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CORRELATION OF IRRADIATION DATA USING ACTIVATION FLUENCES AND IRRADIATION TEMPERATURE

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16. Abstract

A new method of direct correlation of radiation damage with time-integrated activation of neutron dosimeters is described. The method is tested by using a consistently defined set of nil-ductility transition temperature data for A302-B ferritic steel and is compared to a correlation representative of damage function techniques. For the data studied, the new method is as accurate as the damage function method and its advantages over this method are pointed out. Also, the effect of irradiation temperature on damage is treated.

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SUMMARY

Nil-ductility transition temperature data for A302-B ferritic steel were analyzed by using multiple regression. Four independent variables were used. These were specimen temperature during irradiation and time-integrated specific activations (activation fluences) of fast-, intermediate-, and thermal-neutron detectors. The results of these analyses were predictive equations having the increase in transition temperature as the dependent variable. For a reference point, a similar analysis was also performed using the flux integral above 1 MeV and the irradiation temperature as independent variables. Also, the effects of excluding irradiation temperature from these analyses was studied.

The results of these analyses showed that excellent correlation was achieved by using activation fluences and irradiation temperature as the independent variables; that is, 93 percent of the data variability was explained as contrasted with 36 percent when using the flux above 1 MeV. Irradiation temperature was found to account for from 29 to 46 percent of the data variability depending on the model used.

The activation fluence model was also compared with a selected damage function model. This comparison indicated that the activation fluence method was about as accurate (for the data studied) as the damage function method, and had several additional advantages over this method.

INTRODUCTION

Radiation damage and the development of correlations for predicting damage have been the subject of many investigations (refs. 1 to 6 are typical). The correlations generally are attempts at relating a change in the physical property of a material to time-integrated neutron flux above some selected energy, usually 1 MeV but frequently

as low as 0.01 MeV. Those correlations which assume a threshold energy for damage are generally recognized to be inadequate for many situations because of the assumptions made in deriving and using them. The energy distribution of neutrons is usually assumed to be that of a fission spectrum; only neutrons above 1 MeV are assumed to contribute (and those equally) to the damage production; and the observed damage is assumed to depend only on the total exposure and not on the exposure rate (ref. 7). The first two assumptions are clearly not generally true, especially for fast reactor applications. The third has some theoretical (ref. 6) and experimental (refs. 8 and 9) verification.

A more fundamental approach is typified by that of McElroy, Dahl, and Serpan (ref. 10), in which the damage is treated as a cross section (damage function) which when multiplied by the neutron flux yields the damage in both integral and differential form. Damage function methods usually involve complex calculations (ref. 10). They require knowledge of the neutron flux in some form (differential or integral above some selected energy) both for correlating data and for using these correlations to predict damage. This introduces error in predicted values because of uncertainties in cross sections and in the spectrum determination and error propagation in the damage function generation.

The work discussed in this report correlates damage with detector activations. In the discussion that follows, this method is shown to have several advantages for correlating damage when compared with selected integral flux and damage function methods.

METHOD

The activation fluence method is essentially a regression analysis (ref. 11) of data to obtain a polynomial in several variables, their second powers, and their interactions. The variables account for spectrum shape (fast, intermediate, and thermal) and the temperature at which the damage is produced.

In general, the polynomial is

$$Y = A_0 + A_1X_1 + A_2X_2 + \ldots + A_nX_n + A_{12}X_1X_2 + A_{13}X_1X_3 + \ldots + A_{1n}X_1X_n + A_{23}X_2X_3 + A_{24}X_2X_4 + \ldots + A_{n-1,n}X_{n-1}X_n + A_{11}X_1^2 + A_{22}X_2^2 + \ldots + A_{nn}X_n^2 + \epsilon \quad (1)$$

The dependent variable Y is the predicted damage. The coefficients A are determined by using multiple linear regression and are best least-squares estimates assuming only

that the experimental error ϵ is normally distributed with a finite variance and a mean of zero. The X values are functions of the activation fluences for selected neutron detectors (X_1, X_2, . . ., X_{n-1}) and temperature X_n.

Using multiple regression as a means to implement correlation gives an unambiguous and immediately useful model along with quantitative indications of the accuracy that can be expected when using this model. Also, statistical interpretations are possible which may aid in determining which variables are important so that more basic investigations can proceed on an efficient experimental basis.

The activation fluence for any detector is defined by

$$\alpha \equiv \int_0^T \int_0^\infty \sigma(E)\varphi(E,t)dE\ dt \tag{2}$$

where T is the total irradiation time and the other symbols have their usual meanings. In practice, a set of detectors is selected that will cover the complete range of neutron energy. When feasible, these detectors are irradiated with the irradiation specimen; and their activations are used to obtain the activation fluences. Detectors having long half-lives are preferable because they reflect the irradiation history more accurately.

TESTING THE METHOD

In order to test this correlation approach using actual data, a literature search was made for a consistent set of data. A data set was selected (refs. 12 and 13) that was consistently defined and that had accompanying calculated neutron spectra. These data represented a wide range of spectra and were taken in several different test reactors. The data and tabulated spectra along with several helpful comments were provided by C. Z. Serpan of the Naval Research Laboratory, Washington, D.C.

The selected parameter of interest was the change in nil-ductility transition temperature ΔNDT for A302-B ferritic steel. Of the data examined, this parameter had by far the most data points.

Detectors were selected in an attempt to cover the complete range of neutron energies with minimal response overlap. Consideration was also given to availability, ease of counting, performance at high neutron flux levels, and expected accuracy of various detector materials.

The detector[1] set first selected was $Co^{59}(n,\gamma)Co^{60}$, responding from ~0 to 6 eV; $Cu^{63}(n,\gamma)Cu^{64}$, responding from ~0.4 eV to 0.2 MeV when cadmium-covered;

[1]In a fission spectrum 90 percent of the activation that would occur in these detectors is contained within the energy limits shown.

$Np^{237}(n, f)N^{fp}$, responding from ~0.5 MeV to 5 MeV; and $Ti^{46}(n, p)Sc^{46}$, responding from ~3.5 MeV to 9 MeV.

Although copper is not an ideal detector because of its short half-life, it was felt to be adequate for purposes of demonstrating the method. Activation fluences were calculated for each detector in each flux spectrum. Whenever possible, for actual correlations using this method, the activation fluences would not be generated in this manner but would instead be determined from detector activities and would include the effects of irradiation history and half-life. It was necessary to calculate the activation fluences for this test of method because this particular set of detectors was not used with the existing data. It was felt that calculated activations would be adequate to test and demonstrate the method. Measured activations would, however, be better because they do not contain uncertainties in spectra and cross sections.

In anticipation of an irradiation temperature effect, temperature was carried as a separate entity. Thus, the variables used were the activation fluences and the temperature at which the specimens were irradiated.

Four calculations were made to obtain data correlation by four different models:

(1) A calculation (ref. 11) was performed using neutron fluence above 1 MeV (assuming a Watt fission spectrum) as the only independent variable. Although it is generally recognized that the flux above 1 MeV is not a reliable damage predictor, this model was selected as representative of the integral flux models. It was felt that the relative accuracy of this model compared to other models would be of interest. This calculation is referred to as calculation 1.

(2) Calculation 1 was rerun with temperature added as the second independent variable. The purpose of this calculation was to determine the effects of temperature when used with an arbitrarily selected model. This is called calculation 2.

(3) A third calculation was run which used activation fluences and temperature as independent variables. This calculation is designated calculation 3.

(4) The fourth calculation which was performed was the same as calculation 3 except that temperature was dropped as an independent variable. This calculation is called calculation 4 and was performed to determine whether the temperature effect was independent of the selected model, that is, whether the effect would be about the same as it is for calculations 1 and 2.

The damage function method of reference 10 was also used to infer damage values. This made it possible to compare results calculated by the activation fluence method with those results obtained by a different contemporary technique.

The data and values of the independent variables used in these calculations are shown in tables I and II. The flux correlations (calculations 1 and 2) did not incorporate any nonlinear transformations in the fitting procedure as did the activation fluence correlation (calculation 3). The nonlinear transformation (raising variables to the 1/4 power) was a numerical expedient that enabled the matrix inversion to be completed.

The transformation which makes the X_i values lie between 0 and 1.0 is linear and has no effect on R^2 and S. When coupled with the 1/4 power transformation, it improves the matrix conditioning because the transformed values become more homogeneously distributed between 0 and 1.0. The 1/4 power transformation does affect the R^2 and S values somewhat. To examine this effect, calculations 1 and 2 were rerun with the 1/4 power transformation. The effect on R^2 and S was small; thus, this transformation did not account for the better agreement of calculation 3, and the conclusions remain valid. All results of calculations 1 and 2 shown in this report pertain to those calculations in which no 1/4 power transformation was used.

RESULTS

In calculations 3 and 4, a high intercorrelation was found between the neptunium and titanium activities due to the spectral shapes being similar above 1 MeV for about 70 percent of the data. This resulted in difficulty in inverting the least-squares matrix. Since this problem was inherent in the data and because several attempts at transforming the basic independent variables to avoid excessive covariance proved unsatisfactory, the titanium activation was dropped and the neptunium was relied on to sense the complete fast energy range.

The least-squares analysis provides for sequential remodeling by dropping less significant terms from the equation and refitting until all terms retained are significant at some specified level of rejection (ref. 11). The rejection level selected for this study was 90 percent.

The accuracy parameters associated with calculations 1 to 4 are given in table III. The parameters include

(1) R^2, the square of the multiple correlation coefficient, which is that fraction of the total variance in the measured data that is accounted for by the regression equation

(2) S, the standard error of estimate, that is, the standard deviation of fit at the mean of the independent variables

(3) F, the ratio of regression mean sum of squares to residual mean sum of squares

Comparing the calculated value of F ($F_{calculated}$) to the value of F from the statistical tables ($F_{tabulated}$) gives an indication as to whether the regression equation could have the overall accuracy indicated by R^2 by chance alone. A value of $F_{calculated}$ well above the tabulated F value indicates only a 5 percent chance that the fit occurred accidentally. A high ratio of $F_{calculated}$ to $F_{tabulated}$ is prerequisite to trusting the R^2, S, and polynomial coefficients.

(4) F', the ratio of lack-of-fit mean sum of squares to replication mean sum of squares

When $F'_{calculated}$ is higher than $F'_{tabulated}$, the error is probably caused by lack of fit and not experimental error. When the reverse is true, the error is probably caused by experimental error rather than by the model selection. For a more detailed discussion of these statistical parameters, see reference 11.

The fitting coefficients are given for calculation 3 in table IV. Table V is a comparison of measured and predicted points using the first three regression models and the selected damage function technique. The following section gives a detailed comparison of the damage function and activation fluence methods. Table VI shows the neutron flux distributions that were used to generate activation fluences; and table VII is a summary of cross sections obtained for the group structure used by plotting activation cross sections (ref. 14) and numerically averaging these over the indicated groups.

The following are two interpretations that may be made from the data of tables III and V:

(1) These calculations confirm a strong dependence of radiation damage in A302-B on the irradiation test temperatures (from 327 to 561 K). This is evident from the large increase in the multiple correlation coefficient when the temperature is added to calculation 1 (see calculation 2) or calculation 4 (see calculation 3).

(2) The activation fluence model (for this particular set of data which was assumed to be typical and adequate to evaluate relative accuracy) correlates the data much better than the integral flux (>1 MeV) method. The activation fluence method explains 93 percent (see R^2 comparison) of the data variability and has the lowest standard error at the fitted points. The lower R^2 for the integral flux (>1 MeV) method was expected.

The negative coefficient for thermal activation fluence (see table IV) suggests a damage annealing effect due to thermal neutrons. The other coefficients also could possibly be interpreted by using physical arguments. However, no attempt has been made to explain the polynomials from a mechanistic viewpoint since this study was intended more to illustrate a technique rather than to produce a usable correlation. The actual values of the independent variables might have been different from the values constructed using calculated fluxes and cross sections. Also, the data were not taken using an orthogonal experiment design; thus, the coefficients may be improved with judicious experiment planning. It is reasonable to assume, however, that the equations generated from this study will have similar accuracy if the independent variables are calculated as was done here for other ΔNDT (in A302-B) values of interest.

COMPARISON OF ACTIVATION FLUENCE METHOD TO DAMAGE FUNCTION METHOD

It is interesting to note some of the similarities and differences between the activation fluence and damage function methods of reference 10. The activation fluence method

is similar to the damage function approach of reference 10 in that both are fits of data to obtain weighting coefficients. For applications in which activations can be obtained experimentally, the activation fluence method has the advantage of not requiring any of the complex computer codes which must be used to obtain spectrum information for the damage function method. This tends to reduce the propagation and magnification of error. The regression analysis with the activation fluence method can be used with the minimum number of detectors required for accurate data correlation, whereas the damage function model requires detailed spectrum information in many energy groups. The regression analysis fit permits treatment of irradiation temperature as an explicit independent variable, while the damage function approach of reference 10 assumes that the effects of temperature are constant over a temperature range of several hundred degrees. A fundamental difference between the regression analysis method and the damage function method of reference 10 is that the damage function method requires an additional separate correlation between damage and total neutron fluence in order to predict the damage that will occur due to an arbitrary spectrum and fluence. Regression analysis using a single correlation predicts damage within the limits of the data used to derive the coefficients. Finally, the activation fluence method allows explicitly for interaction effects between detectors and between detectors and temperature. This effect is implicit in the damage function method.

The activation fluence polynomial can be solved for that total fluence required to produce a specified change in nil-ductility temperature by replacing the time-integrated activations by the product of total fluence and fluence-weighted activation cross sections, provided that these cross sections are available or can be generated. The damage function method yields the required total fluence directly.

The activation fluence method correlates these data as well as the damage function method (11.5 K standard error for the activation fluence method compared to 12.6 K standard error for the damage function method). Table V shows a detailed comparison of measured and calculated values.

CONCLUSIONS AND RECOMMENDATIONS

Based on the results of this study, it is concluded that the activation fluence method may be an accurate and relatively simple method for correlating radiation damage. Its disadvantage is that for data having high covariance, the aliasing of effects will prevent physical interpretation of individual coefficients. However, this does not hamper its predictive capability, which is the goal of a regression analysis of unplanned data.

It is also concluded that regression analysis provides an easy way to accommodate temperature as an additional variable. Finally, it is noteworthy that a three-energy-group spectrum description appears adequate for correlating A302-B nil-ductility data.

This work in applying the activation fluence method points out the need for standardization of the method of determining the spectrum variable. While the number and choice of detectors will depend on the property and temperature range for the material being studied, a complete description of the dosimetry method will permit other investigators to use the data generated in exploring other methods of data correlation. The methods used at the Plum Brook Reactor Facility are described in part V of "Standard Guides to the Design of Experiments for the Plum Brook Reactor Facility."

With the activation fluence method, when detectors can be placed in-pile with the specimens, they may be left in-pile for the duration of the irradiation if long half-life detectors are selected exclusively. If some short half-life detectors are used, at least one long half-life detector should be used so that the detectors can be normalized to comply with the definition of activation fluence given in equation (2).

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TABLE I. - SUMMARY OF DATA ANALYZED

Replicate set	Reactor	Position (a)	Change in nil-ductility temperature, ΔNDT, K	$\int_0^T \int_{1\ \text{Mev}}^{10\ \text{MeV}} \varphi(E,t)dE\,dt$ (b)	$\int_0^T \int_0^{10\ \text{MeV}} \varphi(E,t)\sigma(E)dE\,dt$ (cobalt)	$\int_0^T \int_0^{10\ \text{MeV}} \varphi(E,t)\sigma(E)dE\,dt$ (copper)	$\int_0^T \int_0^{10\ \text{MeV}} \varphi(E,t)\sigma(E)dE\,dt$ (neptunium)	Irradiation temperature, K
1	LITR	C-18	77.8	0.50×10^{19}	21.4×10^{19}	0.207×10^{19}	0.950×10^{19}	505
1	↓	C-18	77.8	.50	21.4	.207	.950	505
2	↓	C-55	86.1	3.1	164.5	1.543	6.694	561
2	↓	C-55	94.4	3.1	164.5	1.543	6.694	561
3	↓	C-18	94.4	.50	21.4	.207	.950	400
4	↓	↓	72.2	.50	21.4	.207	.950	477
5	↓	↓	111.1	.60	25.7	.249	1.140	366
6	↓	↓	141.7	1.8	77.0	.747	3.420	327
7	↓	C-28	122.2	1.24	72.0	.356	2.279	↓
8	↓	C-53	119.4	1.2	74.2	.586	1.422	↓
9	↓	↓	163.9	1.94	119.9	.948	2.299	↓
10	↓	↓	175.0	2.00	123.6	.977	2.370	↓
11	↓	↓	183.3	3.50	216.4	1.710	4.148	↓
12	↓	C-55	127.8	.85	451.2	.423	1.835	↓
13	↓	C-49	113.9	.75	82.2	.663	1.801	↓
14	IRL	11.75 cm (4.625 in.)	58.3	.26	4.1	.048	.339	↓
15	↓	14.29 cm (5.625 in.)	44.4	.16	1.6	.041	.274	↓
16	↓	16.83 cm (6.625 in.)	27.8	.10	1.0	.035	.211	↓
17	↓	19.38 cm (7.625 in.)	27.8	.06	.8	.029	.158	↓
18	↓	21.91 cm (8.625 in.)	19.4	.04	.8	.027	.127	↓
19	BGR	W-44	113.9	.55	4.1	.130	1.572	↓
20	BGR	W-44	133.3	.75	5.5	.177	2.144	↓
21	HWCTR	Gray rod	105.6	.73	1030.0	1.199	1.515	↓
22	LITR	C-55	111.1	1.4	74.3	.698	3.023	527
23	Yankee	Vessel wall	61.1	.22	19.1	.041	.221	544
24	↓	Accelerated	125.0	5.0	1855.6	.965	6.306	547
25	↓	Accelerated	144.4	7.0	2597.5	1.351	8.831	547
26	↓	Accelerated	172.2	9.0	3339.4	1.736	11.350	547
27	LITR	C-55	86.1	1.5	79.6	.747	3.238	↓
28	↓	C-18	86.1	3.3	141.1	1.369	6.268	↓
29	↓	C-18	91.7	3.0	128.3	1.244	5.700	↓
30	↓	C-55	105.6	4.7	249.5	2.341	10.144	↓

[a] See refs. 12 and 13 for a description of the exact position of the specimen relative to the reactor core.

[b] A Watt fission spectrum is assumed for this integral. All other integrals in this table use the calculated (assumed actual) flux distribution.

TABLE II. - MODEL DESCRIPTION

Calculation	Primary model	Transformation
1	[a] $Y = A_0 + A_1Z + A_{11}Z^2$	$Z = \left[10^{-19} \int_0^T \int_{1\ MeV}^{10\ MeV} \varphi(E,t)dE\ dt\right]$
2	$Y = A_0 + A_1G_1 + A_2G_2 + A_{12}G_1G_2 + A_{11}G_1^2 + A_{22}G_2^2$	$G_1 = \left[10^{-19} \int_0^T \int_{1\ MeV}^{10\ Mev} \varphi(E,t)dE\ dt\right]$ G_2 = Irradiation temperature
3	$Y = A_0 + A_1X_1 + A_2X_2 + A_3X_3 + A_4X_4 + A_{12}X_1X_2 + A_{13}X_1X_3 + A_{14}X_1X_4 + A_{23}X_2X_3 + A_{24}X_2X_4 + A_{34}X_3X_4 + A_{44}X_4^2$	$\alpha_1 = \left[10^{-19} \int_0^T \int_0^{10\ MeV} \varphi(E,t)\sigma(E)dE\ dt\right]$ Co-59 $\alpha_2 = \left[10^{-17} \int_0^T \int_0^{10\ MeV} \varphi(E,t)\sigma(E)dE\ dt\right]$ Cu-63 $\alpha_3 = \left[10^{-17} \int_0^T \int_0^{10\ MeV} \varphi(E,t)\sigma(E)dE\ dt\right]$ Np-237
4	$Y = A_0 + A_1X_1 + A_2X_2 + A_3X_3 + A_{12}X_1X_2 + A_{13}X_1X_3 + A_{23}X_2X_3$	α_4 = Irradiation temperature [b] $\left[X_i = \left(\frac{\alpha_i - \alpha_i^{min}}{\alpha_i^{max} - \alpha_i^{min}}\right)^{1/4}\right]$

[a] Y is the change in nil-ductility transition temperature for A302-B ferritic steel, i.e., ΔNDT of table I.

[b] α_i^{max} and α_i^{min} are the largest and smallest activation fluences (for the i^{th} detector) that were generated from the data used in the regression analysis. The linear transformation from α_i to X_i^4 codes the data so that it will be between 0 and 1.0.

TABLE III. - SUMMARY OF ACCURACY PARAMETERS

Statistical parameter (a)	Symbol	Calculation 1[b]		Calculation 2		Calculation 3		Calculation 4	
		Primary model	Reduced model[c]	Primary model	Reduced model	Primary model	Reduced model	Primary model	Reduced model
$\frac{\text{Regression sum of squares}}{\text{Total sum of squares}}$	R^2	0.355	0.302	0.788	0.764	0.931	0.925	0.644	0.604
$(\text{Residual mean square})^{1/2}$	S	35.0	35.8	21.2	21.2	13.8	13.1	28.0	27.9
$\frac{\text{Regression mean square}}{\text{Residual mean square}}$	$F_{calculated}$	7.97	13.6	19.3	46.8	24.6	38.4	7.53	14.1
	[d] $F_{tabulated}$	3.33	4.18	2.59	3.37	2.32	2.51	2.49	2.99
$\frac{\text{Lack-of-fit mean square}}{\text{Replication mean square}}$	$F'_{calculated}$	75.9	79.2	28.0	27.7	12.1	10.7	49.1	48.3
	[d] $F'_{tabulated}$	19.5	19.5	19.5	19.5	19.4	19.5	19.5	19.5

[a] Total sum of squares is equal to regression sum of squares plus residual sum of squares. Residual sum of squares is equal to lack-of-fit sum of squares plus replication sum of squares.

[b] See RESULTS section of text for a description of each calculation.

[c] Reduced model refers to model obtained after all terms not significant at 90 percent level have been deleted by stepwise backward elimination procedure.

[d] Probability value is 95 percent.

TABLE IV. - SUMMARY OF ACTIVATION

FLUENCE MODEL COEFFICIENTS

FOR CALCULATION 3

[ΔNDT (in K, primary model) = $A_0 + A_1X_1 + A_2X_2 + A_3X_3 + A_4X_4 + A_{12}X_1X_2 + A_{13}X_1X_3 + A_{14}X_1X_4 + A_{23}X_2X_3 + A_{24}X_2X_4 + A_{34}X_3X_4 + A_{44}X_4^2$; ΔNDT (in K, reduced model) = $A_0 + A_1X_1 + A_3X_3 + A_4X_4 + A_{14}X_1X_4 + A_{23}X_2X_3 + A_{34}X_3X_4 + A_{44}X_4^2$.]

Coefficient or constant[a]	Primary model	Reduced model
A_0	-8.33	-8.09
A_1	-177.8	-73.3
A_2	105.4	-------
A_3	87.6	129.3
A_4	241.5	265.6
A_{12}	-120.0	-------
A_{13}	241.3	-------
A_{14}	243.9	190.2
A_{23}	88.1	180.0
A_{24}	461.1	-------
A_{34}	-850.0	-343.4
A_{44}	-139.0	-169.3

[a]See table II for definitions of the independent variables X.

TABLE V. - COMPARISON OF PREDICTED AND MEASURED DATA

Measured change in nil-ductility temperature, ΔNDT, K (see table I)	Predicted change in nil-ductility temperature ΔNDT, K; and error, $\frac{\text{Measured - Calculated}}{\text{Measured}} \times 100$							
	Calculation 1		Calculation 2		Calculation 3		Damage function	
	Calculated	Percent error	Calculated	Percent error	Calculated	Percent error	Calculated	Percent error
77.8	87.3	-12.3	71.9	+7.6	71.4	+8.2	88.3	-13.6
77.8	87.3	-12.3	71.9	+7.6	71.4	+8.2	88.3	-13.6
86.1	115.8	-34.5	93.1	-8.1	97.7	-13.5	(a)	
94.4	115.8	-22.6	93.1	+4.1	97.7	-3.5	(a)	
94.4	87.3	+7.5	80.2	+15.1	98.6	-4.4	88.3	+6.5
72.2	87.3	-20.1	74.1	-2.6	78.1	-8.2	88.3	-22.3
111.1	88.4	+20.4	86.8	+21.9	112.2	-1.0	95.6	+14.0
141.7	101.6	+28.3	146.2	-3.2	161.0	-13.6	138.3	+2.4
122.2	95.4	+21.9	120.2	+1.7	127.1	-4.0	122.2	+0.0
119.4	95.0	+20.5	118.3	+.9	118.3	+.9	125.6	-5.1
163.9	103.1	+37.1	152.7	+6.8	145.6	+11.2	144.4	+11.9
175.0	103.8	+40.7	155.4	+11.2	147.4	+15.8	145.6	+16.8
183.3	120.2	+34.4	225.1	-22.8	187.4	-2.2	167.8	+8.5
127.8	91.2	+28.7	102.1	+20.1	124.8	+2.3	113.9	+10.9
113.9	90.1	+20.9	97.4	+14.4	129.5	-13.7	114.4	-.5
58.3	84.7	-45.2	74.7	-28.1	54.8	+6.0	46.1	+21.0
44.4	83.6	-88.1	70.1	-57.6	50.7	-14.1	37.8	+15.0
27.8	82.9	-199.6	67.3	-142.2	43.8	-57.6	29.4	-6.0
27.8	82.5	-197.0	65.4	-135.5	36.4	-31.9	16.7	+40.0
19.4	82.3	-323.2	64.5	-231.7	4.0	+79.4	9.4	+51.4
113.9	87.9	+22.8	88.2	+22.6	109.7	+3.7	106.7	+6.3
133.3	90.1	+32.4	97.4	+26.9	124.1	+6.9	118.9	+10.8
105.6	89.8	+14.9	96.5	+8.6	112.5	-6.5	107.2	-1.6
111.1	97.2	+12.5	83.7	+24.7	83.8	+24.6	(a)	
61.1	84.3	-37.9	65.4	-7.0	72.7	-19.0		
125.0	136.7	-9.3	122.7	+1.9	131.6	-5.3		
144.4	158.6	-9.8	146.7	-1.5	147.3	-2.0		
172.2	180.5	-4.8	170.7	+.9	161.3	+6.3		
86.1	98.3	-14.1	77.4	+10.2	76.3	+11.4		
86.1	118.0	-37.0	95.1	-10.4	92.2	-7.1		
91.7	114.7	-25.2	92.1	-.5	88.9	+3.0		
105.6	133.4	-26.3	108.8	-3.1	116.2	-10.0		
Standard error, K	35.8		21.2		13.1 [b]11.5		12.6	

[a]Cannot be calculated using the damage function because irradiation temperature is greater than 506 K (450° F).

[b]For same sample of 21 points used for damage function.

TABLE VI. - SUMMARY OF RELATIVE NEUTRON FLUX USED

Energy group	Lowest energy	Reactor													
		LITR					IRL					BGR	HWCTR	Yankee	
		Test specimen location													
		C-18	C-28	C-49	C-53	C-55	11.75 cm (4.625 in.)	14.29 cm (5.625 in.)	16.83 cm (6.625 in.)	19.38 cm (7.625 in.)	21.91 cm (8.625 in.)	W-44	Gray rod	Pressure vessel	Accelerated
		Relative neutron flux[a]													
1	7.79 MeV	0.0872	0.094	0.0140	0.0862	0.0707	0.260	0.133	0.0680	0.0335	0.0177	0.00257	0.0841	0.101	5.41
2	6.07 MeV	.242	.266	.0356	.257	.214	.603	.285	.136	.0649	.0338	.0966	.227	.312	16.8
3	4.72 MeV	.541	.595	.0783	.564	.466	1.12	.518	.257	.119	.0554	.267	.383	.435	29.1
4	3.68 MeV	.874	.958	.117	.862	.690	1.24	.607	.282	.138	.0633	.401	.540	.421	33.9
5	2.87 MeV	1.36	1.52	.146	1.20	.883	1.85	.948	.454	.251	.119	.565	.843	.450	41.8
6	2.23 MeV	1.95	2.11	.316	2.04	1.62	3.10	1.79	.964	.505	.274	1.043	1.25	.627	63.8
7	1.74 MeV	2.05	2.52	.484	2.54	2.15	2.93	1.90	1.15	.592	.369	1.444	.990	.535	49.2
8	1.35 MeV	2.36	2.70	.509	2.87	2.39	3.14	2.23	1.41	.845	.502	1.757	1.01	.536	47.8
9	1.05 MeV	2.26	2.49	.462	2.62	2.12	2.71	2.04	1.42	.963	.580	1.834	.934	.464	31.2
10	.821 MeV	2.17	2.35	.427	2.52	2.00	2.66	2.08	1.54	1.02	.670	1.863	.972	.485	34.9
11	.639 MeV	2.24	2.34	.423	2.56	1.99	2.52	2.00	1.46	1.03	.694	1.944	1.15	.547	39.4
12	.498 MeV	1.86	1.94	.365	2.18	1.68	1.89	1.49	1.11	.799	.554	1.902	.968	.508	33.6
13	.388 MeV	1.46	1.52	.395	1.79	1.44	1.24	.934	.686	.487	.336	1.781	.678	.346	23.0
14	.302 MeV	1.59	1.59	.391	1.88	1.48	1.65	1.32	1.01	.729	.497	1.766	1.10	.433	28.4
15	.235 MeV	1.38	1.39	.338	1.62	1.28	1.30	1.04	.788	.577	.402	1.686	1.26	.387	24.9
16	.183 MeV	1.19	1.18	.304	1.41	1.11	1.26	1.07	.836	.641	.445	1.638	1.30	.351	21.8
17	.414 eV	35.9	28.3	11.8	44.3	34.6	31.0	21.2	14.6	10.0	6.94	9.646	105	9.58	715
18	0	12.6	22.4	5.11	20.0	12.3	8.67	1.57	.477	.230	.243	148	367	18.7	5910

[a]The group fluxes are the fluxes integrated over each group. All nonthermal flux distributions are calculated distributions at the specimen locations. Thermal fluxes were measured at these locations.

TABLE VII. - SUMMARY OF SELECTED CROSS SECTIONS

Energy group	Lowest energy	Cross sections, barns		
		$Co^{59}(n,\gamma)Co^{60}$	$^{a}Cu^{63}(n,\gamma)Cu^{64}$	$Np^{237}(n,f)N^{fp}$
1	7.79 MeV	0.00012	0.0034	2.35
2	6.07 MeV	.00014	.0037	1.80
3	4.72 MeV	.00021	.0042	1.42
4	3.68 MeV	.00045	.0048	1.50
5	2.87 MeV	.00095	.0056	1.55
6	2.23 MeV	.0016	.0066	1.62
7	1.74 MeV	.0025	.0075	1.67
8	1.35 MeV	.0031	.0083	1.64
9	1.05 MeV	.0039	.0094	1.58
10	.821 MeV	.0044	.0107	1.33
11	.639 MeV	.0042	.0125	1.00
12	.498 MeV	.0040	.0145	.650
13	.388 MeV	.0053	.0170	.270
14	.302 MeV	.0062	.0210	.110
15	.235 MeV	.0065	.0235	.0400
16	.183 MeV	.0068	.0280	.0275
17	.414 eV	3.61	.153	.0109
18	0	37	0	.019

[a]These cross sections are with the standard 0.0508 cm cadmium cover.