cross section or the low energy flux. There is no strong evidence that the spread in the Doppler results is correlated with differences in the low energy flux. The USSR solution, which has among the lowest Doppler values, has the largest central control rod results and an average ρ^{B10} result. The Doppler values of the JAERI-3 solution, which have been revised (reduced considerably) from their preliminary values, are large; however, this solution

has the smallest central control rod value. The central control rod results have consistently not correlated with other low energy indicators. There appears to be some difficulty with the Swedish Doppler solution which has the largest sodium-in Doppler and the smallest sodium-void Doppler.

IV. REVIEW OF DETAILED RESULTS

In order to further understand the overall results of the performance and safety parameters, some of the detailed results will now be discussed. As before, only a portion of the submitted solutions are reproduced in this section; however, complete results are given in Appendices B and C.

A. Central Neutron Balance

The major isotopic components of the neutron balance for the inner core of Configuration 1 are given in Table XI. Note that the terms shown in Table XI have been normalized for each solution such that the total fission production in the reactor is 1.0. The following general observations may be made.

- Although the 1σ variation in total fission production is $<1\%$, the 1σ variations in the ^{240}Pu and ^{241}Pu fission production terms are 8.2% and 7.1%, respectively.
- The 1σ variations in the capture terms are much larger. For ^{239}Pu , ^{240}Pu , ^{241}Pu , and ^{238}U , the 1σ variations in capture are 6.7%, 19.3%, 9.7%, and 3.0%, respectively. The 1σ variations in capture in both the structural materials and $\text{Na} + ^{16}\text{O}$ are $\sim 20\%$.

The following observations may be made with regard to particular solutions.

- The Cadarache-1 solution (which has, with UKAEA and Karlsruhe-2, the largest k_{B}^2 , k_{∞} , and k_{eff} values) has: the largest fission and fission production terms in ^{239}Pu ; and the smallest capture in structure (with USSR) and total for the inner core (with UKAEA and Karlsruhe-2). The difference in capture (relative to the mean value) is totally accounted for by the smaller capture value in the structure.

TABLE XI. Isotopic Components of the Inner Core Neutron Balance for Configuration 1

Organization	Fission Production					Fission			Capture					Structure	$^{16}\text{O} + \text{Na}$	k_{Bsquared}^a
	Total	^{239}Pu	^{240}Pu	^{241}Pu	^{238}U	Total	^{239}Pu	^{238}U	Total	^{239}Pu	^{240}Pu	^{241}Pu	^{238}U			
ANL	0.63012	0.41967	0.02405	0.09151	0.08487	0.21669	0.14341	0.03071	0.33996	0.04472	0.01289	0.00611	0.23784	0.03256	0.00374	1.13199
BELGIUM	0.62938	0.40722	0.01927	0.10962	0.08352	0.21517	0.13839	0.02983	0.34836	0.03476	0.01795	0.00598	0.25055	---	0.00260	1.11685
CADARACHE-1	0.64353	0.42506	0.02437	0.09595	0.08825	0.22067	0.14552	0.03076	0.32653	0.04549	0.01247	0.00706	0.23249	0.02229	0.00393	1.17604
CNEN	0.63127	0.41730	0.02755	0.09052	0.08826	0.21695	0.14251	0.03188	0.33409	0.04386	0.01396	0.00596	0.23221	0.03294	0.00420	1.14560
EIR-1	0.62657	0.41914	0.02379	0.09129	0.08237	0.21552	0.14317	0.02994	0.34116	0.04456	0.01267	0.00594	0.23137	0.04177	0.00275	1.12555
EIR-2	0.62784	0.42104	0.02242	0.08993	0.08441	0.21564	0.14360	0.03046	0.33620	0.04378	0.01004	0.00561	0.23949	0.03155	0.00370	1.13772
JAERI-1	0.62852	0.41455	0.02443	0.08827	0.09109	0.21589	0.14141	0.03287	0.33655	0.04617	0.01355	0.00631	0.23274	0.03176	0.00391	1.13772
JAERI-2	0.63140	0.42220	0.02400	0.09127	0.08441	0.21680	0.14370	0.03060	0.32840	0.04160	0.01159	0.00652	0.22960	0.03189	0.00514	1.15811
JAERI-3	0.63220	0.41743	0.02435	0.09084	0.08940	0.21706	0.14215	0.03243	0.33102	0.04462	0.01153	0.00673	0.23131	0.03034	0.00417	1.15348
KARLSRUHE-1	0.63620	0.42580	0.02254	0.09154	0.08635	0.21790	0.14470	0.03098	0.33880	0.04480	0.01437	0.00619	0.23770	---	---	1.14281
KARLSRUHE-2	0.63720	0.42180	0.02446	0.09273	0.08825	0.21780	0.14340	0.03120	0.32660	0.04273	0.00959	0.00734	0.23250	---	---	1.17046
SWEDEN	0.62640	0.42420	0.02210	0.08980	0.08020	0.21540	0.14480	0.02910	0.34830	0.04230	0.00990	0.00550	0.24820	0.03710	0.00330	1.11123
UKAEA	0.63863	0.41970	0.02443	0.09378	0.09078	0.21916	0.14285	0.03282	0.32302	0.04314	0.01413	0.00737	0.22619	0.02767	0.00255	1.17789
USSR	0.63860	0.41090	0.02130	0.10870	0.08780	0.21860	0.14000	0.03120	0.33210	0.04000	0.01730	0.00570	0.24390	0.01930	0.00380	1.15962
Mean	0.63270	0.41900	0.02350	0.09398	0.08643	0.21709	0.14283	0.03106	0.33508	0.04304	0.01300	0.00631	0.23615	0.03083	0.00365	1.14608
Std. Dev.	0.00528	0.00529	0.00192	0.00669	0.00323	0.00165	0.00192	0.00112	0.00780	0.00289	0.00251	0.00061	0.00719	0.00621	0.00075	0.02099

$k_{\text{Bsquared}}^a \equiv \text{Fission Production/Absorption.}$

- The Karlsruhe-2 solution, which is only slightly above the mean values of fission and fission production, has the smallest value (with UKAEA and Cadarache-1) of total capture. This relatively small value for capture is primarily due to small values for capture in ^{240}Pu and ^{238}U .
- The UKAEA solution, which has relatively large values for fission and fission production, has: the highest values (with JAERI-1) of fission and fission production in ^{238}U ; the smallest values of total capture (with Cadarache-1 and Karlsruhe-2) and capture in ^{238}U ; highest value of capture in ^{241}Pu ; and lowest value of capture in $\text{Na} + ^{16}\text{O}$. The difference in capture (relative to the mean value) is totally accounted for by the smaller capture value in ^{238}U .
- The Sweden and Belgium solutions, which have the smallest values of k_{B}^2 , k_{∞} and k_{eff} , also have: the smallest fission and fission production values for ^{238}U and total for the inner core; and highest values for capture in ^{238}U and total for the inner core. Additionally, the Belgium solution has the smallest values of fission in ^{239}Pu and ^{240}Pu and capture in ^{239}Pu .
- The JAERI-2 solution, which has the smallest breeding ratio, has the second smallest value (next to UKAEA) of capture in ^{238}U .
- The CNEN solution has the largest values of fission and fission production in ^{240}Pu .
- The EIR-1 solution has the largest value (35.5% above the mean) of capture in structure. It is also interesting to note that all solutions using the ENDF/B data base have large values (relative to the mean) of capture in structure. (Further observations about groups of solutions are discussed in Section V of this report.)

B. Central Spectrum Related Quantities

1. Central Real and Adjoint Spectra: The central real flux spectrum reported by each participant has been previously discussed in Section II. These results are shown in Figs. 5-7. The central flux centroids have been compared in Table V. As discussed previously, the real flux spectrum of the JAERI-3 solution is clearly the softest; whereas the spectra of USSR and Cadarache-1 are clearly the hardest. There is considerable variation among all solutions in the 10-300 keV energy range.

Participants also reported the central adjoint flux spectrum for Configuration 1. These spectra are shown in Figs. 32-34. The normalization is such that the average adjoint over the energy range from 20 eV to 1 MeV is equal to one. There is a great deal of variation among the various solutions for the adjoint flux spectrum. Four solutions (Belgium, Karlsruhe-1 and -2, and USSR) have a much steeper slope at higher energies, i.e., above 500 keV. These same four solutions are much smaller below 20 eV. On the positive slope portion of the adjoint spectrum (i.e., above ~10 keV), the CNEN solution is the softest (i.e., has least positive slope). The next softest is the ANL solution which is very close to the Cadarache-1 and UKAEA solutions. On the negative slope portion of the adjoint spectrum (i.e., below ~10 keV), the ANL solution has the largest slope and the CNEN, EIR-1, and EIR-2 solutions have least slope.

2. Central Reaction Rate Ratios: Central reaction rate ratios for Configuration 1 are shown in Table XII. The 1 σ variations for $^{28}\text{C}/^{49}\text{f}$, $^{28}\text{f}/^{49}\text{f}$, and $^{49}\text{C}/^{49}\text{f}$ are 3.7%, 4.0%, and 5.8%, respectively. The ANL value for $^{28}\text{C}/^{49}\text{f}$ is 4.6% higher than the UKAEA value and 3.8% higher than the CADARACHE-1 value. The Cadarache-2 value for $^{28}\text{C}/^{49}\text{f}$ is 1.1% higher than the UKAEA value. As noted previously, the $^{28}\text{C}/^{49}\text{f}$ values correlate well with values of k_{eff} , k_{∞} , breeding ratio, and $\langle^{28}\sigma_{\text{C}}\rangle$.

ADJOINT BROAD GROUP SPECTRA AT CORE CENTER

Fig. 32.

ADJOINT BROAD GROUP SPECTRA AT CORE CENTER

Fig. 33.

ADJOINT BROAD GROUP SPECTRA AT CORE CENTER

Fig. 34.

TABLE XII. Central Reaction Rate Ratios for Configuration 1

Solution	Reaction Rate Ratios ^a		
	$^{28}\text{c}/^{49}\text{f}$	$^{28}\text{f}/^{49}\text{f}$	$^{49}\text{c}/^{49}\text{f}$
Mean	0.1659	0.02216	0.2997
Std. Deviation	0.0062	0.00088	0.0174
Percent Difference Relative to the Mean			
ANL	0.37	-1.43	2.80
BELGIUM	9.71	-0.80	-17.22
CADARACHE-1	-3.43	-2.34	2.84
CADARACHE-2	-2.34	-1.12	4.94
CNEN	-1.38	3.03	1.38
EIR-1	-2.11	-3.85	2.75
EIR-2	1.00	-2.46	0.65
HEDL	-	-	-
JAERI-1	-0.36	6.92	7.80
JAERI-2	-3.05	-0.22	-0.56
JAERI-3	-1.50	5.03	3.58
KARLSRUHE-1	-0.61	-1.26	1.81
KARLSRUHE-2	-1.06	-2.92	-0.65
SWEDEN	3.69	-7.39	-3.33
UKAEA	-4.24	5.92	-0.71
USSR	5.31	2.90	-6.07

^aPer atom reaction rate ratios are given, and c and f refer to capture and fission rates, respectively.

3. Central Material Reactivity Worths: The calculated central isotopic reactivity worth for Configurations 1 and 3 are shown in Table XIII. The agreement is remarkably good. The 1σ variations are ~3-5% for ^{239}Pu and ^{238}U ; ~10-13% for Na; ~5-8% for Fe and ^{10}B . The larger variation in the small (scattering) worth of Na is not surprising. For the central worth of ^{239}Pu in Configuration 1, the largest values are obtained by CADARACHE-1 and -2, UKAEA, and KARLSRUHE-1. These solutions, except for KARLSRUHE-1, are derived from adjusted data sets.

Typical C/E values for fissile isotope central reactivity worths in critical assemblies are 1.15-1.20 for ENDF/B-IV calculations and ~1.0 for calculations using the adjusted sets (e.g., CARNAVAL-III and FGL5). These differences could be due to differences in the calculated central worths ($\delta k/k$ units), differences in the β_{eff} 's used in the data reduction or other differences in the experiments. When corrected for differences in the radial flux distributions (discussed below), the calculated central worths for ENDF/B-IV and the adjusted sets are about the same. The β_{eff} values differ by 5.5-7.0%, which leaves about 10% of the central worth discrepancy due to either experimental problems or unaccounted for in this exercise. It is unlikely that a difference as large as 10% can be attributed to the experimental techniques.

The comparison of calculated central worths in the sodium-voided Configuration 3 and the sodium-filled Configuration 1 is interesting. The ratio of these calculated values (Configuration 3/Configuration 1) for each solution is given in Table XIV. It may be noted that the values of ρ^{49} for the Cadarache-1 and -2 solutions, which are the largest in the sodium-in configuration, are reduced and very near (within 1% of) the mean value in the sodium-voided configuration. Similarly, the values of ρ^{28} increase in the

TABLE XIII. Central Isotopic Reactivity Worths ($10^{-6} \Delta k/k$ per 10^{24} atoms/cc)

Solution	Configuration 1 (Na In)					Configuration 3 (Na Void)				
	^{239}Pu	^{238}U	Na	Fe	^{10}B	^{239}Pu	^{238}U	Na	Fe	^{10}B
Mean	118.7	-7.31	-0.69	-0.89	-78.2	133.5	-7.57	-0.93	-1.06	-74.2
Std. Deviation	4.4	0.35	0.07	0.07	4.3	3.5	0.21	0.12	0.05	3.3
Percent Difference Relative to the Mean										
ANL	-0.9	-3.8	1.4	-2.1	-5.0	0.5	-1.5	4.9	1.6	-2.8
BELGIUM ^a	-5.9	4.7	-20.5	13.5	3.0	-4.3	5.5	2.2	-0.2	-3.2
CADARACHE-1 ^a	4.2	4.5	11.5	-11.7	5.3	-0.6	-0.3	11.8	-11.1	2.0
CADARACHE-2 ^a	6.0	5.8	2.9	-5.5	6.0	0.7	0.4	-1.3	-5.8	2.0
CNEN	-0.5	6.6	2.8	0.1	-5.8	4.3	1.8	8.6	3.4	-0.1
EIR-1 ^a	-4.7	-6.4	-13.7	10.1	-6.1	-1.2	-0.6	-25.8	7.2	-0.2
EIR-2 ^a	-3.3	-0.1	-10.7	-3.8	-7.1	-0.4	3.8	-5.8	0.4	-3.6
JAERI-1	-4.0	-4.3	-1.0	-2.6	-6.1	-1.9	-0.8	3.6	1.6	-3.4
JAERI-2	0.4	-5.0	5.7	-0.9	0.3	0.5	-3.7	15.0	1.4	1.7
JAERI-3	-1.5	-1.2	2.9	-5.9	10.0	-1.1	0.9	7.6	0.4	13.0
KARLSRUHE-1 ^a	4.3	-2.2	15.3	5.4	-0.0	2.3	0.3	18.2	0.8	-1.2
KARLSRUHE-2 ^a	0.2	-5.9	8.6	10.5	4.0	-2.6	-5.9	-11.4	5.8	-0.2
SWEDEN	-	-	-	-	-	-	-	-	-	-
UKAEA	4.7	2.1	-2.8	4.1	2.9	5.5	-0.8	-8.1	1.5	1.4
USSR	1.1	5.2	-2.4	-11.2	-1.4	-1.7	0.8	-19.6	-7.0	-5.4

^aThese values are not shown as submitted, but have been divided by a factor of 2.0.

TABLE XIV. Ratio of Calculated Central Worths in the Na-Voided Configuration 3 and the Na-In Configuration 1

Solution	Ratio of Calculated Central Worths in $\frac{\text{Configuration 3}}{\text{Configuration 1}}$				
	^{239}Pu	^{238}U	Na	Fe	^{10}B
ANL	1.141	1.060	1.393	1.242	0.970
BELGIUM	1.142	1.043	1.731	1.051	0.891
CADARACHE-1	1.074	0.988	1.351	1.204	0.919
CADARACHE-2	1.068	0.983	1.292	1.192	0.913
CNEN	1.179	0.988	1.423	1.236	1.005
EIR-1	1.165	1.099	1.158	1.164	1.008
EIR-2	1.158	1.075	1.421	1.247	0.985
JAERI-1	1.149	1.073	1.409	1.247	0.976
JAERI-2	1.125	1.049	1.465	1.223	0.962
JAERI-3	1.130	1.057	1.407	1.276	0.974
KARLSRUHE-1	1.103	1.062	1.381	1.143	0.937
KARLSRUHE-2	1.093	1.034	1.099	1.144	0.910
SWEDEN	-	-	-	-	-
UKAEA	1.133	1.006	1.274	1.165	0.934
USSR	1.092	0.992	1.110	1.252	0.910
Mean	1.125	1.036	1.351	1.199	0.950
Std. Deviation	0.033	0.037	0.157	0.058	0.037

sodium-voided configuration (relative to the reference sodium-in configuration) for all solutions except Cadarache-1 and -2, CNEN, and USSR (which decrease ~1%); and the values of ρ^{B10} decrease (~5%) in the sodium-voided configuration for all solutions except CNEN and EIR-1 (which increase <1%). Edits of the central (first mesh) real and adjoint flux spectra in Configuration 3 would be useful in comparing the calculated central worths in that configuration and should be added to the requested edits of the problem specifications.

C. Spatial Effects

1. Radial and Axial Worth Profiles: One of the more surprising results is the generally poor agreement for the radial worth profiles (e.g., see that of ρ^{B10} in Figs. 35-37). The same is not true for the axial worth profiles, which generally are in much better agreement (e.g., see that of ρ^{B10} in Figs. 38-40). The problem specifications requested edits (D.13-14) of: axial worth profiles of ^{239}Pu , ^{238}U , and Fe; radial and axial worth profiles of ^{10}B and Na; and leakage and non-leakage components of the reactivity worth of sodium. These results are displayed as plots in Appendix C. The following observations may be made.

- The axial worth profiles in Configuration 1 are in good agreement (within ~5-7%), especially for ^{239}Pu , ^{238}U , and ^{10}B . The USSR solution is generally high, especially near the core/blanket interface. The Belgium and UKAEA solutions are slightly low on the non-leakage and total worth of sodium, especially near the core/blanket interface. Although the axial differences are not large, the UKAEA axial worth profiles for ^{239}Pu , ^{238}U , Fe, and ^{10}B fall off more slowly axially and (as discussed below) more rapidly radially through the core regions.

RADIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. 35,
RADIAL DISTANCE (cm)

RADIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. 36.

RADIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. 37.

AXIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. 38.

AXIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. 39.

AXIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. 40.

- The axial worth profiles in Configuration 3 are very similar to the axial profiles in Configuration 1. The EIR-1 solution is low near the core/blanket interface on the leakage and total worth profiles of sodium. The Belgium solution has some structure in the blanket region of the axial worth profiles of the sodium leakage and non-leakage worths.
- The radial worth profiles in Configuration 1 (with variations of ~15-20%) are in much poorer agreement than the axial profiles. The Cadarache-1 and -2 and UKAEA solutions fall off much faster radially through the inner core region. The Belgium solution is rather low near the outer core/radial blanket interface in the radial profiles of leakage and total worth of sodium.
- The radial worth profiles in Configuration 3 are very similar to the radial profiles in Configuration 1. The agreement is perhaps slightly improved, though the considerable differences in curvature through the inner core region remain, with the Cadarache-1 and -2 and UKAEA solutions being the most buckled.

2. Radial Power Density Profiles: Another of the more surprising results of this comparison study is the large spread in the radial power density distributions as shown in Figs. 41-43. The Cadarache-1, UKAEA, and USSR* solutions show the largest inner core bucklings (i.e., fall off faster in the inner core) and the Sweden solution has the smallest inner core buckling.

It is apparent that differing radial distributions affect, by several percent, both the inner core to outer core power split and center to core-average flux and adjoint ratios. These latter ratios, of course, affect the central reactivity worths - bringing the values of Cadarache, UKAEA, and USSR

*It appears the USSR solution has been incorrectly reported as fission rate per ^{239}Pu atom.

RADIAL FISSION RATE DISTRIBUTION FOR CONFIGURATION 1.

Fig. 41.

RADIAL FISSION RATE DISTRIBUTION FOR CONFIGURATION 1.

Fig. 42.

RADIAL FISSION RATE DISTRIBUTION FOR CONFIGURATION 1.

Fig. 43.

up relative to those of the other participants. The effect is about 10% due to 5% effects for both the flux and adjoint. The central control rod is similarly affected. It is also interesting to note that the Cadarache-1 and UKAEA solutions have the largest k_{eff} values while the Swedish solution has the smallest k_{eff} . The functional form of the relationship (complex for a two-zone core) between the radial distributions and k_{eff} is not known.

Because of the relative consistency in the axial profiles, it is concluded that the discrepancies in the radial distributions are not due simply to diffusion coefficient effects resulting from differences in the transport cross sections. In fact, it appears that the cause of the differing radial distributions is related both to the differing inner core material bucklings (or k_{∞}) and to differing k_{∞} increments produced by the inner core to outer core enrichment change. Given the inner core k_{∞} and the core height, the inner core radial curvature is determined. The specified enrichment change produces a k_{∞} increment which depends on the cross sections (which differ between participants); and the k_{∞} split between inner and outer cores determines the radial power profile in the outer core relative to the inner core. The outer core to inner core k_{B2} ratio* is largest for the Swedish solution and smallest for the Cadarache-1 solution which correlates well with the fact that the Swedish solution displays the flattest radial power profile while the Cadarache-1 solution is among the steepest. This correlation between k_{B2} split and inner core/outer core power split was observed for most participants' solutions.

*While the k_{∞} values for the outer core were not calculated, a related parameter can be calculated from the neutron balance data. Called k_{B2} it is the ratio of fission production to absorption for the critically-buckled spectrum.

RADIAL FISSION RATE DISTRIBUTION FOR ANL

Fig. 44.

3. Radial Power Density Profile With and Without Control Rod Inserted:

The radial power density profiles for Configurations 1 (reference sodium-filled), 3 (sodium-voided), 5 (reference with a central boron control rod inserted), and 6 (sodium-voided with a central boron control rod inserted) of the ANL solution are shown in Fig. 44. These edits were not requested in the original problem specifications though, in retrospect, they should be added.

As shown in Fig. 44, the central control rod causes a large perturbation in the radial power profile. The size of the radial flux shift induced by control rod insertion almost certainly depends on the inner core to outer core k_{∞} ratio. Thus, it is expected that the differing inner core to outer core k_{∞} ratios among the participants are contributing to the large observed variations in control rod worth. Also, as shown in Fig. 44, a radial power shift occurs upon sodium voiding. The size of this shift is expected to depend on the inner core to outer core k_{∞} ratio. Further analysis is needed to quantitatively understand these effects. The apparent sensitivity of the radial distributions in large two-zone cores is an interesting and potentially important problem.

One additional radial power distribution was submitted with the Cadarache-2 solution. It is for Configuration 5 (reference with a central boron control rod inserted). For comparison this radial power density is shown in Fig. 45 with the Cadarache-1 solution for Configuration 1 and the ANL solution for Configurations 1 and 5.

RADIAL FISSION RATE DISTRIBUTION FOR ANL AND CADARACHE

Fig. 45.

V. COMPARISON OF RESULTS BY GROUPING OF PARTICIPANTS

The comparison calculation has provided an opportunity to intercompare predictions based on non-adjusted and adjusted data sets. To facilitate such a comparison, the results for one group effective cross sections and for several key integral parameters have been organized into the categories:

- All Solutions
- Adjusted-Data-Set Solutions Only
- Non-Adjusted Data Set Solutions Only
- ENDF/B-IV Data Set Solutions Only

For each of the categories, the mean value and the standard deviation are computed. The results of such a grouping for one group effective cross sections was discussed in Section II (see Table VIII). Certain of the key integral parameters have been organized into categories by basic data type. The means and the standard deviations are displayed in Table XV.

As regard to sizes of variations in the integral parameters, inspection of Table XV leads to the following conclusions:

- The users of adjusted sets generally achieve a smaller variation in integral parameters than do the users of non-adjusted sets.

Parameter	Variation among users of Adjusted Sets	Variation among users of Non-adjusted Sets
k_{∞}	± 0.01	± 0.02
k_{eff}	± 0.005	± 0.010
$^{28}C/^{49}f$	$\pm 1.2\%$	$\pm 3.8\%$
CR(IC)	$\pm 1.3\%$	$\pm 4.3\%$
BR(Config. 1)	$\pm 1.1\%$	± 3.7

- For $^{28}f/^{49}f$, ρ^{49} , ρ^{28} and ρ^{B10} , however, the variations are of about the same size. (The large variations in the central worths, as discussed above, are due in part to the different radial flux profiles.)

TABLE XV. Integral Parameters for Configuration 1 by Basic Data File Type

	k_{∞} (IC)	k_{B^2} (Core)	k_{eff}	Core Leakage
<u>All Sets</u>				
Mean	1.12428	1.2042	1.00509	0.1621
Std. Dev. *100%/Mean	1.83	1.64	1.29	2.47
(Max-Mean)*100%/Mean	+2.50 CADARACHE-1	+2.45 UKAEA	+1.88 KARLSRUHE-2	+4.75 CNEN
(Min-Mean)*100%/Mean	-3.33 SWEDEN	-2.76 SWEDEN	-2.45 SWEDEN	-4.44 JAERI-1
(Max-Min) *100%/Mean	5.84	5.21	4.33	9.19
<u>Adjusted Sets</u>				
Mean	1.14392	1.2227	1.01778	0.1617
Std. Dev. *100%/Mean	0.83	0.72	0.52	2.23
(Max-Mean)*100%/Mean	+0.74 CADARACHE-1	+0.90 UKAEA	+0.61 KARLSRUHE-2	+2.16 UKAEA
(Min-Mean)*100%/Mean	-1.05 JAERI-3	-1.08 JAERI-3	-0.57 JAERI-2	-2.63 JAERI-3
(Max-Min) *100%/Mean	1.79	1.98	1.18	4.79
<u>Non-Adjusted Sets</u>				
Mean	1.11445	1.1919	0.99748	0.1624
Std. Dev. *100%/Mean	1.53	1.17	0.94	2.71
(Max-Mean)*100%/Mean	+2.79 CNEN	+1.94 USSR	+1.10 USSR	+4.55 CNEN
(Min-Mean)*100%/Mean	-2.48 SWEDEN	-1.75 SWEDEN	-1.70 SWEDEN	-4.62 JAERI-1
(Max-Min) (100%/Mean	5.27	3.69	2.81	9.17
<u>ENDF/B-IV Sets</u>				
Mean	1.11930	1.1925	0.99784	0.1646
Std. Dev. *100%/Mean	1.65	0.82	0.47	3.28
(Max-Mean)*100%/Mean	+2.35 CNEN	+0.91 CNEN	+0.43 HEDL	+3.16 CNEN
(Min-Mean)*100%/Mean	-1.33 EIR-1	-0.71 EIR-1	-0.48 ANL	-3.41 EIR-1
(Max-Min) *100%/Mean	3.68	1.63	0.91	6.57

Percent Differences Relative to the Mean of the Adjusted Sets				
Adjusted Sets	---	---	---	---
Non-Adjusted Sets	-2.58	-2.52	-1.99	+0.43
ENDF/B-IV Sets	-2.15	-2.47	-1.96	+1.79

TABLE XV. (Cont'd.)

	c^{28}/f^{49}	f^{28}/f^{49}	CR(IC)	Breeding Ratio
<u>All Sets</u>				
Mean	0.16589	0.02216	1.09262	1.39211
Std. Dev. *100%/Mean	3.71	3.97	4.25	3.46
(Max-Mean)*100%/Mean	+9.71 BELGIUM	+6.90 JAERI-1	+11.66 BELGIUM	+9.19 BELGIUM
(Min-Mean)*100%/Mean	-4.24 UKAEA	-7.40 SWEDEN	-4.41 CADARACHE-1	-3.19 EIR-1
(Max-Min) *100%/Mean	13.96	14.31	16.07	12.38
<u>Adjusted Sets</u>				
Mean	0.16157	0.02232	1.05985	1.36171
Std. Dev. *100%/Mean	1.23	3.76	1.26	1.13
(Max-Mean)*100%/Mean	+1.59 KARLSRUHE-2	+5.16 UKAEA	+2.09 CADARACHE-2	+2.27 CADARACHE-2
(Min-Mean)*100%/Mean	-1.68 UKAEA	-3.62 KARLSRUHE-2	-1.46 CADARACHE-1	-0.73 JAERI-3
(Max-Min) *100%/Mean	3.27	8.78	3.55	3.00
<u>Non-Adjusted Sets</u>				
Mean	0.16877	0.02205	1.11446	1.41035
Std. Dev. *100%/Mean	3.76	4.26	4.31	3.72
(Max-Mean)*100%/Mean	+7.84 BELGIUM	+7.44 JAERI-1	+9.47 BELGIUM	+7.78 BELGIUM
(Min-Mean)*100%/Mean	-3.78 EIR-1	-6.94 SWEDEN	-4.18 EIR-1	-4.44 EIR-1
(Max-Min) *100%/Mean	11.62	14.38	13.65	12.22
<u>ENDF/B-IV Sets</u>				
Mean	0.16416	0.02199	1.08228	1.37907
Std. Dev. *100%/Mean	1.29	3.55	1.22	1.54
(Max-Mean)*100%/Mean	+1.43 ANL	3.81 CNEN	+1.07 ANL	+1.14 ANL
(Min-Mean)*100%/Mean	-1.08 EIR-1	-3.13 EIR-1	-1.33 EIR-1	-2.27 EIR-1
(Max-Min) *100%/Mean	2.50	6.94	2.40	3.41

Percent Differences Relative to the Mean of the Adjusted Sets				
Adjusted Sets	---	---	---	---
Non-Adjusted Sets	+4.46	-1.21	+5.15	+3.58
ENDF/B-IV Sets	+1.61	-1.48	+2.11	+1.28

TABLE XV. (Cont'd.)

	ρ^{49}	ρ^{28}	ρ^{B10}	ρ^{Fe}
<u>All Sets</u>				
Mean	1.187(-4)	-7.314(-6)	-7.824(-5)	-8.894(-7)
Std. Dev. *100%/Mean	3.73	4.73	5.43	7.83
(Max-Mean)*100%/Mean	+5.96 CADARACHE-2	+6.65 CNEN	+10.03 JAERI-3	+13.52 BELGIUM
(Min-Mean)*100%/Mean	-5.86 BELGIUM	-6.41 EIR-1	-7.14 EIR-2	-11.74 CADARACHE-1
(Max-Min) *100%/Mean	11.83	13.06	17.17	25.26
<u>Adjusted Sets</u>				
Mean	1.215(-4)	-7.316(-6)	-8.195(-5)	-8.755(-7)
Std. Dev. *100%/Mean	2.96	4.86	3.13	8.09
(Max-Mean)*100%/Mean	3.57 CADARACHE-2	+5.72 CADARACHE-2	+5.03 JAERI-3	+12.30 KARLSRUHE-2
(Min-Mean)*100%/Mean	-3.76 JAERI-3	-5.90 KARLSRUHE-2	-4.28 JAERI-2	-10.33 CADARACHE-1
(Max-Min) (100%/Mean	7.33	11.63	9.31	22.64
<u>Non-Adjusted Sets</u>				
Mean	1.167(-4)	-7.312(-6)	-7.545(-5)	-8.999(-7)
Std. Dev. *100%/Mean	3.42	4.96	3.78	7.97
(Max-Mean)*100%/Mean	+6.10 KARLSRUHE-1	+6.68 CNEN	+6.83 BELGIUM	+12.21 BELGIUM
(Min-Mean)*100%/Mean	-4.20 BELGIUM	-6.39 EIR-1	-3.71 EIR-2	-12.21 USSR
(Max-Min) *100%/Mean	10.30	13.06	10.54	24.41
<u>ENDF/B-IV Sets</u>				
Mean	1.163(-4)	-7.228(-6)	-7.383(-5)	-9.134(-7)
Std. Dev. *100%/Mean	2.32	6.99	0.62	6.36
(Max-Mean)*100%/Mean	+1.55 CNEN	+7.92 CNEN	+0.69 ANL	+7.24 EIR-1
(Min-Mean)*100%/Mean	-2.67 EIR-1	-5.29 EIR-1	-0.51 EIR-1	-4.68 ANL
(Max-Min) *100%/Mean	4.21	13.21	1.20	11.92

Percent Differences Relative to the Mean of the Adjusted Sets				
Adjusted Sets	---	---	---	---
Non-Adjusted Sets	-3.95	-0.05	-7.93	+2.79
ENDF/B-IV Sets	-4.28	-1.20	-9.91	+4.33

- The variation in integral parameters among participants using the same (ENDF/B-IV) basic data set but different processing codes is, as a rule, as large as the variation among all participants. Thus, variations due to processing alone introduce substantial variation in integral parameter prediction.

Tables VIII and XV display, in the bottom two rows, the percent deviations of the mean values of the non-adjusted and the ENDF/B-IV sets from the corresponding mean values for the adjusted sets. An examination of these data can be made to observe correlations between trends in effective cross section values and the resulting trends in integral parameters. Study of Tables VIII and XV leads to the following conclusions.

- For heavy element cross sections:

• for $\langle \bar{v} \rangle$	Adjusted Sets	0.1% > Non-Adjusted Sets
	" "	0.3% > ENDF/B-IV
• for $\langle \sigma_f^{49} \rangle$	Adjusted Sets	0.9% < Non-Adjusted Sets
	" "	1.2% < ENDF/B-IV
• for $\langle \sigma_c^{28} \rangle$	Adjusted Sets	4.5% < Non-Adjusted Sets
	" "	2.2% < ENDF/B-IV
• for $\langle \alpha^{49} \rangle$	Adjusted Sets	2.3% > Non-Adjusted Sets
	" "	1.3% < ENDF/B-IV
• for $\langle \sigma_f^{28} \rangle$	Adjusted Sets	0.3% < Non-Adjusted Sets
	" "	1.7% < ENDF/B-IV

- For light element cross sections:

• for $\langle \sigma_{tr}^{Fe} \rangle$	Adjusted Sets	11.7% < Non-Adjusted Sets
	" "	17.2% < ENDF/B-IV
• for $\langle \sigma_{tr}^{B^{10}} \rangle$	Adjusted Sets	2.2% > Non-Adjusted Sets
	" "	2.4% > ENDF/B-IV
• for $\langle \sigma_{tr}^{Na} \rangle$	Adjusted Sets	0.8% < Non-Adjusted Sets
	" "	0.6% > ENDF/B-IV
• for $\langle \sigma_{tr}^{O^{16}} \rangle$	Adjusted Sets	1.7% < Non-Adjusted Sets
	" "	2.6% < ENDF/B-IV

Based on these trends one might expect that, relative to non-adjusted and ENDF/B-IV sets, the adjusted sets would:

- underpredict conversion ratio, breeding ratio, c^{28}/f^{49} , and ρ^{28} ,
[smaller $\langle\sigma_c^{28}\rangle$ and similar $\langle\sigma_f^{49}\rangle$ and $(1+\langle\alpha^{49}\rangle)$]
- overpredict k_∞ and f^{28}/f^{49} ,
[smaller $\langle\sigma_c^{28}\rangle$ and small differences in $\langle\sigma_f^{28}\rangle$ and $\langle\sigma_f^{49}\rangle$]
- overpredict core leakage,
[smaller $\langle\sigma_{tr}\rangle$ for light elements and smaller $\langle\sigma_c^{28}\rangle$]
- produce about the same result for k_{eff} , and
[cancellation of items above for k_∞ and core leakage]
- produce about the same result for ρ^{49} and ρ^{B10} .
[small differences in $\langle\sigma_f^{49}\rangle$, $(1+\langle\alpha^{49}\rangle)$, and $\langle\sigma_{abs}^{B10}\rangle$]

Examination of Table XV shows that these expectations are generally met for:

- CR(IC)	Adjusted Sets	5.2% < Non-Adjusted Sets
	" "	2.1% < ENDF/B-IV
- Breeding Ratio	Adjusted Sets	3.6% < Non-Adjusted Sets
	" "	1.3% < ENDF/B-IV
- c^{28}/f^{49}	Adjusted Sets	4.5% < Non-Adjusted Sets
	" "	1.6% < ENDF/B-IV
- k_∞ (IC)	Adjusted Sets	2.6% > Non-Adjusted Sets
	" "	2.2% > ENDF/B-IV
- f^{28}/f^{49}	Adjusted Sets	1.2% > Non-Adjusted Sets
	" "	1.5% > ENDF/B-IV

However, some surprises occurred:

- ρ^{49}	Adjusted Sets	4.0% > Non-Adjusted Sets
	" "	4.3% > ENDF/B-IV
- ρ^{28}	Adjusted Sets	0.1% > Non-Adjusted Sets
	" "	1.2% > ENDF/B-IV
- ρ^{B10}	Adjusted Sets	7.9% > Non-Adjusted Sets
	" "	9.9% > ENDF/B-IV

Little or no difference was expected for ρ^{49} ; a small difference (less than 8-10%) was expected for ρ^{B10} ; and ρ^{28} was expected to change (decrease) much more than observed.

- Core Leakage	Adjusted Sets	0.4% < Non-Adjusted Sets
	" "	1.8% < ENDF/B-IV

The opposite difference was expected.

- k_{eff}	Adjusted Sets	2.0% > Non-Adjusted Sets
	" "	2.0% > ENDF/B-IV

A small difference was expected due to cancellation of k_{∞} and leakage effects; however, the expected change in leakage did not occur and the differences in k_{eff} are significant.

To summarize, k_{∞} , CR(IC), BR, c^{28}/f^{49} , and f^{28}/f^{49} respond as expected to the facts that $\langle \sigma_c^{28} \rangle$ and $\langle \sigma_{tr}^{light} \rangle$ are smaller for the adjusted sets and $\langle \bar{v} \rangle$, $\langle \sigma_f^{49} \rangle$, $\langle \sigma_f^{28} \rangle$, and $(1 + \langle \alpha^{49} \rangle)$ are generally the same. However, the core leakage behaves opposite to expectation, the changes in ρ^{49} and ρ^{B10} are unexpectedly large, and the change in ρ^{28} is unexpectedly small.

VI. SUMMARY OF MAJOR DISCREPANCIES

The afternoon working session of the second day of the Specialists Meeting on these NEACRP International Comparison Calculations was devoted to discussion by participants of three of the major discrepancies resulting from this study. The discrepancies considered by the participants to be most significant were: (A) Radial Distributions, (B) Central Worth Discrepancy, and (C) Central Control Rod Worth. The discussion of these discrepancies was recorded, summarized and reviewed on the following (final) day of the specialists meeting. Item (A) was summarized by D. C. Wade (ANL); items (B) and (C) were summarized by P. J. Collins (ANL). These summaries reflect the discussion at the meeting and are presented below with only editorial changes which resulted due to addition or revision of solutions by participants.

A. Radial Distributions

1. Observations: While a high degree of consistency was observed among the contributed axial worth profiles,* the radial worth and power density distributions displayed a large and unexpected range of variation.

In the radial distributions,

- the Swedish solution was flattest in the inner core, while
- the Cadarache-1 (Carnaval III), UKAEA, and USSR results fell off more rapidly (vs. radius) in the inner core than did the other solutions.

The radial worth profiles were less dispersed for the sodium-voided Configuration 3 than they were for the sodium-filled Configuration 1.

*As noted in Section IV, this statement is correct only in a relative sense. Subsequent review of the axial worth profiles indicates variations of ~5-7% among participants.

The "corner" in the distributions at the inner core/outer core interface observed in the sodium-filled core was made less severe by sodium voiding.

2. Impact of Differing Radial Profiles: The differing radial power density profiles affected the inner core/outer core power split -- variations of ~10% were observed between the minimum (Cadache-1) and maximum (Sweden) values of the outer core to inner core power ratio.

The differing profiles also were indicative of differing ratios of core-center to core-average flux ratio and adjoint ratio.

These ratios, of course, affect the central reactivity worth values -- bringing the values of Cadache-1, UKAEA, and USSR up relative to those of the other participants. This effect is estimated to be ~10% -- 5% for each of the flux and adjoint.

The same can be said of central control rod worth.

3. Possible Causes for the Differing Radial Profiles: The macroscopic transport cross section influences the curvature of the flux and adjoint distributions; and the macroscopic transport cross sections of light elements are known to be discrepant between participants' solutions.

However, the agreement observed for the axial profiles rules out transport cross section variations as a principal cause of the observed radial profile differences.

It appears that the variation in radial profiles is caused in large measure by variations in the inner-core-to-outer-core k_{∞} difference (or ratio).*

The k_{∞} change which accompanies the specified enrichment change between the inner and outer core depends on the cross sections used by the participants, and a variation in k_{∞} change is, in fact, observed in the contributed solutions.

*Note also that the inner core material buckling or k_{∞} controls the profile in the interior of the inner core and that this correlation is seen with Sweden displaying a flat profile and low k_{∞} while UKAEA AND Cadache display a steep profile and high k_{∞} .

- For Sweden, the ratio of outer core to inner core k_{B2} is larger* than for all other participants, while
 - for Cadarache-1 the ratio is smaller than for all other participants.
 - This correlates with the corresponding Swedish and Cadarache-1 maximum and minimum outer core to inner core power ratios.
- The k_{eff} value may also enter in this argument.
- The k_{eff} values for Cadarache-1 and UKAEA are high relative to most other participants, while
 - the k_{eff} value of Sweden is lower than that of other participants.

An analysis is required to determine a physically reasonable functional form for the relationship between the ratio, ϕ outer core/ ϕ inner core, and the quantities k_{eff} , k_{∞} (inner core), and k_{∞} (outer core).

4. Relevant Experience with Critical Assembly Measurements: Predictions of radial reaction rate and reactivity worth profiles in critical experiments have been relatively good among participants who, in this model problem, predict different radial shapes.

- a. For the MZB core, (which had a thin outer core ring), the UKAEA predicted the plutonium fission rate (relative to the inner core value):

- . 4% low in the outer core using diffusion theory
- . 2% low in the outer core using transport theory.

Also a slight change vs. radius of the C/E on rod worth was observed.

* k_{B2} is (fission production)/(absorption) for the critically buckled spectrum.

- b. For clean (no control rods) cores, of 60 to 70 cm radius, CEA computed reaction rate distributions to within 1% of experiment at most points in the core more than 5 cm from the blanket. Also, calculations of material buckling agreed with measurements to within 1%.
- c. Both UKAEA and CEA include the material buckling measurement results in their data adjustment procedures.
- d. For ZPPR-4 (Phase I) the ANL average C/E values on f^{49} were:
 - 1.002 $\pm$ 0.012 for 55 inner core positions
 - 0.982 $\pm$ 0.013 for 47 outer core positions.
 For f^{25} the corresponding values were:
 - 1.028
 - 1.018
- e. For ZPPR-7 (radial parfait core) the C/E for reaction rates was 4-5% higher in the outer core region than in the inner core region. This was reduced by half if σ_c^{28} was lowered by 5%.
- f. The JAERI calculations of reaction rate distribution agree with experiment to within 2%. However, in the radial blanket, maximum deviations of up to 10% underprediction of reaction rates have been observed. The effort on distribution measurements at JAERI has not been stressed up to now.

5. Conclusions and Recommendations: The observed differences in radial distributions help to explain observed differences in central material worths and central boron rod worth.

It appears possible that the cause of the differing radial distributions is related to differing k_{∞} increments produced by the inner core to outer core enrichment change. These increments depend on the cross sections, which differ between participants.

The k_{eff} value may also contribute to the driving function for the inner-core-to-outer-core power ratio. A physically reasonable functional form for the relationship between k_{∞} , k_{eff} , and power ratio is needed.

Participants who predict different radial profiles for the NEACRP model each predict radial profiles in critical assemblies to good accuracy. This is a dilemma which is unresolved. However, it may be remarked that the critical experiments have been on cores smaller and perhaps with different enrichments than the NEACRP model problem. Comparisons of calculation with the larger-core BIZET and ZPPR-9 critical experiments which are currently in progress are of high interest.

B. Central Worth Discrepancy

For the central plutonium worth, the most obvious feature of the results was again the higher values, relative to ENDF/B and other sets, exhibited by the CARNAVAL-III (+5%) and FGL5 (+9%) data.

In comparison with experimental data, the CARNAVAL-III and FGL5 data give C/E's within a few percent of unity both for ^{239}Pu and ^{10}B . On the other hand, the ENDF/B-IV data gives values 15 to 20% higher than experiment for ^{239}Pu in LMFBR cores and C/E's for ^{10}B 5 to 10% lower than for plutonium.

The β_{eff} values for the CARNAVAL-III data are 5.5% higher than ENDF/B-IV and those for FGL5 are 7% higher. If the ENDF/B calculations used the higher β_{eff} values the C/E's would be reduced by 5.5 to 7.0%.

For the benchmark problem, these two adjusted data sets give results compared to ENDF/B-IV which are 15 to 20% higher than would have been expected using available C/E and β_{eff} information. About 10% of this difference would be consistent with the radial power distribution discrepancies as previously discussed.

It was noted that no C/E information was available for cores of the size of the benchmark problems.

C. Central B₄C Control Rod Worth

1. It was established that all calculations were made consistently and as specified with two exceptions.

- a. CNEN used a special cross section preparation for the control rod (i.e., collapsed σ_B over the control rod spectrum).

Also, a reduced mesh size was used inside the control rod.

- b. Sweden used the specified number of mesh points within the control rod but only half the specified number of mesh points for the rest of the reactor. Furthermore, their boron rod worth calculation was not converged.

2. The results were considered in conjunction with those for the central ^{10}B worth. However, when the control rod worths were adjusted for the respective differences in boron worth, the results were not made more consistent overall.

3. The Central Worth of ^{10}B : The results may be summarized as follows:

<u>Data Sets</u>	<u>Percent Difference Relative to Mean</u>
ENDF/B-III, -IV, JAERI-I	-5 to -7
KEDAK-2, KFKINR	+3 to +4
CARNAVAL-III, IV	+5 to +6
JAERI-2, KEDAK-3	0
JAERI-3	+10
FGL5	+13

It was noted that the values for the CARNAVAL-III and FGL5 data sets were about 10% higher than those with ENDF/B data. This difference would be consistent with the differences in the central/average fission rate distributions if a similar behavior were hypothesized for the adjoint fluxes.

4. The Central Control Rod: The results may be summarized as follows:

<u>Data Sets</u>	<u>Percent Difference in Rod Worth Relative to Mean</u>		
	<u>Na In, Relative to fuel</u>	<u>Na In, Relative to CRP</u>	<u>Na Voided, Relative to fuel</u>
ENDF/B-III, IV, JAERI-1, 2, 3	-5 to -13	-6 to -16	-3 to -13
CARNAVAL-III	+20	+21	+19
CARNAVAL-IV	+8	+20	+18
KEDAK-2	-3	-2	-3
KEDAK-3	-1	-3	-1
FGL5, KFKINR	+9	+9	+8

The most striking difference is between results using CARNAVAL and FGL5 data and the others. These data sets gave rod worths approximately 20-30% higher than the others. The central ^{10}B results would account for about 10% of these differences. Results in the sodium voided environment were close but showed the same behavior. It was noted that there was a large difference between CARNAVAL-III and CARNAVAL-IV results when relative to fuel which was not present when relative to the CRP.

5. Discussion: The large differences in control rod worth are connected with differences in calculated power distributions. Perturbation in flux caused by insertion of the rod could amplify the differences seen in the reference core.

Several contributors agreed to supply power distributions for the model with the control rod inserted for comparison. Some other points arising from the discussion were as follows:

- a. It was noted that there was a high steel content in the rod and the effect of this should be examined.
- b. It was assumed that all calculations were adequately converged.
- c. For possible future comparisons other interesting control rod cases might be for
 - i. rods with boron loadings intermediate between those that can be adequately calculated by FOP (central sample) and the heavily loaded rod, and
 - ii. rods distributed off-center.
- d. For the central $^{10}\text{B}/^{239}\text{Pu}$ worth ratio the ENDF/B-IV and FGL5 results were in close agreement whereas the CARNAVAL-III and -IV results were 2 to 3% higher and the JAERI-3 result was 14% higher.

VII. SENSITIVITY ANALYSIS

As a follow-on exercise to understand some of the questions arising from these comparison calculations, ANL has performed a cross section sensitivity study³ for two-zone conventional cores which included ZPPR-2, ZPPR-9, and the NEACRP benchmark model. In a one-dimensional analysis, the differences in k_{eff} , power shape, and central control rod worth have been predicted by assessing the sensitivity of the integral parameters to the nuclear data used. These predicted integral parameter differences were obtained simply from the differences reported herein in the effective one-group cross sections.

A. Introduction

The results of these NEACRP/IAEA international comparison calculations for a large LMFBR benchmark model show surprisingly large discrepancies for several important integral parameters. Since the calculational model was precisely defined, the source of these differences should lie in the multi-group data used by each participant. The differences in predicted parameters were not supported by evidence from integral measurements used by participants in testing their data sets. However, all of the integral information was obtained on cores much smaller than the 1250 MWe reactor being studied.

As an aid in the resolution of these discrepancies, and as an aid in extrapolating integral measurements to larger reactors, sensitivity coefficients for three sizes of conventional cores have been calculated. The first, ZPPR-2, is representative of a reactor of about 350 MWe and typical of several subsequent ZPPR assemblies (ZPPR-3 to ZPPR-6) in which integral data were obtained for the CRBR design. The second core, ZPPR-9, is currently being investigated to study predictions in a core of 700 MWe size. The third case is the 1250 MWe NEACRP benchmark itself. The fissile Pu enrichments for the two core zones of these reactors are given below:

$$\text{Enrichment} \equiv \frac{{}^{239}\text{Pu} + {}^{241}\text{Pu}}{\text{Total U} + \text{Pu}}$$

	Inner Core	Outer Core
ZPPR 2	0.131	0.194
ZPPR-9	0.097	0.142
1250 MWe Core	0.101	0.129

The two parameters which stood out most clearly in the comparisons were the radial power distribution and the central control rod worth; differences in the central reactivity worth appeared to stem largely from the radial power shape. The analysis has concentrated on these two parameters. In addition, sensitivities for k_{eff} were naturally obtained in the process.

The sensitivity studies were made with a 12-group set of cross section data. Attention has initially focused on the radial direction, and a one-dimensional (r) model was used. The benchmark comparisons have shown relatively small changes in the axial direction, but these need further study. However, the differences in radial power distribution have been shown to be strongly correlated with the inner core/outer core k_B^2 ratio.

B. Calculation of the Sensitivities

The studies were initially planned in 12 groups, to be consistent with other sensitivity analysis at Argonne. The multigroup data for each reactor were separately collapsed to form 12-group libraries using a one-dimensional cylindrical model. A constant half-height was found for each case to give a k_{eff} close to 0.986; the value obtained for many ZPPR cores with the ENDF/B-IV data. Calculations for the reference configuration gave sensitivities for both k_{eff} and reaction rate distributions. For the control rod studies, the composition specified for the 1250 MWe benchmark was used and produced a worth of 0.94\$ in the one-dimensional model. For ZPPR-2 and ZPPR-9, control

rod designs used in the experimental programs were modeled. These had calculated worths of 3.9\$ and 3.4\$ respectively. The three reactor models are shown in Fig. 46.

Sensitivities were calculated directly by varying the isotopic data in the model and comparing results back to the reference case. These initial calculations were made by varying selected isotope-reaction cross sections, (e.g. $^{239}\text{Pu}(n,f)$), one at a time, over all groups simultaneously. That is, one-group sensitivities were obtained, but by means of a multigroup calculation. No compensating changes were made to return k_{eff} to the reference value in the generation of these sensitivities since the object was to compare the predictions of different data libraries for the same model. The results for the one-group sensitivities are reported here.

For study of diffusion coefficient sensitivities, the macroscopic Σ_{tr} was varied rather than the isotopic σ_{tr} . In addition, the axial leakage (via DB^2) was kept constant, as D was varied, in order to concentrate on radial effects. All cross-section variations were made by an increase of 10% but the diffusion coefficient was varied by 5%. For the k_{eff} and power shape sensitivities additional studies of diffusion coefficient variations were made with rz models. These gave sensitivities for both axial and radial diffusion coefficient changes.

C. Sensitivities

The sensitivities are defined as the percent change in the given parameter for a one-percent increase in cross section. The following one-group sensitivities are given in the tables, for each of the three reactors:

TABLE XVI. k_{eff}

TABLE XVII. Inner core maximum/minimum (i.e., center/edge of inner core) for the $^{239}\text{Pu}(n,f)$ reaction rate.

ZPPR-2 ($H/2 = 69.801$, $k_{eff} = 0.9836$)

C.R.P. (2)	LC (20)	O.C. (9)	R.B. (13)	R.R. (4)
0.0000 6.3782		64.4200	91.1000	129.0900 140.8200

Key

H/2 Half Height
 I.C. Inner Core
 O.C. Outer Core
 R.B. Radial Blanket
 R.R. Radial Reflector
 C.R.P. Control Rod Position
 (for CR worth calculations only)

All dimensions in cm.
 Number of mesh intervals in ().

ZPPR-9 ($H/2 = 74.151$, $k_{eff} = 0.9844$)

C.R.P. (3)	LC (27)	O.C. (11)	R.B. (8)	R.R. (9)
0.0000 9.3508		88.8720	119.8480	142.5840 159.4480

1250 MWe Core ($H/2 = 65.428$, $k_{eff} = 0.9860$)

C.R.P. (2)	LC (36)	O.C. (9)	R.B. (9)	R.R. (3)
0.0000 7.2026		136.8501	176.5330	223.1990 238.5877

Fig. 46. One-dimensional Models of ZPPR-2, ZPPR-9 and the 1250 MWe Core.

TABLE XVIII. Inner core average/outer core average for $^{239}\text{Pu}(n,f)$.

TABLE XIX. Center control rod worth relative to fuel.

The principal fission and capture reactions are treated in each case. The only scattering cross section presently included in $^{238}\text{U}(n,n')$. The higher plutonium isotopes ^{241}Pu and ^{242}Pu are present only in small quantities in the ZPPR cores and several of these reactions have been omitted. Even in the large benchmark core some of these sensitivities are very small. Some of the smaller sensitivities may be inadequate because of convergence accuracy in the flux calculations and remain to be checked.

TABLE XVI. Sensitivity Coefficients for k_{eff}

Cross Section	ZPPR-2	ZPPR-9	Benchmark
$^{238}\text{U}(n,\gamma)$	-0.201	-0.247	-0.247
$^{238}\text{U}(n,f)$	+0.070	+0.089	+0.083
$^{238}\text{U } \nu$	+0.113	+0.145	+0.135
$^{238}\text{U}(n,n')$	+0.038	-0.060	-0.054
$^{239}\text{Pu}(n,\gamma)$	-0.063	-0.058	-0.050
$^{239}\text{Pu}(n,f)$	+0.575	+0.515	+0.475
$^{239}\text{Pu } \nu$	+0.830	+0.802	+0.667
$^{240}\text{Pu}(n,\gamma)$	-0.0089	-0.0081	-0.015
$^{240}\text{Pu}(n,f)$	+0.017	+0.016	+0.028
$^{241}\text{Pu}(n,\gamma)$	NA	NA	-0.007
$^{241}\text{Pu}(n,f)$	+0.015	+0.011	+0.106
$^{242}\text{Pu}(n,\gamma)$	NA	NA	-0.0013
$^{242}\text{Pu}(n,f)$	NA	NA	+0.0026
Fe(n, γ)	-0.017	-0.014	-0.013
Cr(n, γ)	-0.0062	-0.0059	-0.0067
Ni(n, γ)	-0.0035	-0.0032	-0.0056
Radial D	-0.049	-0.053	-0.032
Axial D	-0.084	-0.073	-0.075

TABLE XVII. Sensitivity Coefficients for
 Inner Core Maximum/Inner Core Minimum
 $^{239}\text{Pu}(n,f)$ Distribution

Cross Section	Assembly		
	ZPPR-2	ZPPR-9	Benchmark
$^{238}\text{U}(n,\gamma)$	-0.084	-0.254	-0.365
$^{238}\text{U}(n,f)$	+0.0096	+0.032	+0.036
$^{238}\text{U } \nu$	+0.014	+0.054	+0.061
$^{238}\text{U}(n,n')$	-0.0014	-0.047	-0.065
$^{239}\text{Pu}(n,\gamma)$	-0.015	-0.021	-0.022
$^{239}\text{Pu}(n,f)$	+0.0005	+0.0005	+0.020
$^{239}\text{Pu } \nu$	-0.0091	-0.050	-0.049
$^{240}\text{Pu}(n,\gamma)$	-0.0016	-0.0022	-0.0043
$^{240}\text{Pu}(n,f)$	-0.0033	-0.0091	-0.024
$^{241}\text{Pu}(n,\gamma)$	NA	NA	-0.0009
$^{241}\text{Pu}(n,f)$	+0.0010	+0.0025	+0.026
$^{242}\text{Pu}(n,\gamma)$	NA	NA	-0.0005
$^{242}\text{Pu}(n,f)$	NA	NA	-0.0027
Fe(n, γ)	-0.0063	-0.0111	-0.020
Cr(n, γ)	-0.0032	-0.0052	-0.010
Ni(n, γ)	-0.0011	-0.0022	-0.0071
Radial D	-0.026	+0.460	+0.833
Axial D	-0.090	-0.027	-0.056

TABLE XVIII. Sensitivity Coefficients for
 Inner Core Average/Outer Core Average
 $^{239}\text{Pu}(n, f)$ Distribution

Cross Section	Assembly		Benchmark
	ZPPR-2	ZPPR-9	
$^{238}\text{U}(n, \gamma)$	-0.041	-0.170	-0.240
$^{238}\text{U}(n, f)$	+0.0054	-0.021	+0.021
$^{238}\text{U } \nu$	+0.0069	-0.036	+0.038
$^{238}\text{U}(n, n')$	+0.0048	-0.023	-0.030
$^{239}\text{Pu}(n, \gamma)$	-0.013	-0.023	-0.015
$^{239}\text{Pu}(n, f)$	-0.0004	-0.0011	+0.014
$^{239}\text{Pu } \nu$	-0.0011	-0.033	+0.030
$^{240}\text{Pu}(n, \gamma)$	-0.0012	-0.0029	-0.0035
$^{240}\text{Pu}(n, f)$	-0.0016	-0.0072	-0.016
$^{241}\text{Pu}(n, f)$	-0.0017	+0.0014	+0.016
Fe(n, γ)	-0.0079	-0.0083	-0.016
Cr(n, γ)	-0.0052	-0.0042	-0.0083
Ni(n, γ)	-0.0040	-0.0016	-0.0001
D	+0.029	+0.233	+0.229

TABLE XIX. Sensitivity Coefficients for
Control Rod Worth Relative to Fuel

Cross Section	Assembly		
	ZPPR-2	ZPPR-9	Benchmark
$^{238}\text{U}(n,\gamma)$	-0.593	-0.833	-0.973
$^{238}\text{U}(n,f)$	-0.0095	+0.022	+0.023
$^{238}\text{U } \nu$	+0.048	+0.123	+0.123
$^{238}\text{U}(n,n')$	+0.040	-0.146	-0.143
$^{239}\text{Pu}(n,\gamma)$	-0.196	-0.156	-0.140
$^{239}\text{Pu}(n,f)$	+0.379	+0.464	+0.389
$^{239}\text{Pu } \nu$	+0.904	+0.826	+0.658
$^{240}\text{Pu}(n,f)$	-0.0056	-0.0071	-0.029
$^{241}\text{Pu}(n,f)$	+0.016	+0.016	+0.143
Fe(n, γ)	-0.053	-0.040	-0.056
Cr(n, γ)	-0.019	-0.019	-0.026
Ni(n, γ)	-0.010	-0.0088	-0.023
D	+0.273	+1.047	+1.433

The following conclusions may be drawn from the tables:

- Sensitivities for k_{eff} are similar for all three cores, except where particular isotopes are present in appreciably different quantities. This implies a similar k_{eff} bias for each case with a given data set.
- For the $^{239}\text{Pu}(n,f)$ distribution, the sensitivities are much larger in the benchmark core than for ZPPR-2, in many cases larger by a factor of four or more. However, the sensitivities for ZPPR-9 are generally about 70% of the benchmark values, except for the higher plutonium isotopes as previously noted. This leads to the important conclusion that experiments in ZPPR-9 may be extrapolated with some confidence to the larger core size, whereas results from the smaller cores may be quite inconclusive.
- Sensitivities for the central control rod show effects similar to those for the power distribution. Values for ZPPR-9 are generally much larger than ZPPR-2 and approach those for the benchmark core.
- Apart from the nu-values which are uncertain by only 0.5 to 1%, the largest sensitivities are for $^{238}\text{U}(n,\gamma)$, $^{239}\text{Pu}(n,f)$, and the diffusion coefficient (for power distributions and rod worth). Sensitivities for other reactions are smaller but may be significant since uncertainties on these data may be higher -- e.g., 25% or more for steel capture components. The high sensitivities to diffusion coefficients in the large cores were unexpected.

D. Comparison of Results with ENDF/B-IV, FGL5 and CARNAVAL-III Data Sets

Some of the most striking differences seen in the benchmark comparison exercise were those between predictions using ENDF/B data and adjusted data sets, particularly CARNAVAL-III (Cadache-1) and FGL5 (UKAEA). The calculated sensitivities and applied data changes for these sets, relative to ENDF/B-IV (ANL), have been used to study the effect on reactor predictions. For the benchmark core, these may be compared with the complete calculations of the contributors.

One-group cross sections were supplied by each participant for some of the principal reaction types. For other capture and fission reactions, the changes in cross-section have been derived from the neutron balances. It may be noted that the percentage changes for cross-sections in the outer core and inner core were practically identical. For CARNAVAL-III, changes in transport cross-sections were not originally supplied and the changes in diffusion coefficient for this data set have not yet been determined. For the FGL5 data, the changes in transport cross sections relative to ENDF/B-IV that were identified were principally for sodium, oxygen, iron and ^{238}U . These isotopes contributed about 85% of the macroscopic diffusion coefficient. Furthermore, changes in D for the inner core, outer core and radial blanket were almost the same.

From the participants solutions submitted for the benchmark, the following differences were obtained relative to ANL (ENDF/B-IV):

	<u>FGL5</u>	<u>CARNAVAL-III</u>
k_{eff}	+2.95%	+2.84%
Inner core power, max/min	+7.2%	+10.7%
Control rod worth (Δk relative to fuel)	+19.5%	+30.6%

It is seen from Tables XX-XXII that the one-group sensitivities predict 60% to 90% of these differences with FGL5 data. The totals for CARNAVAL data could not be obtained because of lack of information on transport cross sections.

The predicted change for k_{eff} is less accurate than for the other two parameters. The change in k_{eff} due to the diffusion coefficient is -0.51%. On the other hand, if the known differences in core leakage are treated as effective absorption cross sections (sensitivity -0.17), these differences of +0.12% (FGL5) and -1.70% (CARNAVAL-III) predict changes in k_{eff} of -0.02% and +0.29% respectively. This discrepancy needs further investigation.

A possible reason for the poor prediction of k_{eff} may be neglect of ^{238}U inelastic scattering effects. The sensitivities to the total inelastic scattering cross sections are given in the tables, but the actual cross section differences are not known. However, reductions of 10% to 20% in $^{238}\text{U}(n,n')$ for FGL5 relative to ENDF/B-IV would not be unfeasible. These would give increases in k_{eff} for the benchmark of 0.5% to 1.1%. These same data changes would give further improvements in prediction for the power shape of +0.7% to +1.3% and for the control rod worth of +1.4% to +2.8%.

The predicted changes in k_{eff} , inner core power shape and control rod worth are given in Tables XX, XXI and XXII. The percentage changes between ENDF/B-IV (ANL) and FGL5 one-group cross-sections were used to predict changes in the three parameters for ZPPR-2, ZPPR-9 and the 1250 MWe benchmark. For CARNAVAL-III data, changes for the benchmark only are given.

It is worth noting that the predicted changes for ZPPR-2 in going from ENDF/B-IV to FGL5 data are not inconsistent with evidence from the analysis of integral experiments for cores of 300 MWe size:

TABLE XX. Predicted Changes in k_{eff} Relative to ENDF/B-IV (ANL)

Cross Section	Percent Change in k_{eff}					
	FGL5 Data			CARNAVAL-III Data		
	Percent Difference	ZPPR-2	ZPPR-9	Benchmark	Percent Difference	Benchmark
$^{238}\text{U}(n,\gamma)$	-5.05	+1.02	+1.25	+1.25	-2.54	+0.63
(n,f)	+6.72	+0.47	+0.60	+0.56	+0.22	+0.02
ν	-0.63	-0.07	-0.09	-0.09	+1.68	+0.23
$^{239}\text{Pu}(n,\gamma)$	-3.95	+0.25	+0.23	+0.20	+1.34	-0.07
(n,f)	-0.56	-0.32	-0.29	-0.27	+1.30	+0.62
ν	+0.30	+0.25	+0.24	+0.20	-0.88	-0.59
$^{240}\text{Pu}(n,\gamma)$	+9.5	-0.09	-0.08	-0.14	-3.5	+0.05
(n,f)	-1.1	-0.02	-0.02	-0.03	+4.5	+0.12
$^{241}\text{Pu}(n,\gamma)$	+20.6	NA	NA	-0.14	+15.4	-0.11
(n,f)	+3.2	+0.05	+0.04	+0.34	+4.8	+0.51
$^{242}\text{Pu}(n,\gamma)$	+6.3	NA	NA	-0.01	+64.9	-0.08
(n,f)	-3.1	NA	NA	-0.01	-7.6	-0.02
$\text{Fe}(n,\gamma)$	+4.5	-0.08	-0.06	-0.06	-28.0	+0.37
$\text{Cr}(n,\gamma)$	-60.9	+0.38	+0.36	+0.41	-58.7	+0.39
$\text{Ni}(n,\gamma)$	-20.4	+0.07	+0.07	+0.11	-38.9	+0.22
Radial D	+4.8 ^a	-0.24	-0.25	-0.15		NA
Axial D	+4.8 ^a	-0.40	-0.35	-0.36		NA
Total		+1.27	+1.65	+1.82		NA

^aEstimated values. See text.

TABLE XXI. Predicted Changes in Inner Core Max/Min
 $^{239}\text{Pu}(n,f)$ Relative to ENDF/B-IV (ANL)

Cross Section	Percent Change in Inner Core Max/Min $^{239}\text{Pu}(n,f)^a$			
	FGL5 Data			CARNAVAL-III Data
	ZPPR-2	ZPPR-9	Benchmark	Benchmark
$^{238}\text{U}(n,\gamma)$	+0.42	+1.28	+1.84	+0.93
$^{238}\text{U}(n,f)$	+0.06	+0.22	+0.24	+0.01
ν	-0.01	-0.03	-0.03	+0.10
$^{239}\text{Pu}(n,\gamma)$	+0.06	+0.08	+0.09	-0.03
$^{239}\text{Pu}(n,f)$	-0.00	-0.00	-0.01	+0.03
ν	-0.00	-0.01	-0.01	+0.04
$^{240}\text{Pu}(n,\gamma)$	-0.15	-0.02	-0.04	+0.02
$^{240}\text{Pu}(n,f)$	+0.00	+0.01	+0.08	+0.10
$^{241}\text{Pu}(n,\gamma)$	NA	NA	-0.02	-0.01
$^{241}\text{Pu}(n,f)$	+0.00	+0.01	+0.08	+0.12
$^{242}\text{Pu}(n,\gamma)$	NA	NA	-0.00	-0.03
$^{242}\text{Pu}(n,f)$	NA	NA	+0.01	+0.02
Fe(n, γ)	-0.03	-0.05	-0.09	+0.57
Cr(n, γ)	+0.19	+0.32	+0.62	+0.60
Ni(n, γ)	+0.02	+0.04	-0.14	-0.28
Radial D	-0.12	+2.21	+4.00	NA
Axial D	-0.43	-0.13	-0.27	NA
Total	+0.01	+3.93	+6.56	NA

^aCorresponding to the percent differences in cross sections given in Table XX.

TABLE XXIII. Predicted Changes in Control Rod Worth
Relative to ENDF/B-IV (ANL)

Cross Section	Percent Change in Control Rod Worth ^a			
	FGL5 Data			CARNAVAL-III Data
	ZPPR-2	ZPPR-9	Benchmark	Benchmark
$^{238}\text{U}(n,\gamma)$	+3.00	+4.21	+4.92	+2.47
(n,f)	-0.06	+0.15	+0.16	+0.01
ν	-0.03	-0.08	-0.08	+0.21
$^{239}\text{Pu}(n,\gamma)$	+0.77	+0.62	+0.55	-0.19
(n,f)	-0.21	-0.26	-0.22	+0.51
ν	+0.27	+0.25	+0.20	-0.58
$^{240}\text{Pu}(n,\gamma)$	NA	NA	NA	NA
(n,f)	-0.05	+0.01	+0.03	-0.13
$^{241}\text{Pu}(n,\gamma)$	NA	NA	NA	NA
(n,f)	+0.05	+0.05	+0.46	+0.69
$^{242}\text{Pu}(n,\gamma)$	NA	NA	NA	NA
(n,f)	NA	NA	NA	NA
$\text{Fe}(n,\gamma)$	-0.24	-0.18	-0.25	+1.55
$\text{Cr}(n,\gamma)$	+1.16	+1.18	+1.60	+1.54
$\text{Ni}(n,\gamma)$	+0.21	+0.18	+0.48	+0.91
Radial D	+1.32	+5.08	+6.95	NA
Total	+6.19	+11.21	+14.80	NA

^aCorresponding to the percent differences in cross sections given in Table XX.

- k_{eff} for the MZB assembly is 1.001 with FGL5 data⁴ compared with values from 0.982 to 0.986 for ZPPR-2, 3, or 4 with ENDF/B-IV⁵. Uncertainties of a few tenths of a percent arise from the treatment of transport, streaming, heterogeneity effects, etc.
- Neither data set shows any significant error in inner core fission distributions within experimental uncertainties of up to 1%.
- Control rod worth C/Es, for a central rod of natural boron, using equivalent diffusion theory calculations are 1.04 in MZC with FGL5 data⁴ and 1.09 in ZPPR-4 with ENDF/B-IV⁵. However, the use of the UK delayed neutron data would reduce the C/E for ZPPR by between 5 and 10%.

E. Conclusions

Although the predicted changes in parameters using FGL5 nuclear data are not as large as obtained for the direct calculations, they do make up a substantial part, approximately 60-90%, of the reported differences. Relatively high sensitivities to diffusion coefficients are found in the large cores. Changes in one-group diffusion coefficients have been inferred from changes in the microscopic transport cross sections. This may not accurately reflect the differences in diffusion coefficients used in the flux calculations. It is possible that significant changes in reactor parameters also arise from unidentified differences in ^{238}U inelastic scattering data.

Two results of this study appear particularly important:

- Sensitivities for power distributions and control rod worths for ZPPR-9 are substantially larger than for ZPPR-2. This will presumably be the case for other properties such as central reactivity worths.

- Sensitivities for ZPPR-9 approach those of the 1250 MWe size reactor enabling a reasonably small extrapolation of integral information.

Finally, analysis of experiments in the ZPPR-9 and BIZET cores will provide tests of some aspects of data libraries not hitherto available from experiments in the smaller cores.

VIII. MONTE CARLO AND TRANSPORT THEORY SOLUTIONS

The benchmark specifications requested a two-dimensional diffusion theory solution for the cylindrical benchmark model. Furthermore, in addition to fixing the dimensions and atom concentrations, the specifications either set or recommended computational parameters, such as number of mesh intervals, number and width of broad energy groups, and flux and eigenvalue convergence criteria, in order that (a) the calculational methods would be consistent among participants and (b) these methods would yield an acceptably accurate solution. Because of the excellent cooperation of the participants in following the benchmark specifications, the submitted solutions are very consistent with regard to calculational methods. That is, the differences in the solutions principally reflect the differences in multigroup data (i.e., differences in basic nuclear data and cross section processing methods). In order to test the accuracy of the recommended calculational methods (i.e., two-dimensional diffusion theory with the specified mesh spacing, and ~27 broad energy groups), the reference configuration has been calculated using Monte Carlo and transport theory methods.

A. Monte Carlo Solution

For the specified geometry and homogeneous compositions of the cylindrical benchmark model, the Monte Carlo methods give basically an exact solution of the Boltzman equation. The Monte Carlo solution was provided by H. Henryson (ANL) using the VIM code⁷ and its associated statistical editing packages. Neutron cross sections for VIM are derived from the ENDF/B Version IV data files and are treated as continuous functions of energy. Resolved resonances are "traced out" by a set of point cross section values suitable for interpolation. Unresolved resonances are treated by a probability table method. The energy range for the VIM calculation was 10 MeV down to 13.71 eV.

Twenty-seven half-lethargy width energy intervals were used for edits. This energy range (and 27 broad energy group structure) was chosen to be identical to that of the ANL (ENDF/B-IV) solution submitted for this benchmark comparison. However, because the VIM cross section library* was derived from the ENDF/B-IV data files for a temperature of 300°K, the VIM Monte Carlo solution is not directly comparable to any of the submitted solutions for the reference configuration (for which cross sections were derived from the specified temperature of 1100°K). In order to make a comparison with this Monte Carlo solution, the ANL calculations for the reference configuration were repeated (as outlined previously in Section II) with the single exception that the temperature was changed to 300°K.

The VIM code provides three eigenvalue estimators (track length, collision, and analog) plus some combinations of these estimators to obtain reduced variance estimates of the eigenvalue. These eigenvalue estimates obtained by VIM for the reference configuration are based on 100,000 neutron histories and are summarized in Table XXIII. The best estimate (i.e., smallest 1σ) by VIM for this eigenvalue is 1.00775 ± 0.001947 , as compared to 1.00711 for the ANL diffusion theory solution using the 300°K cross sections.

Additional results of these two solutions are compared in Tables XXIV-XXVI. Table XXIV summarizes the neutron balance by region for the Monte Carlo and diffusion theory solutions. The normalization for these neutron balance edits is such that the total fission production in the reactor is unity. The agreement between the two solutions (based on 300°K cross sections) is excellent, that is, generally well within the 1σ statistics of the Monte

*The VIM code has been widely used at ANL and has been particularly useful for the validation of cross section homogenization methods for fast reactor critical assembly analysis.⁸

TABLE XXIII. VIM Estimates of Eigenvalue for the Reference Configuration

Estimator	Eigenvalue $\pm$ Standard Deviation
Analog	0.999179 $\pm$ 0.003972
Track Length	1.010699 $\pm$ 0.002365
Collision	1.010791 $\pm$ 0.002219
Combined Analog and Track Length	1.00748 $\pm$ 0.001947
Combined Analog and Collision	1.00775 $\pm$ 0.001833
Combined Track Length and Collision	1.01081 $\pm$ 0.002215
Average of Analog and Track Length	1.00494 $\pm$ 0.002217
Average of Analog and Collision	1.00498 $\pm$ 0.002156
Average of Track Length and Collision	1.01075 $\pm$ 0.002267
Average of all three Estimators	1.00689 $\pm$ 0.001900

TABLE XXIV. Neutron Balance Comparison
Between VIM and MC²-2/ARC Calculations*

	VIM(300°K)	ARC(300°K)	ARC(1100°K)	$\frac{\text{ARC}(300^\circ\text{K})}{\text{VIM}(300^\circ\text{K})}$
<u>Fission Production</u>				
Inner Core	0.63145 ± 0.00387	0.63495	0.63012	1.00554 ± 0.00613
Outer Core	0.34828 ± 0.00366	0.34346	0.34856	0.98616 ± 0.01051
(Core Subtotal)	0.97973 ± 0.00533	0.97841	0.97868	0.99865 ± 0.00544
Radial Blkt.	0.00917 ± 0.00023	0.00940	0.00936	1.02508 ± 0.02526
Axial Blkt.	0.01110 ± 0.00021	0.01219	0.01197	1.09820 ± 0.01879
Reflector	—	—	—	—
(Reactor Total)	1.00000 ± 0.00534	1.00000	1.00000	1.00000 ± 0.00534
<u>Net (n,2n) Source</u>				
Inner Core	0.00072 ± 0.00003	0.00125	0.00124	1.73611 ± 0.04310
Outer Core	0.00036 ± 0.00002	0.00061	0.00063	1.69444 ± 0.05310
(Core Subtotal)	0.00107 ± 0.00004	0.00186	0.00187	1.73832 ± 0.03370
Radial Blkt.	0.00005 ± 0.00001	9.4(-5)	9.6(-5)	1.88000 ± 0.18300
Axial Blkt.	0.00006 ± 0.00001	0.00012	0.00012	2.00000 ± 0.13200
Reflector	5.4(-9) ± 5.4(-9)	1.0(-8)	9.8(-9)	1.85185 ± 1.00000
(Reactor Total)	0.00118 ± 0.00004	0.00207	0.00209	1.75424 ± 0.03282
<u>Capture</u>				
Inner Core	0.33189 ± 0.00238	0.33409	0.33996	1.00663 ± 0.00717
Outer Core	0.14422 ± 0.00166	0.14151	0.14631	0.98121 ± 0.01150
(Core Subtotal)	0.47611 ± 0.00290	0.47560	0.48627	0.99893 ± 0.00609
Radial Blkt.	0.06571 ± 0.00128	0.06450	0.06657	0.98159 ± 0.01950
Axial Blkt.	0.09294 ± 0.00102	0.09242	0.09413	0.99440 ± 0.01100
Reflector	0.00213 ± 0.00008	0.00213	0.00199	1.00000 ± 0.03550
(Reactor Total)	0.63688 ± 0.00333	0.63465	0.64896	0.99650 ± 0.00523

TABLE XXIV. (Cont'd.)

	VIM(300°K)	ARC(300°K)	ARC(1100°K)	$\frac{\text{ARC}(300^\circ\text{K})}{\text{VIM}(300^\circ\text{K})}$
<u>Fission</u>				
Inner Core	0.21733 $\pm$ 0.00133	0.21827	0.21669	1.00433 $\pm$ 0.00613
Outer Core	0.11941 $\pm$ 0.00125	0.11762	0.11941	0.98501 $\pm$ 0.01050
(Core Subtotal)	0.33674 $\pm$ 0.00183	0.33589	0.33610	0.99748 $\pm$ 0.00542
Radial Blkt.	0.00347 $\pm$ 0.00008	0.00352	0.00350	1.01441 $\pm$ 0.02420
Axial Blkt.	0.00421 $\pm$ 0.00007	0.00458	0.00449	1.08789 $\pm$ 0.01770
Reflector	---	---	---	---
(Reactor Total)	0.34442 $\pm$ 0.00183	0.34399	0.34409	0.99875 $\pm$ 0.00531
<u>Summary</u>				
k_{eff}	1.00775 $\pm$ 0.00183	1.00711	0.99304	0.99936 $\pm$ 0.00182
Leakage from				
Inner Core	0.07809 $\pm$ 0.00485	0.07936	0.07914	1.01345 $\pm$ 0.06205
Inner Core + Outer Core	0.16042 $\pm$ 0.00655	0.16187	0.16504	1.00384 $\pm$ 0.04083
Reactor	0.01219 $\pm$ 0.00677	0.01637	0.01605	1.25537 $\pm$ 0.55537

*MC²-2/ARC denotes the ANL cross section preparation (MC²-II) and 2D diffusion theory (ARC) code sequence.

TABLE XXV. Comparison of Inner-Core-Averaged Neutron Flux
Between VIM and MC²-2/ARC Calculations*

Group	ARC (300°K)	VIM (300°K)	$\frac{\text{ARC (300°K)}^a}{\text{VIM (300°K)}}$
1	2.52810(-3)	2.5103(-3)	1.0071 ± 0.0313
2	1.05296(-2)	1.0349(-2)	1.0175 ± 0.0159
3	2.69088(-2)	2.6802(-2)	1.0040 ± 0.0113
4	3.86725(-2)	3.8778(-2)	0.9973 ± 0.00891
5	4.89626(-2)	4.8473(-2)	1.0101 ± 0.00922
6	9.14339(-2)	9.1575(-2)	0.9985 ± 0.00798
7	8.38641(-2)	8.3709(-2)	1.0019 ± 0.00726
8	1.05629(-1)	1.0596(-1)	0.9969 ± 0.00640
9	1.12248(-1)	1.1258(-1)	0.9971 ± 0.00700
10	1.01642(-1)	1.0202(-1)	0.9963 ± 0.00686
11	8.83570(-2)	8.8173(-2)	1.0021 ± 0.00663
12	7.08386(-2)	7.0889(-2)	0.9993 ± 0.00742
13	6.59042(-2)	6.5870(-2)	1.0005 ± 0.00817
14	5.14392(-2)	5.1336(-2)	1.0020 ± 0.00862
15	3.04320(-2)	3.0444(-2)	0.9996 ± 0.0105
16	1.90079(-2)	1.8402(-2)	1.0329 ± 0.00986
17	6.88767(-3)	6.9809(-3)	0.9866 ± 0.00993
18	1.96582(-2)	1.9651(-2)	1.0004 ± 0.0118
19	1.25562(-2)	1.3036(-2)	0.9632 ± 0.0142
20	7.16934(-3)	7.2323(-3)	0.9913 ± 0.0200
21	3.10949(-3)	3.0279(-3)	1.0269 ± 0.0312
22	1.45770(-3)	1.5001(-3)	0.9717 ± 0.0375
23	5.67990(-4)	5.4195(-4)	1.0480 ± 0.0750
24	1.54828(-4)	1.1888(-4)	1.3024 ± 0.144
25	3.23307(-5)	3.7817(-5)	0.8523 ± 0.203
26	7.15729(-6)	9.1185(-6)	0.7849 ± 0.444
27	1.21638(-6)	6.2165(-6)	0.1957 ± 0.467

^aUncertainties represent 1- σ interval from VIM calculations.

*MC²-2/ARC denotes the ANL cross section preparation (MC²-II) and 2D diffusion theory (ARC) code sequence.

TABLE XXVI. Comparison of Inner-Core-Averaged Reaction Rate Ratios
Between VIM and MC²-2/ARC Calculations*

	ARC (300°K)	VIM (300°K) ^a	$\frac{\text{ARC (300°K)}^a}{\text{VIM (300°K)}}$
f^{25}/f^{49}	0.02230	0.02232 ± 0.00021	0.9995 ± 0.0094
f^{28}/f^{49}	0.21073	0.20949 ± 0.00207	1.0060 ± 0.0099
c^{28}/f^{49}	1.58510	1.58432 ± 0.01501	1.0005 ± 0.0095
c^{49}/f^{49}	0.31776	0.31929 ± 0.00347	0.9952 ± 0.0109
f^{40}/f^{49}	0.05393	0.05372 ± 0.00049	1.0040 ± 0.0091
c^{40}/f^{49}	0.09167	0.09231 ± 0.00106	0.9931 ± 0.0115
f^{41}/f^{49}	0.21664	0.21654 ± 0.00202	1.0005 ± 0.0093

^aUncertainties represent 1- σ interval from VIM calculations.

*MC²-2/ARC denotes the ANL cross section preparation (MC²-II) and 2D diffusion theory (ARC) code sequence.

Carlo calculation. The major exception is the calculated (n,2n) reaction rates. These are consistently higher (by 70 to 100%) in the MC²-2/ARC solution relative to the VIM solution. This bias in the (n,2n) reaction rate is due almost totally to the difference in the (n,2n) cross section which is significant only in the first or highest energy group. The fraction of the flux in the first energy group is nearly identical (within 1%) for these two solutions.

The neutron balance comparison in Table XXIV also indicates a small difference in leakage between the Monte Carlo and diffusion theory solutions.* The slightly larger leakage fractions in the diffusion theory solution (relative to the Monte Carlo solution) also produce slightly larger reaction rates in the blanket regions, especially the axial blanket. One might in fact question why the differences between the Monte Carlo and diffusion theory solutions are not larger. However, as will next be discussed, the comparison of diffusion theory and transport theory solutions indicate the transport effects are small (of the order of 0.2% Δk , which is comparable to the 1 σ statistics on the eigenvalue of the VIM Monte Carlo solution).

The inner-core-averaged neutron flux spectra for the VIM and MC²-2/ARC calculations are compared in Table XXV. Inner-core-averaged reaction rate ratios are compared in Table XXVI. As shown in these tables, the agreement between the Monte Carlo and diffusion theory solutions for the inner-core-averaged flux spectra and reaction rate ratios is excellent ($\leq 1\sigma$).

*As stated previously in Section III of this report, the preliminary values of the ANL solution included calculated leakage fractions which were in error due to incorrect normalization. These values have been revised herein.

B. Transport Theory Solution

Several transport theory (S_4 - P_0) solutions were calculated for the reference configuration using the two-dimensional transport theory capability of the DIF3D code.⁹ The S_n calculations utilized the same R-Z cylindrical model (and mesh spacings) and the same 27 broad energy group MC²-2 generated ENDF/B-IV cross sections as the diffusion theory solution. The S_4 - P_0 eigenvalues obtained using both the 300°K and 1100°K cross section sets are given in Table XXVII. It may be noted that the eigenvalue difference (from diffusion theory to S_4) is +0.0013. Since the correction from S_4 to S_∞ would be of comparable magnitude, it may be concluded the total transport correction (i.e., diffusion theory to S_∞) is ~0.1-0.2% Δk , which is comparable to the 1 σ statistics on the VIM Monte Carlo estimate of the eigenvalue. As would be expected, the real and adjoint flux spectra are very similar between the diffusion theory and S_4 solutions calculations. These calculated central real flux spectra are displayed in Fig. 47. Also, the radial and axial fission rate profiles are very similar between the diffusion theory and S_4 solutions (as shown in Figs. 48 and 49). These fission rate profiles display the slightly greater neutron leakage from the core (production) regions to the blanket (absorption) regions in the diffusion theory solution (as compared to the S_n solution). This diffusion-to- S_n bias is slightly larger radially than axially.

TABLE XXVII. Summary of Monte Carlo, S_n ,
and Diffusion Theory Eigenvalues for n ,
Configuration 1

Calculation	Eigenvalue	
	300°K	Cross Section Set 1100°K
VIM (Monte Carlo)	1.00775 ± 0.00183	---
MC ² -2/DIF3D (S_4 - P_0)	1.00842	0.99431
MC ² -2/DIF2D (Diffusion Theory)	1.00710	0.99304
	$\delta k_{\text{Diff.} \rightarrow S_4}$	
$k_{S_4P_0} - k_{\text{Diffusion Theory}}$	+0.00132	+0.00127

SPECTRA AT MESH (01,01) IN CONFIGURATION 1

Nota. These group fluxes are normalized such that the sum of the group fluxes equals one.

— DIF2D Diff. Th.
 - - - DIF3D S_4/P_0

RADIAL FISSION RATE DISTRIBUTION FOR CONFIGURATION 1

Fig. 48.

AXIAL FISSION RATE DISTRIBUTION FOR CONFIGURATION 1

Fig. 49.

C. Calculations to Test Convergence

Although there was little reason to suspect a problem of convergence* to be a significant contribution to the mispredictions (or discrepancies) of the radial power profiles, two-dimensional diffusion theory and S_n calculations were repeated for the reference configuration with different convergence criteria and different mesh spacings. The diffusion theory results are compared in Table XXVIII. The eigenvalue and flux difference criteria were each 10^{-5} for these calculations. The change in eigenvalue between the calculations with the standard mesh and the fine mesh is less than $3 \times 10^{-4} \delta k$; the change in the inner core/outer core power split is less than 0.5%. By fitting the fine mesh and standard mesh results to an exponential of the form $k_{h=0} = k_h e^{\alpha h}$, one finds that the zero mesh results are given by:

$$k_{h=0} = 0.99920 \quad k_{h=3.6013} = 0.99224$$

The effect of changing the mesh spacing upon the calculated flux profiles is shown to be small (<0.3% between the standard and fine mesh solutions throughout the inner and outer core) in the graphs of the radial and axial fission rate profiles shown in Figs. 50 and 51, respectively. Similar agreement was found for DIF3D/ S_4P_0 calculations using the standard mesh spacing and fine mesh spacing.

Finally, a two-dimensional diffusion theory calculation for the reference configuration was repeated using logarithmic boundary conditions on the outer radial and axial surfaces, as compared to the zero flux boundary conditions. The resulting eigenvalue was 0.99311, or a +0.00007 δk change relative to the reference calculation.

*The benchmark specifications had set the eigenvalue convergence criteria at $10^{-5} \delta k$ and the mesh spacing at 3.6 cm and 4.4 cm throughout the inner core and outer core, respectively (which is approximately equal to the mean free path in these regions).

TABLE XXVIII. Diffusion Theory Calculations for the Reference Configuration
Using Standard, Fine, and Coarse Spatial Mesh

	Standard Mesh	Fine Mesh	Percent Diff. Rel. to Standard	Coarse Mesh	Percent Diff. Rel. to Standard
No. of Mesh (R × Z)	59 × 26	88 × 38		40 × 17	
Radial Mesh Interval in Inner Core (cm)	3.6013	2.4009		5.4020	
IBM 370/195 Requirements					
Region (k bytes)	715	1220		400	
Time c.p.u. (sec)	210	361		68	
wait (sec)	180	240		95	
No. of Outers	30	26		26	
Eigenvalue	0.993035	0.992769	-0.026	0.993869	+0.085
Power Split (IC/OC)	1.8321	1.8402	+0.440	1.8092	-1.25

RADIAL FISSION RATE DISTRIBUTION FOR CONFIGURATION 1

Fig. 50.

AXIAL FISSION RATE DISTRIBUTION FOR CONFIGURATION 1

Fig. 51.

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APPENDIX A

PROBLEM SPECIFICATIONS

The problem specifications for these comparison calculations are reproduced in this appendix. These specifications include one correction from their original form. In section D.12, the first order perturbation theory expression for the reactivity worth ($\delta k/k$) had neglected the $\frac{1}{k}$ factor in the fission production term of the perturbation numerator (Eqn. 6). All participants have submitted calculated reactivity worth values based on the correct expression.

SPECIFICATIONS FOR AN INTERNATIONAL COMPARISON CALCULATION
OF A LARGE SODIUM-COOLED FAST BREEDER REACTOR

INTRODUCTION

The NEACRP is sponsoring an international comparison calculation of a large (~ 1250 MWe) sodium-cooled fast breeder reactor. A simple, two-zone cylindrical model without control rods or rod positions is defined in Section A of this document. Homogeneous compositions are specified for each region of the reactor.

The principal goal of the exercise is to determine the influence of the differences in the several sets of basic nuclear data currently in use on a broad range of calculated reactor physics parameters for a reference core and on safety-related physics parameters for several perturbations about the reference configuration.

The configurations to be calculated, and the parameters to be edited are described in Section B. Eight configurations are included:

1. Reference
2. Na Voided Inner Core.
3. Na Voided Inner Core, Outer Core, and Axial Blanket.
4. Reference with a Central Na-Filled Control Rod Position.
5. Reference with a Central Boron Control Rod.
6. Na-Voided Inner Core, Outer Core, and Axial Blanket with a Central Boron Control Rod.
7. Reference with Hot Fuel.
8. Na-Voided Inner Core, Outer Core, and Axial Blanket with Hot Fuel.

Suggested cross section preparation procedures are specified in Section C.

Sections D and E provide detailed specifications and definitions of the physics parameters to be tabulated.

The ANL solution, and a blank form for reporting results will be mailed under separate cover in several weeks.

A. Geometry and Configurations

1. Two-dimensional R,Z Geometry with axial symmetry about $z = 0$.
2. Region dimensions and recommended mesh spacing are shown in Fig. A-1.
3. Use $\phi = 0$ boundary condition at outside boundary of reflector region.
4. The central control rod is not present in all configurations. When it is present its radius is 7.2026 cm (2 mesh intervals).
5. The reactor compositions are fixed in this calculation (i.e., no search to critical enrichment is done). Region compositions and the associated cross section sets to be generated are specified in Table A-1. (Note that different broad group cross sections are generated for each composition to account for the spatial dependence of the spectrum and that separate sets for high temperature fuel are specified for a Doppler coefficient calculation.)
6. An energy group structure of 27 broad groups consisting of 27 groups of $\Delta u = 0.5$ spanning the energy range 10 MeV to 13.7 eV is recommended. A second choice is the Bondarenko 26 group structure.

B. Configurations and Requested Results

1. Eight reactor configurations are defined in Figs. A-2 through A-9, reactor configuration is defined by specifying the composition in each reactor region.
2. The calculations and edits requested for each configuration are listed on the figures by specifying an edit number. The numbers refer to the paragraphs in Section D., "Calculations and Edits", where the requested edits are defined in detail.
3. Basic calculations should use a two-dimensional R,Z diffusion code.

TABLE A-I. Compositions and Associated Broad
Group Cross Section Sets*

Cross Section Set Ident.**	Inner Core		Outer Core		Axial Blanket		Radial Blanket	Refl.
	Na In	Na Void	Na In	Na Void	Na In	Na Void		
	A	B	C	D	E	F	G	G
Na	9.6673-3	0.000	9.6673-3	0.000	9.6673-3	0.000	6.1299-3	9.6673-3
ϕ -16	17.3293-3		17.3293-3		17.8922-3		27.1562-3	---
Fe	11.2693-3	Same as A	11.2693-3	Same as C	11.2693-3	Same as E	8.7384-3	13.9943-3
Ni	2.2435-3		2.2435-3		2.2435-3		1.7397-3	29.5122-3
Cr	3.2840-3		3.2840-3		3.2840-3		2.5464-3	9.6621-3
Mo	0.2339-3		0.2339-3		0.2339-3		0.1814-3	0.2239-3
Mn	0.2842-3		0.2842-3		0.2842-3		0.2203-3	0.3607-3
²³⁵ U	0.0155-3		0.0149-3		0.0184-3		0.0279-3	---
²³⁸ U	7.6440-3		7.3256-3		9.0634-3		13.7561-3	---
²³⁹ Pu	0.7704-3		0.9868-3		---		---	---
²⁴⁰ Pu	0.2208-3		0.2828-3		---		---	---
²⁴¹ Pu	0.1165-3		0.1497-3		---		---	---
²⁴² Pu	0.0279-3		0.0358-3		---		---	---
¹⁰ B	---	---	---	---	---	---	---	---
¹² C	---	---	---	---	---	---	---	---
¹¹ B	---	---	---	---	---	---	---	---

TABLE A-I. (Cont'd.)

Cross Section Set Ident.**	Boron Control Rod		Na Filled Control Rod Position	Hot Fuel Cross Sections***			
				Inner Core		Outer Core	
	Na In	Na Out [†]		Na In	Na Void	Na In	Na Void
	A	B		A ^H	B ^H	C ^H	D ^H
Na	8.5022-3		17.7718-3				
φ-16							
Fe	17.7433-3	Same as Na In Control Rod	10.0239-3				
Ni	3.5324-3		1.3228-3				
Cr	5.1706-3		2.8795-3				
Mo	0.3683-3		0.0208-3				
Mn	0.4474-3		0.2387-3				
²³⁵ U	—		—				
²³⁸ U	—		—				
²³⁹ Pu	—		—				
²⁴⁰ Pu	—		—				
²⁴¹ Pu	—		—				
²⁴² Pu	—		—				
¹⁰ B	14.8031-3		—				
¹² C	7.5013-3		—				
¹¹ B	14.8031-3		—				

*Number densities in atoms/barn-cm.

** Broad group cross sections are obtained by collapsing over the spectra appropriate for the listed compositions.

***The "hot fuel" cross section sets are generated with the temperatures of ²³⁵U, ²³⁸U, ²³⁹Pu, ²⁴⁰Pu, ²⁴¹Pu, ²⁴²Pu, and ¹⁶O set at 2200°K while the temperature of all other isotopes is set at 1100°K.

†Note that the control rod composition contains sodium even though the surrounding core is sodium voided.

Fig. A-1. Geometry of LMFBR International Comparison Calculational Model.

Assignment of Compositions to Regions (Refer to Table I)

<u>Region</u>	<u>Composition</u>	<u>Cross Section Set</u>
Inner Core	Na in	A
Outer Core	Na in	C
Radial Blanket	Radial Blanket	G
Axial Blanket	Na in	E
Reflector	Refl.	G

Required Calculations and Edits (Refer to Section D)

Calc. and Edit No.*

- 1. X
- 2. X
- 3. X
- 4. X
- 5. X
- 6. X
- 7. X
- 8. X
- 9. X
- 10. X
- 11. X
- 12. X
- 13. X
- 14. X
- 15. X
- 16. X
- 17. X
- 18. X

*A check indicates the calculation and edit is to be made.

Fig.A-2, Configuration 1: Reference.

Assignment of Compositions to Regions (Refer to Table I)

<u>Region</u>	<u>Composition</u>	<u>Cross Section Set</u>
Inner Core	Na Void	B
Outer Core	Na In	C
Radial Blanket	Radial Blanket	G
Axial Blanket	Na In	E
Reflector	Refl	G

Required Calculations and Edits (Refer to Section D)

Calc. and Edit No.*

1. X
- 2.
- 3.
- 4.
- 5.
- 6.
- 7.
- 8.
- 9.
- 10.
11. X
- 12.
- 13.
- 14.
- 15.
- 16.
- 17.
- 18.

*A check indicates the calculation and edit is to be made.

Fig.A-3, Configuration 2: Na Voided Inner Core.

Assignment of Compositions to Regions (Refer to Table I)

<u>Region</u>	<u>Composition</u>	<u>Cross Section Set</u>
Inner Core	Na Void	B
Outer Core	Na Void	D
Radial Blanket	Radial Blanket	G
Axial Blanket	Na Void	F
Reflector	Refl	G

Required Calculations and Edits (Refer to Section D)

Calc. and Edit No.*

1.	X
2.	
3.	
4.	
5.	
6.	
7.	
8.	
9.	
10.	
11.	X
12.	X
13.	X
14.	X
15.	X
16.	X
17.	
18.	

*A check indicates the calculation and edit is to be made.

Fig. A-4, Configuration 3: Na Voided Inner Core, Outer Core, and Axial Blanket.

Assignment of Compositions to Regions (Refer to Table I)

<u>Region</u>	<u>Composition</u>	<u>- Cross Section Set</u>
Inner Core	Na In	A
Outer Core	Na In	C
Radial Blanket	Radial Blanket	G
Axial Blanket	Na In	E
Reflector	Refl	G
Control Rod	Na-Filled Control Rod Position	A

Required Calculations and Edits (Refer to Section D)

Calc. and Edit No.*

1. X
- 2.
- 3.
- 4.
- 5.
- 6.
- 7.
- 8.
- 9.
- 10.
- 11.
- 12.
- 13.
- 14.
- 15.
16. X
- 17.
- 18.

*A check indicates the calculation and edit is to be made.

Fig.A-5. Configuration 4: Reference with Central Na-Filled Control Rod Position.

Assignment of Compositions to Regions (Refer to Table I)

<u>Region</u>	<u>Composition</u>	<u>Cross Section Set</u>
Inner Core	Na In	A
Outer Core	Na In	C
Radial Blanket	Radial Blanket	G
Axial Blanket	Na In	E
Reflector	keff	G
Central Rod	B-10 Control Rod	A

Required Calculations and Edits (Refer to Section D)

Calc. and Edit No.*

1. X
- 2.
- 3.
- 4.
- 5.
- 6.
- 7.
- 8.
- 9.
- 10.
- 11.
- 12.
- 13.
- 14.
- 15.
16. X
- 17.
- 18.

*A check indicates the calculation and edit is to be made.

Fig. A-6, Configuration 5: Reference with Central Boron Control Rod.

Assignment of Compositions to Regions (Refer to Table I)

<u>Region</u>	<u>Composition</u>	<u>Cross Section Set</u>
Inner Core	Na Void	B
Outer Core	Na Void	D
Radial Blanket	Radial Blanket	G
Axial Blanket	Na Void	F
Reflector	Refl	G
Control Rod	Control Rod	B

Required Calculations and Edits (Refer to Section D)

Calc. and Edit No.*

1. X
- 2.
- 3.
- 4.
- 5.
- 6.
- 7.
- 8.
- 9.
- 10.
- 11.
- 12.
- 13.
- 14.
- 15.
16. X
- 17.
- 18.

*A check indicates the calculation and edit is to be made.

Fig. A-7, Configuration 6: Na Voided Inner Core, Outer Core and Axial Blanket with Control Rod Inserted.

Assignment of Compositions to Regions (Refer to Table I)

<u>Region</u>	<u>Composition</u>	<u>Cross Section Set</u>
Inner Core	Na In	H
Outer Core	Na In	A ^H C ^H
Radial Blanket	Radial Blanket	G
Axial Blanket	Na In	E
Reflector	Refl	G

Required Calculations and Edits (Refer to Section D)

Calc. and Edit No.*

1. X
- 2.
- 3.
- 4.
- 5.
- 6.
- 7.
- 8.
- 9.
- 10.
- 11.
- 12.
- 13.
- 14.
15. X
- 16.
- 17.
- 18.

*A check indicates the calculation and edit is to be made.

Fig.A-8. Configuration 7: Reference with Hot Fuel.

Assignment of Compositions to Regions (Refer to Table I)

<u>Region</u>	<u>Composition</u>	<u>Cross Section Set</u>
Inner Core	Na Void	B ^H
Outer Core	Na Void	D ^H
Radial Blanket	Radial Blanket	G
Axial Blanket	Na Void	F
Reflector	Ref1	G

Required Calculations and Edits (Refer to Section D)

Calc. and Edit No.*

1. X
- 2.
- 3.
- 4.
- 5.
- 6.
- 7.
- 8.
- 9.
- 10.
- 11.
- 12.
- 13.
- 14.
15. X
- 16.
- 17.
- 18.

*A check indicates the calculation and edit is to be made.

Fig.A-9. Configuration 8: Na Voided Inner Core, Outer Core, and Axial Blanket with Hot Fuel.

C. Cross Section Preparation

1. The primary purpose of this comparison is to evaluate the effects of variations in basic nuclear data. It is, therefore, desirable to minimize the differences in cross section preparation techniques. The prescription described below was used at ANL for the sample calculation. If possible, it is suggested that the participants follow this prescription in their preparation of broad group cross sections. In any case, briefly describe the cross section generation procedure used.
 - a. P1 fundamental mode calculation in the inner core and outer core regions (i.e., search to critical buckling).
 - b. In the blanket regions set $B^2 = 0$ and perform a fixed source slowing down calculation using the ^{239}Pu fission spectrum as the source.
 - c. Use radial blanket cross sections for the reflector region.
 - d. Group structure: Set top of upper group at 10 MeV. Use 27 groups each with $\Delta u = 0.5$. No thermal group. The Bondarenko structure is recommended as a second choice.
 - e. Diffusion coefficient = $1/3 \Sigma_{tr}$.
2. Seven sets of broad group isotopic microscopic cross sections are to be prepared by collapsing fine group sets over the spectra calculated for compositions A through G of Table A-I. In these sets all isotopes are assumed to be at a temperature of 1100°K.
3. Four additional sets of broad group isotopic microscopic cross sections are to be generated. These are denoted by A^H , B^H , C^H , and D^H in Table A-I, and are computed for an assumed temperature of 2200° for the isotopes:

^{235}U , ^{238}U , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu , and ^{16}O and an assumed temperature of 1100°K for all other isotopes. The compositions are the same as A, B, C, and D, respectively. These cross section sets are to be used in configurations 7 and 8 to determine the fuel Doppler coefficient.

4. The following information on the cross section preparation is requested.

- a. Identify the basic data (e.g., ENDF/B-IV) from which the broad group cross sections have been derived.
- b. Provide a brief description of the technique used for the preparation of the broad group cross sections.
- c. Tabulate the inner core broad group cross sections as follows:
 - 1) Fissile and fertile isotopes: σ_c and σ_f .
 - 2) Other isotopes: σ_{tr} .
 - 3) Fission source spectrum. Also indicate the functional form and the temperature, as appropriate, of the fission spectrum.
- d. Calculate $k_\infty = k_{eff}|_{B^2 = 0}$ based on broad group set of cross sections for inner core composition (Set A).

D. Calculations and Edits

Following are listed the calculations and formulas for edits requested for the comparison. The eight configurations to be calculated are shown in Figs. A-2 through A-9. The requested edits are listed on each figure by edit number corresponding to the numbered items below. Definitions of symbols used below are provided in Section E. In all edit items except numbers 5, 6, 7, and 18 capture (c) is meant to mean the sum of (n, γ), (n,p), and (n, α) events as per the definitions given in Section E.4. However, in edit items 5, 6, 7, and 18 (Eqns. 11 and 12) capture (c) is meant to mean (n, γ) events only.

1. $k_{\text{eff}} \cdot \left(\text{Convergence } |k_{\text{eff}_{i+1}} - k_{\text{eff}_i}| < 10^{-5} \right) .$
2. (a) Fission source, net $(n, 2n)$ source, fission, (n, γ) , (n, α) , (n, p) , and capture = $(n, \gamma) + (n, \alpha) + (n, p)$ integrals by isotope for each reactor region.
 (b) Total fission source, net $(n, 2n)$ source, capture, and fission integrals for each region and reactor totals.

Normalize such that reactor fission source $S = 1.0$. The normalization on source, $S = 1.0$, sets the normalization on flux to be used in all edits, unless a different normalization is specifically specified. (See Section E for a definition of symbols.)

3. Net leakage from the core.

$$L_c = 1/k_{\text{eff}} [S_1 + S_2] + S_1^{2n} + S_2^{2n} - [C_1 + C_2 + F_1 + F_2]$$

4. Net leakage from the reactor.

$$L_r = \sum_{j=1,5} \left(\frac{S_j}{k_{\text{eff}}} + S_j^{2n} \right) - \sum_{j=1,5} (C_j + F_j)$$

5. Reactor Breeding Ratio.

$$BR = \frac{{}^{28}\text{C} + {}^{40}\text{C}}{{}^{49}\text{C} + {}^{41}\text{C} + {}^{25}\text{C} + {}^{49}\text{F} + {}^{41}\text{F} + {}^{25}\text{F}}$$

6. Regional components of breeding ratio.

$$BR_j = \frac{{}^{28}\text{C}_j + {}^{40}\text{C}_j}{{}^{49}\text{C}_j + {}^{41}\text{C}_j + {}^{25}\text{C}_j + {}^{49}\text{F}_j + {}^{41}\text{F}_j + {}^{25}\text{F}_j}, \quad \text{region, } j = 1, 2, \dots, 4$$

7. Conversion ratios for inner and outer core regions.

$$CR_j = \frac{{}^{28}\text{C}_j + {}^{40}\text{C}_j}{{}^{49}\text{C}_j + {}^{41}\text{C}_j + {}^{25}\text{C}_j + {}^{49}\text{F}_j + {}^{41}\text{F}_j + {}^{25}\text{F}_j}, \quad j = 1, 2$$

8. (a) The broad group flux spectrum (flux/unit lethargy, $\phi^i/\Delta u^i$) evaluated at $z = 0$, $r = 0$, normalized to unit flux

$$\left(\sum_{i=1}^I \frac{\phi^i}{\Delta u^i} \Delta u^i = 1.0 \right)$$

- (b) The broad group adjoint flux spectrum (flux/unit lethargy, ϕ^{*i}) evaluated at $z = 0$, $r = 0$, normalized to unit adjoint

$$\left(\sum_{i=1}^I \phi^{*i} \Delta u^i = 1.0 \right)$$

9. Radial fission rate distribution. Fissions/cm³-sec versus radius at $z = 0$. Normalize such that the fission/cm³-sec at $r = 0$, $z = 0$ is 1.0.
10. Central reaction rate ratios. Per atom ratios of ²³⁸U capture to ²³⁹Pu fission, ²³⁸U fission to ²³⁹Pu fission, and ²³⁹Pu capture to ²³⁹Pu fission at $r = 0$ and $z = 0$.
11. Region sodium void worths.
- a.
$$\frac{[(k_{\text{eff}} \text{ for configuration 2}) - (k_{\text{eff}} \text{ for configuration 1})]}{(k_{\text{eff}} \text{ for configuration 1})}$$
- b.
$$\frac{[(k_{\text{eff}} \text{ for configuration 3}) - (k_{\text{eff}} \text{ for configuration 1})]}{(k_{\text{eff}} \text{ for configuration 1})}$$
12. First order perturbation central ($r = 0$ and $z = 0$) reactivity worths for ²³⁹Pu, ²³⁸U, Na, Fe and ¹⁰B. List the total worth in units of $\delta k/k$ per 10^{24} atoms according to Eq. (1).

First Order Perturbation Theory Formulas ($\delta k/k$ per unit volume per 10^{24}

atoms of isotope ℓ at position $\vec{r}$).

$$\left. \frac{1}{\Delta V(\vec{r})} \frac{1}{k} \frac{\delta k}{\delta N} \right|_{\vec{r}} = \frac{{}^{\ell}P(\vec{r})}{D} \text{ Total Worth} \quad (1)$$

$$\left. \frac{1}{\Delta V(\vec{r})} \frac{1}{k} \frac{\delta k}{\delta N} \right|_{\vec{r}} = \frac{{}^{\ell}NL(\vec{r})}{D} \text{ Non-leakage Component} \quad (2)$$

$$\left. \frac{1}{\Delta V(\vec{r})} \frac{1}{k} \frac{\delta k}{\delta N} \right|_{\vec{r}} = \frac{{}^{\ell}L(\vec{r})}{D} \text{ Leakage Component} \quad (3)$$

Where D is the perturbation denominator:

$$D = 1/k \int_{\text{Reactor}} d\vec{r} \sum_{i=1}^I \chi(\vec{r}) \phi^{*i}(\vec{r}) \sum_{i'=1}^I v \Sigma_f^{i'}(\vec{r}) \phi^{i'}(\vec{r}) \quad (4)$$

and ${}^{\ell}P(\vec{r})$ is the isotope ℓ perturbation numerator per 10^{24} atoms per unit volume at position $\vec{r}$:

$${}^{\ell}P(\vec{r}) = {}^{\ell}NL(\vec{r}) + {}^{\ell}L(\vec{r}) \quad (5)$$

consisting of the non-leakage component

$$\begin{aligned} {}^{\ell}NL(\vec{r}) = & \frac{1}{k} \sum_{i=1}^I \chi^i(\vec{r}) \phi^{*i}(\vec{r}) \sum_{i'=1}^I v \sigma_f^{i'}(\vec{r}) \phi^{i'}(\vec{r}) \\ & + \sum_{i=1}^I \sum_{i'=1}^I {}^{\ell}\sigma_s^{i' \rightarrow i}(\vec{r}) \phi^{i'}(\vec{r}) \left(\phi^{*i}(\vec{r}) - \phi^{*i'}(\vec{r}) \right) \\ & - \sum_{i=1}^I \left({}^{\ell}\sigma_c^i(\vec{r}) + {}^{\ell}\sigma_f^i(\vec{r}) \right) \phi^i(\vec{r}) \phi^{*i}(\vec{r}) \end{aligned} \quad (6)$$

and the leakage component:

$${}^{\ell}L(\vec{r}) = + \sum_{i=1}^I \nabla \phi^{*i}(\vec{r}) \cdot \nabla \phi^i(\vec{r}) \frac{1}{3(D^i)^2} {}^{\ell}\sigma_{tr}^i \quad (7)$$

13. Per atom axial reactivity worth profiles using first order perturbation theory (Eq. 1) for ^{239}Pu , ^{238}U and Fe at $r = 0$, normalized to 1.0 at $z = 0$.
14. (a) Per atom radial (at $z = 0$) and axial (at $r = 0$) reactivity worth profiles using first order perturbation theory for ^{10}B , Na, (Eq. 1).
- (b) The leakage component (Eq. 3) of sodium and the non-leakage component (eq. 2) of sodium.

For ^{10}B and Na normalize to 1.0 at $r = 0$ and $z = 0$. Normalize the leakage and non-leakage components of sodium such that their algebraic sum is 1.0 at $r = 0$ and $z = 0$. The leakage and non-leakage components of sodium worth are to be shown as a function of r at $Z = 0$ and as a function of Z at $r = 0$.

15. Isothermal Core Fuel Doppler Reactivities

- (a) Na In Doppler

$$\frac{(k_{\text{eff}} \text{ for configuration 7}) - (k_{\text{eff}} \text{ for configuration 1})}{(k_{\text{eff}} \text{ for configuration 1})}$$

- (b) Na Void Doppler

$$\frac{(k_{\text{eff}} \text{ for configuration 8}) - (k_{\text{eff}} \text{ for configuration 3})}{(k_{\text{eff}} \text{ for configuration 3})}$$

16. Reactivity worth of central control rod in Reference and Sodium-Voided Cores.

(a) Na In, Rod Worth Relative to Fuel

$$\frac{[(k_{\text{eff}} \text{ for configuration 5}) - (k_{\text{eff}} \text{ for configuration 1})]}{(k_{\text{eff}} \text{ for configuration 1})}$$

(b) Na In, Rod Worth Relative to Na Filled Control Rod Position

$$\frac{[(k_{\text{eff}} \text{ for configuration 5}) - (k_{\text{eff}} \text{ for configuration 4})]}{(k_{\text{eff}} \text{ for configuration 4})}$$

(c) Na Void, Rod Worth Relative to Fuel

$$\frac{(k_{\text{eff}} \text{ for configuration 6}) - (k_{\text{eff}} \text{ for configuration 3})}{(k_{\text{eff}} \text{ for configuration 3})}$$

17. The effective delayed neutron fraction and the inhour of reactivity.

(a) The effective delayed neutron fraction given by Eq. 9:

where:

$$\beta_m = \frac{1}{D} \int_{\text{reactor}} d\vec{r} \left\{ \sum_{\ell} \left[\sum_{i=1}^I \chi_{d,m}^{\ell i} \phi^{*i}(\vec{r}) \left(\sum_{i'=1}^I \ell_{N(\vec{r})}^{\ell} v_{d,m}^{i'} \sigma_f^{i'} \right) \right] \right\} \quad (8)$$

and:

D is the perturbation denominator given by Eq. 4,

$\ell v_{d,m}^i$ is the number of delayed neutrons emitted into delayed group m when isotope ℓ fissions upon absorption of a neutron having energy corresponding to energy group i.

$\ell \chi_{d,m}^i$ is the fraction of the group-m delayed neutrons (which were generated by fission of isotope ℓ by neutrons of any energy) which are emitted at an energy corresponding to energy group i.

$$\beta_{\text{eff}} = \sum_m \beta_m \quad (9)$$

(b) The inhour of reactivity is given by

$$\rho(3600) \approx \sum_m \frac{\beta_m}{1 + \lambda_m \frac{3600}{3600}} \quad (10)$$

where

λ_m is the decay constant for delayed neutron group m

A negligible term, (generation time/3600), has been neglected in Eq. 10.

18. Breeding Gains (See Reference 1)

(a) Internal breeding gain G_I

$$G_I = \frac{\sum_{j=1,2} \sum_{\ell \neq 28} \ell W_j \left(\ell^{-1} C_j - \ell A_j \right)}{F} \quad (11)$$

(b) Blanket breeding gain, G_B

$$G_B = \frac{\sum_{j=3,4} \sum_{\ell \neq 28} \ell W_j \left(\ell^{-1} C_j - \ell A_j \right)}{F} \quad (12)$$

(c) Total Breeding gain G

$$G = G_I + G_B \quad (13)$$

Here, the sum over isotopes in the numerator is over all heavy isotopes except ^{238}U :

$$\ell = 25, 49, 40, 41, 42$$

and ℓW is an equivalent worth relative to ^{239}Pu computed from the formula

$$\ell_W = \frac{\ell_X - 28X}{49X - 28X} \quad (14)$$

where the X's are evaluated from the core contributions to the neutron balance:

$$\ell_X = \frac{\sum_{j=1,2} (\ell_{S_j} - \ell_{A_j})}{\sum_{j=1,2} \ell_{N_j} \Delta V_j} \quad (15)$$

As discussed in Reference 1 the breeding gain is related to the breeding ratio minus one except that the effectiveness in maintaining the chain reaction (relative to ^{239}Pu) of the heavy metal atoms is accounted for.

E. Definitions of Symbols

1. General

"Reactor" is defined to include the core, blanket, and reflector regions, i.e., the entire model.

$\vec{r}$ denotes spatial position.

z denotes axial dimension.

r denotes radial dimension.

$\ell_{N(\vec{r})}$ denotes number density of isotope, ℓ , at position, $\vec{r}$.

$\phi^i(\vec{r})$ neutron flux in group i, at position, $\vec{r}$

$\phi^{*i}(\vec{r})$ adjoint in group i at position, $\vec{r}$.

Δu^i lethargy width of energy group, i.

k_{eff} eigenvalue

χ^i fission neutron emission spectrum into energy group, i.

2. Subscripts

j reactor region index

j = 1 inner core

j = 2 outer core

j = 3 axial blanket

j = 4 radial blanket

j = 5 reflector

3. Superscripts

i denotes energy group

l isotope index

l = 25 denotes ^{235}U

l = 28 " ^{238}U

l = 49 " ^{239}Pu

l = 40 " ^{240}Pu

l = 41 " ^{241}Pu

4. Microscopic and Macroscopic Cross Sections

σ_c^{li} - microscopic capture cross section (includes (n, γ), (n,p) and (n, α)) for isotope, l, in energy group i.

σ_f^{li} - microscopic fission cross section for isotope, l, in energy group i.

σ_{tr}^{li} - microscopic transport cross section for isotope, l, in energy group i.

$\nu \Sigma_f^i(\vec{r})$ macroscopic fission cross section times number of neutrons emitted per fission.

5. Region, Isotopic, Reaction Rate Integrals

$${}^{\ell}C_j - \text{Region } j \text{ capture integral for isotope, } \ell. \quad \int_{\text{region } j} d\vec{r} \sum_{i=1}^I {}^{\ell}N(\vec{r}) {}^{\ell}\sigma_c^i \phi^i(\vec{r})$$

$${}^{\ell}F_j - \text{Region fission integral for isotope } \ell \quad \int_{\text{region } j} d\vec{r} \sum_{i=1}^I {}^{\ell}N(\vec{r}) {}^{\ell}\sigma_f^i \phi^i(\vec{r})$$

$${}^{\ell}S_j - \text{Region fission source integral, (not normalized by } k_{\text{eff}}) \text{ for isotope } \ell$$

$${}^{\ell}S_j = \int_{\text{region } j} d\vec{r} \sum_i {}^{\ell}N(\vec{r}) {}^{\ell}\nu\sigma_f^i \phi^i(\vec{r})$$

$${}^{\ell}R_j^{2n} = \text{Region } (n,2n) \text{ removal integral for isotope } \ell.$$

$${}^{\ell}R_j^{2n} = \int_{\text{region } j} d\vec{r} \sum_i {}^{\ell}N(\vec{r}) {}^{\ell}\sigma_{n,2n}^i$$

$${}^{\ell}S_j^{2n} - \text{Region } \underline{\text{net}} \text{ } (n,2n) \text{ source integral for isotope } \ell$$

$${}^{\ell}S_j^{2n} \quad {}^{\ell}R_j^{2n}$$

6. Region Reaction Rate Integrals (Summed over Isotopes)

$$C_j$$

$$F_j$$

$$S_j$$

$$S_j^{2n}$$

These are defined the same as in item (5) except that they are summed over isotopes

e.g., C_j - Region, j , capture integral

$$C_j = \sum_{\ell} {}^{\ell}C_j$$

7. Reactor Reaction Rate Integrals by Isotope (Summed over Regions) L_C L_F L_S ${}^L_{S^{2n}}$

These are the same as in item (5) except that they are summed over all regions by isotope; e.g.,

$${}^L_C = \sum_j {}^L_{C_j}$$

8. Reactor Total Reaction Rate Integrals C F S S^{2n}

These are the same as in item (5) except that they are summed over all regions and isotopes; e.g.,

$$C = \sum_L \sum_j {}^L_{C_j}$$

9. Breeding Ratio and Conversion Ratio

$$\text{BR - reactor breeding ratio} = \frac{\sum_j ({}^{28}C_j + {}^{40}C_j)}{\sum_j [({}^{49}C_j + {}^{41}C_j + {}^{25}C_j) + ({}^{49}F_j + {}^{41}F_j + {}^{25}F_j)]}$$

$$\text{BR}_j \text{ - region } j \text{ component of the breeding ratio} = \frac{{}^{28}C_j + {}^{40}C_j}{\sum_j [({}^{49}C_j + {}^{41}C_j + {}^{25}C_j) + ({}^{49}F_j + {}^{41}F_j + {}^{25}F_j)]}$$

$$\text{CR}_j \text{ - region } j \text{ conversion ratio} = \frac{{}^{28}C_j + {}^{40}C_j}{({}^{49}C_j + {}^{41}C_j + {}^{25}C_j) + ({}^{49}F_j + {}^{41}F_j + {}^{25}F_j)}$$

10. Perturbation Theory

(See Eqs. 1 through 8)

D perturbation denominator (Eq. 4)

ℓ_P perturbation numerator per unit volume per 10^{24} atoms of isotope, ℓ (Eqs. 1, 5)

ℓ_{NL} non-leakage component of perturbation numerator (Eq. 6)

ℓ_L leakage component of perturbation numerator (Eq. 7).

F. References

1. A. R. Baker and R. W. Ross, "Comparison of the Value of Plutonium and Uranium Isotopes in Fast Reactors," p. 329, Proceedings of the Conference on Breeding, Economics, and Safety in Large Fast Power Reactors, October 7-10, 1963, at Argonne National Laboratory, ANL-6792, (December 1963).

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APPENDIX B

TABULATED RESULTS

Results of the contributed solutions are given in the following tables.

TABLES

C.4.c.	Summary of One-Group-Averaged Cross Sections.
D.1.	Summary of Calculated Eigenvalues.
D2.a-1.	Inner Core Fission Production by Isotope.
D2.a-2.	Inner Core (n,f) by Isotope.
D2.b,D3,D4.	Summary of Neutron Balance.
D5,D6,D7.	Breeding Ratio by Region for Configuration 1.
D10.	Summary of Calculated Central Reaction Rate Ratios.
D11.	Region Sodium Void Worths.
D12.	Central Reactivity Worths for ^{239}U , ^{238}U , Na, Fe, and ^{10}B .
D15.	Isothermal Core Fuel Doppler Reactivities.
D16.	Reactivity Worth of Central Control Rod.
D17.	Effective Delayed Neutron Fraction and the Inhour of Reactivity for Configuration 1.
D18.	Breeding Gains.

Note that each table is represented in two formats. First, the values contributed by participants are listed with the mean and standard deviation of these values. Values represented by asterisks (*****) were not submitted by that participant. Next, the values are reported as percent differences relative to the mean value.

Table C.4.c. Summary Of One-Group-Averaged Cross Sections.

Organization	Cross Sections										
	k-infinity Inner Core	k(Phi) in keV	²³⁸ U Capture	²³⁸ U Fission	²³⁹ Pu Capture	²³⁹ Pu Fission	²³⁹ Pu Alpha	¹⁰⁸ B Transport	Fe Transport	Na Transport	O Transport
ANL	1.10850	107.720	0.30697	0.04031	0.56781	1.84297	0.30810	5.44836	4.01792	4.15189	3.53076
BELGIUM	1.09500	108.005	0.33170	0.04004	0.45194	1.82496	0.24764	*****	4.71243	4.78848	3.74735
CADARACHE-1	1.15242	117.996	0.29234	0.03867	0.56750	1.81550	0.31259	*****	*****	*****	*****
CADARACHE-2	*****	*****	0.29550	0.03919	0.57910	1.81625	0.31884	*****	*****	*****	*****
CWEN	1.14558	113.793	0.30250	0.04518	0.57407	1.87984	0.30538	5.40132	4.07896	4.10724	3.53955
EIR-1	1.10442	109.117	0.30448	0.03995	0.57734	1.37503	0.30791	5.46198	5.36215	4.54541	3.58767
EIR-2	1.11735	111.779	0.32230	0.04158	0.58030	1.92367	0.30166	5.48159	4.54193	4.36690	3.50487
HEDL	1.11870	*****	*****	*****	*****	*****	*****	*****	*****	*****	*****
JAERI-1	1.11710	108.758	0.30991	0.04457	0.60820	1.89247	0.32309	5.52344	4.79261	4.45146	3.51690
JAERI-2	1.13564	108.916	0.29624	0.04048	0.52981	1.84266	0.28698	5.51229	3.93776	4.14213	3.49352
JAERI-3	1.13189	101.988	0.30905	0.04381	0.58704	1.89176	0.31731	5.79390	3.97266	4.38019	3.50360
KARLSRUHE-1	1.11840	108.797	0.30858	0.04097	0.57101	1.87191	0.30504	*****	3.12493	4.06499	3.34958
KARLSRUHE-2	1.14843	112.058	0.30406	0.04159	0.54818	1.85915	0.29328	*****	3.85887	4.55270	3.54034
SWEDEN	1.09680	108.845	0.30856	0.03868	0.54087	1.38136	0.28749	5.47947	3.76949	4.18664	3.48434
UKAEA	1.15123	111.991	0.29137	0.04302	0.54539	1.83265	0.29760	5.41029	3.53615	4.09901	3.31038
USSR	1.13270	118.106	0.30915	0.04009	0.49816	1.75992	0.28146	5.35558	4.06790	4.29630	3.44045
Mean	1.12429	110.562	0.30617	0.04121	0.55505	1.85467	0.29016	5.48682	4.13644	4.31794	3.50379
Std. Dev.	0.02054	4.213	0.01069	0.00207	0.03909	0.03889	0.01250	0.11975	0.59546	0.22086	0.10543

Table C.4.c. Summary Of One-Group-Averaged Cross Sections.

Organization	Cross Sections										
	k-infinity Inner Core	E(Phi) in keV	²³⁸ U Capture	²³⁸ U Fission	²³⁹ Pu Capture	²³⁹ Pu Fission	²³⁹ Pu Alpha	¹⁰ B Transport	Fe Transport	Na Transport	O Transport
Mean	1.12428	110.562	0.30617	0.04121	0.55505	1.85467	0.29916	5.48682	4.13644	4.31794	3.50379
	Percent Difference Relative To Reference ¹										
ANL	-1.403	-2.571	0.227	-2.181	2.299	-0.631	2.988	-0.701	-2.865	-3.846	0.770
BELGIUM	-2.604	-2.313	8.337	-2.836	-18.576	-1.602	-17.220	*****	13.925	10.897	6.951
CADARACHE-1	2.503	6.723	-4.518	-6.160	2.243	-2.112	4.489	*****	*****	*****	*****
CADARACHE-2	*****	*****	-3.486	-4.899	4.333	-2.071	6.580	*****	*****	*****	*****
CNEN	1.895	2.923	-1.200	9.637	3.427	1.357	2.081	-1.558	-1.390	-4.880	1.021
EIR-1	-1.766	-1.307	-0.553	-3.054	4.016	1.098	2.926	-0.453	29.632	5.268	2.394
EIR-2	-0.616	1.100	5.257	0.901	4.550	3.720	0.837	-0.095	9.803	1.134	0.031
HEDL	-0.496	*****	*****	*****	*****	*****	*****	*****	*****	*****	*****
JAERI-1	-0.638	-1.631	1.220	8.157	9.576	1.499	7.999	0.667	15.863	3.092	0.374
JAERI-2	1.011	-1.489	-3.244	-1.768	-4.727	-0.647	-4.070	0.464	-4.803	-4.072	-0.293
JAERI-3	0.677	-7.755	0.939	6.313	5.764	2.000	3.729	5.597	-3.959	1.442	-0.005
KARLSRUHE-1	-0.523	-1.597	0.786	-0.579	2.876	0.930	1.967	*****	-24.454	-5.858	-4.401
KARLSRUHE-2	2.148	1.353	-0.690	0.925	-1.237	0.781	-1.965	*****	-6.710	5.437	1.043
SWEDEN	-3.333	-1.553	0.779	-6.136	-2.554	1.439	-3.901	-0.134	-8.871	-3.041	-0.555
UKAEA	2.398	1.293	-4.835	4.396	-1.740	-1.187	-0.522	-1.395	-14.512	-5.070	-5.520
USSR	0.749	6.824	0.972	-2.715	-10.249	-4.570	-5.916	-2.392	-1.657	-0.501	-1.808

¹Percent difference is taken to be $\frac{(C.S. - C.S.(\text{Mean}))}{C.S.(\text{Mean})} \times 100\%$.

Table D1. Summary of Calculated Eigenvalues.

Organization	Configuration ¹							
	1	2	3	4	5	6	7	8
	Eigenvalue							
ANL	0.99304	1.01648	1.01737	0.99246	0.98971	1.01281	0.98591	1.01279
BELGIUM	0.98413	0.99817	0.99559	0.98362	0.98375	0.99113	0.97721	0.99128
CADARACHE-1	1.02124	1.04427	1.04400	1.02057	1.01689	1.03827	1.01354	1.03893
CADARACHE-2	1.01274	1.03463	1.03425	1.01247	1.00886	1.02865	1.00560	1.02948
CNEN	1.00164	1.02450	1.02372	1.00105	0.99826	1.01960	0.99422	1.02004
EIR-1	0.99459	1.01874	1.02025	0.99407	0.99150	1.01591	0.98852	1.01685
EIR-2	1.00490	1.02458	1.02447	1.00436	1.00171	1.02007	0.99718	1.01941
HEDL	1.00211	1.02193	1.02174	1.00156	0.99875	1.01724	0.99478	1.01677
JAERI-1	1.00660	1.02875	1.02967	1.00605	1.00341	1.02527	0.99867	1.02445
JAERI-2	1.01199	1.03586	1.03654	1.01140	1.00882	1.03218	1.00531	1.03225
JAERI-3	1.01442	1.03830	1.03937	1.01382	1.01127	1.03507	1.00664	1.03403
KARLSRUHE-1	0.99877	1.02001	1.01941	0.99817	0.99527	1.01478	0.99195	1.01516
KARLSRUHE-2	1.02400	1.04376	1.04271	1.02336	1.02004	1.03756	1.01747	1.03850
SWEDEN	0.98051	1.00181	1.00183	0.97992	0.97716	1.00134	0.97118	0.99885
UKAEA	1.02232	1.04325	1.04283	1.02167	1.01834	1.03763	1.01488	1.03835
USSR	1.00850	1.02700	1.02520	1.00770	1.00380	1.02000	1.00200	1.02180
Mean	1.00509	1.02638	1.02618	1.00452	1.00153	1.02172	0.99792	1.02181
Std. Dev.	0.01292	0.01376	0.01408	0.01264	0.01264	0.01338	0.01306	0.01372

¹Configuration Key:

1. Reference.
2. Na-Voided Inner Core.
3. Na-Voided Inner Core, Outer Core and Axial Blanket.
4. Reference with Central Na-Filled Control Rod Position.

5. Reference with Central Boron Control Rod.
6. Na-Voided Inner Core, Outer Core and Axial Blanket with Control Rod Inserted.
7. Reference with Hot Fuel.
8. Na-Voided Inner Core, Outer Core and Axial Blanket with Hot Fuel.

Table D1. Summary of Calculated Eigenvalues.

Organization	Configuration ¹							
	1	2	3	4	5	6	7	8
	Eigenvalue							
Mean	1.00509	1.02638	1.02618	1.00452	1.00153	1.02172	0.99782	1.02181
	Eigenvalue Difference ²							
ANL	-0.01205	-0.00990	-0.00881	-0.01206	-0.01182	-0.00891	-0.01191	-0.00902
BELGIUM	-0.02096	-0.02821	-0.03059	-0.02069	-0.02078	-0.03059	-0.02061	-0.03053
CADARACHE-1	0.01615	0.01789	0.01782	0.01605	0.01536	0.01655	0.01572	0.01712
CADARACHE-2	0.00765	0.00325	0.00807	0.00795	0.00733	0.00693	0.00784	0.00767
CNEN	-0.00345	-0.00188	-0.00240	-0.00347	-0.00327	-0.00212	-0.00360	-0.00177
EIR-1	-0.01050	-0.00764	-0.00593	-0.01045	-0.01003	-0.00581	-0.00930	-0.00496
EIR-2	-0.00019	-0.00180	-0.00171	-0.00016	0.00016	-0.00165	-0.00064	-0.00240
HEDL	-0.00298	-0.00445	-0.00444	-0.00296	-0.00278	-0.00448	-0.00304	-0.00504
JAERI-1	0.00151	0.00237	0.00349	0.00153	0.00188	0.00355	0.00085	0.00264
JAERI-2	0.00690	0.00948	0.01036	0.00688	0.00729	0.01046	0.00749	0.01044
JAERI-3	0.00933	0.01192	0.01319	0.00930	0.00974	0.01335	0.00882	0.01222
KARLSRUHE-1	-0.00632	-0.00637	-0.00677	-0.00634	-0.00626	-0.00694	-0.00587	-0.00665
KARLSRUHE-2	0.01890	0.01739	0.01652	0.01885	0.01851	0.01584	0.01965	0.01669
SWEDEN	-0.02458	-0.02457	-0.02435	-0.02460	-0.02437	-0.02038	-0.02664	-0.02296
UKAEA	0.01723	0.01687	0.01665	0.01715	0.01681	0.01591	0.01706	0.01654
USSE	0.00341	0.00062	-0.00098	0.00318	0.00227	-0.00172	0.00418	-0.00001

¹Configuration Key:

1. Reference.	5. Reference with Central Boron Control Rod.
2. Na-Voided Inner Core.	6. Na-Voided Inner Core, Outer Core and Axial Blanket with Control Rod Inserted.
3. Na-Voided Inner Core, Outer Core and Axial Blanket.	7. Reference with Hot Fuel.
4. Reference with Central Na-Filled Control Rod Position.	8. Na-Voided Inner Core, Outer Core and Axial Blanket with Hot Fuel.

²Eigenvalue difference is taken to be $k - k(\text{Mean})$.

Table D2.a-1 Inner Core Fission Production by Isotope.

Organization	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	²³⁵ U	²³⁸ U	O	Na	Fe	Ni	Cr	Mn	Mo	Total
ANL	.41967	.02405	.09151	.00228	.00774	.08487	*****	*****	*****	*****	*****	*****	*****	.63012
BELGIUM	.40722	.01927	.10962	.00180	.00794	.08352	*****	*****	*****	*****	*****	*****	*****	.62938
CADARACHE-1	.42506	.02437	.09595	.00214	.00776	.08825	*****	*****	*****	*****	*****	*****	*****	.64353
CNEN	.41730	.02755	.09052	*****	.00764	.08826	*****	*****	*****	*****	*****	*****	*****	.63127
EIR-1	.41914	.02379	.09129	.00225	.00774	.08237	*****	*****	*****	*****	*****	*****	*****	.62657
EIR-2	.42104	.02242	.08993	.00222	.00781	.08441	*****	*****	*****	*****	*****	*****	*****	.62784
JAERI-1	.41455	.02443	.08827	.00232	.00785	.09109	*****	*****	*****	*****	*****	*****	*****	.62852
JAERI-2	.42220	.02400	.09127	.00221	.00736	.08441	*****	*****	*****	*****	*****	*****	*****	.63140
JAERI-3	.41743	.02435	.09084	.00229	.00788	.08940	*****	*****	*****	*****	*****	*****	*****	.63220
KARLSRUHE-1	.42580	.02254	.09154	.00202	.00793	.08635	*****	*****	*****	*****	*****	*****	*****	.63620
KARLSRUHE-2	.42180	.02446	.09273	.00217	.00783	.08825	*****	*****	*****	*****	*****	*****	*****	.63720
SWEDEN	.42420	.02210	.08980	.00220	.00790	.08020	*****	*****	*****	*****	*****	*****	*****	.62640
UKAEA	.41970	.02443	.09378	.00230	.00764	.09078	*****	*****	*****	*****	*****	*****	*****	.63863
USSR	.41090	.02130	.10870	.00200	.00790	.08780	*****	*****	*****	*****	*****	*****	*****	.63860
Mean	.41900	.02350	.09398	.00217	.00778	.08643	*****	*****	*****	*****	*****	*****	*****	.63270
Std. Dev.	.00529	.00192	.00669	.00015	.00016	.00323	*****	*****	*****	*****	*****	*****	*****	.00528

Table D2.a-1 Inner Core Fission Production by Isotope.

Organization	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	²³⁵ U	²³⁸ U	O	Na	Fe	Ni	Cr	Mn	Mo	Total
Reference Values														
Mean	.41900	.02350	.09398	.00217	.00778	.08643	*****	*****	*****	*****	*****	*****	*****	.63270
Percent Difference Relative to Reference ¹														
ANL	0.16	2.32	-2.63	5.09	-0.53	-1.80	*****	*****	*****	*****	*****	*****	*****	-0.41
BELGIUM	-2.81	-18.00	16.64	-16.94	2.09	-3.36	*****	*****	*****	*****	*****	*****	*****	-0.53
CADARACHE-1	1.45	3.68	2.09	-1.36	-0.27	2.11	*****	*****	*****	*****	*****	*****	*****	1.71
CNEN	-0.41	17.21	-3.68	*****	-1.82	2.12	*****	*****	*****	*****	*****	*****	*****	-0.23
EIR-1	0.03	1.22	-2.87	3.71	-0.54	-4.69	*****	*****	*****	*****	*****	*****	*****	-0.97
EIR-2	0.49	-4.61	-4.31	2.24	0.42	-2.33	*****	*****	*****	*****	*****	*****	*****	-0.77
JAERI-1	-1.06	3.94	-6.08	6.93	0.88	5.40	*****	*****	*****	*****	*****	*****	*****	-0.66
JAERI-2	0.76	2.11	-2.89	1.77	-5.36	-2.33	*****	*****	*****	*****	*****	*****	*****	-0.21
JAERI-3	-0.37	3.60	-3.34	5.70	1.29	3.44	*****	*****	*****	*****	*****	*****	*****	-0.08
KARLSRUHE-1	1.62	-4.10	-2.60	-6.99	1.94	-0.09	*****	*****	*****	*****	*****	*****	*****	0.55
KARLSRUHE-2	0.67	4.07	-1.33	0.25	0.69	2.11	*****	*****	*****	*****	*****	*****	*****	0.71
SWEDEN	1.24	-5.97	-4.45	1.40	1.53	-7.20	*****	*****	*****	*****	*****	*****	*****	-1.00
UKAEA	0.17	3.93	-0.21	6.03	-1.85	5.04	*****	*****	*****	*****	*****	*****	*****	0.94
USSR	-1.93	-9.38	15.66	-7.82	1.53	1.59	*****	*****	*****	*****	*****	*****	*****	0.93

¹Percent difference is taken to be $\frac{(R.R. - R.R.(\text{Mean}))}{R.R.(\text{Mean})} \times 100\%$.

Table D2.a-2 Inner Core Capture by Isotope.

Organization	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	²³⁵ U	²³⁸ U	O	Na	Fe	Ni	Cr	Mn	Mo	Total
ANL	.04472	.01289	.00611	.00112	.00098	.23784	.00150	.00224	.01295	.00709	.00642	.00212	.00398	.33996
BELGIUM	.03476	.01795	.00598	.00140	.00098	.25055	.00111	.00149	*****	.00602	.00282	*****	.00381	.34836
CADARACHE-1	.04549	.01247	.00706	.00184	.00096	.23249	.00163	.00230	.00938	.00436	.00269	.00197	.00389	.32653
CNEN	.04386	.01396	.00596	*****	.00095	.23221	.00187	.00233	.01396	.00711	.00593	.00211	.00383	.33409
EIR-1	.04456	.01267	.00594	.00114	.00095	.23137	.00134	.00141	.02024	.00704	.00892	.00154	.00403	.34116
EIR-2	.04378	.01004	.00561	.00107	.00097	.23949	.00147	.00223	.01321	.00684	.00582	.00200	.00368	.33620
JAERI-1	.04617	.01355	.00631	.00112	.00098	.23274	.00198	.00193	.01386	.00796	.00434	.00210	.00350	.33655
JAERI-2	.04160	.01159	.00652	.00098	.00104	.22960	.00145	.00369	.01285	.00697	.00632	.00199	.00376	.32840
JAERI-3	.04462	.01153	.00673	.00123	.00108	.23131	.00151	.00266	.01096	.00693	.00586	.00243	.00416	.33102
KARLSRUHE-1	.04480	.01437	.00619	.00136	.00098	.23770	*****	*****	*****	*****	*****	*****	*****	.33880
KARLSRUHE-2	.04273	.00959	.00734	.00170	.00093	.23250	*****	*****	*****	*****	*****	*****	*****	.32660
SWEDEN	.04230	.00990	.00550	.00100	.00100	.24820	.00120	.00210	.01460	.00690	.00620	.00200	.00740	.34830
UKAEA	.04314	.01413	.00737	.00112	.00085	.22619	.00120	.00135	.01356	.00566	.00255	.00206	.00384	.32302
USSR	.04000	.01730	.00570	.00130	.00100	.24390	.00200	.00160	.00760	.00520	.00310	*****	.00340	.33210
Mean	.04304	.01300	.00631	.00126	.00097	.23615	.00152	.00211	.01302	.00651	.00508	.00203	.00411	.33508
Std. Dev.	.00289	.00251	.00061	.00026	.00005	.00719	.00030	.00065	.00322	.00100	.00198	.00022	.00106	.00780

Table D2.a-2 Inner Core Capture by Isotope.

Organization	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	²³⁵ U	²³⁸ U	O	Na	Fe	Ni	Cr	Mn	Mo	Total
Reference Values														
Mean	.04304	.01300	.00631	.00126	.00097	.23615	.00152	.00211	.01302	.00651	.00508	.00203	.00411	.33508
Percent Difference Relative to Reference ¹														
ANL	3.91	-0.81	-3.15	-11.12	0.53	0.72	-1.44	6.09	-0.50	8.97	26.36	4.34	-3.10	1.46
BELGIUM	-19.23	38.12	-5.21	11.10	0.94	6.10	-27.33	-29.39	*****	-7.52	-44.54	*****	-7.16	3.96
CADARACHE-1	5.70	-4.05	11.91	46.02	-1.52	-1.55	7.10	8.93	-27.93	-32.99	-47.05	-3.04	-5.29	-2.55
CNEN	1.91	7.42	-5.53	*****	-2.55	-1.67	22.87	10.35	7.26	9.28	16.72	3.85	-6.75	-0.29
EIR-1	3.54	-2.51	-5.84	-9.53	-2.55	-2.02	-11.95	-33.22	55.51	8.20	75.56	-24.21	-1.88	1.82
EIR-2	1.72	-22.74	-11.08	-15.09	-0.50	1.41	-3.41	5.61	1.49	5.13	14.55	-1.57	-10.40	0.33
JAERI-1	7.28	4.26	0.02	-11.12	0.53	-1.44	30.10	-8.60	6.49	22.34	-14.58	3.36	-14.78	0.44
JAERI-2	-3.34	-10.82	3.32	-22.44	6.27	-2.77	-4.46	74.99	-1.27	7.20	24.47	-2.01	-8.43	-1.99
JAERI-3	3.68	-11.28	6.76	-2.31	11.30	-2.05	-0.45	26.07	-15.79	6.45	15.28	19.45	1.34	-1.21
KARLSRUHE-1	4.09	10.57	-1.88	8.01	0.69	0.66	*****	*****	*****	*****	*****	*****	*****	1.11
KARLSRUHE-2	-0.72	-26.19	16.33	35.07	-5.08	-1.55	*****	*****	*****	*****	*****	*****	*****	-2.53
SWEDEN	-1.71	-23.82	-12.82	-20.64	2.58	5.10	-21.15	-0.54	12.17	6.05	22.03	-1.57	80.17	3.95
UKAEA	0.24	8.73	16.82	-11.12	-13.22	-4.22	-21.28	-36.06	4.18	-13.01	-49.81	1.39	-6.50	-3.60
USSR	-7.06	33.12	-9.65	3.17	2.58	3.28	31.41	-24.22	-41.61	-20.08	-38.99	*****	-17.22	-0.89

¹Percent difference is taken to be $\frac{(R.R. - R.R.(\text{Mean}))}{R.R.(\text{Mean})} \times 100\%$.

Table D2.a-3 Inner Core (n,f) by Isotope.

Organization	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	²³⁵ U	²³⁸ U	O	Na	Fe	Ni	Cr	Mn	Mo	Total
ANL	.14341	.00784	.03081	.00075	.00317	.03071	*****	*****	*****	*****	*****	*****	*****	.21669
BELGIUM	.13839	.00642	.03672	.00059	.00323	.02983	*****	*****	*****	*****	*****	*****	*****	.21517
CADARACHE-1	.14552	.00818	.03234	.00069	.00318	.03076	*****	*****	*****	*****	*****	*****	*****	.22067
CNEN	.14251	.00897	.03047	*****	.00313	.03188	*****	*****	*****	*****	*****	*****	*****	.21695
EIR-1	.14317	.00776	.03075	.00074	.00316	.02994	*****	*****	*****	*****	*****	*****	*****	.21552
EIR-2	.14360	.00742	.03024	.00073	.00319	.03046	*****	*****	*****	*****	*****	*****	*****	.21564
JAERI-1	.14141	.00796	.00297	.00076	.00318	.03287	*****	*****	*****	*****	*****	*****	*****	.21589
JAERI-2	.14370	.00793	.03072	.00073	.00317	.03060	*****	*****	*****	*****	*****	*****	*****	.21680
JAERI-3	.14215	.00797	.03055	.00075	.00320	.03243	*****	*****	*****	*****	*****	*****	*****	.21706
KARLSRUHE-1	.14470	.00774	.03056	.00066	.00323	.03098	*****	*****	*****	*****	*****	*****	*****	.21790
KARLSRUHE-2	.14340	.00778	.03149	.00068	.00319	.03120	*****	*****	*****	*****	*****	*****	*****	.21780
SWEDEN	.14480	.00730	.03020	.00070	.00320	.02910	*****	*****	*****	*****	*****	*****	*****	.21540
UKAEA	.14285	.00776	.03186	.00076	.00311	.03282	*****	*****	*****	*****	*****	*****	*****	.21916
USSR	.14000	.00710	.03640	.00070	.00320	.03120	*****	*****	*****	*****	*****	*****	*****	.21860
Mean	.14283	.00772	.02972	.00071	.00318	.03106	*****	*****	*****	*****	*****	*****	*****	.21709
Std. Dev.	.00192	.00058	.00799	.00005	.00003	.00112	*****	*****	*****	*****	*****	*****	*****	.00165

Table D2.a-3 Inner Core (n,f) by Isotope.

Organization	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu	²³⁵ U	²³⁸ U	O	Na	Fe	Ni	Cr	Mn	Mo	Total
Reference Values														
Mean	.14283	.00772	.02972	.00071	.00318	.03106	*****	*****	*****	*****	*****	*****	*****	.21709
Percent Difference Relative to Reference ¹														
ANL	0.41	1.50	3.67	5.57	-0.36	-1.11	*****	*****	*****	*****	*****	*****	*****	-0.18
BELGIUM	-3.11	-16.93	23.55	-17.46	1.62	-3.94	*****	*****	*****	*****	*****	*****	*****	-0.88
CADARACHE-1	1.88	5.91	8.82	-2.87	-0.04	-0.95	*****	*****	*****	*****	*****	*****	*****	1.65
CNEN	-0.22	16.13	2.52	*****	-1.62	2.65	*****	*****	*****	*****	*****	*****	*****	-0.06
EIR-1	0.24	0.47	3.47	4.16	-0.67	-3.59	*****	*****	*****	*****	*****	*****	*****	-0.72
EIR-2	0.54	-3.93	1.75	2.76	0.27	-1.92	*****	*****	*****	*****	*****	*****	*****	-0.67
JAERI-1	-0.99	3.06	-90.01	6.98	-0.04	5.84	*****	*****	*****	*****	*****	*****	*****	-0.55
JAERI-2	0.61	2.67	3.36	2.76	-0.36	-1.47	*****	*****	*****	*****	*****	*****	*****	-0.13
JAERI-3	-0.48	3.19	2.79	5.85	0.58	4.42	*****	*****	*****	*****	*****	*****	*****	-0.01
KARLSRUHE-1	1.31	0.27	2.83	-7.29	1.43	-0.24	*****	*****	*****	*****	*****	*****	*****	0.37
KARLSRUHE-2	0.40	0.77	5.96	-3.93	0.27	0.46	*****	*****	*****	*****	*****	*****	*****	0.33
SWEDEN	1.38	-5.49	1.62	-1.47	0.58	-6.30	*****	*****	*****	*****	*****	*****	*****	-0.78
UKAEA	0.01	0.47	7.20	6.42	-2.25	5.68	*****	*****	*****	*****	*****	*****	*****	0.95
USSR	-1.98	-8.08	22.48	-1.47	0.58	0.46	*****	*****	*****	*****	*****	*****	*****	0.70

¹Percent difference is taken to be $\frac{(R.R. - R.R.(\text{Mean}))}{R.R.(\text{Mean})} \times 100\%$.

Table D2.b, D3, D4. Summary of Neutron Balance.

Organization	k_{Bsq}^1	Fission Production		Capture ²		Fission		Absorption		Leakage	
	Core	Core	Reactor	Core	Reactor	Core	Reactor	Core	Reactor	Core	Reactor
ANL	1.1901	0.9787	1.0000	0.4863	0.6490	0.3361	0.3441	0.8224	0.9930	0.1650	0.0160
BELGIUM	1.1739	0.9793	1.0000	0.4998	0.6592	0.3344	0.3421	0.8343	1.0013	0.1624	0.0165
CADARACHE-1	1.2298	0.9771	1.0000	0.4598	0.6173	0.3347	0.3430	0.7945	0.9603	0.1622	0.0189
CADARACHE-2	1.2207	0.9773	1.0000	0.4666	0.6270	0.3340	0.3423	0.8006	0.9693	0.1644	0.0181
CNEN	1.2034	0.9776	1.0000	0.4769	0.6398	0.3355	0.3439	0.8124	0.9837	0.1698	0.0147
EIR-1	1.1840	0.9793	1.0000	0.4907	0.6490	0.3364	0.3442	0.8271	0.9932	0.1590	0.0139
EIR-2	1.1966	0.9794	1.0000	0.4825	0.6400	0.3359	0.3436	0.8185	0.9837	0.1585	0.0141
HEDL	*****	*****	1.0000	0.4817	0.6423	0.3367	0.3446	0.8184	0.9869	*****	0.0149
JAERI-1	1.1947	0.9763	1.0000	0.4823	0.6399	0.3349	0.3437	0.8172	0.9835	0.1549	0.0124
JAFRI-2	1.2173	0.9791	1.0000	0.4687	0.6284	0.3357	0.3436	0.8044	0.9720	0.1640	0.0173
JAERI-3	1.2095	0.9776	1.0000	0.4731	0.6272	0.3352	0.3436	0.8082	0.9708	0.1575	0.0173
KARLSRUHE-1	1.1983	0.9772	1.0000	0.4812	0.6418	0.3343	0.3428	0.8155	0.9846	0.1629	0.0185
KARLSRUHE-2	1.2254	0.9765	1.0000	0.4635	0.6188	0.3334	0.3419	0.7969	0.9607	0.1569	0.0174
SWEDEN	1.1710	0.9814	1.0000	0.5010	0.6653	0.3371	0.3439	0.8381	1.0093	0.1644	0.0124
UKAEA	1.2337	0.9768	1.0000	0.4570	0.6168	0.3347	0.3434	0.7917	0.9602	0.1652	0.0196
USSR	1.2150	0.9783	1.0000	0.4707	0.6288	0.3345	0.3425	0.8052	0.9713	0.1648	0.0203
Mean	1.2042	0.9781	1.0000	0.4776	0.6369	0.3352	0.3433	0.8128	0.9802	0.1621	0.0164
Std. Dev.	0.0197	0.0016	0.0	0.0131	0.0144	0.0010	0.0008	0.0137	0.0147	0.0040	0.0025

¹ k_{Bsq} = Fission Production / Absorption

²Capture = (n,gamma) + (n,alpha) + (n,p)

Table D2.b, D3, D4. Summary of Neutron Balance.

Organization	¹ k Bsquared	Fission Production		Capture ²		Fission		Absorption		Leakage	
	Core	Core	Reactor	Core	Reactor	Core	Reactor	Core	Reactor	Core	Reactor
Reference Values											
Mean	1.2042	0.9781	1.0000	0.4776	0.6369	0.3352	0.3433	0.8128	0.9802	0.1621	0.0164
Eigenvalue Difference ³ Percent Difference Relative To Reference ⁴											
ANL	-0.0141	0.0571	0.0	1.8137	1.8926	0.2613	0.2220	1.1734	1.3075	1.7930	-2.1415
BELGIUM	-0.0303	0.1223	0.0	4.6524	3.5048	-0.2366	-0.3634	2.6361	2.1499	0.1647	0.5412
CADARACHE-1	0.0256	-0.1045	0.0	-3.7306	-3.0846	-0.1444	-0.0809	-2.2516	-2.0325	0.0722	15.1742
CADARACHE-2	0.0165	-0.0840	0.0	-2.3047	-1.5553	-0.3652	-0.2994	-1.5049	-1.1154	1.3982	10.3575
CNEN	-0.0009	-0.0549	0.0	-0.1546	0.4511	0.0854	0.1643	-0.0557	0.3506	4.7539	-10.3714
EIR-1	-0.0202	0.1184	0.0	2.7392	1.8989	0.3448	0.2453	1.7517	1.3197	-1.9139	-15.0674
EIR-2	-0.0076	0.1276	0.0	1.0327	0.4905	0.2135	0.0909	0.6948	0.3506	-2.2654	-14.2138
HEDI	*****	*****	0.0	0.8485	0.8438	0.4462	0.3647	0.6826	0.6760	*****	-8.9703
JAERI-1	-0.0096	-0.1893	0.0	0.9909	0.4639	-0.1086	0.0968	0.5374	0.3353	-4.4550	-24.2740
JAERI-2	0.0130	0.1051	0.0	-1.8713	-1.3402	0.1479	0.0880	-1.0386	-0.8400	1.1824	5.2969
JAERI-3	0.0053	-0.0564	0.0	-0.9522	-1.5192	-0.0102	0.0793	-0.5637	-0.9593	-2.8822	5.2969
KARLSRUHE-1	-0.0059	-0.0942	0.0	0.7522	0.7685	-0.2757	-0.1537	0.3282	0.4454	0.4731	13.0403
KARLSRUHE-2	0.0212	-0.1658	0.0	-2.9538	-2.8428	-0.5442	-0.4159	-1.9601	-1.9928	-3.2276	6.3334
SWEDEN	-0.0332	0.3393	0.0	4.9041	4.4629	0.5477	0.1812	3.1074	2.9631	1.3736	-24.5179
UKAEA	0.0295	-0.1382	0.0	-4.3189	-3.1615	-0.1444	0.0239	-2.5973	-2.0458	1.8917	19.7471
USSR	0.0108	0.0182	0.0	-1.4463	-1.2727	-0.2160	-0.2411	-0.9389	-0.9114	1.6450	23.7711

¹k = Fission Production / Absorption
Bsquared

²Capture = (n,gamma) + (n,alpha) + (n,p)

³Eigenvalue difference is taken to be $k - k(\text{Mean})$.

⁴Percent difference is taken to be $\frac{(R.R. - R.P.(\text{Mean}))}{R.R.(\text{Mean})} \times 100\%$.

Table D5, D6, D7. Breeding Ratio by Region for Configuration 1.

Organization	Conversion Ratio		Breeding Ratio						
	Inner Core	Outer Core	Inner Core	Outer Core	Core Subtotal	Radial Blanket	Axial Blanket	Blanket Subtotal	Reactor Total
ANL	1.09390	0.82102	0.70057	0.28900	(0.93957)	0.17312	0.23205	(0.40517)	1.39474
BELGIUM	1.22000	0.92544	0.77825	0.32779	(1.10604)	0.17786	0.23614	(0.41400)	1.52004
CADARACHE-1	1.04440	0.78730	0.68310	0.26630	(0.94940)	0.17050	0.23570	(0.40620)	1.35560
CADARACHE-2	1.08200	0.80200	0.70700	0.27200	(0.97900)	0.17260	0.24100	(0.41360)	1.39260
CNEN	1.08505	0.81563	0.69653	0.28576	(0.98229)	0.17295	0.23350	(0.40645)	1.38375
EIR-1	1.06790	0.80320	0.67970	0.28590	(0.96560)	0.16470	0.21740	(0.38210)	1.34770
EIR-2	1.09730	0.82210	0.69950	0.29190	(0.99140)	0.16830	0.22560	(0.39390)	1.38530
HEDL	*****	*****	0.69744	0.28687	(0.98431)	0.17017	0.23061	(0.40078)	1.38509
JAERI-1	1.08130	0.81278	0.69194	0.28659	(0.97853)	0.16750	0.22741	(0.39491)	1.37344
JAERI-2	1.06400	0.79850	0.67960	0.27800	(0.95760)	0.16970	0.22740	(0.39710)	1.35470
JAERI-3	1.06348	0.80252	0.68324	0.28059	(0.96383)	0.16561	0.22237	(0.38798)	1.35181
KARLSRUHE-1	1.09360	0.82710	0.70730	0.28560	(0.99290)	0.17130	0.23230	(0.40360)	1.39650
KARLSRUHE-2	1.05660	0.79650	0.68440	0.27440	(0.95880)	0.16850	0.22850	(0.39700)	1.35580
SWEDEN	1.13730	0.84510	0.72120	0.30330	(1.02450)	0.17540	0.23220	(0.40760)	1.43210
UKAEA	1.04862	0.79068	0.68132	0.27076	(0.95208)	0.17265	0.23500	(0.40765)	1.35973
USSR	1.15380	0.87170	0.75730	0.30430	(1.06160)	0.17550	0.24280	(0.41830)	1.47990
Mean	1.09262	0.82144	0.70302	0.28682	0.93984	0.17102	0.23125	0.40227	1.39211
Std. Dev.	0.04640	0.03623	0.02815	0.01519	0.04205	0.00364	0.00651	0.00972	0.04823

Table D5, D6, D7. Breeding Ratio by Region for Configuration 1.

Organization	Conversion Ratio		Breeding Ratio						
	Inner Core	Outer Core	Inner Core	Outer Core	Core Subtotal	Radial Blanket	Axial Blanket	Blanket Subtotal	Reactor Total
Reference Values									
Mean	1.09262	0.82144	0.70302	0.28682	(0.93984)	0.17102	0.23125	(0.40227)	1.39211
Percent Difference Relative to Reference ¹									
ANL	0.11757	-0.05086	-0.34907	0.76135	(-0.02734)	1.22665	0.34650	(0.72064)	0.18887
BELGIUM	11.65865	12.66100	10.70034	14.28568	(11.73911)	3.99822	2.11516	(2.91568)	9.18956
CADARACHE-1	-4.41289	-4.15585	-2.83405	-7.15312	(-4.03557)	-0.30530	1.92489	(0.97669)	-2.62267
CADARACHE-2	-0.97155	-2.36630	0.56555	-5.16579	(-1.09519)	0.92260	4.21681	(2.81626)	0.03514
CNEN	-0.69216	-0.70654	-0.92345	-0.36760	(-0.76241)	1.12551	0.97569	(1.03934)	-0.24169
EIR-1	-2.26204	-2.22022	-3.31768	-0.31947	(-2.44894)	-3.69668	-5.98867	(-5.01429)	-3.19017
EIR-2	0.42874	0.08062	-0.50127	1.77245	(0.15754)	-1.59169	-2.44271	(-2.08095)	-0.48920
HEDI	*****	*****	-0.79444	0.02045	(-0.55834)	-0.50061	-0.27533	(-0.37115)	-0.50413
JAERI-1	-1.03562	-1.05398	-1.57663	-0.07891	(-1.14267)	-2.05947	-1.66000	(-1.82987)	-1.34114
JAERI-2	-2.61903	-2.79239	-3.33190	-3.07385	(-3.25715)	-0.77309	-1.66433	(-1.28547)	-2.68734
JAERI-3	-2.66660	-2.30300	-2.81413	-2.17119	(-2.62786)	-3.16167	-3.84075	(-3.55209)	-2.89484
KARLSRUHE-1	0.09008	0.68931	0.60822	-0.42409	(0.30907)	0.16248	0.45462	(0.33038)	0.31533
KARLSRUHE-2	-3.29627	-3.03586	-2.64914	-4.32901	(-3.13592)	-1.47476	-1.18865	(-1.31033)	-2.60828
SWEDEN	4.08967	2.88059	2.58540	5.74711	(3.50150)	2.55981	0.41137	(1.32472)	2.87258
UKAEA	-4.02666	-3.74438	-3.08724	-5.59811	(-3.81482)	0.95184	1.62218	(1.33715)	-2.32597
USSR	5.59977	6.11882	7.72035	6.09576	(7.24957)	2.61829	4.99519	(3.98463)	6.30622

¹Percent difference is taken to be
$$\frac{(B.P. - B.R.(\text{Mean}))}{B.R.(\text{Mean})} \times 100\%$$

Table D10. Summary of Calculated Central Reaction Rate Ratios

Organization	^{238}U capture / ^{239}Pu fission	^{238}U fission / ^{239}Pu fission	^{239}Pu capture / ^{239}Pu fission
ANL	0.16650	0.02184	0.30806
BELGIUM	0.18200	0.02198	0.24810
CADARACHE-1	0.16020	0.02164	0.30820
CADARACHE-2	0.16200	0.02191	0.31449
CNEN	0.16359	0.02283	0.30382
EIR-1	0.16239	0.02130	0.30793
EIR-2	0.16755	0.02161	0.30165
JAERI-1	0.16529	0.02369	0.32308
JAERI-2	0.16083	0.02211	0.29803
JAERI-3	0.16340	0.02327	0.31044
KARLSRUHE-1	0.16487	0.02188	0.30511
KARLSRUHE-2	0.16413	0.02151	0.29776
SWEDEN	0.17200	0.02052	0.28972
UKAEA	0.15885	0.02347	0.29757
USSR	0.17470	0.02280	0.28150
Mean	0.16589	0.02216	0.29970
Std. Dev.	0.00615	0.00088	0.01742

Table D10. Summary of Calculated Central Reaction Rate Ratios

Organization	^{238}U capture / ^{239}Pu fission	^{238}U fission / ^{239}Pu fission	^{239}Pu capture / ^{239}Pu fission
Reference Values			
Mean	0.16589	0.02216	0.29970
Percent Difference Relative to Reference ¹			
ANL	0.36987	-1.43381	2.79678
BELGIUM	9.71362	-0.80198	-17.21667
CADARACHE-1	-3.42789	-2.33643	2.83682
CADARACHE-2	-2.34284	-1.11790	4.93560
CNEN	-1.38435	3.03416	1.37535
EIR-1	-2.10771	-3.85285	2.74674
EIR-2	1.00283	-2.45829	0.65130
JAERI-1	-0.35953	6.91542	7.80181
JAERI-2	-3.04813	-0.22430	-0.55659
JAERI-3	-1.49886	5.03346	3.58324
KARLSRUHE-1	-0.61273	-1.26233	1.80579
KARLSRUHE-2	-1.05881	-2.92314	-0.64669
SWEDEN	3.68541	-7.39111	-3.32937
UKAEA	-4.24173	5.92255	-0.71007
USSR	5.31301	2.39876	-6.07213

¹Percent difference is taken to be
$$\frac{(R.R. - R.R.(\text{Mean}))}{R.R.(\text{Mean})} \times 100\%$$

Table D11. Region Sodium Void Worths.

Organization	$k_{\text{Voided}} - k_{\text{Ref}}$		$(k_{\text{Voided}} - k_{\text{Ref}}) / k_{\text{Ref}}$	
	Inner Core	Inner Core, Outer Core and Axial Blanket	Inner Core	Inner Core, Outer Core and Axial Blanket
ANL	0.02344	0.02433	0.02360	0.02450
BELGIUM	0.01404	0.01146	0.01427	0.01164
CADARACHE-1	0.02303	0.02276	0.02255	0.02229
CADARACHE-2	0.02190	0.02151	0.02161	0.02127
CNEN	0.02286	0.02208	0.02282	0.02204
EIR-1	0.02415	0.02566	0.02428	0.02580
EIR-2	0.01968	0.01957	0.01958	0.01947
HEDL	0.01982	0.01978	0.01963	0.01959
JAERI-1	0.02214	0.02307	0.02200	0.02292
JAERI-2	0.02365	0.02455	0.02359	0.02426
JAERI-3	0.02388	0.02495	0.02354	0.02459
KARLSRUHE-1	0.02124	0.02064	0.02126	0.02067
KARLSRUHE-2	0.01976	0.01871	0.01930	0.01827
SWEDEN	0.02130	0.02132	0.02172	0.02174
UKAEA	0.02093	0.02051	0.02047	0.02006
USSR	0.01850	0.01670	0.01930	0.01660
Mean	0.02127	0.02110	0.02116	0.02098
Std. Dev.	0.00257	0.00354	0.00255	0.00352

Table D11. Region Sodium Void Worths.

Organization	$k_{\text{Voided}} - k_{\text{Ref}}$		$(k_{\text{Voided}} - k_{\text{Ref}}) / k_{\text{Ref}}$	
	Inner Core	Inner Core, Outer Core and Axial Blanket	Inner Core	Inner Core, Outer Core and Axial Blanket
Worth of Voided Sodium				
Mean	0.02127	0.02110	0.02116	0.02098
Percent Difference Relative to Reference ¹				
ANL	10.20326	15.30853	11.54483	16.76587
BELGIUM	-33.99089	-45.68697	-32.57208	-44.50046
CADARACHE-1	8.27564	7.86774	6.58203	6.23313
CADARACHE-2	2.96294	1.94355	2.13915	1.37186
CNEN	7.47638	4.64498	7.85819	5.04164
EIR-1	13.54133	21.61188	14.75883	22.96162
EIR-2	-7.47440	-7.25080	-7.45559	-7.20686
HEDL	-6.81619	-6.26501	-7.21926	-6.63971
JAERI-1	4.09129	9.33694	3.98248	9.23567
JAERI-2	11.19057	16.35120	11.49757	15.62206
JAERI-3	12.27191	18.24693	11.26124	17.21866
KARLSRUHE-1	-0.15416	-2.17494	0.49907	-1.50678
KARLSRUHE-2	-7.09828	-11.32664	-8.77901	-12.92601
SWEDEN	0.14204	1.04307	2.65907	3.61185
UKAEA	-1.59751	-2.79580	-3.24904	-4.39495
USSR	-13.02218	-20.85274	-13.50549	-20.88515

¹Percent difference is taken to be
$$\frac{(\text{Worth} - \text{Worth}(\text{Mean}))}{\text{Worth}(\text{Mean})} \times 100\% .$$

Table D12. Central Reactivity Worths for ^{239}Pu , ^{238}U , Na, Fe, and ^{10}B .

Organization	Configuration 1					Configuration 3				
	^{239}Pu	^{238}U	Na	Fe	^{10}B	^{239}Pu	^{238}U	Na	Fe	^{10}B
Reactivity Worth $\text{dk/k}/10^{24}$ atoms										
ANL	1.176E-04	-7.038E-06	-7.006E-07	-8.706E-07	-7.434E-05	1.342E-04	-7.460E-06	-9.756E-07	-1.081E-06	-7.214E-05
BELGIUM ¹	1.118E-04	-7.660E-06	-5.495E-07	-1.010E-06	-8.060E-05	1.277E-04	-7.988E-06	-9.510E-07	-1.062E-06	-7.181E-05
CADARACHE-1 ¹	1.236E-04	-7.640E-06	-7.700E-07	-7.850E-07	-8.235E-05	1.327E-04	-7.550E-06	-1.040E-06	-9.450E-07	-7.570E-05
CADARACHE-2 ¹	1.258E-04	-7.735E-06	-7.107E-07	-8.405E-07	-8.296E-05	1.344E-04	-7.630E-06	-9.185E-07	-1.002E-06	-7.573E-05
CNEN	1.181E-04	-7.800E-06	-7.100E-07	-8.900E-07	-7.370E-05	1.392E-04	-7.710E-06	-1.010E-06	-1.100E-06	-7.410E-05
EIR-1 ¹	1.132E-04	-6.845E-06	-5.960E-07	-9.795E-07	-7.345E-05	1.319E-04	-7.525E-06	-6.900E-07	-1.140E-06	-7.405E-05
EIR-2 ¹	1.148E-04	-7.305E-06	-6.170E-07	-8.555E-07	-7.265E-05	1.329E-04	-7.855E-06	-8.765E-07	-1.067E-06	-7.155E-05
JAERI-1	1.140E-04	-7.000E-06	-6.840E-07	-8.660E-07	-7.350E-05	1.310E-04	-7.510E-06	-9.640E-07	-1.080E-06	-7.170E-05
JAERI-2	1.192E-04	-6.949E-06	-7.302E-07	-8.812E-07	-7.845E-05	1.341E-04	-7.292E-06	-1.070E-06	-1.078E-06	-7.549E-05
JAERI-3	1.169E-04	-7.225E-06	-7.112E-07	-8.368E-07	-8.608E-05	1.321E-04	-7.638E-06	-1.001E-06	-1.068E-06	-8.385E-05
KARLSRUHE-1 ¹	1.238E-04	-7.154E-06	-7.968E-07	-9.376E-07	-7.820E-05	1.365E-04	-7.594E-06	-1.100E-06	-1.072E-06	-7.330E-05
KARLSRUHE-2 ¹	1.189E-04	-6.884E-06	-7.500E-07	-9.832E-07	-8.135E-05	1.300E-04	-7.120E-06	-8.242E-07	-1.125E-06	-7.405E-05
SWEDEN	*****	*****	*****	*****	*****	*****	*****	*****	*****	*****
UKAEA	1.244E-04	-7.465E-06	-6.711E-07	-9.261E-07	-8.054E-05	1.409E-04	-7.512E-06	-8.549E-07	-1.079E-06	-7.526E-05
USSR	1.201E-04	-7.693E-06	-6.740E-07	-7.900E-07	-7.714E-05	1.312E-04	-7.631E-06	-7.480E-07	-9.890E-07	-7.022E-05
Mean	1.187E-04	-7.314E-06	-6.908E-07	-8.894E-07	-7.824E-05	1.335E-04	-7.570E-06	-9.303E-07	-1.063E-06	-7.421E-05
Std. Dev.	4.433E-06	3.456E-07	6.735E-08	6.968E-08	4.251E-06	3.516E-06	2.123E-07	1.197E-07	5.226E-08	3.288E-06

¹These values are not shown as submitted, but have been divided by a factor of 2.0.

Table D12. Central Reactivity Worths for ²³⁹Pu, ²³⁸U, Na, Fe, and ¹⁰B.

Organization	Configuration 1					Configuration 3				
	²³⁹ Pu	²³⁸ U	Na	Fe	¹⁰ B	²³⁹ Pu	²³⁸ U	Na	Fe	¹⁰ B
Reactivity Worth dk/k/10 ²⁴ atoms										
Mean	1.187E-04	-7.314E-06	-6.909E-07	-8.894E-07	-7.824E-05	1.335E-04	-7.570E-06	-9.303E-07	-1.063E-06	-7.421E-05
Percent Difference Relative to Reference ²										
ANL	-0.94181	-3.77096	1.42282	-2.11055	-4.98269	0.52888	-1.46393	4.86635	1.62906	-2.78911
BELGIUM ¹	-5.86343	4.73292	-20.45387	13.52390	3.02318	-4.34597	5.51642	2.22597	-0.15386	-3.22772
CADARACHE-1 ¹	4.15164	4.45945	11.46589	-11.73979	5.25908	-0.58514	-0.26928	11.79286	-11.13709	2.00739
CADARACHE-2 ¹	5.96261	5.75837	2.88157	-5.49974	6.03777	0.68845	0.39119	-1.26755	-5.77716	2.04781
CNEN	-0.52318	6.64709	2.78024	0.06571	-5.79316	4.28447	1.84422	8.56800	3.43824	-0.14864
EIR-1 ¹	-4.65050	-6.41033	-13.72251	10.12843	-6.11771	-1.18448	-0.59952	-25.82971	7.19968	-0.21601
EIR-2 ¹	-3.30281	-0.12090	-10.68253	-3.81324	-7.14326	-0.43530	3.75958	-5.78226	0.38212	-3.58482
JAERI-1	-3.97665	-4.29107	-0.98354	-2.63269	-6.05380	-1.85872	-0.79765	3.62337	1.55757	-3.38269
JAERI-2	0.38652	-4.98838	5.70441	-0.92371	0.27318	0.47869	-3.67201	15.03913	1.36950	1.72306
JAERI-3	-1.53058	-1.21266	2.94744	-5.91125	10.02633	-1.05037	0.89579	7.63709	0.38777	12.98700
KARLSRUHE-1 ¹	4.26114	-2.18547	15.34548	5.41753	-0.04635	2.26920	0.31194	18.23161	0.81470	-1.22666
KARLSRUHE-2 ¹	0.15067	-5.87027	8.57067	10.54442	3.97991	-2.60040	-5.94270	-11.40415	5.81735	-0.21601
SWEDEN	*****	*****	*****	*****	*****	*****	*****	*****	*****	*****
UKAEA	4.74969	2.06947	-2.84372	4.12905	2.94458	5.52809	-0.77387	-8.10519	1.47290	1.41044
USSR	1.12775	5.18411	-2.43115	-11.17763	-1.40251	-1.71638	0.80069	-19.59511	-6.99964	-5.38242

¹These values are not shown as submitted, but have been divided by a factor of 2.0.

²Percent difference is taken to be
$$\frac{(\text{Worth} - \text{Worth}(\text{Mean}))}{\text{Worth}(\text{Mean})} * 100\%$$

Table D15. Isothermal Core Fuel Doppler Reactivities.

Organization	$k_{2200} - k_{1100}$		$(k_{2200} - k_{1100}) / k_{1100}$	
	Configuration 1 Na In	Configuration 3 Na Voided	Configuration 1 Na In	Configuration 3 Na Voided
Doppler Worth				
ANL	-0.00713	-0.00458	-0.00718	-0.00450
BELGIUM	-0.00692	-0.00431	-0.00703	-0.00433
CADARACHE-1	-0.00770	-0.00507	-0.00754	-0.00486
CADARACHE-2	-0.00708	-0.00477	-0.00699	-0.00461
CHEM	-0.00742	-0.00446	-0.00741	-0.00435
HEDL	-0.00733	-0.00497	-0.00731	-0.00486
JAERI-1	-0.00794	-0.00522	-0.00788	-0.00507
JAERI-2	-0.00668	-0.00429	-0.00660	-0.00413
JAERI-3	-0.00778	-0.00534	-0.00767	-0.00514
KARLSRUHE-1	-0.00682	-0.00425	-0.00683	-0.00417
KARLSRUHE-2	-0.00652	-0.00421	-0.00637	-0.00404
SWEDEN	-0.00933	-0.00298	-0.00952	-0.00297
UKAEA	-0.00744	-0.00448	-0.00728	-0.00430
USSR	-0.00650	-0.00340	-0.00640	-0.00330
Mean	-0.00733	-0.00445	-0.00729	-0.00433
Std. Dev.	0.00074	0.00065	0.00079	0.00062

Table D15. Isothermal Core Fuel Doppler Reactivities.

Organization	$k_{2200} - k_{1100}$		$(k_{2200} - k_{1100}) / k_{1100}$	
	Configuration 1 Na In	Configuration 3 Na Voided	Configuration 1 Na In	Configuration 3 Na Voided
Doppler Worth				
Mean	-0.00733	-0.00445	-0.00729	-0.00433
Percent Difference Relative to Reference ¹				
ANL	-2.70299	2.86687	-1.46068	3.91072
BELGIUM	-5.62326	-3.12993	-3.55359	0.03140
CADARACHE-1	5.07535	13.87225	3.48002	12.22360
CADARACHE-2	-3.38526	7.13430	-4.06828	6.45077
CNEN	1.25443	0.17169	1.69586	0.44705
HEDL	0.02623	11.62630	0.32348	12.22360
JAERI-1	8.35040	17.24130	8.14623	17.07272
JAERI-2	-8.84376	-3.64651	-9.42065	-4.63304
JAERI-3	6.16700	19.93648	5.26413	18.68915
KARLSRUHE-1	-6.88826	-4.50003	-6.24080	-3.68629
KARLSRUHE-2	-10.97256	-5.48826	-12.56349	-6.80363
SWEDEN	27.31853	-33.06912	30.65381	-31.41888
UKAEA	1.52731	0.62083	-0.08824	-0.70753
USSR	-11.30005	-23.63591	-12.16553	-23.79877

¹Percent difference is taken to be
$$\frac{(Worth - Worth(Mean))}{Worth(Mean)} \times 100\%$$

Table D16. Reactivity Worth of Central Control Rod.

Organization	$k_{\text{Rod}} - k_{\text{Ref}}$			$(k_{\text{Rod}} - k_{\text{Ref}}) / k_{\text{Ref}}$		
	Na In, Relative to Fuel	Na In, Relative to Na-Filled CRP	Na Void, Relative to Fuel	Na In, Relative to Fuel	Na In, Relative to Na-Filled CRP	Na Void, Relative to Fuel
Worth of Rod						
ANL	-0.00333	-0.00275	-0.00456	-0.00335	-0.00277	-0.00448
BELGIUM	-0.00338	-0.00287	-0.00446	-0.00343	-0.00292	-0.00448
CADARACHE-1	-0.00435	-0.00368	-0.00573	-0.00426	-0.00361	-0.00549
CADARACHE-2	-0.00388	-0.00361	-0.00560	-0.00383	-0.00357	-0.00541
CNEN	-0.00338	-0.00279	-0.00412	-0.00337	-0.00279	-0.00402
EIR-1	-0.00309	-0.00257	-0.00434	-0.00311	-0.00259	-0.00425
EIR-2	-0.00319	-0.00265	-0.00440	-0.00317	-0.00264	-0.00429
HEDL	-0.00336	-0.00281	-0.00450	-0.00335	-0.00281	-0.00440
JAERI-1	-0.00320	-0.00265	-0.00440	-0.00317	-0.00263	-0.00427
JAERI-2	-0.00317	-0.00258	-0.00436	-0.00313	-0.00256	-0.00421
JAERI-3	-0.00315	-0.00255	-0.00430	-0.00310	-0.00251	-0.00414
KARLSRUHE-1	-0.00350	-0.00290	-0.00463	-0.00350	-0.00290	-0.00454
KARLSRUHE-2	-0.00396	-0.00332	-0.00515	-0.00397	-0.00325	-0.00494
UKAEA	-0.00398	-0.00333	-0.00520	-0.00389	-0.00326	-0.00499
USSR	-0.00470	-0.00390	-0.00520	-0.00470	-0.00390	-0.00510
Mean	-0.00357	-0.00300	-0.00473	-0.00355	-0.00298	-0.00460
Std. Dev.	0.00049	0.00045	0.00051	0.00047	0.00043	0.00047

Table D16. Reactivity Worth of Central Control Rod.

Organization	$k_{\text{Rod}} - k_{\text{Ref}}$			$(k_{\text{Rod}} - k_{\text{Ref}}) / k_{\text{Ref}}$		
	Na In, Relative to Fuel	Na In, Relative to Na-Filled CRP	Na Void, Relative to Fuel	Na In, Relative to Fuel	Na In, Relative to Na-Filled CRP	Na Void, Relative to Fuel
Worth of Rod						
Mean	-0.00357	-0.00300	-0.00473	-0.00355	-0.00298	-0.00460
Percent Difference Relative to Reference ¹						
ANL	-6.83937	-8.26845	-3.59674	-5.61231	-7.08931	-2.62666
BELGIUM	-5.44057	-4.13219	-5.71080	-3.23148	-1.99432	-2.62666
CADARACHE-1	21.69627	22.75346	21.13835	20.02731	21.08575	19.32584
CADARACHE-2	8.54751	20.41849	18.38998	7.91190	19.74409	17.58701
CNEN	-5.44057	-6.93418	-12.89876	-5.04880	-6.41848	-12.62479
EIR-1	-13.55366	-14.27270	-8.24779	-12.37441	-13.12683	-7.62572
EIR-2	-10.75694	-11.60414	-6.97926	-10.68388	-11.44975	-6.75627
REDL	-6.00009	-6.26703	-4.86519	-5.61231	-5.74764	-4.36540
JAERI-1	-10.47628	-11.60414	-6.97926	-10.68388	-11.78516	-7.19100
JAERI-2	-11.31556	-13.93913	-7.82495	-11.81090	-14.13308	-8.49510
JAERI-3	-11.87509	-14.93984	-9.09340	-12.51529	-15.64247	-10.01660
KARLSRUHE-1	-2.13658	-3.29160	-2.05342	-1.31837	-2.57795	-1.21819
KARLSRUHE-2	10.75753	10.91178	8.85540	8.92621	8.97719	7.32808
UKAEA	11.34512	11.07857	9.93358	9.60242	9.34615	8.45826
USSR	31.48748	30.09200	9.93358	32.42445	30.81285	10.84913

¹Percent difference is taken to be $\frac{(\text{Worth} - \text{Worth}(\text{Mean}))}{\text{Worth}(\text{Mean})} \times 100\%$.

Table D17. Effective Delayed Neutron Fraction and the Inhour of Reactivity for Configuration 1.

Organization	Beta-Effective	Inhour of Reactivity
ANL	3.74125E-03	1.13549E-05
BELGIUM	*****	*****
CADARACHE-1	3.95080E-03	*****
CNEN	3.95800E-03	*****
EIR-1	3.74900E-03	1.13800E-05
EIR-2	3.80500E-03	1.14900E-05
JAERI-1	3.82000E-03	1.14400E-05
JAERI-2	3.75580E-03	1.13780E-05
JAERI-3	3.83974E-03	1.14991E-05
KARLSRUHE-1	3.98020E-03	1.17750E-05
KARLSRUHE-2	3.97830E-03	1.17550E-05
SWEDEN	*****	*****
UKAEA	4.01230E-03	1.17850E-05
USSR	3.82000E-03	1.13000E-05
Mean	3.86753E-03	1.15157E-05
Std. Dev.	1.01255E-04	1.86645E-07

Table D17. Effective Delayed Neutron Fraction and the Inhour of Reactivity for Configuration 1.

Organization	Beta-Effective	Inhour of Reactivity
Reference Values		
Mean	3.86753E-03	1.15157E-05
Percent Difference Relative to Reference ¹		
ANL	-3.26514	-1.39607
BELGIUM	*****	*****
CADARACHE-1	2.15297	*****
CNEN	2.33916	*****
EIR-1	-3.06476	-1.17837
EIR-2	-1.61680	-0.22315
JAERI-1	-1.22896	-0.65734
JAERI-2	-2.88893	-1.19574
JAERI-3	-0.71856	-0.14413
KARLSRUHE-1	2.91314	2.25173
KARLSRUHE-2	2.86402	2.07806
SWEDEN	*****	*****
UKAEA	3.74315	2.33857
USSR	-1.22896	-1.87307

¹Percent difference is taken to be
$$\frac{(X - X(\text{Mean}))}{X(\text{Mean})} \times 100\% .$$

Table D18. Breeding Gains.

Organization	Internal Breeding Gain	Blanket Breeding Gain	Total Breeding Gain
ANL	-0.03788	0.41510	0.37723
BELGIUM	0.03030	0.41100	0.44130
CADARACHE-1	-0.09300	0.41800	0.32500
CADARACHE-2	-0.06681	0.42630	0.35955
CNEN ¹	-0.04370	0.41141	0.36771
EIR-1	-0.06380	0.39230	0.32850
EIR-2	-0.03400	0.40260	0.36860
JAERI-1	-0.04370	0.40790	0.36420
JAERI-2	-0.06258	0.40370	0.34112
JAERI-3	-0.06348	0.39475	0.33127
KARLSRUHE-1 ¹	-0.03426	0.41285	0.37859
KARLSRUHE-2 ¹	-0.07653	0.40400	0.32747
SWEDEN	0.00140	0.41850	0.41990
UKAEA	-0.08019	0.41220	0.33201
USSR	-0.02650	0.42460	0.39810
Mean	-0.04631	0.41035	0.36404
Std. Dev.	0.03212	0.00977	0.03529

¹These values were computed by ANL from reported values in Table 2a.

Table D18. Breeding Gains.

Organization	Internal Breeding Gain	Blanket Breeding Gain	Total Breeding Gain
Reference Values			
Mean	-0.04631	0.41035	0.36404
Percent Difference Relative to Reference ²			
ANL	-18.21707	1.15885	3.62298
BELGIUM	-165.42165	0.15895	21.22400
CADARACHE-1	100.79950	1.86483	-10.72331
CADARACHE-2	44.25168	3.88750	-1.23251
CWEN ¹	-5.65322	0.25922	1.01034
EIR-1	37.75273	-4.39816	-9.76188
EIR-2	-26.58940	-1.88808	1.25349
JAERI-1	-5.64582	-0.59650	0.04483
JAERI-2	35.11642	-1.62002	-6.29520
JAERI-3	37.06177	-3.80109	-9.00096
KARLSRUHE-1 ¹	-26.03018	0.61052	3.99882
KARLSRUHE-2 ¹	65.24071	-1.54594	-10.04399
SWEDEN	-103.02277	1.98667	15.34547
UKAEA	73.14085	0.45139	-8.79769
USSR	-42.78291	3.47321	9.35706

¹These values were computed by ANL from reported values in Table 2a.

²Percent difference is taken to be
$$\frac{(B.G. - B.G.(\text{Mean}))}{B.G.(\text{Mean})} \times 100\%$$

APPENDIX C
PLOTTED RESULTS

Portions of the contributed solutions have been displayed in figures in previous sections of this report. These results include:

	<u>Figures</u>
Central Real Flux Spectrum	5-7
Central Adjoint Flux Spectrum	32-34
Radial Fission Rate Distribution	41-43
Fission Source Spectrum	2-4
Broad Group Cross Sections	8-31

The calculated reactivity worth profiles are displayed in this appendix.

	<u>Figures</u>
<u>Axial Worth Profiles in Configuration 1</u>	
Axial Reactivity Worth Profile of Pu-239	C1-C3
Axial Reactivity Worth Profile of U-238	C4-C6
Axial Reactivity Worth Profile of Fe	C7-C9
Axial Reactivity Worth Profile of B-10	C10-C12
Axial Reactivity Worth Profile of Na	C13-C15
Axial Reactivity Worth Profile of Na Leakage	C16-C18
Axial Reactivity Worth Profile of Na Non-Leakage	C19-C21
<u>Axial Worth Profiles in Configuration 3</u>	
Axial Reactivity Worth Profile of Pu-239	C22-C24
Axial Reactivity Worth Profile of U-238	C25-C27
Axial Reactivity Worth Profile of Fe	C28-C30
Axial Reactivity Worth Profile of B-10	C31-C33
Axial Reactivity Worth Profile of Na	C34-C36
Axial Reactivity Worth Profile of Na Leakage	C37-C39
Axial Reactivity Worth Profile of Na Non-Leakage	C40-C42
<u>Radial Worth Profiles in Configuration 1</u>	
Radial Reactivity Worth Profile of B-10	C43-C45
Radial Reactivity Worth Profile of Na	C46-C48
Radial Reactivity Worth Profile of Na Leakage	C49-C51
Radial Reactivity Worth Profile of Na Non-Leakage	C52-C54
<u>Radial Worth Profiles in Configuration 3</u>	
Radial Reactivity Worth Profile of B-10	C55-C57
Radial Reactivity Worth Profile of Na	C58-C60
Radial Reactivity Worth Profile of Na Leakage	C61-C63
Radial Reactivity Worth Profile of Na Non Leakage	C64-C66

AXIAL REACTIVITY WORTH PROFILE OF Pu-239 FOR CONFIGURATION 1.

Fig. C1.
AXIAL DISTANCE (cm)

AXIAL REACTIVITY WORTH PROFILE OF Pu-239 FOR CONFIGURATION 1.

Fig. C2.

AXIAL REACTIVITY WORTH PROFILE OF Pu-239 FOR CONFIGURATION 1.

Fig. C3.

AXIAL REACTIVITY WORTH PROFILE OF U-238 FOR CONFIGURATION 1.

Fig. C4.

AXIAL REACTIVITY WORTH PROFILE OF U-238 FOR CONFIGURATION 1.

Fig. C5.

AXIAL REACTIVITY WORTH PROFILE OF U-238 FOR CONFIGURATION 1.

Fig. C6.

AXIAL REACTIVITY WORTH PROFILE OF Fe FOR CONFIGURATION 1.

Fig. C7.

AXIAL REACTIVITY WORTH PROFILE OF Fe FOR CONFIGURATION 1.

Fig. C8.

AXIAL REACTIVITY WORTH PROFILE OF Fe FOR CONFIGURATION 1.

Fig. C9.

AXIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. C10.

AXIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. C11.

AXIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

AXIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 1.

Fig. C13.

AXIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 1.

Fig. C14.

AXIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 1.

Fig. C15.

AXIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 1.

Fig. C16.

AXIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 1.

Fig. C27.

AXIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 1.

Fig. C18.

AXIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 1.

Fig. C19.

AXIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 1. ●

Fig. C20.

AXIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 1.

Fig. C21.

AXIAL REACTIVITY WORTH PROFILE OF Pu-239 FOR CONFIGURATION 3.

Fig. 122.

AXIAL REACTIVITY WORTH PROFILE OF Pu-239 FOR CONFIGURATION 3.

Fig. C23.

AXIAL REACTIVITY WORTH PROFILE OF Pu-239 FOR CONFIGURATION 3. ●

Fig. C24.

AXIAL REACTIVITY WORTH PROFILE OF U-238 FOR CONFIGURATION 3.

Fig. C25.

AXIAL REACTIVITY WORTH PROFILE OF U-238 FOR CONFIGURATION 3. ●

Fig. C26.

AXIAL REACTIVITY WORTH PROFILE OF U-238 FOR CONFIGURATION 3.

Fig. C27.

AXIAL REACTIVITY WORTH PROFILE OF Fe FOR CONFIGURATION 3.

Fig. C28.

AXIAL REACTIVITY WORTH PROFILE OF Fe FOR CONFIGURATION 3.

Fig. C29.

AXIAL REACTIVITY WORTH PROFILE OF Fe FOR CONFIGURATION 3.

Fig. C30.

AXIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 3.

Fig. C31.

AXIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 3.

Fig. C32.

AXIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 3.

Fig. C33.

AXIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 3.

Fig. C34.

AXIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 3.

Fig. C35.

AXIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 3.

Fig. C36.

AXIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 3.

Fig. C37.

AXIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 3.

Fig. C38.

AXIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 3.

Fig. C39.

AXIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 3. ●

Fig. C40.

AXIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 3.

Fig. C41.

AXIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 3. ●

Fig. C42.

RADIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. C43.

RADIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. C44.

RADIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 1.

Fig. C45,

RADIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 1.

Fig. C46.

RADIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 1.

Fig. C47.

RADIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 1. ●

Fig. C48.

RADIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 1.

Fig. C49.

RADIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 1.

RADIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 1.

Fig. C51.

RADIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 1. ●

Fig. C52.

RADIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 1.

Fig. C53.

RADIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 1. ●

Fig. C54.

RADIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 3.

Fig. C55.

RADIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 3.

Fig. C56.

RADIAL REACTIVITY WORTH PROFILE OF B-10 FOR CONFIGURATION 3.

Fig. C57.

RADIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 3.

Fig. C58:

RADIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 3.

Fig. C59.

RADIAL REACTIVITY WORTH PROFILE OF Na FOR CONFIGURATION 3.

Fig. C60.

RADIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 3.

Fig. C61.

RADIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 3.

Fig. C62.

RADIAL REACTIVITY WORTH PROFILE OF Na LEAKAGE FOR CONF. 3.

Fig. C63.

RADIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 3.

Fig. C64.

RADIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 3.

Fig. C65.

RADIAL REACTIVITY WORTH PROFILE OF Na NON-LEAKAGE FOR CONF. 3.

Fig. C66.

APPENDIX D

LISTING OF CODES TO GENERATE TABLES

Because the contributed solutions to these comparison calculations represent such a large mass of data, a series of simple FORTRAN programs were written to manipulate and display these results. Computer generation of the tables permitted not only very simple (and hopefully error-free) reproduction of the results but also calculation of values of the mean, standard deviation, and percent differences of each solution relative to each other solution and the mean. The same deck, including input data to be tabulated, required changing only one flag to compute the mean and standard deviation of any desired subset of the solutions. Needless to say, the use of these codes greatly simplified the additions, deletions, and corrections made to their solutions by many participants.

Even though each of these programs is very simple (and almost identical in structure), one is listed herein to serve as an example for anyone wishing to work from it to achieve a similar task. The program listed in Figure D-1 was used to generate Table D-1 (Summary of Calculated Eigenvalues) in the comparison study. The input description is given in Figure D-2.

```

-- //KFTAB201 JOB (A12644,40,,6),KEE,REGION=180K,CLASS=A,PRTY=T
FREESE K      12644      41605-00116
-- //**FORMAT PR,DDNAME=FT06FO01,TRAIN=TN,DEST=LUCAL,CUPIES=4
-- //STEP1 EXEC FTHCLG
-- //FTH.SYSIN DD *
      REAL*8 URG(40),DORG(40)
      DIMENSION DATA(12,20),DIFF(12,20),ANS(12,20)
      DIMENSION SUM1(20),SUM2(20),EN(20),AVG(20),SD(20),IA(20)
      M=8
10  FORMAT(12A6)
12  FORMAT (6F12.5)
      READ(5,11)NP
11  FORMAT(I3)
      NP2=2*NP
      NP3=NP2+2
      MP=NP+1
      READ(5,10)(URG(I),I=1,MP3)
      READ(5,13)(IA(I),I=1,MP)
13  FORMAT(20I1)
      DO 15 J=1,MP
      READ (5,12) (DATA(I,J),I=1,8)
15  CONTINUE
      DO 102 I=1,M
      SUM1(I)=0.0
      SUM2(I)=0.0
      EN(I)=0.0
      DO 100 J=1,MP
      IF(IA(J).NE.0) GO TO 100
      IF(DATA(I,J).EQ.0.0) GO TO 100
      SUM1(I)=SUM1(I)+DATA(I,J)
      SUM2(I)=SUM2(I)+DATA(I,J)*DATA(I,J)
      EN(I)=EN(I)+1.0
100 CONTINUE
      DATA(I,MP)=SUM1(I)/EN(I)
      V=(SUM2(I)-SUM1(I)*SUM1(I)/EN(I))/(EN(I)-1.0)
      SD(I)=SQRT(V)
102 CONTINUE
      WRITE(6,51)
      WRITE(6,52)
      WRITE(6,53)
      WRITE(6,54)
      WRITE(6,55)
      WRITE(6,56)
      WRITE(6,57)
      DO 16 I=1,MP
      I2=2*I
      I1=I2-1
      WRITE(6,58)DORG(I1),DORG(I2),(DATA(J,I),J=1,8)
16  CONTINUE
      WRITE(6,81)
      WRITE(6,82)(DATA(I,MP),I=1,M)
      WRITE(6,83)(SD(I),I=1,8)
      WRITE(6,62)
      WRITE(6,63)
      WRITE(6,64)
      WRITE(6,65)
      WRITE(6,66)
      WRITE(6,67)
      WRITE(6,68)
      WRITE(6,69)

```

Fig. D-1.

```

WRITE(6,70)
WRITE(6,71)
WRITE(6,72)
NP=MP
DO 99 IREF = 1, NP
  IR2=2*IREF
  IR1=IR2-1
  DO 20 I = 1, NP3
20  DORG(I) = ORG(I)
    IF( IREF-1, EQ, 0 ) GO TO 25
    DORG(I) = ORG(IR1)
    DORG(2)=ORG(IR2)
    DO 24 K=4, IR2, 2
    DORG(K-1)=ORG(K-3)
24  DORG(K) = ORG(K-2)
25  CONTINUE
    DO 32 I = 1, 5
    DO 30 J = 1, NP
      DIFF(I,J) = DATA(I,J) - DATA(I,IREF)
30  ANS(I,1) = DIFF(I,1)
32  ANS(I,1) = DATA(I,IREF)
    IF( IREF-1, EQ, 0 ) GO TO 35
    DO 34 K = 2, IREF
    DO 34 I = 1, 8
34  ANS(I,K) = DIFF(I,K-1)
35  CONTINUE
51  FORMAT (1H1)
    WRITE (6,51)
    WRITE (6,52)
52  FORMAT (      1H ,35X'TABLE D1.  SUMMARY OF CALCULATED EIGENVALUES.
1')
    WRITE(6,53)
53  FORMAT( /,58X,'CONFIGURATION11')
    WRITE(6,54)
54  FORMAT(1H ,18X,93(' '))
    WRITE(6,55)
55  FORMAT(1H0,'ORGANIZATION',10X,'1',11X,'2',11X,'3',11X,'4',11X,'5',
111X,'6',11X,'7',11X,'8')
    WRITE(6,56)
56  FORMAT(1H ,12(' '),3X,8(3X,9(' ')))
    WRITE(6,57)
57  FORMAT(1H0,60X,'EIGENVALUE')
    WRITE(6,58)DORG(1),DORG(2),(ANS(J,1),J=1,8)
58  FORMAT(1H0,      2A6,3X,8F12.5)
    WRITE(6,59)
59  FORMAT(1H0,54X,'EIGENVALUE DIFFERENCE2')
    DO 60 I=2, NP
    I2=2*I
    I1=I2-1
    WRITE(6,58)DORG(I1),DORG(I2),(ANS(J,I),J=1,8)
60  CONTINUE
    WRITE(6,61)
    WRITE(6,62)
    WRITE(6,63)
    WRITE(6,64)
    WRITE(6,65)
    WRITE(6,66)
    WRITE(6,67)
    WRITE(6,68)
    WRITE(6,69)

```

```

WRITE(6,70)
WRITE(6,71)
WRITE(6,72)
61 FORMAT(1H,12(' '))
62 FORMAT(1H0,6X,'1CONFIGURATION KEY:')
63 FORMAT(1H0,7X,'1. REFERENCE.',40X,'5. REFERENCE WITH CENTRAL BOR
1ON')
64 FORMAT(1H,65X,'CONTROL ROD.')
65 FORMAT(1H,7X,'2. NA-VOIDED INNER CORE.')
66 FORMAT(1H,61X,'6. NA-VOIDED INNER CORE, OUTER CORE AND')
67 FORMAT(1H,7X,'3. NA-VOIDED INNER CORE, OUTER CORE',22X,
1'AXIAL BLANKET WITH CONTROL ROD INSERTED.')
68 FORMAT(1H,11X,'AND AXIAL BLANKET.')
69 FORMAT(1H,61X,'7. REFERENCE WITH HOT FUEL.')
70 FORMAT(1H,7X,'4. REFERENCE WITH CENTRAL A-FILLED')
71 FORMAT(1H,11X,'CONTROL ROD POSITION.',29X,'8. NA-VOIDED INNER CO
XRE, OUTER CORE')
72 FORMAT(1H,65X,'AND AXIAL BLANKET WITH HOT FUEL.')
73 FORMAT(1,2,' - 2EIGENVALUE DIFFERENCE IS TAKEN TO BE  $K - K($ ,
12A6,').')
81 FORMAT(1H,111(' '))
82 FORMAT(1H0,'MEAN',11X,BF12.5)
83 FORMAT(1H0,'STD. DEV.',6X,BF12.5)
WRITE(6,73) ODRG(1),ODRG(2)
86 FORMAT(//,6X,'*MEAN AND STD. DEV. INCLUDES ONLY THOSE PARTICIPANT,
X.')
99 CONTINUE
STOP
END
//GO,SYSIN DD *
***** INPUT DATA HERE *****
/*

```

Fig. D-1 (Cont'd.)

INPUT DESCRIPTION

<u>Card Type</u>	<u>Format</u>	<u>Columns</u>	<u>Description</u>
1	I3	1-3	Number of participants
2	6A12	1-72	Names of participants, six per card in fields of twelve. Up to 19 participants may be used. In field following last participant name, the word "mean" must be punched for calculation of mean in table.
3	19I1	1-19	A flag corresponding to each of up to 19 participants signals whether to include data from that participant in calculation of mean and standard deviation. 0 = include data 1 = exclude data
4	6E12.5	1-72	Data to be tabulated for each participant. Repeat card type 4 for each participant.

Fig. D-2.

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