MASTER

1000 MWe

L M F B R

Liquid Metal Fast Breeder Reactor

FOLLOW-ON STUDY

WESTINGHOUSE ELECTRIC

TASK II AND III REPORT

Topical Report

Comparisons of the Westinghouse 1000 MWe

LMFBR Reference Design Calculations

Using the Westinghouse 6602-M Data

and the MC² Code with Neutron

Cross-section Data from the ENDF/B File

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REACTOR TECHNOLOGY (TID-4500)

COMPARISONS OF THE WESTINGHOUSE 1000-MWe LMFBR
REFERENCE DESIGN CALCULATIONS
USING
THE WESTINGHOUSE 6602-M DATA
AND THE
MC² CODE WITH NEUTRON CROSS-SECTION DATA
FROM
THE ENDF/B FILE

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CONTENTS

Title	Page
INTRODUCTION.....	1
DESCRIPTION OF CROSS-SECTION SETS.....	1
Standard Westinghouse 6602-M Library.....	1
MC ² Library.....	3
Description of ENDF/B File.....	3
Description of MC ² Computer Code.....	4
Subset Descriptions.....	4
COMPARISON OF CROSS-SECTION SETS.....	5
Pu-239 Fission Cross Sections.....	5
Pu-239 Capture-to-Fission Cross-Section Ratio (α).....	5
U-238 Absorption Cross Sections.....	6
Cross Sections for Lumped Fission Products.....	6
Transport and Capture Cross Sections for Sodium.....	6
Capture and Transport Cross Sections for Stainless Steel.....	6
DISCUSSION OF RESULTS.....	6
Effective Multiplication Factor.....	6
Breeding Ratio.....	8
Reactivity Effect of Sodium Voiding.....	8
Reactivity Effect of Doppler Broadening.....	8
COMPARISON OF CALCULATIONS WITH MC ² /ENDF/B AND ZPR III, ASSEMBLY 48 EXPERIMENTS.....	9
SUMMARY OF CROSS-SECTION COMPARISONS AND CALCULATIONAL RESULTS.....	9
CONCLUSIONS.....	9
APPENDIX I DESCRIPTION OF REACTOR.....	11
APPENDIX II CROSS-SECTION SETS.....	15
APPENDIX III CALCULATIONAL METHOD.....	22
APPENDIX IV COMPARISON OF FLUX SPECTRA.....	25
REFERENCES.....	28

FIGURES

Figure	Title	Page
I-1	One Quadrant of Four-Module Reference Design.....	12
IV-1	Core Averaged Flux Spectra Calculated with MOBI Code with 6602-M and MC ² Sets of Cross Sections.....	26

TABLES

Table	Title	Page
1	Fission Product Yield Ratios	7
2	Calculated Parameters.....	10
I-1	Equivalent Reactor Dimensions Used in Calculations.....	13
I-2	Volume Fractions of Reactor Constituents.....	14
II-1	Acquisition Numbers of Materials.....	16
II-2	Pu-239 Parameters.....	17
II-3	Absorption Cross Sections for U-238.....	18
II-4	Capture Cross Sections for Lumped Fission Products.....	19
II-5	Sodium Cross Sections.....	20
II-6	Stainless Steel Cross Sections.....	21
IV-4	Core Spectra 1000-MWe Reference Design.....	27

INTRODUCTION

Neutronic calculations were performed for the Westinghouse 1000-MWe LMFBR Task III reference design using both the standard Westinghouse 6602-M cross-section library and the ENDF/B cross-section data.

The parameters which formed the basis of the calculations were:

- a. effective neutron multiplication factor,
- b. breeding ratio,
- c. reactivity effect of voiding the core and axial blanket of sodium, and
- d. reactivity effect of Doppler broadening.

These ground rules were followed throughout this study:

- a. all calculations performed on the four-module Task III Reference Design,
- b. all calculations performed at mid-equilibrium core conditions, and
- c. all calculations performed with both the 6602-M and MC² cross-section libraries.

The results of the calculations are presented along with a comparison of the two cross-section libraries used.

DESCRIPTION OF CROSS-SECTION SETS

STANDARD WESTINGHOUSE 6602-M SET

The 6602-M multigroup microscopic cross-section library was generated at Westinghouse to study a broad class of fast reactors. This library contains the cross sections for about 40 nuclides in an 18-group energy structure covering an energy range from 30 ev to 10 Mev. The first 13 energy groups from 10 Mev down to 9.1 Kev are generally 0.5 lethargy units in width and are identical to the 16-group library published by Yiftah, Okrent, and Moldauer in 1960^[1] (YOM library). The groups below 9.1 Kev in the 6602-M library have more break points and extend below the YOM library, which has a minimum energy of 500 ev. This modification was made to more accurately

represent the softer spectra characteristic of large power fast reactors. Although the lower groups are less important than the high energy cross sections for the determination of criticality, they are very important for the determination of reactivity coefficients, such as Doppler coefficient and coolant void coefficient. These lower groups are necessary in calculations comparing the kinetic behavior and safety characteristics of various fast reactor concepts.

The YOM library has served as the basis for the 6602-M library. The cross-sections above 9.1 Kev for 22 of the nuclides were taken from YOM; however, certain modifications were made that take into account cross-section data made available since 1960, as discussed later. The cross sections for the remainder of the nuclides were taken from MUFT and GAM libraries.[2] Cross sections were compacted from the MUFT and GAM energy structure to the 18-group structure.

At energies below 100 Kev, the neutron source is due mainly to the moderation of the light elements. Therefore, accurate values of elastic removal cross sections of these elements are needed to determine reliable spectra and reactivity coefficients. Further-more, the presence of elastic scattering resonances of light elements over most of the energy range important to fast reactors produces strong perturbations in the neutron flux spectrum. The ELMOE Code, which has a very fine group structure, was written by Rago and Hummel[3] to take into account the problem of elastic moderation by light elements having elastic scattering resonances. The code produces flux-averaged multigroup microscopic elastic-removal cross sections for these elements. It does not directly produce averaged microscopic transport cross sections, but instead provides multigroup diffusion coefficients. Unfortunately, these averaged cross sections for the light elements are composition dependent because of the interaction of their resonances; therefore, a unique "best" library of moderate group structure for all types of fast reactors cannot be developed. The values of the elastic removal and transport cross sections of C, O, Na, and Type 304 stainless steel were selected for a large sodium-cooled power reactor from the work of Hummel, Bhide, and Rago. [4,5] The inelastic and absorption cross sections are from YOM except for σ_a for sodium, which is based on the work of Hummel et al [5] and Troubetszkoy et al.[6]

In compiling the 6602-M library, special attention was given to the important fuel isotopes, and efforts were made to update the values of $\bar{\nu}$ and $\alpha(E)$ appearing in the YOM library because fast reactor parameters are very sensitive to these values.[7] The $\bar{\nu}$ values for U-235, U-233, U-238, and Pu-239 were selected from the work of A. B. Smith presented at the International Symposium 1961[8] and the compilations appearing in the 1965 edition of BNL-325.[9] Straight lines for $\bar{\nu}$ versus energy were obtained by Smith by applying a least-square fit procedure to the data available in 1961. These data include the experimental study of D. Butler et al [10] and B. C. Diven and J. C. Hopkins.[11]

For U-235, the experiments indicate a non-linearity of $\bar{\nu}$ versus energy. The data could be fitted to two straight lines: one for energies up to 2 Mev, and the other for energies greater than 2 Mev. The Westinghouse Advanced Reactors Division's (WARD) 18-group library has adopted the first straight

line, namely: $\bar{\nu} = 2.42 + 0.138E$.

This value of $\bar{\nu}$ for U-235 is lower than that given in the YOM library. However, this is in agreement with the results of C. Cook and W. O. Davey.[12]

For Pu-240, $\bar{\nu}(E)$ was obtained by taking $\bar{\nu}(1.67 \text{ Mev}) = 3.1$ rather than 3.3 as used in YOM. The new values soften the spectrum which is in agreement with the measurements obtained in the core of Zephyr.[13] The Westinghouse Atomic Power Division's (WAPD) values of $\bar{\nu}$ for Pu-239 in the range of zero to 1 Mev are slightly lower than those of YOM, but are greater than YOM above 1 Mev.

The values of α for U-233 and Pu-239 are based on Argonne National Laboratory's (ANL) set 635[14], which is the YOM set modified to agree with the measurements of Diven and Hopkins[15], as well as on the work of Schmidt.[16] For Pu-240, values of capture cross section reported by Hummel et al[17] were used for energies less than 30 Kev. However, for design studies, the fission and capture cross sections below 30 Kev for fuel isotopes are re-evaluated for the appropriate composition and temperature by use of the ARES[18] Code. This code employs the Breit-Wigner single-level resonance theory and averages the resonance integrals over the Porter-Thomas distribution of resonance parameters.

The down scattering matrix for U-238 was taken from the British cross-section library FD-2.[19] In addition, the transport cross sections of U-238 were corrected to be consistent with the change in total scattering cross section. Values of $\nu\sigma_f$ for Pu-240 were replaced by values supplied by N. Azziz.[20]

MC² LIBRARY

A set of 18-group cross-sections was developed for the materials in the reactor by means of the MC²[21] computer code with the data in the Evaluated Nuclear Data File/B[22] (ENDF/B).

Description of ENDF/B File

The ENDF/B file is a library of evaluated nuclear data maintained on magnetic tape in a form that is readily usable by processing programs. The data in the file has been evaluated by users and subsequently accepted by the ENDF center for incorporation into the ENDF/B file.

Evaluation is defined as "the process of digesting the experimental data, combining it with the predictions of nuclear model calculations, and attempting to extract the true value of a cross section".[22]

After a user evaluates the data for a material (a material being either an isotope or a collection of isotopes) and finds it satisfactory, he then transmits the data to the ENDF center. If the data is accepted for ENDF/B, it is assigned a unique acquisition number. Other users may request that particular set of data by requesting the material having the assigned acquisition number.

Description of MC² Computer Code

MC² is a computer code written at ANL for generating "a complete cross-section set, using an Evaluated Nuclear Data File, suitable for direct input to neutronics codes without performing ancillary calculations or manipulations".

The code contains certain features, one or more of which are not found in previous cross-section codes.

MC² performs a detailed calculation in the resonance region. If the user desires, the code will calculate a weighting spectrum appropriate to a particular composition. Flexibility in regard to how many energy groups may be used to average over a given energy range is also a feature of the program.

Fine group calculations are performed for slowly varying data; whereas the rapidly varying data, such as elastic scattering and resonant cross sections, may be calculated with either ultra-fine groups or fine groups, depending on the options chosen.

Doppler-broadened line shapes are used to calculate cross sections in the resolved resonance region. The interference with overlapping resonances in other isotopes is allowed, together with interference between resonance and potential scattering.

Unresolved resonance region calculations are performed by averaging over suitable Porter-Tomas distributions of the neutron and fission widths. Both s- and p-wave calculations are performed; these include a summation over spin states in each case.

Fundamental mode weighting spectrum calculations may be performed in either ordinary P₁, consistent P₁, or B₁ approximations.

Subset Descriptions

The MC² cross-section set used in this study may best be considered as a collection of four subsets. Three of the 18-group subsets were calculated with core flux spectra, and one subset was calculated with a flux spectrum typical of the radial blanket region.

The first two subsets (one at 750°K and the other at 1500°K) were calculated for the isotopic conditions in the core with the coolant sodium present. The third subset was also calculated at 750°K, but with the coolant sodium removed.

The fourth subset was calculated for an isotopic condition typical of the blanket regions, with a temperature of 750°K and with the coolant sodium present.

All subsets were calculated with the following options in MC²:

- a. P-1 fundamental mode calculation,
- b. all fine group problems,
- c. constant weighting spectrum for averaging pointwise data over fine groups,
- d. weighting spectrum calculated internally,
- e. homogeneous resonance self-shielding option.

The core was represented as an infinite medium by entering a total buckling of 10^{-20} , whereas the blanket was represented as a finite medium by entering a negative buckling calculated for the radial blanket region.

In this report, the subsets of cross sections developed from the ENDF/B data will be referred to, collectively, as the MC² set.

COMPARISON OF CROSS-SECTION SETS

An MC² cross-section set was generated in a manner described in the previous section for the Westinghouse four-module design at the mid-equilibrium composition. Tabulated values for the cross-sections discussed are given in Appendix II.

A detailed comparison of the MC² set and the 6602-M cross sections revealed the differences presented below:

Pu-239 FISSION CROSS SECTIONS

The fission cross sections of Pu-239 in the MC² set were generally lower than those in the 6602-M set by about 10 to 20% in the energy range above 9 Kev.

Pu-239 CAPTURE-TO-FISSION CROSS-SECTION RATIO (α)

The ratio of capture-to-fission cross-section ratio (α) of Pu-239 was in reasonable agreement in the two sets for neutron energies above 20 Kev. Below this energy, the α values in the MC² set were approximately 20% lower than those in the 6602-M set.

U-238 ABSORPTION CROSS SECTIONS

The absorption cross sections for U-238 in the MC² set were somewhat lower than in the 6602-M set for the energy range above 15 Kev. From 15 to 1 Kev, the MC² set was higher by as much as 20%. In the energy ranges below 1 Kev, the MC² set was lower by as much as 40%.

CROSS SECTIONS FOR LUMPED FISSION PRODUCTS

The fission product capture cross sections used with the MC² set were calculated with Xe-135, Sm-149, Rapidly, Slowly, and Non-Saturating group data, which were weighted with the fission product yield ratios given in Table 1.

Since scattering cross-section data were not available in ENDF/B for the fission products, the transport cross sections from the 6602-M library were used in the MC² set. This was done with the realization that such usage would cause negligible changes in the neutronic calculations.

The capture cross sections of the lumped fission products were lower in the MC² set by nearly a factor of two.

TRANSPORT AND CAPTURE CROSS SECTIONS FOR SODIUM

The capture cross sections for sodium showed no uniform variations except that those in the MC² set were slightly higher than those in the 6602-M set for energies above 300 Kev.

The sodium transport cross sections were lower in the MC² set by approximately 20% for neutron energies below 1.35 Mev.

CAPTURE AND TRANSPORT CROSS SECTIONS FOR STAINLESS STEEL

The capture cross sections for stainless steel showed a uniform variation below 40 Kev, with the MC² cross sections being higher than those in 6602-M by as much as a factor of 3. In the range from 40 to 110 kev, the MC² set was lower by nearly a factor of 2. Above 110 Kev, the variations were random.

The MC² transport cross sections for stainless steel were generally lower than those in 6602-M by nearly 10 to 20%.

DISCUSSION OF RESULTS

EFFECTIVE MULTIPLICATION FACTOR

Initial calculations for the Reference Design showed that the MC² set predicted a K_{eff} approximately 7% lower than the K_{eff} calculated with the 6602-M data.

Table 1

Fission Product Yield Ratios^[22]

Group	Yield Ratio
Xe-135	0.0715
Sm-149	0.0130
Rapidly Saturating	0.0134
Slowly Saturating	0.4010
Non-Saturating	1.5011

This difference can be attributed, primarily, to the lower fission cross sections of Pu-239 in the MC² set.

Once the initial K_{eff} calculations were performed, the concentration of Pu-239 in the core regions was increased until a K_{eff} comparable to that calculated with the 6602-M set was obtained. In order to obtain the desired K_{eff}, the at.% of Pu-239 in the fuel had to be increased approximately 2% in the core regions.

All subsequent calculations with the MC² set were made with the higher Pu-239 concentration. The calculations with the 6602-M set were made with the lower value.

BREEDING RATIO

The breeding ratio calculated in Task II with the 6602-M cross sections was 1.33 at the mid-equilibrium condition. With the MC² set, the breeding ratio was found to be 1.25 at the same time in core life.

REACTIVITY EFFECT OF SODIUM VOIDING

The sodium removal reactivity effect was calculated assuming that the coolant was lost from only one of the four modules. Sodium was voided from both the core and the axial blanket regions for that one module, with the remainder of the reactor maintaining its normal complement of coolant.

When the MC² set was used to calculate the voiding effect, the reactivity changed by -33¢. The 6602-M set predicted that a reactivity change of -57¢ would occur when the sodium was removed from the same regions.

REACTIVITY EFFECT OF DOPPLER BROADENING

When the effect of Doppler broadening was calculated in Task III, the Doppler coefficient ($T_{dk} \frac{d\lambda}{dT}$) was -0.0024. Calculation of the Doppler coefficient for an isothermal temperature rise in the core and blankets using the MC² set also yielded a value of -0.0024.

A comparison by Adkins, Murley, and Dyos^[24] of Doppler coefficient calculations using MC² and a more rigorous method using a combination of the PSEUDO^[25] and RABBLE^[26] codes was performed. The comparison shows that the less rigorous treatment used in MC² to calculate resonance parameters in the unresolved resonance region yield Doppler coefficients that are in good agreement with the more sophisticated method using the PSEUDO-RABBLE combination.

COMPARISON OF CALCULATIONS WITH MC²/ENDFB AND ZPR III, ASSEMBLY 48 EXPERIMENTS

Calculations were performed by Pitterle, Page, and Yamamoto^[27] that compare criticality values calculated with the ENDF/B data and experimental values obtained from ZPR III, Assembly 48. The results of their study show that the ENDF/B data predict an effective multiplication factor that is approximately 1.5% too low. Since the Assembly 48 isotopic composition is similar to the composition of the Task III Reference Design studied here, one may conclude that the 6602-M library would be over-reactive in the criticality calculation by approximately 5.5%.

SUMMARY OF CROSS-SECTION COMPARISONS AND CALCULATIONAL RESULTS

Calculations with the two sets of cross sections showed that the 6602-M set₂ gave an effective multiplication factor approximately 7% higher than the MC² set. This effect was caused primarily by the differences in the Pu-239 fission cross sections in the upper energies and U-238 absorption cross sections in the resonance region.

The MC² set also predicted a lower breeding ratio, but at least a portion of the change can be attributed to the lower fertile-to-fissile atom ratio used when performing the calculations with MC².

A slightly less negative reactivity change for sodium voiding was calculated with MC², and the Doppler coefficient was identical with the two sets of cross sections.

The numerical values of the parameters are given in Table 2.

CONCLUSIONS

It can be seen from the results given above that the 6602-M data library is not a satisfactory library for design analyses. Future calculations will be made with the ENDF/B library modified as we feel is required.

Table 2
Calculated Parameters

	6602-M	MC ²
<u>Pu</u> Inner Core	12.9%	14.8%
<u>U+Pu</u> Outer Core	15.4%	18.4%
 <u>Fertile</u> <u>Fissile</u> (atom ratio)	 5.43	 4.70
 Breeding ratio	 1.33	 1.25
 Doppler coefficient	 -0.0024	 -0.0024
 Sodium void reactivity	 -57¢	 -33¢
 Percent of fission in U-238		
Core region 1:	12.1	12.2
Core region 2:	9.8	9.8
Radial blanket 1:	41.6	44.6
Radial blanket 2:	17.7	18.5

$$\beta_{\text{eff}} = 0.0037$$

APPENDIX I

DESCRIPTION OF REACTOR

A complete description of the reference reactor is contained in Reference 28, but a brief description of the calculational model used is given here for completeness.

All neutronic calculations were performed on one of the four modules, and the results were extrapolated to all four modules by techniques discussed in Appendix III. Figure I-1 shows one quadrant of the actual core that was replaced in the calculational model by a cylinder of equal volume and number densities. The dimensions of the equivalent geometry are given in Table I-1. Volume fractions of the reactor constituents are listed in Table I-2.

The reactor consists of four identical modules, each having two core regions surrounded by blanket material. The radial blanket is UO_2 , whereas the axial blanket, above and below the core, is loaded with UC. An inconel reflector surrounds the outer radial blanket.

Figure I-1 One Quadrant of Four-Module Reference Design

Table I-1

Equivalent Reactor Dimensions Used in Calculations

Region	Outer (in.)	Radius (in.)	Volume (M ³)
Inner fuel	16.63	42.23	0.559
Outer fuel	23.32	59.24	0.541
Inner radial blanket	28.89	73.39	0.588
Outer radial blanket	33.56	85.23	0.588
Reflector	36.36	92.35	0.396

	Region (in.)	Height (cm)	
Core and radial blanket	39.30	99.82	2.276
Top blanket	15.0	38.1	0.420
Bottom blanket	15.0	38.1	0.420

Table I-2

Volume Fractions of Reactor Constituents

Material	Core	Axial Blanket	Radial Blanket
(U-Pu)C	0.2759	--	--
UC	--	0.2759	--
UO ₂	--	--	0.4618
Cr	0.0091	0.0091	--
Na	0.5897	0.5723	0.3041
Type 316 stainless steel cladding	0.0720	0.0894	0.1649
Mo	0.0533	0.0533	0.0692

APPENDIX II

CROSS-SECTION SETS

The 18 energy group cross sections for both the 6602-M and the MC² sets are contained in this Appendix. The cross sections presented are for a core temperature of 750° with sodium present throughout the reactor. The MC² set was collapsed from 70 fine groups using a typical core spectrum.

Table II-1 lists the ENDF/B identification numbers of all materials used in this study. The cross-section data are tabulated as follows:

<u>Parameter</u>	<u>Table</u>
σ_f and α (Pu-239)	II-2
σ_a (U-238)	II-3
σ_c - Fission Products	II-4
σ_{tr} and σ_a (Na)	II-5
σ_a and σ_{tr} (Stainless Steel)	II-6

Table II-1

Acquisition Numbers of Materials

<u>Material</u>	<u>ENDF/B ID</u>
Pu-239	1051
Pu-240	1053
Pu-241	1054
Pu-242	1055
U-235	1044
U-238	1047
C	1010
Na-23	1059
Cr	1018
Fe	1020
Ni	1021
Mo	1025
016	1013
<u>Pu-239 Fission Products</u>	
Rapidly saturating	1052
Slowly saturating	1070
Non-saturating	1071
Xe-135	1026
Sm-149	1027

Table II-2

Pu-239 Parameters

Energy Group	Lower Energy of Group (ev)	σ_f		α	
		6602-M	MC ²	6602-M	MC ²
1	3.6678×10^6	1.83	1.956	0.00109	0.00920
2	2.2246×10^6	1.99	2.011	0.00503	0.00
3	1.3493×10^6	1.96	1.921	0.01020	0.00521
4	8.1839×10^5	1.81	1.752	0.02486	0.02683
5	4.9638×10^5	1.70	1.619	0.05529	0.05621
6	3.0107×10^5	1.70	1.500	0.0900	0.09067
7	1.8261×10^5	1.70	1.481	0.1100	0.1215
8	1.1076×10^5	1.70	1.516	0.1400	0.1385
9	6.7166×10^4	1.80	1.514	0.1400	0.1448
10	4.0745×10^4	1.93	1.478	0.1503	0.1996
11	2.4713×10^4	2.225	1.620	0.3219	0.3056
12	1.4989×10^4	2.507	1.835	0.4750	0.3935
13	9.0915×10^3	2.488	2.096	0.6150	0.4575
14	4.0×10^3	2.381	2.484	0.6728	0.5073
15	1.0×10^3	3.336	4.294	0.7698	0.6181
16	3.0×10^2	6.329	6.790	0.8300	0.6495
17	1.0×10^2	9.932	15.00	0.8619	0.5587
18	4.1399×10^{-2}	13.529	25.76	0.8522	0.2283

Table II-3

Absorption Cross Sections for U-238

<u>Group</u>	<u>6602-M</u>	<u>MC²</u>
1	0.619	0.6551
2	0.601	0.5869
3	0.490	0.4866
4	0.154	0.1710
5	0.140	0.1410
6	0.130	0.1348
7	0.170	0.1510
8	0.220	0.1814
9	0.280	0.2358
10	0.350	0.3361
11	0.456	0.4456
12	0.538	0.5697
13	0.612	0.6953
14	0.710	0.8495
15	0.860	1.006
16	1.276	1.252
17	2.341	1.387
18	2.594	1.871

Table II-4

Capture Cross Sections for Lumped Fission Products

<u>Group</u>	<u>6602-M</u>	<u>MC²</u>
1	0.020	0.0100
2	0.040	0.0145
3	0.063	0.01967
4	0.090	0.02487
5	0.092	0.03032
6	0.095	0.03834
7	0.110	0.05105
8	0.125	0.0700
9	0.140	0.09398
10	0.165	0.12580
11	0.200	0.16570
12	0.250	0.21780
13	0.330	0.28580
14	1.50	0.3900
15	3.00	0.77110
16	14.00	1.5800
17	25.00	2.8130
18	26.00	6.6790

Table II-5

Sodium Cross Sections

Group	σ_{tr}		σ_a	
	6602-M	MC ²	6602-M	MC ²
1	1.30	1.416	0.0005	0.01723
2	1.51	1.650	0.0001	0.00018
3	1.67	1.910	0.0001	0.00020
4	2.76	2.558	0.0002	0.00023
5	4.76	3.671	0.0003	0.00032
6	3.76	3.096	0.0005	0.00037
7	4.08	3.426	0.0007	0.00073
8	5.57	3.315	0.0	0.00075
9	4.38	3.600	0.0	0.00028
10	4.97	4.620	0.003	0.0083
11	5.18	4.019	0.0	0.00174
12	4.19	4.446	0.0	0.00070
13	5.05	5.031	0.0	0.00129
14	10.02	7.753	0.0014	0.00260
15	11.25	10.92	0.053	0.02298
16	3.01	3.148	0.0048	0.00572
17	3.02	2.876	0.006	0.00678
18	3.01	2.910	0.010	0.00989

Table II-6

Stainless Steel Cross Sections

Group	σ_a		σ_{tr}	
	6602-M	MC ²	6602-M	MC ²
1	0.00140	0.04048	1.96	2.057
2	0.00120	0.01260	2.39	2.312
3	0.00210	0.00387	2.00	2.330
4	0.00370	0.00352	1.96	2.076
5	0.00503	0.00473	2.31	2.287
6	0.00503	0.00525	2.92	2.852
7	0.00612	0.00540	3.02	2.648
8	0.00520	0.00574	4.01	3.434
9	0.01040	0.00610	4.73	3.596
10	0.01110	0.00686	4.39	4.034
11	0.01790	0.02096	7.92	5.153
12	0.01050	0.01747	4.7	4.258
13	0.01520	0.02094	7.5	7.591
14	0.02130	0.06433	13.55	12.21
15	0.02830	0.07594	8.75	7.85
16	0.03640	0.04195	10.67	9.15
17	0.04450	0.05356	11.88	10.27
18	0.06580	0.07317	10.86	10.38

APPENDIX III

CALCULATIONAL METHOD

To obtain an equivalent value of K_{eff} for the multimodule configuration, weighted terms of the neutron balance determined from the one-module calculations were evaluated for two different outside boundary conditions.

a. $\phi = 0$, flux equals zero at the outer boundary, and

b. $J = 0$, net current equals zero at the outer boundary.

The weighting of the terms in the neutron balance was done by numbers proportional to the solid angle subtended at the center of the core by the other core modules and by the outer boundary of the configuration. Only vertical surfaces were considered. It was assumed that half the vertical surface of each outside module faced other module, and half faced the core exterior. The inside module faced other modules only. Assigning a weight of one-half to each of the first conditions, gives:

$$W_{\phi=0} = 4 \times \frac{1}{2} = 2$$

$$W_{J=0} = 4 \times \frac{1}{2} = 2$$

For the condition where one module was voided of sodium and three were flooded, weighting factors are:

$$\text{Voided: } W_{\phi=0} = 1 \times \frac{1}{2} = \frac{1}{2}$$

$$W_{J=0} = 1 \times \frac{1}{2} = \frac{1}{2}$$

$$\text{Flooded: } W_{\phi=0} = 3 \times \frac{1}{2} = \frac{3}{2}$$

$$W_{J=0} = 3 \times \frac{1}{2} = \frac{3}{2}$$

The value of K_{eff} for a single module and a multimodule core are related through the terms of their neutron balance equation. Let B_R^2 and B_T^2 represent the radial and transverse components of the geometric buckling, and let $\phi=0$ and $J=0$ denote the quantities obtained from calculations with the respective boundary conditions. For a single module with a homogeneous boundary condition (either $\phi=0$ or $J=0$), K_{eff} is given by:

$$K_{eff} = \frac{\sum_{i=1}^{18} \sum_{j=1}^5 (\nu \Sigma_f \phi)_{ij}}{\sum_{i=1}^{18} \sum_{j=1}^5 \left[(\Sigma_a \phi)_{ij} + (DB_R^2 \phi)_{ij} + (DB_T^2 \phi)_{ij} \right]}$$

Subscript "i" refers to the energy groups and subscript "j" refers to the core and blanket regions.

For modules with a zero current condition ($J=0$), the radial leakage term

$$\sum_{i=1}^{18} \sum_{j=1}^5 (DB_R^2 \phi)_{ij}$$

is zero. For a reactor composed of modules sharing the boundary conditions in different proportions, the value of K_{eff} for the configuration was obtained by weighting the factors in the neutron balance as follows:

$$K_{eff} = \frac{W_{J=0} \left[\sum_{i=1}^{18} \sum_{j=1}^5 (\nu \Sigma_f \phi)_{ij} \right]_{J=0} + W_{\phi=0} \left[\sum_{i=1}^{18} \sum_{j=1}^5 (\nu \Sigma_f \phi)_{ij} \right]_{\phi=0}}{W_{J=0} \left[\sum_{i=1}^{18} \sum_{j=1}^5 (\Sigma_a \phi)_{ij} + (DB_T^2 \phi)_{ij} \right]_{J=0} + W_{\phi=0} \left[\sum_{i=1}^{18} \sum_{j=1}^5 (\Sigma_a \phi)_{ij} + (DB_T^2 \phi)_{ij} + (DB_R^2 \phi)_{ij} \right]_{\phi=0}}$$

The values of the weights $W_{J=0}$ and $W_{\phi=0}$ were given above, and all the remaining factors came from MOBI [29,30,31] neutron balance evaluations.

To calculate the reactivity effect due to sodium voiding, the following set of ground rules were used:

- For the four-module core, void one of the modules; assume the other three modules flooded.
- In case(a), also void the top and bottom axial blanket of the voided module.

- c. Use isotopic inventory at the mid-equilibrium cycle concentrations.
- d. Compute the sodium void reactivity as follows:

$$\left[\frac{\Delta K_{\text{eff}}}{K_{\text{eff}}} \right]_{\text{Void}} = \frac{\left[K_{\text{eff}} \right]_{\text{Voided}} - \left[K_{\text{eff}} \right]_{\text{Flooded}}}{\left[K_{\text{eff}} \right]_{\text{Voided}}}$$

All K_{eff} calculations were made with the one-dimensional diffusion theory code MOBI. Iterations on the material bucklings were performed on both a neutron energy group and reactor region basis.

APPENDIX IV

FLUX SPECTRA

Figure IV-1 shows the core-averaged flux spectra, calculated with the MOBI code, with both the 6602-M and the MC² sets of cross sections. The core conditions for the calculations were:

- a. mid-equilibrium cycle isotopic concentrations,
- b. temperature of materials at 750°K, and
- c. sodium present in all four modules.

The tabulated spectra are given in Table IV-1.

Figure IV-1 Core Averaged Flux Spectra Calculated with MOBI Code with 6602-M and MC² Sets of Cross Sections

Table IV-1

Core Spectra 1000-MWe Reference Design

<u>E_{Lower}</u> (ev)	<u>6602-M</u>	<u>MC²</u>
	<u>Relative Flux</u>	<u>Relative Flux</u>
3.667×10^6	0.01694	0.01981
2.225×10^6	0.03134	0.03408
1.349×10^6	0.051802	0.04887
8.184×10^6	0.066362	0.06597
4.964×10^5	0.102802	0.09547
3.011×10^5	0.12749	0.12593
1.826×10^5	0.122551	0.12553
1.108×10^5	0.109331	0.11716
6.718×10^4	0.094561	0.10121
4.075×10^4	0.086818	0.07646
2.471×10^4	0.057453	0.06294
1.499×10^4	0.049653	0.05223
9.092×10^3	0.031604	0.03312
4.00×10^3	0.021647	0.02380
1.00×10^3	0.004785	0.00335
3.00×10^2	0.004785	0.00335
1.00×10^2	0.000431	0.00017
3.00×10^1	0.000016	0.000012

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