Public Service
Electric and Gas
Company

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April 17, 1990
NLR-N90086

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

ADDITIONAL INFORMATION REGARDING
PSE&G'S RESPONSE TO GENERIC LETTER 88-11
DOCKET NO. 50-354
HOPE CREEK GENERATING STATION

Public Service Electric and Gas Company (PSE&G) hereby provides material requested by Mr. Clyde Shiraki, NRC Licensing Project Manager, regarding our response to Generic Letter 88-11 for the Hope Creek Generating Station.

In response to an NRC consultant - reviewer's question on the period of validity for our reactor vessel pressure-temperature curves, a clarification (Attachment 1) was provided by General Electric Company's (GE) Mr. Thomas A. Caine, who prepared the original study for the HCGS response to the subject Generic Letter. The clarification letter supports our current Technical Specification limits as valid for 32 EFPY...even using Regulatory Guide 1.99, Revision 2 methods. This is in agreement with the NRC reviewer's assessment.

Also, in response to Mr. Shiraki's request, Attachment 2 provides a copy of the March 1989 GE evaluation of a Flux Wire Dosimeter removed from the HCGS reactor vessel at the end of the first fuel cycle.

Should you have any further questions, we will be pleased to discuss them with you.

Sincerely,

Thomas M Crimmins Jr

Attachments

A008 1/1

C Mr. C. Y. Shiraki
USNRC Licensing Project Manager

Mr. T. P. Johnson
USNRC Senior Resident Inspector

Mr. T. T. Martin, Administrator
USNRC Region I

Mr. K. Tosch, Chief
Bureau of Nuclear Engineering
New Jersey Department of Environmental Protection

Reference: NLR-N90086

ATTACHMENT 1

GE CLARIFICATION LETTER FOR GL 88-11 RESPONSE

GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

April 11, 1990

cc: C. J. Papandrea
P. C. Ray, GE NSM

Mr. Jim Perrin
Public Service Electric & Gas

Subject: **CLARIFICATION OF CONCLUSIONS IN GE RESPONSE TO GENERIC LETTER 88-11 FOR HOPE CREEK**

Dear Jim,

As we have discussed by phone, NRC review of your Generic Letter 88-11 submittal has resulted in a question on one of the conclusions on the period of validity of the pressure-temperature (P-T) curves. The submittal stated that the curves, valid to 32 EFPY using 1.99, Revision 1, were only valid for 6.7 EFPY using 1.99, Revision 2. The NRC reviewer concluded that, even with 1.99, Revision 2, the curves are still valid for 32 EFPY. The statement in the submittal is clarified below. I agree with the NRC's assessment; the current Tech Spec curves are valid for 32 EFPY using 1.99, Revision 2.

The current P-T curves are attached. As seen, curves A', B' and C' are the beltline limits including 20°F shift, which corresponds to the 1.99, Revision 1 shift for 32 EFPY. The figure also notes that A, B and C are more limiting and are therefore the applicable P-T limits for 32 EFPY. In the 88-11 submittal, the conclusion that the P-T curves are valid for 6.7 EFPY referred to curve A', B' and C'. In other words, the shift of 20°F shown in the figure is only valid for 6.7 EFPY using 1.99, Revision 2. After that, the A', B' and C' curves would have to shift further to the right to be correct.

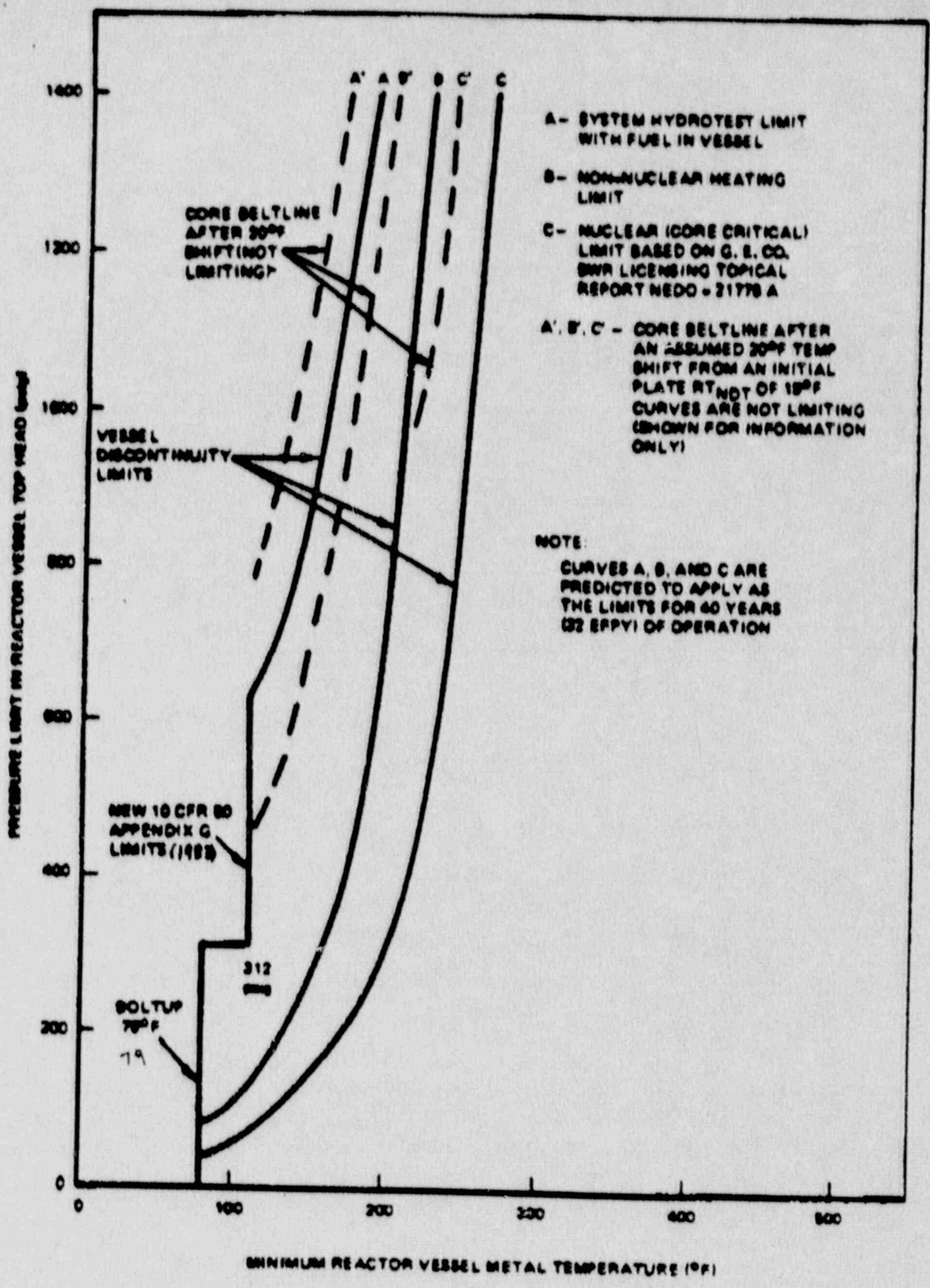
The 88-11 submittal did not address the period of validity of curves A, B and C, which would be more than 6.7 EFPY. As has already been suggested by the NRC review, if the A', B' and C' curves are shifted to cover the limiting conditions using 1.99, Revision 2 (66°F adjusted reference temperature instead of 39°F), curves A, B and C are still limiting. Therefore, the current Tech Spec limits are valid for 32 EFPY using 1.99, Revision 2 methods. Changes required to the P-T curves are only cosmetic, such as moving the A', B' and C' curves as appropriate. Furthermore, the changes do not need to be done before two fuel cycles are completed, because the Hope Creek P-T curves are conservative.

If you have further questions on this subject, please call me at the number below.

Regards,

Tom Caine

T. A. Caine, Senior Engineer
Materials Monitoring & Structural Analysis Services
(408) 925-4047, Mail Code 747



MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

Figure 3.4.6.1-1

Reference: NLR-N90086

ATTACHMENT 2

FLUX WIRE DOSIMETER EVALUATION FOR HCGS

SASR 89-23

DRF A00-02764-1

March 1989

FLUX WIRE DOSIMETER EVALUATION
FOR THE
HOPE CREEK GENERATING STATION

Prepared by: T.A. Caine

T.A. Caine, Senior Engineer
Materials Monitoring &
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Verified by: C.J. Papandrea

C. J. Papandrea, Engineer
Materials Monitoring &
Structural Analysis Services

Reviewed by: S Ranganath

S. Ranganath, Manager
Materials Monitoring &
Structural Analysis Services

GE Nuclear Energy

IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT
PLEASE READ CAREFULLY

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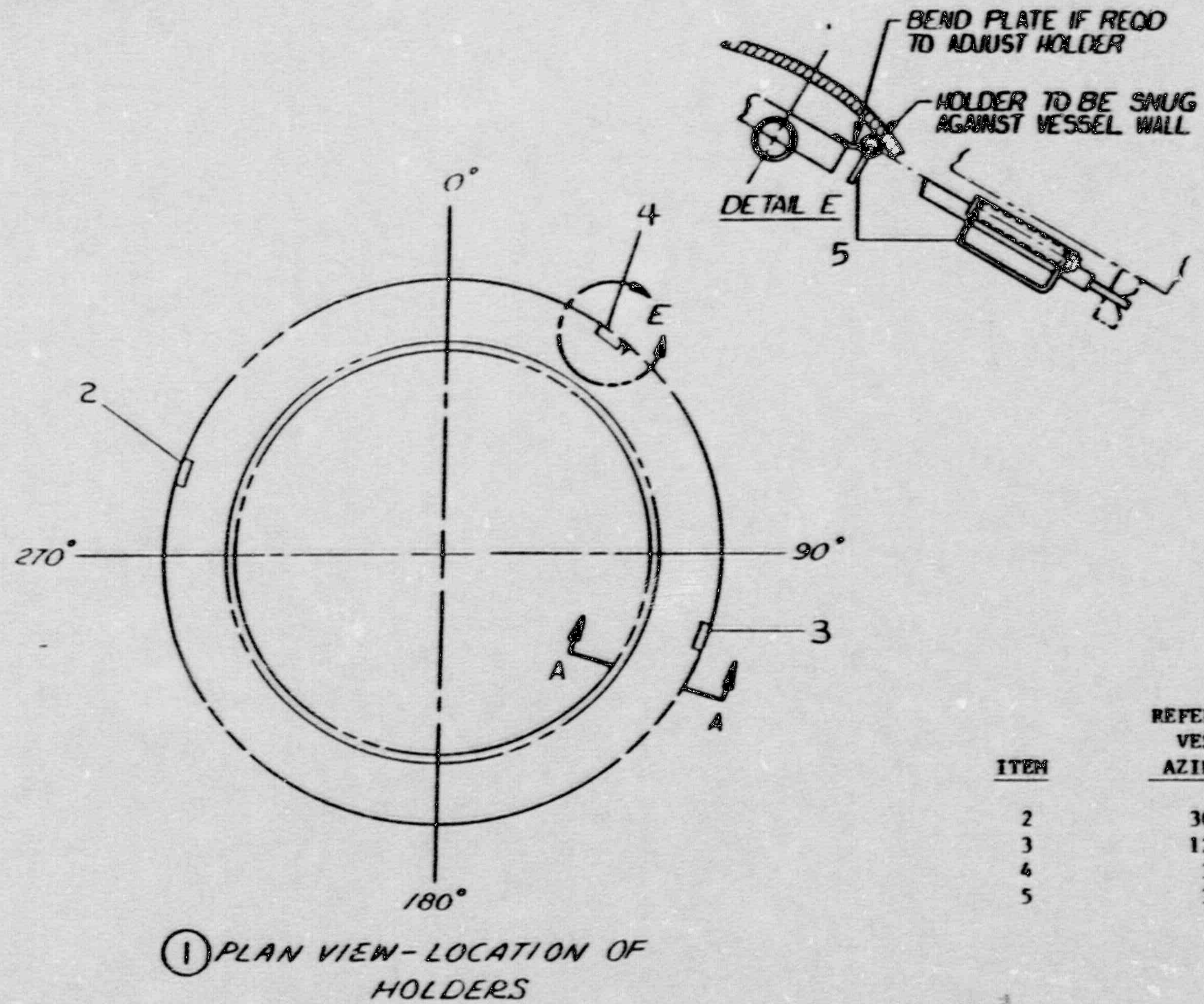
A.	TEST REPORT FOR FLUX WIRE DOSIMETER REMOVED FROM HOPE CREEK AT END OF CYCLE 1	A-1

1. INTRODUCTION

In February, 1988, Hope Creek Generating Station (Hope Creek) completed its first fuel cycle. During the outage that followed, the flux wire dosimeter attached to the surveillance capsule at the vessel 30° azimuth was removed. The dosimeter was shipped to the General Electric Vallecitos Nuclear Center (VNC) in Pleasanton, CA for testing. The test results and the associated determination of peak vessel flux and fluence are presented in this report.

The surveillance program for Hope Creek consists of three surveillance capsules and one flux wire dosimeter. Each surveillance capsule contains Charpy and tensile specimens of the beltline base, weld and HAZ materials, and a set of flux wires used to determine the fluence experienced by the capsule. The surveillance capsules are scheduled to be withdrawn periodically during plant life (the current schedule required by ASTM E185-82 [1] is a capsule at 6, 15, and 32 effective full power years). In addition to the flux wires in the surveillance capsules, a flux wire dosimeter is attached to the capsule at 30°, as shown in Figure 1-1, for removal after the first fuel cycle. Since the vessel fluence is proportional to thermal power produced, the results of the flux wire dosimeter test are used to provide a calibration point of vessel fluence versus accumulated thermal power. A linear extrapolation provides an estimate of the fluence at 32 effective full power years (EFPY). It should be noted that the flux wires that will be removed later as part of the surveillance capsules will have an irradiation history more typical of normal operation, and will be useful for re-calibrating the 32 EFPY fluence estimate.



ITEM	REFERENCE VESSEL AZIMUTH
2	300°
3	120°
4	30°
5	30°

Figure 1-1. Schematic of Capsule Locations for Surveillance Program

2. ANALYSIS

The determination of the peak 32 EFPY fluence is basically a two-step process. First, the flux wires are analyzed to determine the flux and fluence at the dosimeter location. Then, lead factors are calculated which relate the flux magnitude at the dosimeter location to that at the location of peak flux.

The flux wire dosimeter was disassembled at VNC and the iron and copper flux wires were cleaned and weighed. Gamma spectrometry was used to determine the rate of disintegrations. The daily power history of the first fuel cycle was used, along with cross-section data developed for BWRs to transform the disintegration data into rates of irradiation, or flux (n/cm^2-s). The detailed procedure used in evaluating the flux wires is contained in the test report in Appendix A.

Determination of lead factors was done recently for a 251 inch diameter BWR/4 vessel with 764 fuel bundles. This matches the Hope Creek configuration. The lead factors are essentially geometry dependent. Plant-specific operating characteristics of the flux are accounted for in the results of the flux wire test. Furthermore, the lead factors were calculated with the latest methods, assuming an equilibrium fuel cycle, which is representative of a typical normal operation core power distribution. Therefore, the lead factors provide the best available means of predicting peak 32 EFPY fluence from the flux wire data.

Determination of the lead factors for the RPV peak location at the inside wall and 1/4 T depth was done using a combination of one-dimensional and two-dimensional finite difference computer analyses. The two-dimensional analysis established the relative fluence in the azimuthal direction at the vessel surface and 1/4 T depth. A series of one-dimensional analyses was done to determine the core height of the axial flux peak and its relationship to the surveillance capsule height. The combination of azimuthal and axial distribution results provides the lead factor between the dosimeter location and the peak flux location.

The two-dimensional DOT computer program was used to solve the Boltzman transport equation using the discrete ordinate method on an (R,θ) geometry, assuming a fixed source. One eighth core symmetry was used with periodic boundary conditions at 0° and 45°. Neutron cross sections were determined for 26 energy groups, with angular scattering approximated by a third-order Legendre expansion. A total of 113 radial intervals and 45 azimuthal intervals were used. The model, shown in Figure 2-1, consists of an inner and outer core region, the shroud, water regions inside and outside the shroud, the vessel wall, and an air region representing the drywell. Flux as a function of azimuth was calculated, establishing the azimuth of the peak flux and its magnitude relative to the flux at the dosimeter location of 30°. This factor, the azimuthal component of the lead factor, is shown in Figure 2-2.

The one-dimensional computer code (SN1D) was used to calculate radial flux distribution for several core elevations at the peak azimuth angle. The elevation of the peak flux was determined, as well as its magnitude relative to the flux at the dosimeter elevation. This factor is the axial component of the lead factor. The lead factor between the peak and dosimeter locations was calculated as the azimuthal component times the axial component.

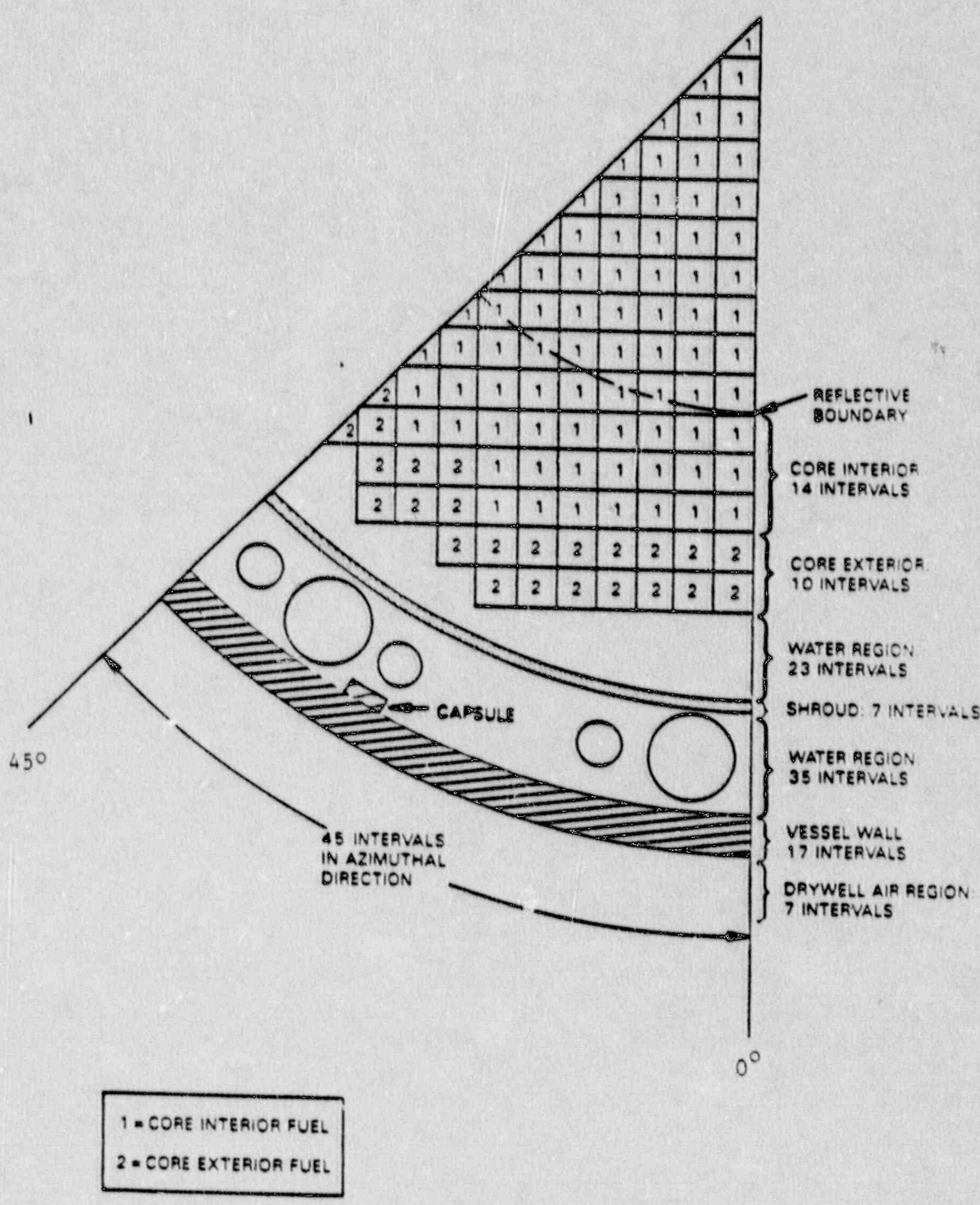


Figure 2-1. Schematic of Model for Two-Dimensional Flux Distribution Analysis

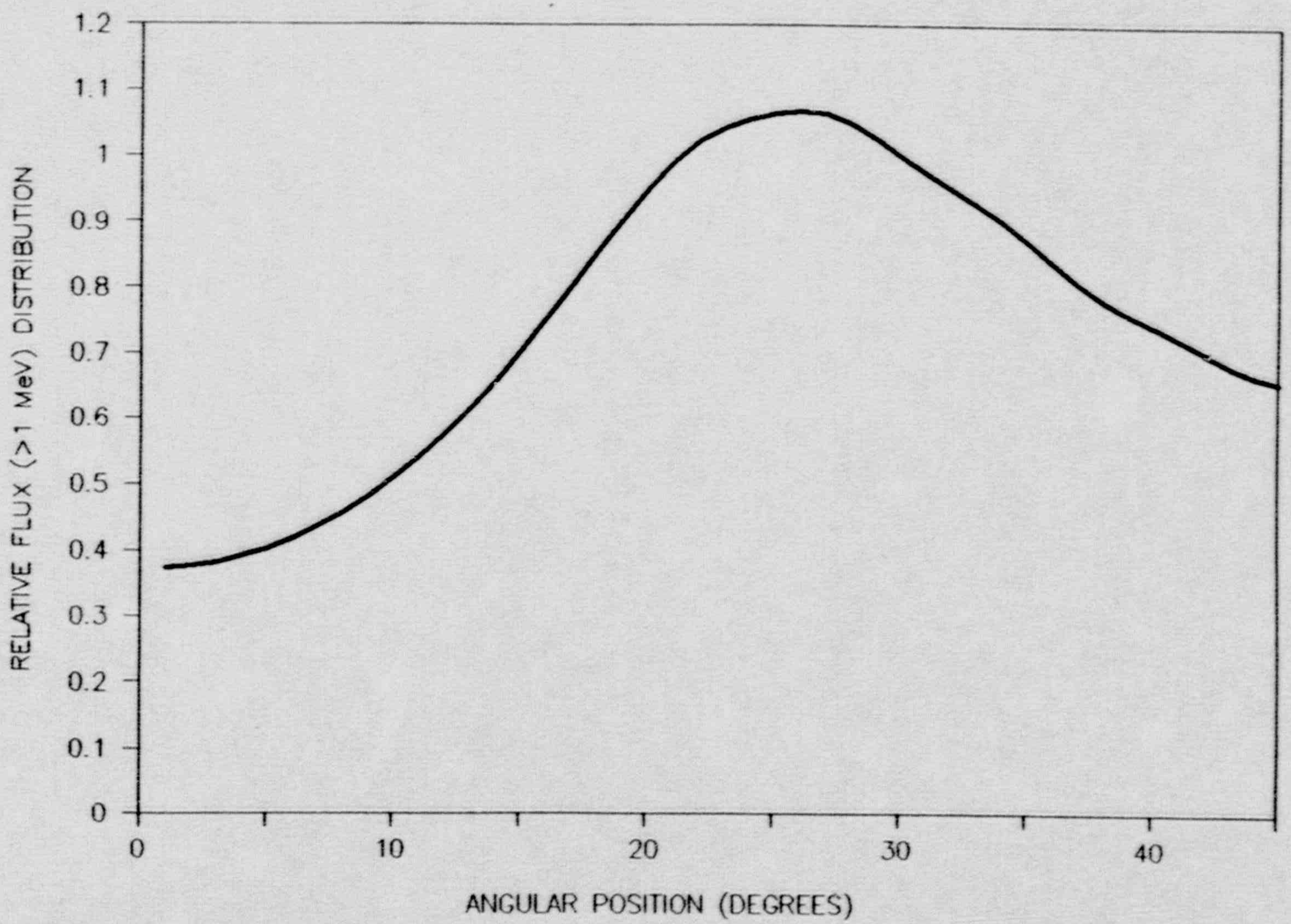


Figure 2-2. Relative Fast Neutron Flux Variation with Vessel Azimuth

3. RESULTS

The flux wire dosimeter test results are presented in detail in Appendix A. A summary of the >1 MeV flux and fluence values for the dosimeter are presented in Table 3-1. As discussed in the test report, there is an uncertainty of ±25% on the >1 MeV flux and fluence. Table 3-1 shows the upper bound values with the nominal values.

The lead factors for the peak location inside surface and 1/4 T depth are presented in Table 3-1 with the dosimeter test results. The I.D. lead factor is used to predict the peak I.D. fluence according to the following equation:

Peak Fluence = (Dosimeter Flux)*(Full Power Seconds)/Lead Factor

The first fuel cycle for Hope Creek consisted of 452 days of operation with an average capacity factor of 0.825. This is equivalent to 372.9 days at full power, or 1.02 EFPY. These values are used to calculate the fluence values at the end of cycle one (EOC1) and at 32 EFPY, as shown in Table 3-1.

The fluences at the peak location I.D. and 1/4 T are plotted as a function of EFPY in Figure 3-1. Regulatory Guide 1.99, Revision 2 [2] (Rev 2) includes a method for determining the 1/4 T fluence from the I.D. fluence. The Rev 2 method is to take the I.D. fluence, f_{surf}, and attenuate that value to the depth x according to the relationship

$$f_x = f_{surf} \, (e^{-0.24x}).$$

This method has been used to calculate the 1/4 T fluence plotted in Figure 3-1, assuming a vessel thickness of 6.1 inches. If the computed lead factor for the 1/4 T were used, the 1/4 T peak fluence would be slightly lower (for 32 EFPY, 4.5×10^{17} instead of 4.6×10^{17}).

Table 3-1

FLUENCE DETERMINATION FOR THE PEAK LOCATION IN THE HOPE CREEK VESSEL

Time at Power:

EOC1	1.02 EFPY = 3.22×10^{7} seconds
32 EFPY	32 EFPY = 1.01×10^{9} seconds

Lead Factors:

I.D.	0.95
1/4 T	1.38 [(a)]

Dosimeter Flux ($n/cm^2 \cdot s$) 6.2×10^{8} (nominal) 7.8×10^{8} (upper bound)

FLUENCE (n/cm^2):

	NOMINAL	UPPER BOUND
EOC1 Peak I.D.	2.1×10^{16}	2.6×10^{16}
EOC1 Peak 1/4 T	1.5×10^{16}	1.8×10^{16}
32 EFPY Peak I.D.	6.6×10^{17}	8.2×10^{17}
32 EFPY Peak 1/4 T	4.6×10^{17}	5.7×10^{17}

(a) 1/4 T lead factor provided for information only.
1/4 T fluence calculated from I.D. fluence, according to Rev 2.

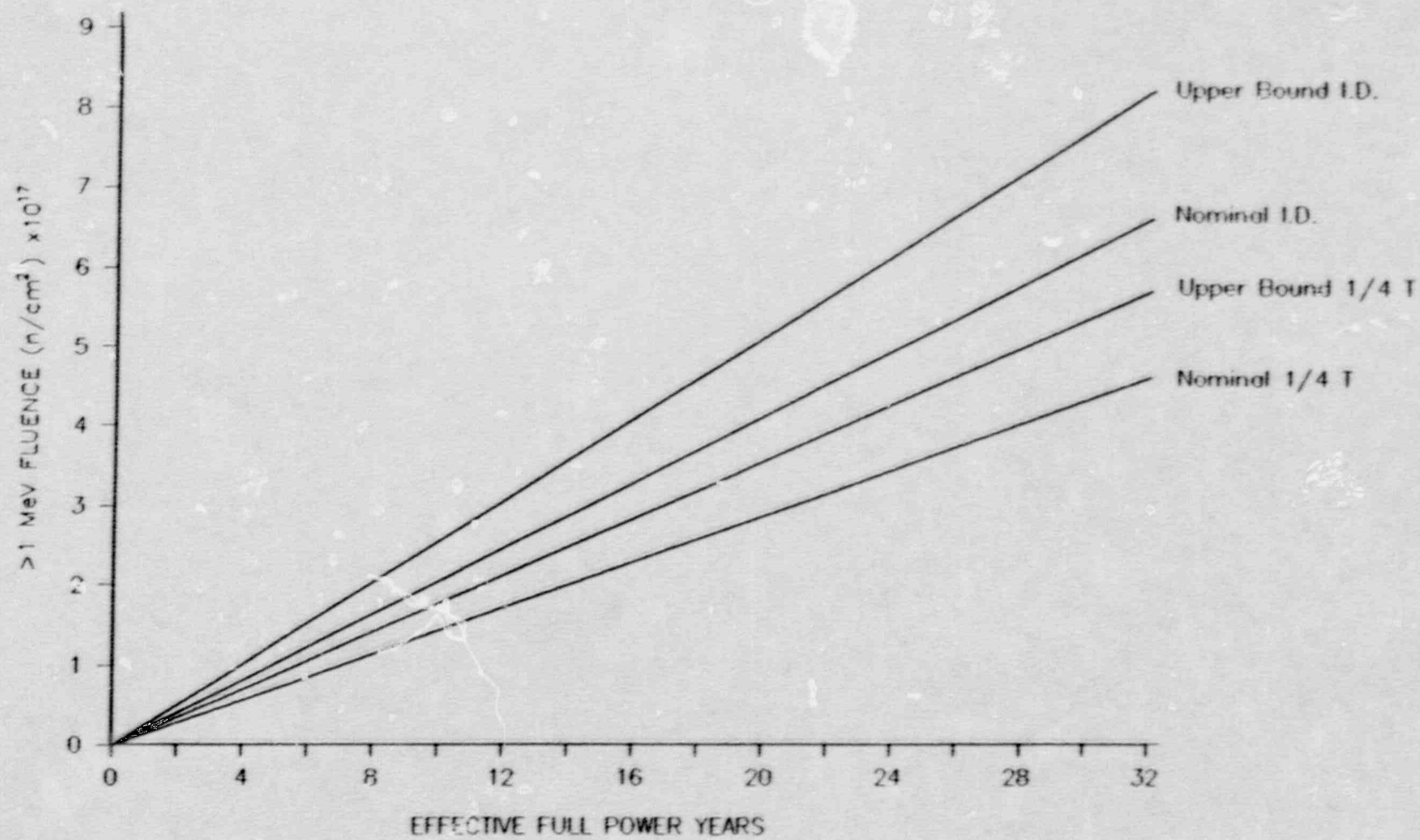


Figure 3–1. Vessel Peak Fluence Based on Dosimetry Results

4. CONCLUSIONS

The flux wire test results summarized in Table 3-1 show a nominal peak fluence on the vessel ID at 32 EFPY of 6.6×10^{17} n/cm². The fluence determined by dosimetry is substantially lower than the design I.D. fluence value of 1.7×10^{18} n/cm² shown in Appendix 5A of the FSAR. This is not surprising, since the FSAR value was intended to be a conservative design-basis fluence. Furthermore, flux wire tests for other 251 inch BWR/4 vessels with 764 fuel bundles have given similar results.

The results from the flux wire testing are generally used to modify the pressure-temperature curves in the Technical Specification. In particular, the NRC has issued Generic Letter 88-11 [3], which requires a revision of the pressure-temperature (P-T) curves within two outages if the Rev 2 prediction of vessel embrittlement is less conservative than that used as the basis for the P-T curves. Therefore, it is recommended that the Hope Creek P-T curves be revised, taking into account Rev 2 embrittlement predictions and the fluence prediction based on the flux wire test.

Figure 3-1 presents best-estimate and upper bound fluence values. It is certainly conservative to use the upper bound value in revising embrittlement predictions. However, it is reasonable to use the best-estimate. This conclusion is based on the fact that the prediction methodology developed for Rev 2 was based on commercial reactor surveillance data, using best-estimate fluence values. The size of the Margin term in Rev 2 was determined to account for scatter in the data base due to uncertainties such as fluence determination and Charpy test interpretation. Therefore, calculating embrittlement shift with Rev 2 on the basis of best-estimate fluence is consistent with the use of the Margin term.

5.0 REFERENCES

[1] "Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels," Annual Book of ASTM Standards, E185-82, July 1982.

[2] "Radiation Embrittlement of Reactor Vessel Materials," USNRC Regulatory Guide 1.99, Revision 2, May 1988.

[3] "NRC Position on Radiation Embrittlement of Reactor Vessel Material and Its Impact on Plant Operations," USNRC Generic Letter 88-11, July 1988.

APPENDIX A

TEST REPORT
FOR
FLUX WIRE DOSIMETER
REMOVED FROM
HOPE CREEK
AT
END OF CYCLE 1

GE Nuclear Energy

BWR Technology

FMT TRANSMITTAL
NO. 89-212-0010

FUEL MATERIALS TECHNOLOGY

TEST REPORT

DETERMINATION OF FAST NEUTRON FLUX DENSITY AND FLUENCE: HOPE CREEK GENERATING STATION

February 23, 1989

WA KTZ903595
DRF A00-02764

Prepared By: G. C. Martin — 2/23/89
G. C. Martin
Fuel Materials Technology
Date

Reviewed By: Lois K. Kessler — 2/23/89
L. K. Kessler
Fuel Materials Technology
Date

Approved By: R B Adamson — 2-24-89
R. B. Adamson, Manager
Fuel Materials Technology
Date

FMT

FMT TRANSMITTAL
NO. 89-212-0010

DETERMINATION OF FAST NEUTRON FLUX DENSITY AND FLUENCE: HOPE CREEK GENERATING STATION

SUMMARY

The fast neutron flux density and fluence (integrated neutron flux) at a capsule near the reactor vessel wall of the Hope Creek Generating Station of the Public Service Electric and Gas Company have been determined to be:

$6.2 \times 10^{8}\ n \cdot cm^{-2} \cdot s^{-1}$	>1 MeV full-power flux density
$1.0 \times 10^{9}\ n \cdot cm^{-2} \cdot s^{-1}$	>0.1 MeV full-power flux density
$2.0 \times 10^{16}\ n \cdot cm^{-2}$	>1 MeV fluence
$3.2 \times 10^{16}\ n \cdot cm^{-2}$	>0.1 MeV fluence

following the analysis of irradiated iron and copper flux dosimeters, in accordance with the GE CM&S Method No. 10.1.6.0 R3.

EXPERIMENTAL

Wires of iron and copper (three each) were irradiated in a GE pressure vessel capsule holder at Hope Creek from startup to February 13, 1988 (end of cycle 1). Each wire was removed from the capsule, cleaned with 4N or 8N HNO_3, weighed, mounted on a counting card, and analyzed for its radioactivity content by gamma spectrometry. Each iron wire was analyzed for Mn-54 content and each copper wire for Co-60 at a calibrated 4 cm source-to-detector distance with 100-cc and 80-cc Ge(Li) detector systems.

From the monthly cycle summary report and monthly histograms, the irradiation time periods were calculated. Operating days for each of two periods and the reactor average power fraction are shown in Table 1. The zero power days between the fuel periods are also listed.

TABLE 1. Hope Creek Irradiation Periods (Cycle 1)

Period	Date	Days	Full Power* Fraction	Between Period Time (Days)
1	10/12/86 - 09/18/87	342	0.785**	38
2	10/27/87 - 02/13/88	110	0.949	
		452 (Total)	0.825 (Av)	

* Full power was 3293 MW_t.
** Includes low-power irradiations of August and September, 1986.

DISCUSSION OF RESULTS

From the activity measurements and power history, reaction rates for $^{54}Fe(n,p)^{54}Mn$ and $^{63}Cu(n,\alpha)^{60}Co$ were calculated. These data appear in Table 2. The Hope Creek >1 MeV flux density reaction cross sections for iron and copper were calculated to be 0.212 barns and 0.00374 barns, respectively. These values were obtained from measured cross section data functions from more than 65 spectral determinations for BWRs and for the General Electric Test Reactor using activation monitors and spectral unfolding techniques. These data functions were applied to BWR pressure vessel locations based on water gap (fuel to pressure vessel) distances. The >1 MeV/>0.1 MeV cross section ratio at BWR pressure vessel locations is approximately 1.6.

The Hope Creek full-power >1 MeV flux density results were consistent for the two dosimeter types (Fe, Cu) (see Table 2). These results were 6.3×10^8 and 6.2×10^8 $n \cdot cm^{-2} \cdot s^{-1}$, respectively. The determined full-power flux density and actual fluence results at the reactor vessel wall capsule holder location are given in Table 2. The average >1 MeV and >0.1 MeV values of 6.2×10^8 and 1.0×10^9 $n \cdot cm^{-2} \cdot s^{-1}$ from the flux monitors were calculated by dividing the reaction rate measurement data for the reactions $^{54}Fe(n,p)^{54}Mn$ and $^{63}Cu(n,\alpha)^{60}Cu$ by the appropriate cross sections. The corresponding fluence results, 2.0×10^{16} and 3.2×10^{16} $n \cdot cm^{-2}$ for >1 MeV and >0.1 MeV, respectively, were obtained by multiplying the full-power flux density values by the product of the total seconds irradiated (3.905×10^7 s) and the full-power fraction (0.825).

The 2σ errors of the values in Table 2 are estimated to be:

± 5% for dps/g
± 10% for dps nucleus (sat'd)
± 25% for ϕ and ϕt >1 MeV
± 35% for ϕ and ϕt >0.1 MeV

TABLE 2. FLUX DENSITY AND FLUENCE DETERMINATIONS - HOPE CREEK IRRADIATION: STARTUP - FEBRUARY 13, 1988

Wire (Element)	Wire Weight (g)	dps/g Element (at end of Irradiation)	Reaction Rate [dps/nucleus (sat'd)]		ϕFP^* Flux Density ($n \cdot cm^{-2} \cdot s^{-1}$) >1 MeV	ϕFP^* Flux Density ($n \cdot cm^{-2} \cdot s^{-1}$) >0.1 MeV	ϕt Fluence ($n \cdot cm^{-2}$) >1 MeV	ϕt Fluence ($n \cdot cm^{-2}$) >0.1 MeV
Iron A	0.1142	4.20×10^{4}	1.32×10^{-16}					
Iron B	0.1250	4.24×10^{4}	1.34×10^{-16}					
Iron C	0.1282	4.20×10^{4}	1.32×10^{-16}					
			1.33×10^{-16}	(Av)	6.3×10^{8}	1.0×10^{9}	2.0×10^{16}	3.2×10^{16}
Copper A	0.4302	1.85×10^{3}	2.29×10^{-18}					
Copper B	0.4266	1.87×10^{3}	2.32×10^{-18}					
Copper C	0.4566	1.86×10^{3}	2.31×10^{-18}					
			2.31×10^{-18}	(Av)	6.2×10^{8}	9.9×10^{8}	2.0×10^{16}	3.2×10^{16}
			This Report:		6.2×10^{8}	1.0×10^{9}	2.0×10^{16}	3.2×10^{16}

* At full Power (3293 MW_t).