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ANALYTICAL PROCEDURES AND APPLICATIONS OF FLUENCE DETERMINATIONS FROM EBR-II FLUX WIRES

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U.S. ATOMIC ENERGY COMMISSION
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PROJECT AGREEMENT 10

BREEDER REACTOR DEVELOPMENT OPERATION

GENERAL ELECTRIC

SUNNYVALE, CALIFORNIA

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ANALYTICAL PROCEDURES AND APPLICATIONS
OF FLUENCE DETERMINATIONS FROM EBR-II FLUX WIRES

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ABSTRACT

A detailed review has been prepared of the General Electric methods and data used in determining flux levels in the EBR-II by use of flux-wire analysis. Comments are offered on the accuracies involved in these methods as compared to the methods employed by the Argonne National Laboratory. In addition, other measurements of fluence are analyzed. Neutron energy ranges which can be best expected to show correlations with possible damage mechanisms are discussed with suggestions as to the best means of "measuring" the neutron fluxes over the energy ranges of interest.

SECTION I

SUMMARY

Three experimental subassemblies containing materials, capsules, and flux wires were irradiated in the EBR-II to study the effects of fast-neutron exposure upon the mechanical and physical properties of austenitic stainless steels.

The nature of the (n,p) threshold reactions in the analysis of iron and nickel flux wires makes this method particularly applicable to the measurement of fluences in the MeV range of neutron energies. When flux wires were analyzed it was determined that EBR-II subassemblies XG06, XO09, and XO14 were irradiated to a fluence greater than 1 MeV, of 6.7×10^{21} , 3.9×10^{21} , and 3.0×10^{21} n/cm², respectively.

Total fluence may be inaccurate if flux-wire data are used as the basis for extrapolation.

The present flux-wire analyses may have errors well in excess of 20 percent if applied to measurements of the total fluence of the samples. The corresponding inaccuracy in the power normalization approach (ANL) should be considerably less. Conversely, the flux-wire analysis can be expected to give more accurate measurements of the high energy fluence (above 1-2 MeV) than the ANL approach. It appears that the differences in the two approaches may be caused by an inexact calculated flux spectra, which is too hard (too high above 1 MeV) because of cross section uncertainties and spatial variations in the fission neutron source.

It was determined that for the three EBR-II subassemblies analyzed, the following conversion factors apply for estimating fluence at 45 MW:

- Total fluence:

Row 2	4.7×10^{18} (n/cm ²)/MWd
Row 4	3.7×10^{18} (n/cm ²)/MWd

- Fast fluence ($E_n > 1$ MeV):

Row 2	8.2×10^{17} (n/cm ²)/MWd
Row 4	7.2×10^{17} (n/cm ²)/MWd

It was found that ANL total flux estimates were 60% higher than the General Electric results in a row-2 position and about 40% higher in a row-4 position.

Correlations between various types of radiation damage in stainless steel and neutron fluence should consider the particular damage mechanism and the part of the neutron energy spectrum producing it.

SECTION II

CONCLUSIONS

Total neutron fluence of materials irradiated in EBR-II is currently best determined by neutron flux computations normalized to the reactor power.

Fluence above 1 MeV is more accurately determined by an analysis of iron and nickel flux wires located adjacent to the samples.

The integrated fast-neutron exposures ($E_n > 1$ MeV) for EBR-II subassemblies XG06

(series L-2'), XO09 (series L-4), and XO14 (series L-4, and MT-5, 6) are 6.7×10^{21} , 3.9×10^{21} , and 3.0×10^{21} n/cm², respectively, as determined from a flux-wire analysis.

The ANL power normalization approach for determining neutron fluence in EBR-II yield a value 60% greater than the General Electric dosimetry methods for a row-2 location and 40% higher values at a row-4 position.

SECTION III

INTRODUCTION

Fast-neutron damage to austenitic stainless steels at elevated temperatures is thought to be associated with the production of helium and of cavities⁽¹⁻⁴⁾ and their effects on mechanical and physical properties. Mechanisms regarding the formation of cavities are still subject to question although it is generally agreed that they are the result of a combination of two discrete phenomena: gas producing neutronic reactions $[(n, \alpha) \text{ reactions primarily with iron, chromium, nickel, and nitrogen}]$ and vacancy-producing displacement collisions.

Lattice displacements begin to occur at neutron energies around 10 keV⁽⁵⁾ and helium production is believed to have a reaction thresh-

old at about 6 MeV. Thus the knowledge of the total flux as well as the high energy region of the neutron spectrum in an experimental fast reactor permits proper investigation of fast-neutron damage in austenitic stainless steels.

Flux wires have been used in material irradiation experiments in the EBR-II as a means for determining the integrated exposure over a particular energy range during reactor exposure. At General Electric, the integrated fast-neutron exposure was monitored by the reaction Fe-54 (n,p) Mn-54, which has a half life of 303 days and Ni-58 (n,p) Co-58 with a half life of 71 days.

The objective of this report is to review the methods used by General Electric in determining flux levels in the EBR-II, to present the dosimetry results from three irradiation experi-

ments, and compare these results with ANL calculations, and to comment on dosimetry information as related to radiation-damage mechanisms.

SECTION IV

EXPERIMENTAL PROCEDURE

4.1 RADIATION HISTORY

This report describes the dosimetry results from four General Electric irradiation experiments in the EBR-II. These are designated series L-2', L-4, L-4', MT-5 and 6. Series L-2' consisted of five capsules; L-4 contained two capsules, while L-4' consisted of five capsules; the MT-5 and 6 series contained two capsules. Table 4-1 summarizes the exposure history for each of these series. The reactor loading diagram for the three sub-assemblies is seen in Figure 4-1, while Figures 4-2 through 4-4 present the capsule positions within each subassembly. Series L-2' experienced 18 power cycles while series L-4 was cycled for 8 times. The combined series L-4' and MT-5 and 6 was cycled for 6 times. The time out-of-pile (the time from completion of the irradiation exposure to counting the flux-wire activity) was 152 days (L-2'), 199 days (L-4), and 224 days (L-4'/MT-5 and 6).

4.2 ANALYSIS

The nature of the reactions involved in the flux wire analyses of Mn-54 and Co-58 activities makes this measurement particularly applicable to the measurement of fluences in the MeV range of neutron energies. The thresholds of the (n,p) reactions which produce these activities are between 1 and 2 MeV. Consequently, the measured activities are independent of the neutron flux spectrum below 1 MeV, and any application of these techniques to the measurement of fluences below this

energy must rely upon either a measured or computed flux spectrum. Methods for measuring flux spectra from a few keV to 10 MeV are currently being developed, but a carefully computed flux spectrum can be expected to give equally accurate results for this application. The calculated neutron energy spectrum which was used for EBR-II analysis is shown in Figure 4-5.

In generating the cross sections used in the EBR-II flux-wire analyses, the following procedure was used. In measuring flux levels by use of (n,p) reactions, a single average value of the (n,p) cross section was used in the expression*:

$$\phi = \frac{\lambda [N_f(t_2) e^{\lambda t_2} - N_f(0) e^{-\lambda t_1}]}{N_I [1 - e^{-\lambda t_1}] \bar{\sigma}} \quad (1)$$

where:

λ = the decay constant of the product isotope of the (n,p) reaction

$N_f(0)$ = nuclear concentration of the product isotope prior to irradiation

t_1 = irradiation time

t_2 = time between end of irradiation and counting

* Equation (1) is for a single irradiation period. Expansion to multiple irradiation periods does not affect the relationship of ϕ to $\bar{\sigma}$.

TABLE 4-1

EBR-II EXPOSURE HISTORY FOR SERIES L-2', L-4, L-4', AND MT-5 AND 6SERIES L-2'

Subassembly Loaded	XG06 9/3/65	Reactor Position Removed	4E2 (Row 4) 2/20/67
-----------------------	----------------	-----------------------------	------------------------

<u>Capsule</u>	<u>Subassembly Position No.</u>
L-2K	2
L-2M	8
L-2O	17
L-2P	15
L-2Q	16

Runs No. 9 through 24 (9317 MWd, 207 EFPD*)

SERIES L-4

Subassembly Loaded	XO09 3/24/66	Reactor Position Removed	4A2 (Row 4) 11/14/66
-----------------------	-----------------	-----------------------------	-------------------------

<u>Capsule</u>	<u>Subassembly Position No.</u>
L-4C	2
L-4D	17

Runs No. 15 through 22 (5355 MWd, 119 EFPD*)

SERIES L-4'/MT-5 AND 6

Subassembly Loaded	XO14 7/17/66	Reactor Position Removed	2D1 (Row 2) 4/10/67
-----------------------	-----------------	-----------------------------	------------------------

<u>Capsule</u>	<u>Subassembly Position No.</u>
L-4A	8
L-4B	5
L-4E	3
L-4F	4
L-4G	2
MT-5	11
MT-6	19

Runs No. 20 through 24 (3674 MWd, 82 EFPD*)

* Effective full power days at 45 MW = Total MWd/45 MW/day

FIGURE 4-1. EBR-II LOADING PATTERN

FIGURE 4-2. SUBASSEMBLY DIAGRAM NO. XG06 (Core Location 4E2)

FIGURE 4-3. SUBASSEMBLY DIAGRAM NO. X009 (Core Location 4A2)

FIGURE 4-4. SUBASSEMBLY DIAGRAM NO. X014 (Core Location 2D1)

FIGURE 4-5. EBR-II NEUTRON ENERGY SPECTRUM

$N_f(t_2)$ = nuclear concentration of product isotope at time of counting

N_I = nuclear concentration of isotope having (n,p) cross section (assumed constant)

Equation (1) is derived from the more general form:

$$\int_0^\infty \sigma(E) \phi(E) dE = \frac{\lambda [N_f(t_2) e^{\lambda t_2} - N_f(0) e^{\lambda t_1}]}{N_I [1 - e^{-\lambda t_1}]} \quad (2)$$

where the energy dependence of the cross section is maintained. To obtain Equation (1) from Equation (2), the average cross section must be defined by:

$$\begin{aligned} \bar{\sigma}_{>E_L} &= \frac{\int_0^\infty \sigma(E) \phi(E) dE}{\int_{E_L}^\infty \phi(E) dE} \\ &= \frac{\int_0^\infty \sigma(E) \phi(E) dE}{\phi_{>E_L}} \end{aligned} \quad (3)$$

where E_L is an arbitrary lower energy limit. [Since the (n,p) reaction is effectively a threshold reaction, the lower limit of integration in the numerator can be set at a value just below this threshold energy, E_{th} .] Substituting the $\bar{\sigma}_{>E_L}$ obtained from Equation (3) into Equation

(1) gives the integrated flux above energy E_L . Thus, Equation (1) should be written:

$$\phi_{>E_L} = \frac{\lambda [N_f(t_2) e^{\lambda t_2} - N_f(0) e^{-\lambda t_1}]}{N_I [1 - e^{-\lambda t_1}] \bar{\sigma}_{>E_L}} \quad (4)$$

Tables 4-2 and 4-3 list values of $\bar{\sigma}_{>E_L}$ for (n,p) reactions in Fe-54 and Ni-58 and for $E_L = 0, 0.1 \text{ MeV}, 0.5 \text{ MeV}, 1 \text{ MeV}, \text{ and } 3 \text{ MeV}$. These values of $\bar{\sigma}_{>E_L}$ were computed by averaging the evaluated pointwise cross sections for the (n,p) reaction in Fe-54 and Ni-58 over an energy range from 3.7 MeV to 20 MeV, using a flux weighting factor proportional to the fission neutron spectrum for U-235. Average cross sections were then computed over energy ranges above 0, 0.1, 0.5, 1, and 3 MeV, using flux weighting factors which were obtained from a 60-group diffusion theory solution for three positions in EBR-II and normalized to the fission neutron spectrum above 3.7 MeV. Integrated fluxes above the specified energies E_L were computed with Equation (4), using the appropriate values of $\bar{\sigma}_{>E_L}$ from Tables 4-2 and 4-3.

Sections of the irradiated iron and nickel flux wire were weighed and dissolved in acid and appropriate aliquots were mounted to assay for Mn-54 and Co-58.

Corrections for decay and activation rates versus reactor power history were made in the following equation, which describes the neutron exposure above a particular energy level:

$$\phi_t > X \text{ MeV} = \frac{A e^{\lambda t_0} \sum_n P_n T_n}{(10^{-24}) N \sigma_x \sum_n P_n (1 - e^{-\lambda T_n}) (e^{-\lambda (D_n + T_n + 1)})} \quad (5)$$

TABLE 4-2

AVERAGE CROSS SECTION VALUES FOR THE (n,p) REACTION IN Fe-54

		$\bar{\sigma}_{>E_L}$				
E_L (MeV) =		0	0.1	0.5	1	3
1.	U-235 Fission Neutron Spectrum	--	--	85	109	365
2.	EBR-II					
	Row 2	24	26	48	85	365
	Row 4	23	25	47	85	365
	Row 6	13	15	35	77	390

TABLE 4-3

AVERAGE CROSS SECTION VALUES FOR THE (n,p) REACTION IN Ni-58

		$\bar{\sigma}_{>E_L}$				
E_L (MeV) =		0	0.1	0.5	1	3
1.	U-235 Fission Neutron Spectrum	--	--	126	161	540
2.	EBR-II					
	Row 2	35	38	69	125	550
	Row 4	33	36	68	124	545
	Row 6	19	22	51	112	565

where:

$\phi_t > X \text{ MeV}$	= total neutron exposure above X MeV	P_n	= power level* for cycle
A	= $\frac{\text{dps Mn-54}}{\text{gm Fe}}$ or $\frac{\text{dps Co-58}}{\text{gm Ni}}$	T_n	= irradiation time* for cycle
λ	= decay constant $T_{1/2} = 303 \text{ days - Mn-54}$ $T_{1/2} = 71 \text{ days - Co-58}$	D_n	= "down time" between cycles n and n+1*
t_o	= time out-of-pile	σ_x	= average reaction cross section (barns) for neutrons greater than X MeV (Tables 4-2 and 4-3)
N	= atoms of iron or nickel		
n	= identifies the power cycle*		

Calculations for dps/gm were obtained from the following equation:

$$\text{dps/gm} = \frac{(\gamma\text{-photopeak counts}) (\text{counting factor}) (\text{dilution factor})}{(\text{counting time}) (\text{sample weight}) (60)}$$

Twelve examples of these data are tabulated in Table 4-4 for six-material capsules (two capsules from each of the three series examined). Column 8 lists specific activity at count time. 2σ counting errors are $\pm 3\%$ (relative) and $\pm 7\%$ (absolute). Weighing and aliquoting errors are $< 3\%$.

In reviewing this procedure, the following estimates of uncertainties were made:

- The evaluated (n,p) pointwise cross sections are those recommended by Schmidt⁽⁶⁾. There is a fair amount of consistent data for these cross sections and Schmidt's values still look reasonable. The error in these values is estimated to be in the range of 10 to 20 percent.
- A comparison of the results shown in Tables 4-2 and 4-3 for the U-235 fission

neutron spectrum and the EBR-II spectrum shows that care must be used in applying the usual assumption that the high energy neutron flux spectrum follows the fission neutron spectrum. The integrated flux above 1 MeV in EBR-II is 20% higher than that obtained assuming a fission neutron spectrum, while the integrated flux above 0.5 MeV is a factor of 2 higher. From this it may be concluded that the fission neutron spectrum approximation should not be used for fluxes below 3 MeV. Above 3.7 MeV the assumption that the flux spectrum follows the fission neutron spectrum should be good and the average cross sections in this range should be accurate to within 10 to 20 percent.

- The use of a computed flux spectrum to extend the average cross sections to lower energy limits introduces a sizeable possible error because of the coarseness of the high energy groups used in the flux solution.

* Obtained from EBR-II quarterly run reports.

TABLE 4-4

FLUX-WIRE ANALYSIS

Code	Isotope	Date Counted	γ -Photopeak Counts ($\times 10^5$)	Dilution Factor	Counting Time (seconds)	Sample Weight (mg)	dps/gm $\times 10^{10}$	Row Position (S/A)	Axial Dist. from Core Bottom (inches)	Exposure Runs
L2K-3	Mn-54	6-2-67	1.43	498	120	17.0	0.443	4(XG06)	3-1/8	Run No. 9 through Run No. 24
L2K-3	Co-58		1.19	498	120	30.1	3.77		3-1/8	
L2Q-2	Mn-54		1.20	498	120	15.0	0.421		10-3/8	
L2Q-2	Co-58		0.982	498	120	30.0	3.12		10-3/8	
L4C-5	Mn-54		1.28	588	240	16.9	0.235	4(XO09)	2-1/8	Run No. 15 through Run No. 22
L4C-5	Co-58		1.03	498	240	30.3	1.62		2-1/8	
L4C-4	Mn-54		0.817	498	120	15.5	0.295		6-6/8	
L4C-4	Co-58		1.15	498	240	26.8	2.05			
L4E-4	Mn-54	8-12-67	0.867	250	120	9.6	0.238	2(XO14)		Run No. 20 through Run No. 24
L4E-4	Co-58		0.495	500	120	25.4	1.86			
L4G-4	Mn-54		0.919	250	120	10.1	0.240			
L4G-4	Co-58		0.492	500	120	27.3	1.72			

Counting factors:

Mn-54; 7.60×10^3 Co-58; 1.38×10^5

uncertainties in nuclear data, and possible spatial variations in the flux spectrum. These errors depend upon the accuracy of the computed ratio of flux below 3.7 MeV to that above. Therefore, the accuracy of this step decreases as the lower energy limit of the range considered, E_L , is decreased.

The error in the average cross section used for determining fluences above 3 MeV

should not be much greater than the 10 to 20 percent error of the cross section above 3.7 MeV. This is substantiated by the independent fluence results for the two different activations agreeing within this accuracy. At the other extreme, the average cross sections for total fluence ($E_L = 0$) may have an error considerably in excess of 20 percent (perhaps as high as 80 percent), which would then emerge as a sizeable error in the total measured fluence.

SECTION V

RESULTS

The counting data obtained with flux wires from capsule series L-2', L-4, L-4', and MT-5 and 6 irradiated in the EBR-II were analyzed using the described General Electric methods. In Tables 5-1 and 5-2 are listed the dosimetry results for iron and nickel flux wires, respectively. These tables show the fluence values for various axial positions at four neutron energy levels: (1) $E_n > 0.1$ MeV, (2) $E_n > 0.5$ MeV, (3) $E_n > 1$ MeV, and (4) $E_n > 3$ MeV. Figure 5-1 illustrates the fluence profile, at $E_n > 1.0$ MeV for the three subassemblies examined.

Capsule series L-2' received a measured neutron fluence of 2.5×10^{22} n/cm², $E_n > 0.1$ MeV (6.7×10^{21} n/cm², $E_n > 1$ MeV); the series L-4 exposure was 1.4×10^{22} n/cm², $E_n > 0.1$ MeV (3.9×10^{21} n/cm², $E_n > 1$ MeV); and series L-4' and MT-5 and 6 were irradiated

to 1.1×10^{22} n/cm², $E_n > 0.1$ MeV (3.0×10^{21} n/cm², $E_n > 1$ MeV).

Argonne National Laboratory, at the EBR-II site uses a neutronics calculation based upon a power normalization to estimate the total fluence. Table 5-3 shows a comparison between the fluence results obtained from the ANL computation and the General Electric flux-wire analysis. It was assumed that the fast flux ($E_n > 1$ MeV) was about 28% of the total flux (see Figure 4-5). Also tabulated are the ANL conversion factors used to estimate the fluence and the conversion factors obtained from the General Electric dosimetry analysis.

On the average, the ANL flux estimates seem to be about 60% higher than the General Electric dosimetry results for row-2 and about 40% higher values in row-4 position.

TABLE 5-1

DOSIMETRY DATA FOR REACTION Fe-54 (n,p) Mn-54

Capsule Section No.	dps Mn-54 gm Iron	$\phi 45\text{MW n/cm}^2$ ($E_n > 0.5\text{MeV}$)	$\phi t, \text{n/cm}^2$ ($E_n > 0.1\text{MeV}$)	$\phi t, \text{n/cm}^2$ ($E_n > 0.5\text{MeV}$)	$\phi t, \text{n/cm}^2$ ($E_n > 1.0\text{MeV}$)	$\phi t, \text{n/cm}^2$ ($E_n > 3.0\text{MeV}$)	Approximate Distance from Bottom of Core (inches)
<u>Series L-2'</u> (Counted at 6/2/67)							
L2K-1	1.08×10^9	1.7×10^{14}	5.5×10^{21}	2.9×10^{21}	1.6×10^{21}	3.8×10^{20}	17-3/8
L2K-3	4.43×10^9	6.8×10^{14}	2.3×10^{22}	1.2×10^{22}	6.7×10^{21}	1.6×10^{21}	3-1/8
L2M-2	4.59×10^9	7.1×10^{14}	2.4×10^{22}	1.2×10^{22}	6.9×10^{21}	1.6×10^{21}	10-3/8
L2M-3*	4.04×10^9	1.4×10^{15}	1.8×10^{22}	9.7×10^{21}	5.5×10^{21}	1.3×10^{21}	3-1/8
L2O-1	9.83×10^8	1.5×10^{14}	5.0×10^{21}	2.7×10^{21}	1.5×10^{21}	3.4×10^{20}	17-3/8
L2P-2	3.96×10^9	6.1×10^{14}	2.0×10^{22}	1.1×10^{22}	5.9×10^{21}	1.4×10^{21}	10-3/8
L2P-3	3.88×10^9	6.0×10^{14}	2.0×10^{22}	1.0×10^{22}	5.8×10^{21}	1.4×10^{21}	3-1/8
L2Q-1	8.91×10^8	1.4×10^{14}	4.6×10^{21}	2.4×10^{21}	1.3×10^{21}	3.1×10^{20}	17-3/8
L2Q-2	4.21×10^9	6.5×10^{14}	2.2×10^{22}	1.1×10^{22}	6.3×10^{21}	1.5×10^{21}	10-3/8
<u>Series L-4</u> (Counted at 6/2/67)							
L4C-1	2.11×10^8	5.2×10^{13}	1.0×10^{21}	5.5×10^{20}	3.0×10^{20}	7.1×10^{19}	20-4/8
L4C-2	6.65×10^8	1.6×10^{14}	3.3×10^{21}	1.7×10^{21}	9.6×10^{20}	2.2×10^{20}	16
L4C-3	2.17×10^9	5.3×10^{14}	1.1×10^{22}	5.7×10^{21}	3.1×10^{21}	7.3×10^{20}	11-3/8
L4C-4	2.95×10^9	7.2×10^{14}	1.4×10^{22}	7.7×10^{21}	4.3×10^{21}	9.9×10^{20}	6-6/8
L4C-5	2.35×10^9	5.8×10^{14}	1.2×10^{22}	6.1×10^{21}	3.4×10^{21}	7.9×10^{20}	2-1/8
L4D-4	2.41×10^9	5.9×10^{14}	1.2×10^{22}	6.3×10^{21}	3.5×10^{21}	8.1×10^{20}	6-6/8
<u>Series L-4'</u> (Counted at 8/12/67)							
L4A-1	1.58×10^8	5.5×10^{13}	7.0×10^{20}	3.8×10^{20}	2.1×10^{20}	5.0×10^{19}	20-4/8
L4A-2	6.10×10^8	2.1×10^{14}	2.7×10^{21}	1.5×10^{21}	8.3×10^{20}	1.9×10^{20}	16
L4A-3	1.79×10^9	6.2×10^{14}	8.0×10^{21}	4.3×10^{21}	2.4×10^{21}	5.7×10^{20}	11-3/8
L4A-4	2.24×10^9	7.7×10^{14}	1.0×10^{22}	5.4×10^{21}	3.0×10^{21}	7.1×10^{20}	6-6/8
L4A-5	1.81×10^9	6.3×10^{14}	8.0×10^{21}	4.4×10^{21}	2.5×10^{21}	5.7×10^{20}	2-1/8
L4B-2	5.74×10^8	2.0×10^{14}	2.5×10^{21}	1.4×10^{21}	7.8×10^{20}	1.8×10^{20}	16

* Counted at 8/12/67.

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TABLE 5-1 (Continued)

DOSIMETRY DATA FOR REACTION Fe-54 (n, p) Mn-54

Capsule Section No.	dps Mn-54 gm Iron	ϕ_{45MW} n/cm ² (E _n > 0.5 MeV)	ϕ_t , n/cm ² (E _n > 0.1 MeV)	ϕ_t , n/cm ² (E _n > 0.5 MeV)	ϕ_t , n/cm ² (E _n > 1.0 MeV)	ϕ_t , n/cm ² (E _n > 3.0 MeV)	Approximate Distance from Bottom of Core (inches)
<u>Series L-4'</u> (Continued)							
L4B-4	2.21×10^9	7.6×10^{14}	9.8×10^{21}	5.3×10^{21}	3.0×10^{21}	7.0×10^{20}	6-6/8
L4B-5	1.84×10^9	6.3×10^{14}	8.1×10^{21}	4.4×10^{21}	2.5×10^{21}	5.8×10^{20}	2-1/8
L4E-1	1.39×10^8	4.8×10^{13}	6.2×10^{20}	3.4×10^{20}	1.9×10^{20}	4.4×10^{19}	20-4/8
L4E-3	1.95×10^9	6.7×10^{14}	8.6×10^{21}	4.7×10^{21}	2.6×10^{21}	6.2×10^{20}	11-3/8
L4E-4	2.38×10^9	8.2×10^{14}	1.0×10^{22}	5.7×10^{21}	3.2×10^{21}	7.5×10^{20}	6-6/8
L4F-1	1.70×10^8	5.9×10^{13}	7.6×10^{20}	4.1×10^{20}	2.3×10^{20}	5.4×10^{19}	20-4/8
L4F-3	1.68×10^9	5.8×10^{14}	7.4×10^{21}	4.0×10^{21}	2.3×10^{21}	5.3×10^{20}	11-3/8
L4F-5	1.70×10^9	5.8×10^{14}	7.5×10^{21}	4.1×10^{21}	2.3×10^{21}	5.4×10^{20}	2-1/8
L4G-2	5.47×10^8	1.9×10^{14}	2.4×10^{21}	1.3×10^{21}	7.4×10^{20}	1.7×10^{20}	16
L4G-3	1.88×10^9	6.5×10^{14}	8.3×10^{21}	4.5×10^{21}	2.6×10^{21}	5.9×10^{20}	11-3/8
L4G-4	2.40×10^9	8.3×10^{14}	1.0×10^{22}	5.8×10^{21}	3.3×10^{21}	7.6×10^{20}	6-6/8
<u>Series MT-5 and 6</u> (Counted at 10/20/67)							
	1.78×10^9	7.1×10^{14}	9.3×10^{21}	5.0×10^{21}	2.8×10^{21}	6.6×10^{20}	3-3/4
	1.98×10^9	7.9×10^{14}	1.0×10^{22}	5.9×10^{21}	3.2×10^{21}	7.3×10^{20}	7-1/4
	1.64×10^9	6.6×10^{14}	8.5×10^{21}	4.6×10^{21}	2.6×10^{21}	6.1×10^{20}	10-3/4
	1.18×10^8	4.7×10^{13}	6.1×10^{20}	3.3×10^{20}	1.9×10^{20}	4.4×10^{19}	20-1/2
	3.40×10^7	1.4×10^{13}	1.8×10^{20}	9.6×10^{19}	5.4×10^{19}	1.3×10^{19}	24
	4.98×10^6	2.0×10^{12}	2.6×10^{19}	1.4×10^{19}	7.9×10^{18}	1.8×10^{18}	31
	2.12×10^6	8.5×10^{11}	1.1×10^{19}	6.0×10^{18}	3.4×10^{18}	7.9×10^{17}	34-1/2
	$7.44 \times 10^{8**}$	2.6×10^{14}	3.3×10^{21}	1.8×10^{21}	1.0×10^{21}	2.4×10^{20}	14-1/4
	$1.38 \times 10^{8**}$	4.8×10^{13}	6.2×10^{20}	3.3×10^{20}	1.9×10^{20}	4.4×10^{19}	20-1/2
	$2.5 \times 10^{6**}$	8.8×10^{11}	1.1×10^{19}	6.1×10^{18}	3.4×10^{18}	8.0×10^{17}	34-1/2

** Counted at 8/16/67.

TABLE 5-2

DOSIMETRY DATA FOR REACTION Ni-58 (n,p) Co-58

Capsule Section No.	dps Co-58 gm Nickel	ϕ_{45MW} n/cm ² (E _n >0.5MeV)	ϕ_t , n/cm ² (E _n >0.1MeV)	ϕ_t , n/cm ² (E _n >0.5MeV)	ϕ_t , n/cm ² (E _n >1.0MeV)	ϕ_t , n/cm ² (E _n >3.0MeV)	Approximate Distance from Bottom of Core (inches)
<u>Series L-2'</u> (Counted at 6/2/67)							
L2K-1	7.72×10^9	1.4×10^{14}	4.7×10^{21}	2.5×10^{21}	1.4×10^{21}	3.1×10^{20}	17-3/8
L2K-3	3.77×10^{10}	6.9×10^{14}	2.3×10^{22}	1.2×10^{22}	6.6×10^{21}	1.5×10^{21}	3-1/8
L2M-2	3.57×10^{10}	6.5×10^{14}	2.2×10^{22}	1.2×10^{22}	6.3×10^{21}	1.4×10^{21}	10-3/8
L2O-1	7.58×10^9	1.4×10^{14}	4.6×10^{21}	2.4×10^{21}	1.3×10^{21}	3.0×10^{20}	17-3/8
L2O-3	3.48×10^{10}	6.4×10^{14}	2.1×10^{22}	1.1×10^{22}	6.1×10^{21}	1.4×10^{21}	3-1/8
L2P-2	3.29×10^{10}	6.0×10^{14}	2.0×10^{22}	1.1×10^{22}	5.8×10^{21}	1.3×10^{21}	10-3/8
L2P-3	3.27×10^{10}	6.0×10^{14}	2.0×10^{22}	1.0×10^{22}	5.8×10^{21}	1.3×10^{21}	3-1/8
L2Q-1	7.12×10^9	1.3×10^{14}	4.3×10^{21}	2.3×10^{21}	1.3×10^{21}	2.9×10^{20}	17-3/8
L2Q-3	3.12×10^{10}	5.7×10^{14}	1.9×10^{22}	1.0×10^{22}	5.5×10^{21}	1.3×10^{21}	3-1/8
<u>Series L-4</u> (Counted at 6/2/67)							
L4C-1	1.38×10^9	4.6×10^{13}	9.3×10^{20}	4.9×10^{20}	2.7×10^{20}	6.2×10^{19}	20-1/2
L4C-2	4.92×10^9	1.7×10^{14}	3.4×10^{21}	1.8×10^{21}	9.7×10^{20}	2.2×10^{20}	16
L4C-3	1.57×10^{10}	5.3×10^{14}	1.1×10^{22}	5.7×10^{21}	3.1×10^{21}	7.1×10^{20}	11-3/8
L4C-4	2.05×10^{10}	6.9×10^{14}	1.4×10^{22}	7.4×10^{21}	4.0×10^{21}	9.2×10^{20}	6-3/4
L4C-5	1.62×10^{10}	5.5×10^{14}	1.1×10^{22}	5.8×10^{21}	3.2×10^{21}	7.3×10^{20}	2-1/8
L4D-4	1.94×10^{10}	6.6×10^{14}	1.3×10^{22}	7.0×10^{21}	3.8×10^{21}	8.8×10^{20}	6-3/4
<u>Series L-4'</u> (Counted at 8/12/67)							
L4A-1	1.51×10^9	6.7×10^{13}	8.4×10^{20}	4.6×10^{20}	2.6×10^{20}	5.8×10^{19}	20-1/2
L4A-2	4.69×10^9	2.1×10^{14}	2.6×10^{21}	1.4×10^{21}	8.0×10^{20}	1.8×10^{20}	16
L4A-3	1.37×10^{10}	6.0×10^{14}	7.6×10^{21}	4.2×10^{21}	2.3×10^{21}	5.3×10^{20}	11-3/8
L4A-4	1.68×10^{10}	7.4×10^{14}	9.4×10^{21}	5.2×10^{21}	2.9×10^{21}	6.5×10^{20}	6-3/4
L4A-5	1.33×10^{10}	5.9×10^{14}	7.4×10^{21}	4.1×10^{21}	2.3×10^{21}	5.1×10^{20}	2-1/8
L4B-2	4.51×10^9	2.0×10^{14}	2.5×10^{21}	1.4×10^{21}	7.7×10^{20}	1.7×10^{20}	16
L4B-4	1.70×10^{10}	7.5×10^{14}	9.5×10^{21}	5.2×10^{21}	2.9×10^{21}	6.6×10^{20}	6-3/4

TABLE 5-2 (Continued)

DOSIMETRY DATA FOR REACTION Ni-58 (n, p) Co-58

Capsule Section No.	dps Co-58 gm Nickel	$\phi_{45\text{MW}}$ n/cm ² (E _n > 0.5 MeV)	ϕ_t , n/cm ² (E _n > 0.1 MeV)	ϕ_t , n/cm ² (E _n > 0.5 MeV)	ϕ_t , n/cm ² (E _n > 1.0 MeV)	ϕ_t , n/cm ² (E _n > 3.0 MeV)	Approximate Distance from Bottom of Core (inches)
Series L-4' (Continued)							
L4B-5	1.28×10^{10}	5.6×10^{14}	7.1×10^{21}	3.9×10^{21}	2.2×10^{21}	4.9×10^{20}	2-1/8
L4E-1	1.30×10^9	5.7×10^{13}	7.3×10^{20}	4.0×10^{20}	2.2×10^{20}	5.0×10^{19}	20-1/2
L4E-3	1.48×10^{10}	6.5×10^{14}	8.3×10^{21}	4.6×10^{21}	2.5×10^{21}	5.7×10^{20}	11-3/8
L4E-4	1.86×10^{10}	8.2×10^{14}	1.0×10^{22}	5.7×10^{21}	3.2×10^{21}	7.2×10^{20}	6-3/4
L4F-1	1.36×10^9	6.0×10^{13}	7.6×10^{20}	4.2×10^{20}	2.3×10^{20}	5.3×10^{19}	20-1/2
L4F-3	1.47×10^{10}	6.5×10^{14}	8.2×10^{21}	4.5×10^{21}	2.5×10^{21}	5.7×10^{20}	11-3/8
L4F-5	1.42×10^{10}	6.3×10^{14}	7.9×10^{21}	4.4×10^{21}	2.4×10^{21}	5.5×10^{20}	2-1/8
L4G-2	4.60×10^9	2.0×10^{14}	2.6×10^{21}	1.4×10^{21}	7.8×10^{20}	1.8×10^{20}	16
L4G-3	1.42×10^{10}	6.3×10^{14}	7.9×10^{21}	4.4×10^{21}	2.4×10^{21}	5.5×10^{20}	11-3/8
L4G-4	1.72×10^{10}	7.6×10^{14}	9.6×10^{21}	5.3×10^{21}	2.9×10^{21}	6.6×10^{20}	6-3/4
Series MT-5 and 6 (Counted at 12/31/66)							
	1.38×10^{11}	6.7×10^{14}	8.5×10^{21}	4.7×10^{21}	2.6×10^{21}	5.9×10^{20}	3-3/4
	1.51×10^{11}	7.3×10^{14}	9.3×10^{21}	5.1×10^{21}	2.8×10^{21}	6.5×10^{20}	7-1/4
	1.28×10^{11}	6.2×10^{14}	7.9×10^{21}	4.4×10^{21}	2.4×10^{21}	5.5×10^{20}	10-3/4
	5.86×10^{10}	2.8×10^{14}	3.6×10^{21}	2.0×10^{21}	1.1×10^{21}	2.5×10^{20}	14-1/4
	9.31×10^9	4.5×10^{13}	5.8×10^{20}	3.2×10^{20}	1.8×10^{20}	4.0×10^{19}	20-1/2
	3.02×10^9	1.5×10^{13}	1.9×10^{20}	1.0×10^{20}	5.7×10^{19}	1.3×10^{19}	24
	4.15×10^8	2.0×10^{12}	2.6×10^{19}	1.4×10^{19}	7.8×10^{18}	1.8×10^{18}	31
	1.61×10^8	7.8×10^{11}	1.0×10^{19}	5.5×10^{18}	3.0×10^{18}	6.9×10^{17}	34-1/2
	$1.63 \times 10^{10*}$	7.5×10^{14}	9.5×10^{21}	5.2×10^{21}	2.9×10^{21}	6.5×10^{20}	3-3/4
	$1.38 \times 10^{10*}$	6.33×10^{14}	8.0×10^{21}	4.4×10^{21}	2.4×10^{21}	5.5×10^{20}	10-3/4
	$6.15 \times 10^9*$	2.8×10^{14}	3.6×10^{21}	2.0×10^{21}	1.1×10^{21}	2.5×10^{20}	14-1/4
	$3.67 \times 10^8*$	1.7×10^{13}	2.1×10^{20}	1.2×10^{20}	6.5×10^{19}	1.5×10^{19}	24
	$1.67 \times 10^7*$	7.7×10^{11}	9.7×10^{18}	5.3×10^{18}	2.9×10^{18}	6.7×10^{17}	34-1/2

* Counted at 8/16/67.

FIGURE 5-1. EBR-II NEUTRON FLUENCE DISTRIBUTION FOR CAPSULE SERIES L-2', L-4, L-4', AND MT 5 AND 6

TABLE 5-3

NEUTRON FLUENCE COMPARISONS BETWEEN
GENERAL ELECTRIC AND ARGONNE NATIONAL LABORATORY

	<u>General Electric</u>	<u>Argonne National Laboratory</u>
1. TOTAL FLUENCE, n/cm^2		
Series L-2'	2.5×10^{22}	3.4×10^{22}
Series L-4	1.4×10^{22}	2.0×10^{22}
Series L-4'	1.1×10^{22}	1.7×10^{22}
2. FAST FLUENCE ($E_n > 1$ MeV), n/cm^2		
Series L-2'	6.7×10^{21}	9.5×10^{21}
Series L-4	3.9×10^{21}	5.6×10^{21}
Series L-4'	3.0×10^{21}	4.8×10^{21}
3. CONVERSION FACTORS, @ 45 MW		
Total Fluence, (n/cm^2)/MWd		
Row 2	2.7×10^{18}	4.7×10^{18}
Row 4	2.7×10^{18}	3.7×10^{18}
Fast Fluence, ($E_n > 1$ MeV) (n/cm^2)/MWd		
Row 2	8.2×10^{17}	1.3×10^{18}
Row 4	7.2×10^{17}	1.0×10^{18}

SECTION VI

DISCUSSION

6.1 ANALYTICAL PROCEDURES

The disagreement between total fluences, in EBR-II, as determined by ANL and by extrapolation of flux-wire analyses may be associated with errors in computed flux spectra combined with differences in normalization. As indicated in Section 4.2, the flux-wire analyses are probably quite inaccurate for determining total fluences because they are normalized to a measurement above 1 MeV and depend upon an imprecise calculation of the ratio of the flux below 3.7 MeV to that above. The determination of total fluence from a neutronics computation which is based upon the actual reactor power, as the ANL value apparently is, may be more accurate. It should be pointed out that this is a more normal application of neutronics computations. In most former applications of neutronics techniques, the energy range above 3.7 MeV has been of minor importance to the desired results. Hence, high accuracy for the flux above 3.7 MeV has not been a previous goal.

Conversely, the flux-wire analysis should provide a more accurate local fluence above several MeV than a power normalized, neutronics computation of the normal type. This is not to say that a much more accurate neutronics computation could not be performed. A spatial flux solution in which many groups are used between 1 MeV and 15 MeV (perhaps even to the extent of the ultrafine group codes ELMOE or MC²) could significantly reduce the disagreement in the two normalizations.

From all indications, the flux spectrum computed by ANL in 22 groups agrees well with the General Electric computed spectrum. For example, the computed integrated fluxes above 0.82 MeV are 0.30 and 0.33. Also, values for nvt/MWd were computed which are

slightly larger than the ANL values when normalized to total power. Since the General Electric computed spectrum tends to undercompute the total fluence when normalized to the wire measurements, it can be concluded that the fluence above 1 or 2 MeV would be greater as computed by the ANL method (i.e., power normalization) than by the flux-wire method. The fairly good agreement ANL obtains for the U^{238}/U^{235} fission ratio between calculation and experiment would indicate that the higher value is the more correct. This does not, however, take into account spatial variations in the spectrum, and the calculated fission ratio, itself, is quite uncertain due to lack of definitive cross-section data for the U^{238} fission threshold. This agreement should be given only a small weight in determining if the computed high energy flux spectrum is correct.

6.2 NEUTRON ENERGY LEVEL—
RADIATION DAMAGE CORRELATION

Attempts have been made to correlate radiation damage, as measured by mechanical properties, with total neutron fluence. This has principally been due to the lack of measurements of energy differential fluences and to the complexities of the primary radiation damage mechanisms.

Since most neutron damage mechanisms are attributed to neutrons of particular energies, such a correlation cannot be expected to be found for tests performed in reactors of vastly different neutron flux spectra (i.e., fast and thermal reactors). However, if a total fluence correlation can be shown for tests performed in similar, but not identical, spectra, i.e., the spectra in rows 2 and 7 of EBR-II, it would indicate that there may also be a good correlation between radiation damage and fuel burnup

for such spectra, and extrapolation to LMFBR burnup values should be fairly accurate.

Total fluence is currently best determined by neutron flux computations, normalized to the reactor power. For those samples yet to be irradiated, the computations may be checked or adjusted by use of foil irradiations at the placement of the sample. Improved correlation parameters (rather than total fluence) can be computed from the analytical spectrum using displacement production and energy deposition cross sections.*⁽⁵⁾ A threshold activation material might also be found which roughly parallels the displacement production cross-section curve. The measurement of such an activation would provide an improved correlation parameter for lattice displacements and energy deposition.

If such a correlation between radiation damage and total fluence cannot be found, attempts should be made to find a good correlation with partial fluences (energy-wise) or reaction rates such that the test results can be analytically extrapolated to FCR flux conditions. For this purpose, fluence energy ranges which can be expected to correlate with a few particular damage mechanisms are defined in the following paragraphs. Methods for obtaining the best possible "estimates" of these partial fluences are also suggested. With such definitions of fluence energy ranges and "estimations" of the magnitudes of these partial fluences for a wide range of test spectra, a comprehensive correlation study might shed some light on which damage mechanisms are the most important for a given sample environment.

6.2.1 Trace Elements by Transmutation

Production of trace materials by transmutation due to neutron capture may occur

* The lattice displacement production and energy deposition cross sections given in reference 5 are not necessarily the best values to use. An up-to-date literature search for values has not been performed.

over any neutron energy range, depending upon the particular transmutation of interest. For example, the reaction B-10 (n, α) Li-7 has a cross section which increases with decreasing energy, while the Ni-58 (n, p) Co-58 reaction has a 1 to 2 MeV cross-section threshold. The best fluence to use for correlation purposes must be determined for each trace element to be considered. In general, these fluences are best determined by calculations of spectra, with flux-wire analysis being the best approach for high energy threshold reactions.

Hydrogen is produced by (n, p) reactions in principally Fe and Ni, the same reactions used in the flux-wire analysis. Thus, fluences above 1 to 2 MeV should be used for correlation purposes.

Helium production in austenitic stainless steel is thought to be primarily caused by the (n, α) reaction in Fe, Cr and Ni, which have thresholds at about 6 MeV and N with a 1 MeV threshold. Since the flux spectrum above a few MeV should not vary considerably from irradiation to irradiation, fluences above 1 to 3 MeV should be used for correlation studies. Stainless steel may contain varying amounts of boron, which also will contribute to the helium production. The (n, α) cross section of boron extends down to thermal energies and total fluences would be required for correlation purposes.

For most stainless steels, however, the boron content burns out after relatively short irradiation periods, and the helium production from Fe, Cr, Ni, and N tends to over-shadow it. Threshold (n, α) cross sections are difficult to obtain experimentally and direct computations of the helium production rate are of low accuracy.

Partial fluences for both helium and hydrogen production can best be measured by the flux-wire analyses. For cases where measurements are not available and for extrapolation purposes, a power-normalized analytical flux solution using many groups, as described in Section 5.1, can be used. Very crude

estimates of these fluences can be made by assuming that they are proportional to the core power density (per unit volume of core) at the point of irradiation. **This type of approximation should only be used for comparing irradiations performed in highly multiplying regions (reactor cores) of approximately the same composition.**

6.2.2 Lattice Displacements and Energy Deposition

Radiation damage due to lattice displacements begins to occur at neutron energies around

10 keV.⁽⁵⁾ The curve of the displacement production cross section increases to a peak at about 2 MeV. Because of the energy variation of this cross section the total fluence should be used in correlation studies.

Direct energy deposition due to neutrons is closely related to the displacement rate. In addition, the formation or migration of voids may be associated with energy deposition. As with lattice displacements, the energy deposition probability increases with neutron energy, and the total fluence should be considered.

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