Westinghouse Energy Systems

Westinghouse Energy Systems

ATTACHMENT I TO IPN-89-060

WCAP-11045 REVISION 1
"INDIAN POINT UNIT 3 REACTOR VESSEL
FLUENCE AND RT_{PTS} EVALUATIONS"

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

WCAP-11045
Revision 1

INDIAN POINT UNIT 3 REACTOR VESSEL FLUENCE AND RT_{PTS} EVALUATIONS

S. E. Yanichko
S. L. Anderson
N. K. Ray

Work Performed for New York Power Authority

June 1989

Approved by: T. A. Meyer
T. A. Meyer, Manager
Structural Materials and
Reliability Technology

Work Performed Under Shop Order IVGP-108

WESTINGHOUSE ELECTRIC CORPORATION
Energy Systems Division
P.O. Box 2728
Pittsburgh, Pennsylvania 15230-2728

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

SECTION I

INTRODUCTION

The purpose of this report is to submit the reference temperature for pressurized thermal shock (RT_{PTS}) values for the Indian Point Unit 3 reactor vessel to address the Pressurized Thermal Shock (PTS) Rule. Section I discusses the Rule and provides the methodology for calculating RT_{PTS}. Section II presents the results of the neutron exposure evaluation assessing the effects that past and present core management strategies have had on neutron fluence levels in the reactor vessel. Section III provides the reactor vessel beltline region material properties. Section IV provides the RT_{PTS} calculations from present through the projected end-of-license fluence values.

I.1 THE PRESSURIZED THERMAL SHOCK RULE

The Pressurized Thermal Shock (PTS) Rule [1] was approved by the U.S. Nuclear Regulatory Commissioners on June 20, 1985, and appeared in the Federal Register on July 23, 1985. The date that the Rule was published in the Federal Register is the date that the Rule became a regulatory requirement.

The Rule outlines regulations to address the potential for PTS events on pressurized water reactor (PWR) vessels in nuclear power plants that are operated with a license from the United States Nuclear Regulatory Commission (USNRC). PTS events have been shown from operating experience to be transients that result in a rapid and severe cooldown in the primary system coincident with a high or increasing primary system pressure. The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The Rule establishes the following requirements for all domestic, operating PWRs:

* Establishes the RT_{PTS} (measure of fracture resistance) Screening Criterion for the reactor vessel beltline region

  270°F for plates, forgings, axial welds
  300°F for circumferential weld materials

* 6 Months From Date of Rule: All plants must submit their present RT_{PTS} values (per the prescribed methodology) and projected RT_{PTS} values at the expiration date of the operating license. The date that this submittal must be received by the NRC for plants with operating licenses is January 23, 1986.

* 9 Months From Date of Rule: Plants projected to exceed the PTS Screening Criterion shall submit an analysis and a schedule for implementation of such flux reduction programs as are reasonably practicable to avoid reaching the Screening Criterion. The data for this submittal must be received by the NRC for plants with operating licenses by April 23, 1986.

* Requires plant-specific PTS Safety Analyses before a plant is within 3 years of reaching the Screening Criterion, including analyses of alternatives to minimize the PTS concern.

* Requires NRC approval for operation beyond the Screening Criterion.

In the Rule, the NRC provides guidance regarding the calculation of the toughness state of the reactor vessel materials - the "reference temperature for nil ductility transition" (RT_{NDT}). For purposes of the Rule, RT_{NDT} is now defined as "the reference temperature for pressurized thermal shock" (RT_{PTS}) and calculated as prescribed by 10 CFR 50.61(b) of the Rule. Each USNRC licensed PWR must submit a projection of RT_{PTS} values from the time of the submittal to the license expiration date. This assessment must be submitted within 6 months after the effective date of the Rule, on January 23, 1986, with updates whenever changes in core loadings, surveillance measurements, or other information indicate a significant change in projected values. The calculation must be made for each weld and plate, or forging, in the reactor vessel beltline. The purpose of this report is to provide ~~the~~ UPDATED RT_{PTS} values for Indian Point Unit 3.

I.2 THE CALCULATION OF RT_{PTS}

In the PTS Rule, the NRC Staff has selected a conservative and uniform method for determining plant-specific values of RT_{PTS} at a given time.

The prescribed equations in the PTS rule for calculating RT_{PTS} are actually one of several ways to calculate RT_{NDT}. For the purpose of comparison with the Screening Criterion, the value of RT_{PTS} for the reactor vessel must be calculated for each weld and plate, or forging in the beltline region as given below. For each material, RT_{PTS} is the lower of the results given by Equations 1 and 2.

Equation 1:

$$RT_{PTS} = I + M + [-10 + 470(Cu) + 350(Cu)(Ni)]\ f^{0.270}$$

Equation 2:

$$RT_{PTS} = I + M + 283\ f^{0.194}$$

where

I = the initial reference transition temperature of the unirradiated material measured as defined in the ASME Code, NB-2331. If a measured value is not available, the following generic mean values must be used: 0°F for welds made with Linde 80 flux, and -56°F for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes.

M = the margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel content, fluence, and calculation procedures. In Equation 1, M=48°F if a measured value of I was used, and M=59°F if the generic mean value of I was used. In Equation 2, M=0°F if a measured value of I was used, and M=34°F if the generic mean value of I was used.

Cu and Ni = the best estimate weight percent of copper and nickel in the material.

f = the maximum neutron fluence, in units of $10^{19} n/cm^2$ (E greater than or equal to 1 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question.

Note, since the chemistry values given in equations 1 and 2 are best estimate mean values, and the margin, M, causes the RT_{PTS} values to be upper bound predictions, the mean material chemistry values are to be used, when available, so as not to compound conservatism. The basis for the Cu and Ni values used in the RT_{PTS} calculations for Indian Point Unit 3 are discussed in Section III.2.

SECTION II
NEUTRON EXPOSURE EVALUATION

This section describes a discrete ordinates Sn transport analysis performed for the Indian Point Unit 3 reactor to determine the neutron radiation environment within the reactor vessel and surveillance capsules. Fast neutron exposure parameters in terms of fast neutron fluence (E>1.0 MeV) and iron atom displacements (dpa) are established on a plant and fuel cycle specific basis for the first seven reactor operating fuel cycles; and, based on these evaluations, projections of vessel exposure for future operating periods using the most recent cycle design are estimated. Neutron dosimetry results from the first three surveillance capsules withdrawn from the Indian Point Unit 3 reactor are integrated with the analytically derived exposure values to provide an overall "best estimate" of the current and future exposure of the pressure vessel.

The use of fast neutron fluence (E>1.0 MeV) to correlate measured material properties changes to the neutron exposure of the material for light water reactor applications has traditionally been accepted for development of damage trend curves as well as for the implementation of trend curve data to assess vessel condition. In recent years, however, it has been suggested that an exposure model that accounts for differences in neutron energy spectra among surveillance capsule locations and positions within the pressure vessel wall could lead to an improvement in the uncertainties associated with damage trend curves as well as to a more accurate evaluation of damage gradients through the pressure vessel wall.

Because of this potential shift away from a threshold fluence toward an energy dependent damage function for data correlation, ASTM Standard Practice E853, "Analysis and Interpretation of Light Water Reactor Surveillance Results", recommends reporting iron atom displacements (dpa) along with fluence (E>1.0 MeV) to provide a data base for future reference. The energy dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693, "Characterising Neutron Exposures in Ferritic Steels in Terms of Displacements per Atom". The Application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the pressure vessel wall has already been promulgated in Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials". Therefore, in keeping with the philosophy espoused in the current ASTM standards governing pressure vessel exposure evaluations, dpa data is also included in this section.

II.1 METHOD OF ANALYSIS

In performing the fast neutron exposure evaluations for the Indian Point Unit 3 reactor pressure vessel, two distinct sets of transport calculations were carried out. The first, a single computation in the conventional forward mode, was used primarily to obtain relative neutron energy distributions throughout the reactor geometry as well as to establish relative radial distributions of exposure parameters [fluence (E>1.0 MeV) and dpa] through the vessel wall. The neutron spectral information is required for the interpretation of neutron dosimetry withdrawn from surveillance capsules as well as for the determination of exposure parameter ratios; i.e., dpa/fluence (E>1.0 MeV), within the pressure vessel geometry. The relative radial gradient information is required to permit the projection of measured exposure parameters to locations interior to the pressure vessel wall; i.e., the 1/4T, 1/2T, and 3/4T locations.

The second set of calculations consisted of a series of adjoint analyses relating the fast neutron flux (E>1.0 MeV) at surveillance capsule positions and several azimuthal locations on the pressure vessel inner radius to neutron source distributions within the reactor core. The importance functions generated from these adjoint analyses provide the basis for all absolute exposure projections and comparison with measurement. These importance functions, when combined with cycle specific neutron source distributions, yielded absolute predictions of neutron exposure at the locations of interest for each of the first seven operating fuel cycles; and established the means to perform similar predictions and dosimetry evaluations for subsequent fuel cycles. It is important to note that the cycle specific neutron source distributions utilized in these analyses included not only spatial variations of fission rates within the reactor core; but, also accounted for the effects of varying neutron yield per fission and fission spectrum introduced by the build-in of plutonium as the burnup of individual fuel assemblies increased.

The absolute cycle specific data from the adjoint evaluations together with relative energy spectra and radial distribution information from the forward calculation provide the means to:

1. Evaluate neutron dosimetry obtained from the surveillance capsule program.

2. Extrapolate dosimetry results to key locations at the inner radius and through the thickness of the pressure vessel wall.

3. Enable a direct comparison of analytical prediction with measurement.

4. Establish a mechanism for projection of pressure vessel exposure as the design of each new fuel cycle evolves.

A plan view of the Indian Point Unit 3 reactor geometry at the core midplane is shown in figure II.1-1. Since the reactor exhibits 1/8 core symmetry only a 0-45 degree sector is depicted. In addition to the core, reactor internals, pressure vessel, and primary biological shield, the model also included explicit representations of the surveillance capsules, the pressure vessel cladding, and the insulation located external to the pressure vessel.

From a neutronic standpoint, the inclusion of the surveillance capsules in the analytical model is significant. Since the presence of the capsules and associated support structure has a marked impact on both the flux magnitude and energy spectra at dosimetry locations, a meaningful comparison of measurement and calculation can be made only if these perturbation effects are properly accounted for in the analysis. These analytical/experimental comparisons are important since, on a plant specific basis, they establish the overall accuracy of the analytical technique. A plan view of a single surveillance capsule attached to the thermal shield is shown in figure II.1-2.

The forward transport calculation for the reactor model summarized in figures II.1-1 and II.1-2 was carried out in R, Theta geometry using the DOT two-dimensional discrete ordinates code [2] and the SAILOR cross-section library [3]. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In this analysis anisotropic scattering was treated with a P3 expansion of the scattering cross-sections and the angular discretization was modeled with an S8 order of angular quadrature. The reference forward calculation was normalized to a core midplane power density characteristic of operation at a thermal power level of 3025 MWt.

The reference core power distribution utilized in the forward analysis was derived from statistical studies of long term operation of Westinghouse 4-loop plants. Inherent in the development of this reference 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 2 sigma uncertainty derived from the statistical evaluation of plant to plant and cycle to cycle variations in peripheral power was added to the nominal assembly power level for all fuel assemblies adjacent to the core baffle plates. An axial peaking factor of 1.20 was also employed to scale the axially averaged power distribution to the midplane value. Since it is unlikely that a single reactor would have a power distribution at the nominal + 2 sigma level and would maintain an axial peaking factor of 1.20 for a large number of cycles, the use of this reference case is expected to yield somewhat conservative results. This is especially true in cases where low leakage fuel management has been employed. The reference core power distribution data used in the analysis is provided in Appendix A to this report. The data listed in Appendix A represents cycle averaged assembly powers.

All adjoint analyses were also carried out using a P3 cross-section approximation from the SAILOR library and an S8 order of angular quadrature. Adjoint source locations were chosen at several locations at the pressure vessel inner radius and at the geometric center of surveillance capsules positioned at 4 degrees and 40 degrees relative to the core cardinal axes. Again, these calculations were run in R,Theta geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case flux (E > 1.0 MeV). Having the importance functions and appropriate core source distributions, the response of interest could be calculated as:

$$\phi(Ro,\theta o) = \int_R \int_\theta \int_E I(R,\theta,E)\ S(R,\theta,E)\ RdRd\theta dE$$

where: $\phi(Ro,\theta o)$ = Neutron flux (E > 1.0 MeV) at the adjoint source location of radius Ro and an Azimuthal angle θo

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

$S(R,\theta,E)$ = Neutron source strength at core location R,O and energy E

Although the adjoint importance functions used in the Indian Point Unit 3 analyses were based on a response function defined by the threshold neutron flux (E > 1.0 MeV), prior calculations have shown that, while the implementation of low leakage loading patterns significantly impact the magnitude and spatial distribution of the neutron field, changes in the relative neutron energy spectrum are of second order. Thus, for a given location the ratio of dpa/fluence (E > 1.0 MeV) is insensitive to changing core source distributions. In the application of these importance functions to the Indian Point Unit 3 reactor, therefore, iron atom displacements (dpa) were computed on a cycle specific basis by using dpa/fluence (E > 1.0 MeV) ratios from the forward analysis in conjunction with the cycle specific fluence (E > 1.0 MeV) solutions from the individual adjoint evaluations.

The core power distributions used in the plant specific adjoint evaluations were taken from the appropriate fuel cycle design reports for the Indian Point Unit 3 reactor [4 through 10]. The data extracted from the design reports represented cycle averaged relative assembly powers, burnups, and axial peaking factors. Therefore, the adjoint results provided data in terms of fuel cycle averaged neutron flux which, when multiplied by the appropriate fuel cycle length, in turn yielded the incremental fast neutron exposure for the fuel cycle. In constructing the cycle specific energy dependent source distributions account was taken of the burnup dependent inventory of fissioning isotopes, including U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242. For comparison purposes with the reference core power distribution used in the forward analysis, the cycle specific relative assembly power distributions used in the adjoint analyses are also provided in Appendix A to this report.

The transport methodology, both forward and adjoint, using the SAILOR cross-section library has been benchmarked against neutron dosimetry data obtained at the ORNL PCA facility [11]. Extensive comparisons of analytical predictions with measurements from power reactor surveillance capsules and reactor cavity dosimetry programs have also been made. The benchmarking studies indicate that the use of SAILOR cross-sections and the generic reference core power distribution 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 a tendency to underpredict fluence levels at surveillance capsule positions by from 5-10%; while calculations applicable to reactor cavity locations tend to be biased low by approximately 10-20% depending on the thickness of the pressure vessel. In performing the exposure evaluation for the Indian Point Unit 3 reactor the comparisons of predictions with measurements obtained from previously withdrawn surveillance capsules (T,Y, and Z) were factored into the overall analysis to provide best estimate exposure levels with a minimum uncertainty.

II.2 FAST NEUTRON FLUENCE RESULTS

Best estimate fast neutron (E > 1.0 MeV) exposure results for the Indian Point Unit 3 reactor are presented in Tables II.2-1 through II.2-7 and in Figures II.2-1 and II.2-2. 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. Factors to convert the data provided in Tables II.2-1 through II.2-7 and Figures II.2-1 and II.2-2 to exposure in terms of iron atom displacements rather than fluence (E > 1.0 MeV) are given in Tables II.2-8 and II.2-9.

In Tables II.2-1 through II.2-4 cycle specific maximum neutron flux and fluence levels at 0, 15, 30, and 45 degrees on the reactor vessel inner radius of Indian Point Unit 3 are listed for the period of operation up to May 31,1989 and projected fluence levels are provided for irradiation periods encompassing the expiration date of the current operating license (8/13/2009). Also presented are the reference fluence levels predicted using the generic 4-loop core power distribution at the nominal + 2 sigma level. Similar data applicable to the center of surveillance capsules located at 4 degrees and 40 degrees are given in Tables II.2-5 and II.2-6, respectively.

In regard to the cycle specific average flux and fluence levels listed in Tables II.2-1 through II.2-6, it should be noted that the calculated neutron radiation levels have been multiplied by a factor of 1.086 to account for biases observed between cycle specific calculations and the results of neutron dosimetry for the first three surveillance capsules [12,13,14] removed from the Indian Point Unit 3 reactor. The factor of 1.086 was derived by taking the average of the measurement to calculation ratios (M/C) as follows.

	CALCULATED FLUENCE (n/cm^2)	MEASURED FLUENCE (n/cm^2)	M/C
CAPSULE T	2.66E+18	2.99E+18	1.124
CAPSULE Y	7.03E+18	7.58E+18	1.078
CAPSULE Z	1.04E+19	1.10E+19	1.058
AVERAGE			1.086

These three surveillance capsules were withdrawn from the 40 degree location at the end of cycles 1, 3, and 5, respectively. Application of this bias factor to the analytical results assures that the vessel exposure predictions represent conservative values with a minimum uncertainty.

Graphical representations of the plant specific fast neutron fluence at key locations on the reactor vessel are shown in Figure II.2-1. Reactor vessel exposure data is presented for the 45 degree location on the circumferential weld and shell plates as well as for all of the longitudinal welds located in the beltline region (see Section III.1). In regard to Figure II.2-1, the solid portions of the fluence curves are based directly on the cycle specific core loadings for the first six fuel cycles. The dashed portions of these curves, however, involve a projection into the future. The neutron flux calculated for cycle 7 was used to project these future fluence levels.

It should be noted that implementation of a more severe low leakage pattern than that represented by the cycle 7 loading pattern would reduce the projections of fluence at key locations. On the other hand, relaxation of the cycle 7 loading pattern or a return to out-in fuel management would increase those projections. The RTpts assessment must be updated per 10CFR50.61(b) [1] whenever, among other things, changes in core loadings significantly impact the fluence and RTpts projections.

In Figure II.2-2, the azimuthal variation of maximum fast neutron ($E > 1.0$ MeV) fluence at the inner radius of the reactor vessel is presented. Data are depicted for both current (EOC 6) and projected conditions. In Table II.2-7, the relative radial variation of fast neutron flux and fluence within the reactor vessel wall is listed for the azimuthal locations of each of the key vessel materials. Similar data showing the cycle specific relative axial variation of fast neutron flux and fluence over the beltline region of the reactor are included with the remainder of the cycle specific core power distribution information in Appendix A to this report. A two-dimensional description of the maximum fast neutron exposure of the reactor vessel wall can be constructed using the data given in Tables II.2-1 through II.2-7 along with the relation

$$\Phi(R,\theta) = \Phi(\theta)\ F(R)$$

where: $\Phi(R,\theta)$ = Fast neutron fluence at location R,θ within the reactor vessel wall

$\Phi(\theta)$ = Fast neutron fluence at azimuthal location r on the reactor vessel inner radius from Tables II.2-1 through II.2-4

F(R) = Relative fast neutron fluence at radius R into the vessel wall from Table II.2-7

Analysis has shown that the radial 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.

If, in addition to the R,Theta variation of maximum neutron exposure, an axial variation over the beltline region of the vessel is desired, these distributions can be constructed by using the cycle specific axial power distributions listed in Appendix A and reintegrating over the appropriate irradiation periods.

All of the best estimate fast neutron (E > 1.0 MeV) exposure data can be converted to iron atom displacements (dpa) by making use of the data listed in Tables II.2-8 and II.2-9. The exposure parameter ratios given in Table II.2-8 can be employed in conjunction with the fluence information from Tables II.2-1 through II.2-4 to calculate the dpa values at the pressure vessel inner radius. Distributional information within the vessel wall may then be calculated by normalizing the inner radius values to the relative radial gradient data given in Table II.2-9.

TABLE II.2-1

INDIAN POINT UNIT 3
FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE
REACTOR VESSEL INNER RADIUS - 0 DEGREE AZIMUTHAL ANGLE (a)

IRRADIATION INTERVAL	ELAPSED IRRADIATION TIME (EFPY)	AVG. FLUX (b) (n/cm2-sec)	BELTLINE REGION CUMULATIVE FLUENCE (n/cm2) PLANT SPECIFIC (b)	REFERENCE (c)
CY-1	1.37	6.47E+09	2.80E+17	3.23E+17
CY-2	2.23	7.70E+09	4.88E+17	5.26E+17
CY-3	3.29	7.95E+09	7.53E+17	7.76E+17
CY-4	4.41	6.97E+09	1.00E+18	1.04E+18
CY-5	5.55	6.41E+09	1.23E+18	1.31E+18
CY-6 (d)	6.73	5.91E+09	1.45E+18	1.59E+18
CY7-EOL (e)	21.88	5.53E+09	4.10E+18	5.16E+18

(a) Applicable to longitudinal weld 2-042C in the intermediate shell

(b) Includes an analytical bias factor of 1.086

(c) Reference fast neutron flux = 7.47E+09 n/cm2-sec at 3025 MWt

(d) Current neutron fluences are defined as of the end of cycle 6

(e) Fuel cycle projections are based on the average neutron flux for cycle 7 and an assumed capacity factor of 0.75

TABLE II.2-2

INDIAN POINT UNIT 3
FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE
REACTOR VESSEL INNER RADIUS - 15 DEGREE AZIMUTHAL ANGLE (a)

IRRADIATION INTERVAL	ELAPSED IRRADIATION TIME (EFPY)	AVG. FLUX (b) (n/cm2-sec)	BELTLINE REGION CUMULATIVE FLUENCE (n/cm2) PLANT SPECIFIC (b)	REFERENCE (c)
CY-1	1.37	1.03E+10	4.46E+17	5.23E+17
CY-2	2.23	1.22E+10	7.75E+17	8.52E+17
CY-3	3.29	1.25E+10	1.19E+18	1.26E+18
CY-4	4.41	1.05E+10	1.56E+18	1.68E+18
CY-5	5.55	9.34E+09	1.90E+18	2.12E+18
CY-6 (d)	6.73	8.84E+09	2.23E+18	2.57E+18
CY7-EOL (e)	21.88	8.95E+09	6.51E+18	8.35E+18

(a) Applicable to longitudinal welds 3-042A and 3-042B in the lower shell

(b) Includes an analytical bias factor of 1.086

(c) Reference fast neutron flux = 1.21E+10 n/cm2-sec at 3025 MWt

(d) Current neutron fluences are defined as of the end of cycle 6

(e) Fuel cycle projections are based on the average neutron flux for cycle 7 and an assumed capacity factor of 0.75

TABLE II.2-3

INDIAN POINT UNIT 3
FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE
REACTOR VESSEL INNER RADIUS - 30 DEGREE AZIMUTHAL ANGLE (a)

IRRADIATION INTERVAL	ELAPSED IRRADIATION TIME (EFPY)	AVG. FLUX (b) (n/cm2-sec)	BELTLINE REGION CUMULATIVE FLUENCE (n/cm2) PLANT SPECIFIC (b)	REFERENCE (c)
CY-1	1.37	1.27E+10	5.50E+17	6.49E+17
CY-2	2.23	1.59E+10	9.79E+17	1.06E+18
CY-3	3.29	1.40E+10	1.45E+18	1.56E+18
CY-4	4.41	1.13E+10	1.85E+18	2.09E+18
CY-5	5.55	9.34E+09	2.18E+18	2.63E+18
CY-6 (d)	6.73	8.98E+09	2.52E+18	3.19E+18
CY7-EOL (e)	21.88	7.98E+09	6.33E+18	1.04E+19

(a) Applicable to longitudinal welds 2-042A and 2-042B in the intermediate shell

(b) Includes an analytical bias factor of 1.086

(c) Reference fast neutron flux = 1.50E+10 n/cm2-sec at 3025 MWt

(d) Current neutron fluences are defined as of the end of cycle 6

(e) Fuel cycle projections are based on the average neutron flux for cycle 7 and an assumed capacity factor of 0.75

TABLE II.2-4

INDIAN POINT UNIT 3

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE

REACTOR VESSEL INNER RADIUS - 45 DEGREE AZIMUTHAL ANGLE (a)

			BELTLINE REGION CUMULATIVE FLUENCE (n/cm2)	
IRRADIATION INTERVAL	ELAPSED IRRADIATION TIME (EFPY)	AVG. FLUX (b) (n/cm2-sec)	PLANT SPECIFIC (b)	REFERENCE (c)
CY-1	1.37	1.94E+10	8.40E+17	1.00E+18
CY-2	2.23	2.53E+10	1.52E+18	1.63E+18
CY-3	3.29	2.09E+10	2.22E+18	2.41E+18
CY-4	4.41	1.67E+10	2.81E+18	3.23E+18
CY-5	5.55	1.38E+10	3.31E+18	4.06E+18
CY-6 (d)	6.73	1.30E+10	3.79E+18	4.93E+18
CY7-EOL (e)	21.88	1.05E+10	8.81E+18	1.60E+19

(a) Applicable to longitudinal weld 3-042C in the lower shell, the intermediate to lower shell circumferential weld 9-042, and all shell plates

(b) Includes an analytical bias factor of 1.086

(c) Reference fast neutron flux = 2.32E+10 n/cm2-sec at 3025 MWt

(d) Current neutron fluences are defined as of the end of cycle 6

(e) Fuel cycle projections are based on the average neutron flux for cycle 7 and an assumed capacity factor of 0.75

TABLE II.2-5

INDIAN POINT UNIT 3
FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE
4 DEGREE SURVEILLANCE CAPSULE CENTER

IRRADIATION INTERVAL	ELAPSED IRRADIATION TIME (EFPY)	AVG. FLUX (a) (n/cm2-sec)	BELTLINE REGION CUMULATIVE FLUENCE (n/cm2) PLANT SPECIFIC (a)	BELTLINE REGION CUMULATIVE FLUENCE (n/cm2) REFERENCE (b)
CY-1	1.37	2.16E+10	9.37E+17	1.08E+18
CY-2	2.23	2.58E+10	1.63E+18	1.76E+18
CY-3	3.29	2.66E+10	2.52E+18	2.60E+18
CY-4	4.41	2.33E+10	3.35E+18	3.48E+18
CY-5	5.55	2.14E+10	4.12E+18	4.38E+18
CY-6 (c)	6.73	1.98E+10	4.85E+18	5.31E+18
CY7-EOL (d)	21.88	1.85E+10	1.37E+19	1.73E+19

(a) Includes an analytical bias factor of 1.086

(b) Reference fast neutron flux = 2.50E+10 n/cm2-sec at 3025 MWt

(c) Current neutron fluences are defined as of the end of cycle 6

(d) Fuel cycle projections are based on the average neutron flux for cycle 7 and an assumed capacity factor of 0.75

TABLE II.2-6

INDIAN POINT UNIT 3
FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE
40 DEGREE SURVEILLANCE CAPSULE CENTER

IRRADIATION INTERVAL	ELAPSED IRRADIATION TIME (EFPY)	AVG. FLUX (a) (n/cm2-sec)	BELTLINE REGION CUMULATIVE FLUENCE (n/cm2) PLANT SPECIFIC (a)	REFERENCE (b)
CY-1	1.37	6.67E+10	2.89E+18	3.47E+18
CY-2	2.23	8.79E+10	5.26E+18	5.64E+18
CY-3	3.29	7.14E+10	7.64E+18	8.33E+18
CY-4	4.41	5.67E+10	9.65E+18	1.12E+19
CY-5	5.55	4.64E+10	1.14E+19	1.40E+19
CY-6 (c)	6.73	4.34E+10	1.30E+19	1.70E+19
CY7-EOL (d)	21.88	3.40E+10	2.92E+19	5.54E+19

(a) Includes an analytical bias factor of 1.086

(b) Reference fast neutron flux = 8.02E+10 n/cm2-sec at 3025 MWt

(c) Current neutron fluences are defined as of the end of cycle 6

(d) Fuel cycle projections are based on the average neutron flux for cycle 7 and an assumed capacity factor of 0.75

TABLE II.2-7

RELATIVE RADIAL DISTRIBUTIONS OF NEUTRON FLUX (E > 1.0 MeV) WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	0 Degrees	15 Degrees	30 Degrees	45 Degrees
220.27(a)	1.000	1.000	1.000	1.000
220.64	0.977	0.978	0.979	0.977
221.66	0.884	0.887	0.889	0.885
222.99	0.758	0.762	0.765	0.756
224.31	0.641	0.644	0.648	0.637
225.63	0.537	0.540	0.545	0.534
226.95	0.448	0.451	0.455	0.443
228.28	0.372	0.375	0.379	0.367
229.60	0.309	0.310	0.315	0.303
230.92	0.255	0.257	0.261	0.250
232.25	0.211	0.212	0.216	0.206
233.57	0.174	0.175	0.178	0.169
234.89	0.143	0.144	0.147	0.138
236.22	0.117	0.118	0.121	0.113
237.54	0.0961	0.0963	0.0989	0.0912
238.86	0.0783	0.0783	0.0807	0.0736
240.19	0.0635	0.0632	0.0656	0.0584
241.51	0.0511	0.0501	0.0519	0.0454
242.17(b)	0.0483	0.0469	0.0487	0.0422

(a) Indicates base metal inner radius

(b) Indicates base metal outer radius

TABLE II.2-8

dpa/fluence (E > 1.0 MeV) RATIOS AT THE PRESSURE VESSEL INNER RADIUS AND AT THE CENTER OF SURVEILLANCE CAPSULES

	[dpa]/fluence (E > 1.0 MeV)
0 deg. Vessel IR	1.63E-21
15 deg. Vessel IR	1.61E-21
30 deg. Vessel IR	1.61E-21
45 deg. Vessel IR	1.63E-21
4 deg. Capsule	1.62E-21
40 deg. Capsule	1.71E-21

TABLE II.2-9

RELATIVE RADIAL DISTRIBUTIONS OF IRON ATOM DISPLACEMENT RATE (dpa)
WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	0 Degrees	15 Degrees	30 Degrees	45 Degrees
220.27(a)	1.000	1.000	1.000	1.000
220.64	0.983	0.983	0.984	0.983
221.66	0.913	0.914	0.918	0.915
222.99	0.818	0.819	0.827	0.820
224.31	0.728	0.728	0.739	0.730
225.63	0.647	0.646	0.659	0.647
226.95	0.574	0.573	0.587	0.573
228.28	0.510	0.507	0.523	0.507
229.60	0.453	0.450	0.466	0.449
230.92	0.402	0.399	0.414	0.397
232.25	0.356	0.353	0.368	0.349
233.57	0.315	0.312	0.327	0.307
234.89	0.277	0.275	0.289	0.269
236.22	0.243	0.241	0.254	0.233
237.54	0.212	0.210	0.222	0.201
238.86	0.182	0.181	0.192	0.170
240.19	0.155	0.154	0.164	0.141
241.51	0.131	0.128	0.137	0.113
242.17(b)	0.125	0.122	0.130	0.106

(a) Indicates base metal inner radius

(b) Indicates base metal outer radius

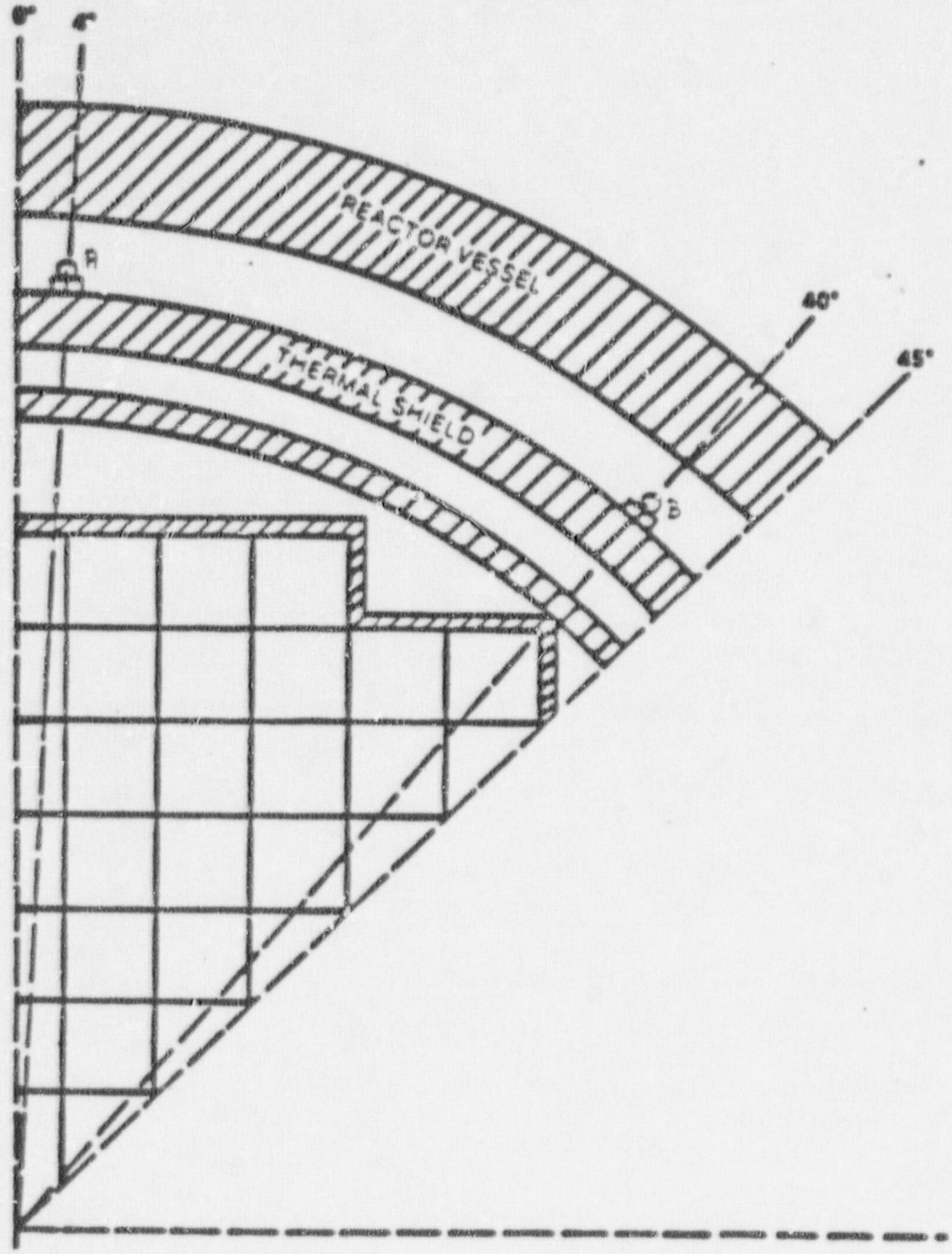


Figure II.1-1. Indian Point Unit 3 R,Theta Reactor Geometry

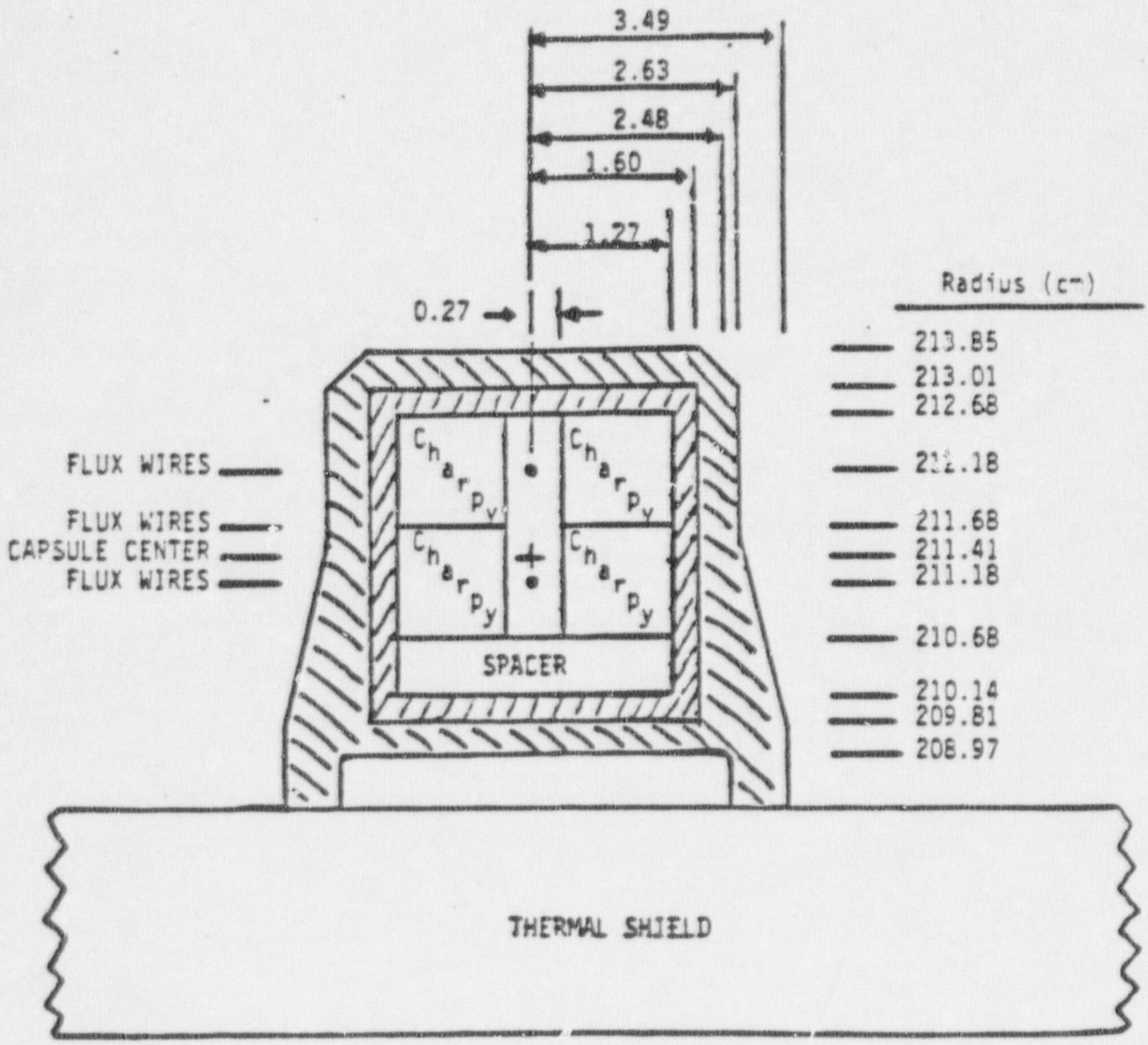


Figure II.1-2. Surveillance Capsule Geometry

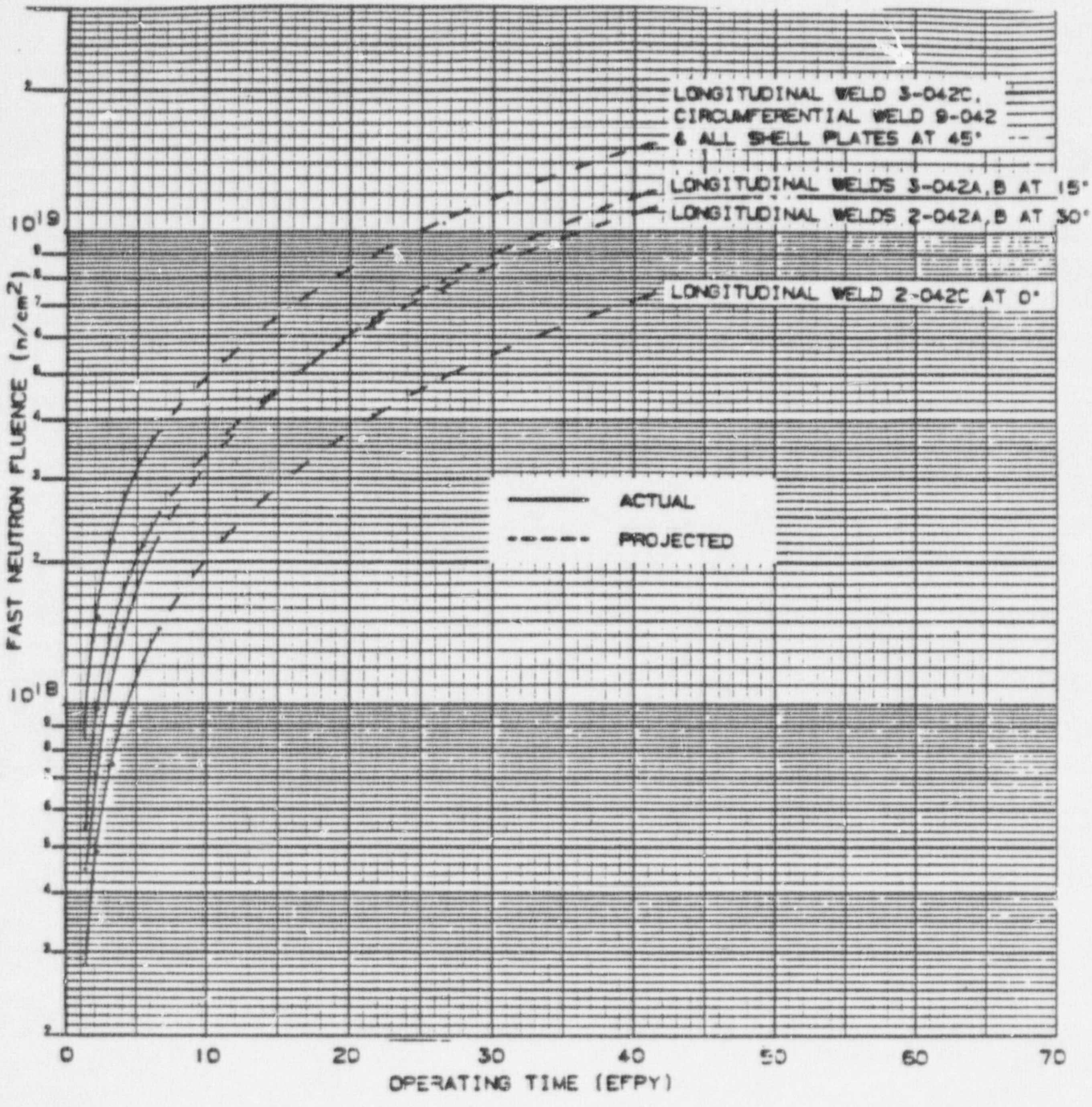


Figure II.2-1. Indian Point Unit 3 Fast Neutron (E > 1.0 Mev) Fluence at the Vessel Beltline as a Function of Operating Time

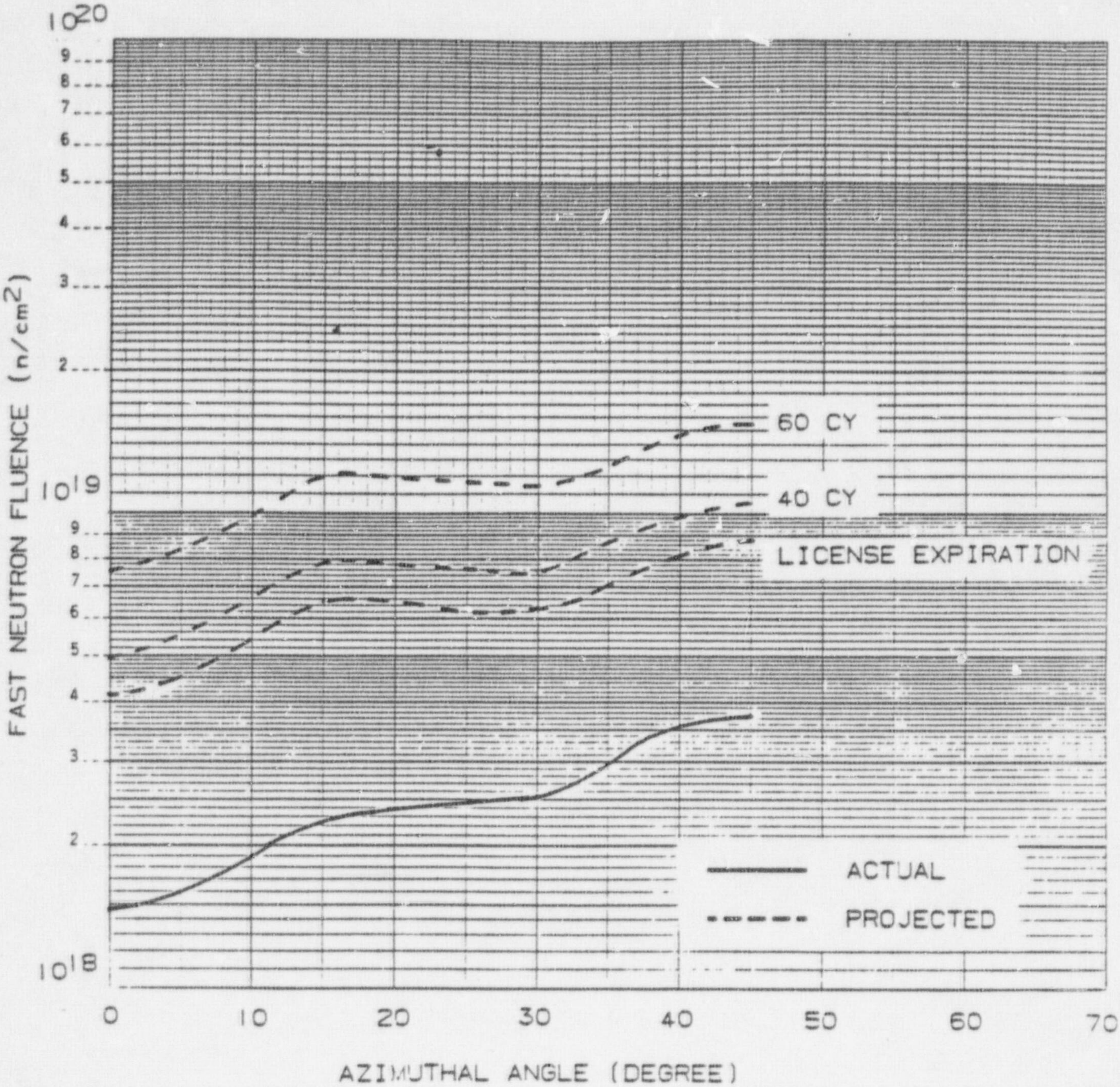


Figure II.2-2. Azimuthal Variation of Fast Neutron (E > 1.0 Mev) Fluence at the Vessel Beltline as a Function of Operating Time

SECTION III

MATERIAL PROPERTIES

For the RT_{PTS} calculation, the best estimate copper and nickel chemical composition of the reactor vessel beltline material is necessary. The material properties for the Indian Point Unit 3 beltline region will be presented in this section.

III.1 IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIALS

The beltline region is defined by the Rule [1] to be "the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron irradiation damage to be considered in the selection of the most limiting material with regard to radiation damage." Figure III.1-1 identifies the location of all beltline region materials for the reactor vessel.

III.2 DEFINITION AND SOURCE OF MATERIAL PROPERTIES FOR ALL VESSEL LOCATIONS

Material property values for the shell plates, which have been docketed with the NRC in Reference 12, were derived from vessel fabrication test certificates. The property data for the welds were derived from weld qualification test records and have also been reported in Reference 15. The tests were performed by the reactor vessel vendor, Combustion Engineering (CE), at the time of fabrication.

Fast neutron irradiation-induced changes in the tensile, fracture, and impact properties of reactor vessel materials are largely dependent on chemical composition, particularly in the copper concentration. The variability in irradiation-induced property changes, which exists in general, is compounded by the variability of copper concentration within the weldments.

For each weld in the Indian Point Unit 3 beltline region. a material data search was performed using the WOG Reactor Vessel Beltline Region Weld Materials Data Base. The WOG data base, which was developed in 1984 and is continually being updated, contains information from weld qualification records (including those for Indian Point Unit 3), surveillance capsule reports, the B&W report BAW-1799, and the Materials Properties Council (MPC) and the NRC Mender MATSURV data bases.

Searches were performed for materials having the identical weld wire heat number as those in the Indian Point Unit 3 vessel, but any combination of wire and flux was allowed. For all of the data found for a particular wire, the copper, nickel, phosphorous and silicon values were averaged and the standard deviations were calculated. Although phosphorous and silicon are not needed for the PTS Rule, they are provided for the sake of completeness. The information obtained from the data base searches is found in Appendix B.

III.3 SUMMARY OF PLANT-SPECIFIC MATERIAL PROPERTIES

A summary of the pertinent chemical and mechanical properties of the beltline region plate and weld materials of the Indian Point Unit 3 reactor vessel are given in Table III.3-1.

All of the initial RT_{NDT} values (I) are given in Table III.3-1. The longitudinal weld initial RT_{NDT} is the generic mean value as defined by the PTS rule [1].

The data in Table III.3-1 is used to evaluate the RT_{PTS} values for the Indian Point Unit 3 reactor vessel.

TABLE III.3-1

INDIAN POINT UNIT 3 REACTOR VESSEL BELTLINE REGION MATERIAL PROPERTIES

	Cu (Wt.%)	Ni (Wt.%)	P (Wt.%)	I[a] (°F)
Intermediate Shell Plate B2802-1:	.20	.50	.010	5
Intermediate Shell Plate B2802-2:	.22	.53	.015	-4
Intermediate Shell Plate B2802-3:	.20	.49	.011	17
Lower Shell Plate B2803-1:	.19	.47	.012	49
Lower Shell Plate B2803-2:	.22	.52	.011	-5
Lower Shell Plate B2803-3:	.24	.52	.012	74
Longitudinal Welds - 2-042 A,B,C and 3-042 A,B,C, Wire Heat 34B009, Flux Linde 1092				
WOG Data Base	.19	1.00	.012	-56
Circum. Weld 9-042 - Intermed. to Lower Shell, Wire Heat 13253, Flux Linde 1092				
Weld Qualification Value	.27	.74	.023	-70
WOG Data Base Mean	.25	.72	.020	-54

(a) All Plate and Circumferential Weld RT_{NDT} (I) values are actual values. The remaining value for the Longitudinal welds is a generic mean value as defined by the PTS rule [1].

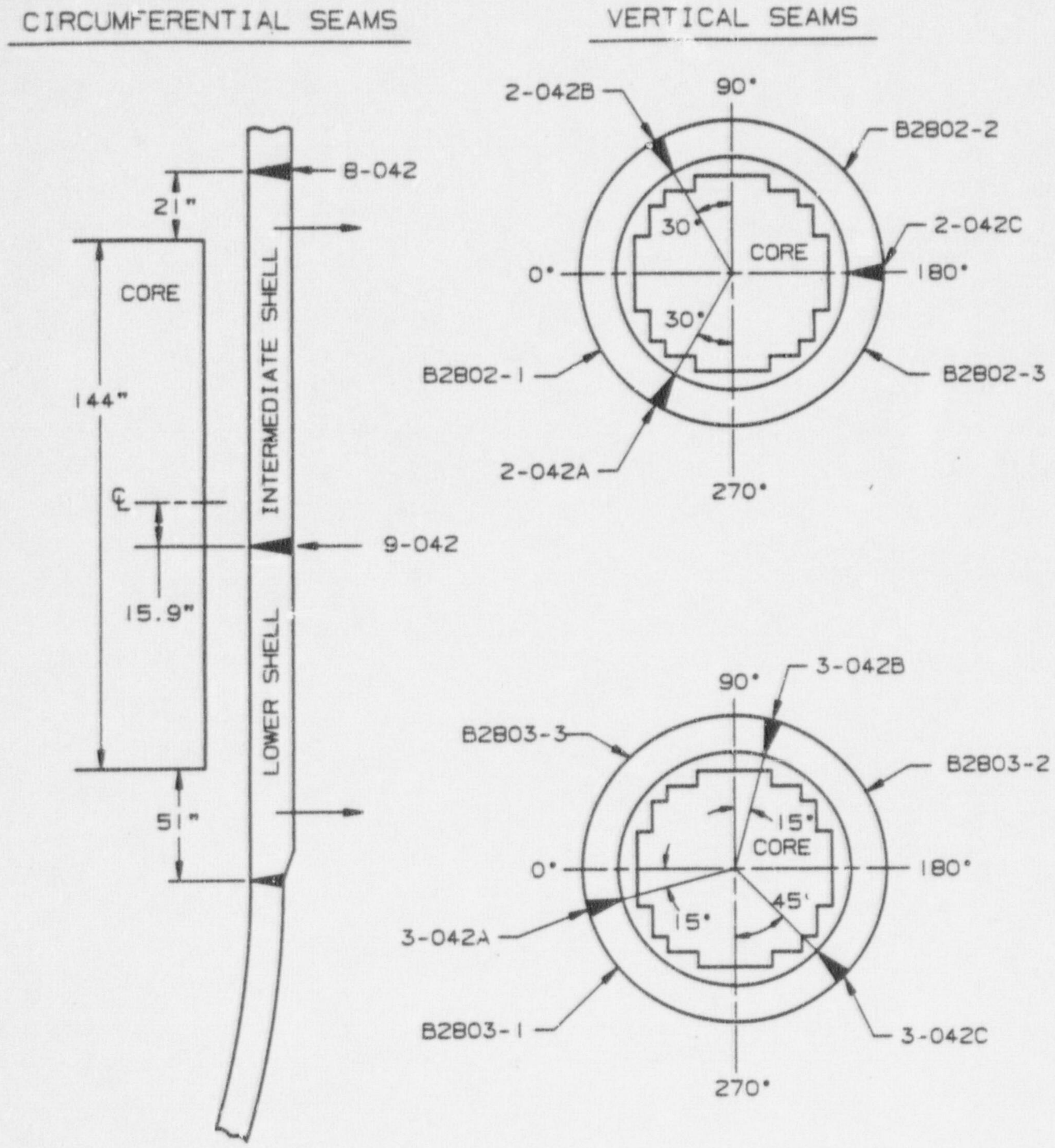


Figure III.1-1 Identification and Location of Beltline Region Material for the Indian Point Unit 3 Reactor Vessel

SECTION IV

DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIALS

Using the methodology prescribed in Section I.2, the results of the fast neutron exposure provided in Section II, and the material properties discussed in Section III, the RT_{PTS} values for Indian Point Unit 3 can now be determined.

IV.1 STATUS OF REACTOR VESSEL INTEGRITY IN TERMS OF RT_{PTS} VERSUS FLUENCE RESULTS

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Indian Point Unit 3 reactor vessel as a function of several fluence values and pertinent vessel lifetimes. The tabulated results from the total evaluation are presented in Appendix C for all beltline region materials for both units.

Figure IV.1-1 presents the RT_{PTS} values for the limiting longitudinal weld, circumferential weld and limiting basemetal of the Indian Point Unit 3 vessel in terms of RT_{PTS} versus fluence* curves. The curves in these figures can be used:

- o to provide guidelines to evaluate fuel reload options in relation to the NRC RT_{PTS} Screening Criterion for PTS (i.e., RT_{PTS} values can be readily projected for any options under consideration, provided fluence is known), and
- o to show the current (6.73 EFPY), and end-of-license (21.88 EFPY) RT_{PTS} values using actual and projected fluence.

---

*The EFPY can be determined using Figure II.2-1.

Table IV.1-1 provides a summary of the RT_{PTS} values for all beltline region materials for the lifetime of interest.

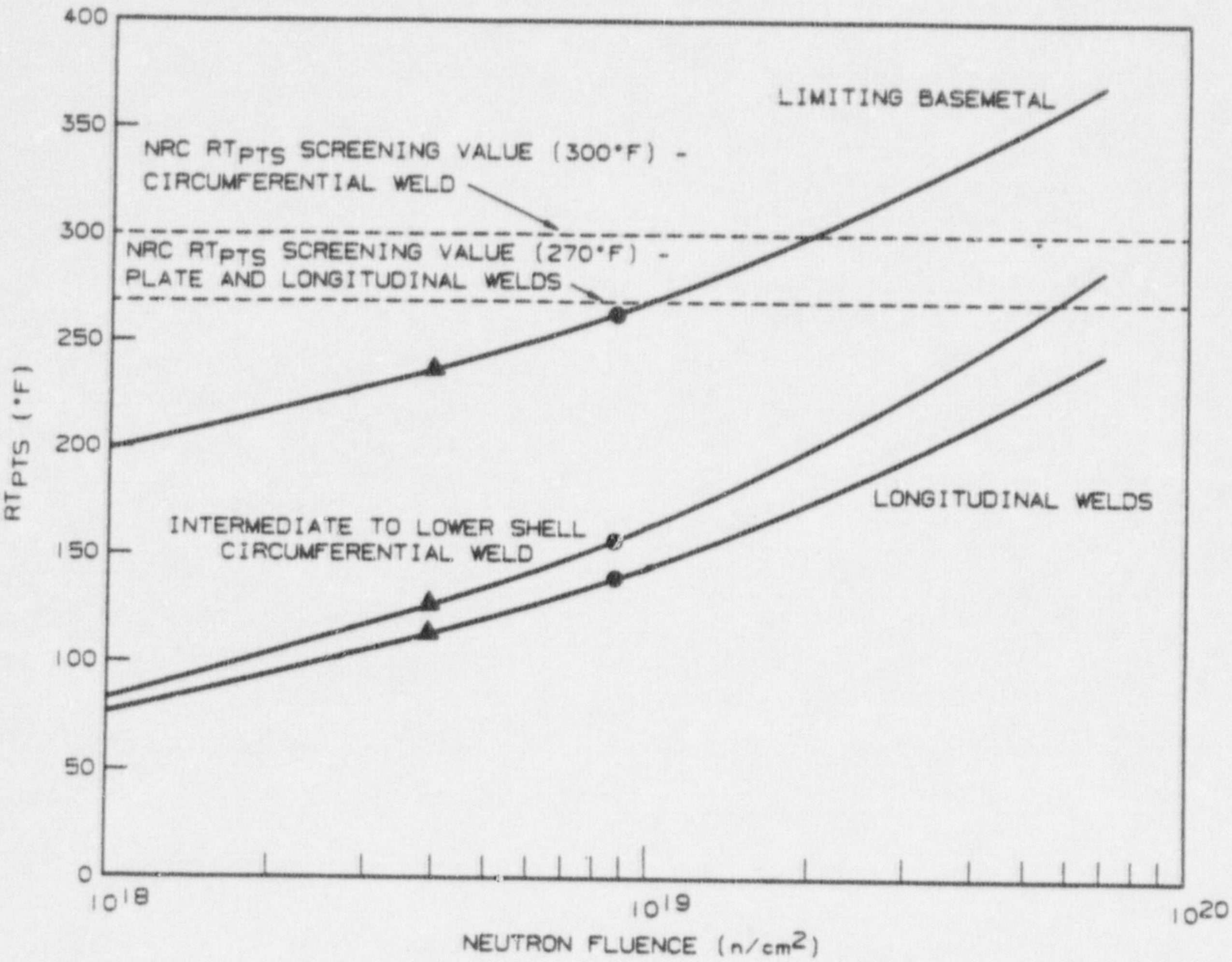
IV.2 DISCUSSION OF RESULTS

As shown in Figure IV.1-1 and Table IV.1-1, lower shell plate B2803-3 is the limiting location relative to PTS. At license expiration, plate B2803-3 is seen to have an RT_{PTS} value of 264°F. All of the RT_{PTS} values, including that for plate B2803-3, are below the NRC screening criteria through license expiration.

TABLE IV.1-1

RT_{PTS} VALUES FOR INDIAN POINT UNIT 3

Location	Vessel Material	RT_{PTS} Values (°F) Present (6.73 EFPY)	End-of-License (21.88 EFPY)	Screening Criteria
1	Intermediate shell plate B2802-1	145	168	270
2	Intermediate shell plate B2802-2	147	174	270
3	Intermediate shell plate B2802-3	156	179	270
4	Lower shell plate B2803-1	182	204	270
5	Lower shell plate B2803-2	146	172	270
6	Lower shell plate B2803-3	235	264	270
7	Limiting Longitudinal Weld 3-042C	115	144	270
8	Intermediate to lower shell circumferential weld 9-042	125	159	300
-	Longitudinal weld 2-042C	90	118	270
-	Longitudinal weld 3-042A,B	100	133	270
-	Longitudinal weld 2-042A,B	104	132	270



LEGEND: Δ = Current Life (6.73 EFPY) and • = End of License 21.88 EFPY) RT_{PTS} Values Using Plant Specific and Projected Plant Specific Fluence Values (Limiting Fluences Used for all Locations)

Figure IV.1-1 Indian Point Unit 3 RT_{PTS} Curves Per PTS Rule Method [1] Docketed Base Material and WOG Data Base Mean Material Properties

SECTION V
CONCLUSIONS AND RECOMMENDATIONS

Calculations have been completed in order to submit RT_{PTS} values for the Indian Point Unit 3 reactor vessel in meeting the requirements of the NRC Rule for Pressurized Thermal Shock [1]. This work entailed a neutron exposure evaluation and a reactor vessel material study in order to determine the RT_{PTS} values.

Detailed fast neutron exposure evaluations using plant specific cycle by cycle core power distributions and state-of-the-art neutron transport methodology have been completed for the Indian Point Unit 3 reactor vessel. Explicit calculations were performed for the first fuel cycles. Projection of the fast neutron exposure beyond Cycle 6 was based on continued implementation of low leakage fuel management similar to that employed during cycle 7.

Plant specific evaluations have demonstrated that during fuel cycles using out-in-in fuel management, the maximum fast neutron (E > 1.0 MeV) flux incident on the reactor vessel was, on the average, 12 percent less than predictions based on the design basis core power distributions. With regard to the low leakage fuel management strategy in place at Indian Point Unit 3, the plant specific evaluations have shown that the average fast neutron (E > 1.0 MeV) flux at the 45° azimuthal position (peak location) was reduced by 2C percent relative to that prior to the implementation of low leakage.

It should be noted that significant deviations from the low leakage scheme already in place will affect the exposure projections beyond the current operating cycle. A move toward a more severe form of low leakage (lower relative power on the periphery) would reduce the projection. On the other hand, a relaxation of the loading pattern toward higher relative power on the core periphery would increase the projections beyond those reported. As each future fuel cycle evolves, the loading patterns should be evaluated to determine their potential impact on projections made in this report.

The fast neutron fluence values from the plant specific calculations have been compared directly with measured fluence levels derived from neutron dosimetry contained in surveillance capsules withdrawn from Indian Point Unit 3. The ratio of measured to calculated fluence values ranges from 1.058 to 1.124 for the three capsule data points. This reasonably good agreement between calculation and measurement supports the use of this analytical approach to perform a plant specific evaluation for the Indian Point Unit 3 reactor.

Material property values for the Indian Point Unit 3 reactor vessel beltline region components were determined. The pertinent chemical and mechanical properties for the shell plates remain the same as those that were originally reported in the vessel fabrication test certificates. The weld material properties are obtained from the WOG Material Data Base.

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Indian Point Unit 3 reactor vessel as a function of several fluence values and pertinent vessel lifetimes. All of the RT_{PTS} values remain below the NRC screening values for PTS using the projected fluence exposure through the expiration date of the operating license. The most limiting value at end-of-license is 264°F for the lower shell plate B2803-3. A review of the material properties of shell plate B2803-3 indicates that a high initial value of RT_{NDT} is the primary factor causing the high RT_{PTS} value.

The results in this report are provided to enable New York Power Authority to comply with the requirements of the PTS rule.

SECTION VI

REFERENCES

1. Nuclear Regulatory Commission, 10CFR Part 50, "Analysis of Potential Pressurized Thermal Shock Events," Federal Register, Vol. 50, No. 141, July 23, 1985.

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. WCAP-8360, "Core Physics Characteristics of the Indian Point Nuclear Plant Unit III, Cycle 1," P. J. Sipush, et al., October 1974. (Westinghouse Proprietary)

5. WCAP-9244, "The Nuclear Design and Core Management of the Indian Point Plant Unit No. 3, Cycle 2," D. M. Lucoff, et al., January 1978. (Westinghouse Proprietary)

6. WCAP-9599, "The Nuclear Design and Core Management of the Indian Point Plant Unit No. 3, Cycle 3," J. A. Penkrot, et al., September 1979. (Westinghouse Proprietary)

7. WCAP-10051, "The Nuclear Design and Core Management of the Indian Point Plant Unit No. 3, Cycle 4," M. A. Kotun and M. F. Muenks, March 1982. (Westinghouse Proprietary)

8. WCAP-10839, "The Nuclear Design and Core Management of the Indian Point Plant Unit No. 3, Cycle 5," M. A. Kotun and R. H. Pitulski, June 1985. (Westinghouse Proprietary)

9. WCAP-11537 "Nuclear Parameters and Operations Package for Indian Point Unit 3 Nuclear Power Plant Cycle 6," F. J. Silva, et al., July 1987. (Westinghouse Proprietary)

10. WCAP-12169 "Nuclear Parameters and Operations Package for Indian Point Unit 3 Nuclear Power Plant Cycle 7," T. Downs, J. Cole and F. Silva, April 1989. (Westinghouse Proprietary)

11. NUREG/CR-1861, "LWR Pressure Vessel Surveillance Dosimetry Improvement Program - PCA Experiment and Blend Tests," W. N. McElroy, July 1981.

12. WCAP-9491, "Analysis of Capsule T from the Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," J. A. Davidson, S. L. Anderson and W. T. Kaiser, April 1979.

13. WCAP-10300, "Analysis of Capsule Y from the Power Authority of the State of New York, Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S. E. Yanichko, S. L. Anderson and W. T. Kaiser, March 1983.

14. WCAP-11815, "Analyses of Capsule Z from the New York Power Authority Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," S. E. Yanichko, S. L. Anderson, and L. Albertin, March 1988.

15. Letter from W. J. Cahill, Jr., of Consolidated Edison to the Director of Nuclear Reactor Regulation, Mr. R. W. Reid, Docket 50-286, March 8, 1978.

APPENDIX A

CORE POWER DISTRIBUTIONS

Core power distributions used in the plant specific fast neutron exposure analysis of the Indian Point Unit 3 reactor vessel were derived from the following fuel cycle nuclear design reports:

Fuel Cycle	Nuclear Design Report
1	WCAP-8360
2	WCAP-9244
3	WCAP-9599
4	WCAP-10051
5	WCAP-10839
6	WCAP-11537
7	WCAP-12169

A schematic diagram of the core configuration applicable to Indian Point Unit 3 is shown in Figure A-1. Cycle averaged relative assembly powers for each fuel cycle 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. Three regions consisting of subsets of fuel assemblies are defined. In performing the adjoint evaluations, the relative power in the fuel assemblies comprising Region 3 has been adjusted to account for known biases in the prediction of power in the peripheral fuel assemblies while the relative power in the fuel 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 the fuel assemblies comprising Region 1 do not contribute significantly to the neutron exposure of either the surveillance capsules or the reactor vessel. Therefore, core power distribution data for fuel assemblies in Region 1 are not listed in Table A-1.

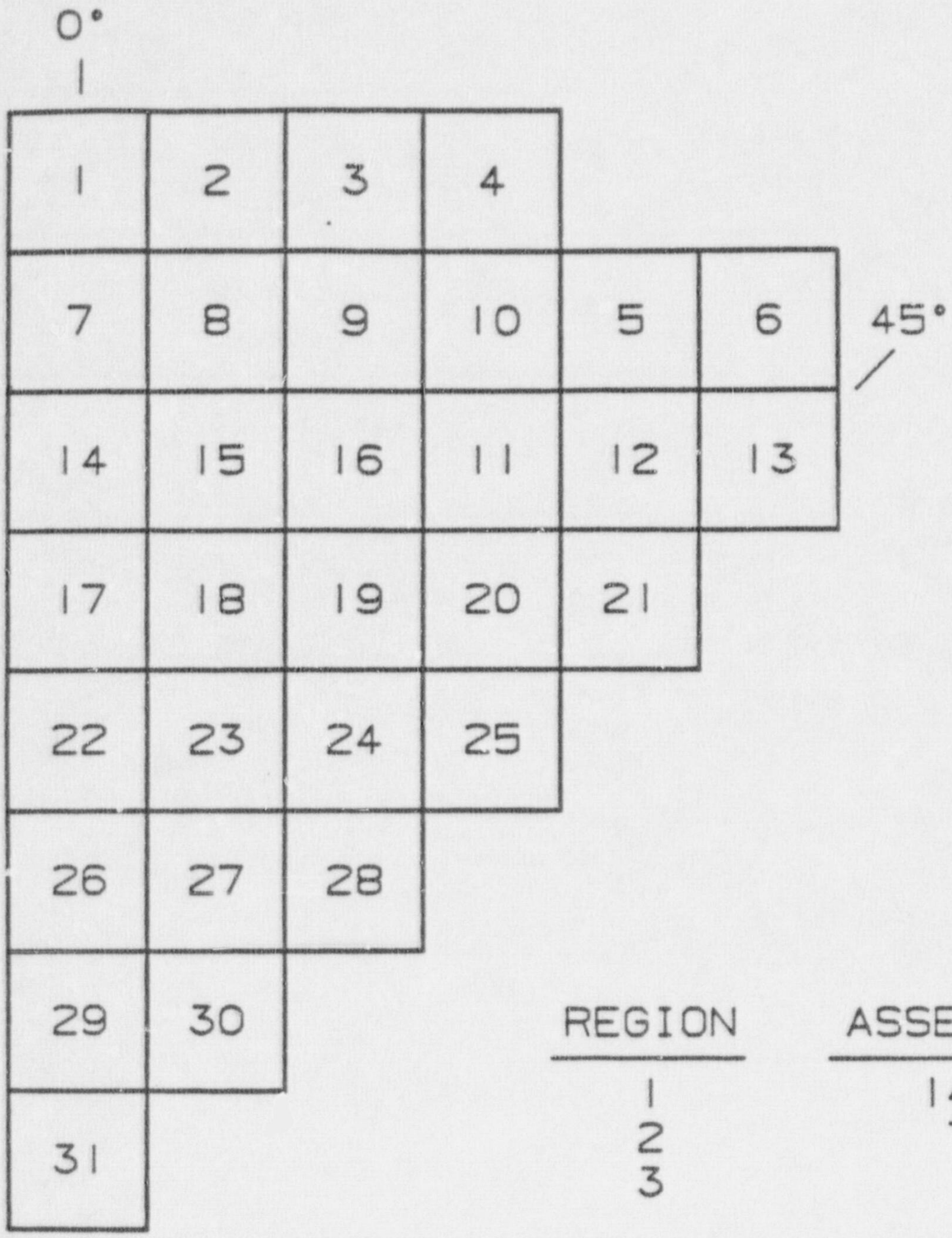
In each of the adjoint evaluations, within assembly spatial gradients have been superimposed on the average assembly power levels. For the peripheral assemblies (Region 3), these spatial gradients also include adjustments to account for analytical deficiencies that tend to occur near the boundaries of the core region.

TABLE A-1

INDIAN POINT UNIT 3
CORE POWER DISTRIBUTIONS USED IN THE FLUENCE ANALYSIS

Fuel Assembly	Design Basis Relative Power	Cycle Averaged Relative Assembly Power, Fuel Cycle 1	2	3	4	5	6	7
1	1.06	0.78	0.93	1.04	0.95	0.75	0.66	0.60
2	1.09	0.84	0.94	1.02	0.94	0.90	0.83	0.62
3	1.01	0.74	0.88	0.95	0.89	0.79	0.65	0.81
4	0.81	0.63	0.70	0.76	0.52	0.40	0.45	0.46
5	1.15	0.88	0.99	0.98	0.90	0.68	0.63	0.61
6	0.75	0.54	0.72	0.61	0.44	0.34	0.31	0.21
7	1.02	1.01	0.99	1.08	1.20	1.10	1.10	0.87
8	1.10	1.00	1.23	1.20	1.15	1.22	1.26	1.24
9	1.00	0.98	0.96	0.95	1.21	1.05	1.02	1.05
10	1.05	0.98	1.16	1.23	1.12	1.05	1.08	1.14
11	1.07	1.06	1.01	0.93	1.21	1.09	1.08	1.18
12	1.00	0.94	1.07	0.97	1.09	1.13	1.15	1.18
13	1.05	0.96	1.16	0.70	0.87	0.79	0.74	0.60

Note: Refer to Figure A-1 for fuel assembly location.



REGION	ASSEMBLIES
1	14-31
2	7-13
3	1-6

Figure A-1 Indian Point Unit 3 Core Description for Power Distribution Map

APPENDIX B

WELD CHEMISTRY

Table B.1-1 provides the weld data output from the WOG Material Data Base. The pertinent material chemical compositions are given, along with the wire/flux identification. The mean chemistry values and the population standard deviation are then calculated. The mean values of copper and nickel are used in the RT_{PTS} analysis.

Weld Chemistry Data Source and Plant:

AEP	-	Donald C. Cook Unit 1
Cu	-	Weight % of Copper
INT	-	Indian Point Unit 3
KEP	-	Mihama Unit 1
Ni	-	Weight % of Nickel
P	-	Weight % of Phosphorous
PGE	-	Diablo Canyon Unit 1
PNJ	-	Salem Unit 2
PSE	-	Salem Unit 1
SC	-	Surveillance Capsule
Si	-	Weight % of Silicon
WQ	-	Weld Qualification

TABLE B-1

INDIAN POINT UNIT 3 INTERMEDIATE TO LOWER SHELL

WELD CHEMISTRY FROM WOG MATERIALS DATA BASE

WIRE HEAT NUMBER 13253

ID	WIRE HEAT	WIRE TYPE	FLUX TYPE	FLUX LOT	WELDCHEM DATA SOURCE	Cu	Ni	P	Si	PLANT	DESCRIPTION
0290	13253	B-4 MOD	LINDE 1092	3774	PNJ,SC	0.230	0.710	0.017	0.290	PGE	Nozzle to Inter Shell
0294	13253	B-4 MOD	LINDE 1092	3791	AEP,SC	0.270	0.740	0.023	0.180	AEP	Surveillance Weld
0294	13253									INT	Inter to Lower Shell
0294	13253									KEP	Nozzle to Inter Shell
0294	13253									PSE	Inter to Lower Shell
mean						0.250000	0.725000	0.020000	0.235000		
std. dev.						0.028284	0.021213	0.004243	0.077782		

3786s/061089:10

APPENDIX C

RT_{PTS} VALUES OF INDIAN POINT UNIT 3 REACTOR VESSEL BELTLINE REGION MATERIALS

Tables C.1-1 through C.1-3 provide the RT_{PTS} values, as a function of both constant fluence and constant EFPY (assuming the projected fluences values), for all beltline region materials of the Indian Point Unit 3 reactor vessel. The RT_{PTS} values are calculated in accordance with the PTS rule, which is Reference [1] in the main body of this report. The vessel location numbers in the following tables correspond to the vessel materials identified below and in Table III.3-1 of the main report.

Location	Vessel Material
1	Intermediate shell plate B2802-1
2	Intermediate shell plate B2802-2
3	Intermediate shell plate B2802-3
4	Lower shell plate B2803-1
5	Lower shell plate B2803-2
6	Lower shell plate B2803-3
7	Limiting Longitudinal Weld 3-042C
8	Intermediate to lower shell Circumferential Weld 9-042

TABLE C.1-1

RTPTS VALUES FOR THE INDIAN POINT UNIT 3 REACTOR

VESSEL BELTLINE REGION MATERIALS AT VARIOUS FLUENCES

								RTPTS VALUE AT FLUENCE			
ID	PLANT	CU	NI	P	I	VALUE	TYPE	.10E+19	.50E+19	.10E+20	.20E+20
1	INT	0.20	0.50	0.010	5	ACTUAL	B.M.	117	152	172	196
2	INT	0.22	0.53	0.015	-4	ACTUAL	B.M.	116	155	178	206
3	INT	0.20	0.49	0.011	17	ACTUAL	B.M.	129	163	183	208
4	INT	0.19	0.47	0.012	49	ACTUAL	B.M.	156	189	208	230
5	INT	0.22	0.52	0.011	-5	ACTUAL	B.M.	115	154	176	204
6	INT	0.24	0.52	0.012	74	ACTUAL	B.M.	201	243	268	299
7	INT	0.19	1.00	0.012	-56	GENERIC	L.W.	81	124	149	179
8	INT	0.25	0.72	0.020	-54	ACTUAL	C.W.	86	135	165	200

Notes: ID = Location of vessel material (see page C-1)

I = Initial value of RTNDT, actual or GENERIC

Value = "ACTUAL" or "GENERIC" denotes type of initial RTNDT value

B.M. = Base Metal

L.W. = Longitudinal Weld

C.W. = Circumferential Weld

Reference temperatures are in deg F

3786s/061089:10

TABLE C.1-1 (CONT)

RTPTS VALUES FOR THE INDIAN POINT UNIT 3 REACTOR

VESSEL BELTLINE REGION MATERIALS AT VARIOUS FLUENCES

ID	PLANT	CU	NI	P	I	VALUE	TYPE	RTPTS VALUE AT FLUENCE .40E+20	.60E+20	.70E+20
1	INT	0.20	0.50	0.010	5	ACTUAL	B.M.	226	246	254
2	INT	0.22	0.53	0.015	-4	ACTUAL	B.M.	239	262	271
3	INT	0.20	0.49	0.011	17	ACTUAL	B.M.	237	257	265
4	INT	0.19	0.47	0.012	49	ACTUAL	B.M.	253	276	284
5	INT	0.22	0.52	0.011	-5	ACTUAL	B.M.	237	259	269
6	INT	0.24	0.52	0.012	74	ACTUAL	B.M.	335	360	370
7	INT	0.19	1.00	0.012	-56	GENERIC	L.W.	215	240	250
8	INT	0.25	0.72	0.020	-54	ACTUAL	C.W.	242	271	282

3786s/061089:10

TABLE C.1-2

RTPTS VALUES FOR THE INDIAN POINT UNIT 3 REACTOR VESSEL
BELTLINE REGION MATERIALS AT CURRENT LIFE (6.73 EFPY)
FLUENCE VALUES

ID	PLANT	CU	NI	P	I	VALUE	TYPE	FLUENCE	RTPTS
1	INT	0.20	0.50	0.010	5	ACTUAL	B.M.	0.38E+19	145
2	INT	0.22	0.53	0.015	-4	ACTUAL	B.M.	0.38E+19	147
3	INT	0.20	0.49	0.011	17	ACTUAL	B.M.	0.38E+19	156
4	INT	0.19	0.47	0.012	49	ACTUAL	B.M.	0.38E+19	182
5	INT	0.22	0.52	0.011	-5	ACTUAL	B.M.	0.38E+19	146
6	INT	0.24	0.52	0.012	74	ACTUAL	B.M.	0.38E+19	235
7	INT	0.19	1.00	0.012	-56	GENERIC	L.W.	0.38E+19	115
8	INT	0.25	0.72	0.020	-54	ACTUAL	C.W.	0.38E+19	125

TABLE C.1-3

RTPTS VALUES FOR THE INDIAN POINT UNIT 3 REACTOR VESSEL

BELTLINE REGION MATERIALS AT END OF LIFE (21.88 EFPY)

FLUENCE VALUES

ID	PLANT	CU	NI	P	I	VALUE	TYPE	FLUENCE	RTPTS
1	INT	0.20	0.50	0.010	5	ACTUAL	B.M.	0.88E+19	168
2	INT	0.22	0.53	0.015	-4	ACTUAL	B.M.	0.88E+19	174
3	INT	0.20	0.49	0.011	17	ACTUAL	B.M.	0.88E+19	179
4	INT	0.19	0.47	0.012	49	ACTUAL	B.M.	0.88E+19	204
5	INT	0.22	0.52	0.011	-5	ACTUAL	B.M.	0.88E+19	172
6	INT	0.24	0.52	0.012	74	ACTUAL	B.M.	0.88E+19	264
7	INT	0.19	1.00	0.012	-56	GENERIC	L.W.	0.88E+19	144
8	INT	0.25	0.72	0.020	-54	ACTUAL	C.W.	0.88E+19	159