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May 16, 1986

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Director of Nuclear Reactor, Region 1
U. S. Nuclear Regulatory Commission
Attn: Mr. Peter S. Tam, Project Manager
Project Directorate No. 2
Division of PWR Licensing - A
Washington, DC 20555
- Mail Stop 340 -

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License NO. DPR-66
Reactor Vessel Neutron Fluence Additional Information

Gentlemen:

Enclosed is our response to your request for additional information dated April 16, 1986 regarding reactor vessel neutron fluence estimates. Forty (40) copies of this response are being provided to the Document Management Branch in accordance with Generic Letter 84-18. Each request is listed and is followed by our response which references the applicable Section, Figure or Table included in the attachment.

1. Request - List the assumptions in the computer codes and the data used in the derivation of those values.

   Response: Section II.1: Method of Analysis, Appendix A: Power Distributions

2. Request - Give the azimuthal distribution of the ($E \geq 1.0$Mev) neutron fluence to the inner surface of the pressure vessel, present and at the end of life.

   Response: Section II.2: Fast Neutron Fluence Results, Figure II.2-3: Maximum Fast Neutron ($E > 1$Mev) Fluence at the Pressure Vessel Inner Radius as a Function of Azimuthal Angle.

3. Request - Have core power distribution from the previous cycles been used in the evaluation of the present fluence.

   Response: Yes, Appendix A: Power Distributions

4. Request - Was a benchmarked code used for the estimation of the fluence?

   Response: Yes, Section II.1: Method of Analysis

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5. Request - Have the fluence values been compared to the results of surveillance capsules?

   Response: Yes, Section II.2: Fast Neutron Fluence Results, Tables II.2-5 and II.2-6

6. Request - List the assumptions for future power distributions.

   Response: Section II.2: Fast Neutron Fluence Results

If you have any questions concerning this matter, please contact my office.

Very truly yours,

J. J. Carey

J. J. Carey

Attachment

cc: W. M. Troskoski, Resident Inspector
U. S. Nuclear Regulatory Commission
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SECTION II

NEUTRON EXPOSURE EVALUATION

II.1 METHOD OF ANALYSIS

A plan view of the Beaver Valley Unit 1 reactor geometry at the core midplane is shown in Figure II.1-1. Since the reactor exhibits 1/8th core symmetry only a 0°-45° sector is depicted. Eight irradiation capsules attached to the thermal shield are included in the design to constitute the reactor vessel surveillance program. The capsules are located at 45°, 55°, 65°, 165°, 245°, 285°, 295° and 305° relative to the reactor geometry flat at 0°.

A plan view of a single surveillance capsule attached to the thermal shield is shown in Figure II.1-2. The stainless steel specimen container is 1-inch square and approximately 40 inches in height. The containers are positioned axially such that the specimens are centered on the core midplane, thus spanning the central 3.33 feet of the 12-foot high reactor core.

Two sets of transport calculations were carried out in performing the fast neutron exposure evaluations for the reactor geometry shown in Figures II.1-1 and II.1-2. The first, a single computation in the conventional forward mode, was utilized to provide baseline data derived from a design basis core power distribution against which cycle by cycle plant specific calculations can be compared. The second set of calculations consisted of a series of adjoint analyses relating the response of interest at the surveillance capsules and several pressure vessel locations within the reactor geometry to the power distributions in the reactor core. These adjoint importance functions when combined with cycle specific core power distributions yield the plant specific exposure data for each operating fuel cycle.

The forward transport calculation was carried out in R,θ geometry using the DOT discrete ordinates code [2] and the SAILOR cross-section library [3]. The SAILOR library is a 47 group, ENDF-B/IV based data set produced specifically

Figure II 1-1. Beaver Valley Unit 1 Reactor Geometry

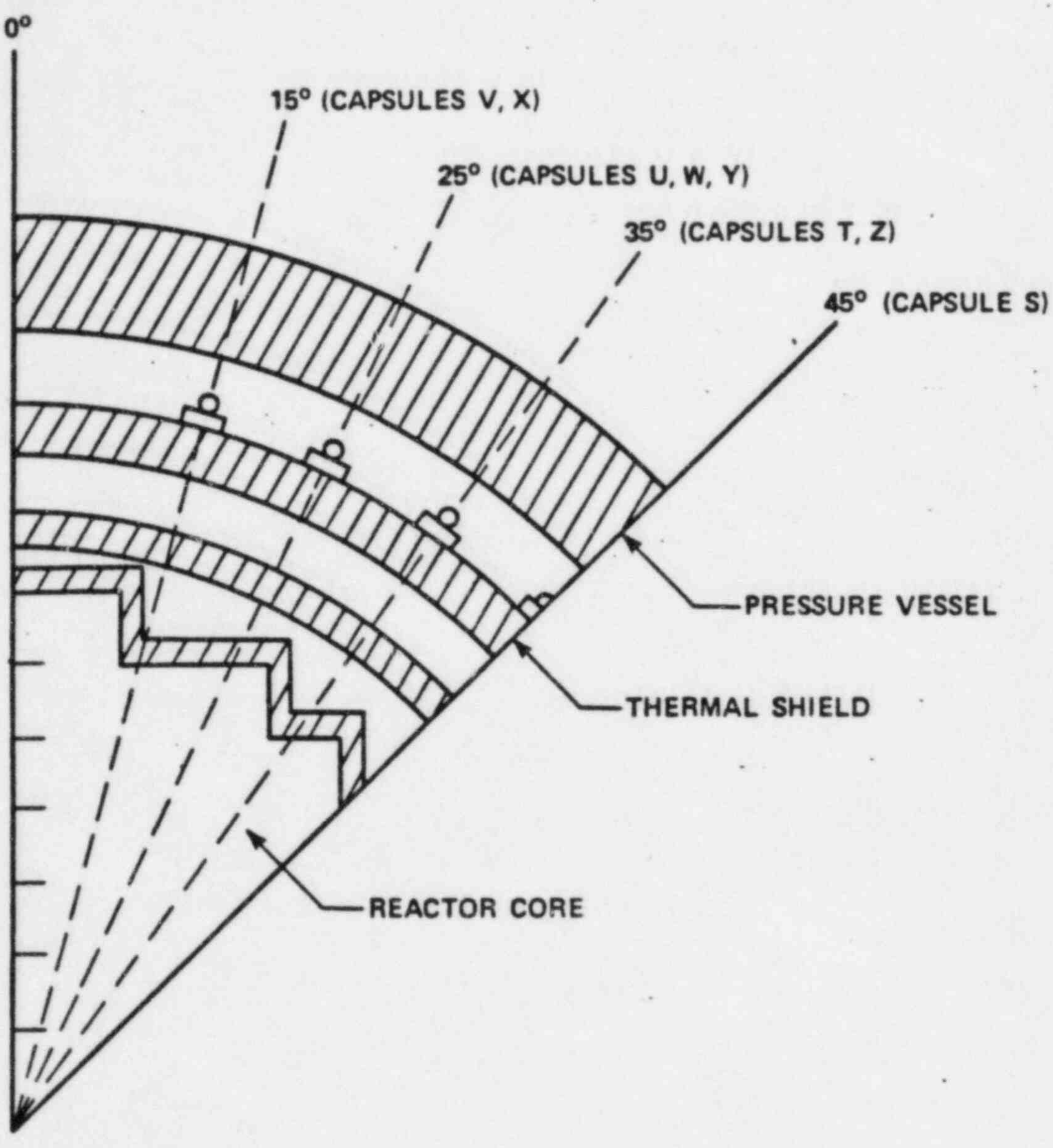
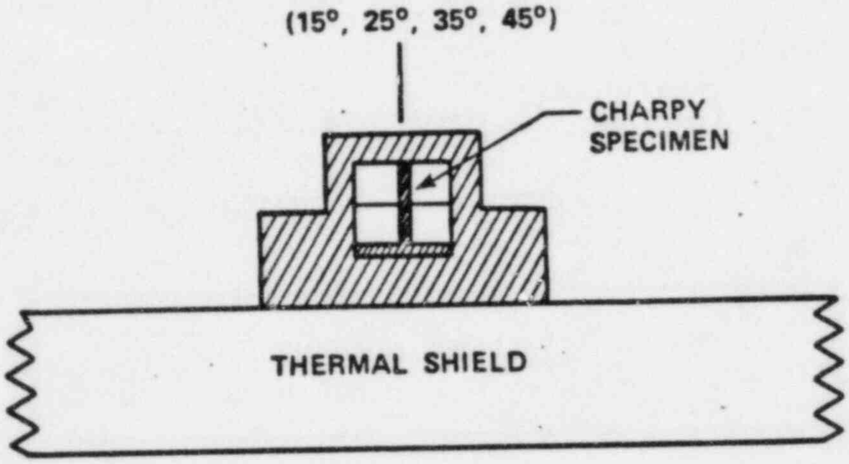


Figure II 1-2. Reactor Vessel Surveillance Capsule



for light water reactor applications. Anisotropic scattering is treated with a P_3 expansion of the cross-sections.

The design basis core power distribution utilized in the forward analysis was derived from statistical studies of long-term operation of Westinghouse 3-loop plants. Inherent in the development of this design basis core power distribution is the use of an out-in fuel management strategy; i.e., fresh fuel on the core periphery. Furthermore, for the peripheral fuel assemblies, a 2σ uncertainty derived from the statistical evaluation of plant to plant and cycle to cycle variations in peripheral power was used. Since it is unlikely that a single reactor would have a power distribution at the nominal $+2\sigma$ level for a large number of fuel cycles, the use of this design basis distribution is expected to yield somewhat conservative results. The design basis power distribution data used in the analysis is provided in Appendix A of this report. The data listed in Appendix A represents cycle averaged relative assembly powers.

The adjoint analyses were also carried out using the P_3 cross section approximation from the SAILOR library. Adjoint source locations were chosen at the center of each of the surveillance capsules as well as at positions along the inner diameter of the pressure vessel. Again, these calculations were run in R,θ geometry to provide power distribution importance functions for the exposure parameters of interest. Having the adjoint importance functions and appropriate core power distributions, the response of interest is calculated as

$$R_{R,\theta} = \int_R \int_\theta I(R,\theta)\, F(R,\theta)\, R\, dR\, d\theta$$

where:

$R_{R,\theta}$ = Response of interest (ϕ ($E > 1.0$ MeV), dPa, etc.) at radius R and azimuthal angle θ.

$I(R,\theta)$ = Adjoint importance function at radius R and azimuthal angle θ

$F(R,\theta)$ = Full power fission density at radius R and azimuthal angle θ

It should be noted that as written in the above equation, the importance function $I(R,\theta)$ represents an integral over the fission distribution so that the response of interest can be related directly to the spatial distribution of fission density within the reactor core.

Core power distributions for use in the Beaver Valley Unit 1 plant specific fluence evaluations were taken from fuel cycle design reports for each operating cycle to date. The specific power distribution data used in the analysis is provided in Appendix A of this report. The data listed in Appendix A represents cycle averaged relative assembly powers. Therefore, the adjoint results are in terms of fuel cycle averaged neutron flux which when multiplied by the fuel cycle length yields the incremental fast neutron fluence.

The transport methodology, both forward and adjoint, using the SAILOR cross-section library has been benchmarked against the ORNL PCA facility as well as against the Westinghouse power reactor surveillance capsule data base [4]. The benchmarking studies indicate that the use of SAILOR cross-sections and generic design basis power distributions produces flux levels that tend to be conservative by from 7-22%. When plant specific power distributions are used with the adjoint importance functions, the benchmarking studies show that fluence predictions are within $\pm$ 15% of measured values at surveillance capsule locations.

II.2 FAST NEUTRON FLUENCE RESULTS

Calculated fast neutron ($E > 1.0$ MeV) exposure results for Beaver Valley Unit 1 are presented in Tables II.2-1 through II.2-8 and in Figures II.2-1 through II.2-5. Data is presented at several azimuthal locations on the inner radius of the pressure vessel as well as at the center of each surveillance capsule. The fluence levels are based on a reactor thermal power level of 2652 MW.

Tables II.2-1 through II.2-4 list plant specific maximum neutron flux levels at 0°, 15°, 30°, and 45° on the pressure vessel inner radius for the first five operating cycles. Plant specific beltline cumulative fluence levels for each completed fuel cycle (1-4), and design basis cumulative fluence levels based on generic 3-loop core power distribution at the nominal + 2σ level are also presented for each completed fuel cycle. Similar data for the center of surveillance capsules located at 15°, 25°, 35°, and 45° are given in Tables II.2-5 through II.2-8, respectively. Measured fluence data from surveillance capsules withdrawn at the end of Cycle 1 and Cycle 4 are also presented for comparison with analytical results.

Several observations regarding the data presented in Tables II.2-1 through II.2-8 are worthy of note. These observations are summarized as follows:

1. Calculated plant specific fast neutron ($E > 1.0$ MeV) fluence level at the surveillance capsules are in good agreement with measured data. The maximum difference between the plant specific calculations and the measurements is approximately 11%. Differences of this magnitude are within the uncertainty of the experimental results.

2. The peak fast neutron ($E > 1.0$ MeV) flux incident on the pressure vessel (0° azimuthal position) during the fuel cycles using out-in fuel management (cycles 1-3) was, on the average, 12% less than predictions based on the design basis core power distributions.

3. Low leakage fuel management introduced following cycle 3 has reduced the average peak fast neutron ($E > 1.0$ MeV) flux on the pressure vessel by about 23% relative to that existing prior to implementation of low leakage.

TABLE II.2-1

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE VESSEL INNER RADIUS - 0° AZIMUTHAL ANGLE

Cycle No.	Irradiation Time (EFPS)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Plant Specific	Design Basis[(a)]
1	3.66×10^7	5.36×10^{10}	1.96×10^{18}	2.37×10^{18}
2	2.26×10^7	5.76×10^{10}	3.26×10^{18}	3.83×10^{18}
3	2.49×10^7	6.00×10^{10}	4.76×10^{18}	5.44×10^{18}
4	2.91×10^7	4.35×10^{10}	6.02×10^{18}	7.32×10^{18}
5	(b)	4.45×10^{10}	--	--

(a) Design Basis $\phi = 6.47 \times 10^{10}$ n/cm^2-sec

(b) Ongoing fuel cycle

TABLE II.2-2

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE VESSEL INNER RADIUS - 15° AZIMUTHAL ANGLE

Cycle No.	Irradiation Time (EFPS)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Plant Specific	Design Basis[(a)]
1	3.66×10^7	2.56×10^{10}	9.37×10^{17}	1.09×10^{18}
2	2.26×10^7	2.82×10^{10}	1.57×10^{18}	1.76×10^{18}
3	2.49×10^7	2.89×10^{10}	2.29×10^{18}	2.50×10^{18}
4	2.91×10^7	2.12×10^{10}	2.91×10^{18}	3.36×10^{18}
5	(b)	2.16×10^{10}	--	--

(a) Design Basis $\phi = 2.96 \times 10^{10}$ n/cm^2-sec

(b) Ongoing fuel cycle

TABLE II.2-3

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE VESSEL INNER RADIUS - 30° AZIMUTHAL ANGLE

Cycle No.	Irradiation Time (EFPS)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Plant Specific	Design Basis[(a)]
1	3.66×10^7	1.36×10^{10}	4.98×10^{17}	6.33×10^{17}
2	2.26×10^7	1.56×10^{10}	8.50×10^{17}	1.02×10^{18}
3	2.49×10^7	1.51×10^{10}	1.23×10^{18}	1.45×10^{18}
4	2.91×10^7	1.09×10^{10}	1.54×10^{18}	1.96×10^{18}
5	(b)	1.10×10^{10}	—	—

(a) Design Basis $\phi = 1.73 \times 10^{10}$ n/cm^2-sec

(b) Ongoing fuel cycle

TABLE II.2-4

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE VESSEL INNER RADIUS - 45° AZIMUTHAL ANGLE

Cycle No.	Irradiation Time (EFPS)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Plant Specific	Design Basis[(a)]
1	3.66×10^7	9.03×10^9	3.30×10^{17}	3.85×10^{17}
2	2.26×10^7	1.06×10^{10}	5.70×10^{17}	6.22×10^{17}
3	2.49×10^7	9.73×10^9	8.12×10^{17}	8.84×10^{17}
4	2.91×10^7	7.12×10^9	1.02×10^{18}	1.19×10^{18}
5	(b)	7.34×10^9	--	—

(a) Design Basis $\phi = 1.05 \times 10^{10}$ n/cm^2-sec

(b) Ongoing fuel cycle

TABLE II.2-5

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE 15° SURVEILLANCE CAPSULE CENTER

Cycle No.	Irradiation Time (EFPS)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Plant Specific	Design Basis[(a)]	Capsule "V Data
1	3.66×10^7	8.85×10^{10}	3.24×10^{18}	3.75×10^{18}	2.91×10^{18}
2	2.26×10^7	9.75×10^{10}	5.44×10^{18}	6.06×10^{18}	
3	2.49×10^7	9.97×10^{10}	7.93×10^{18}	8.61×10^{18}	
4	2.91×10^7	7.28×10^{10}	1.00×10^{19}	1.16×10^{19}	
5	(b)	7.44×10^{10}	--	--	

(a) Design Basis $\phi = 1.02 \times 10^{11}$ n/cm^2-sec

(b) Ongoing fuel cycle

TABLE II.2-6

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE 25° SURVEILLANCE CAPSULE CENTER

Cycle No.	Irradiation Time (EFPS)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Plant Specific	Design Basis[(a)]	Capsule "U Data
1	3.66×10^7	5.52×10^{10}	2.02×10^{18}	2.38×10^{18}	
2	2.26×10^7	6.25×10^{10}	3.43×10^{18}	3.85×10^{18}	
3	2.49×10^7	6.21×10^{10}	4.98×10^{18}	5.47×10^{18}	
4	2.91×10^7	4.55×10^{10}	6.30×10^{18}	7.36×10^{18}	6.54×10^{18}
5	(b)	4.56×10^{10}	--	--	

(a) Design Basis $\phi = 6.50 \times 10^{10}$ n/cm^2-sec

(b) Ongoing fuel cycle

TABLE II.2-7

FAST NELTRON (E > 1.0 MeV) EXPOSURE AT THE 35° SURVEILLANCE CAPSULE CENTER

Cycle No.	Irradiation Time (EFPS)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Plant Specific	Design Basis[(a)]
1	3.66×10^7	3.71×10^{10}	1.36×10^{18}	1.63×10^{18}
2	2.26×10^7	4.31×10^{10}	2.33×10^{18}	2.63×10^{18}
3	2.49×10^7	4.09×10^{10}	3.35×10^{18}	3.74×10^{18}
4	2.91×10^7	2.93×10^{10}	4.20×10^{18}	5.03×10^{18}
5	(b)	2.98×10^{10}	--	--

(a) Design Basis $\phi = 4.44 \times 10^{10}$ n/cm^2-sec

(b) Ongoing fuel cycle

TABLE II.2-8

FAST NEUTRON (E > 1.0 MeV) EXPUSURE AT THE 45° SURVEILLANCE CAPSULE CENTER

Cycle No.	Irradiation Time (EFPS)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Plant Specific	Design Basis[(a)]
1	3.66×10^7	2.91×10^{10}	1.07×10^{18}	1.29×10^{18}
2	2.26×10^7	3.42×10^{10}	1.84×10^{18}	2.08×10^{18}
3	2.49×10^7	3.13×10^{10}	2.62×10^{18}	2.95×10^{18}
4	2.91×10^7	2.27×10^{10}	3.28×10^{18}	3.98×10^{18}
5	(b)	2.35×10^{10}	--	--

(a) Design Basis $\phi = 3.51 \times 10^{10}$ n/cm^2-sec

(b) Ongoing fuel cycle

Graphical presentation of the plant specific fast neutron ($E > 1.0$ MeV) fluence at key locations on the pressure vessel and at the center of the surveillance capsules are shown in Figures II.2-1 and II-2-2 as a function of full power operating time. Pressure vessel data is presented for the 0° azimuthal location on the circumferential weld and for the beltline region on the longitudinal welds. Surveillance capsule data is presented for the 15°, 25°, 35° and 45° locations.

The solid portions of the fluence curves in Figures II.2-1 and II.2-2 are based directly on the cycles 1-4 plant specific evaluations presented in this report. The dashed portions of these curves however involve a projection into the future. Since Beaver Valley Unit 1 has been committed to a consistent form of low leakage fuel management for several cycles, the average neutron flux at the key locations over the low leakage fuel cycles was used for all temporal projections. In particular, the neutron flux average over cycles 4 and 5 was used to project future fluence levels for Unit 1.

It should be noted that implementation of a more severe low leakage pattern would act to reduce the projections of fluence at key locations. On the other hand, relaxation of the current low leakage patterns or a return to out-in fuel management would increase those projections. In any event it would be prudent to update the fluence analysis as the design of each future fuel cycle evolves.

The azimuthal variation of maximum fast neutron ($E > 1.0$ MeV) fluence at the inner radius of the pressure vessel is presented in Figure II.2-3 as a function of azimuthal angle. Data are presented for both current and projected end-of-life conditions. In Figure II.2-4, the relative radial variation of fast neutron flux and fluence within the pressure vessel wall is presented. Similar data showing the relative axial variation of fast neutron flux and fluence over the beltline region of the pressure vessel is shown in Figure II.2-5. A three-dimensional description of the fast neutron exposure of the pressure vessel wall can be constructed using the data given in Figures II.2-3 through II.2-5 along with the relation

$$\phi(R,\theta,Z) = \phi(\theta)\ F(R)\ G(Z)$$

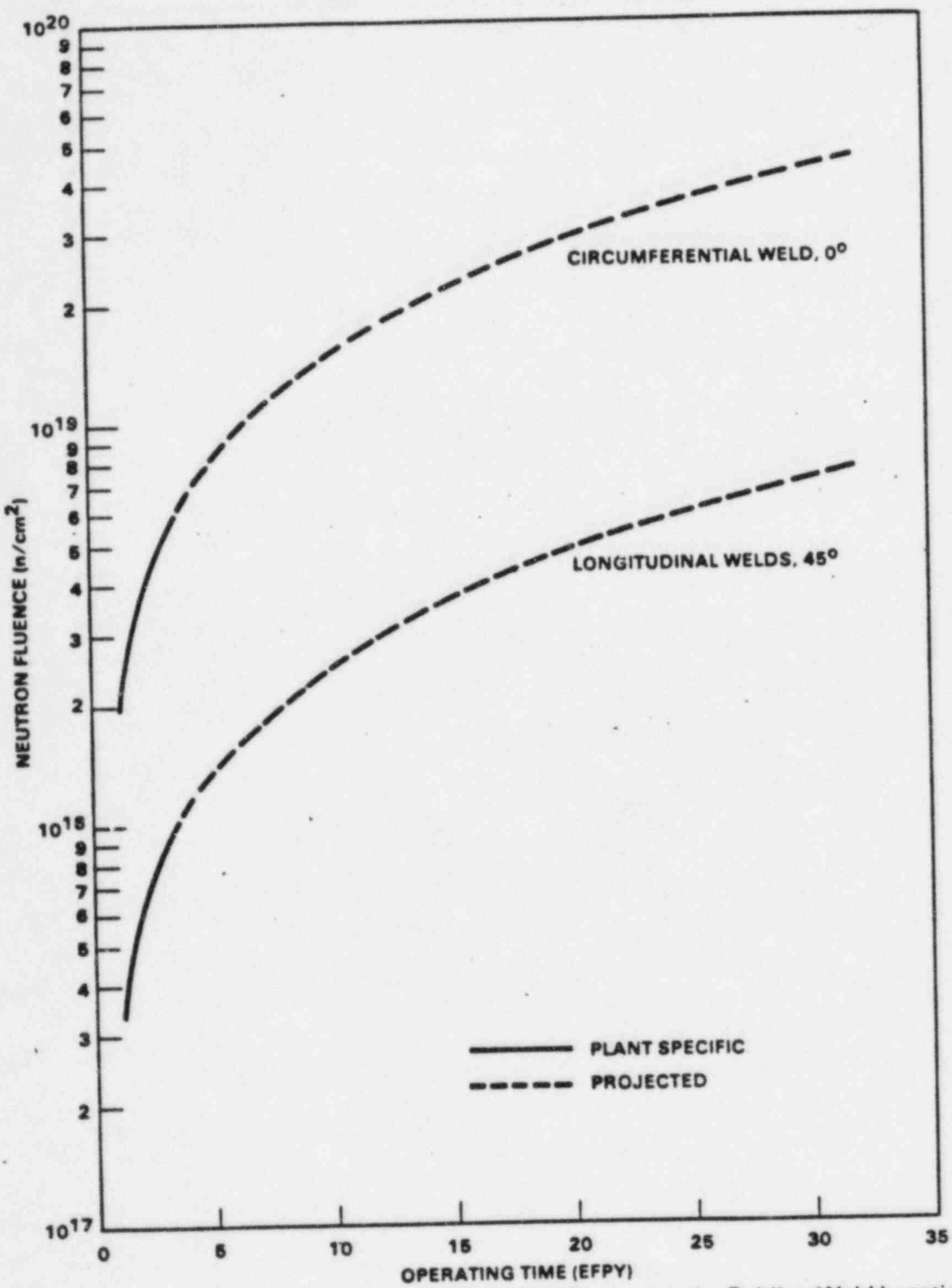


Figure II.2-1. Maximum Fast Neutron (E>1 MeV) Fluence at the Beltline Weld Locations as a Function of Full Power Operating Time

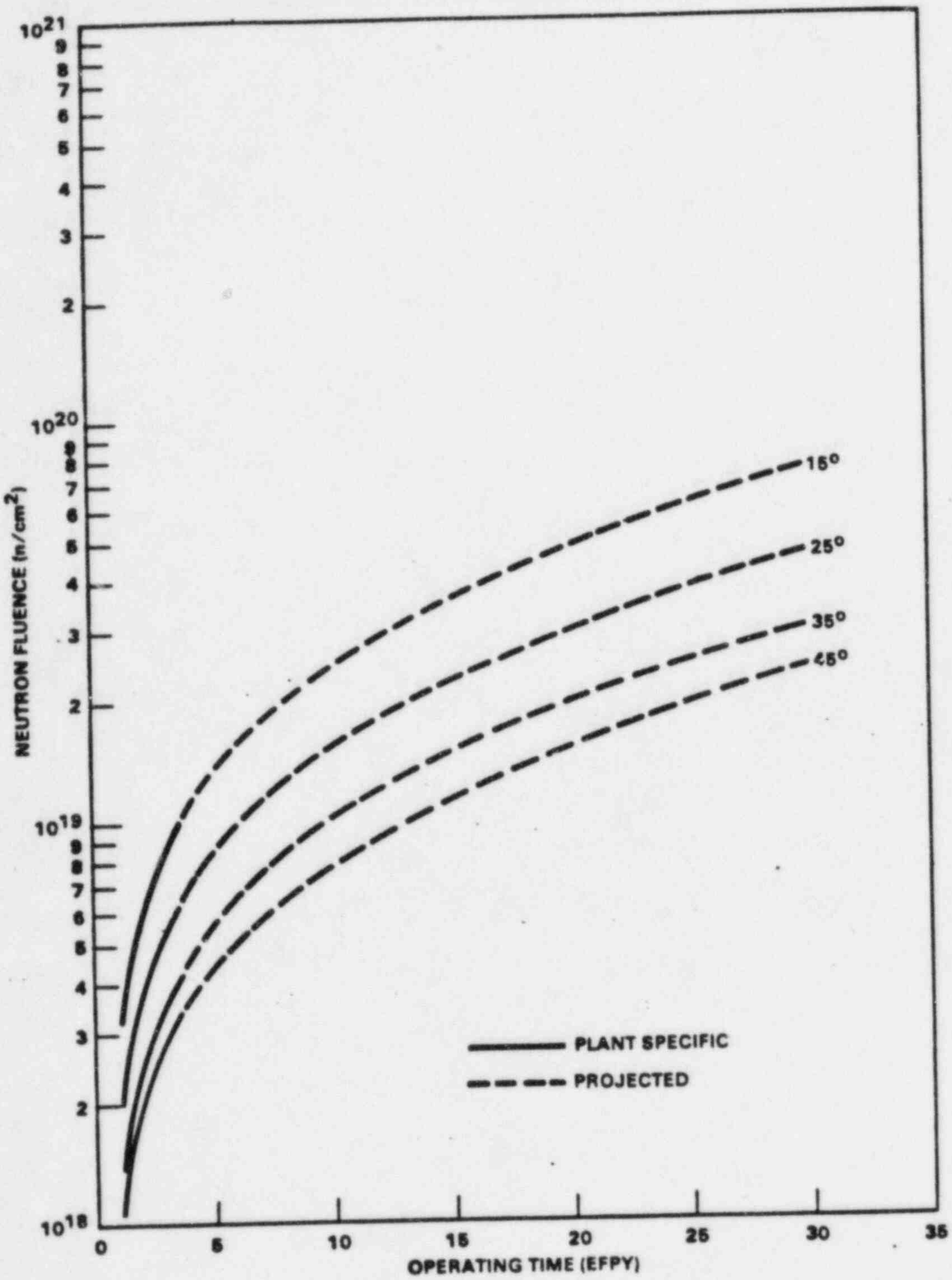


Figure II 2-2. Maximum Fast Neutron (E > 1 MeV) Fluence at the Center of the Surveillance Capsules as a Function of Full Power Operation Time

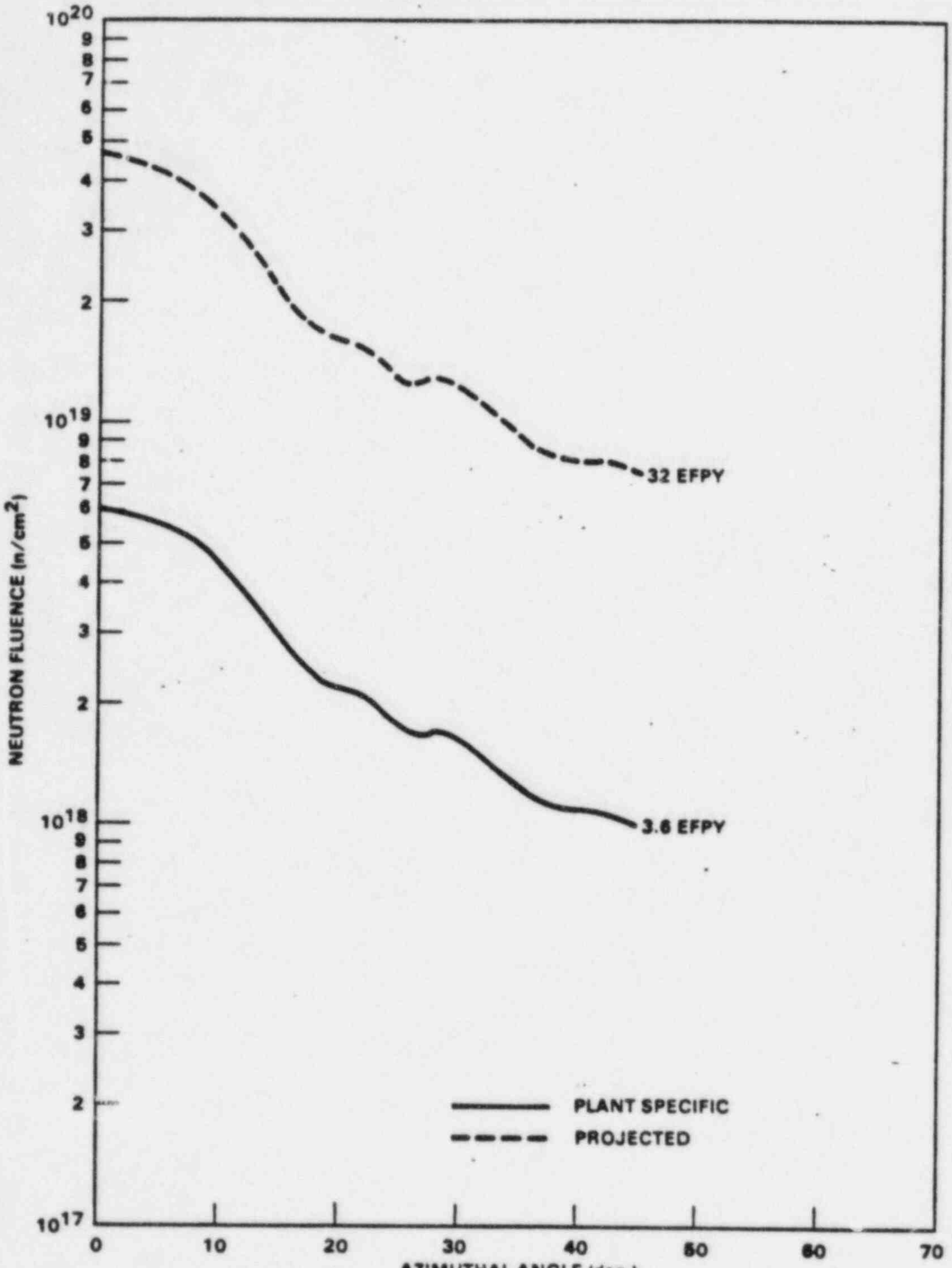


Figure II.2-3. Maximum Fast Neutron (E>1.0 MeV) Fluence at the Pressure Vessel Inner Radius as a Function of Azimuthal Angle

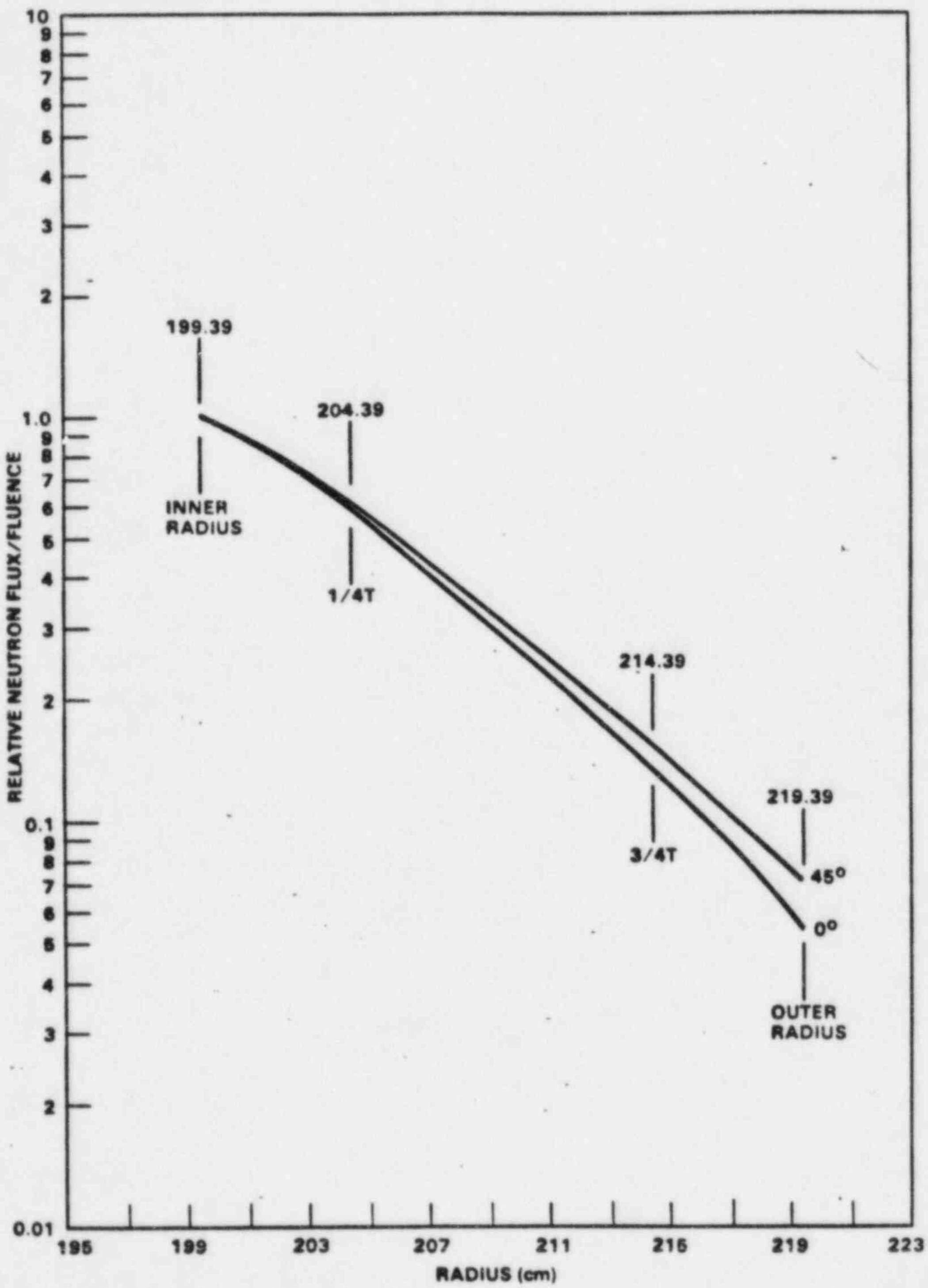


Figure II.2-4. Relative Radial Distribution of Fast Neutron (E > 1.0 MeV) Flux and Fluence Within the Pressure Vessel Wall

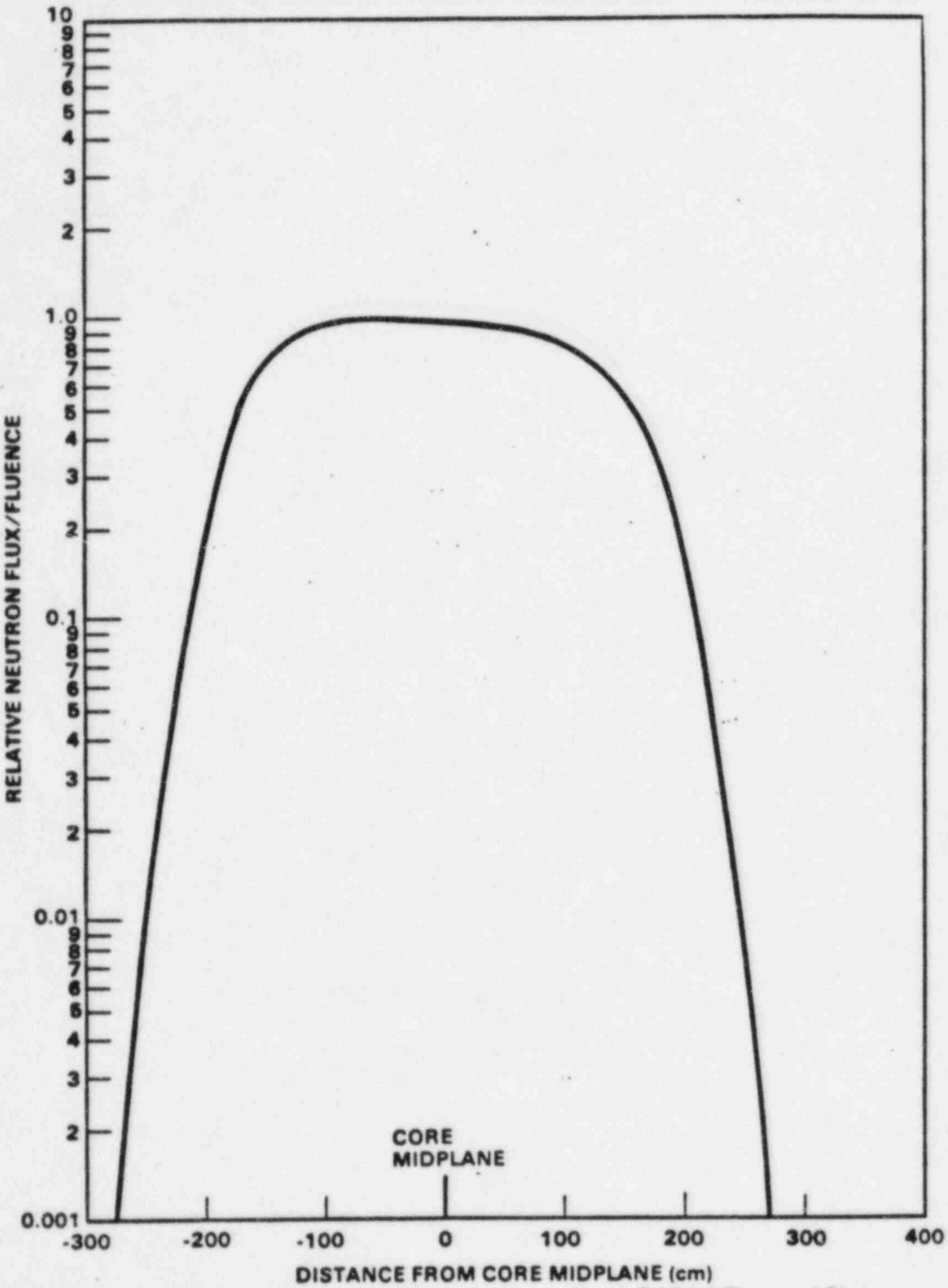


Figure II 2-5. Relative Axial Variation of Fast Neutron (E > 1.0 MeV) Flux and Fluence Within the Pressure Vessel Wall where: $\phi(R,\theta,Z)$ = Fast neutron fluence at location R, θ, Z within the pressure vessel wall

$\phi(\theta)$ = Fast neutron fluence at azimuthal location θ on the pressure vessel inner radius from Figure II.2-3

F (R) = Relative fast neutron flux at radius R into the pressure vessel from Figure II.2-4

G (Z) = Relative fast neutron flux at axial position Z from Figure II.2-5

Analysis has shown that the radial and axial variations within the vessel wall are relatively insensitive to the implementation of low leakage fuel management schemes. Thus, the above relationship provides a vehicle for a reasonable evaluation of fluence gradients within the vessel wall.

REFERENCES

2. Soltesz, R. G., Disney, R. K., Jedruch, J. and Ziegler, S. L., "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation Vol. 5 - Two Dimensional, Discrete Ordinates Transport Technique," WANL-PR(LL)034, Vol. 5, August 1970.

3. "Sailor RSIC Data Library Collection DLC-76," Coupled, Self-Shielded, 47 Neutron, 20 Gamma-Ray, P_3, Cross Section Library for Light Water Reactors.

4. Benchmark Testing of Westinghouse Neutron Transport Analysis Methodology - to be published.

APPENDIX A

POWER DISTRIBUTIONS

Core power distributions used in the plant specific fast neutron exposure analysis of the Beaver Valley Unit 1 pressure vessel was derived from the following fuel cycle design reports:

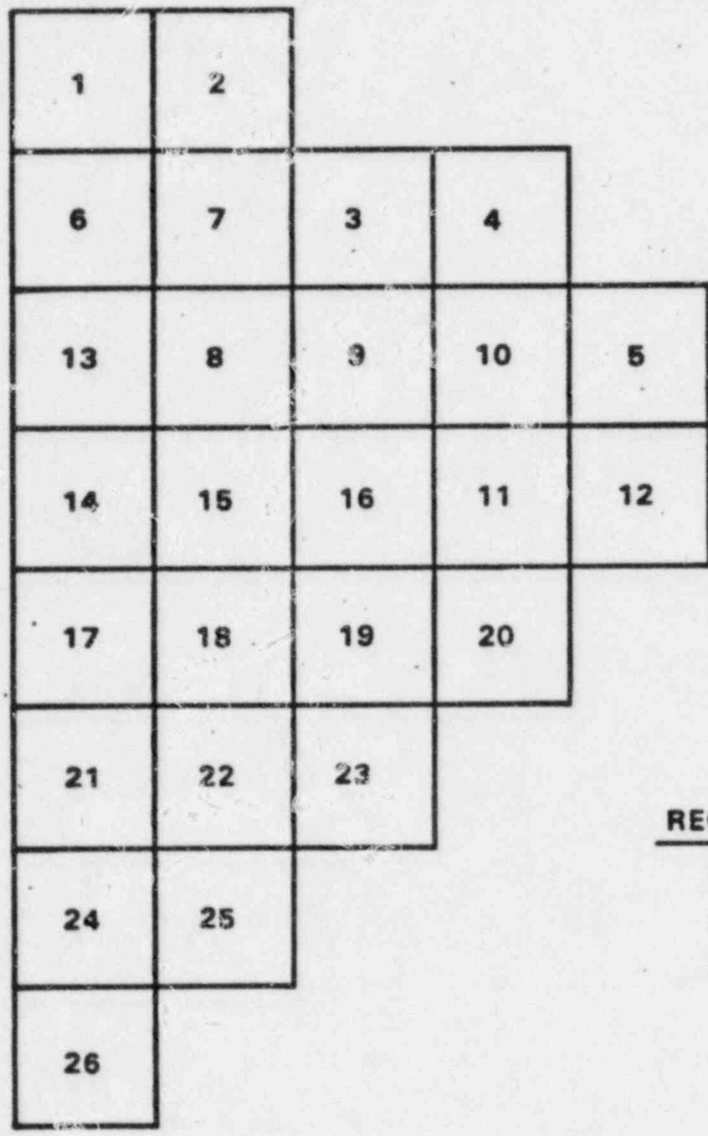
Fuel Cycle	Unit 1
1	WCAP-8441
2	WCAP-9505
3	WCAP-10037
4	WCAP-10330
5	WCAP-10660

A schematic diagram of the core configuration applicable to Beaver Valley Unit 1 is shown in Figure A-1. Cycle averaged relative assembly powers for each operating fuel cycle of Beaver Valley Unit 1 are listed in Table A-1 along with the design basis core power distribution.

On Figure A-1 and in Table A-1 an identification number is assigned to each fuel assembly location; and three regions consisting of subsets of fuel assemblies are defined. In performing the adjoint evaluations, the relative power in assemblies comprising Region 3 has been adjusted to account for known biases in the prediction of power in the peripheral assemblies while the relative power in assemblies comprising Region 2 has been maintained at the cycle average value. Due to the extreme self-shielding of the reactor core neutrons born in fuel assemblies comprising Region 1, Region 1 does not contribute significantly to the neutron exposure either at the surveillance capsules or at the pressure vessel. Therefore, power distribution data for assemblies in Region 1 are not listed in Table A-1.

In each of the adjoint evaluations, within assembly spatial gradients have been superimposed on the average assembly power levels. For the peripheral

Figure A-1. Beaver Valley Unit 1 Core Description for Power Distribution Map



REGION	ASSEMBLIES
1	13-26
2	6-12
3	1-5

TABLE A-1

CORE POWER DISTRIBUTIONS USED IN BEAVER VALLEY UNIT 1 FLUENCE ANALYSIS

Assembly	Design Basis Relative Assembly Power	Plant Specific Cycle Averaged Relative Assembly Power — Fuel Cycle 1	2	3	4	5
1	1.00	0.81	0.85	0.89	0.69	0.69
2	0.83	0.64	0.71	0.73	0.46	0.49
3	1.21	0.92	1.01	1.01	0.93	0.92
4	0.86	0.64	0.74	0.73	0.45	0.45
5	0.92	0.67	0.80	0.71	0.47	0.48
6	0.98	1.02	0.92	1.13	0.90	0.97
7	1.10	1.02	1.16	1.18	1.10	1.03
8	1.00	1.10	0.96	0.95	1.22	1.14
9	1.05	1.04	0.96	1.12	1.08	1.21
10	1.08	0.97	1.17	1.15	1.07	1.05
11	1.06	1.04	0.96	1.09	1.21	1.18
12	0.95	0.88	1.02	0.87	0.87	1.06

assemblies (Region 3), these spatial gradients also include adjustments to account for analytical deficiencies that tend to occur near the boundaries of the core region.