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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Documents results of 830125 meeting between util & NRC to resolve pressurized thermal shock issue.Requests another meeting in early Mar to discuss methods of obtaining addl thermal margin in reactor core design.

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CP&L

Carolina Power & Light Company

FEB 09 1983

Mr. H. R. Denton, Director
Office Of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
PRESSURIZED THERMAL SHOCK

Dear Mr. Denton:

Summary

On January 25, 1983 representatives from Carolina Power & Light Company (CP&L) met with your staff to provide an update on CP&L's program to resolve the Pressurized Thermal Shock (PTS) issue. The purpose of this letter is to document the results of that meeting and the information provided by CP&L at that meeting.

Detailed Discussion

At the meeting on January 25, 1983 information was provided on the following subjects:

1) Fluence Calculations (Attachment 1)
2) Surveillance Capsule Results (Attachment 2)
3) Vessel Weld Material Properties and Damage Mechanisms (Attachment 3)
4) Inservice Inspection (Attachment 4)
5) Flux Reduction Techniques (Attachment 5)

Copies of CP&L's presentation were provided to the Staff at the meeting, but major points of the presentation are repeated in this letter and its attachments.

At the conclusion of the meeting, CP&L requested that another meeting be held in early March to discuss methods of obtaining additional thermal margin in the H. B. Robinson reactor core design. This additional

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margin would then allow CP&L to potentially design a core reload pattern which would achieve a flux reduction in excess of the factor of 2 attained by the present core. It is CP&L's intent to present to the NRC in March justification for approval of the changes in methodology outlined in Attachment 6. As outlined in Attachment 6, CP&L would then seek an expedited review by the NRC Staff of these justifications on the schedule provided in that attachment in order that CP&L could proceed with an indepth design and reload analysis for the next cycle. Acceptance of the schedule shown and dedication of the NRC resources to the required reviews and approvals are essential if further flux reductions are to be achieved.

Finally, as outlined in Attachment 1 and discussed at the meeting of January 25, 1983, CP&L's most recent fluence calculations show the H. B. Robinson Reactor Vessel reaching the Generic PTS Screening Criteria in 1993 or later. Further flux reductions as outlined above have the potential of extending that time to greater than the year 2000. Although CP&L's efforts to resolve the PTS issue for H. B. Robinson continue to focus on achieving that resolution in the short term, the NRC needs to recognize that this process can and should occur in an atmosphere of careful deliberation and reasoned analysis and not in a crisis environment. As mentioned in previous correspondence and meetings with the NRC, CP&L is committed to resolution of the PTS issue as quickly as possible but believes that the time available for resolution is more than ample and any further elevated regulatory action by the NRC is unwarranted. Therefore, subject to review of our calculations, CP&L requests that NRC acknowledge in future reports to the Commissioners, correspondence or news releases that the H. B. Robinson Plant will not exceed the NRC Generic Screening Criteria until 1993.

Conclusions

As demonstrated by the meeting held January 25, 1983 between CP&L and the NRC, it should be apparent that CP&L has a large ongoing program to resolve the PTS issue for H. B. Robinson. The major points arising out of that meeting which are documented in this letter are:

1) The H. B. Robinson Reactor Vessel will not reach the NRC's PTS Screening Criteria until 1993 or later.

2) Flux reduction beyond the factor of 2 attained in the present cycle could extend the time of reaching the screening criteria significantly.

3) Further flux reductions will require acceptance by the NRC of certain changes in methodology in order to obtain more margin to core thermal limits.

4) Carolina Power & Light Company will request a meeting in early March to provide justification for these changes. Expedited NRC Staff review will be required in order to allow CP&L to complete a core reload design and analysis prior to the next fuel cycle.

We trust that the contents of this letter are responsive to your needs. If you have any questions on this submittal or CP&L's PTS Program, please do not hesitate to contact me or a member of my staff.

Yours very truly,

M A McDuffie

for E. E. Utley
Executive Vice President
Power Supply and
Engineering & Construction

JJS/EEU/kjr (6119JJS)

cc: Mr. J. P. O'Reilly (NRC-II)
Mr. S. Weise (NRC-HBR)
Mr. G. Requa (NRC)
Mr. T. Speis (NRC)
Mr. F. Schroeder (NRC)
Mr. R. Woods (NRC)
Mr. W. J. Dircks
Chairman N. J. Palladino
Commissioner J. F. Ahearne
Commissioner V. Gilinsky
Commissioner T. M. Roberts
Commissioner J. K. Asselstine

Attachment 1

Fluence Calculations

The following discussion provides the basis for CP&L's most recent fluence calculations:

Calculated Fast Neutron Exposure of the Pressure Vessel

In the 150-day letter submitted to the NRC on January 25, 1982, a calculated value for the peak fast neutron flux ($E_0 > 1.0$ Mev) at the inner radius of the pressure vessel was listed as 6.5×10^{10} n/cm^2 sec.

This calculated flux level was derived using the benchmarked methodology described in WCAP 10019. In particular, the following assumptions used in the analysis are of importance:

1. Results are based on 2D R,Θ discrete ordinates analysis.

2. Nominal generic reactor dimensions are used throughout.

3. Generic long term core power distributions based on an out-in fuel management scheme are employed.

4. GAMBIT cross-sections with a P_1, scattering approximation are used.

Based on the power reactor surveillance capsule dosimetry data also summarized in WCAP 10019, this overall methodology including assumptions 1, 2, 3, and 4 has been benchmarked and an uncertainty of ±20% has been assigned to the analytical results.

Subsequent to the 150-day letter submittal and during continuing discussions with the NRC and its contractor Brookhaven National Laboratory (BNL), Westinghouse performed a second calculation using the SAILWR ENDF IV based cross-section library which includes a P_3 scattering approximation. This computation still employed design basis core power distributions and nominal reactor dimensions.

In an attempt to further improve the applicability of the transport calculations to the H. B. Robinson pressure vessel, a further study was undertaked to compare the actual burnup averaged power distribution for the first 8 cycles of operation with the design basis distribution that was used in the original calculations.

Figure 1-1 depicts the calculated average peripheral power distribution for the first 8 cycles along with the design basis values. Also, shown on Figure 1 is the peripheral power distribution for the low leakage pattern installed at the start of cycle 9. An examination of Figure 1-1 indicates that a modest reduction in pressure vessel exposure could be realized by consideration of the actual rather than design basis power distributions for the first 8 cycles of operation. Furthermore, it is evident that the incorporation of the low leakage core in cycle 9 will result in a rather large reduction in neutron flux at several azimuthal locations on the pressure vessel beltline.

In order to fully quantify the effects of these actual power distributions on pressure vessel exposure, additional neutron transport calculations were performed for the 8 cycle average and the cycle 9 low leakage power distributions. These computations also employed the SAILWR cross-section library with a P_3 scattering approximation. The results of these analyses are summarized in Figure 1-2. The reduction in neutron flux at the pressure vessel beltline is clearly evident for both core loading patterns.

Using data from Figure 1-2 as well as from the original 150-day submittal, the calculated maximum fast neutron flux (E > 1.0 Mev) at the midplane of the reactor core and at the axial location of the circumferential weld is summarized as follows:

max (E > 1.0 Mev) at RV Inner Radius

	Design Basis Power Distribution	Actual Power Distribution
150 Day Letter	6.50×10^{10}	--
Cycles 1-8	7.51×10^{10}	6.62×10^{10}
Cycle 9	--	3.82×10^{10}

max (E > 1.0 Mev) at Circumferential Weld

	Design Basis Power Distribution	Actual Power Distribution
150 Day Letter	5.6×10^{10}	--
Cycles 1-8	6.53×10^{10}	5.76×10^{10}
Cycle 9	--	3.32×10^{10}

To perform best estimate fluence projections for the H. B. Robinson circumferential weld, the values of neutron flux obtained from calculations based on actual core power distributions should be used. That is, for the first 7 cycles $\Phi = 5.76 \times 10^{10}$ n/cm^2 sec. and for the low leakage cycle $\Phi = 3.32 \times 10$ n/cm^2 sec.

In addition to the two dimensional transport calculations outlined above, a series of one-dimensional computations was also undertaken to assess the impact of differences in core and stainless steel nuclear densities used by BNL and Westinghouse in their analysis of the H. B. Robinson reactor. The results of this study indicate that:

1. If the same nuclear densities are used in the analysis, the BNL and Westinghouse calculations are in essential agreement.

2. If the missing elements in the CP&L supplied stainless steel nuclear densities are assumed to be iron rather than void, the BNL and Westinghouse calculations will agree within approximately 3%.

It is our understanding that fluence levels for use in PTS evaluations are intended to be nominal values with uncertainties covered by the 2σ increase in the RT_{NDT} trend curve. Further, it is our belief that the approach taken by

Westinghouse; i.e., assuming that stainless steel residual elements are iron, reflects common industry practice. A letter has been obtained from Frank Kam, of Oak Ridge National Laboratory (ORNL) which concurs in this latter belief.

A revision of the stainless steel number densities for the baffle, corebarrel, thermal shield and vessel given in a letter dated October 7, 1982 from S. R. Zimmerman to S. A. Varga is given in Table 1-1.

The maximum circumferential weld fluence has been plotted versus effective full power years and compared with the BNL calculations on Figure 1-3. The principal assumptions, including those accounting for prior differences, are noted on the graph.

Conclusions

Based on the above, CP&L believes that the following values should be used for fluence at the critical weld for H. B. Robinson:

13.5×10^{18} n/cm^2 @ 7.48 EFPY

1.05×10^{18} n/cm^2 per EFPY additional accumulation assuming a flux reduction factor of 2 in Cycle 9 and all subsequent cores

Based on these calculations, the H. B. Robinson vessel would reach the screening criteria sometime in excess of 13 total effective full power years (EFPY). Based on CP&L's planned operating and outage schedule for H. B. Robinson, the screening criteria would be reached in 1993 or later.

TABLE 1-1
H. B. Robinson Unit No. 2
Number Densities of Core Region Structurals (Revised)

	Fe	Cr	Ni	Mn
Baffle:				
Weight Percent	69.52	18.65	9.60	1.54
Number Density	5.95×10^{-2}	1.71×10^{-2}	7.82×10^{-3}	1.34×10^{-3}
Core Barrel:				
Weight Percent	69.49	18.66	9.68	1.55
Number Density	5.95×10^{-2}	1.71 x 10-2	7.89×10^{-3}	1.35×10^{-3}
Thermal Shield:				
Weight Percent	69.99	18.41	9.42	1.56
Number Density	5.99×10^{-2}	1.69×10^{-2}	7.68×10^{-3}	1.36×10^{-3}
Reactor Vessel:				
Weight Percent	97.75			1.32
Number Density	8.27×10^{-2}			1.20×10^{-3}

	P	S	Si	C
Baffle:				
Weight Percent	.026	.018	.60	.051
Number Density	4.1×10^{-5}	2.7×10^{-5}	1.02×10^{-3}	2.03×10^{-4}
Core Barrel:				
Weight Percent	.031	.014	.53	.050
Number Density	4.9×10^{-5}	2.1×10^{-5}	9×10^{-4}	1.99×10^{-4}
Thermal Shield:				
Weight Percent	.024	.018	.52	.056
Number Density	3.8×10^{-5}	2.7×10^{-5}	8.8×10^{-4}	2.23×10^{-4}
Reactor Vessel:				
Weight Percent	.009	.018	.23	.193
Number Density	1.38×10^{-5}	2.22×10^{-5}	1.0×10^{-4}	7.49×10^{-4}

Figure 1 - 1

H.B. Robinson Peripheral Power Distribution

0° / 45°

	H	G	F	E	D	C
15	0.93 0.81 0.44	0.77 0.67 0.33				
14		1.07 1.06 1.04	1.12 0.97 1.01	0.80 0.70 0.75		
13				1.04 1.07 1.11	0.85 0.76 0.83	
12					0.92 0.87 1.14	

← DESIGN BASIS
← ACTUAL 8 CYCLE AVERAGE
← CYCLE 9 LOW LEAKAGE

NOTE: For Fuel Positions H-15 & G-15, the following ratios apply

	H-15	G-15
8 CYCLE AVG. / DESIGN BASIS	0.87	0.87
CYCLE 9 / DESIGN BASIS	0.47	0.43

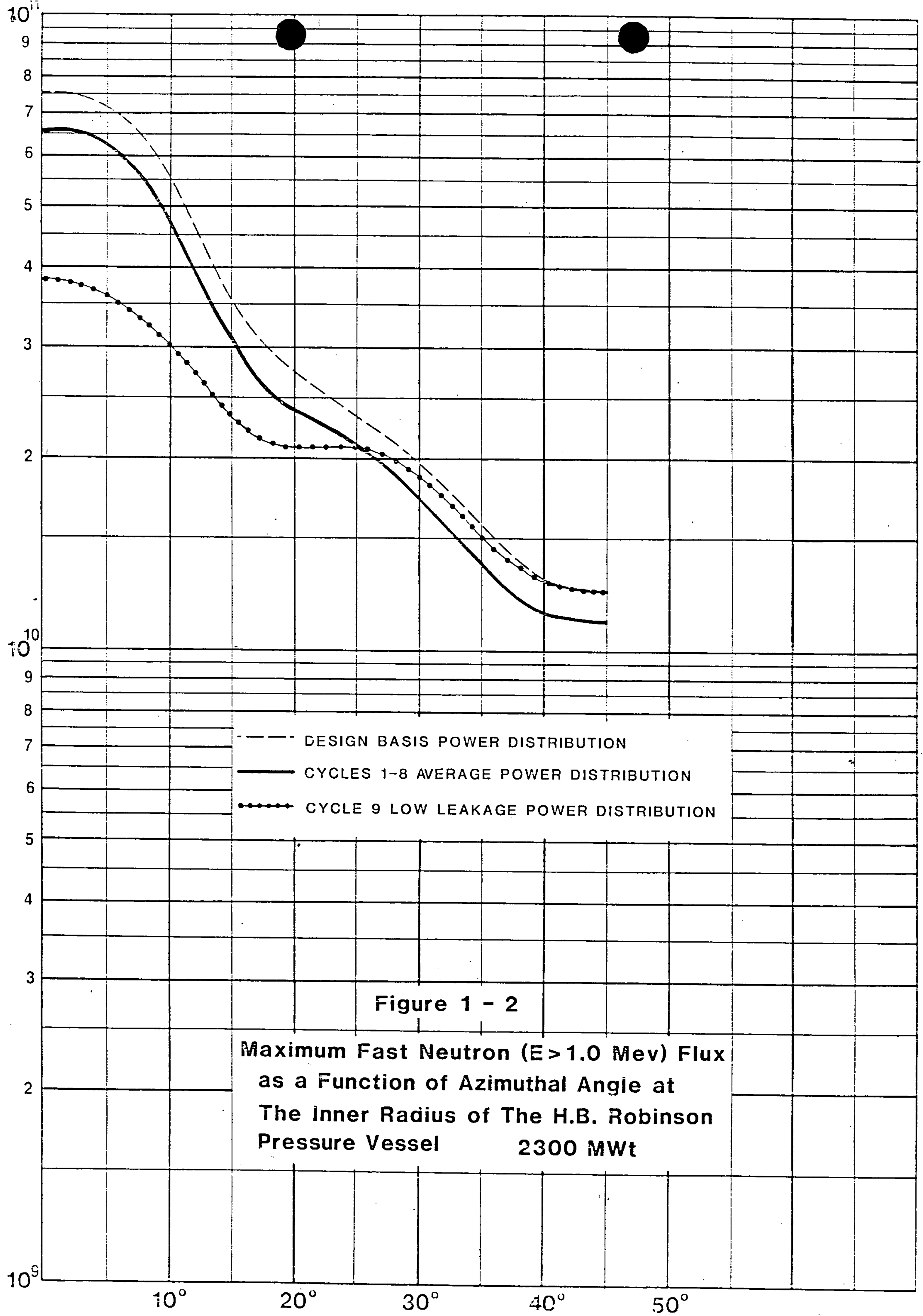


Figure 1 - 2

Maximum Fast Neutron (E>1.0 Mev) Flux as a Function of Azimuthal Angle at The Inner Radius of The H.B. Robinson Pressure Vessel 2300 MWt

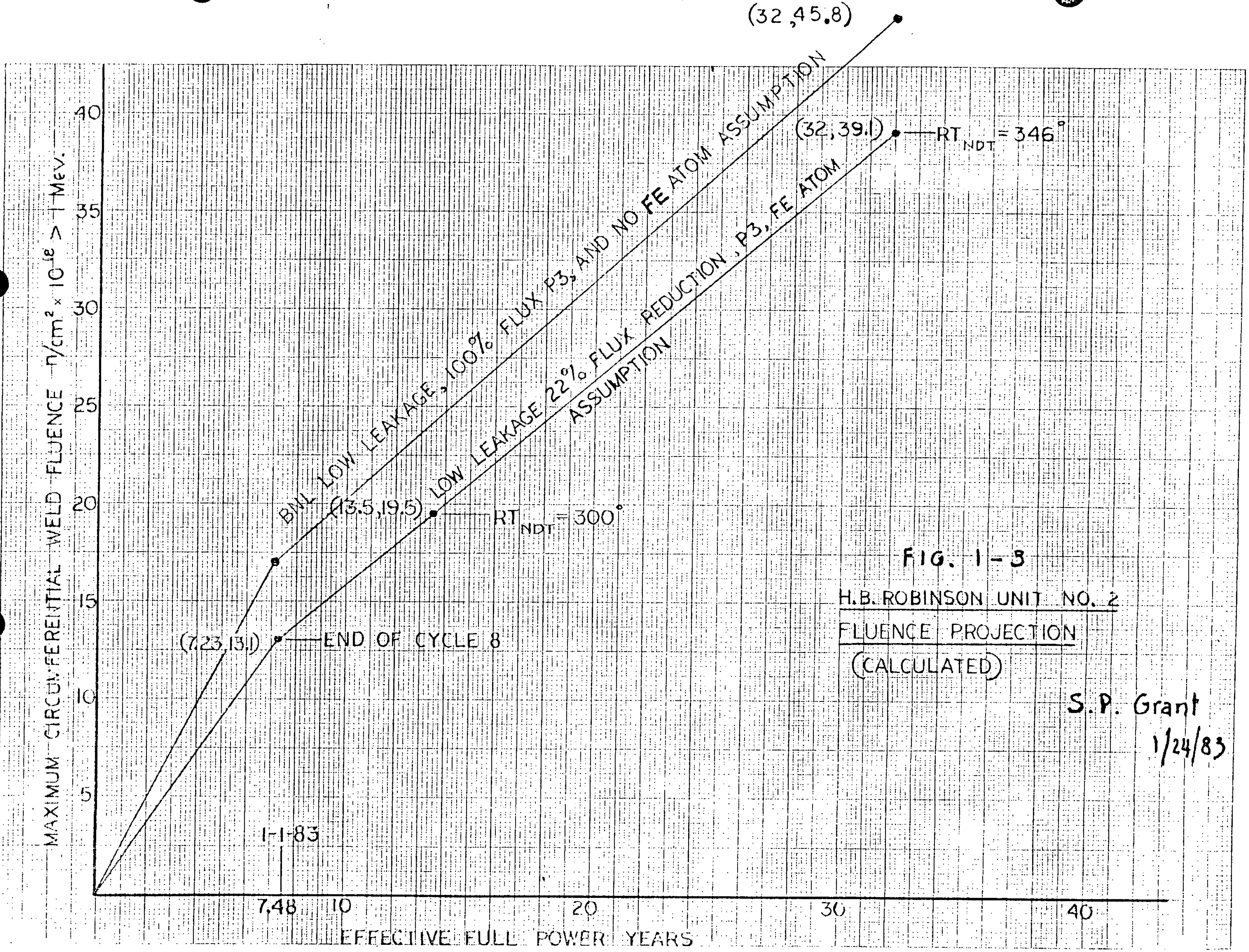


FIG. 1-3

H.B. ROBINSON UNIT NO. 2
FLUENCE PROJECTION
(CALCULATED)

S.P. Grant
1/24/83

Attachment 2

Surveillance Capsule Test Results

Capsule T was removed from the Robinson Unit 2 at the end of the 8th cycle and shipped to the Westinghouse Waltz Mill facility for testing as part of an EPRI surveillance test project. Testing has been completed, but formal issuance of a Westinghouse-EPRI report is not expected until April, 1983. The test results affecting radiation damage and fluence calculation of reactor lifetime projections are included herein.

Charpy V notch test points were grouped in the transition regions as opposed to shelf regions in order to better define the 3 ft-lb shift. The upper shelf is > 60 ft-lb at a fluence greater than that projected for the H. B. Robinson vessel for 32 EFPY. Concurrence in this approach was obtained from Warren Hazelton of the NRC staff. The shift is shown on Figure 2-1 to be 285°F at 30 ft-lb or 50 ft-lb.

The Charpy V curves for heat-affected zone, plate and correlation monitor materials are shown in Figures 2-2, 2-3, and 2-4. Strength and ductility results are shown on Figure 2-5 for the weld. Comparisons with Reg. Guide 1.99 are shown on Figure 2-6.

Temperature monitors at 570°F and 590°F did not melt.

Dosimetry measurements and calculations are reported in Tables 2-1 and 2-2.

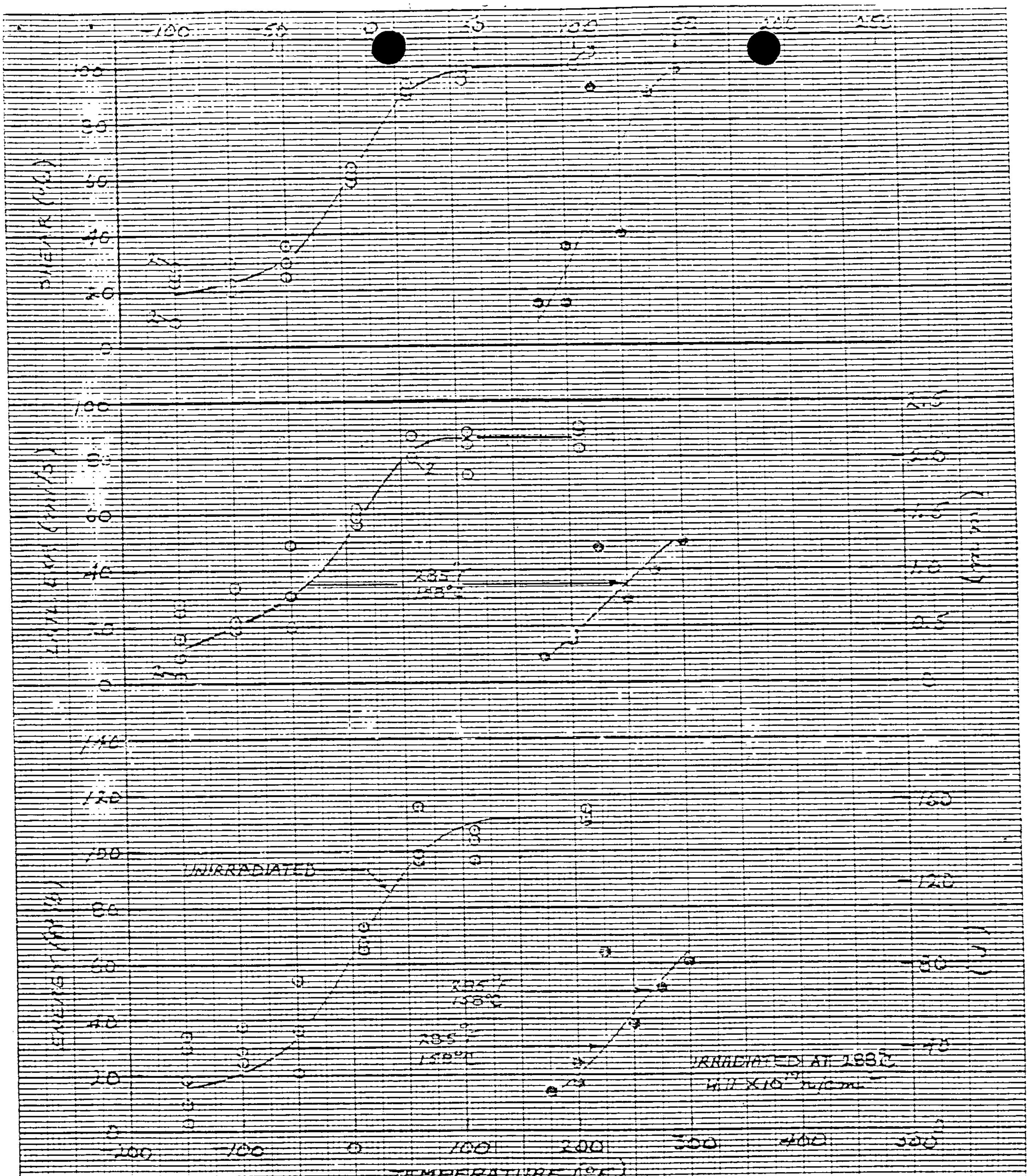


FIGURE 2-1 CHARPY V-NOTCH IMPACT DATA FOR H.B. ROBINSON UNIT 2 REACTOR VESSEL WELD METAL

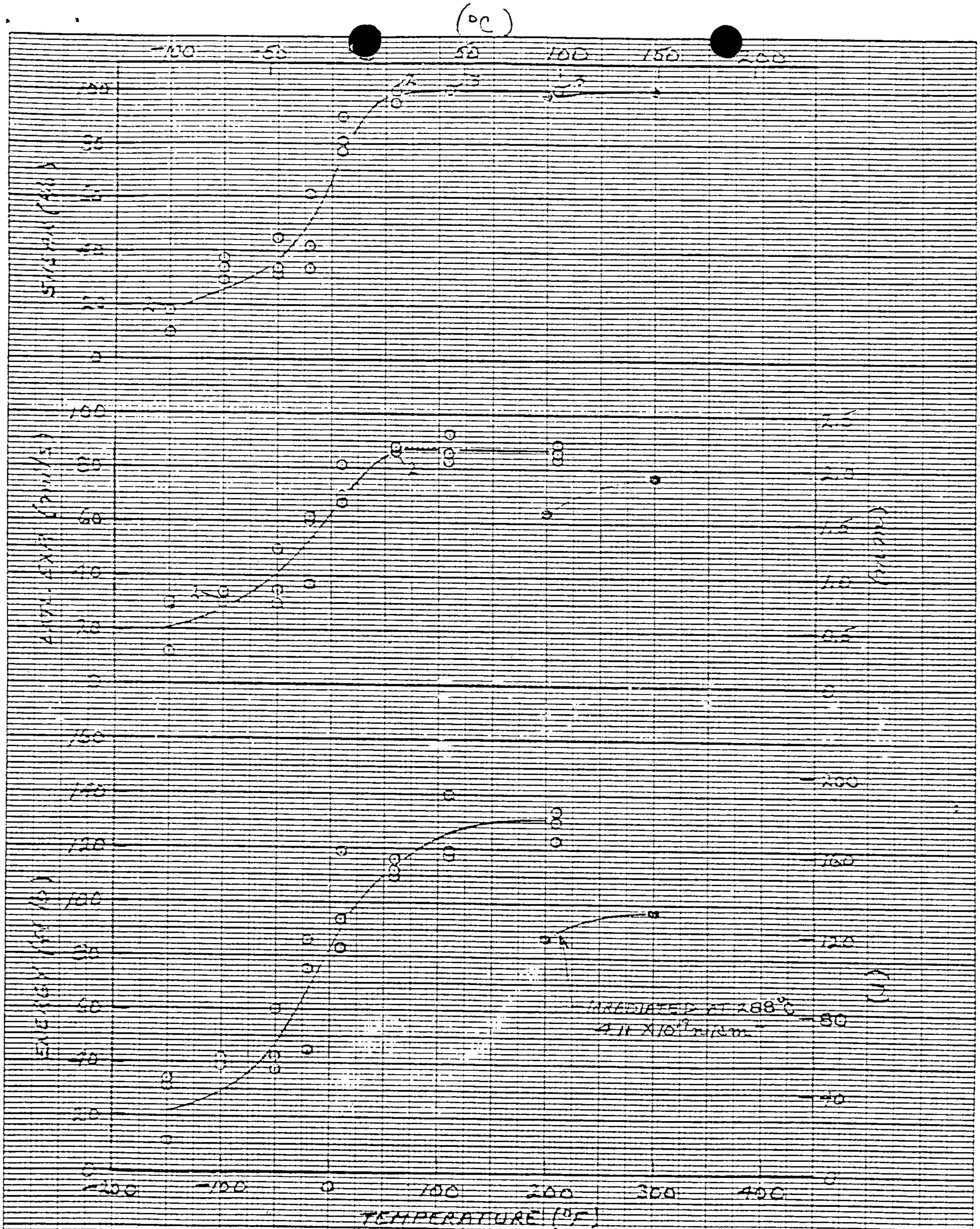


FIGURE 2-2 CHARPY V-NOTCH IMPACT DATA FOR H.B. ROBINSON UNIT 2 REACTOR VESSEL WELD HEAT AFFECTED ZONE MATERIAL

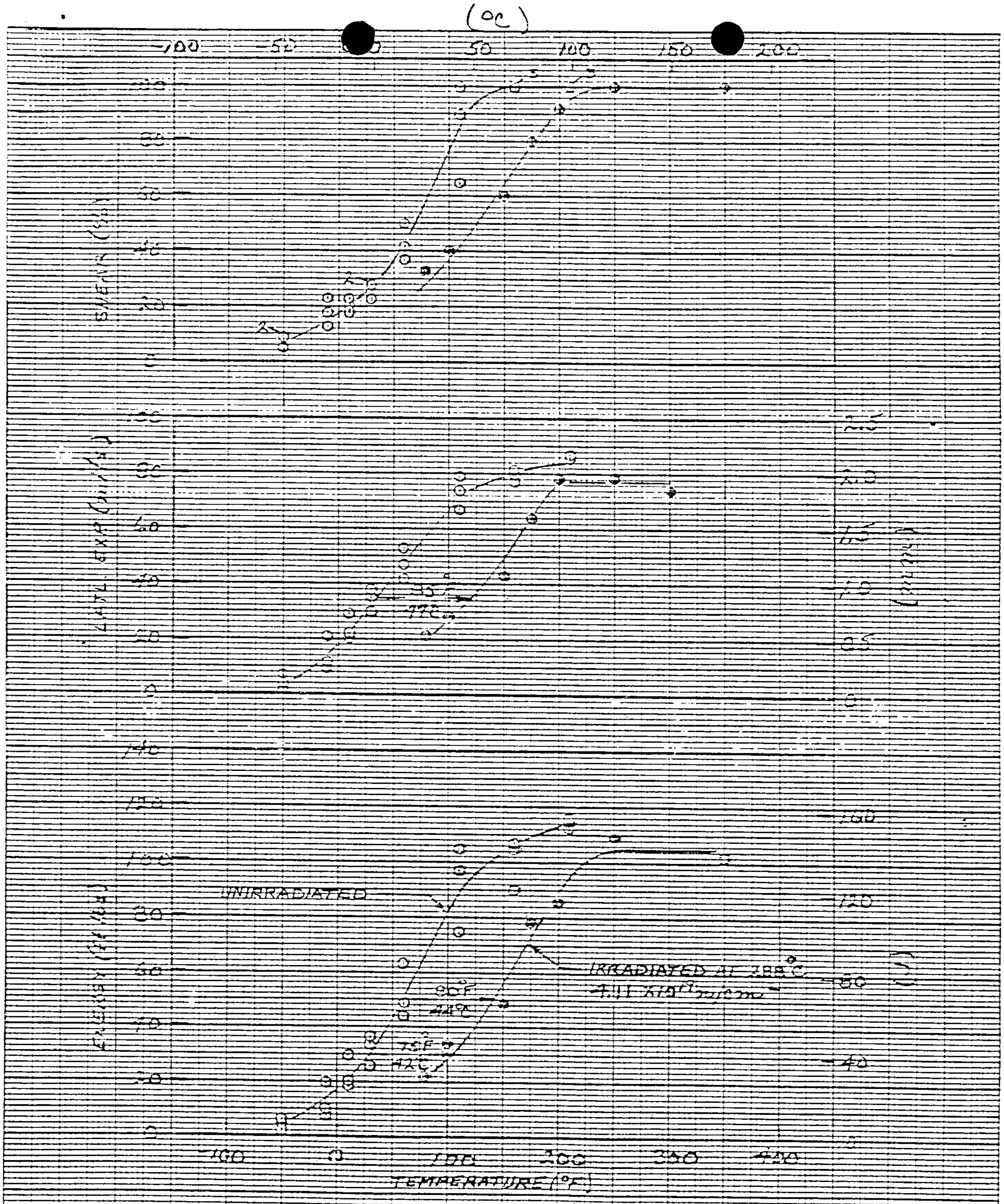


FIGURE 2-3 CHARPY V-NOTCH IMPACT DATA FOR H.B. ROBINSON UNIT 2 REACTOR VESSEL SHELL PLATE W10201-6

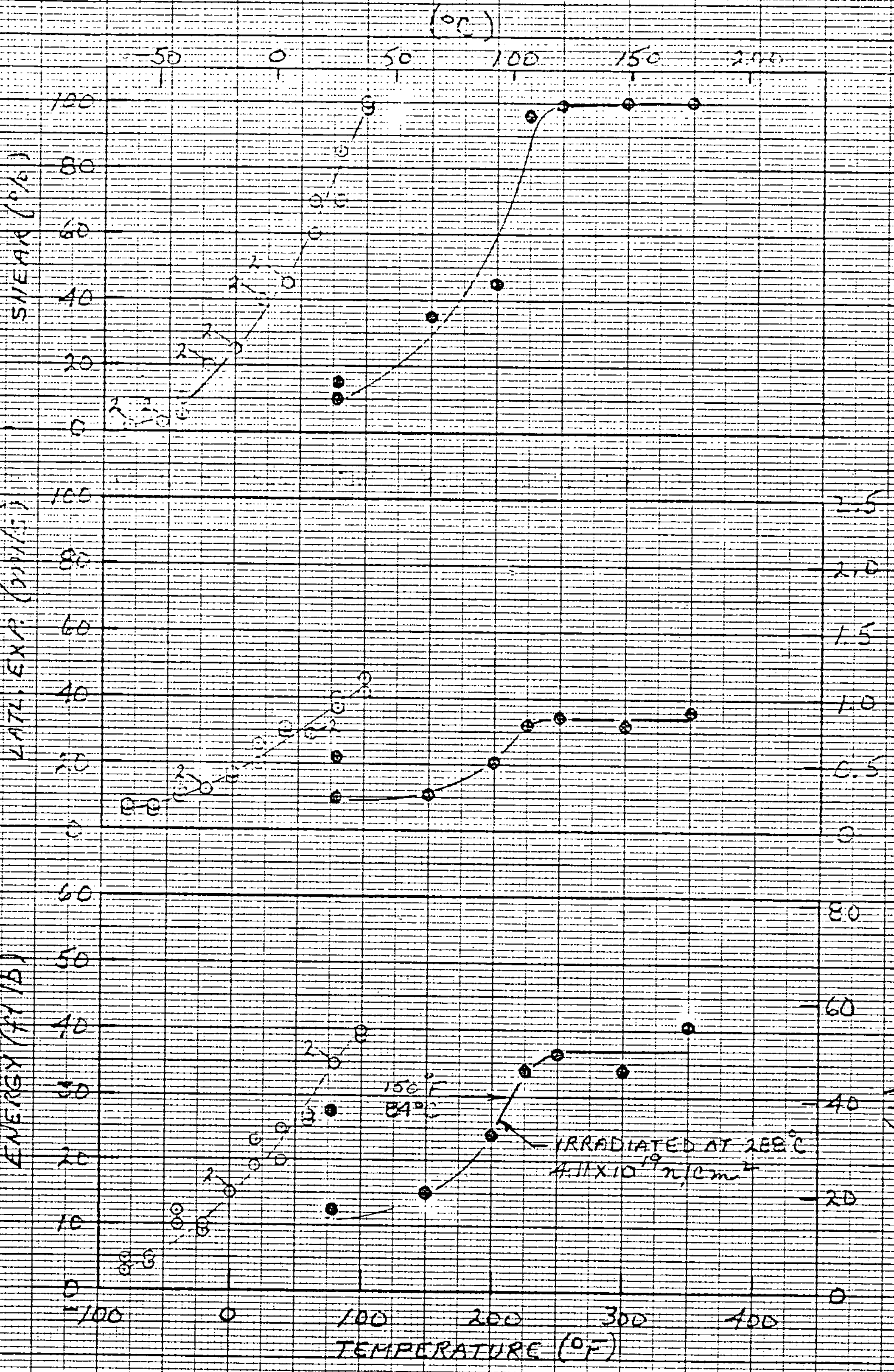


FIGURE 2-4 CHARPY V-NOTCH IMPACT DATA FOR H.B. ROBINSON UNIT 2 ASTM A302B CORRELATION MONITOR MATERIAL (TRANSVERSE ORIENTATION)

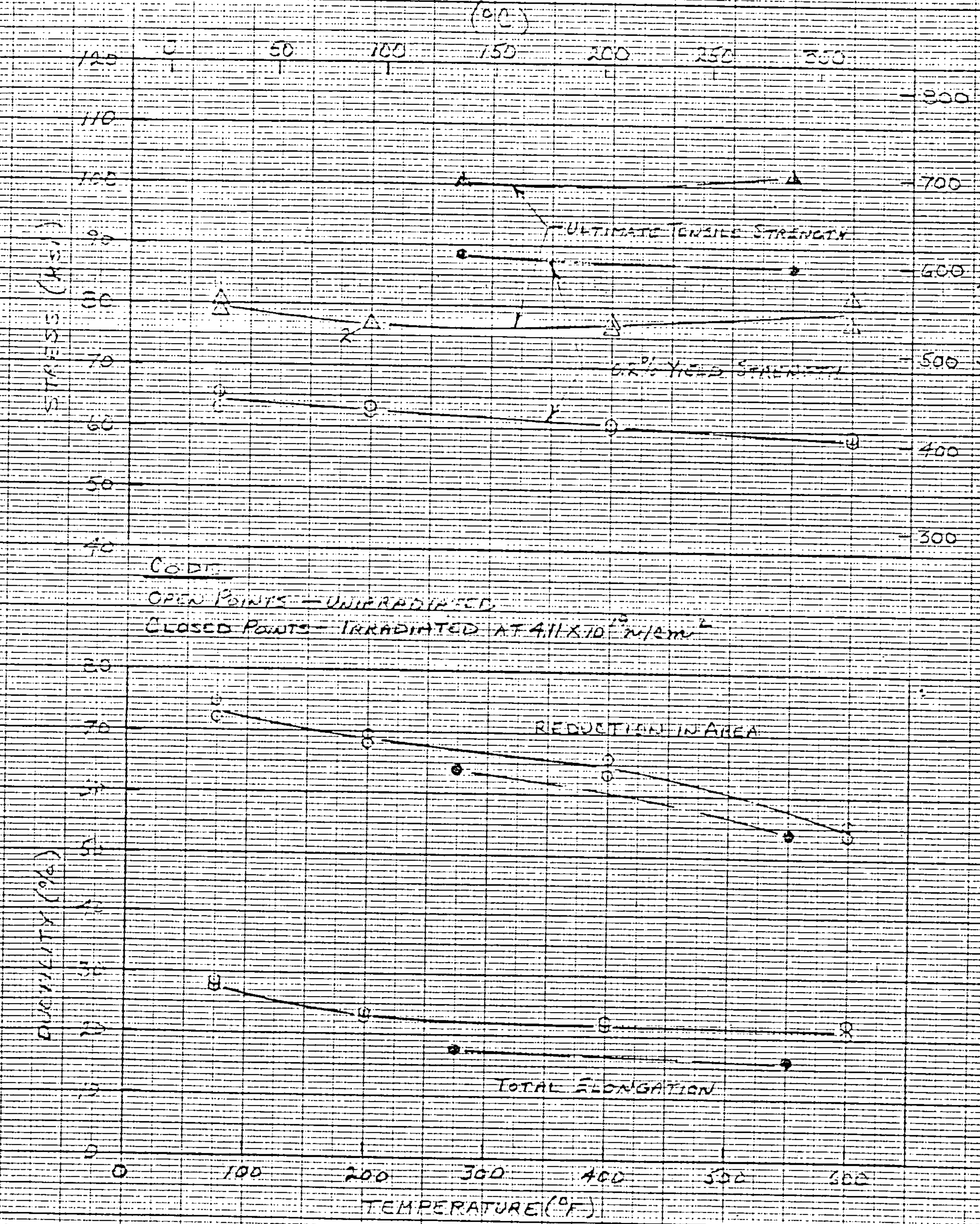


FIGURE 2-5 TENSILE PROPERTIES FOR H. B. ROBINSON UNIT 2
REACTOR VESSEL WELD METAL

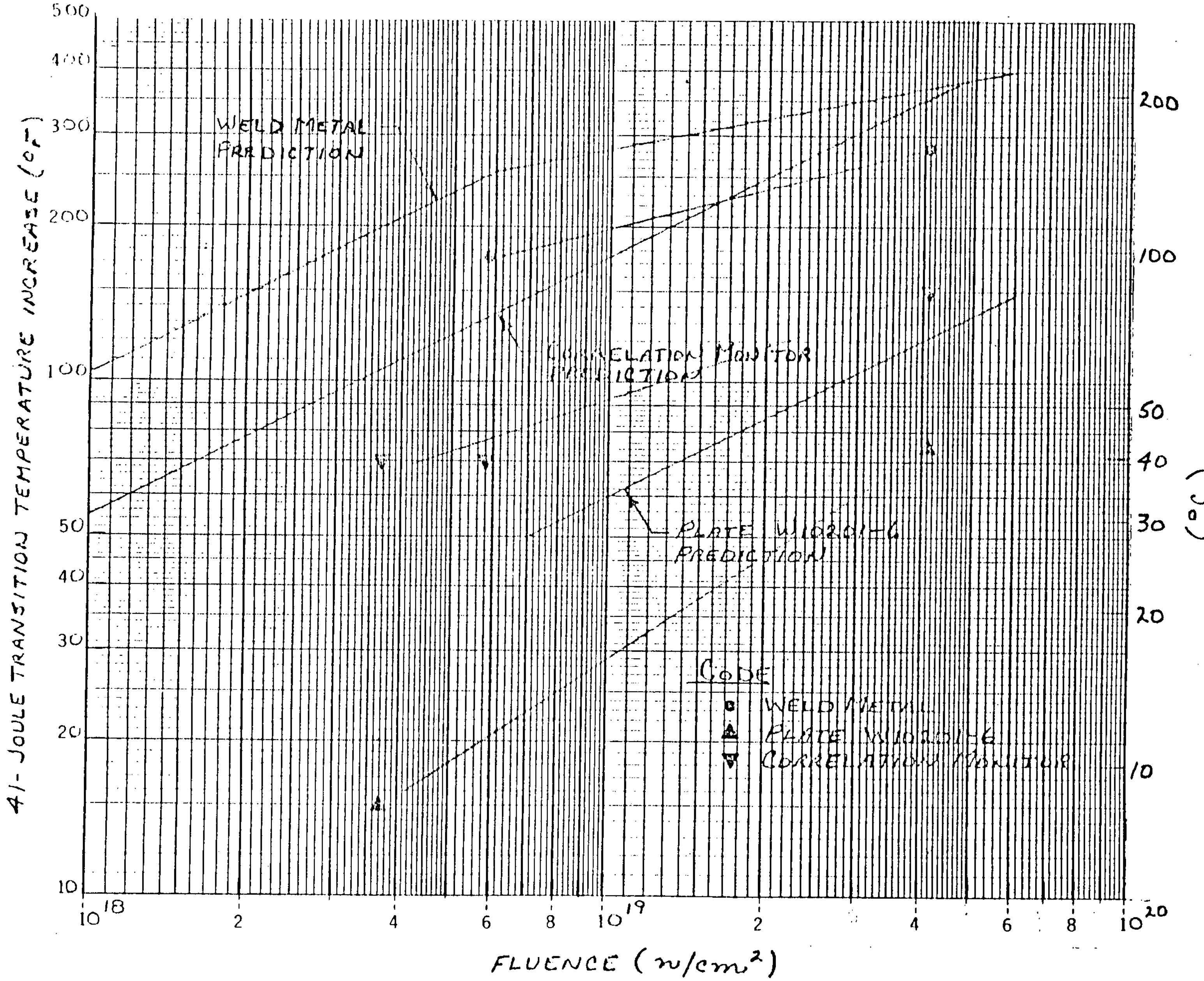


FIGURE 2-6 COMPARISON OF ACTUAL VERSUS PREDICTED 41-JOULE TRANSITION TEMPERATURE INCREASE FOR H.B. ROBINSON UNIT 2 REACTOR VESSEL MATERIALS USING THE PREDICTION METHODS OF REGULATORY GUIDE 1.99 REVISION 1

Table 2 - 1

Results of Fast Neutron Dosimetry for Capsule T

REACTION	ADJUSTED SATURATED ACTIVITY (dps/gm)		ϕ (E > 1.0 Mev) (n/cm^2-sec)		Φ (E > 1.0 Mev) (n/cm^2)	
	MEASURED	CALCULATED	MEASURED	CALCULATED	MEASURED	CALCULATED
Fe^{54}(n,p) Mn^{54}	8.87×10^6	7.96×10^6	1.81×10^{11}	1.68×10^{11}	4.11×10^{19}	3.81×10^{19}
Cu^{63}(n,α) Co^{60}	7.70×10^5	5.70×10^5	2.19×10^{11}		4.97×10^{19}	
Ni^{58}(n,p) Co^{58}	1.24×10^8	1.16×10^8	1.77×10^{11}		4.02×10^{19}	
Np^{237}(n,f) Cs^{137}	8.72×10^7	7.15×10^7	2.05×10^{11}		4.65×10^{19}	
U^{238}(n,f) Cs^{137}	1.32×10^7	9.65×10^6	2.27×10^{11}		5.16×10^{19}	

NOTE: Irradiation Time 2.27×10^8 EFPS

Table 2 - 2

Summary of Neutron Dosimetry Results for Capsule T

BASIS	IRRADIATION TIME (EFPS)	Ø (E > 1.0 Mev) (n/cm^2 -sec)	Φ (E > 1.0 Mev) (n/cm^2)	LEAD FACTOR	VESSEL FLUENCE (n/cm^2)	CALCULATED VESSEL FLUENCE (n/cm^2)
$Fe^{54}(n,p)Mn^{54}$	2.27×10^8	1.81×10^{11}	4.11×10^{19}	2.63	1.56×10^{19}	1.45×10^{19}
Dosimeter Avg.	2.27×10^8	2.02×10^{11}	4.58×10^{19}	2.63	1.74×10^{19}	1.45×10^{19}

Attachment 3

Vessel Weld Material Properties

During a meeting held with the NRC on November 12, 1982, a presentation was made with the hypothesis that the H. B. Robinson vessel welds contained less copper than previously assumed. The basis for the hypothesis was a comparison of the known copper and nickel levels of several welds made in the same time frame and with similar materials as the Robinson welds. Carolina Power & Light Company is continuing to evaluate that hypothesis and to gather additional information. As of this date, there is still insufficient evidence to draw a final conclusion with regard to that hypothesis.

Carolina Power & Light Company, however, is continuing to evaluate other areas associated with material properties. These areas are discussed below:

Confirmation of Fast Neutron Fluence Calculations

Results of surveillance tests at H. B. Robinson Unit No. 2 and two sister plants have been plotted in Figure 3-1 to provide a comparison with the design fluence calculation. These comparisons show very good correlation with the design calculations and support the data presented in Attachment 1.

Comparison of Surveillance Results with Trend Curves

It is noted that RT_{NDT} for Capsule T is 85°F below the Reg. Guide 1.99 curve and much farther below the Guthrie trend curve for .34%Cu and 1.20%Ni. Capsule T weld test results when plotted with Capsule V results fall well below the .35% copper line for the high nickel plot given by P. N. Randall[3] of the NRC staff (Figure 3-2).

The H. B. Robinson data together with Maine Yankee and Turkey Point data lie close to the .35% copper line of Randall's low nickel plot[3]. Randall's slope of .15% appears more appropriate than .27% or .35% (Figure 3-3).

If one plots all of the data from Main Yankee, it is seen that it cannot be plotted as a straight line on log-log paper, but must be of transcendental or other character. The damage at high fluence appears to approach a limit (Figure 3-4).

Fundamental Research into Mechanism of Radiation Damage

A theory that radiation damage by copper is due to precipitation of copper atoms was published in 1972[1]. Fundamental research by U. S. Steel with a field ion microscope (FIM) and time of flight mass spectroscopy has proven the theory by analyzing atom layers to determine that agglomeration of copper atoms occurred while the matrix was depleted.[2]

The above work was performed on a weld with .20% copper and low (.18%) nickel irradiated to 2 X 10^{19} n/cm > 1 Mev. A weld sample from the H. B. Robinson surveillance capsule T with high nickel and copper which received a fluence of 4.1 x 10^{19} is now available for test. The FIM will be used to investigate the effect of nickel on copper and phosphorus precipitation and the behavior of nickel atoms alone. An additional sample will be thermally annealed at 850°F to find out if precipitates grow or redissolve. Precipitation theory predicts that the particle will grow, which reduces the yield strength and the embrittlement effect. Precipitation theory further predicts that the original embrittlement rate cannot be re-established after annealing and upon re-exposure to fast neutrons.

Conclusions

Carolina Power & Light Company believes that consideration of the mechanism of radiation embrittlement in the temperature, spectral, and power ranges of commercial power reactor operation together with most pertinent surveillance data will show that embrittlement is approaching a limit peculiar to particular steels and reactors even if the nickel content of the steel or weld is high. Sufficient time remains before H. B. Robinson Unit No. 2 arrives at the PTS screening criterion to develop convincing documentation that damage saturation occurs for the H. B. Robinson critical weld. Carolina Power & Light Company will provide further information on these subjects as it is developed.

REFERENCES:

1. Effect of Neutron Fluence on Steel Weld Metal for Reactor Vessels, S. P. Grant and E. Fortner, Metals Engineering Quaterly, August 1972, pp. 17-24.

2. FIM/Atom Probe Study of Irradiated Pressure Vessel Steels, M. K. Miller and S. S. Brenner, U. S. Steel Research Laboratory. To be published in Res Mechanica.

3. The Status of Trend Curves and Surveillance Results in U. S. NRC Regulatory Activities, P. N. Randall. Presented at IAEA Specialists Meeting on Irradiation Embrittlement, October 18, 1981.

4. Analysis of the Maine Yankee Reactor Vessel Second Accelerated Surveillance Capsule, WCAP9875, T. R. Mager, March 1981.

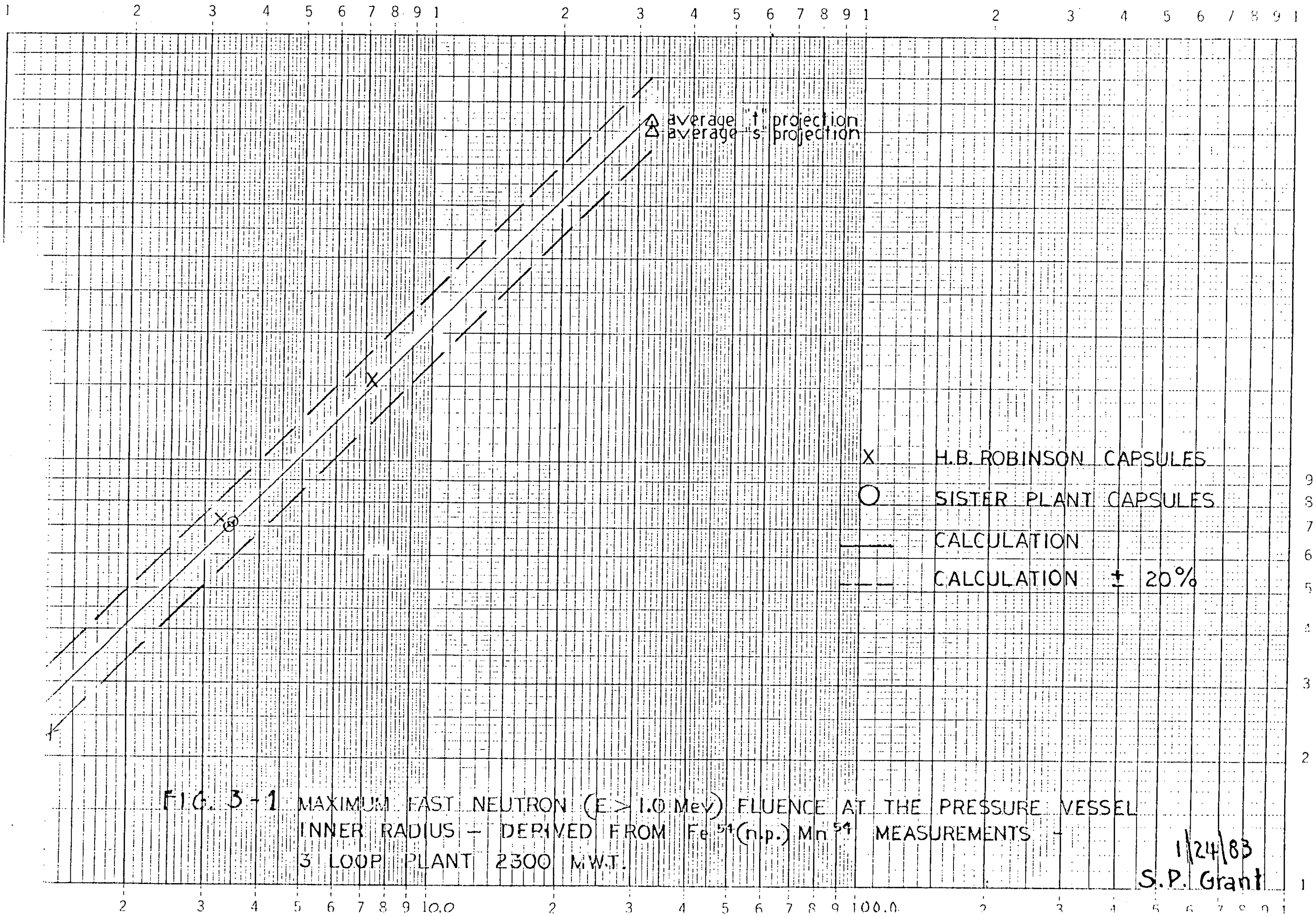


FIG. 3-1 MAXIMUM FAST NEUTRON (E > 1.0 Mev) FLUENCE AT THE PRESSURE VESSEL INNER RADIUS – DERIVED FROM $Fe^{54}(n,p) Mn^{54}$ MEASUREMENTS – 3 LOOP PLANT 2300 MWT.

1/24/83
S.P. Grant

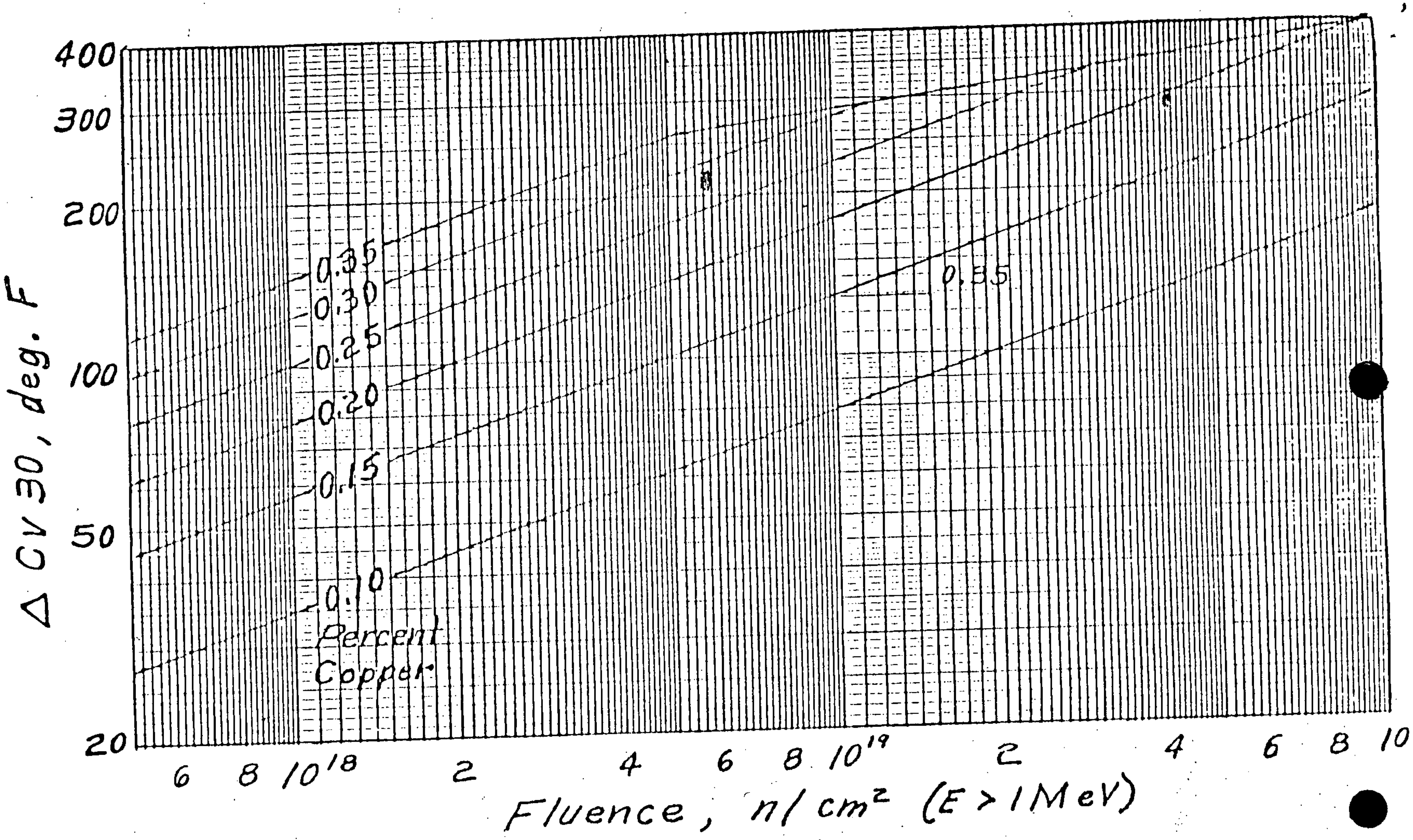


FIGURE 3-2 TENTATIVE TREND CURVES FOR HIGH-NICKEL MATERIALS (>0.5% Ni)

$$\Delta Cv30 = [30 + 1000\ (\%Cu - 0.05)]\ \frac{f^{0.35}}{10^{19}}$$

For % Cu $\leq$ 0.05, $\Delta Cv30 = \frac{f^{0.35}}{10^{19}}$

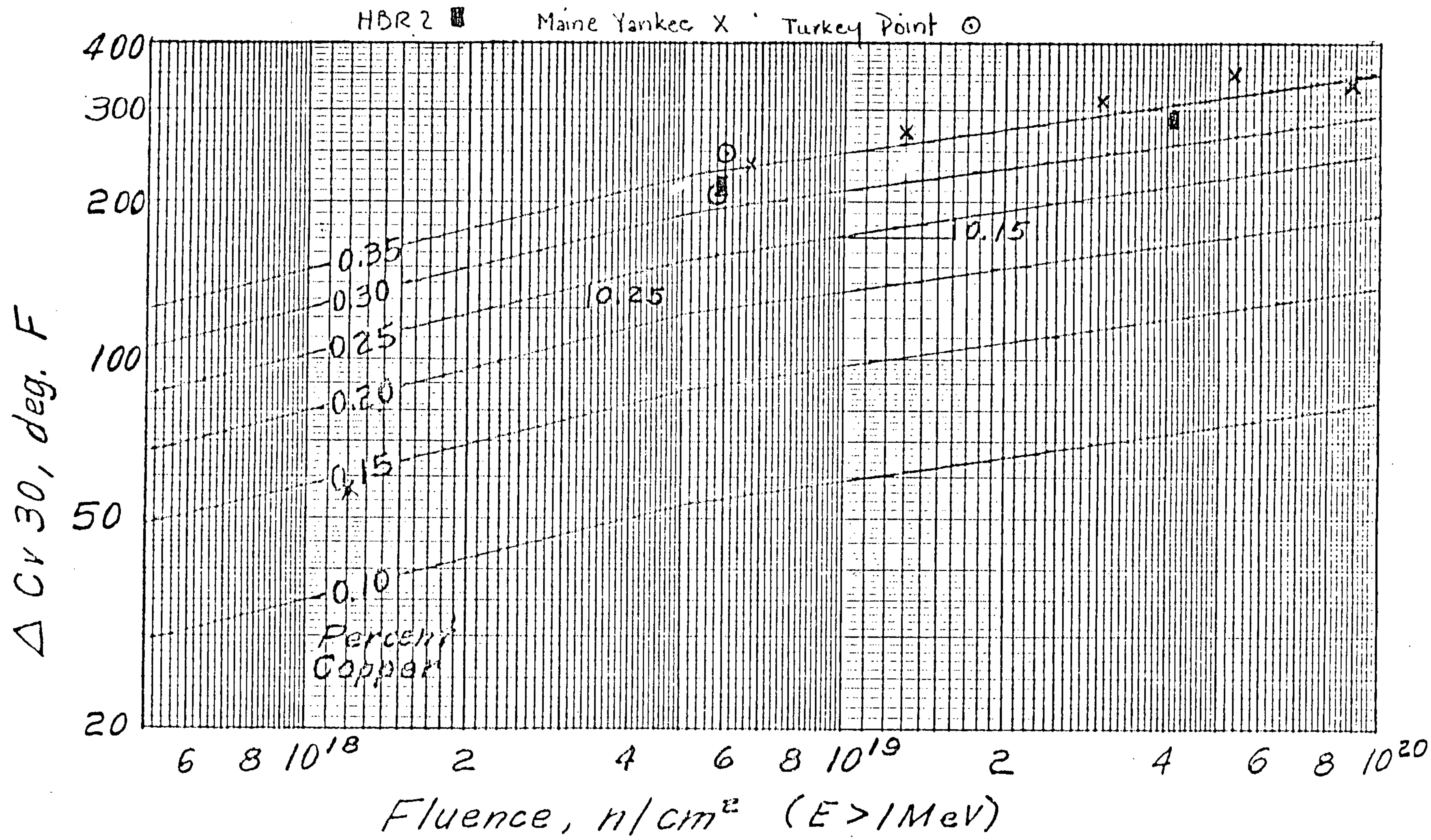


FIGURE 3-3 TENTATIVE TREND CURVES FOR LOW-NICKEL MATERIALS (< 0.2% Ni)

FLUENCE $\leq 5 \times 10^{18}$ n/cm^2
$$\Delta Cv30 = [23 + 800\,(Cu - 0.05)]\left(\frac{f}{10^{19}}\right)^{0.25}$$

FLUENCE $> 5 \times 10^{18}$ n/cm^2
$$\Delta Cv30 = [21 + 747\,(Cu - 0.05)]\left(\frac{f}{10^{19}}\right)^{0.15}$$

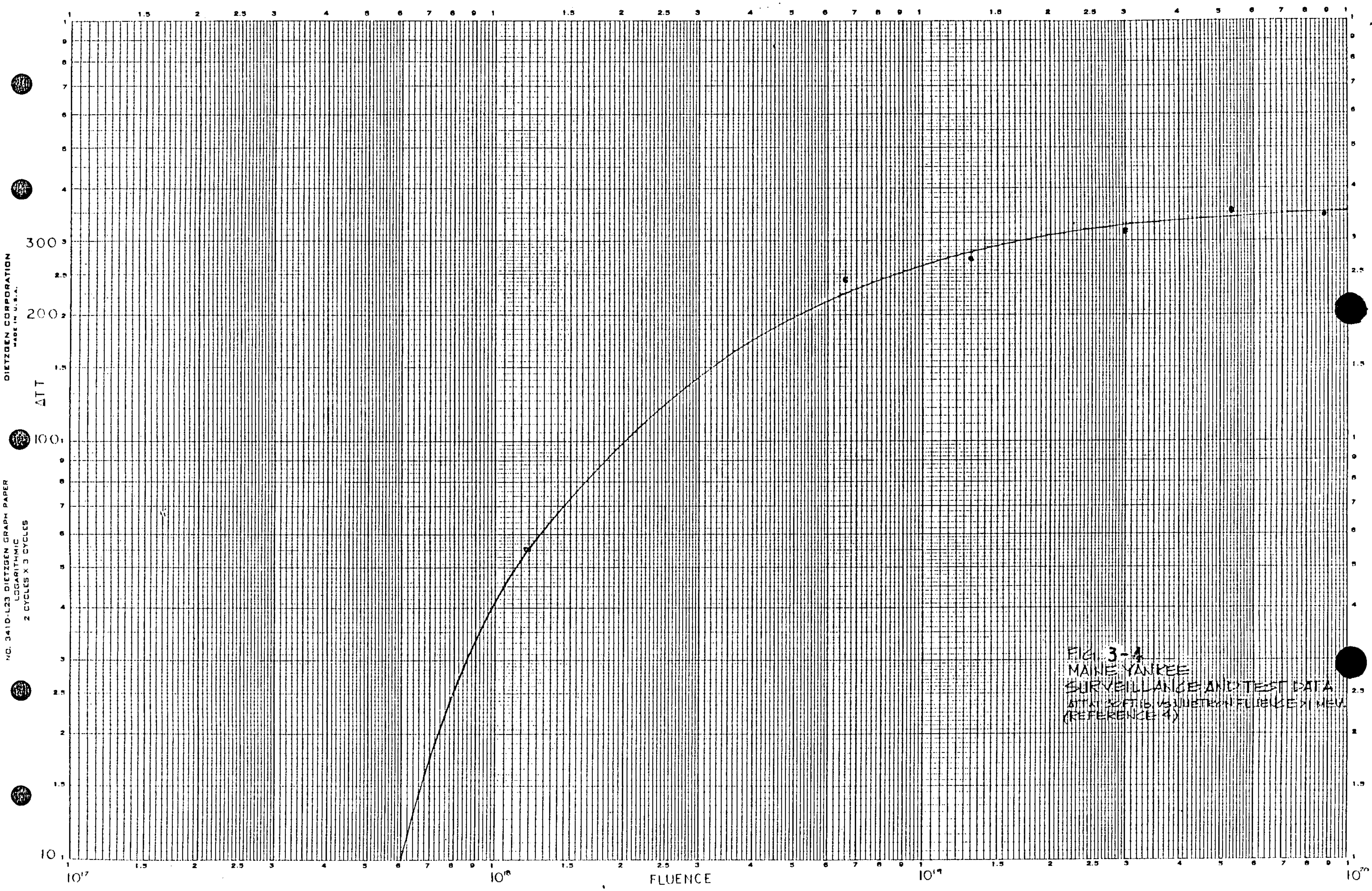


FIG. 3-4
MAINE YANKEE
SURVEILLANCE AND TEST DATA
ΔTT AT 50 FT-LB VS NEUTRON FLUENCE >1 MEV.
(REFERENCE 4)

Attachment 4
Inservice Inspection

In meetings held May 6, 1982 and November 12, 1982, CP&L discussed the results of the ten year Inservice Inspection conducted at H. B. Robinson during the spring of 1982. The inspection conducted was state-of-the-art and included a near surface examination and involved an extensive look at the Reactor Vessel beltline welds. No reportable indications were found in the beltline welds. Based on that inspection, CP&L is reviewing the possibility of establishment of a Robinson specific postulated flaw distribution curve for probabilistic fracture mechanics calculations. To do this, CP&L is examining with a number of vendors additional interpretive techniques for the data obtained during the inspection and will examine the original vessel construction nondestructive testing examination records. Insufficient data is presently available to draw a conclusion on the validity of proposing a plant specific flaw distribution to replace the presently assumed OCTAVIA or Marshall distributions. Carolina Power & Light Company will keep NRC appraised of its efforts in this area.

Attachment 5

Flux Reduction Techniques

The following discussion will explain CP&L's methods for analyzing reactor cores, review aspects of core design which relate to Pressurized Thermal Shock (PTS), and compare the costs and benefits of various flux reduction techniques.

Core Design

For the purpose of this discussion, core design is defined as the development of fuel assembly types and the corresponding loading pattern to meet the cycle design objectives. Therefore, core design in this discussion refers to a feasibility proof rather than a full-blown design effort. Carolina Power & Light Company methodology for designing core patterns for PTS concerns is to investigate promising candidate designs for the relative cost/benefit using diffusion theory and the approximation that flux at a point on the reactor vessel is proportional to that in the nearest fuel assemblies. Using this methodology, the most promising alternatives can then be evaluated more accurately using detailed transport theory and ultimately a complete operational design of the optimal design alternative can be initiated.

In utilizing the above methodology, CP&L and its fuel vendor, Exxon Nuclear Corporation (ENC), utilize the following codes and techniques:

XPIN: Transport theory code to generate group constants for strong absorbers such as gadolinia.

XPOSE: LEOPARD-based zero-dimensional spectral code using a modified MUFT/SOFOCATE model with ABH disadvantage factors generates most group constants.

XTGPWR: Nodal simulator (coarse-mesh "group-and-a-half" diffusion theory code) to perform two- and three-dimensional core calculations.

PDQ07: Two-group diffusion theory code to perform detailed pin-wise core calculations.

Transport theory analysis and calculations are usually contracted through an outside vendor as will be the case for the next low leakage core.

In developing a core design, a number of constraints must be satisfied. Satisfying all of the operational, safety, and economic considerations imposed by a utility, the NRC, and the Public Utilities Commission is not a trivial task. Some of these constraints are shown in Figure 5-1.

Additionally, it needs to be recognized that core calculations for low-leakage cores are intrinsically more difficult, costly and time consuming than old style, simpler out-in high-leakage cores. This is true because:

- Cross-section constants for exotic fuel (such as gadolinia) are more difficult to generate.

- Redetermination of boundary conditions and benchmarking may be required.

- Heavy burnable poison loadings make power peaking very complex and dynamic. Loading patterns may no longer be determined principally on the beginning of core state point, but must be examined throughout an entire cycle depletion as power peaks shift.

Robinson's most recent cycle design effort (Cycle 10) was expedited due to the existence of resources developed for the present core design (Cycle 9). These included:

- Cross section libraries for fuel and gadolinia were available.

- Because the flux distribution target for Cycle 10 was to be the same as Cycle 9 boundary conditions and benchmarks developed for Cycle 9 were still appropriate.

Even with these resources available, however, nodal simulator calculations for the loading pattern development alone involved scores of pattern alternatives and ten man-weeks of effort merely for the preliminary design.

In summary of the current generation of core designs, it may again be emphasized that:

- Complex and contradictory requirements are imposed by different constraints so that an optimal solution is a carefully balanced compromise.

- Even a feasibility study for a fairly typical low-leakage loading pattern will require man-months of effort and tens of thousands of dollars in computer charges.

- More exotic alternatives are even more difficult.

Carolina Power & Light Company, however, is expending these resources, and is dedicated to achieving additional flux reductions.

<u>PTS Considerations</u>

The concern over PTS centers on the consequences of fast neutron embrittlement of reactor vessel plate and weld materials. Although the accumulated fluence already present cannot be reduced, flux reduction is a useful method to reduce the rate of additional damage due to irradiation by fast neutrons, thereby buying additional time before the safety or usefulness of the reactor vessel becomes a concern.

Carolina Power & Light Company is committed to the use of flux reduction to aid in the resolution of PTS. This commitment is demonstrated by our previous flux reduction efforts. When the level of PTS concern deepened during the previous cycle (Cycle 8), CP&L immediately instituted flux reduction measures to the maximum extent possible without compromising plant operations. These efforts involved:

- An expedited program for low-leakage pattern development.

- Although the original Cycle 9 fuel was already built and delivered, gadolinia burnable poison fuel was procured. (CP&L is among the pioneers in PWR Gadolinia use).

- Twenty-eight fuel assemblies were shipped cross-country, disassembled, refabricated, and returned to Robinson.

- In the process, all of the margin to the $F_{\Delta H}$ thermal limit was sacrificed, all margin to the APDMS turn-on limit for F_{XY} was abandoned, and much of the margin to F_Q thermal limit was sacrificed.

These efforts produced a flux reduction factor of 2 in the current cycle (Cycle 9) which in turn has significantly lengthened the time available prior to H. B. Robinson reaching the generic screening criteria.

The loading pattern utilized in Cycle 9 is shown on Figure 5-2. A comparison of power distributions between the previous cycle and Cycle 9 is shown on Figure 5-3. The loading pattern presently designed for Cycle 10 is similiar to Cycle 9. A comparison of selected thermal limits for Cycles 8, 9, and 10 is shown in Figures 5-4 and 5-5.

Based on the above information and figures it can be seen that the present core design installed at H. B. Robinson has been very costly from a design effort and thermal limit viewpoint. Additional flux reductions, therefore, will have to provide extensive benefits to justify the expected increased costs of additional design efforts and sacrificing of margins to thermal limits. Additional considerations are described below.

Figure 5-6 shows the relative locations of the H. B. Robinson reactor vessel welds. The circumferential weld between the intermediate and lower shell is the critical weld for PTS concerns. Figure 5-7 shows a typical azimuthal distribution of fast neutron flux. As described earlier, the reactor vessel flux is affected primarily by the nearest fuel assemblies and the vessel flux from 0-20^{o} is dominated by the fuel on the flats of the core. In order to lower the peak flux on the vessel, the leakage on the flats should be restricted. This, however, may cause an increase in flux on the minor axis of the core. The goal should be, then, to levelize the flux seen by the vessel over plant life. From Figure 5-7 it can be seen that if flux could be decreased by five timeson the flats without substantially increasing flux on the minor axes, fluence at angles $> 20^{o}$ would not catch up to fluence from 0-20^{o} for 25 effective full power years. Therefore, unless the flux on the flats is reduced by five times or greater, there is little incentive to achieve flux reductions for the remaining peripheral fuel assemblies.

Flux Reduction Alternatives

In response to the PTS issue, CP&L has looked at a number of alternative loading patterns. The loading patterns considered are shown in Figure 5-8 through 5-14. Figure 5-15 summarizes the results of analyzing these patterns. As denoted on Figure 5-15, the cost of achieving a flux reduction in excess of a factor of 2 is large in terms of loss of cycle length and power derating of the plant. Figures 5-16 through 5-20 graphically show the cost payed in individual factors as the flux reduction is increased beyond a factor of 2. Therefore, to minimize these costs, relief in the area of thermal limits will be necessary in order to justify additional flux reductions. The relief required is described in Attachment 6.

H. B. ROBINSON UNIT 2

RELOAD DESIGN CONSIDERATIONS

(ALL ITEMS MUST BE CONSIDERED)

ITEM	PRIMARY EMPHASIS
* PEAKING FACTORS	SAFETY
* TEMPERATURE COEFFICIENT	SAFETY
* CONTROL ROD WORTH	SAFETY/COST-BENEFIT
* EXPOSURE LIMIT OF FUEL	SAFETY/COST-BENEFIT
* POWER CAPABILITY	COST-BENEFIT
* EFFICIENT FUEL UTILIZATION	COST-BENEFIT
* SPENT FUEL PIT	COST-BENEFIT
* CYCLE LENGTH	COST-BENEFIT
* SG TUBE DEGRADATION	SAFETY
* VESSEL FLUENCE	SAFETY

(Figure 5-1)

Figure 5-2

H.B. Robinson Unit 2 - Cycle 9

Full Core Loading Pattern

180°

R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
						3	2	3						
				0	0	0	0	0	0	0				
			0	1	1	2	2	2	1	1	0			
		0	0	1	2	1	2	1	2	1	2	0		
	0	1	1	2	1	2	0	2	1	2	1	1	0	
	0	1	2	1	2	1	2	1	2	1	2	1	0	
3	0	2	1	2	1	2	1	2	1	2	1	2	0	3
2	0	2	2	0	2	1	2	1	2	0	2	2	0	2
3	0	2	1	2	1	2	1	2	1	2	1	2	0	3
	0	1	2	1	2	1	2	1	2	1	2	1	0	
	0	1	1	2	1	2	0	2	1	2	1	1	0	
		0	0	1	2	1	2	1	2	1	2	0		
			0	1	1	2	2	2	1	1	0			
				0	0	0	0	0	0	0				
						3	2	3						

0°

90° 270°

N41 N43 N44 N42

A loop C loop B loop

0 NEW FUEL

1 ONCE BURNED

2 TWICE BURNED

3 THRICE BURNED

FIGURE 5-3

COMPARISON OF INCORE MEASURED RELATIVE POWERS

H.B.ROBINSON UNIT 2

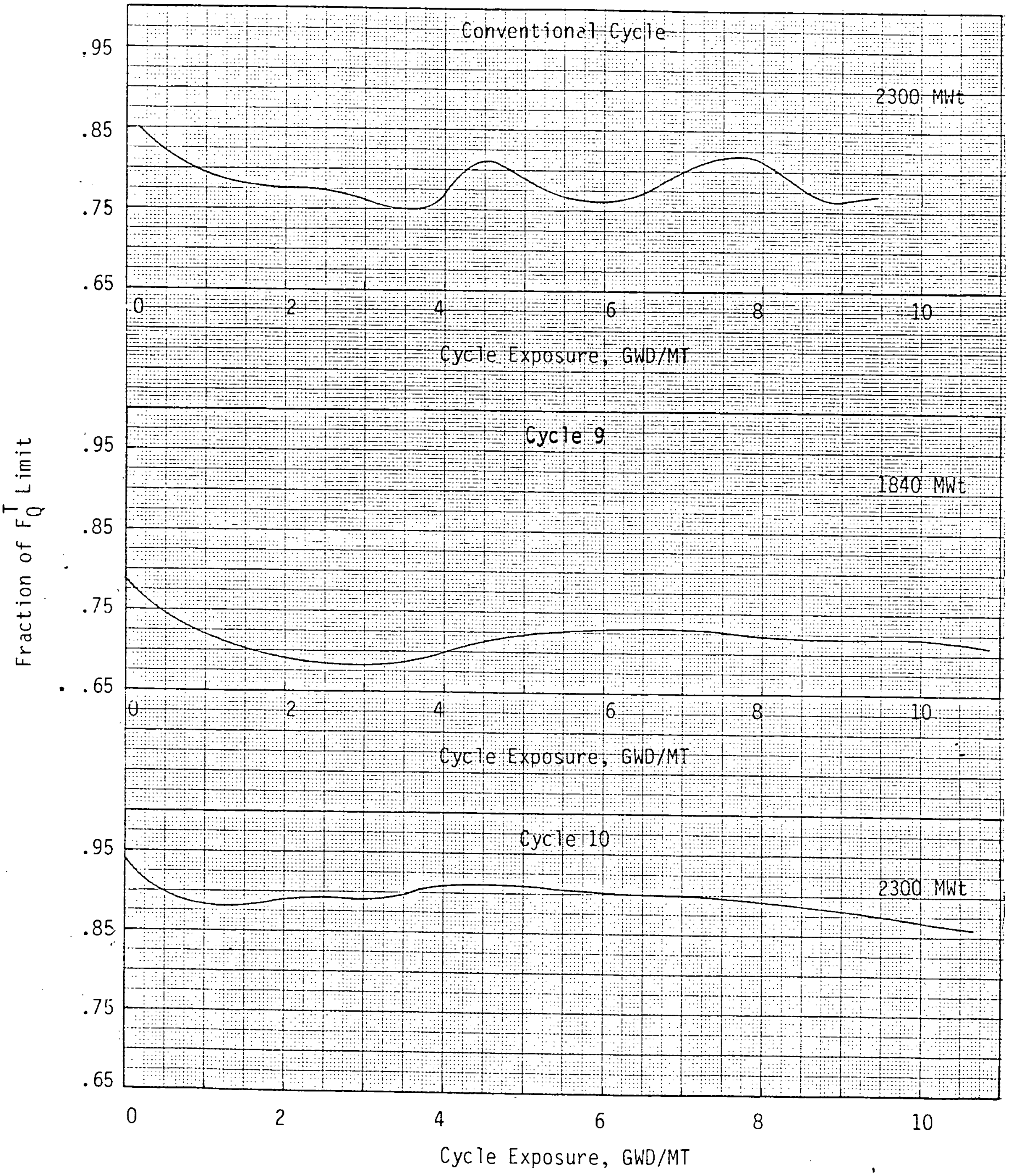
1.149	1.363	1.093	1.105	0.972	0.896	0.869	0.323
0.651	1.075	0.955	0.939	1.227	1.168	0.915	0.802
76.50	26.79	14.45	17.68	-20.78	-23.29	-5.03	-59.73
1.357	1.189	1.274	1.050	1.267	1.011	0.809	0.245
1.072	0.940	1.195	1.016	1.240	0.965	1.217	0.671
26.59	26.49	6.61	3.35	2.18	4.77	-33.53	-63.49
1.105	1.286	1.151	1.233	1.136	1.273	0.853	
0.947	1.192	0.981	1.229	0.988	1.118	1.025	
16.68	7.89	17.33	0.33	14.98	13.86	-16.78	
1.129	1.065	1.238	1.075	1.179	1.166	0.709	
0.924	1.002	1.221	1.118	0.934	1.178	0.711	
22.19	6.29	1.39	-3.85	26.23	-1.02	-0.28	
0.994	1.276	1.110	1.167	1.007	0.777		
1.193	1.210	0.979	0.928	0.902	0.712		
-16.68	5.45	13.38	25.75	11.64	9.13		
0.882	1.006	1.252	1.152	0.774			
1.125	0.934	1.100	1.167	0.710			
-21.60	7.71	13.82	-1.29	9.01			
0.858	0.806	0.853	0.702	MEASURED (A)			
0.877	1.166	0.995	0.702	MEASURED (B)			
-2.17	-30.87	-14.27	0.0	% DIFF=(A-B)/B*100			
0.344	0.369						
0.772	0.649						
-55.44	-43.14						

A = CYCLE 9 BOC MEASURED MAP413

B = CYCLE 8 BOC MEASURED MAP371

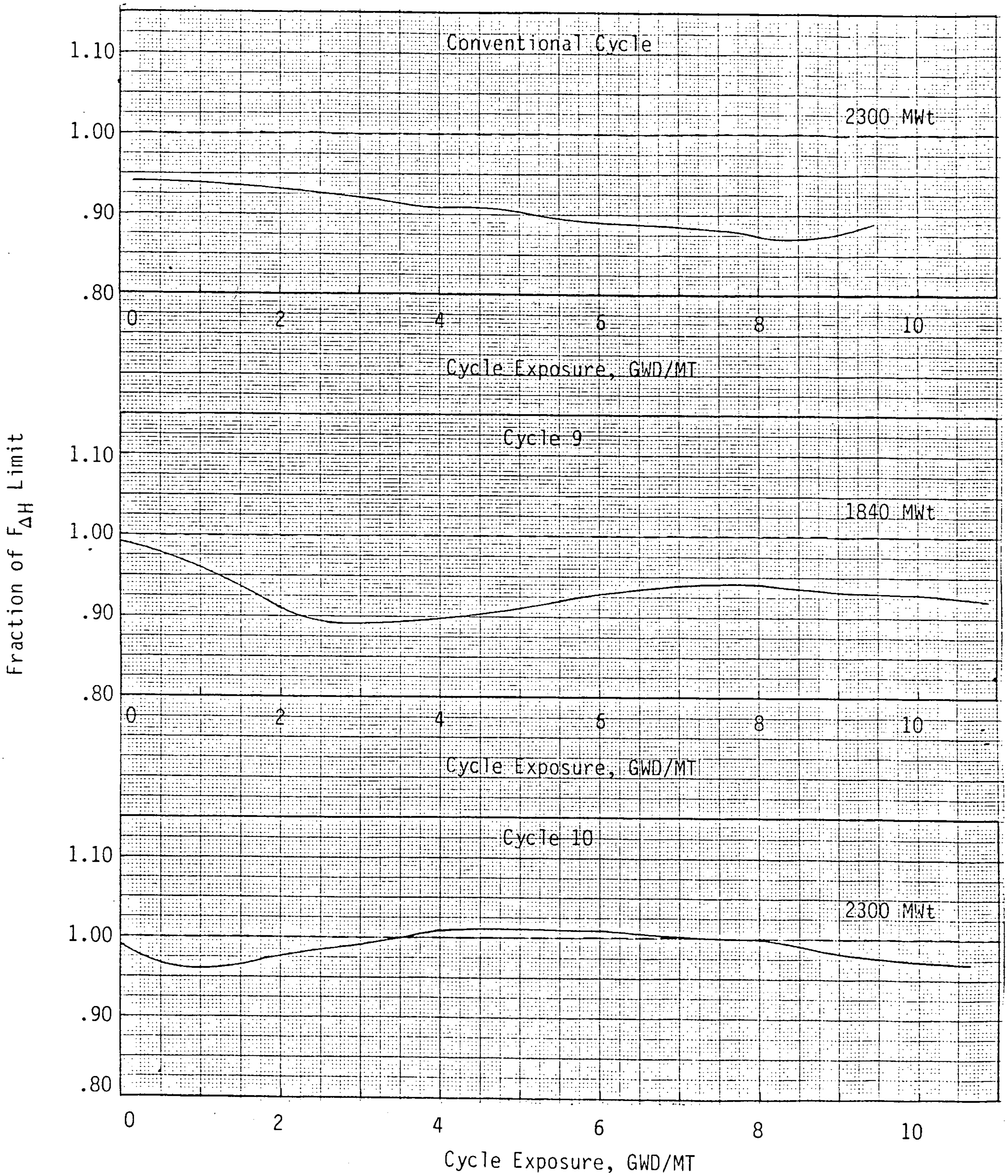
AVERAGE ABSOLUTE PER CENT DIFFERENCE =17.2%

Figure 5-4



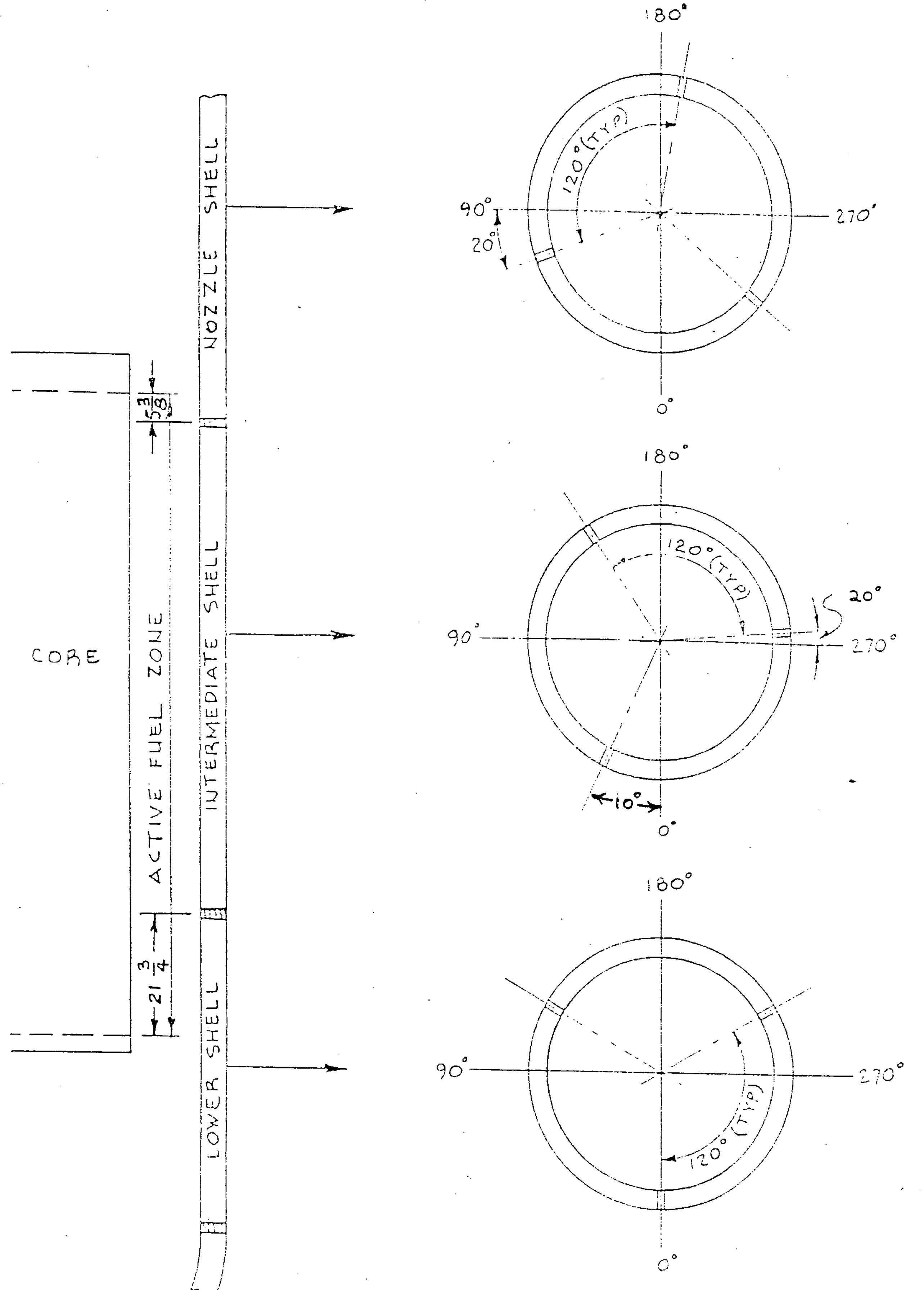
H. B. Robinson Unit 2, F_Q^T Limit vs. Cycle Exposure

Figure 5-5



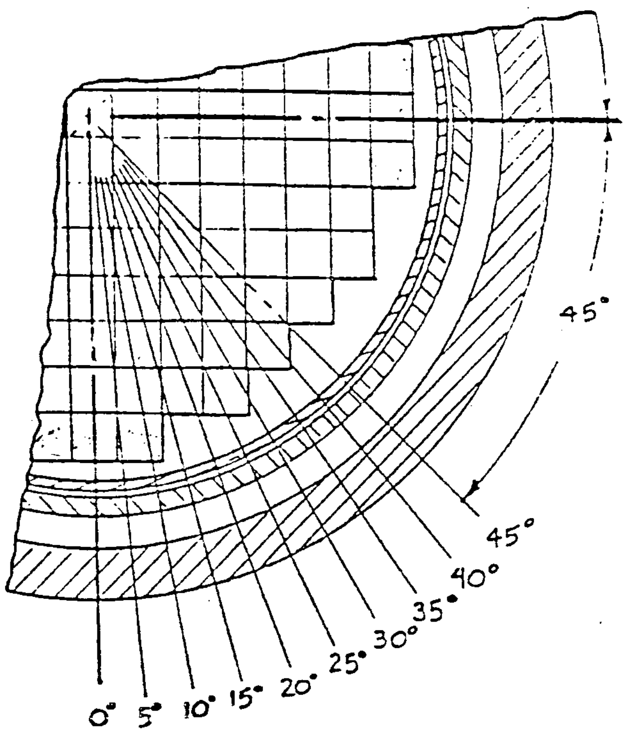
H. B. Robinson Unit 2, $F_{\Delta H}$ versus Cycle Exposure

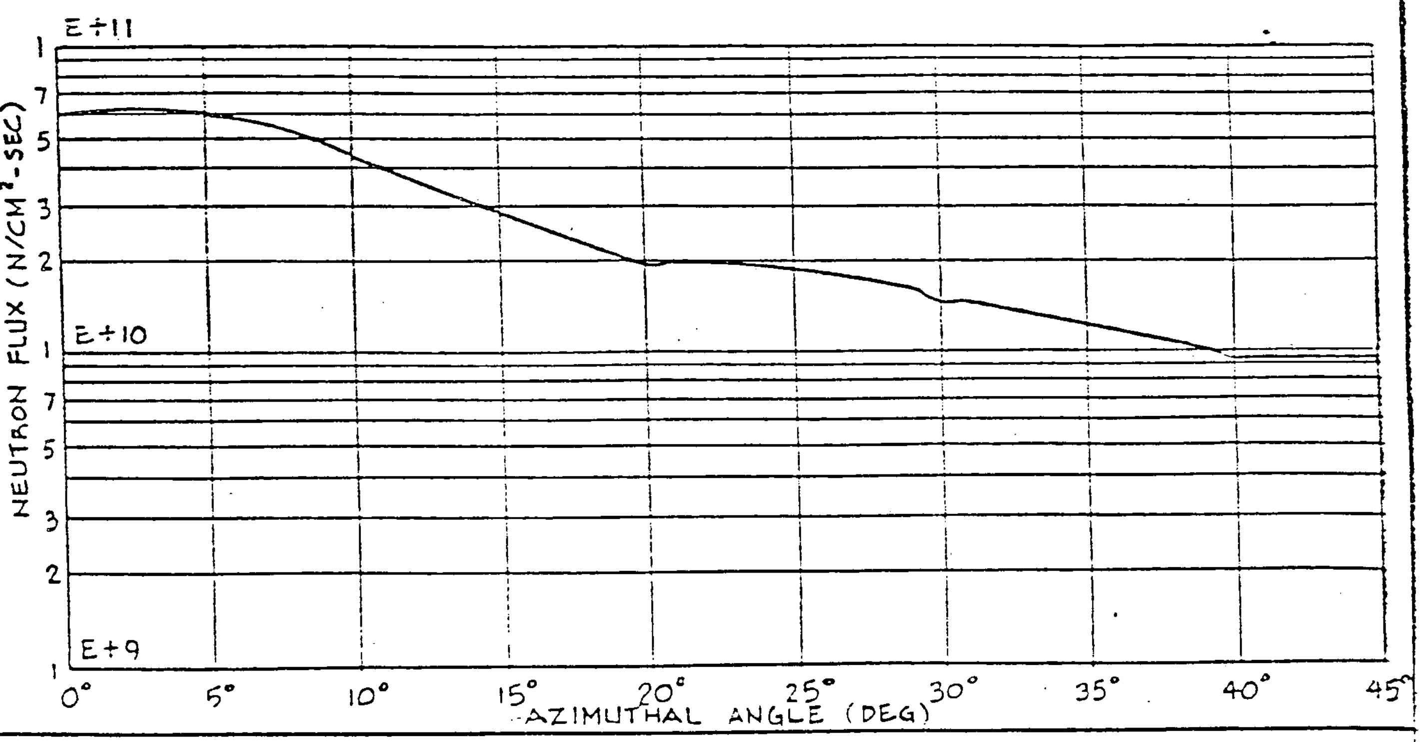
Figure 5-6



LOCATION OF WELDS FOR
THE H. B. ROBINSON UNIT 2
REACTOR VESSEL

Figure 5-7





AZIMUTHAL DISTRIBUTION OF
NEUTRON FLUX (N/CM^2-SEC)
AT REACTOR VESSEL INNER SURFACE

FIGURE 5-8

H. B. ROBINSON UNIT 2, CYCLE 8

TYPICAL EARLY HIGH LEAKAGE CORE

	H	G	F	E	D	C	B	A
8	1 4	2 9 1	3 17 2	4 24 2	5 31 1	6 37 1	7 42 2	8 46 0
9	9 1	10 2	11 18 1	12 25 2	13 32 1	14 38 2	15 43 0	16 47 0
10	17 2	18 1	19 2	20 26 1	21 33 2	22 39 1	23 44 0	
11	24 2	25 2	26 1	27 1	28 34 2	29 40 0	30 45 0	
12	31 1	32 1	33 2	34 2	35 1	36 41 0		
13	37 1	38 2	39 1	40 0	41 0			
14	42 2	43 0	44 0	45 0				
15	46 0	47 0						

FIGURE 5-9

H. B. ROBINSON UNIT 2, CYCLE 10

BASE CASE

	H	G	F	E	D	C	B	A
8	1 0	2 9 1	3 17 2	4 24 0	5 31 1	6 37 2	7 42 0	8 46 2
9	9 1	10 1	11 18 1	12 25 2	13 32 1	14 38 1	15 43 0	16 47 3
10	17 2	18 1	19 2	20 26 0	21 33 2	22 39 2	23 44 0	
11	24 0	25 2	26 0	27 2	28 34 1	29 40 0	30 45 0	
12	31 1	32 1	33 2	34 1	35 0	36 41 2		
13	37 2	38 1	39 2	40 0	41 2			
14	42 0	43 0	44 0	45 0				
15	46 2	47 3						

FIGURE 5-10

H. B. ROBINSON UNIT 2, CYCLE 10

ALTERNATIVE #1

	H	G	F	E	D	C	B	A
8	1 0	2 9 1	3 17 2	4 24 0	5 31 1	6 37 2	7 42 0	8 46 3
9	9 1	10 1	11 18 1	12 25 2	13 32 1	14 38 1	15 43 0	16 47 3
10	17 2	18 1	19 2	20 26 0	21 33 2	22 39 2	23 44 0	
11	24 0	25 2	26 0	27 2	28 34 1	29 40 0	30 45 0	
12	31 1	32 1	33 2	34 1	35 0	36 41 2		
13	37 2	38 1	39 2	40 0	41 2			
14	42 0	43 0	44 0	45 0				
15	46 3	47 3						

FIGURE 5-11

H. B. ROBINSON UNIT 2, CYCLE 10

ALTERNATIVE #2

	H	G	F	E	D	C	B	A
8	1 0	2 9 1	3 17 2	4 24 0	5 31 1	6 37 2	7 42 0	8 46 NATURAL UO_2
9	9 1	10 1	11 18 1	12 25 2	13 32 1	14 38 1	15 43 0	16 47 NATURAL UO_2
10	17 2	18 1	19 2	20 26 0	21 33 2	22 39 2	23 44 0	
11	24 0	25 2	26 0	27 2	28 34 1	29 40 0	30 45 0	
12	31 1	32 1	33 2	34 1	35 0	36 41 2		
13	37 2	38 1	39 2	40 0	41 2			
14	42 0	43 0	44 0	45 0				
15	46 NATURAL UO_2	47 NATURAL UO_2						

FIGURE 5-12

H. B. ROBINSON UNIT 2, CYCLE 10

ALTERNATIVE #3

	H	G	F	E	D	C	B	A
8	1 0	2 9 1	3 17 2	4 24 0	5 31 1	6 37 2	7 42 0	8 46 NATURAL UO_2
9	9 1	10 1	11 18 1	12 25 2	13 32 1	14 38 1	15 43 3	16 47 NATURAL UO_2
10	17 2	18 1	19 2	20 26 0	21 33 2	22 39 2	23 44 0	
11	24 0	25 2	26 0	27 2	28 34 1	29 40 0	30 45 0	
12	31 1	32 1	33 2	34 1	35 0	36 41 2		
13	37 2	38 1	39 2	40 0	41 2			
14	42 0	43 3	44 0	45 0				
15	46 NATURAL UO_2	47 NATURAL UO_2						

FIGURE 5-13

H. B. ROBINSON UNIT 2, CYCLE 10

ALTERNATIVE #4

	H	G	F	E	D	C	B	A
8	1 0	2 9 1	3 17 2	4 24 0	5 31 1	6 37 2	7 42 0	8 46 2 POISONED
9	9 1	10 1	11 18 1	12 25 2	13 32 1	14 38 1	15 43 0	16 47 3 POISONED
10	17 2	18 1	19 2	20 26 0	21 33 2	22 39 2	23 44 0	
11	24 0	25 2	26 0	27 2	28 34 1	29 40 0	30 45 0	
12	31 1	32 1	33 2	34 1	35 0	36 41 2		
13	37 2	38 1	39 2	40 0	41 2			
14	42 0	43 0	44 0	45 0				
15	46 2 POISONED	47 3 POISONED						

FIGURE 5-14

H. B. ROBINSON UNIT 2, CYCLE 10

ALTERNATIVE #5

	H	G	F	E	D	C	B	A
8	1 0	2 9 1	3 17 2	4 24 0	5 31 1	6 37 2	7 42 0	8 46 DUMMY SHIELD
9	9 1	10 1	11 18 1	12 25 2	13 32 1	14 38 1	15 43 0	16 47 DUMMY SHIELD
10	17 2	18 1	19 2	20 26 0	21 33 2	22 39 2	23 44 0	
11	24 0	25 2	26 0	27 2	28 34 1	29 40 0	30 45 0	
12	31 1	32 1	33 2	34 1	35 0	36 41 2		
13	37 2	38 1	39 2	40 0	41 2			
14	42 0	43 0	44 0	45 0				
15	46 DUMMY SHIELD	47 DUMMY SHIELD						

Figure 5-15

DEGREE OF FLUX REDUCTION VERSUS OPERATIONAL IMPACT

DESCRIPTION	(ILLUSTRATIVE FIGURE NO.)	RELATIVE FLUX AT REACTOR VESSEL	PEAKING FACTORS F_{XY}	F_Q	$F_{\Delta H}$	CYCLE LENGTH OF 2300 MWT OPERATION (EFPD)	LOSS IN CYCLE LENGTH (EFPD)	POWER DERATE CAUSED BY $F_{\Delta H}$
CYCLE 8	(FIG. 5-8)	1.00[1]	1.417	1.829	1.457	305	0	0
CYCLE 10								
BASE CASE	(FIG. 5-9)	0.50[2]	1.816	2.078	1.560	300	0	3.2%
ALTERNATIVE #1	(FIG. 5-10)	0.45[2]	1.817	2.073	1.570	298	2	6.5%
ALTERNATIVE #2	(FIG. 5-11)	0.32[2]	1.881	2.076	1.587	296	4	11.9%
ALTERNATIVE #3	(FIG. 5-12)	0.18[2]	2.073	2.323	1.772	262	38	71.6%
ALTERNATIVE #4	(FIG. 5-13)	0.20[2,3]	1.907	2.141	1.636	287	13	27.7%
ALTERNATIVE #5	(FIG. 5-14)	0.10[3]	1.967	2.198	1.681	279	21	42.3%

[1] CYCLE 8 IS TYPICAL OF THE ORIGINAL EQUILIBRIUM-CYCLE LOADINGS AT HBR2; HENCE, THE RELATIVE FLUX IS DEFINED TO BE 1.00 AS A BASIS FOR COMPARISON.

[2] DEFINED BY A COMPARISON OF THE RELATIVE POWER IN THE PERIPHERAL "FLATS" ASSEMBLY LOCATIONS.

[3] NRC ESTIMATE FROM "PRELIMINARY ASSESSMENT OF TECHNIQUES FOR FLUENCE RATE REDUCTION TO PWR PRESSURE VESSELS."

FIGURE 5-16

H. B. ROBINSON UNIT 2, CYCLE 10

FLUX REDUCTION IMPACT ON F_{XY}

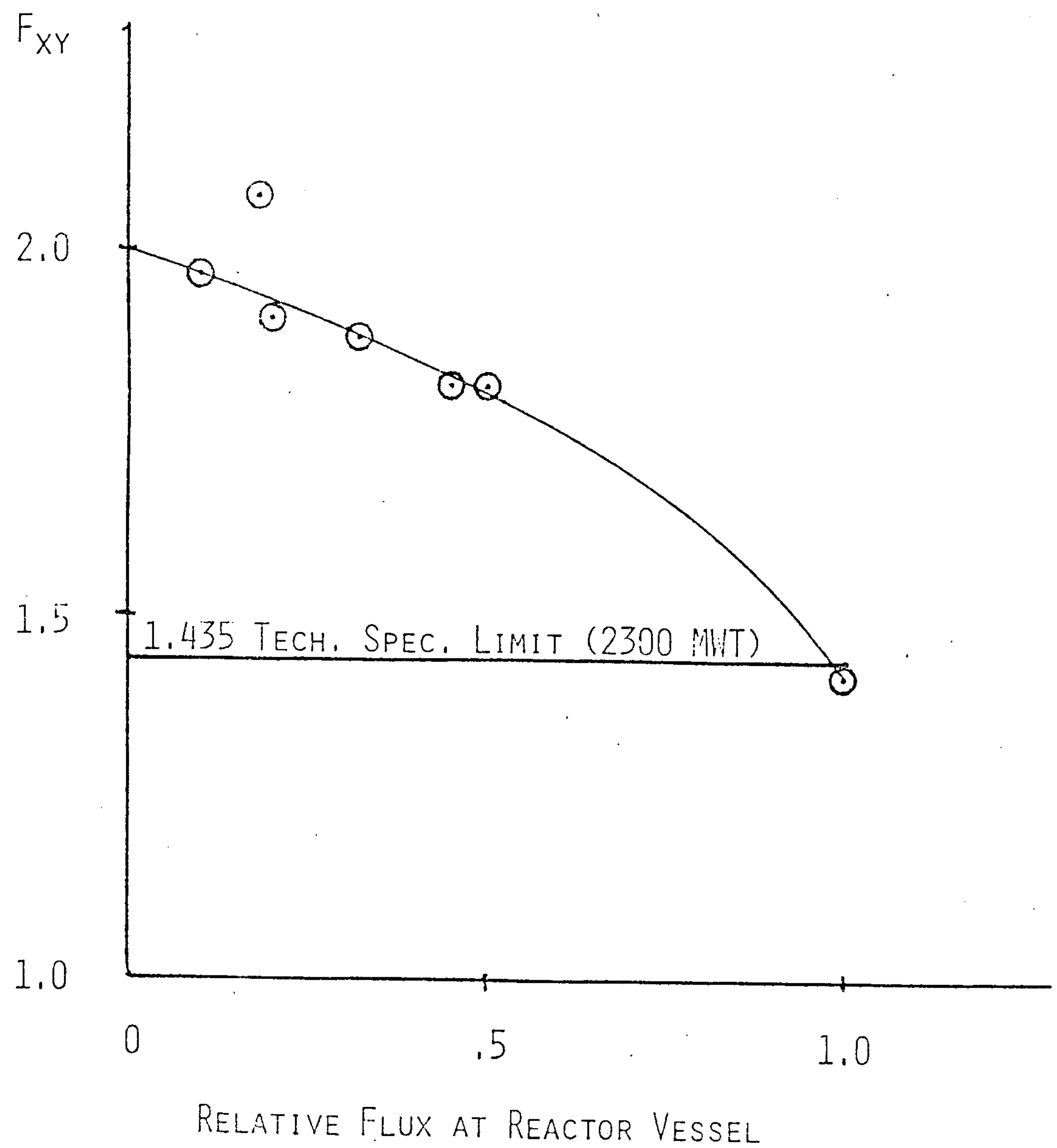


FIGURE 5-17

H. B. ROBINSON UNIT 2, CYCLE 10

FLUX REDUCTION IMPACT ON F_Q

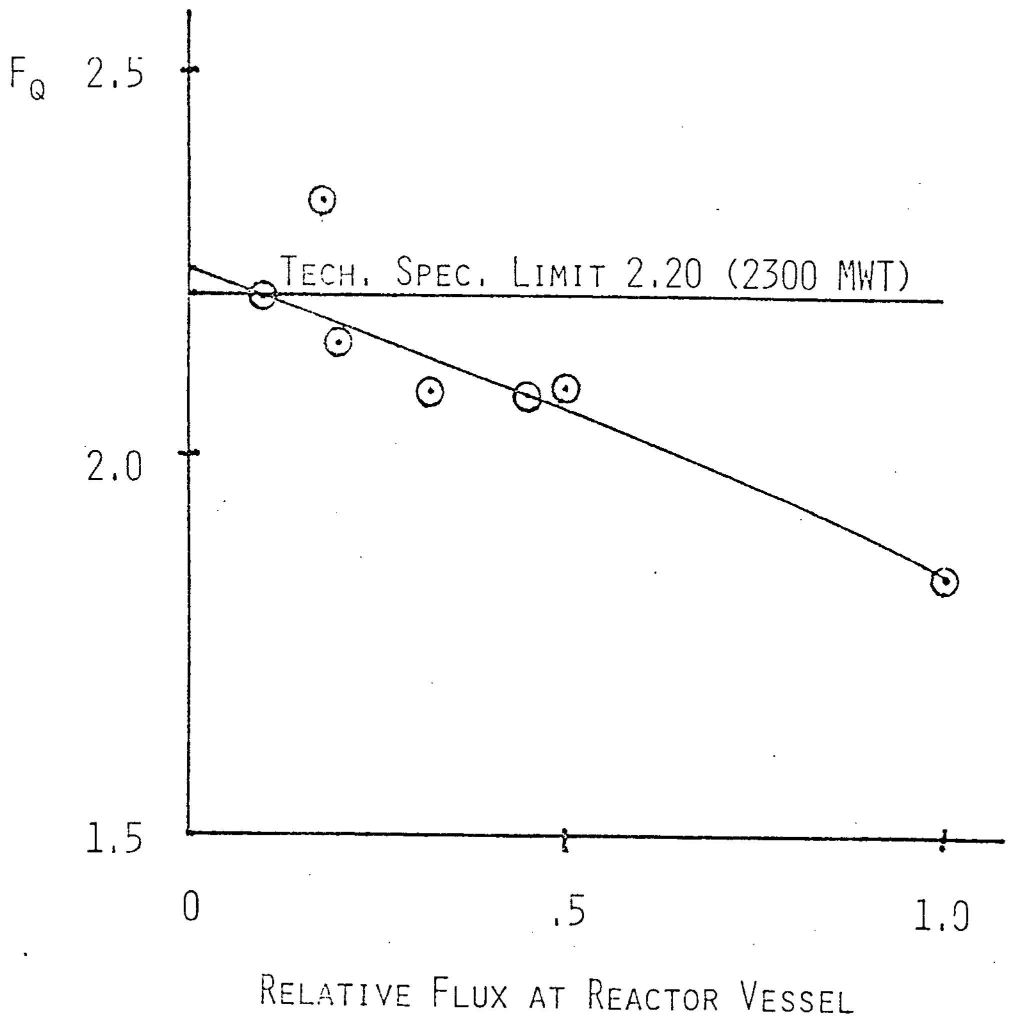


FIGURE 5-18

H. B. ROBINSON UNIT 2, CYCLE 10

FLUX REDUCTION IMPACT ON $F_{\Delta H}$

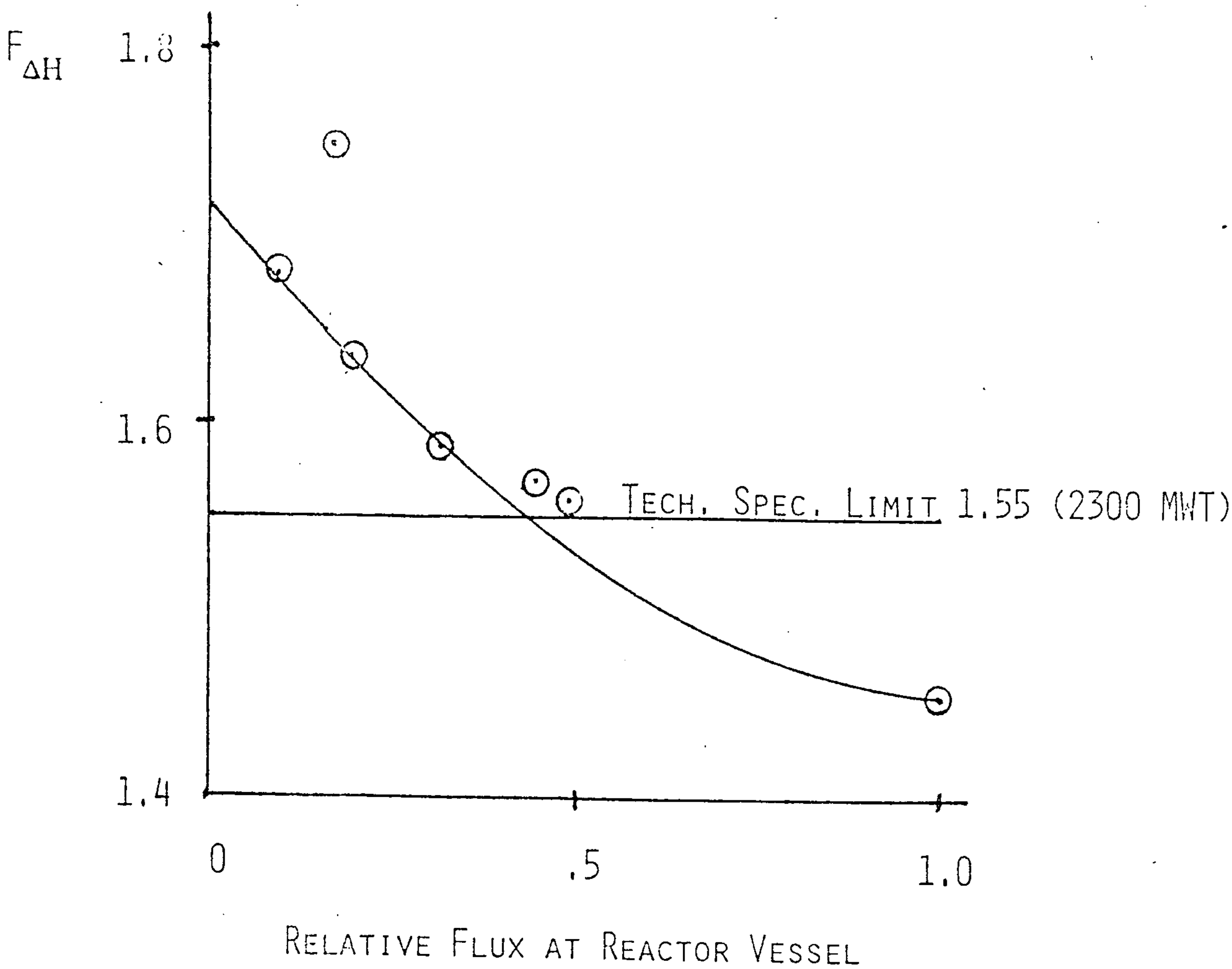


FIGURE 5-19

H. B. ROBINSON UNIT 2, CYCLE 10

FLUX REDUCTION IMPACT ON CYCLE LENGTHS

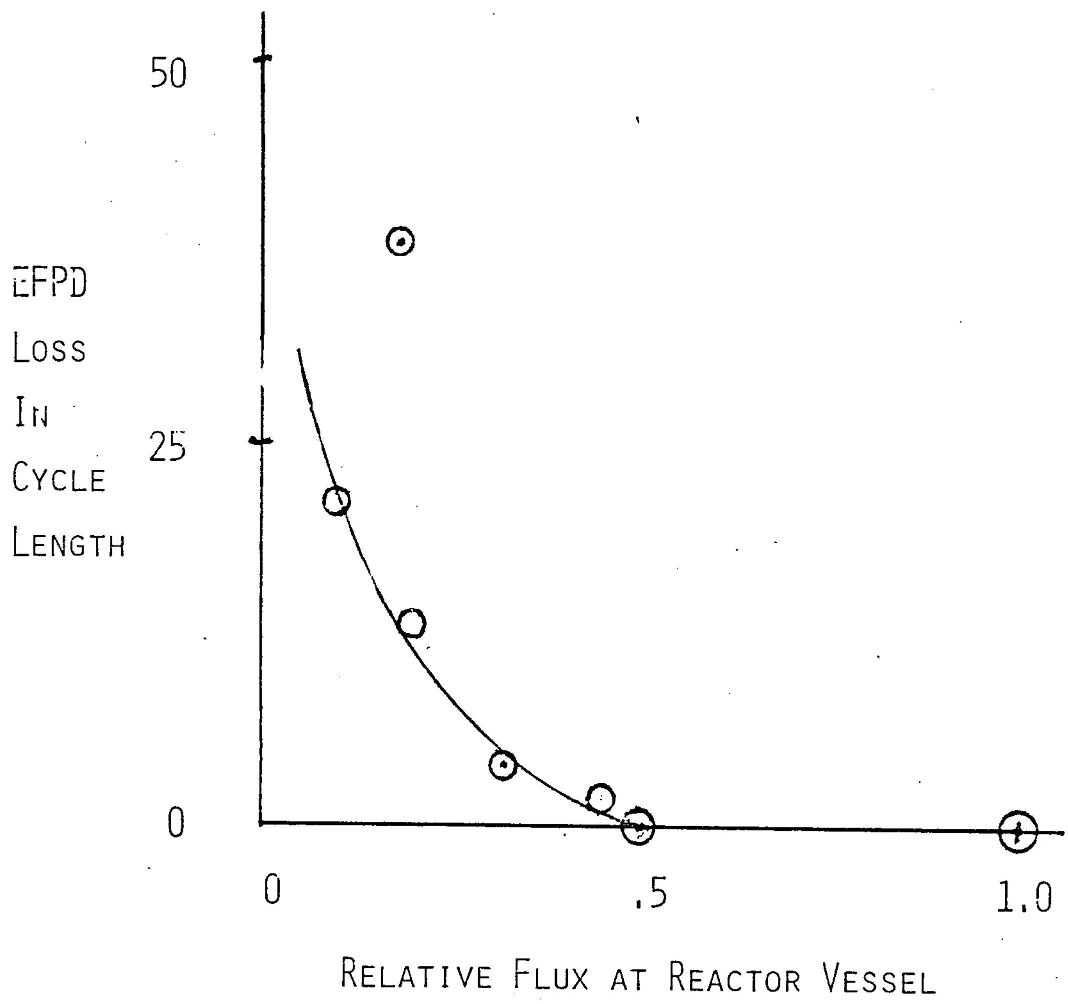
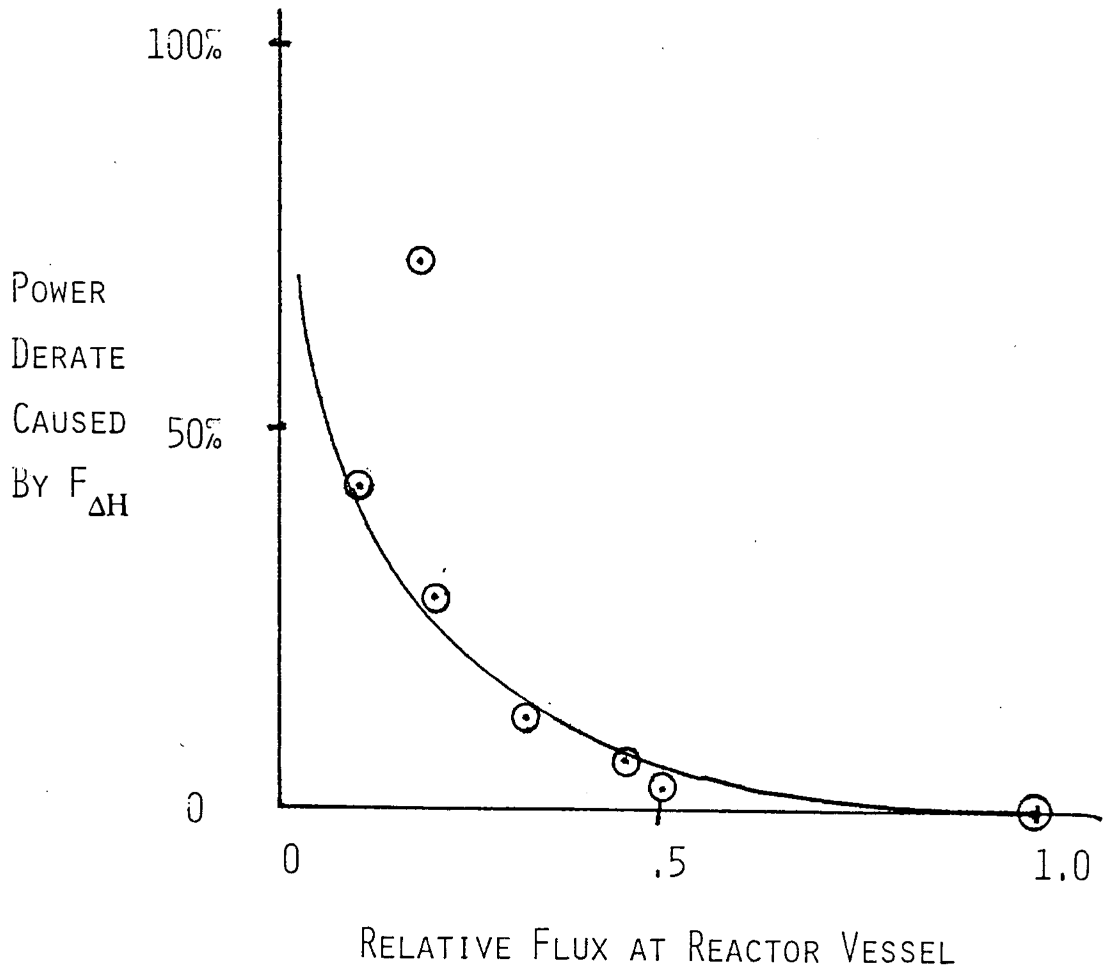


FIGURE 5-20

H. B. ROBINSON UNIT 2, CYCLE 10

FLUX REDUCTION IMPACT ON POWER



Attachment 6

Attainment of Further Flux Reduction

As described in the previous attachment, additional flux reductions for H. B. Robinson are presently limited by thermal limit restrictions. These limits presently are:

$$F_Q = 2.2 \text{ at } 2300 \text{ MW}_T$$
$$= 2.32 \text{ at } 1955 \text{ MW}_T \text{ (Low } T_{AVG} \text{ Program)}$$
$$F_{\Delta H} = 1.55$$

Based on our previous analysis, CP&L believes that significant additional flux reduction can be achieved if the thermal limits could be increased to the following after Steam Generator replacement:

$$F_Q = 2.3 \text{ at } 2300 \text{ MW}_T$$
$$F_{\Delta H} = 1.65 \text{ at } 2300 \text{ MW}_T$$

To achieve these target values, CP&L will provide to the NRC the necessary justification to allow the use of the following techniques in the next reload analysis (Cycle 10):

1) Evaluation of the spectrum of axial power distributions to reduce "Design Axial" conservatism ($+F_{\Delta H}$).

2) Reevaluation of setpoints and bandwidths to develop more certainty in initial conditions ($+F_{\Delta H}$).

3) Use of an improved rod bow model ($+F_{\Delta H}$).

4) Use of XCOBRA-IIIC Methodology with Automated Crossflow Boundary Conditions ($+F_{\Delta H}$).

5) Use of excess DNB Margin ($+F_{\Delta H}$).

6) Use of reworked analysis with most current models ($+F_Q$) to trade off F_Q for a new higher $F_{\Delta H}$

7) Modification of Technical Specifications to eliminate the 5 to 1 power loss for excess power ($+F_{\Delta H}$).

Additionally, in the future (beyond Cycle 10), CP&L intends to discuss with the NRC the use of the following techniques to gain additional margin to allow for greater flux reduction and operational flexibility:

1) Use of statistical combination of uncertainties of plant parameters for transient MDNBR evaluations ($+F_{\Delta H}$).

2) Use of a more realistic reflood model with deentrainmention in reactor vessel upper plenum ($+F_Q$)

3) Use of a more realistic refill model including heat transfer ($+F_Q$).

4) Use of a revised stored energy model (+F_Q).

5) Use of integrated reflood - containment pressure calculation (+F_Q).

6) Relaxation of use of ANS Standard +20% decay heat (+F_Q).

7) Relaxation of Baker-Just Metal-Water Reaction Relationship (+F_Q).

8) Use of Best-estimate blowdown heat transfer model (+F_Q).

9) Use of correlations and models that have strong data base that demonstrate validity (+F_Q and $F_{\Delta H}$).

In order to proceed with alternation of the core design of Cycle 10 to achieve greater flux reductions, CP&L needs to receive feedback from the NRC on the feasibility of utilizing the techniques described for Cycle 10 optimization. Carolina Power & Light Company therefore proposes the following schedule:

3/8/83	-	Meeting between NRC & CP&L wherein CP&L provides justification for the relief in thermal margins being sought for Cycle 10.
3/25/83	-	Carolina Power & Light Company documents the information provided in the Thermal Margin Meeting by a formal submittal on the docket.
5/6/83	-	NRC provides a written evaluation to CP&L of the acceptability of utilizing these techiques. Carolina Power & Light Company begins preparing detailed design and analysis for Cycle 10.
1/84	-	End of Cycle 9. Beginning of Steam Generator Outage.
3/84	-	Submittal of Detailed Reload Analysis for Cycle 10. Begin fuel fabrication.
6/84	-	Completion of NRC review of Cycle 10 Reload Analysis and issuance of SER.
9/84	-	Begin fuel load for Cycle 10.
10/84	-	Startup for Cycle 10.

Therefore, in summary, CP&L is dedicated to the attainment of additional flux reductions beyond the current factor of 2. To do so requires a cooperative effort between the NRC and CP&L. To make that effort successful, commitment to the above schedule and assignment of sufficient NRC resources to attain it are essential. Carolina Power & Light is already committed to this effort and seeks an equal commitment from the NRC.