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ASSESSMENT OF NUCLEAR DATA FILES VIA BENCHMARK CALCULATIONS -- A PRELIMINARY REPORT ON THE NEACRP/IAEA INTERNATIONAL COMPARISON CALCULATION OF A LARGE LMFBR

L. G. LeSage, D. C. Wade, R. D. McKnight and K. E. Freese
Argonne National Laboratory
Applied Physics Division
9700 South Cass Avenue
Argonne, Illinois 60439 USA

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 fourteen 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.

INTRODUCTION AND BACKGROUND

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" comparison[1] 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 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 specification which was 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.

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.

The purpose of this paper is to present a preliminary report on the results of the comparison and of the specialist's meeting. A detailed report of the meeting will be issued as a combined ANL/NEACRP report later this year. 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	R. D. McKnight, L. G. LeSage, D. C. Wade
BELGIUM	Belgium	KEDAK-2	No	G. Minsart, R. deWouters
CADARACHE	France	CARNAVAL-3	Yes	Y. Y. Bouget, P. Hammer
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. R. Marr, R. W. Hardie
JAERI-1	Japan	JENDL	No	H. Yoshida, H. Nishimura, S. Iijima T. Osugi, Y Kikuchi, T. Nakagawa, K Ikawa, and H 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	Germany	KEDAK-3	No	C. Broeders
SWEDEN	Sweden	ENDF/B-III	No	Klas Jirlow
UKAEA	England	FGL-5	Yes	J. L. Rolands, C. J. Dean, C. G. Campbell
USSR	USSR	BNAB-70	No	M. Troyanov

*Additional solutions from Karlsruhe and Cadarache are not included.

PROBLEM SPECIFICATION

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
- 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, and
- Isothermal core fuel Doppler reactivities

SURVEY OF RESULTS

The fourteen 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. Although it is possible to include only a part of this material, the most important and interesting results are summarized in Tables II-V and Figures 1-4.

TABLE II. Eigenvalue, k_∞, Leakage Fraction, Reaction Rate Ratios and Breeding Ratio

	Configuration 1		Leakage Fraction		Reaction Rate Ratios[a]			Breeding Ratio		
Solution	k_{eff}	k_∞(IC)	Core	Reactor	$^{28}c/^{49}f$	$^{28}f/^{49}f$	$^{49}c/^{49}f$	Core	Blanket	Total
Mean	1.0032	1.1273	0.163	0.0167	0.166	0.0222	0.299	0.934	0.402	1.395
Std. Dev.	0.0124	0.0186	0.005	0.0031	0.007	0.0009	0.018	0.046	0.010	0.050
	$\Delta k \times 10^2$		Percent Difference Relative to Mean							
ANL	-1.02	-1.67	4.9	37.7	0.1	-1.7	3.1	0.4	0.8	0
Belgium	-1.91	-2.87	-0.2	-1.2	9.4	-1.1	-16.9	11.3	3.0	8.9
Cadarache	1.80	2.23	-0.3	13.1	-3.7	-2.6	3.2	-4.5	1.0	-2.8
CNEN	-0.16	1.62	4.3	-12.0	-1.6	2.7	1.7	-1.1	1.1	-0.5
EIR-1	-0.86	-2.03	-2.3	-16.6	-2.4	-4.2	3.1	-2.8	-5.0	-3.4
EIR-2	0.17	-0.89	-2.7	-15.8	0.7	-2.8	1.0	-0.2	2.0	-0.7
HEDL	-0.11	-0.77	-	-10.6	-	-	-	-	-	-0.7
JAERI-1	0.34	-0.91	-4.8	-25.6	-0.6	6.6	8.2	-1.5	-1.8	-1.6
JAERI-2	0.88	0.74	0.8	3.4	-3.3	-0.5	-0.2	-3.1	-0.6	-2.4
JAERI-3	1.12	0.40	-3.3	3.4	-1.8	4.7	3.9	-3.0	-3.5	-3.1
Karlsruhe	-0.44	-	0.1	11.0	-0.9	-1.6	2.1	-0.1	0.4	0.1
Sweden	-2.27	1.56	1.0	-25.9	3.4	-7.7	-3.0	3.1	1.4	2.6
UKAEA	1.91	2.12	1.5	17.6	-4.5	5.6	-0.4	-4.2	1.4	-2.5
USSR	0.53	0.48	1.2	21.5	5.0	2.6	-5.8	6.8	4.0	6.1

[a] Per atom reaction rate ratios are given, and f and c refer to capture and fission rates respectively.

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

	Central B_4C Control Rod Worth $(k_{rod}-k_{ref})/k_{ref}$			Na-Void Worth $(k_{void}-k_{ref})/k_{ref}$		Isothermal Fuel Doppler Worth $(k_{2200}-k_{1100})/k_{1100}$	
Solution	Na In Rel. to Fuel	Na In Rel. to Na	Na Void Rel. to Fuel	Inner Core	Core Plus Axial Blanket	Conf. 1 Na In	Conf. 3 Na Void
Mean	-0.00350	-0.00291	-0.00451	0.0213	0.0212	-0.00743	-0.00439
Std. Dev.	0.00047	0.00041	0.00043	0.0027	0.0037	0.00096	0.00085
			Percent Difference Relative to Mean				
ANL	-4.2	-4.8	-0.7	11.0	15.8	-3.3	2.6
Belgium	-1.8	0.5	-0.7	-32.9	-45.0	-5.4	-1.2
Cadarache	21.8	24.1	21.7	6.1	5.4	1.5	10.8
CNEN	-3.6	-4.1	-10.9	7.3	4.2	-0.2	-0.8
EIR-1	-11.1	-11.0	-5.8	14.2	22.0	-17.9	-24.1
EIR-2	-9.4	-9.2	-4.9	-7.9	-8.0	3.4	12.6
HEDL	-4.2	-3.4	-2.5	-7.7	-7.4	-1.6	10.8
JAERI-1	-9.4	-9.6	-5.4	3.5	8.3	6.1	15.6
JAERI-2	-10.5	-12.0	-6.7	11.0	14.7	-11.1	-5.8
JAERI-3	-11.2	-13.5	-8.3	10.7	16.3	24.3	43.4
Karlsruhe	0.1	-0.1	0.7	0.0	-2.3	-8.0	-4.9
Sweden	-2.2	-3.0	-	2.2	2.8	28.2	-32.3
UKAEA	11.2	12.1	10.6	-3.7	-5.2	-2.0	-2.0
USSR	34.4	34.1	13.0	-13.9	-21.5	-13.8	-24.6

TABLE IV. One-Group Cross Sections and Mean Flux Energy

Solution	$\overline{E}$(¢) (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	110.4	0.308	0.0415	0.554	1.860	0.298	5.49	4.16	4.30	3.50
Std. Dev.	4.4	0.010	0.0020	0.042	0.038	0.019	0.12	0.61	0.22	0.11
					Percent Difference Relative to Mean					
ANL	-2.5	-0.3	-2.8	2.4	-0.9	3.5	-0.7	-3.4	-3.4	0.9
Belgium	-2.2	7.8	-3.4	-18.5	-1.9	-16.8	-	13.3	11.4	7.0
Cadarache[a]	6.8	-2.8	-2.6	3.8	0.3	3.5	-	-	-	-
CNEN	3.0	-1.7	9.0	3.6	1.0	2.6	-1.6	-1.9	-4.4	1.1
EIR-1	-1.2	-1.0	-3.7	4.2	0.8	3.4	-0.5	28.9	5.7	2.5
EIR-2	1.2	4.8	0.3	4.7	3.4	1.3	-0.1	9.2	1.6	0.1
JAERI-1	-1.5	0.7	7.5	9.7	1.2	8.5	0.7	15.2	3.6	0.5
JAERI-2	-1.4	-3.7	-2.4	-4.6	-1.0	-3.6	0.5	-5.3	-3.6	-0.2
JAERI-3	-7.7	0.4	5.6	5.9	1.7	4.2	5.6	-4.5	1.9	0.1
Karlsruhe	-1.5	0.3	-1.2	3.0	0.6	2.4	-	-24.9	-5.4	-4.3
Sweden	-1.5	0.3	-6.7	-2.4	1.1	-3.4	-0.1	-9.4	-2.6	-0.5
UKAEA	1.4	-5.3	3.7	-1.6	-1.5	-0.1	-1.4	-15.0	-4.6	-5.4
USSR	6.9	0.5	-3.3	-10.1	-4.9	-5.5	-2.4	-2.2	0	-1.7

[a] One group cross section values calculated by ANL from neutron balance data and other information.

TABLE V. Central Isotopic Reactivity Worths (10^{-6} Δk/k per 10^{24} atoms/cc)

	Configuration 1 (Na In)					Configuration 3 (Na Void)				
Solution	^{239}Pu	^{238}U	Na	Fe	^{10}B	^{239}Pu	^{238}U	Na	Fe	^{10}B
Mean	118.5	-7.3	-0.70	-0.87	-77.5	134.1	-7.6	-0.94	-1.06	-74.1
Std. Dev.	5.0	0.3	0.09	0.08	4.2	4.6	0.2	0.13	0.05	3.5
				Percent Difference Relative to Mean						
ANL	-0.7	-3.7	0.5	-0.3	-4.2	0.1	-1.8	3.8	1.6	-2.6
Belgium	-5.7	4.8	-21.1	15.6	3.9	-4.8	5.2	1.2	-0.1	-3.1
Cadarache	4.4	4.6	10.5	-10.1	6.1	-1.0	-0.6	10.6	-11.1	2.2
CNEN	-0.3	6.8	1.9	1.9	-5.0	3.8	1.5	7.4	3.4	0.0
EIR-1	-4.4	-6.3	-14.4	12.2	-5.3	-1.6	-0.9	-26.6	7.2	-0.1
EIR-2	-3.1	0.0	-11.5	-2.0	-6.4	-0.9	3.4	-6.8	0.4	-3.4
JAERI-1	-3.8	-4.2	-1.8	-0.8	-5.2	-2.3	-1.1	2.5	1.6	-3.2
JAERI-2	0.6	-4.9	26.5	-16.3	1.1	-0.1	-4.0	13.8	1.3	1.8
JAERI-3	-1.3	-1.1	2.1	-4.1	11.0	-1.5	0.6	6.5	0.4	13.2
Karlsruhe	4.5	-2.1	14.3	7.4	0.8	1.8	0.0	17.0	0.8	-1.1
UKAEA	8.6	0.9	-3.7	6.1	3.8	8.6	-2.6	-9.1	1.5	1.6
USSR	1.3	5.3	-3.3	-9.5	-0.6	-2.2	0.5	-20.4	-7.0	-5.2

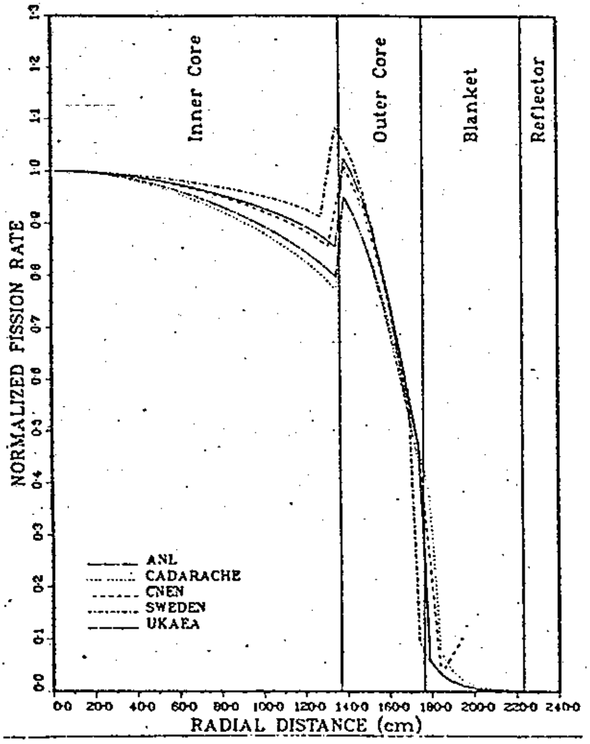

Fig. 1. Radial Fission Rate Distributions for Configuration 1.

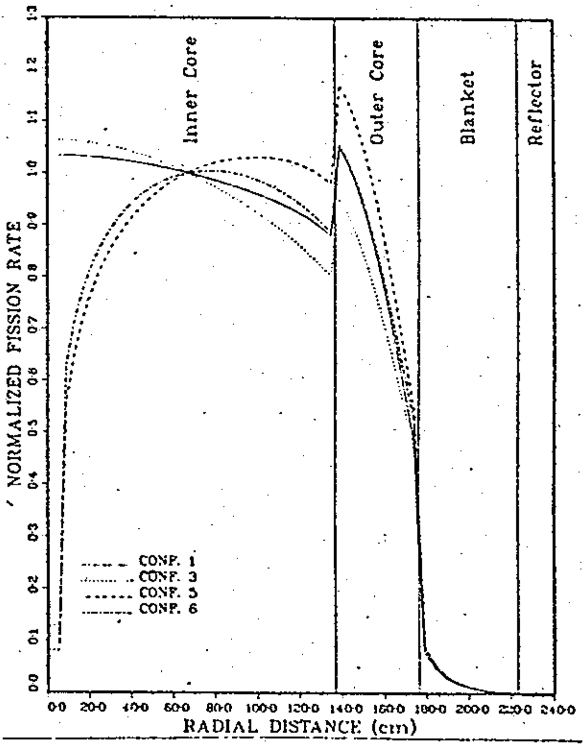

Fig. 2. Radial Fission Rate Distributions with and without Inserted Central Control Rod (ANL).

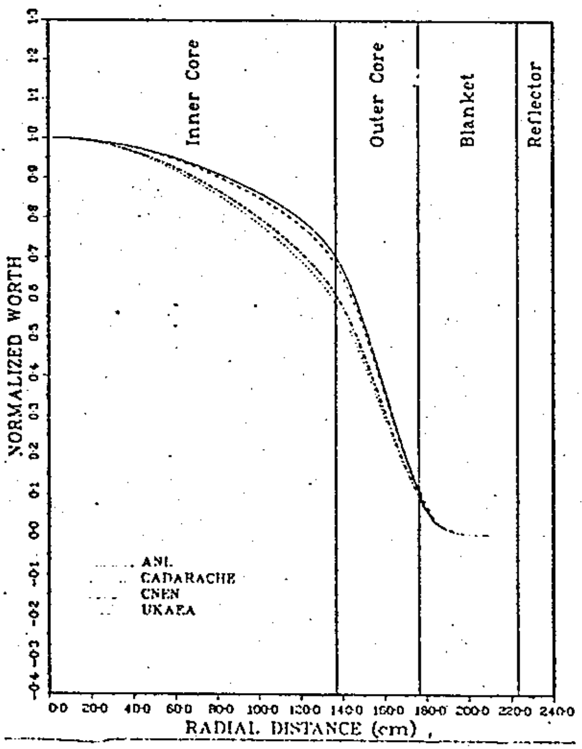

Fig. 3. Radial Reactivity Worth Profile for ^{10}B for Configuration 1.

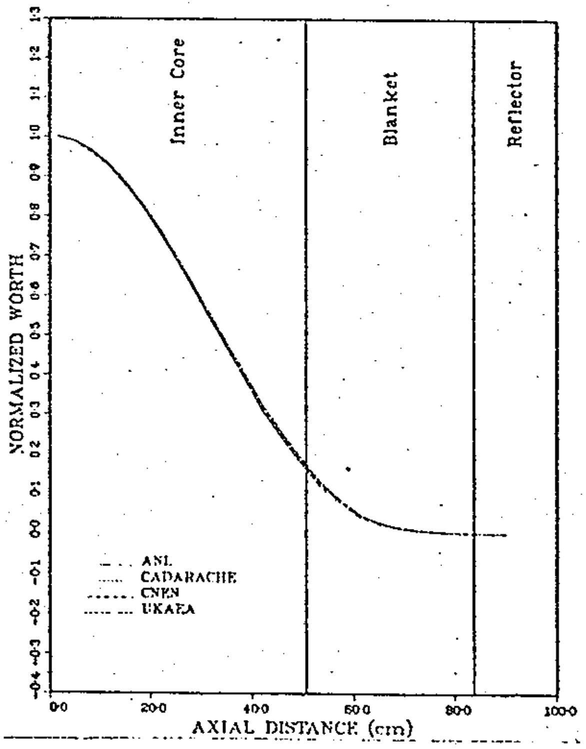

Fig. 4. Axial Reactivity Worth Profile for ^{10}B for Configuration 1.

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 III. 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).

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 possibly discrepant and
- the correlations among the results for different parameters.

In the following discussions effective one-group cross sections are indicated by $<\sigma>$, 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.

Eigenvalue and k_∞

Relatively large 1σ variations of 1.2% and 1.6% are observed for the eigenvalue and k_∞ (see Table II). These results are comparable to those obtained in the earlier "Baker Model" comparison.[1] The solutions with the largest eigenvalues (UKAEA and Cadarache) also have:

- the largest k_∞,
- the smallest $^{28}c/^{49}f$,
- the smallest $<\sigma_c^{28}>$ (with the JAERI-2 Solution),
- the largest ρ^{49} (with the Karlsruhe solution) and
- the smallest core breeding ratio.

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

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

There appears to be no significant correlation between k_{eff} and $<\sigma_f^{49}>$, ρ^{28}, or core or reactor leakage. The k_∞ values generally correlate with k_{eff} values but there are some exceptions. The Swedish solution has a high k_∞, the smallest k_{eff} and the smallest reactor leakage, a combination that seems to be inconsistent with the other solutions. That the k_{eff} for the USSR solution is 0.5% above the mean is surprising since it has among the largest values of $^{28}c/^{49}f$ and the smallest $<\sigma_f^{49}>$; however, it does have a small $^{49}c/^{49}f$. Using calculated sensitivity coefficients, Collins[2] has shown that the k_{eff} differences between ANL and UKAEA and between ANL and Cadarache can be explained by the cross section differences.

Core and Reactor Leakage

The 1σ variation of the core leakage (see Table II) is only 3% while that for the reactor leakage is 19%. This is not surprising due to the relative small fraction (~1.7%) of the neutrons that leak from the reactor. The ANL solution indicates a reactor leakage considerably larger than the other solutions. In order to check this result, one of the authors (McKnight) ran a VIM Monte Carlo calculation using the same ENDF/B-IV data base as used for the ANL diffusion 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 VIM result apparently confirms that the difference between ANL and the other results is not due to approximations in ANL diffusion code, and leaves the question of the cause of the difference unanswered.

Breeding Ratio

The 1σ variation for the core breeding ratio (4.9%) is about the same as that for $^{28}c/^{49}f$ (4.2%), as expected, but is about double the value for the blanket breeding ratio (2.5%). The solutions with the smallest breeding ratios (UKAEA and Cadarache) also have:

- the smallest $^{28}c/^{49}f$,
- the smallest $<\sigma_c^{28}>$ (with the JAERI-2 solution),
- the largest k_{eff} and k_∞ and
- the largest ρ^{49} (with the Karlsruhe solution).

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

- the largest $^{28}c/^{49}f$,
- the smallest $<\sigma_f^{49}>$,
- the smallest k_{eff} and k_∞ (Belgium only),
- the largest ρ^{28} (with the CNEN and Cadarache solutions) and
- the smallest sodium-void.

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 $^{28}c/^{49}f$ and breeding ratio, k_{eff} and k_∞ is very apparent.

Central Control Rod Worth

The large inconsistency in these results (see Table III) was one of the real surprises associated with this exercise. The 1σ variations for the sodium-in and sodium-voided cases of 14% and 10% are unusually large compared to normal accuracy requirements for control rod calculations. The solutions with the largest central control rod worths (USSR, Cadarache and UKAEA) are not well-correlated with the solutions with the largest central ρ^{B10} or $<\sigma_{tr}^{B10}>$, 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 Fig. 1) of the UKAEA and Cadarache 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 2 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. Further analysis is needed on this point.

Another low energy indicator is the Doppler effect, but there appears to be no correlation with the control rod worths. The UKAEA and Cadarache 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 $<\sigma_{tr}^{B10}>$, ρ^{B10} and Doppler values.*

Sodium-Void Reactivity

Considering the traditional difficulty in sodium-void calculations, the 1σ variations of 13% and 17% in the results (see Table III) are surprisingly good. The Belgium values seem to be somewhat inconsistent and contribute significantly to the 1σ variation and the maximum spread in the results. The sodium-void results do not correlate well with $<\sigma_{tr}^{Na}>$ 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 $<\sigma_{tr}^{Na}>$ which appears to be inconsistent.

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*Note added in press: The JAERI-3 Doppler result will be revised and the EIR-1 Doppler result does not include the unresolved resonance contribution and may be revised.

Doppler Reactivity

The 1σ variations of the Doppler values (13% and 18%) are slightly larger than for the sodium-void values. 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 JAERI-3 solution* has large Doppler values; however, it 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.

Radial Distributions

Another of the more surprising results is the large spread in the radial worth and power distributions as seen in Figs. 1 and 3. This spread is not indicated in the axial worth profiles (shown in Fig. 4) which are very consistent. The Cadarache, UKAEA and USSR radial solutions show the largest inner core buckling while the Swedish solution had 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 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 and UKAEA solutions have the largest k_{eff} values while the Swedish solution has the smallest k_{eff}. The functional form of the relationship (if any) between the radial distributions and k_{eff} is not known.

Because of the consistency in the axial profiles (e.g., see that of ρ^{B10} in Fig. 4) 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 to differing k_∞ increments produced by the inner core to outer core enrichment change. Given the inner core k_∞, 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

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*Note added in press. The JAERI-3 Doppler result will be revised and the EIR-1 Doppler result does not include the unresolved resonance contribution and may be revised.

outer cores determines the radial power profile. The inner core to outer core k_{B^2} ratio* is largest for the Swedish solution and smallest for the Cadarache solution which correlates well with the fact that the Swedish solution displays the flattest radial power profile while the Cadarache solution is among the steepest. This correlation between k_{B^2} split and inner core/outer core power split was observed for most participant's solutions.

As shown in Fig. 2, 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. 2, 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.

Structural Cross Sections

Large variations in the structural cross sections are observed. The 1σ variations are 15% for $<\sigma_{tr}^{Fe}>$ (see Table IV) and 25%, 39% and 15%, respectively, for the iron, chromium and nickel capture rates (not shown in the tables).

Central Worth Discrepancy

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., CARANVAL-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 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.

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*While the k_∞ values for the outer core were not calculated, a related parameter can be calculated from the neutron balance data. Called k_{B^2} it is the ratio of fission production to absorption for the critically-buckled spectrum.

COMPARISON OF RESULTS FROM ADJUSTED AND NON-ADJUSTED DATA SETS

The comparison calculation 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. Tables VI and VII show the results of such a grouping.

As regards the sizes of variations in cross sections, inspection of Table VI leads to the following conclusions:

- For participants using the same (ENDF/B-IV) basic data set, but different processing codes:
  - The variation in heavy element $<\sigma>$'s is small ($\overset{\sim}{<}1\%$) except for $<\sigma_f^{28}>$ where a 7% standard deviation is seen, but
  - the variation in light element $<\sigma_{tr}>$ is large particularly for Fe (±16.92%).
- It appears then, that even with a given basic data set, cross section processing differences lead to nontrivial differences in cross sections.
- For participants using adjusted basic data sets:
  - variations in both heavy element and light element $<\sigma>$'s are of the same characteristic size as the variations observed among the participants using non-adjusted data sets.
- Independent adjustment processes then, do not lead to a common set of cross sections -- though (as will be discussed next) they may lead to a common set of calculated integral parameters.

As regards sizes of variations in the integral parameters, inspection of Table VII 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.

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*The HEDL results are not included in any of the groupings since they were submitted after the initial compilation of the results.

TABLE VI. One-Group Effective Cross Sections By Basic Data File Type

	$\hat{E}$(keV)	$\langle\bar{\nu}\rangle$	$\langle\sigma_c^{28}\rangle$	$\langle\sigma_f^{28}\rangle$	$\langle\sigma_c^{49}\rangle$	$\langle\sigma_f^{49}\rangle$	$\langle\bar{\alpha}^{49}\rangle$	$\langle\sigma_{tr}^{Fe}\rangle$	$\langle\sigma_{tr}^{Na}\rangle$	$\langle\sigma_{tr}^{\phi}\rangle$	$\langle\sigma_{tr}^{B10}\rangle$
All Sets mean Stand Dev.	110.45 ±4.36	2.915	0.3077 ±3.40%	0.04147 ±4.87%	0.5543 ±7.55%	1.8605 ±2.06%	0.2978 ±6.32%	4.1596 ±14.55%	4.2984 ±5.09%	3.5007 ±3.16%	5.4868 ±2.18%
Adjusted Sets mean Stand Dev.	110.22 ±6.66	2.914	0.2989 ±2.50%	0.04193 ±4.17%	0.5592 ±4.79%	1.8585 ±1.42%	0.3008 ±3.57%	3.8155 ±6.36%	4.2071 ±3.60%	3.4358 ±3.16%	5.5722 ±3.57%
Non-Adjusted Sets mean Stand Dev.	110.55 ±3.45	2.916	0.3116 ±3.01%	0.04126 ±5.33%	0.5522 ±8.76%	1.8614 ±2.36%	0.2964 ±7.39%	4.2743 ±15.34%	4.3288 ±5.46%	3.5224 ±3.07%	5.4502 ±1.03%
ENDF/B-IV Sets mean Stand Dev.	110.21 ±3.18	2.915	0.3046 ±0.72%	0.04181 ±6.98%	0.5731 ±0.86%	1.8660 ±1.07%	0.3071 ±0.49%	4.4863 ±16.92%	4.2682 ±5.65%	3.5527 ±0.86%	5.4572 ±0.61%
Per Cent Differences Relative To the Mean of the Adjusted Sets											
Adjusted Sets	---	---	---	---	---	---	---	---	---	---	---
Non Adjusted Sets	~Same	~Same	+4.25%	-1.60%	-1.25%	+0.16%	-1.46%	+12.02%	+2.89%	+2.52%	-2.19%
ENDF/B-IV Sets	~Same	~Same	+1.91%	-0.29%	+2.48%	+0.40%	+2.09%	+17.58%	+1.45%	+3.40%	-2.06%

TABLE VII. Integral Parameters By Basic Data File Type

	k_∞(IC)	k_{B^2}(IC)	k (conf 1)	Core Leakage (conf 1)	c^8/f^9	f^8/f^9	ρ^{49}	ρ^{28}	ρ^{B10}	CR(IC)	BR(conf 1)
All Sets mean Stand Dev.	1.12806 ±0.01926	1.2020 ±0.0192	1.00328 ±0.01296	0.1628 ±2.83%	0.1663 ±3.91%	0.02223 ±4.18%	1.185-4 ±4.20%	-7.307-6 ±4.46%	-7.758-5 ±5.47%	1.0962 ±4.45%	1.3960 ±3.74%
Adjusted Sets mean Stand Dev.	1.14279 ±0.01053	1.2226 ±0.0112	1.01749 ±0.00507	0.1622 ±2.10%	0.1608 ±1.19%	0.02262 ±3.93%	1.221-4 ±4.27%	-7.297-6 ±3.94%	-8.186-5 ±3.95%	1.0551 ±0.96%	-1.3574 ±0.34%
Non-Adjusted Sets mean Stand Dev.	1.12069 ±0.01869	1 1928 ±0.0141	0.99696 ±0.00982	0.1631 ±3.19%	0.1688 ±3.76%	0.02205 ±4.26%	1.167-4 ±3.42%	-7.312-6 ±4.96%	-7.545-5 ±3.78%	1.1145 ±4.31%	1.4132 ±3.88%
ENDF/B-IV Sets mean Stand Dev.	1.11950 ±0.02267	1.1953 ±0.0100	0.99642 ±0.00463	0.1666 ±3.90%	0.1642 ±1.29%	0.02199 ±3.55%	1.163-4 ±2.32%	-7.228-6 ±6.99%	-7.383-5 ±0.62%	1.0823 ±1.22%	1.3771 ±1.86%
Per Cent Differences Relative to the Mean of The Adjusted Sets											
Adjusted Sets	---	---	---	---	---	---	---	---	---	---	---
Non Adjusted Sets	-1.93%	-2.44%	-2.02%	+0.55%	+4.98%	-2.52%	-4.42%	+0.71%	-7.83%	+5.63%	+4.11%
ENDF/B-IV Sets	-2.04%	-2.23%	-2.07%	+2.71%	+2.11%	-2.79%	-4.75%	-0.95%	-9.81%	+2.58%	+1.45%

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
$^{28}c/^{49}f$	±1.2%	±3.7%
CR(IC)	±1%	±4%
BR(Config. 1)	±0.3%	±3.9%

- For $^{28}c/^{49}f$, ρ^{49}, ρ^{28} and ρ^{B10}, however, the variations are of about the same size. This is unexpected since these measurable parameters were used in the adjustment procedure. (The large variations in the central worths, as discussed above, are due in part to the different radial flux profiles.)

- 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 VI and VII 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 VI and VII leads to the following conclusions.

- For heavy element cross sections:

• for $<\bar{\nu}>$ and $<\sigma_f^{49}>$	<1% Difference Between Groupings
• for $<\sigma_c^{28}>$	Adjusted Sets ∿4% < Non-Adjusted
	" " ∿2% < ENDF/B-IV
• for $<\alpha^{49}>$	Adjusted Sets ∿1.5% > Non-Adjusted
	" " ∿2% < ENDF/B-IV
• for $<\sigma_f^{28}>$	Adjusted Sets ∿1.6% > Non-Adjusted
	" " ∿Same as ENDF/B-IV

- For light element cross sections:

• for $<\sigma_{tr}^{Fe}>$	Adjusted Sets ∿12 to 18% <Non-Adjusted and ENDF/B-IV
• for $<\sigma_{tr}^{Na}>$ and $<\sigma_{tr}^{\phi}>$	Adjusted Sets ∿1.5 to 3% < Non-Adjusted and ENDF/B-IV
$<\sigma_{tr}^{B10}>$	Adjusted Sets ∿2% > Non-Adjusted and ENDF/B-IV

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

- underpredict conversion ratio, breeding ratio, $^{28}c/^{49}f$ and ρ^{28},
- overpredict k_∞, $^{28}f/^{49}f$, and core leakage and
- produce about the same result on ρ^{49}, ρ^{B10}, and k_{eff}

Examination of Table VII shows that these expectations are generally met for core conversion ratio, breeding ratio, $^{28}c/^{49}f$, k_∞, and $^{28}f/^{49}f$. However, some surprises occurred:

- ρ^{49} and ρ^{B10} changed more than expected,
- ρ^{28} changed less than expected and
- core leakage changed opposite to expectation.

CONCLUSIONS

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, $<\sigma_c^{28}>$, and $^{28}c/^{49}f$ (e.g., solutions having high $<\sigma_c^{28}>$ and $^{28}c/^{49}f$, 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 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 $^{28}c/^{49}f$. On the other hand, the scatter in $^{28}f/^{49}f$, 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 data have not observed this discrepancy is inconsistant 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.

REFERENCES

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