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WCAP-12795, Rev. 3

Reactor Cavity Neutron Measurement Program
for
Wisconsin Electric Power Company
Point Beach Unit 2

Stanwood L. Anderson
August 1995

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EXECUTIVE SUMMARY

At the conclusion of Fuel Cycle 14, a reactor cavity measurement program was instituted at Point Beach Unit 2 to provide a continuous monitoring of the reactor pressure vessel and reactor vessel support structure. The use of the cavity measurement program coupled with available surveillance capsule measurements provides a plant specific data base that enables the evaluation of the vessel exposure and the uncertainty associated with that exposure over the service life of the unit.

During the first cycle of irradiation with cavity dosimetry installed (Cycle 15), the reactor was operating with a conventional low leakage fuel management strategy. At the onset of Cycle 16 additional neutron flux reduction at the beltline circumferential weld was achieved by the introduction of part length hafnium absorbers in selected peripheral fuel assemblies. A direct comparison of the Cycles 15, 16, 17, and 18 through 20 cavity measurements demonstrated the following incremental flux reduction for the circumferential weld:

MEASURED CAVITY FLUX (E > 1.0 MeV)
[n/cm^2-sec]

	CYCLE 15	CYCLE 16	CYCLE 17	CYCLES 18/20
0°	1.87E+09	1.35E+09	1.40E+09	1.32E+09
15°	1.80E+09	1.28E+09	1.21E+09	1.26E+09
30°	1.32E+09	1.02E+09	1.05E+09	1.18E+09
45°	1.12E+09	1.03E+09	9.76E+08	9.87E+08

MEASURED REDUCTION FACTOR

	CYCLE 15	CYCLE 16	CYCLE 17	CYCLES 18/20
0°	1.00	0.722	0.749	0.706
15°	1.00	0.711	0.672	0.700
30°	1.00	0.773	0.795	0.894
45°	1.00	0.920	0.871	0.881

Due to the relatively short axial extent of the hafnium inserts, the flux reduction impact on the intermediate and lower shell forgings is less dramatic than in the case of the circumferential weld.

Based on the continued use of the current (average of Cycles 16 through 20) fuel loading pattern with the part length hafnium absorbers, the projected maximum fast neutron exposure of the vessel beltline materials at 32 and 48 effective full power years of operation is summarized as follows:

	NEUTRON FLUENCE (E > 1.0 MeV) [n/cm^2] 32 EFPY	48 EFPY
Beltline Circumferential Weld	2.49E+19	3.42E+19
Intermediate Shell Forging	3.01E+19	4.32E+19
Lower Shell Forging	2.52E+19	3.59E+19
Upper/Int. Shell Circ. Weld	5.48E+18	7.84E+18
Lower Shell/Head Circ. Weld	<1.00E+17	<1.00E+17

As further data are accumulated from subsequent irradiations, the neutron environment in the vicinity of the Unit 2 pressure vessel will become better characterized and the uncertainties in the vessel exposure projections will be reduced. Thus, the measurement program will permit the assessment of vessel condition to be based on realistic exposure levels with known uncertainties and will eliminate the need for any unnecessary conservatism in the determination of vessel operating parameters.

In addition, the excellent three-dimensional fluence profiles established by the measurements, enables the true effects of three-dimensional and potentially non-symmetric flux reduction measures to be accurately accounted for in a manner that would be difficult using analysis alone.

All of the calculations and dosimetry evaluations presented in this report have been based on the latest available nuclear cross-section data derived from ENDF/B-VI and are intended to be consistent with the requirements of Draft Regulatory Guide DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence". As such, the data provided here are intended to supersede prior evaluations documented in References 3, 4, and 5.

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SECTION 1.0

OVERVIEW OF THE PROGRAM

The Reactor Cavity Neutron Measurement Program[1] initiated at Point Beach Unit 2 at the start of Fuel Cycle 15 was designed to provide a mechanism for the long term continuous monitoring of the neutron exposure of those portions of the reactor vessel and vessel support structure which may experience radiation induced increases in reference nil ductility transition temperature (RT_{NDT}) over the nuclear power plant lifetime. When used in conjunction with dosimetry from internal surveillance capsules[2] and with the results of neutron transport calculations, the reactor cavity neutron dosimetry provides an extensive plant specific measurement data base that can be used to determine the best estimate neutron exposure of the pressure vessel and to project embrittlement gradients through the vessel wall with a minimum uncertainty. Minimizing the uncertainty in the neutron exposure projections will, in turn, help to assure that the reactor can be operated in the least restrictive mode possible with respect to

1 - 10CFR50 Appendix G pressure/temperature limit curves for normal heatup and cooldown of the reactor coolant system.

2 - Emergency Response Guideline (ERG) pressure/temperature limit curves.

3 - Pressurized Thermal Shock (PTS) RT_{PTS} screening criteria.

In addition, an accurate measure of the neutron exposure of the reactor vessel and support structure can provide a sound basis for requalification should operation of the plant beyond the current design and/or licensed lifetime prove to be desirable.

In the assessment of the state of embrittlement of light water reactor pressure vessels, an accurate evaluation of the neutron exposure of the materials comprising the beltline region of the vessel is required. This exposure evaluation must, in general, include assessments not only at locations of maximum exposure at the inner diameter of the vessel, but, also, as a function of axial, azimuthal, and radial location throughout the vessel wall.

A schematic of the beltline region of the Point Beach Unit 2 reactor pressure vessel is provided in Figure 1-1. In this case, the beltline region is constructed of two (2) shell forgings and two (2) circumferential welds; one joining the intermediate and lower shell forgings and one joining the intermediate shell forging to the nozzle shell course. Each of these four materials must be considered in the overall embrittlement assessments of the pressure vessel.

In order to satisfy the requirements of 10CFR50 Appendix G for the calculation of pressure/temperature limit curves for normal heatup and cooldown of the reactor coolant system, fast neutron exposure levels must be defined at depths within the vessel wall equal to 25 and 75 percent of the wall thickness for each of the materials comprising the beltline region. These locations are commonly referred to as the 1/4T and 3/4T positions in the vessel wall. The 1/4T exposure levels are also used in the determination of upper shelf fracture toughness as specified in 10CFR50 Appendix G.

In the determination of values of RT_{PTS} for comparison with applicable pressurized thermal shock screening criteria for plates and circumferential welds, maximum neutron exposure levels experienced by each of the beltline materials are required. These maximum levels will, of course, occur at the vessel inner radius.

In the event that a probabalistic fracture mechanics evaluation of the pressure vessel is performed, or if an evaluation of thermal annealing and subsequent material re-embrittlement is undertaken, a complete embrittlement profile is required for the entire volume of the pressure vessel beltline. The determination of this embrittlement profile would, in turn, necessitate the evaluation of neutron exposure gradients throughout the entire beltline.

The methodology used to provide these required best estimate neutron exposure evaluations for the Point Beach Unit 2 pressure vessel is based on the underlying philosophy that, in order to minimize the uncertainties associated with vessel exposure projections, plant specific neutron transport calculations must be supported by benchmarking of the analytical approach, comparison with industry wide power reactor data bases of surveillance capsule and reactor cavity dosimetry, and, ultimately, by validation with plant specific surveillance capsule and reactor cavity dosimetry data bases.

That is, as a progression is made from the use of a purely analytical approach tied to experimental benchmarks to an approach that makes use of industry and plant specific power reactor measurements to remove potential biases in the analytical method, knowledge regarding the neutron environment applicable to a specific reactor vessel is increased and the uncertainty associated with vessel exposure projections is minimized.

With this overall methodology in mind, the Reactor Cavity Measurement Program was established to meet the following objectives:

1 - Provide a measurement data base sufficient to:
   a) remove biases that may be present in analytical predictions of neutron exposure; and
   b) support the methodology for the projection of exposure gradients through the thickness of the pressure vessel wall.

2- Establish uncertainties in the best estimate fluence projections for the pressure vessel wall.

3 - Provide a long term continuous monitoring capability for the beltline region of the pressure vessel.

This report provides the results of neutron dosimetry evaluations performed subsequent to the completion of Fuel Cycle 20. Fast neutron exposure in terms of fast neutron fluence (E > 1.0 MeV) and dpa is established for all measurement locations in the reactor cavity. The analytical formalism describing the relationship among the measurement points and locations within the pressure vessel wall is described and used to project the exposure of the vessel itself.

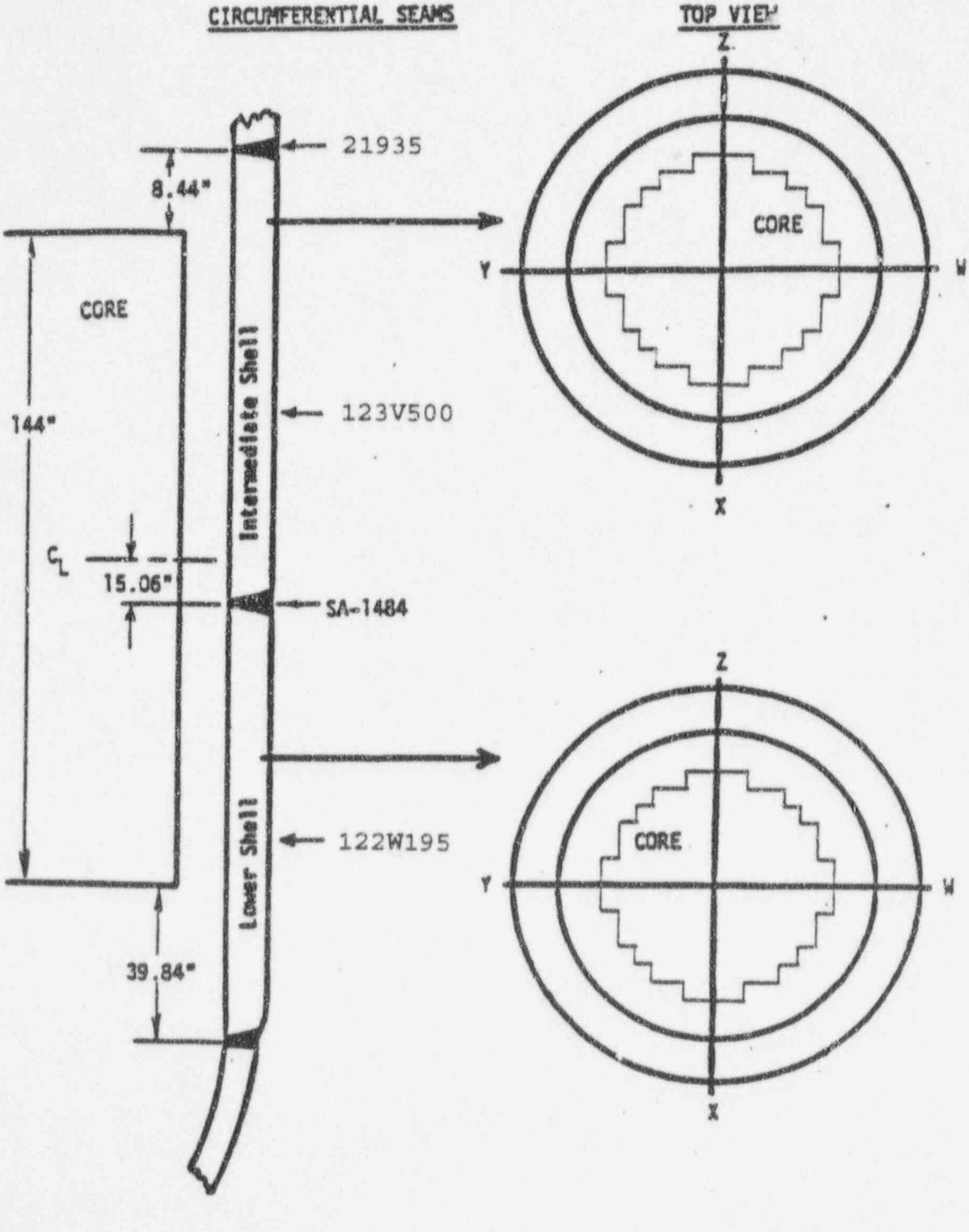
Results of exposure evaluations from surveillance capsule dosimetry withdrawn at the end of Fuel Cycles 1, 3, 5, and 16 as well as cavity dosimetry results from Cycles 15, 16, and 17 are incorporated to provide the integrated exposure of the pressure vessel from plant startup through the end of Cycle 20. Also, uncertainties associated with the derived exposure parameters at the measurement locations and with the projected exposure of the pressure vessel are provided.

In addition to the evaluation of the current exposure of the reactor vessel beltline materials, projections of the future exposure of the vessel are also provided. Current evaluations and future projections are provided for each of the beltline weldments as well as for the forgings comprising both the intermediate and lower shells.

All of the calculations and dosimetry evaluations presented in this report are intended to meet the requirements of Draft Regulatory Guide DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"; and, have been based on the latest available nuclear cross-section data derived from ENDF/B-VI. As such, the data provided here are intended to supersede prior evaluations documented in references 3, 4, and 5.

FIGURE 1.0-1

DESCRIPTION OF PRESSURE VESSEL BELTLINE MATERIALS



SECTION 2.0

DESCRIPTION OF THE MEASUREMENT PROGRAM

2.1 - Description of Reactor Cavity Dosimetry

To achieve the goals of the Reactor Cavity Neutron Measurement Program, comprehensive multiple foil sensor sets including radiometric monitors (RM) and solid state track recorders (SSTR) were installed at several locations in the reactor cavity to characterize the neutron energy spectra within the beltline region of the reactor vessel. In addition, gradient chains were used in conjunction with the encapsulated sensors to complete the azimuthal and axial mapping of the neutron environment over the regions of interest.

Placement of the multiple foil sensor sets was such that spectra evaluations could be made at four azimuthal locations at an axial elevation representative of the midplane of the reactor core. The intent here was to determine changes in spectra caused by varying amounts of water located between the core and the pressure vessel. Due to the irregular shape of the reactor core, water thickness varies significantly as a function of azimuthal angle. In addition to the four midplane sensor sets, two multiple foil packages were positioned opposite the top and bottom of the active core at the azimuthal angle corresponding to the maximum neutron flux. Here the intent was to measure variations in neutron spectra over the the core height; particularly near the top of the fuel where backscattering of neutrons from primary loop nozzles and vessel support structures could produce significant perturbations. At each of the four azimuthal locations selected for core midplane spectra measurements, gradient chains extended over a fourteen foot height centered on the core midplane.

The sensor set deployment described in the preceding paragraphs is characteristic of the basic long term monitoring program designed to provide fast neutron exposure assessments for materials comprising the beltline region of the reactor pressure vessel. During the Cycle 15 irradiation, an additional multiple foil sensor set was included in the vicinity of the reactor vessel supports in order to determine the exposure rate and neutron spectrum at this location well above the beltline region of the reactor vessel. This capsule placement represented a one time measurement that was not repeated as a part of the long term monitoring efforts.

2.1.1 Sensor Placement in the Reactor Cavity

A detailed description of the cavity dosimetry hardware and plant specific installation can be found in Reference 1. However, the following information is provided in this report to orient the reader to the plant geometry and the specifics of the sensor sets.

The placement of the individual multiple foil sensor sets and gradient chains within the reactor cavity is illustrated in Figures 2.1-1 and 2.1-2. In Figure 2.1-1 a plan view of the azimuthal locations of the four strings of sensor sets is depicted. The strings were located at azimuthal positions of 0, 15, 30, and 45 degrees relative to the core cardinal axis. The sensor strings were hung in the annular gap between the pressure vessel insulation and the primary biological shield at a nominal radius of 79 inches relative to the core centerline.

In Figure 2.1-2, the axial extent of each of the sensor set strings is illustrated along with the locations of the multiple foil holders. At the 0 degree azimuth, multiple foil sets were positioned at the core midplane, opposite the top and bottom of the active fuel, and, during Cycle 15 only, at the elevation of the reactor vessel support. At the 15, 30, and 45 degree azimuthal locations, multiple foil sets were positioned only opposite the core midplane. In all cases, stainless steel gradient chains extended $\pm$ 7 feet relative to the midplane of the active core.

The sensor sets and gradient chains were suspended from two support bars mounted on a support frame assembled around the outlet nozzle support shoe of primary loop A. The bottom edges of the support bars were positioned 26.625 inches above the top of the active fuel. The sensor sets and gradient chains were retained and supported at the bottom by chain clamps attached to stainless steel eye nuts with stainless steel threaded chain connectors. The eye nuts were, in turn, atteched to threaded studs embedded in the sump wall. The design of the dosimetry support bars and frames along with the gradient chains and stops ensured correct axial and azimuthal positioning of the dosimetry relative to well known reactor support features.

2.1.2 Description of Irradiation Capsules

The sensor sets used to characterize the neutron spectra within the reactor cavity were retained in 3.87 inch x 1.00 inch x 0.50 inch rectangular aluminum 6061 capsules such as that shown in Figure 2.1-3. Each capsule included three compartments to hold the neutron sensors. The top compartment (position 1) was intended to accomodate bare radiometric monitors and SSTR packages, whereas, the two remaining compartments (positions 2 and 3) were meant to house cadmium shielded packages. The separation between positions 1 and 2 was such that cadmium shields inserted into position 2 did not introduce perturbations in the thermal flux in position 1. Aluminum 6061 was selected for the dosimeter capsules in order to minimize neutron flux perturbations at the sensor set locations as well as to limit the radiation levels associated with post-irradiation shipping and handling of the capsules. A summary of the contents of the multiple foil capsules used during each cycle of irradiation is provided in the appendices to this report.

2.1.3 - Description of Gradient Chains

Along with the multiple foil sensor sets placed at discrete locations within the reactor cavity, gradient chains were employed to obtain axial variations of fast neutron exposure along each of the four traverses. Subsequent to irradiation these gradient chains were removed from the cavity and segmented to provide neutron reaction rate measurements at one foot intervals over the height of the axial traverses. These gradient chains consisted of Type 304 stainless steel bead chain of 0.188 inch diameter.

When coupled with a chemical analysis, the stainless steel gradient chains yielded activation results for the Fe-54(n,p), Ni-58(n,p), and Co-59(n,γ) reactions. The high purity iron, nickel, and cobalt-aluminum foils contained in the multiple foil sensor sets established a direct correlation with the measured reaction rates from the stainless steel chain; and provided an overcheck on the chemical analysis of the type 304 steel.

FIGURE 2.1-1

AZIMUTHAL LOCATION OF SENSOR STRINGS

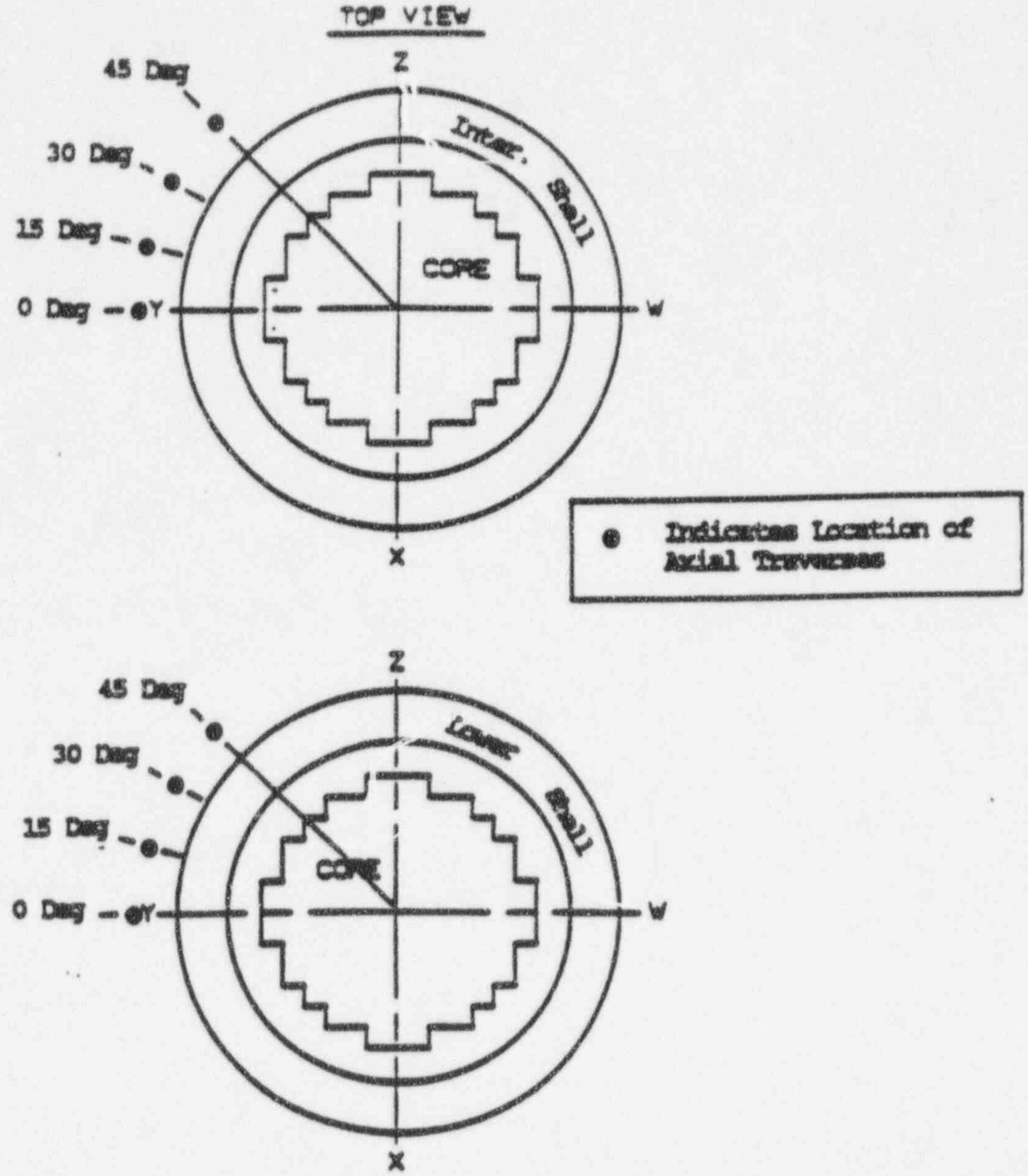


FIGURE 2.1-2

AXIAL LOCATION OF MULTIPLE FOIL SENSOR SETS

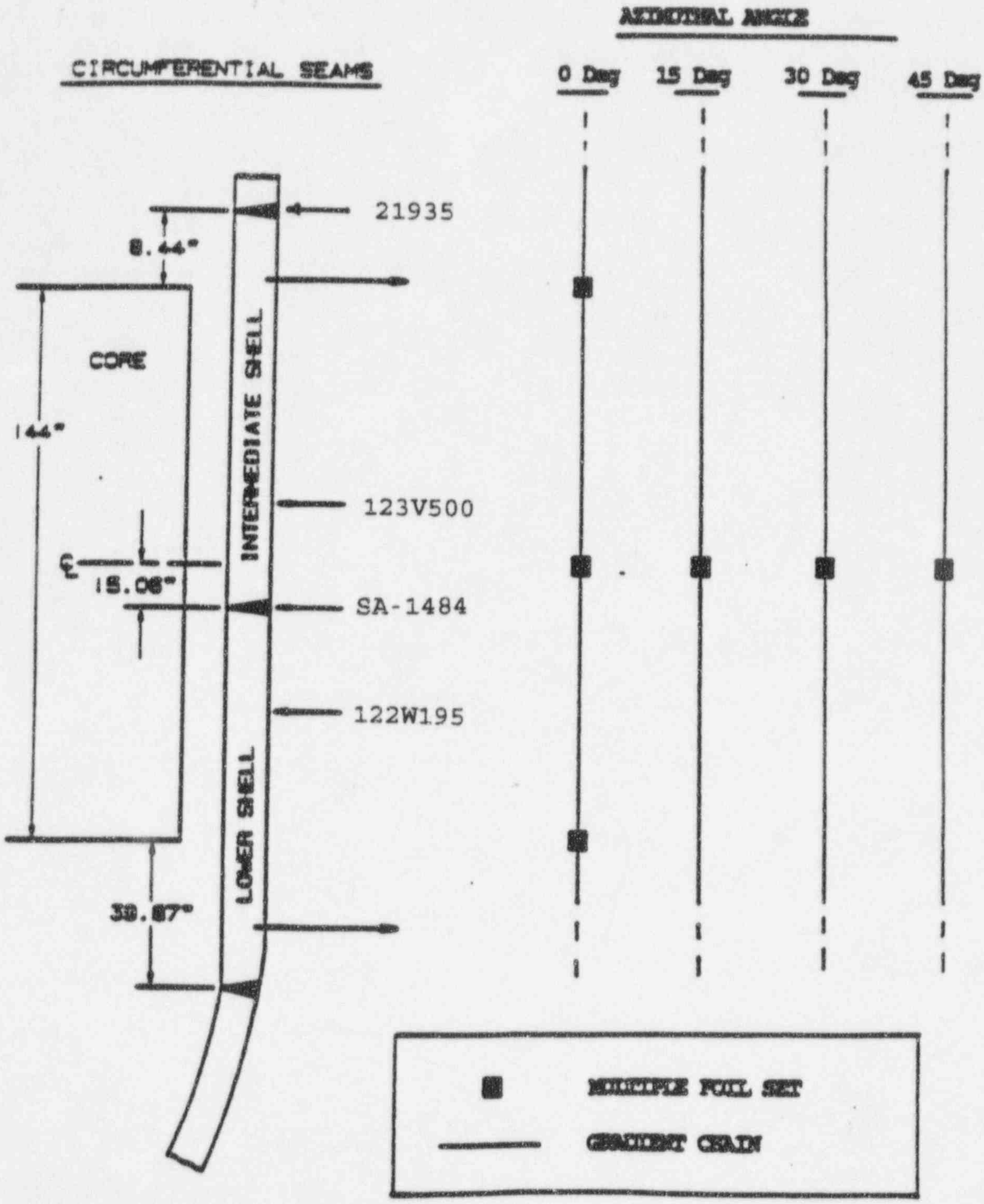
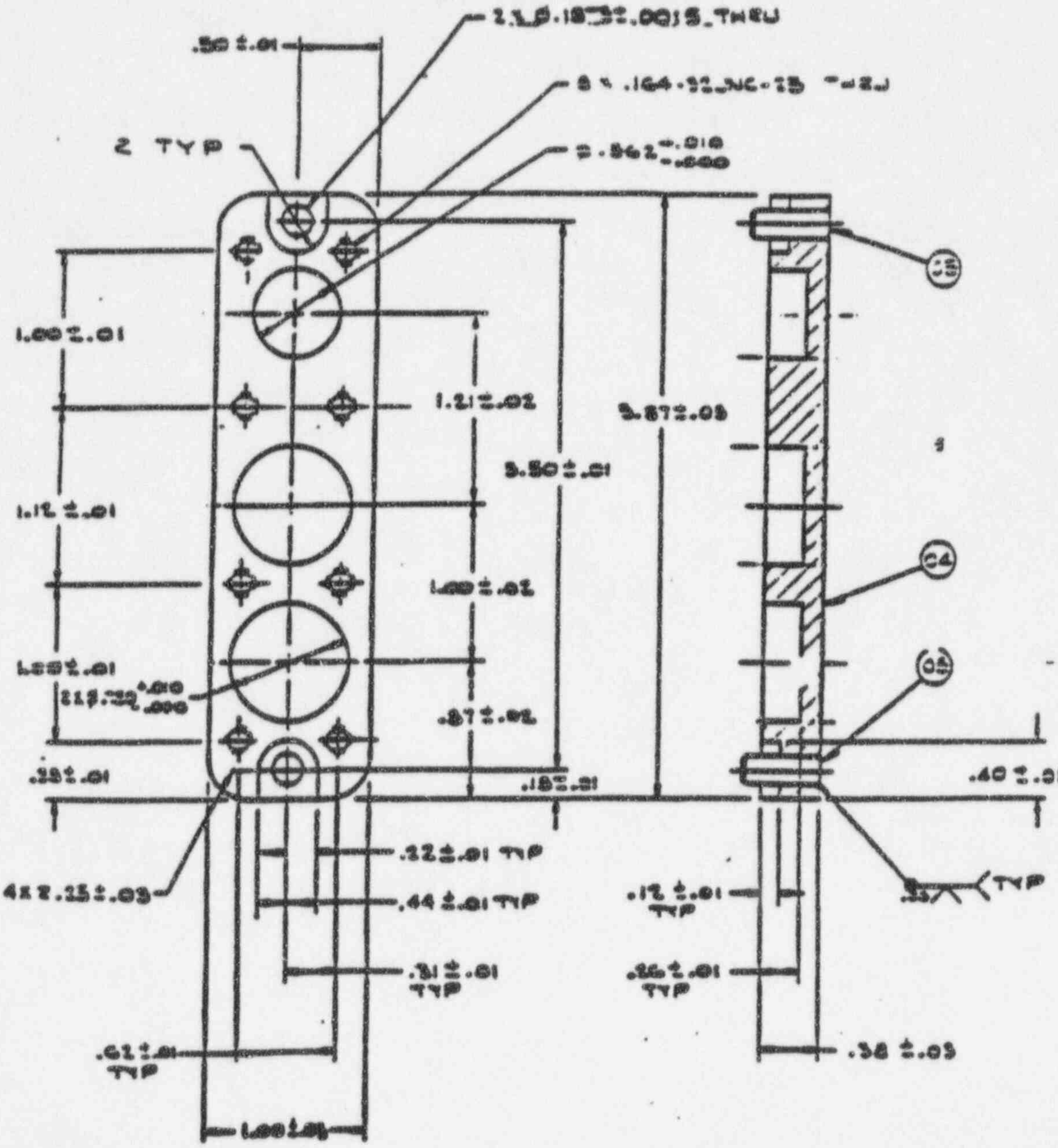


FIGURE 2.1-3

IRRADIATION CAPSULE FOR CAVITY SENSOR SETS



2.2 - Description of Surveillance Capsule Dosimetry

Over the course of the first 16 fuel cycles at Point Beach Unit 2, four materials surveillance capsules were withdrawn from their positions between the thermal shield and the reactor vessel. The neutron dosimetry contained within these capsules provided a measure of the integral exposure received by each of the capsules during its respective irradiation period; and established a measurement continuity between the initial startup of the reactor and the initiation of the Reactor Cavity Measurement Program. The specific withdrawal dates of these four capsules were as follows:

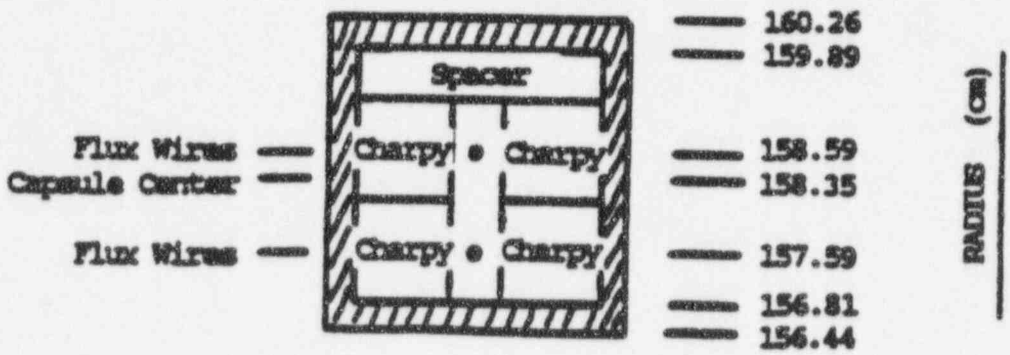
Capsule V	End of Cycle 1	10/74
Capsule T	End of Cycle 3	03/77
Capsule R	End of Cycle 5	03/79
Capsule S	End of Cycle 16	10/90

The type and location of the neutron sensors included in the materials surveillance program are described in some detail in Reference 2; and, are illustrated schematically in Figure 2.2-1 of this report.

Relative to Figure 2.2-1, copper, nickel, and cobalt-aluminum monitors, in wire form, were placed in holes drilled in spacers at several axial levels within each capsule. The cadmium-shielded uranium and neptunium fission monitors were accomodated within a dosimeter block located near the center of the capsule. In addition to these high purity sensors, iron dosimeters were also obtained by removing samples from several charpy test specimens from various locations within the capsule. Specific information pertinent to the individual sensor sets included in Capsules V, T, R and S is provided in the appendices to this report.

FIGURE 2.2-1

NEUTRON SENSOR LOCATIONS WITHIN INTERNAL SURVEILLANCE CAPSULES



SECTION 3.0

NEUTRON TRANSPORT AND DOSIMETRY EVALUATION METHODOLOGIES

As noted in Section 1.0 of this report, the best estimate exposure of the reactor pressure vessel was developed using a combination of absolute plant specific neutron transport calculations and plant specific measurements from the reactor cavity and internal surveillance capsules. In this section, the neutron transport and dosimetry evaluation methodologies are discussed in some detail; and the approach used to combine the calculations and measurements to produce the best estimate vessel exposure is presented.

3.1 - Neutron Transport Analysis Methods

Fast neutron exposure calculations for the reactor and cavity geometry were carried out using both forward and adjoint discrete ordinates transport techniques. A single forward calculation provided the relative energy distribution of neutrons for use as input to neutron dosimetry evaluations as well as for use in relating measurement results to the actual exposure at key locations in the pressure vessel wall. A series of adjoint calculations, on the other hand, established the means to compute absolute exposure rate values using fuel cycle specific core power distributions; thus, providing a direct comparison with all dosimetry results obtained over the operating history of the reactor.

In combination, the absolute cycle specific data from the adjoint evaluations together with relative neutron energy spectra distributions from the forward calculation provided the means to:

1 - Evaluate neutron dosimetry from reactor cavity and surveillance capsule locations.

2 - Enable a direct comparison of analytical prediction with measurement.

3 - Determine plant specific bias factors to be used in the evaluation of the best estimate exposure of the reactor pressure vessel.

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

3.1.1 - Reference Forward Calculation

A plan view of the reactor geometry at the core midplane elevation is shown in Figure 3.1-1. 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 the primary biological shield, the model also included explicit representations of the surveillance capsules, the pressure vessel cladding, and the mirror insulation located external to the vessel.

The model depicted in Figure 3.1-1 was developed using nominal design dimensions for all components. Specified tolerances in the design dimensions are reflected in the overall uncertainty assessments associated with projected neutron exposures. This modeling approach is consistent with the guidelines of DG-1025.

A description of a single surveillance capsule attached to the thermal shield is shown in Figure 3.1-2. From a neutronic standpoint, the inclusion of the surveillance capsules and associated support structures in the analytical model is significant. Since the presence of the capsules and structure has a marked impact on the magnitude of the neutron flux as well as on the relative neutron energy spectra at dosimetry locations within the capsules, a meaningful comparison of measurement and calculation can be made only if these perturbation effects are properly accounted for in the analysis.

In contrast to the relatively massive stainless steel and carbon steel structures associated with the internal surveillance capsules, the small aluminum capsules used in the reactor cavity measurement program were designed to minimize perturbations in the neutron flux and, thus, to provide free field data at the measurement locations. Therefore, explicit modeling of these small capsules in the forward transport model was not required.

The forward transport calculation for the reactor model depicted in Figures 3.1-1 and 3.1-2 was carried out in r,θ geometry using the DORT two-dimensional discrete ordinates code[6] and the BUGLE-93 cross-section library[7]. The BUGLE-93 library is a 47 neutron group, ENDFB-VI based, data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P_3 expansion of the scattering cross-sections and the angular discretization was modeled with an S_8 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 1518 MWt.

The spatial core power distribution utilized in the reference forward calculation was derived from statistical studies of long-term operation of Westinghouse 2-loop plants. Inherent in the development of this reference core power distribution was 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.

Due to the use of this bounding spatial power distribution, the results from the reference forward calculation establish conservative exposure projections for reactors of this design operating at the stretch rating of 1518 MWt. Since it is unlikely that actual reactor operation would result in the implementation of a power distribution at the nominal +2σ level for a large number of fuel cycles and, further, because of the widespread implementation of low leakage fuel management strategies, the fuel cycle specific calculations for this reactor result in exposure rates well below these conservative predictions. This difference between the conservative forward calculation and the fuel cycle specific computations is illustrated by a comparison of the analytical results given in Section 4.0 of this report.

3.1.2 - Cycle Specific Adjoint Calculations

All adjoint analyses were also carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation from the BUGLE-93 library. Adjoint source locations were chosen at each of the azimuthal locations containing cavity dosimetry as well as at several key azimuths on the pressure vessel inner radius. In addition, adjoint calculations were carried out for sources positioned at the geometric center of capsules located at 13, 23, and 33 degrees relative to the core cardinal axes. Again, these calculations were run in r,θ geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case, $\phi(E > 1.0\ MeV)$.

The importance functions generated from these individual adjoint analyses provided 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 fuel cycles to date; and, established the means to perform similar predictions and dosimetry evaluations for all subsequent fuel cycles.

Having the importance functions and appropriate core source distributions, the response of interest can be calculated as:

$$\phi(R_0,\theta_0) = \int_r \int_\theta \int_E I(r,\theta,E)\ S(r,\theta,E)\ r\ dr\ d\theta\ dE$$

where: $\phi(R_0,\theta_0)$ = Neutron flux (E > 1.0 MeV) at radius R_0 and azimuthal angle θ_0.

$I(r,\theta,E)$ = 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,θ and energy E.

It is important to note that the cycle specific neutron source distributions, $S(r,\theta,E)$, utilized with the adjoint importance functions, $I(r,\theta,E)$, permited the use not only of fuel cycle specific spatial variations of fission rates within the reactor core; but, also allowed for the inclusion of the effects of the differing neutron yield per fission and the variation in fission spectrum introduced by the build-in of plutonium isotopes as the burnup of individual fuel assemblies increased.

Although the adjoint importance functions used in these analyses were based on a response function defined by the threshold neutron flux (E > 1.0 MeV), prior calculations[8] have shown that, while the implementation of low leakage loading patterns significantly impact the magnitude and the spatial distribution of the neutron field, changes in the relative neutron energy spectrum are of second order. Thus, for a given location the exposure parameter ratios such as [dpa/sec]/[ϕ(E > 1.0 MeV)] are insensitive to changing core source distributions. In the application of these adjoint importance functions to the current evaluations, therefore, calculation of the iron displacement rates (dpa/sec) and the neutron flux (E > 0.1 MeV) were computed on a cycle specific basis by using the appropriate [dpa/sec]/[ϕ(E > 1.0 MeV)] and [ϕ(E > 0.1 MeV)]/[ϕ(E > 1.0 MeV)] ratios from the reference forward analysis in conjunction with the cycle specific ϕ(E > 1.0 MeV) solutions from the individual adjoint evaluations.

In particular, after defining the following exposure rate ratios,

$$R_1 = \frac{[dpa/sec]}{\phi(E > 1.0\ MeV)}$$

$$R_2 = \frac{\phi(E > 0.1\ MeV)}{\phi(E > 1.0\ MeV)}$$

the corresponding fuel cycle specific exposure rates at the adjoint source locations were computed from the following relations:

$$dpa/sec = [\phi(E > 1.0\ MeV)]\ R_1$$

$$\phi(E > 0.1\ MeV) = [\phi(E > 1.0\ MeV)]\ R_2$$

All fuel cycle specific absolute calulations were also normalized to the current rated power level for Point Beach Unit 2, 1518 MWt.

FIGURE 3.1-1

REACTOR GEOMETRY SHOWING A 45° r,θ SECTOR

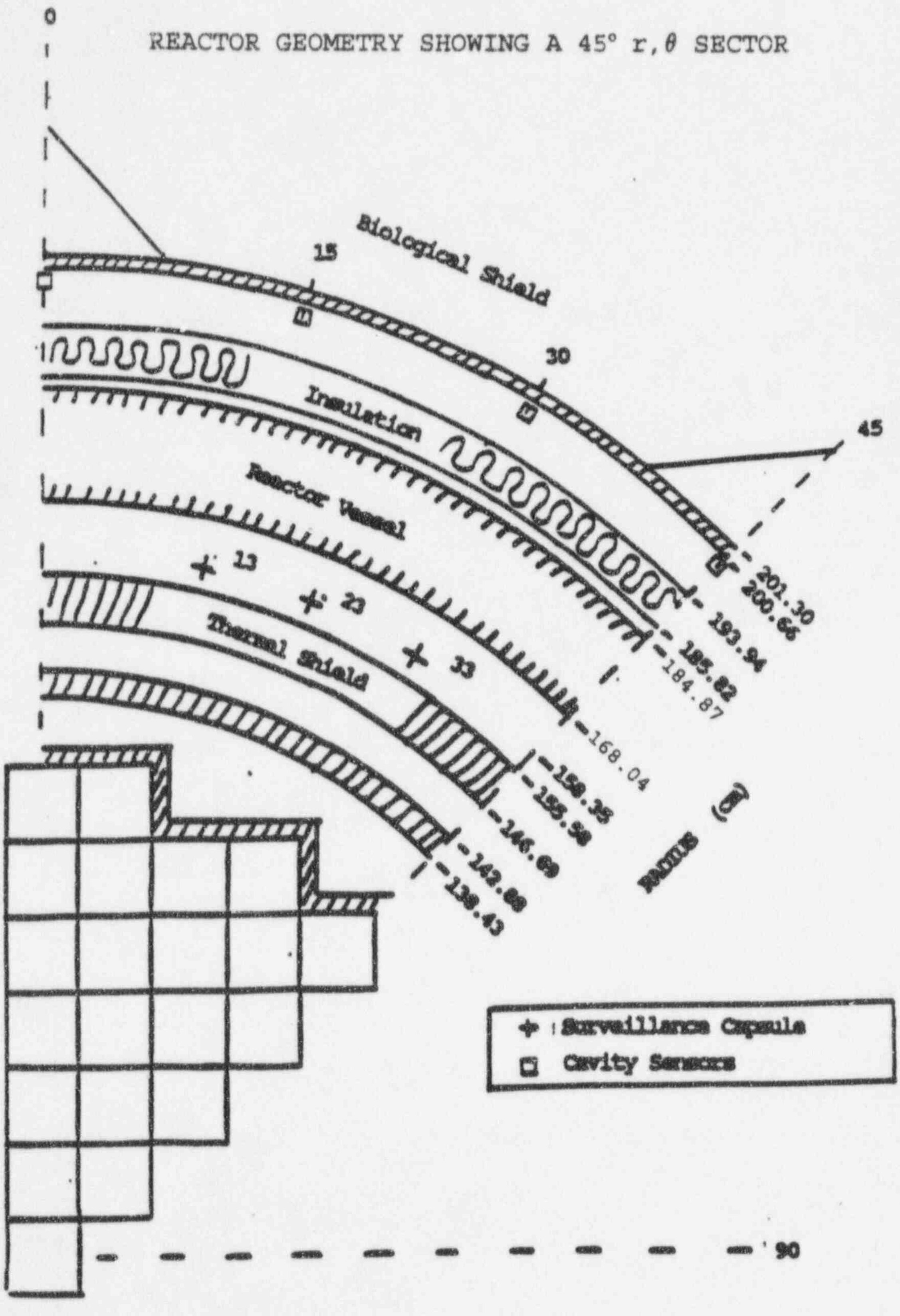
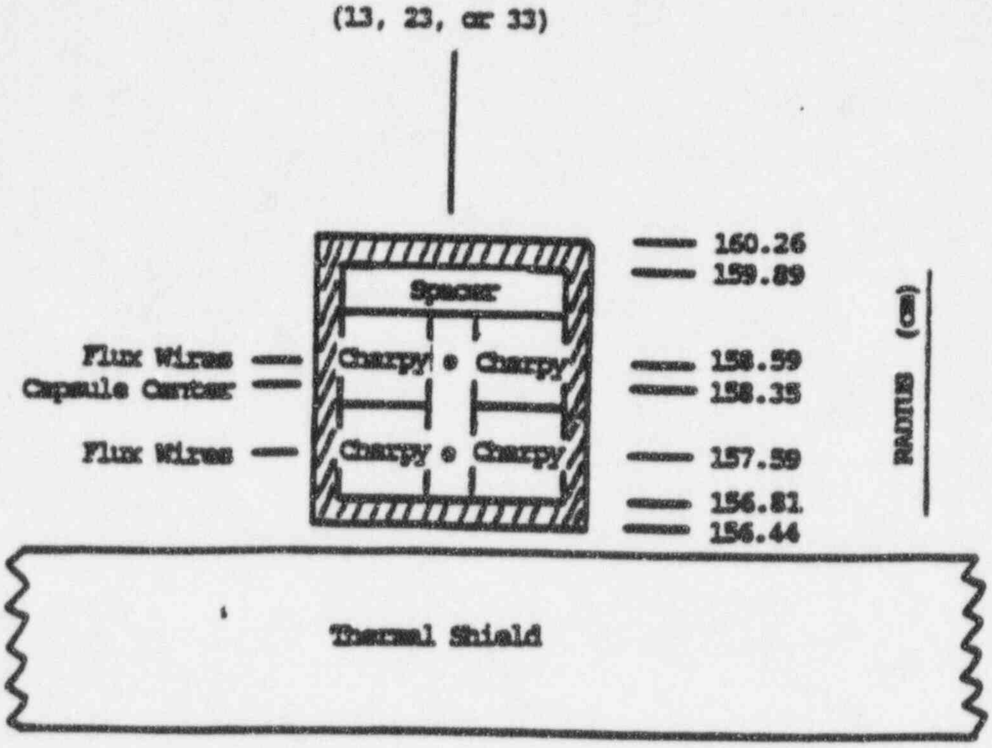


FIGURE 3.1-2

INTERNAL SURVEILLANCE CAPSULE GEOMETRY



3.2 - Neutron Dosimetry Evaluation Methodology

The use of passive neutron sensors such as those included in the internal surveillance capsule and reactor cavity dosimetry sets does not yield a direct measure of the energy dependent neutron flux level at the measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average flux level and, hence, time integrated exposure (fluence) experienced by the sensors may be developed from the measurements only if the sensor characteristics and the parameters of the irradiation are well known. In particular, the following variables are of interest:

1 - The measured specific activity of each sensor
2 - The physical characteristics of each sensor
3 - The operating history of the reactor
4 - The energy response of each sensor
5 - The neutron energy spectrum at the sensor location

In this section the procedures used by Westinghouse to determine sensor specific activities, to develop reaction rates for individual sensors from the measured specific activities and the operating history of the reactor, and to derive key fast neutron exposure parameters from the measured reaction rates are described.

For the most part, these procedures apply to all of the evaluations provided in this report. However, in the case of internal surveillance capsule V, the specific activities of the multiple foil sensor set were determined from prior analysis performed by Battelle Columbus Laboratory. In this case, the source of the measured specific activity data was referenced and the remainder of the data evaluation proceeded using the methodology described in this section.

3.2.1 - Determination of Sensor Reaction Rates

Following irradiation, the multiple foil sensor sets from surveillance capsule and cavity irradiations along with reactor cavity gradient chains were recovered and transported to Pittsburgh for evaluation. Analysis of all radiometric foils and gradient chains was performed at the Westinghouse Analytical Services Laboratory; while the evaluation of the SSTR sensors from the cavity irradiations was carried out at the Westinghouse Science and Technology Center Track Recorder Laboratory.

3.2.1.1 - Radiometric Sensors

The specific activity of each of the radiometric sensors and gradient chain segments was determined using established ASTM procedures[9 through 19]. Following sample preparation and weighing, the specific activity of each sensor was determined by means of a lithium drifted germanium, Ge(Li), gamma spectrometer. In the case of the surveillance capsule and cavity multiple foil sensor sets, these analyses were performed by direct counting of each of the individual foils or wires; or, as in the case of U-238 and Np-237 fission monitors from internal surveillance capsules, by direct counting preceded by dissolution and chemical separation of cesium from the sensor. For the stainless steel gradient chains used in the cavity irradiations, individual sensors were obtained by cutting the chains or wires into a series of segments to provide data points at one foot intervals over an axial span encompassing $\pm$ 7 feet relative to the reactor core midplane.

The irradiation history of the reactor over its operating lifetime was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report". In particular, operating data were extracted from that report on a monthly basis from reactor startup to the end of the current evaluation period. For the sensor sets utilized in surveillance capsule and reactor cavity irradiations, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation at 1518 MWt were determined from the following equation:

$$R = \frac{A}{N_0 \, F \, Y \sum_j \frac{P_j}{P_{ref}} C_j \, [1 - e^{-\lambda t_j}] \, e^{-\lambda t_d}}$$

where:

A	=	measured specific activity (dps/gm)
R	=	reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus).
N_0	=	number of target element atoms per gram of sensor.
F	=	weight fraction of the target isotope in the sensor material.
Y	=	number of product atoms produced per reaction.
P_j	=	average core power level during irradiation period j (MW).
P_{ref}	=	maximum or reference core power level of the reactor (MW).
C_j	=	calculated ratio of $\phi(E > 1.0 \text{ MeV})$ during irradiation period j to the time weighted average $\phi(E > 1.0 \text{ MeV})$ over the entire irradiation period.
λ	=	decay constant of the product isotope (sec^{-1}).
t_j	=	length of irradiation period j (sec).
t_d	=	decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the total irradiation period.

In the above equation, the ratio P_j/P_{ref} accounts for month by month variation of power level within a given fuel cycle. The ratio C_j is calculated for each fuel cycle using the adjoint transport methodology and accounts for the change in sensor reaction rates caused by variations in flux level due to changes in core power spatial distributions from fuel cycle to fuel cycle. For a single cycle irradiation $C_j = 1.0$. However, for multiple cycle irradiations, particularly those employing low leakage fuel management the additional C_j correction must be utilized.

3.2.1.2 - Solid State Track Recorders

Following preparation of the mica discs, all of the solid state track recorders were scanned either manually or with the Westinghouse STC Automated Track Scanner to determine the number of fissions that occured during the course of the irradiation of the sensor sets. Since the SSTR sensors are integrating devices not susceptible to radioactive decay of a product isotope, the measurements of total fissions per atom, A, were converted directly to reaction rates using the following relationship:

$$R = \frac{A}{\sum_j \frac{P_j}{P_{ref}} t_j}$$

where the denominator in the above equation represents the total effective full power seconds of reactor operation during the irradiation of the solid state track recorders.

The SSTR fissionable deposits were designed for reuse in the long term monitoring program. Therefore, following processing each sensor was carefully examined to assure that the deposits were neither damaged nor contaminated during irradiation, handling, and post-irradiation processing.

In particular, these examinations were designed to assure that, in all cases, the fission tracks were confined to an area corresponding to the active portion of the fissionable deposit and that the edges of the active area were sharply defined with a sufficient drop-off in track density to indicate acceptable signal to background ratios for the measurements. Each mica SSTR and fissionable deposit was also closely inspected under a microscope to verify that no physical damage had occured during exposure or shipment. Selected deposits were also subjected to mass recalibration to verify that no deposit mass had been lost during shipping or exposure.

3.2.1.3 - Corrections to Reaction Rate Data

Prior to using the measured reaction rates in the least squares adjustment procedure discussed in Section 3.2.2 of this report, additional corrections were made to the U-238 foil and SSTR measurements to account for the presence of U-235 impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. These corrections were location and fluence dependent and were derived from a combination of data from the reference forward transport calculation and the cycle specific adjoint analyses as well as from measurements made with the U-235 solid state track recorders.

In addition to the corrections made for the presence of U-235 in the U-238 fission sensors, corrections were also made to both the U-238 and Np-237 sensor reaction rates to account for gamma ray induced fission reactions occuring over the course of the irradiation. These photo-fission corrections were, likewise, location dependent and were based on the reference transport calculations described in Section 3.1.1.

In performing the dosimetry evaluations for the internal surveillance capsules, the sensor reaction rates measured at the locations shown in Figure 2.2-1 were indexed to the geometric center of the capsules prior to use in the spectrum adjustment procedure. This indexing procedure required correcting the measured reaction rates by the application of analytically determined spatial gradients. For the Point Beach Unit 2 surveillance capsules, the gradient correction factors for each sensor reaction were obtained from the reference forward transport calculation and were used in a multiplicative fashion to relate individual measured reaction rates to the corresponding value at the geometric center of the surveillance capsule. In the case of the reactor cavity sensors, all of the monitors were located at the same radial and azimuthal location. Thus, gradient corrections were not required in the evaluation of these dosimetry sets.

3.2.2 - Least Squares Adjustment Procedure

Values of key fast neutron exposure parameters were derived from the measured reaction rates using the FERRET least squares adjustment code[20]. The FERRET approach used the measured reaction rate data, sensor reaction cross-sections, and a calculated trial spectrum as input and proceeded to adjust the group fluxes from the trial spectrum to produce a best fit (in a least squares sense) to the measured reaction rate data. The "measured" exposure parameters along with the associated uncertainties were then obtained from the adjusted spectrum.

In the FERRET evaluations, a log-normal least squares algorithm weights both the trial values and the measured data in accordance with the assigned uncertainties and correlations. In general, the measured values f are linearly related to the flux ϕ by some response matrix A:

$$f_i^{(s,\alpha)} = \sum_g A_{ig}^{(s)} \phi_g^{(\alpha)}$$

where i indexes the measured values belonging to a single data set s, g designates the energy group, and α delineates spectra that may be simultaneously adjusted. For example,

$$R_i = \sum_g \sigma_{ig} \phi_g$$

relates a set of measured reaction rates R_i to a single spectrum ϕ_g by the multigroup reaction cross-section σ_{ig}. The log-normal approach automatically accounts for the physical constraint of positive fluxes, even with large assigned uncertainties.

In the least squares adjustment, the continuous quantities (i.e., neutron spectra and cross-sections) were approximated in a multi-group format consisting of 53 energy groups. The trial input spectrum was converted to the FERRET 53 group structure using the SAND-II code[21]. This procedure was carried out by first expanding the 47 group calculated spectrum into the SAND-II 620 group structure using a SPLINE interpolation procedure in regions where group boundaries do not coincide. The 620 point spectrum was then re-collapsed into the group structure used in FERRET.

The sensor set reaction cross-sections, obtained from the ENDF/B-VI dosimetry file[22], were also collapsed into the 53 energy group structure using the SAND-II code. In this instance, the trial spectrum, as expanded to 620 groups, was employed as a weighting function in the cross-section collapsing procedure. Reaction cross-section uncertainties in the form of a 53 x 53 covariance matrix for each sensor reaction were also constructed from the information contained on the ENDF/B-VI data files. These matrices included energy group to energy group uncertainty correlations for each of the individual reactions. However, correlations between cross-sections for different sensor reactions were not included. The omission of this additional uncertainty information does not significantly impact the results of the adjustment.

Due to the importance of providing a trial spectrum that exhibits a relative energy distribution close to the actual spectrum at the sensor set locations, the neutron spectrum input to the FERRET evaluation was obtained from the plant specific calculation for each dosimetry location. While the 53 x 53 group covariance matrices applicable to the sensor reaction cross-sections were developed from the cross-section data files, the covariance matrix for the input trial spectrum was constructed from the following relation:

$$M_{gg'} = R_n^2 + R_g R_{g'} P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty (i.e., complete correlation) for the set of values.

The fractional uncertainties R_g specify additional random uncertainties for group g that are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\ \delta_{gg'} + \theta\ e^{-H}$$

where:

$$H = \frac{(g-g')^2}{2\ \gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes short range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1 when g = g' and 0 otherwise. For the trial spectrum used in the current evaluations, a short range correlation of γ = 6 groups was used. This choice implies that neighboring groups are strongly correlated when θ is close to 1. Strong long range correlations (or anti-correlations) were justified based on information presented by R. E. Maerker[23]. Maerker's results are closely duplicated when γ = 6. For the integral reaction rate covariances, simple normalization and random uncertainties were combined as deduced from experimental uncertainties.

In performing the least squares adjustment with the FERRET code, the input spectra from the reference forward transport calculations were normalized to the absolute calculations from the cycle specific adjoint analyses. The specific normalization factors for individual evaluations depended on the location of the sensor set as well as on the neutron flux level at that location.

The specific assignment of uncertainties in the measured reaction rates and the input (trial) spectra used in the FERRET evaluations was as follows:

REACTION RATE UNCERTAINTY	5%
FLUX NORMALIZATION UNCERTAINTY	30%
FLUX GROUP UNCERTAINTIES	
(E > 0.0055 MeV)	30%
(0.68 ev < E < 0.0055 MeV)	58%
(E < 0.68 ev)	104%
SHORT RANGE CORRELATION	
(E > 0.0055 MeV)	0.9
(0.68 ev < E < 0.0055 MeV)	0.5
(E < 0.68 ev)	0.5
FLUX GROUP CORRELATION RANGE	
(E > 0.0055 MeV)	6
(0.68 ev < E < 0.0055 MeV)	3
(E < 0.68 ev)	2

It should be noted that the uncertainties listed for the upper energy ranges extend down to the lower range. Thus, the 58% group uncertainty in the second range is made up of a 30% uncertainty with a 0.9 short range correlation and a range of 6, and a second part of magnitude 50% with a 0.5 correlation and a range of 3.

These input uncertainty assignments were based on prior experience in using the FERRET least squares adjustment approach in the analysis of neutron dosimetry from surveillance capsule, reactor cavity, and benchmark irradiations. The values are liberal enough to permit adjustment of the input spectrum to fit the measured data for all practical applications.

3.3 - Determination of Best Estimate Pressure Vessel Exposure

As noted earlier in this report, the best estimate exposure of the reactor pressure vessel was developed using a combination of absolute plant specific transport calculations based on the methodology discussed in Section 3.1 and plant specific measurement data determined using the measurement evaluation techniques described in Section 3.2. In particular, the best estimate vessel exposure is obtained from the following relationship:

$$\Phi_{Best\ Est.} = K\ \Phi_{Calc.}$$

where: $\Phi_{Best.\ Est.}$ = The best estimate fast neutron exposure at the location of interest.

K = The plant specific measurement/calculation (M/C) bias factor derived from all available surveillance capsule and reactor cavity dosimetry data.

$\Phi_{Calc.}$ = The absolute calculated fast neutron exposure at the location of interest.

The approach defined in the above equation is based on the premise that the measurement data represent the most accurate plant specific information available at the locations of the dosimetry; and, further that the use of the measurement data on a plant specific basis essentially removes biases present in the analytical approach and mitigates the uncertainties that would result from the use of analysis alone. That is, at the measurement points the uncertainty in the best estimate exposure is dominated by the uncertainties in the measurement process. At locations within the pressure vessel wall, additional uncertainty is incurred due to the analytically determined relative ratios among the various measurement points and locations within the pressure vessel wall.

The implementation of this approach acts to remove plant specific biases associated with the definition of the core source, actual vs. assumed reactor dimensions, and operational variations in water density within the reactor. As a result, the overall uncertainty in the best estimate exposure projections within the vessel wall depend on the individual uncertainties in the measurement process, the uncertainty in the dosimetry location, and in the uncertainty in the calculated ratio of the neutron exposure at the point of interest to that at the measurement location.

The uncertainties in the measured flux were derived directly from the results of the least squares evaluation of dosimetry data. The positioning uncertainties were taken from parametric studies of sensor position performed as part of an analytical sensitivity evaluation of the Point Beach Unit 2 reactor. The uncertainties in the exposure ratios relating dosimetry results to positions within the vessel wall were based on analytical sensitivity studies of the vessel thickness tolerance for the cavity data and on downcomer water density variations and vessel inner radius tolerance for the surveillance capsule measurements.

SECTION 4.0

RESULTS OF NEUTRON TRANSPORT CALCULATIONS

4.1 Reference Forward Calculation

As noted in Section 3.0 of this report, data from the reference forward transport calculation were used in evaluating dosimetry from both reactor cavity and surveillance capsule irradiations as well as in relating the results of these evaluations to the neutron exposure of the pressure vessel wall. In this section, the key data extracted from the reference forward calculation is presented and its relevance to the dosimetry evaluations and vessel exposure projections is discussed. The reader should recall that the results of the reference forward transport calculation were intended for use on a relative basis and, therefore, should not be used for absolute comparison with measurement. All absolute comparisons were based on the results of the fuel cycle specific adjoint calculations discussed in Section 4.2.

4.1.1 - Cavity Sensor Set Locations

Data from the reference forward calculation pertinent to cavity sensor evaluations are provided in Tables 4.1-1 and 4.1-2.

In Table 4.1-1, the calculated neutron energy spectra applicable to the sensor locations at 0.0, 15.0, 30.0 and 45.0 degrees relative to the core cardinal axes are listed. These data represent the trial spectra used as the starting guess in the FERRET least squares adjustment evaluations of the cavity sensor sets. On a relative basis these calculated energy distributions establish a baseline against which adjusted spectra may be compared; and, when coupled with the adjoint results of Section 4.2, provide an analytical prediction of absolute neutron spectra at the sensor set locations for each irradiation period.

In Table 4.1-2, the calculated neutron sensor reaction rates associated with the spectra from Table 4.1-1 are provided along with the reference exposure rates in terms of $\phi(E > 1.0$ MeV), $\phi(E > 0.1$ MeV) and dpa/sec. Also listed are the associated exposure rate ratios calculated for each of the cavity sensor set locations.

The reference reaction rates, exposure rates, and exposure rate ratios were used in conjunction with fuel cycle specific adjoint transport calculations from Section 4.2 to provide calculated sensor set reaction rates and to project sensor set exposures in terms of $\phi(E > 0.1$ MeV) and dpa/sec for each irradiation period.

4.1.2 - Surveillance Capsule Locations

Data from the reference forward calculation pertinent to surveillance capsule evaluations are provided in Tables 4.1-3 through 4.1-5.

In Table 4.1-3, the calculated neutron energy spectra at the geometric center of surveillance capsules located at 13, 23, and 33 degrees relative to the core cardinal axes are listed. In Table 4.1-4, the calculated neutron sensor reaction rates and exposure rate ratios associated with the spectra from Table 4.1-3 are provided along with the calculated exposure rates in terms of $\phi(E > 1.0$ MeV), $\phi(E > 0.1$ MeV) and dpa/sec. Again, these data are applicable to the geometric center of each surveillance capsule. These tabulated data were used in the surveillance capsule dosimetry evaluations and exposure calculations in the same fashion as was the case for the cavity sensor sets.

As noted earlier in this report, surveillance capsule dosimetry evaluations also require spatial gradient corrections to be applied to measured reaction rates in sensors dispersed throughout the capsule. In the case of the Point Beach Unit 2 surveillance capsules, neutron sensors were positioned within the specimen array as shown in Figure 2.2-1. In Table 4.1-5, gradient correction factors applicable to the various dosimetry locations are provided for each sensor reaction.

4.1.3 - Pressure Vessel Wall

Data from the reference forward calculation pertinent to the pressure vessel wall are provided in Tables 4.1-6 through 4.1-9.

In Table 4.1-6, the calculated azimuthal distribution of exposure rates in terms of $\phi(E > 1.0 \text{ MeV})$, $\phi(E > 0.1 \text{ MeV})$, and dpa/sec are listed at approximately 5 degree intervals over the reactor geometry. These data are applicable to the clad/base metal interface. Also given in Table 4.1-6 are the exposure rate ratios $[\phi(E > 0.1 \text{ MeV})]/[\phi(E > 1.0 \text{ MeV})]$ and $[\text{dpa/sec}]/[\phi(E > 1.0 \text{ MeV})]$ that provide an indication of the variation in neutron spectrum as a function of azimuthal angle at the pressure vessel inner radius.

Radial gradient information for $\phi(E > 1.0 \text{ MeV})$, $\phi(E > 0.1 \text{ Mev})$, and dpa/sec is given in Tables 4.1-7, 4.1-8, and 4.1-9, respectively. These data are presented on a relative basis for each exposure parameter at the 0, 15, 30, and 45 degree azimuthal locations. Exposure rate distributions within the vessel wall are obtained by normalizing the calculated or best estimate exposure at the vessel inner radius to the gradient data given in Tables 4.1-7 through 4.1-9.

TABLE 4.1-1

CALCULATED REFERENCE NEUTRON ENERGY SPECTRA
AT CAVITY SENSOR SET LOCATIONS [n/cm^2-sec]
1518 MWt; F_a = 1.2

LOWER ENERGY (MeV)	AZIMUTHAL ANGLE 0.0°	15.0°	30.0°	45.0°
1.42E+01	6.55E+05	5.19E+05	3.99E+05	3.83E+05
1.22E+01	1.94E+06	1.52E+06	1.15E+06	1.10E+06
1.00E+01	8.02E+06	6.14E+06	4.59E+06	4.31E+06
8.61E+00	1.50E+07	1.14E+07	8.44E+06	7.86E+06
7.41E+00	2.39E+07	1.79E+07	1.31E+07	1.21E+07
6.07E+00	5.15E+07	3.83E+07	2.78E+07	2.54E+07
4.97E+00	7.54E+07	5.62E+07	4.03E+07	3.61E+07
3.68E+00	1.52E+08	1.14E+08	8.00E+07	6.98E+07
3.01E+00	1.28E+08	9.65E+07	6.70E+07	5.76E+07
2.73E+00	1.06E+08	7.96E+07	5.47E+07	4.69E+07
2.47E+00	1.34E+08	1.02E+08	7.02E+07	5.95E+07
2.37E+00	8.19E+07	6.26E+07	4.30E+07	3.63E+07
2.35E+00	2.52E+07	1.89E+07	1.28E+07	1.09E+07
2.23E+00	1.09E+08	8.25E+07	5.61E+07	4.77E+07
1.92E+00	3.06E+08	2.32E+08	1.58E+08	1.34E+08
1.65E+00	4.28E+08	3.26E+08	2.22E+08	1.86E+08
1.35E+00	7.32E+08	5.69E+08	3.86E+08	3.20E+08
1.00E+00	1.69E+09	1.34E+09	9.09E+08	7.44E+08
8.21E-01	1.58E+09	1.27E+09	8.60E+08	6.98E+08
7.43E-01	7.57E+08	6.42E+08	4.40E+08	3.44E+08
6.08E-01	3.27E+09	2.68E+09	1.82E+09	1.47E+09
4.98E-01	3.08E+09	2.60E+09	1.78E+09	1.40E+09
3.69E-01	3.20E+09	2.73E+09	1.89E+09	1.48E+09

NOTE: The upper energy of group 1 is 17.33 Mev.

TABLE 4.1-1 (continued)

CALCULATED REFERENCE NEUTRON ENERGY SPECTRA
AT CAVITY SENSOR SET LOCATIONS [n/cm^2-sec]
1518 MWt; F_a = 1.2

LOWER ENERGY (MeV)	AZIMUTHAL ANGLE 0.0°	15.0°	30.0°	45.0°
2.97E-01	4.34E+09	3.67E+09	2.53E+09	2.00E+09
1.83E-01	5.44E+09	4.84E+09	3.37E+09	2.58E+09
1.11E-01	5.93E+09	5.26E+09	3.68E+09	2.84E+09
6.74E-02	3.96E+09	3.57E+09	2.51E+09	1.93E+09
4.09E-02	3.02E+09	2.74E+09	1.94E+09	1.49E+09
3.18E-02	9.38E+08	8.69E+08	6.21E+08	4.72E+08
2.61E-02	4.40E+08	4.16E+08	2.96E+08	2.20E+08
2.42E-02	1.61E+09	1.37E+09	9.63E+08	7.79E+08
2.19E-02	9.99E+08	8.88E+08	6.23E+08	4.84E+08
1.50E-02	1.89E+09	1.77E+09	1.25E+09	9.29E+08
7.10E-03	2.52E+09	2.36E+09	1.69E+09	1.28E+09
3.36E-03	2.87E+09	2.65E+09	1.90E+09	1.46E+09
1.59E-03	2.42E+09	2.24E+09	1.62E+09	1.25E+09
4.54E-04	3.73E+09	3.45E+09	2.50E+09	1.95E+09
2.14E-04	1.87E+09	1.74E+09	1.27E+09	9.87E+08
1.01E-04	2.00E+09	1.85E+09	1.35E+09	1.06E+09
3.73E-05	2.57E+09	2.37E+09	1.73E+09	1.37E+09
1.07E-05	3.00E+09	2.76E+09	2.02E+09	1.61E+09
5.04E-06	1.63E+09	1.50E+09	1.10E+09	8.76E+08
1.86E-06	2.02E+09	1.86E+09	1.36E+09	1.09E+09
8.76E-07	1.42E+09	1.30E+09	9.59E+08	7.71E+08
4.14E-07	1.07E+09	9.87E+08	7.26E+08	5.86E+08
1.00E-07	2.52E+09	2.26E+09	1.67E+09	1.40E+09
0.00	6.55E+09	5.21E+09	3.79E+09	3.64E+09

NOTE: The upper energy of group 1 is 17.33 Mev.

TABLE 4.1-2

REFERENCE NEUTRON SENSOR REACTION RATES AND EXPOSURE PARAMETERS AT THE CAVITY SENSOR SET LOCATIONS

1518 MWt; F_{z} = 1.20

	AZIMUTHAL ANGLE			
	0.0°	15.0°	30.0°	45.0°
	Reaction Rate (rps/nucleus)			
Cu-63(n,α) (Cd)	2.05E-18	1.54E-18	1.13E-18	1.04E-18
Ti-46(n,p) (Cd)	3.06E-17	2.29E-17	1.66E-17	1.51E-17
Fe-54(n,p) (Cd)	1.99E-16	1.50E-16	1.06E-16	9.31E-17
Ni-58(n,p) (Cd)	2.87E-16	2.16E-16	1.52E-16	1.33E-16
U-238(n,f) (Cd)	1.18E-15	9.03E-16	6.22E-16	5.29E-16
Np-237(n,f) (Cd)	1.61E-14	1.30E-14	8.94E-15	7.24E-15
Co-59(n,γ)	3.88E-13	3.34E-13	2.44E-13	2.10E-13
Co-59(n,γ) (Cd)	1.82E-13	1.68E-13	1.23E-13	9.68E-14
U-235(n,f)	3.64E-12	3.01E-12	2.18E-12	2.00E-12
U-235(n,f) (Cd)	6.47E-13	5.92E-13	4.31E-13	3.42E-13
U-238(γ,f)	4.25E-17	3.43E-17	2.49E-17	2.27E-17
Np-237(γ,f)	1.29E-16	1.04E-16	7.15E-17	6.52E-17
	Neutron Flux (n/cm²-sec)			
φ(E > 1.0 MeV)	4.14E+09	3.20E+09	2.19E+09	1.83E+09
φ(E > 0.1 MeV)	3.25E+10	2.76E+10	1.90E+10	1.50E+10
	dpa/sec			
Displacement Rate	1.17E-11	9.70E-12	6.70E-12	5.36E-12
φ(E > 0.1)/φ(E > 1.0)	7.76	8.63	8.72	8.20
[dpa/sec]/φ(E > 1.0)	2.83E-21	3.03E-21	3.06E-21	2.93E-21
U238(γ,f)/U238(n,f)	0.036	0.038	0.040	0.043
Np237(γ,f)/Np237(n,f)	0.008	0.008	0.008	0.009

TABLE 4.1-3

CALCULATED REFERENCE NEUTRON ENERGY SPECTRA
SURVEILLANCE CAPSULE CENTER [n/cm^2-sec]
1518 MWt; F_a = 1.2

LOWER ENERGY (MeV)	AZIMUTHAL ANGLE 13.0°	23.0°	33.0°
1.42E+01	1.63E+07	1.34E+07	1.16E+07
1.22E+01	5.40E+07	4.34E+07	3.75E+07
1.00E+01	2.47E+08	1.94E+08	1.67E+08
8.61E+00	4.97E+08	3.83E+08	3.32E+08
7.41E+00	9.11E+08	6.82E+08	5.94E+08
6.07E+00	2.28E+09	1.69E+09	1.47E+09
4.97E+00	3.71E+09	2.64E+09	2.33E+09
3.68E+00	8.25E+09	5.50E+09	4.98E+09
3.01E+00	7.23E+09	4.53E+09	4.19E+09
2.73E+00	5.72E+09	3.55E+09	3.29E+09
2.47E+00	6.89E+09	4.22E+09	3.94E+09
2.37E+00	4.02E+09	2.47E+09	2.30E+09
2.35E+00	1.12E+09	6.91E+08	6.44E+08
2.23E+00	4.91E+09	2.99E+09	2.80E+09
1.92E+00	1.39E+10	8.32E+09	7.84E+09
1.65E+00	1.72E+10	1.00E+10	9.54E+09
1.35E+00	2.71E+10	1.55E+10	1.48E+10
1.00E+00	5.51E+10	3.04E+10	2.93E+10
8.21E-01	3.95E+10	2.13E+10	2.06E+10
7.43E-01	2.01E+10	1.08E+10	1.05E+10
6.08E-01	6.22E+10	3.24E+10	3.17E+10
4.98E-01	5.22E+10	2.69E+10	2.63E+10
3.69E-01	5.72E+10	2.95E+10	2.88E+10

NOTE: The upper energy of group 1 is 17.33 Mev.

TABLE 4.1-3 (continued)

CALCULATED REFERENCE NEUTRON ENERGY SPECTRA
SURVEILLANCE CAPSULE CENTER [n/cm^2-sec]
1518 MWt; F_a = 1.2

LOWER ENERGY (MeV)	AZIMUTHAL ANGLE 13.0°	23.0°	33.0°
2.97E-01	5.58E+10	2.84E+10	2.78E+10
1.83E-01	7.43E+10	3.79E+10	3.71E+10
1.11E-01	7.06E+10	3.56E+10	3.50E+10
6.74E-02	5.41E+10	2.72E+10	2.67E+10
4.09E-02	4.48E+10	2.25E+10	2.21E+10
3.18E-02	1.64E+10	8.24E+09	8.10E+09
2.61E-02	7.35E+09	3.59E+09	3.51E+09
2.42E-02	1.49E+10	7.43E+09	7.33E+09
2.19E-02	9.31E+09	4.63E+09	4.56E+09
1.50E-02	2.39E+10	1.20E+10	1.18E+10
7.10E-03	4.41E+10	2.22E+10	2.18E+10
3.36E-03	5.02E+10	2.52E+10	2.47E+10
1.59E-03	4.68E+10	2.33E+10	2.30E+10
4.54E-04	7.73E+10	3.85E+10	3.79E+10
2.14E-04	4.26E+10	2.12E+10	2.08E+10
1.01E-04	4.70E+10	2.33E+10	2.30E+10
3.73E-05	6.16E+10	3.04E+10	3.00E+10
1.07E-05	7.50E+10	3.69E+10	3.65E+10
5.04E-06	4.29E+10	2.11E+10	2.08E+10
1.86E-06	5.63E+10	2.78E+10	2.74E+10
8.76E-07	4.16E+10	2.06E+10	2.02E+10
4.14E-07	3.52E+10	1.76E+10	1.72E+10
1.00E-07	8.60E+10	4.39E+10	4.25E+10
0.00	2.45E+11	1.31E+11	1.23E+11

NOTE: The upper energy of group 1 is 17.33 Mev.

TABLE 4.1-4

REFERENCE NEUTRON SENSOR REACTION RATES AND EXPOSURE PARAMETERS AT THE CENTER OF SURVEILLANCE CAPSULES
1518 MWt; F_a = 1.20

	13.0°	23.0°	33.0°
	Reaction Rate (rps/nucleus)		
Cu-63(n,α)	8.11E-17	6.06E-17	5.29E-17
Fe-54(n,p)	9.91E-15	6.64E-15	6.03E-15
Ni-58(n,p)	1.38E-14	9.14E-15	8.29E-15
U-238(n,f) (Cd)	5.14E-14	3.16E-14	2.94E-14
Np-237(n,f) (Cd)	4.43E-13	2.48E-13	2.38E-13
Co-59(n,γ)	1.13E-11	5.87E-12	5.62E-12
Co-59(n,γ) (Cd)	4.40e-12	2.17E-12	2.15E-12
U-238(γ,f)	2.58E-15	1.45E-15	1.40E-15
Np-237(γ,f)	7.21E-15	4.06E-15	3.91E-15
	Neutron Flux (n/cm²-sec)		
φ(E > 1.0 MeV)	1.59E+11	9.35E+10	8.83E+10
φ(E > 0.1 MeV)	6.02E+11	3.22E+11	3.11E+11
	dpa/sec		
Displacement Rate	2.83E-10	1.59E-10	1.52E-10
φ(E > 0.1)/φ(E > 1.0)	3.79	3.44	3.52
[dpa/sec]/φ(E > 1.0)	1.78E-21	1.70E-21	1.72E-21
U238(γ,f)/U238(n,f)	0.050	0.046	0.048
Np237(γ,f)/Np237(n,f)	0.016	0.016	0.016

TABLE 4.1-5

RADIAL GRADIENT CORRECTIONS FOR SENSORS CONTAINED IN INTERNAL SURVEILLANCE CAPSULES

13° CAPSULE

		Radial Location (cm)		
		157.59	158.35	158.59
Cu-63(n,α)		0.88	1.00	1.04
Fe-54(n,p)		0.87	1.00	1.05
Ni-58(n,p)		0.87	1.00	1.05
U-238(n,f)	Cd	0.87	1.00	1.05
Np-237(n,f)	Cd	0.87	1.00	1.05
Co-59(n,γ)		0.95	1.00	0.99
Co-59(n,γ)	Cd	0.83	1.00	1.07

23° CAPSULE

		Radial Location (cm)		
		157.59	158.35	158.59
Cu-63(n,α)		0.87	1.00	1.04
Fe-54(n,p)		0.86	1.00	1.04
Ni-58(n,p)		0.86	1.00	1.04
U-238(n,f)	Cd	0.87	1.00	1.05
Np-237(n,f)	Cd	0.87	1.00	1.05
Co-59(n,γ)		0.95	1.00	0.99
Co-59(n,γ)	Cd	0.84	1.00	1.08

33° CAPSULE

		Radial Location (cm)		
		157.59	158.35	158.59
Cu-63(n,α)		0.87	1.00	1.04
Fe-54(n,p)		0.86	1.00	1.05
Ni-58(n,p)		0.86	1.00	1.05
U-238(n,f)	Cd	0.86	1.00	1.05
Np-237(n,f)	Cd	0.87	1.00	1.05
Co-59(n,γ)		0.95	1.00	0.99
Co-59(n,γ)	Cd	0.83	1.00	1.08

TABLE 4.1-6

SUMMARY OF EXPOSURE RATES AT THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

THETA (deg)	FLUX (n/cm²-sec) (E > 1.0)	(E > 0.1)	dpa/sec	[E > 0.1] / [E > 1.0]	dpa/sec / [E > 1.0]
0.25	5.32E+10	1.46E+11	8.68E-11	2.74	1.63E-21
4.75	5.11E+10	1.40E+11	8.35E-11	2.75	1.64E-21
9.75	4.37E+10	1.23E+11	7.22E-11	2.81	1.65E-21
15.00	3.25E+10	9.45E+10	5.46E-11	2.91	1.68E-21
19.75	2.58E+10	7.13E+10	4.27E-11	2.76	1.65E-21
24.75	2.29E+10	6.36E+10	3.79E-11	2.78	1.66E-21
30.00	2.22E+10	6.05E+10	3.65E-11	2.73	1.64E-21
34.75	2.05E+10	5.72E+10	3.40E-11	2.79	1.66E-21
39.75	1.96E+10	5.17E+10	3.19E-11	2.63	1.62E-21
44.75	1.87E+10	4.91E+10	3.03E-11	2.63	1.62E-21

TABLE 4.1-7

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

RADIUS (cm)	AZIMUTHAL ANGLE 0°	15°	30°	45°
168.04	1.000	1.000	1.000	1.000
168.27	0.987	0.987	0.985	0.987
168.88	0.940	0.942	0.937	0.942
169.75	0.862	0.865	0.857	0.866
170.93	0.754	0.757	0.749	0.760
172.25	0.639	0.644	0.636	0.647
173.53	0.540	0.546	0.539	0.550
174.98	0.444	0.451	0.444	0.454
176.46	0.362	0.370	0.363	0.372
177.58	0.308	0.317	0.311	0.318
179.03	0.250	0.259	0.253	0.260
180.66	0.196	0.206	0.201	0.206
181.63	0.169	0.179	0.175	0.178
182.60	0.144	0.154	0.151	0.154
184.06	0.110	0.122	0.120	0.122
184.87	0.101	0.113	0.112	0.113

Note: Base Metal Inner Radius = 168.04 cm.
Base Metal 1/4T = 172.25 cm.
Base Metal 1/2T = 176.46 cm.
Base Metal 3/4T = 180.66 cm.
Base Metal Outer Radius = 184.87 cm.

TABLE 4.1-8

RELATIVE RADIAL DISTRIBUTION OF NEUTRON FLUX (E > 0.1 MeV)
WITHIN THE PRESSURE VESSEL WALL

RADIUS	AZIMUTHAL ANGLE			
(cm)	0°	15°	30°	45°
168.04	1.000	1.000	1.000	1.000
168.27	1.005	1.007	1.005	1.007
168.88	1.002	1.007	1.004	1.008
169.75	0.980	0.990	0.985	0.992
170.93	0.934	0.948	0.945	0.953
172.25	0.873	0.891	0.889	0.899
173.53	0.809	0.831	0.831	0.841
174.98	0.736	0.763	0.763	0.773
176.46	0.662	0.693	0.694	0.703
177.58	0.606	0.640	0.642	0.650
179.03	0.536	0.573	0.577	0.582
180.66	0.461	0.502	0.507	0.509
181.63	0.416	0.458	0.466	0.465
182.60	0.369	0.415	0.423	0.421
184.06	0.298	0.348	0.361	0.357
184.87	0.276	0.327	0.343	0.339

Note: Base Metal Inner Radius = 168.04 cm.
Base Metal 1/4T = 172.25 cm.
Base Metal 1/2T = 176.46 cm.
Base Metal 3/4T = 180.66 cm.
Base Metal Outer Radius = 184.87 cm.

TABLE 4.1-9

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

RADIUS	AZIMUTHAL ANGLE			
(cm)	0°	15°	30°	45°
168.04	1.000	1.000	1.000	1.000
168.27	0.988	0.990	0.988	0.989
168.88	0.951	0.955	0.950	0.954
169.75	0.889	0.896	0.889	0.857
170.93	0.804	0.814	0.805	0.812
172.25	0.712	0.726	0.716	0.723
173.53	0.630	0.648	0.638	0.644
174.98	0.547	0.568	0.558	0.563
176.46	0.472	0.495	0.486	0.490
177.58	0.420	0.445	0.436	0.439
179.03	0.360	0.386	0.379	0.380
180.66	0.301	0.328	0.322	0.322
181.63	0.267	0.296	0.291	0.289
182.60	0.234	0.264	0.261	0.258
184.06	0.187	0.219	0.220	0.216
184.87	0.173	0.206	0.208	0.205

Note: Base Metal Inner Radius = 168.04 cm.
Base Metal 1/4T = 172.25 cm.
Base Metal 1/2T = 176.46 cm.
Base Metal 3/4T = 180.66 cm.
Base Metal Outer Radius = 184.87 cm.

4.2 - Fuel Cycle Specific Adjoint Calculations

Results of the fuel cycle specific adjoint transport calculations for the first 20 cycles of operation at Point Beach Unit 2 are summarized in Tables 4.2-1 through 4.2-18. The data listed in these tables establish the means for absolute comparison of analysis and measurement for the Cycles 15, 16, 17, and 18/20 cavity dosimetry irradiations as well as for the four sets of surveillance capsule dosimetry withdrawn to date. These results also provide the fuel cycle specific relationship among the surveillance capsule and reactor cavity measurement locations and key positions at the inner radius of the pressure vessel wall.

The core power distributions used in the cycle specific fast neutron exposure calculations for Fuel Cycles 1 through 20 were taken from the fuel cycle design reports applicable to Point Beach Unit 2[24 through 43]. The data extracted from the fuel cycle design reports represented cycle averaged relative fuel assembly powers and burnups as well as cycle averaged relative axial distributions. Therefore, the results of the adjoint evaluation provided data in terms of fuel cycle averaged neutron flux which, when multiplied by the appropriate fuel cycle length, produced the incremental fast neutron exposure for the fuel cycle.

The calculated fast neutron flux (E > 1.0 MeV) and cumulative fast neutron fluence at the center of surveillance capsules located at 13, 23, and 33 degrees are provided for each of the 20 operating fuel cycles in Tables 4.2-1 and 4.2-2, respectively. The data as tabulated are applicable to the axial core midplane. Similar data applicable to the pressure vessel inner radius are given in Tables 4.2-3 and 4.2-4 and data pertinent to the cavity dosimetry sensor locations are listed in Tables 4.2-5 and 4.2-6.

Exposure parameter ratios necessary to convert the cycle specific data listed in Tables 4.2-1 through 4.2-6 to other key fast neutron exposure units are given in Section 4.1 of this report. Application of these ratios to the data from Tables 4.2-1 through 4.2-6 yielded corresponding exposure data in terms of flux/fluence (E > 0.1 MeV) (Tables 4.2.7 through 4.2.12) and iron atom displacements (Tables 4.2.13 through 4.2.18).

TABLE 4.2-1

CALCULATED FAST NEUTRON FLUX (E > 1.0 MeV) AT THE CENTER OF REACTOR VESSEL SURVEILLANCE CAPSULES

	NEUTRON FLUX (n/cm^2-sec)		
CYCLE No.	13°	23°	33°
1	1.35E+11	7.71E+10	7.18E+10
2	1.37E+11	8.14E+10	7.84E+10
3	1.35E+11	7.99E+10	7.77E+10
4	1.28E+11	7.72E+10	7.42E+10
5	1.35E+11	8.09E+10	7.44E+10
6	1.06E+11	7.74E+10	7.68E+10
7	1.04E+11	6.73E+10	6.46E+10
8	1.05E+11	6.60E+10	6.16E+10
9	1.09E+11	6.71E+10	6.24E+10
10	9.88E+10	6.60E+10	6.39E+10
11	9.45E+10	6.86E+10	6.31E+10
12	9.90E+10	6.81E+10	6.02E+10
13	9.16E+10	6.44E+10	5.71E+10
14	9.43E+10	6.69E+10	6.21E+10
15	9.11E+10	6.47E+10	5.69E+10
16	7.32E+10	5.23E+10	4.98E+10
17	7.30E+10	5.32E+10	5.12E+10
18	7.14E+10	5.41E+10	5.28E+10
19	7.30E+10	5.35E+10	5.13E+10
20	7.24E+10	5.41E+10	5.42E+10

TABLE 4.2-2

CALCULATED FAST NEUTRON FLUENCE (E > 1.0 MeV) AT THE CENTER OF REACTOR VESSEL SURVEILLANCE CAPSULES

END OF CYCLE	IRRADIATION TIME (EFPS)	CUMULATIVE FLUENCE (n/cm^2) 13°	23°	33°
1	4.81E+07	6.50E+18	3.71E+18	3.45E+18
2	8.13E+07	1.11E+19	6.41E+18	6.05E+18
3	1.09E+08	1.48E+19	8.61E+18	8.19E+18
4	1.36E+08	1.83E+19	1.07E+19	1.02E+19
5	1.64E+08	2.20E+19	1.30E+19	1.23E+19
6	1.91E+08	2.49E+19	1.51E+19	1.44E+19
7	2.20E+08	2.79E+19	1.70E+19	1.62E+19
8	2.47E+08	3.07E+19	1.88E+19	1.79E+19
9	2.72E+08	3.35E+19	2.05E+19	1.94E+19
10	3.09E+08	3.72E+19	2.29E+19	2.18E+19
11	3.36E+08	3.97E+19	2.48E+19	2.35E+19
12	3.61E+08	4.22E+19	2.65E+19	2.51E+19
13	3.87E+08	4.45E+19	2.81E+19	2.65E+19
14	4.14E+08	4.71E+19	3.00E+19	2.82E+19
15	4.39E+08	4.94E+19	3.16E+19	2.96E+19
16	4.66E+08	5.14E+19	3.30E+19	3.10E+19
17	4.93E+08	5.33E+19	3.44E+19	3.23E+19
18	5.20E+08	5.52E+19	3.59E+19	3.38E+19
19	5.46E+08	5.71E+19	3.73E+19	3.51E+19
20	5.74E+08	5.92E+19	3.88E+19	3.66E+19

TABLE 4.2-3

CALCULATED FAST NEUTRON FLUX (E > 1.0 MeV) AT THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

	NEUTRON FLUX (n/cm^2-sec)			
CYCLE No.	0°	15°	30°	45°
1	4.53E+10	2.75E+10	1.81E+10	1.54E+10
2	4.60E+10	2.80E+10	1.96E+10	1.70E+10
3	4.59E+10	2.76E+10	1 94E+10	1.69E+10
4	4.31E+10	2.63E+10	1.86E+10	1.59E+10
5	4.39E+10	2.76E+10	1.90E+10	1.50E+10
6	3.41E+10	2.28E+10	1.93E+10	1.57E+10
7	3.60E+10	2.20E+10	1.63E+10	1.52E+10
8	3.74E+10	2.21E+10	1.57E+10	1.45E+10
9	3.84E+10	2.29E+10	1.59E+10	1.47E+10
10	3.40E+10	2.10E+10	1.61E+10	1.47E+10
11	2.83E+10	2.06E+10	1.63E+10	1.37E+10
12	3.07E+10	2.14E+10	1.58E+10	1.27E+10
13	2.84E+10	1.99E+10	1.50E+10	1.22E+10
14	2.86E+10	2.05E+10	1.60E+10	1.42E+10
15	2.77E+10	1.98E+10	1.50E+10	1.19E+10
16	2.26E+10	1.60E+10	1.27E+10	1.13E+10
17	2.27E+10	1.60E+10	1.30E+10	1.19E+10
18	2.17E+10	1.58E+10	1.34E+10	1.22E+10
19	2.28E+10	1.61E+10	1.31E+10	1.17E+10
20	2.24E+10	1.60E+10	1.36E+10	1.27E+10

TABLE 4.2-4

CALCULATED FAST NEUTRON FLUENCE (E > 1.0 MeV) AT THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

END OF CYCLE	IRRADIATION TIME (EFPS)	CUMULATIVE FLUENCE (n/cm^2) 0°	15°	30°	45°
1	4.81E+07	2.18E+18	1.32E+18	8.73E+17	7.40E+17
2	8.13E+07	3.71E+18	2.25E+18	1.52E+18	1.30E+18
3	1.09E+08	4.97E+18	3.01E+18	2.06E+18	1.77E+18
4	1.36E+08	6.15E+18	3.73E+18	2.57E+18	2.20E+18
5	1.64E+08	7.38E+18	4.50E+18	3.10E+18	2.62E+18
6	1.91E+08	8.31E+18	5.13E+18	3.62E+18	3.05E+18
7	2.20E+08	9.32E+18	5.75E+18	4.08E+18	3.48E+18
8	2.47E+08	1.03E+19	6.34E+18	4.51E+18	3.87E+18
9	2.72E+08	1.13E+19	6.92E+18	4.90E+18	4.24E+18
10	3.09E+08	1.26E+19	7.71E+18	5.51E+18	4.79E+18
11	3.36E+08	1.33E+19	8.26E+18	5.95E+18	5.16E+18
12	3.61E+08	1.41E+19	8.80E+18	6.35E+18	5.48E+18
13	3.87E+08	1.48E+19	9.31E+18	6.73E+18	5.79E+18
14	4.14E+08	1.56E+19	9.86E+18	7.16E+18	6.17E+18
15	4.39E+08	1.63E+19	1.04E+19	7.54E+18	6.48E+18
16	4.66E+08	1.69E+19	1.08E+19	7.88E+18	6.78E+18
17	4.93E+08	1.75E+19	1.12E+19	8.23E+18	7.10E+18
18	5.20E+08	1.81E+19	1.16E+19	8.59E+18	7.42E+18
19	5.46E+08	1.87E+19	1.21E+19	8.93E+18	7.73E+18
20	5.74E+08	1.93E+19	1.25E+19	9.31E+18	8.09E+18

TABLE 4.2-5

CALCULATED FAST NEUTRON FLUX (E > 1.0 MeV) AT THE CAVITY SENSOR SET LOCATIONS

	NEUTRON FLUX (n/cm^2-sec)			
CYCLE No.	0°	15°	30°	45°
1	3.46E+09	2.67E+09	1.78E+09	1.48E+09
2	3.52E+09	2.73E+09	1.90E+09	1.62E+09
3	3.50E+09	2.70E+09	1.88E+09	1.60E+09
4	3.29E+09	2.57E+09	1.80E+09	1.52E+09
5	3.38E+09	2.65E+09	1.82E+09	1.48E+09
6	2.67E+09	2.23E+09	1.78E+09	1.51E+09
7	2.75E+09	2.16E+09	1.58E+09	1.40E+09
8	2.82E+09	2.18E+09	1.54E+09	1.34E+09
9	2.91E+09	2.25E+09	1.57E+09	1.36E+09
10	2.60E+09	2.07E+09	1.55E+09	1.37E+09
11	2.27E+09	1.96E+09	1.53E+09	1.30E+09
12	2.43E+09	2.04E+09	1.50E+09	1.23E+09
13	2.25E+09	1.90E+09	1.42E+09	1.18E+09
14	2.28E+09	1.95E+09	1.52E+09	1.32E+09
15	2.21E+09	1.88E+09	1.42E+09	1.16E+09
16	1.80E+09	1.54E+09	1.21E+09	1.06E+09
17	1.81E+09	1.55E+09	1.24E+09	1.10E+09
18	1.74E+09	1.52E+09	1.26E+09	1.13E+09
19	1.81E+09	1.55E+09	1.24E+09	1.09E+09
20	1.79E+09	1.55E+09	1.29E+09	1.17E+09

TABLE 4.2-6

CALCULATED FAST NEUTRON FLUENCE (E > 1.0 MeV) AT THE CAVITY SENSOR SET LOCATIONS

END OF CYCLE	IRRADIATION TIME (EFPS)	CUMULATIVE FLUENCE (n/cm^2) 0°	15°	30°	45°
1	4.81E+07	1.67E+17	1.28E+17	8.55E+16	7.12E+16
2	8.13E+07	2.83E+17	2.19E+17	1.49E+17	1.25E+17
3	1.09E+08	3.80E+17	2.93E+17	2.00E+17	1.69E+17
4	1.36E+08	4.70E+17	3.64E+17	2.49E+17	2.11E+17
5	1.64E+08	5.64E+17	4.38E+17	3.00E+17	2.52E+17
6	1.91E+08	6.37E+17	4.99E+17	3.49E+17	2.93E+17
7	2.20E+08	7.15E+17	5.60E+17	3.94E+17	3.33E+17
8	2.47E+08	7.91E+17	6.19E+17	4.35E+17	3.69E+17
9	2.72E+08	8.63E+17	6.75E+17	4.74E+17	4.03E+17
10	3.09E+08	9.61E+17	7.53E+17	5.33E+17	4.54E+17
11	3.36E+08	1.02E+18	8.05E+17	5.74E+17	4.89E+17
12	3.61E+08	1.08E+18	8.57E+17	6.12E+17	5.20E+17
13	3.87E+08	1.14E+18	9.05E+17	6.48E+17	5.51E+17
14	4.14E+08	1.20E+18	9.58E+17	6.89E+17	5.87E+17
15	4.39E+08	1.26E+18	1.01E+18	7.25E+17	6.16E+17
16	4.66E+08	1.31E+18	1.05E+18	7.58E+17	6.45E+17
17	4.93E+08	1.36E+18	1.09E+18	7.91E+17	6.74E+17
18	5.20E+08	1.40E+18	1.13E+18	8.25E+17	7.04E+17
19	5.46E+08	1.45E+18	1.17E+18	8.57E+17	7.33E+17
20	5.74E+08	1.50E+18	1.21E+18	8.93E+17	7.65E+17

TABLE 4.2-7

CALCULATED FAST NEUTRON FLUX (E > 0.1 MeV) AT THE CENTER OF REACTOR VESSEL SURVEILLANCE CAPSULES

	NEUTRON FLUX (n/cm^2-sec)		
CYCLE No.	13°	23°	33°
1	5.12E+11	2.65E+11	2.53E+11
2	5.20E+11	2.80E+11	2.77E+11
3	5.13E+11	2.75E+11	2.74E+11
4	4.85E+11	2.66E+11	2.62E+11
5	5.10E+11	2.79E+11	2.62E+11
6	4.01E+11	2.67E+11	2.71E+11
7	3.95E+11	2.32E+11	2.28E+11
8	3.98E+11	2.27E+11	2.17E+11
9	4.15E+11	2.31E+11	2.20E+11
10	3.74E+11	2.27E+11	2.26E+11
11	3.58E+11	2.36E+11	2.23E+11
12	3.75E+11	2.35E+11	2.12E+11
13	3.47E+11	2.22E+11	2.01E+11
14	3.57E+11	2.30E+11	2.19E+11
15	3.45E+11	2.23E+11	2.01E+11
16	2.77E+11	1.80E+11	1.76E+11
17	2.76E+11	1.83E+11	1.81E+11
18	2.70E+11	1.86E+11	1.86E+11
19	2.77E+11	1.84E+11	1.81E+11
20	2.74E+11	1.86E+11	1.91E+11

TABLE 4.2-8

CALCULATED FAST NEUTRON FLUENCE (E > 0.1 MeV) AT THE CENTER OF REACTOR VESSEL SURVEILLANCE CAPSULES

END OF CYCLE	IRRADIATION TIME (EFPS)	CUMULATIVE FLUENCE (n/cm^2) 13°	23°	33°
1	4.81E+07	2.46E+19	1.28E+19	1.22E+19
2	8.13E+07	4.19E+19	2.21E+19	2.14E+19
3	1.09E+08	5.60E+19	2.96E+19	2.89E+19
4	1.36E+08	6.93E+19	3.69E+19	3.61E+19
5	1.64E+08	8.35E+19	4.47E+19	4.34E+19
6	1.91E+08	9.45E+19	5.19E+19	5.08E+19
7	2.20E+08	1.06E+20	5.85E+19	5.72E+19
8	2.47E+08	1.16E+20	6.46E+19	6.31E+19
9	2.72E+08	1.27E+20	7.04E+19	6.86E+19
10	3.09E+08	1.41E+20	7.89E+19	7.71E+19
11	3.36E+08	1.50E+20	8.53E+19	8.30E+19
12	3.61E+08	1.60E+20	9.12E+19	8.84E+19
13	3.87E+08	1.69E+20	9.68E+19	9.35E+19
14	4.14E+08	1.78E+20	1.03E+20	9.95E+19
15	4.39E+08	1.87E+20	1.09E+20	1.05E+20
16	4.66E+08	1.95E+20	1.14E+20	1.09E+20
17	4.93E+08	2.02E+20	1.18E+20	1.14E+20
18	5.20E+08	2.09E+20	1.23E+20	1.19E+20
19	5.46E+08	2.16E+20	1.28E+20	1.24E+20
20	5.74E+08	2.24E+20	1.33E+20	1.29E+20

TABLE 4.2-9

CALCULATED FAST NEUTRON FLUX (E > 0.1 MeV) AT THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

	NEUTRON FLUX (n/cm^2-sec)			
CYCLE No.	0°	15°	30°	45°
1	1.24E+11	8.00E+10	4.94E+10	4.05E+10
2	1.26E+11	8.17E+10	5.35E+10	4.47E+10
3	1.26E+11	8.05E+10	5.28E+10	4.44E+10
4	1.18E+11	7.66E+10	5.07E+10	4.20E+10
5	1.20E+11	8.05E+10	5.17E+10	3.95E+10
6	9.34E+10	6.65E+10	5.26E+10	4.12E+10
7	9.85E+10	6.42E+10	4.44E+10	4.00E+10
8	1.02E+10	6.45E+10	4.28E+10	3.81E+10
9	1.05E+10	6.68E+10	4.33E+10	3.86E+10
10	9.31E+10	6.13E+10	4.39E+10	3.88E+10
11	7.74E+10	6.00E+10	4.44E+10	3.60E+10
12	8.42E+10	6.22E+10	4.30E+10	3.34E+10
13	7.76E+10	5.80E+10	4.08E+10	3.22E+10
14	7.82E+10	5.97E+10	4.35E+10	3.74E+10
15	7.59E+10	5.78E+10	4.08E+10	3.13E+10
16	6.20E+10	4.66E+10	3.47E+10	2.98E+10
17	6.22E+10	4.67E+10	3.55E+10	3.13E+10
18	5.94E+10	4.61E+10	3.65E+10	3.22E+10
19	6.23E+10	4.68E+10	3.57E+10	3.09E+10
20	6.13E+10	4.65E+10	3.71E+10	3.35E+10

TABLE 4.2-10

CALCULATED FAST NEUTRON FLUENCE (E > 0.1 MeV) AT THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

END OF CYCLE	IRRADIATION TIME (EFPS)	CUMULATIVE FLUENCE (n/cm^2)			
		0°	15°	30°	45°
1	4.81E+07	5.97E+18	3.85E+18	2.38E+18	1.95E+18
2	8.13E+07	1.02E+19	6.56E+18	4.15E+18	3.43E+18
3	1.09E+08	1.36E+19	8.78E+18	5.61E+18	4.65E+18
4	1.36E+08	1.68E+19	1.09E+19	7.00E+18	5.80E+18
5	1.64E+08	2.02E+19	1.31E+19	8.44E+18	6.90E+18
6	1.91E+08	2.27E+19	1.49E+19	9.88E+18	8.03E+18
7	2.20E+08	2.55E+19	1.67E+19	1.11E+19	9.16E+18
8	2.47E+08	2.83E+19	1.85E+19	1.23E+19	1.02E+19
9	2.72E+08	3.09E+19	2.02E+19	1.34E+19	1.12E+19
10	3.09E+08	3.44E+19	2.25E+19	1.50E+19	1.26E+19
11	3.36E+08	3.65E+19	2.41E+19	1.62E+19	1.36E+19
12	3.61E+08	3.86E+19	2.56E+19	1.73E+19	1.44E+19
13	3.87E+08	4.06E+19	2.71E+19	1.83E+19	1.52E+19
14	4.14E+08	4.27E+19	2.87E+19	1.95E+19	1.63F+19
15	4.39E+08	4.47E+19	3.02E+19	2.06E+19	1.71E+19
16	4.66E+08	4.63E+19	3.15E+19	2.15E+19	1.79E+19
17	4.93E+08	4.80E+19	3.27E+19	2.24E+19	1.87E+19
18	5.20E+08	4.96E+19	3.39E+19	2.34E+19	1.95E+19
19	5.46E+08	5.12E+19	3.52E+19	2.43E+19	2.04E+19
20	5.74E+08	5.29E+19	3.65E+19	2.54E+19	2.13E+19

TABLE 4.2-11

CALCULATED FAST NEUTRON FLUX ($E > 0.1$ MeV) AT THE CAVITY SENSOR SET LOCATIONS

	NEUTRON FLUX (n/cm^2-sec)			
CYCLE No.	0°	15°	30°	45°
1	2.72E+10	2.30E+10	1.55E+10	1.22E+10
2	2.76E+10	2.35E+10	1.65E+10	1.33E+10
3	2.75E+10	2.33E+10	1.63E+10	1.32E+10
4	2.59E+10	2.21E+10	1.56E+10	1.25E+10
5	2.66E+10	2.29E+10	1.59E+10	1.21E+10
6	2.10E+10	1.92E+10	1.55E+10	1.24E+10
7	2.16E+10	1.86E+10	1.38E+10	1.15E+10
8	2.22E+10	1.88E+10	1.34E+10	1.10E+10
9	2.29E+10	1.94E+10	1.36E+10	1.12E+10
10	2.04E+10	1.78E+10	1.35E+10	1.12E+10
11	1.78E+10	1.69E+10	1.33E+10	1.07E+10
12	1.91E+10	1.75E+10	1.30E+10	1.01E+10
13	1.76E+10	1.63E+10	1.24E+10	9.69E+09
14	1.79E+10	1.68E+10	1.32E+10	1.09E+10
15	1.74E+10	1.62E+10	1.23E+10	9.53E+09
16	1.42E+10	1.32E+10	1.05E+10	8.69E+09
17	1.42E+10	1.33E+10	1.08E+10	9.04E+09
18	1.37E+10	1.31E+10	1.10E+10	9.27E+09
19	1.42E+10	1.33E+10	1.08E+10	8.97E+09
20	1.41E+10	1.33E+10	1.12E+10	9.57E+09

TABLE 4.2-12

CALCULATED FAST NEUTRON FLUENCE (E > 0.1 MeV) AT THE CAVITY SENSOR SET LOCATIONS

END OF CYCLE	IRRADIATION TIME (EFPS)	CUMULATIVE FLUENCE (n/cm^2) 0°	15°	30°	45°
1	4.81E+07	1.31E+18	1.10E+18	7.43E+17	5.84E+17
2	8.13E+07	2.23E+18	1.89E+18	1.29E+18	1.02E+18
3	1.09E+08	2.98E+18	2.53E+18	1.74E+18	1.39E+18
4	1.36E+08	3.69E+18	3.13E+18	2.17E+18	1.73E+18
5	1.64E+08	4.43E+18	3.77E+18	2.61E+18	2.07E+18
6	1.91E+08	5.01E+18	4.30E+18	3.03E+18	2.41E+18
7	2.20E+08	5.61E+18	4.82E+18	3.42E+18	2.73E+18
8	2.47E+08	6.21E+18	5.33E+18	3.78E+18	3.03E+18
9	2.72E+08	6.78E+18	5.81E+18	4.13E+18	3.31E+18
10	3.09E+08	7.55E+18	6.48E+18	4.63E+18	3.73E+18
11	3.36E+08	8.03E+18	6.94E+18	4.99E+18	4.02E+18
12	3.61E+08	8.51E+18	7.38E+18	5.32E+18	4.27E+18
13	3.87E+08	8.96E+18	7.80E+18	5.64E+18	4.52E+18
14	4.14E+08	9.45E+18	8.25E+18	6.00E+18	4.82E+18
15	4.39E+08	9.89E+18	8.67E+18	6.31E+18	5.06E+18
16	4.66E+08	1.03E+19	9.02E+18	6.59E+18	5.29E+18
17	4.93E+08	1.06E+19	9.38E+18	6.88E+18	5.53E+18
18	5.20E+08	1.10E+19	9.73E+18	7.17E+18	5.78E+18
19	5.46E+08	1.14E+19	1.01E+19	7.45E+18	6.01E+18
20	5.74E+08	1.18E+19	1.04E+19	7.77E+18	6.28E+18

TABLE 4.2-13

CALCULATED IRON ATOM DISPLACEMENT RATE AT THE CENTER OF REACTOR VESSEL SURVEILLANCE CAPSULES

	DISPLACEMENT RATE (dpa/sec)		
CYCLE No.	13°	23°	33°
1	2.41E-10	1.31E-10	1.23E-10
2	2.44E-10	1.39E-10	1.35E-10
3	2.41E-10	1.36E-10	1.34E-10
4	2.28E-10	1.32E-10	1.28E-10
5	2.40E-10	1.38E-10	1.28E-10
6	1.88E-10	1.32E-10	1.32E-10
7	1.86E-10	1.15E-10	1.11E-10
8	1.87E-10	1.12E-10	1.06E-10
9	1.99E-10	1.14E-10	1.07E-10
10	1.76E-10	1.12E-10	1.10E-10
11	1.68E-10	1.17E-10	1.09E-10
12	1.76E-10	1.16E-10	1.03E-10
13	1.63E-10	1.10E-10	9.81E-11
14	1.68E-10	1.14E-10	1.07E-10
15	1.62E-10	1.10E-10	9.79E-11
16	1.30E-10	8.92E-11	8.57E-11
17	1.30E-10	9.06E-11	8.81E-11
18	1.27E-10	9.22E-11	9.08E-11
19	1.30E-10	9.11E-11	8.82E-11
20	1.29E-10	9.21E-11	9.32E-11

TABLE 4.2-14

CALCULATED IRON ATOM DISPLACEMENTS AT THE
CENTER OF REACTOR VESSEL SURVEILLANCE CAPSULES

END OF CYCLE	IRRADIATION TIME (EFPS)	CUMULATIVE DISPLACEMENTS (dpa) 13°	23°	33°
1	4.81E+07	1.16E-02	6.32E-03	5.94E-03
2	8.13E+07	1.97E-02	1.09E-02	1.04E-02
3	1.09E+08	2.63E-02	1.47E-02	1.41E-02
4	1.36E+08	3.26E-02	1.83E-02	1.76E-02
5	1.64E+08	3.92E-02	2.21E-02	2.11E-02
6	1.91E+08	4.44E-02	2.57E-02	2.48E-02
7	2.20E+08	4.96E-02	2.90E-02	2.79E-02
8	2.47E+08	5.47E-02	3.20E-02	3.08E-02
9	2.72E+08	5.95E-02	3.48E-02	3.34E-02
10	3.09E+08	6.62E-02	3.91E-02	3.76E-02
11	3.36E+08	7.07E-02	4.22E-02	4.05E-02
12	3.61E+08	7.51E-02	4.51E-02	4.31E-02
13	3.87E+08	7.93E-02	4.79E-02	4.56E-02
14	4.14E+08	8.38E-02	5.10E-02	4.85E-02
15	4.39E+08	8.80E-02	5.38E-02	5.10E-02
16	4.66E+08	9.15E-02	5.62E-02	5.33E-02
17	4.93E+08	9.49E-02	5.87E-02	5.56E-02
18	5.20E+08	9.83E-02	6.11E-02	5.81E-02
19	5.46E+08	1.02E-01	6.35E-02	6.04E-02
20	5.74E+08	1.05E-01	6.61E-02	6.30E-02

TABLE 4.2-15

CALCULATED IRON ATOM DISPLACEMENT RATE AT THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

CYCLE No.	DISPLACEMENT RATE (dpa/sec) 0°	15°	30°	45°
1	7.40E-11	4.62E-11	2.98E-11	2.50E-11
2	7.51E-11	4.71E-11	3.22E-11	2.75E-11
3	7.50E-11	4.64E-11	3.18E-11	2.74E-11
4	7.03E-11	4.42E-11	3.06E-11	2.59E-11
5	7.16E-11	4.64E-11	3.12E-11	2.44E-11
6	5.57E-11	3.83E-11	3.17E-11	2.54E-11
7	5.87E-11	3.70E-11	2.68E-11	2.46E-11
8	6.10E-11	3.72E-11	2.58E-11	2.35E-11
9	6.26E-11	3.85E-11	2.61E-11	2.38E-11
10	5.55E-11	3.53E-11	2.65E-11	2.39E-11
11	4.61E-11	3.46E-11	2.68E-11	2.22E-11
12	5.02E-11	3.59E-11	2.59E-11	2.06E-11
13	4.63E-11	3.34E-11	2.45E-11	1.98E-11
14	4.66E-11	3.44E-11	2.62E-11	2.30E-11
15	4.53E-11	3.33E-11	2.46E-11	1.93E-11
16	3.70E-11	2.69E-11	2.09E-11	1.84E-11
17	3.71E-11	2.69E-11	2.14E-11	1.93E-11
18	3.54E-11	2.66E-11	2.20E-11	1.98E-11
19	3.71E-11	2.70E-11	2.15E-11	1.90E-11
20	3.65E-11	2.68E-11	2.23E-11	2.06E-11

TABLE 4.2-16

CALCULATED IRON ATOM DISPLACEMENTS AT THE PRESSURE VESSEL CLAD/BASE METAL INTERFACE

END OF CYCLE	IRRADIATION TIME (EFPS)	CUMULATIVE DISPLACEMENTS (dpa) 0°	15°	30°	45°
1	4.81E+07	3.56E-03	2.22E-03	1.43E-03	1.20E-03
2	8.13E+07	6.05E-03	3.79E-03	2.50E-03	2.12E-03
3	1.09E+08	8.11E-03	5.06E-03	3.38E-03	2.87E-03
4	1.36E+08	1.00E-02	6.27E-03	4.21E-03	3.58E-03
5	1.64E+08	1.20E-02	7.57E-03	5.08E-03	4.26E-03
6	1.91E+08	1.36E-02	8.62E-03	5.95E-03	4.95E-03
7	2.20E+08	1.52E-02	9.66E-03	6.70E-03	5.65E-03
8	2.47E+08	1.69E-02	1.07E-02	7.40E-03	6.28E-03
9	2.72E+08	1.84E-02	1.16E-02	8.05E-03	6.88E-03
10	3.09E+08	2.05E-02	1.30E-02	9.05E-03	7.78E-03
11	3.36E+08	2.18E-02	1.39E-02	9.77E-03	8.38E-03
12	3.61E+08	2.30E-02	1.48E-02	1.04E-02	8.89E-03
13	3.87E+08	2.42E-02	1.56E-02	1.11E-02	9.40E-03
14	4.14E+08	2.55E-02	1.66E-02	1.18E-02	1.00E-02
15	4.39E+08	2.66E-02	1.74E-02	1.24E-02	1.05E-02
16	4.66E+08	2.76E-02	1.82E-02	1.29E-02	1.10E-02
17	4.93E+08	2.86E-02	1.89E-02	1.35E-02	1.15E-02
18	5.20E+08	2.95E-02	1.96E-02	1.41E-02	1.21E-02
19	5.46E+08	3.05E-02	2.03E-02	1.47E-02	1.26E-02
20	5.74E+08	3.15E-02	2.10E-02	1.53E-02	1.31E-02

TABLE 4.2-17

CALCULATED IRON ATOM DISPLACEMENT RATE AT THE CAVITY SENSOR SET LOCATIONS

	DISPLACEMENT RATE (dpa/sec)			
CYCLE No.	0°	15°	30°	45°
1	9.76E-12	8.08E-12	5.44E-12	4.34E-12
2	9.92E-12	8.29E-12	5.82E-12	4.74E-12
3	9.87E-12	8.19E-12	5.75E-12	4.70E-12
4	9.29E-12	7.78E-12	5.50E-12	4.47E-12
5	9.53E-12	8.07E-12	5.58E-12	4.34E-12
6	7.54E-12	6.76E-12	5.45E-12	4.43E-12
7	7.74E-12	6.56E-12	4.85E-12	4.11E-12
8	7.95E-12	6.61E-12	4.72E-12	3.94E-12
9	8.20E-12	6.81E-12	4.80E-12	4.00E-12
10	7.33E-12	6.27E-12	4.75E-12	4.01E-12
11	6.39E-12	5.94E-12	4.70E-12	3.83E-12
12	6.85E-12	6.17E-12	4.59E-12	3.62E-12
13	6.33E-12	5.75E-12	4.35E-12	3.46E-12
14	6.43E-12	5.92E-12	4.66E-12	3.89E-12
15	6.23E-12	5.71E-12	4.33E-12	3.41E-12
16	5.08E-12	4.66E-12	3.71E-12	3.11E-12
17	5.10E-12	4.68E-12	3.80E-12	3.23E-12
18	4.92E-12	4.62E-12	3.87E-12	3.31E-12
19	5.11E-12	4.70E-12	3.80E-12	3.21E-12
20	5.05E-12	4.69E-12	3.94E-12	3.42E-12

TABLE 4.2-18

CALCULATED IRON ATOM DISPLACEMENTS AT THE CAVITY SENSOR SET LOCATIONS

END OF CYCLE	IRRADIATION TIME (EFPS)	CUMULATIVE FLUENCE (n/cm^2) 0°	15°	30°	45°
1	4.81E+07	4.70E-04	3.89E-04	2.62E-04	2.09E-04
2	8.13E+07	7.99E-04	6.64E-04	4.55E-04	3.66E-04
3	1.09E+08	1.07E-03	8.89E-04	6.13E-04	4.96E-04
4	1.36E+08	1.33E-03	1.10E-03	7.64E-04	6.18E-04
5	1.64E+08	1.59E-03	1.33E-03	9.19E-04	7.39E-04
6	1.91E+08	1.80E-03	1.51E-03	1.07E-03	8.60E-04
7	2.20E+08	2.02E-03	1.70E-03	1.21E-03	9.76E-04
8	2.47E+08	2.23E-03	1.87E-03	1.33E-03	1.08E-03
9	2.72E+08	2.43E-03	2.05E-03	1.45E-03	1.18E-03
10	3.09E+08	2.71E-03	2.28E-03	1.63E-03	1.33E-03
11	3.36E+08	2.88E-03	2.44E-03	1.76E-03	1.44E-03
12	3.61E+08	3.06E-03	2.60E-03	1.87E-03	1.53E-03
13	3.87E+08	3.22E-03	2.74E-03	1.98E-03	1.62E-03
14	4.14E+08	3.39E-03	2.90E-03	2.11E-03	1.72E-03
15	4.39E+08	3.55E-03	3.05E-03	2.22E-03	1.81E-03
16	4.66E+08	3.69E-03	3.17E-03	2.32E-03	1.89E-03
17	4.93E+08	3.82E-03	3.30E-03	2.42E-03	1.98E-03
18	5.20E+08	3.95E-03	3.42E-03	2.52E-03	2.07E-03
19	5.46E+08	4.09E-03	3.55E-03	2.62E-03	2.15E-03
20	5.74E+08	4.23E-03	3.68E-03	2.73E-03	2.25E-03

SECTION 5.0

EVALUATIONS OF SURVEILLANCE CAPSULE DOSIMETRY

In this section, the results of the evaluations of the four neutron sensor sets withdrawn as a part of the Point Beach Unit 2 Reactor Vessel Materials Surveillance Program are presented. The capsule designation, location within the reactor, and time of withdrawal of each of these dosimetry sets were as follows:

CAPSULE ID	AZIMUTHAL LOCATION	WITHDRAWAL TIME	IRRADIATION TIME (EFPS)
V	13°	END OF CYCLE 1	4.81E+07
T	23°	END OF CYCLE 3	1.09E+08
R	13°	END OF CYCLE 5	1.64E+08
S	33°	END OF CYCLE 16	4.66E+08

5.1 - Measured Reaction Rates

With the exception of Capsule V, radiometric counting of each of these capsule dosimetry data sets was accomplished by Westinghouse using the procedures discussed in Section 3.0 of this report[43,44,45]. The measured specific activities are included in Appendix A to this report. Radiometric counting of the sensors from Capsule V, on the other hand, was carried out by the Battelle Memorial Institute[46]. However, in this case, the measured specific activities were not reported.

The irradiation history of the Point Beach Unit 2 reactor during the first 16 fuel cycles is also listed in Appendix A. The irradiation history was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" for the applicable operating periods. In addition to the reactor power history, for the multiple cycle irradiations Capsules T,R, and S, the flux level adjustment factors for each cycle are also tabulated in Appendix A. These adjustment factors were determined from the fuel cycle specific adjoint calculations described in Section 4.2 of this report.

Based on the irradiation history, the individual sensor characteristics, capsule gradient corrections, and the measured specific activities, reaction rates averaged over the appropriate irradiation periods and referenced to a core power level of 1518 MWt were computed for the sensor sets removed from Capsules T, R, and S. In the case of Capsule V, reaction rates were developed directly from the derived neutron flux and spectrum averaged reaction cross-sections reported in reference 46. The computed reaction rates for the multiple foil sensor sets from each of the four internal surveillance capsules are provided in Table 5.1-1.

In regard to the data listed in Table 5.1-1, the fission rate measurements for the U-238 sensors include corrections for U-235 impurities, the build-in of Plutonium isotopes during the long irradiations, and for the effects of γ,f reactions. Likewise, the fission rate measurements for the Np-237 sensors include adjustments for γ,f reactions occuring over the course of the respective irradiation periods.

5.2 - Results of the Least Squares Adjustment Procedure

The results of the application of the least squares adjustment procedure to the four sets of surveillance capsule dosimetry are provided in Table 5.2-1 through 5.2-4. In these tables, the derived exposure experienced by the capsule along with data illustrating the fit of both the trial and adjusted spectra to the measurements are given. Also included in the tabulations are the 1σ uncertainties associated with each of the derived exposure rates.

In regard to the comparisons listed in Table 5.2-1 and 5.2-2, it should be noted that the columns labeled "trial calc" were obtained by normalizing the neutron spectral data from Table 4.1-3 to the absolute calculated neutron flux (E > 1.0 MeV) averaged over the applicable irradiation periods (Cycle 1 for Capsule V, Cycles 1 through 3 for Capsule T, Cycles 1 through 5 for Capsule R, and Cycles 1 through 16 for Capsule S) as discussed in Section 3.0. Thus, the comparisons illustrated in Tables 5.2-1 through 5.2-4 indicate the degree to which the calculated neutron energy specta matched the measured sensor data before and after adjustment. Absolute comparisons are discussed further in Section 7.0 of this report.

TABLE 5.1-1

SUMMARY OF REACTION RATES DERIVED FROM MULTIPLE FOIL SENSOR SETS WITHDRAWN FROM INTERNAL SURVEILLANCE CAPSULES

REACTION	REACTION RATE (rps/nucleus) CAPSULE V	CAPSULE T	CAPSULE R	CAPSULE S
Cu63(n,α)Co60	6.28E-17	4.86E-17	6.77E-17	4.26E-17
Fe54(n,p)Mn54	7.79E-15	5.53E-15	7.86E-15	
Ni58(n,p)Co58		7.34E-15	1.12E-14	5.51E-15
U238(n,f)Cs137 Cd	4.84E-14	2.69E-14	4.54E-14	2.46E-14
Np237(n,f)Cs137 Cd	4.24E-13	2.37E-13	4.19E-14	1.95E-13
Co59(n,γ)Co60	7.60E-12	5.13E-12	9.34E-12	3.72E-12
Co59(n,γ)Co60 Cd	3.10E-12	1.97E-12	3.79E-12	1.60E-12

TABLE 5.2.1

DERIVED EXPOSURE RATES FROM SURVEILLANCE CAPSULE V DOSIMETRY
WITHDRAWN AT THE END OF FUEL CYCLE 1

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.35E+11	1.44E+11	8%
ϕ(E > 0.1 MeV)	5.12E+11	5.71E+11	15%
ϕ(E < 0.414 eV)	2.46E+11	1.92E+11	20%
dpa/sec	2.41E-10	2.60E-10	11%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
SURVEILLANCE CAPSULE V

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)		6.28E-17	6.90E-17	6.26E-17	0.91	1.00
Fe-54(n,p)		7.79E-15	8.44E-15	7.96E-15	0.92	0.98
U-238(n,f)	Cd	4.84E-14	4.37E-14	4.54E-14	1.11	1.07
Np-237(n,f)	Cd	4.24E-13	3.77E-13	4.18E-13	1.13	1.01
Co-59(n,γ)		7.60E-12	9.62E-12	7.61E-12	0.79	1.00
Co-59(n,γ)	Cd	3.10E-12	3.74E-12	3.10E-12	0.83	1.00

TABLE 5.2.2

DERIVED EXPOSURE RATES FROM SURVEILLANCE CAPSULE T DOSIMETRY
WITHDRAWN AT THE END OF FUEL CYCLE 3

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
$\phi(E > 1.0\ MeV)$	7.91E+10	8.21E+10	8%
$\phi(E > 0.1\ MeV)$	2.72E+11	2.97E+11	15%
$\phi(E < 0.414\ eV)$	1.28E+11	1.29E+11	18%
dpa/sec	1.35E-10	1.43E-10	10%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
SURVEILLANCE CAPSULE T

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)		4.86E-17	5.13E-17	4.86E-17	0.95	1.00
Fe-54(n,p)		5.53E-15	5.62E-15	5.49E-15	0.98	1.01
Ni-58(n,p)		7.34E-15	7.73E-15	7.43E-15	0.95	0.99
U-238(n,f)	Cd	2.69E-14	2.67E-14	2.70E-14	1.01	1.00
Np-237(n,f)	Cd	2.37E-13	2.10E-13	2.30E-13	1.13	1.03
Co-59(n,γ)		5.13E-12	4.97E-12	5.13E-12	1.03	1.00
Co-59(n,γ)	Cd	1.97E-12	1.84E-12	1.97E-12	1.07	1.00

TABLE 5.2.3

DERIVED EXPOSURE RATES FROM SURVEILLANCE CAPSULE R DOSIMETRY
WITHDRAWN AT THE END OF FUEL CYCLE 5

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
φ(E > 1.0 MeV)	1.34E+11	1.39E+11	8%
φ(E > 0.1 MeV)	5.09E+11	5.61E+11	15%
φ(E < 0.414 eV)	2.45E+11	2.34E+11	19%
dpa/sec	2.39E-10	2.55E-10	11%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
SURVEILLANCE CAPSULE R

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)		6.77E-17	6.86E-17	6.71E-17	0.99	1.01
Fe-54(n,p)		7.86E-15	8.39E-15	8.00E-15	0.94	0.98
Ni-58(n,p)		1.12E-14	1.17E-14	1.13E-14	0.96	0.99
U-238(n,f)	Cd	4.54E-14	4.35E-14	4.38E-14	1.04	1.04
Np-237(n,f)	Cd	4.19E-13	3.75E-13	4.08E-13	1.12	1.03
Co-59(n,γ)		9.34E-12	9.56E-12	9.34E-12	0.98	1.00
Co-59(n,γ)	Cd	3.79E-12	3.72E-12	3.79E-12	1.02	1.00

TABLE 5.2.4

DERIVED EXPOSURE RATES FROM SURVEILLANCE CAPSULE S DOSIMETRY
WITHDRAWN AT THE END OF FUEL CYCLE 16

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
φ(E > 1.0 MeV)	6.65E+10	6.81E+10	8%
φ(E > 0.1 MeV)	2.35E+11	2.54E+11	15%
φ(E < 0.414 eV)	1.08E+11	8.91E+10	20%
dpa/sec	1.14E-10	1.20E-10	10%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
SURVEILLANCE CAPSULE S

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)		4.26E-17	3.98E-17	4.18E-17	1.07	1.02
Ni-58(n,p)		5.51E-15	6.24E-17	5.77E-15	0.88	0.96
U-238(n,f)	Cd	2.46E-14	2.21E-14	2.25E-14	1.11	1.09
Np-237(n,f)	Cd	1.95E-13	1.79E-13	1.91E-13	1.09	1.02
Co-59(n,γ)		3.72E-12	4.23E-12	3.73E-12	0.88	1.00
Co-59(n,γ)	Cd	1.60E-12	1.62E-12	1.60E-12	0.99	1.00

SECTION 6.0

EVALUATIONS OF REACTOR CAVITY DOSIMETRY

In this section, the results of the evaluations of all neutron sensor sets irradiated since the inception of the Reactor Cavity Measurement Program are presented. At Point Beach Unit 2 the program was initiated prior to the startup of Cycle 15; and, to date, has included measurement evaluations at the conclusion of Cycles 15, 16, 17 and 20. The evaluation of each of these sets of measured data was accomplished using a consistent approach based on the methodology discussed in Section 3.0, resulting in an accurate data base to be used in defining the best estimate neutron exposure of the reactor vessel wall.

6.1 - Cycle 15 Results

6.1.1 - Measured Reaction rates

During the Cycle 15 irradiation, seven multiple foil sensor sets, and four stainless steel gradient chains were deployed in the reactor cavity as depicted in Figures 2.1-1 and 2.1-2. The capsule identifications associated with each of the multiple foil sensor sets were as follows:

	CAPSULE IDENTIFICATION			
AZIMUTH (degrees)	VESSEL SUPPORT	CORE TOP	CORE MIDPLANE	CORE BOTTOM
0.0	XX	G	H	I
15.0			J	
30.0			K	
45.0			L	

The contents of each of these irradiation capsules is specified in Appendix B to this report.

The irradiation history of the Point Beach Unit 2 reactor during Cycle 15 is also listed in Appendix B. The irradiation history was obtained from NUREG-0020, "Licensed Operating Reactors Status

Summary Report" for the applicable operating period. Based on this reactor operating history, the individual sensor characteristics, and the measured specific activities given in Appendix B, cycle average reaction rates referenced to a core power level of 1518 MWt were computed for each multiple foil sensor and gradient chain segment.

The computed reaction rates for the multiple foil sensor sets, including radiometric foils and solid state track recorders, irradiated during Cycle 15 are provided in Table 6.1-1. Corresponding reaction rate data from the the four stainless steel gradient chains are recorded in Tables 6.1-2 through 6.1-4 for the Fe-54(n,p), Ni-58(n,p), and Co-59(n,γ) reactions, respectively.

In regard to the data listed in Table 6.1-1, the Fe-54(n,p) reaction rates represent an average of the bare and cadmium covered measurements for each capsule. Likewise, the U-238(n,f) reaction rates were obtained by averaging the results of the radiometric foil and solid state track recorder data. In addition, the fission rate measurements include corrections for U-235 impurities in the U-238 sensors as well as corrections for photo-fission reactions in both the U-238 and Np-237 sensors.

6.1.2 - Results of the Least Squares Adjustment Procedure

The results of the application of the least squares adjustment procedure to the seven sets of multiple foil measurements obtained from the Cycle 15 irradiation are provided in Tables 6.1-5 through 6.1-11. In these tables, the derived exposure experienced at each sensor set location along with data illustrating the fit of both the trial and adjusted spectra to the measurements are given. Also included in the tabulations are the 1σ uncertainties associated with each of the derived exposure rates.

In regard to the comparisons listed in Tables 6.1-5 through 6.1-11, it should be noted that the columns labeled "trial calc" were obtained by normalizing the neutron spectral data from Table 4.1-2 to the absolute calculated neutron flux (E > 1.0 MeV) averaged over the Cycle 15 irradiation period as discussed in Section 3.0. Thus, the comparisons illustrated in Tables 6.1-5 through 6.1-11 indicate the degree to which the calculated neutron energy spectra matched the measured data before and after adjustment. Absolute comparisons of calculation and measurement are discussed further in Section 7.0 of this report.

Complete traverses of fast neutron exposure rates in the reactor cavity were developed by combining the results of the least squares adjustment of the multiple foil data with the Fe-54(n,p) reaction rate measurements from the gradient chains. The gradient data were employed to establish relative axial distributions over the measurement range and these relative distributions were then normalized to the FERRET results from the midplane sensor sets to produce axial distributions of exposure rates in terms of $\phi(E > 1.0\ MeV)$, $\phi(E > 0.1\ MeV)$, and dpa/sec in the reactor cavity.

The resultant axial distributions of $\phi(E > 1.0\ MeV)$, $\phi(E > 0.1\ MeV)$, and dpa/sec from the gradient chain measurements are given in Tables 6.1-12, 6.1-13, and 6-14, respectively. The distributions of $\phi(E > 1.0\ MeV)$ are depicted graphically in Figures 6.1-1 through 6.1-4. In these graphical presentations, results for axial locations of -6.0, 0.0, and +6.0 feet relative to the core midplane represent the explicit results of the FERRET evaluations summarized in Tables 6.1-5 through 6.1-10, while results at the remaining axial locations depict the normalized data from the gradient chains.

TABLE 6.1-1

SUMMARY OF REACTION RATES DERIVED FROM MULTIPLE FOIL SENSOR SETS
CYCLE 15 IRRADIATION

REACTION RATE (rps/nucleus)

		CAPSULE H	CAPSULE J	CAPSULE K	CAPSULE L	CAPSULE G	CAPSULE I	CAPSULE XX
Cu-63(n,α)	Cd	1.07E-18	9.85E-19	7.45E-19	7.14E-19	3.86E-19	3.83E-19	3.72E-20
Ti-46(n,p)	Cd	1.66E-17	1.52E-17	1.14E-17	1.08E-17	6.55E-18	6.49E-18	6.23E-19
Fe-54(n,p)	Cd	9.49E-17	8.52E-17	6.58E-17	5.52E-17	3.39E-17	3.64E-17	3.76E-18
Ni-58(n,p)	Cd	1.38E-16	1.21E-16	9.02E-17	8.06E-17	5.35E-17	5.24E-17	5.92E-18
U-238(n,f)	Cd	5.49E-16	5.02E-16	3.73E-16	3.31E-16	2.19E-16	1.97E-16	3.27E-17
Np-237(n,f)	Cd	8.02E-15	7.82E-15	5.61E-15	4.89E-15	3.39E-15	3.23E-15	7.58E-16
Co-59(n,γ)		1.14E-13	1.42E-13	1.16E-13	7.34E-14	3.86E-14	4.53E-14	1.41E-14
Co-59(n,γ)	Cd	6.71E-14	8.11E-14	6.60E-14	4.81e-14	2.85E-14	2.94E-14	1.01E-14
U-235(n,f)		9.60E-13	1.18E-12	1.19E-12	5.47E-13	3.09E-13	3.52E-13	1.16E-13
U-235(n,f)	Cd	3.12E-13	4.05E-13	3.12E-13	2.05E-13	8.26E-14	7.52E-14	5.16E-14

Note: Cd indicates that the sensor was cadmium covered.

TABLE 6.1-2

Fe-54(n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLE 15 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	1.99E-17	1.54E-17	1.30E-17	1.15E-17
+5.5	4.85E-17	4.25E-17	3.17E-17	2.65E-17
+4.5	7.10E-17	6.46E-17	4.86E-17	4.25E-17
+3.5	8.31E-17	7.16E-17	5.35E-17	4.73E-17
+2.5	9.27E-17	7.68E-17	5.71E-17	5.22E-17
+1.5	9.15E-17	7.61E-17	5.44E-17	4.98E-17
+0.5	8.89E-17	7.68E-17	5.50E-17	5.01E-17
-0.5	9.27E-17	8.19E-17	5.95E-17	5.15E-17
-1.5	8.51E-17	7.93E-17	5.68E-17	5.05E-17
-2.5	8.12E-17	7.87E-17	6.03E-17	4.96E-17
-3.5	7.99E-17	7.36E-17	5.16E-17	4.73E-17
-4.5	7.16E-17	6.72E-17	4.51E-17	4.53E-17
-5.5	4.83E-17	4.31E-17	2.85E-17	2.84E-17
-6.5	1.78E-17	1.65E-17	1.24E-17	1.17E-17

TABLE 6.1-3

Ni-58(n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLE 15 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	2.96E-17	2.34E-17	1.91E-17	1.76E-17
+5.5	7.35E-17	5.81E-17	4.89E-17	4.07E-17
+4.5	1.08E-16	9.40E-17	6.95E-17	6.15E-17
+3.5	1.21E-16	1.01E-16	7.80E-17	6.83E-17
+2.5	1.32E-16	1.14E-16	8.43E-17	7.35E-17
+1.5	1.30E-16	1.12E-16	8.54E-17	7.17E-17
+0.5	1.22E-16	1.11E-16	8.09E-17	7.12E-17
-0.5	1.28E-16	1.16E-16	8.37E-17	7.52E-17
-1.5	1.18E-16	1.12E-16	8.09E-17	7.23E-17
-2.5	1.21E-16	1.06E-16	8.03E-17	7.12E-17
-3.5	1.16E-16	1.06E-16	7.63E-17	6.61E-17
-4.5	1.06E-16	9.74E-17	6.83E-17	6.21E-17
-5.5	7.12E-17	6.43E-17	4.52E-17	4.30E-17
-6.5	2.79E-17	2.56E-17	1.89E-17	1.72E-17

TABLE 6.1-4

Co-59(n,γ) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLE 15 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	2.40E-14	2.47E-14	2.18E-14	1.83E-14
+5.5	5.40E-14	7.97E-14	6.10E-14	3.75E-14
+4.5	6.95E-14	1.15E-13	8.76E-14	5.21E-14
+3.5	8.53E-14	1.33E-13	1.03E-13	5.99E-14
+2.5	9.49E-14	1.44E-13	1.10E-13	6.73E-14
+1.5	9.78E-14	1.44E-13	1.15E-13	7.06E-14
+0.5	1.01E-13	1.40E-13	1.19E-13	7.06E-14
-0.5	1.14E-13	1.44E-13	1.17E-13	7.18E-14
-1.5	1.06E-13	1.37E-13	1.14E-13	7.01E-14
-2.5	1.01E-13	1.29E-13	1.09E-13	6.73E-14
-3.5	9.27E-14	1.20E-13	9.89E-14	6.16E-14
-4.5	7.46E-14	1.00E-13	7.86E-14	5.11E-14
-5.5	4.94E-14	7.12E-14	4.21E-14	3.63E-14
-6.5	3.81E-14	3.85E-14	2.93E-14	2.76E-14

TABLE 6.1-5

DERIVED EXPOSURE RATES FROM THE CAPSULE H DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
$\phi(E > 1.0$ MeV)	2.22E+09	1.95E+09	8%
$\phi(E > 0.1$ MeV)	1.74E+10	1.62E+10	17%
$\phi(E < 0.414$ eV)	4.17E+09	1.90E+09	25%
dpa/sec	1.24E-12	5.72E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	1.07E-18	1.10E-18	1.07E-18	0.97	1.00
Ti-46(n,p)	Cd	1.66E-17	1.64E-17	1.63E-17	1.01	1.02
Fe-54(n,p)	Cd	9.49E-17	1.07E-16	9.65E-17	0.89	0.98
Ni-58(n,p)	Cd	1.38E-16	1.54E-16	1.38E-16	0.90	1.00
U-238(n,f)	Cd	5.49E-16	6.35E-16	5.58E-16	0.87	0.98
Np-237(n,f)	Cd	8.02E-15	8.62E-15	7.86E-15	0.93	1.02
Co-59(n,γ)		1.14E-13	2.08E-13	1.14E-13	0.55	1.00
Co-59(n,γ)	Cd	6.71E-14	9.78E-14	6.73E-14	0.69	1.00
U-235(n,f)		9.60E-13	1.95E-12	9.71E-13	0.49	0.99
U-235(n,f)	Cd	3.12E-13	3.47E-13	3.05E-13	0.90	1.02

TABLE 6.1-6

DERIVED EXPOSURE RATES FROM THE CAPSULE J DOSIMETRY EVALUATION
15.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.89E+09	1.78E+09	8%
ϕ(E > 0.1 MeV)	1.63E+10	1.66E+10	17%
ϕ(E < 0.414 eV)	3.76E+09	2.40E+09	25%
dpa/sec	5.73E-12	5.74E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
15.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	9.85E-19	9.11E-19	9.83E-19	1.08	1.00
Ti-46(n,p)	Cd	1.52E-17	1.36E-17	1.49E-17	1.12	1.02
Fe-54(n,p)	Cd	8.52E-17	8.83E-17	8.65E-17	0.97	0.99
Ni-58(n,p)	Cd	1.21E-16	1.27E-16	1.23E-16	0.95	0.98
U-238(n,f)	Cd	5.02E-16	5.33E-16	5.04E-16	0.94	1.00
Np-237(n,f)	Cd	7.82E-15	7.70E-15	7.65E-15	1.02	1.02
Co-59(n,γ)		1.42E-13	1.98E-13	1.41E-13	0.72	1.01
Co-59(n,γ)	Cd	8.11E-14	9.93E-14	8.20E-14	0.82	0.99
U-235(n,f)		1.18E-12	1.78E-12	1.22E-12	0.66	0.97
U-235(n,f)	Cd	4.05E-13	3.49E-13	3.87E-13	1.16	1.05

TABLE 6.1-7

DERIVED EXPOSURE RATES FROM THE CAPSULE K DOSIMETRY EVALUATION
30.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.42E+09	1.31E+09	8%
ϕ(E > 0.1 MeV)	1.24E+10	1.21E+10	17%
ϕ(E < 0.414 eV)	3.01E+09	2.21E+09	24%
dpa/sec	4.35E-12	4.20E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
30.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	7.45E-19	7.34E-19	7.46E-19	1.02	1.00
Ti-46(n,p)	Cd	1.14E-17	1.08E-17	1.12E-17	1.06	1.02
Fe-54(n,p)	Cd	6.58E-17	6.85E-17	6.59E-17	0.96	1.00
Ni-58(n,p)	Cd	9.02E-17	9.84E-17	9.18E-17	0.92	0.98
U-238(n,f)	Cd	3.73E-16	4.03E-16	3.75E-16	0.93	1.00
Np-237(n,f)	Cd	5.60E-15	5.80E-15	5.54E-15	0.97	1.01
Co-59(n,γ)		1.16E-13	1.58E-13	1.20E-13	0.73	0.97
Co-59(n,γ)	Cd	6.60E-14	7.97E-14	6.51E-14	0.83	1.01
U-235(n,f)		1.19E-12	1.42E-12	1.10E-12	0.84	1.08
U-235(n,f)	Cd	3.12E-13	2.80E-13	3.08E-13	1.11	1.01

TABLE 6.1-8

DERIVED EXPOSURE RATES FROM THE CAPSULE L DOSIMETRY EVALUATION
45.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.17E+09	1.12E+09	8%
ϕ(E > 0.1 MeV)	9.60E+09	9.99E+09	17%
ϕ(E < 0.414 eV)	2.75E+09	1.03E+09	26%
dpa/sec	3.43E-12	3.49E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
45.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	7.14E-19	6.68E-19	7.10E-19	1.07	1.01
Ti-46(n,p)	Cd	1.08E-17	9.68E-18	1.05E-17	1.12	1.03
Fe-54(n,p)	Cd	5.52E-17	5.96E-17	5.71E-17	0.93	0.97
Ni-58(n,p)	Cd	8.06E-17	8.49E-17	8.14E-17	0.95	0.99
U-238(n,f)	Cd	3.31E-16	3.39E-16	3.26E-16	0.98	1.02
Np-237(n,f)	Cd	4.88E-15	4.64E-15	4.73E-15	1.05	1.03
Co-59(n,γ)		7.34E-14	1.35E-13	7.37E-14	0.54	1.00
Co-59(n,γ)	Cd	4.81E-14	6.20E-14	4.81E-14	0.78	1.00
U-235(n,f)		5.47E-13	1.28E-12	5.56E-13	0.43	0.98
U-235(n,f)	Cd	2.05E-13	2.19E-13	2.01E-13	0.94	1.02

TABLE 6.1-9

DERIVED EXPOSURE RATES FROM THE CAPSULE G DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE TOP

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	7.84E+08	7.73E+08	8%
ϕ(E > 0.1 MeV)	6.16E+09	6.41E+09	16%
ϕ(E < 0.414 eV)	1.47E+09	5.37E+08	25%
dpa/sec	2.21E-12	2.26E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE TOP

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	3.86E-19	3.88E-19	3.88E-19	1.00	1.00
Ti-46(n,p)	Cd	6.55E-18	5.81E-18	6.30E-18	1.13	1.04
Fe-54(n,p)	Cd	6.39E-17	3.78E-17	3.57E-17	0.90	0.95
Ni-58(n,p)	Cd	5.35E-17	5.42E-17	5.31E-17	0.99	1.01
U-238(n,f)	Cd	2.19E-16	2.24E-16	2.17E-16	0.98	1.01
Np-237(n,f)	Cd	3.38E-15	3.05E-15	3.22E-15	1.11	1.05
Co-59(n,γ)		3.86E-14	7.35E-14	4.12E-14	0.53	0.94
Co-59(n,γ)	Cd	2.85E-14	3.46E-14	2.73E-14	0.82	1.04
U-235(n,f)		3.09E-13	6.90E-13	2.82E-13	0.45	1.10
U-235(n,f)	Cd	8.26E-14	1.23E-13	8.84E-14	0.67	0.93

TABLE 6.1-10

DERIVED EXPOSURE RATES FROM THE CAPSULE I DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE BOTTOM

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	8.42E+08	7.42E+08	8%
ϕ(E > 0.1 MeV)	6.62E+09	5.93E+09	16%
ϕ(E < 0.414 eV)	1.58E+09	6.82E+08	23%
dpa/sec	2.38E-12	2.12E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE BOTTOM

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	3.83E-19	4.17E-19	3.87E-19	0.92	0.99
Ti-46(n,p)	Cd	6.49E-18	6.24E-18	6.29E-18	1.04	1.03
Fe-54(n,p)	Cd	3.64E-17	4.05E-17	3.68E-17	0.90	0.99
Ni-58(n,p)	Cd	5.24E-17	5.83E-17	5.25E-17	0.90	1.00
U-238(n,f)	Cd	1.97E-16	2.41E-16	2.10E-16	0.82	0.94
Np-237(n,f)	Cd	3.22E-15	3.27E-15	3.05E-15	0.99	1.06
Co-59(n,γ)		4.53E-14	7.89E-14	4.68E-14	0.57	0.97
Co-59(n,γ)	Cd	2.94E-14	3.71E-14	2.87E-14	0.79	1.02
U-235(n,f)		3.52E-13	7.41E-13	3.36E-13	0.48	1.05
U-235(n,f)	Cd	7.52E-14	1.32E-13	8.09E-14	0.57	0.93

TABLE 6.1-11

DERIVED EXPOSURE RATES FROM THE CAPSULE XX DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - VESSEL SUPPORT

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	8.70E+07	1.25E+08	8%
ϕ(E > 0.1 MeV)	6.83E+08	1.46E+09	17%
ϕ(E < 0.414 eV)	1.63E+08	1.78E+08	27%
dpa/sec	2.45E-13	4.74E-13	12%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - VESSEL SUPPORT

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	3.72E-20	4.31E-20	3.75E-20	0.86	0.99
Ti-46(n,p)	Cd	6.23E-19	6.44E-19	6.06E-19	0.97	1.03
Fe-54(n,p)	Cd	3.76E-18	4.19E-18	3.94E-18	0.90	0.95
Ni-58(n,p)	Cd	5.92E-18	6.02E-18	6.03E-18	0.98	0.98
U-238(n,f)	Cd	3.28E-17	2.49E-17	3.09E-17	1.32	1.06
Np-237(n,f)	Cd	7.58E-16	3.38E-16	6.86E-16	2.24	1.11
Co-59(n,γ)		1.41E-14	8.15E-15	1.45E-14	1.73	0.97
Co-59(n,γ)	Cd	1.01E-14	3.83E-15	9.95E-15	2.64	1.02
U-235(n,f)		1.16E-13	7.65E-14	1.11E-13	1.52	1.05
U-235(n,f)	Cd	5.16E-14	1.36E-14	4.83E-14	3.79	1.07

TABLE 6.1-12

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 15 IRRADIATION

FEET FROM MIDPLANE	NEUTRON FLUX (n/cm^2-sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	4.27E+08	3.45E+08	2.97E+08	2.53E+08
+5.5	1.04E+09	9.53E+08	7.25E+08	5.84E+08
+4.5	1.52E+09	1.45E+09	1.11E+09	9.37E+08
+3.5	1.79E+09	1.61E+09	1.23E+09	1.04E+09
+2.5	1.99E+09	1.72E+09	1.31E+09	1.15E+09
+1.5	1.96E+09	1.71E+09	1.25E+09	1.10E+09
+0.5	1.91E+09	1.72E+09	1.26E+09	1.10E+09
-0.5	1.99E+09	1.84E+09	1.36E+09	1.14E+09
-1.5	1.83E+09	1.78E+09	1.30E+09	1.11E+09
-2.5	1.74E+09	1.77E+09	1.38E+09	1.09E+09
-3.5	1.72E+09	1.65E+09	1.18E+09	1.04E+09
-4.5	1.54E+09	1.51E+09	1.03E+09	1.00E+09
-5.5	1.04E+09	9.68E+08	6.51E+08	6.26E+08
-6.5	3.82E+08	3.70E+08	2.84E+08	2.58E+08

TABLE 6.1-13

FAST NEUTRON FLUX (E > 0.1 MeV) AS A FUNCTION OF AXIAL POSITION WITHIN THE REACTOR CAVITY CYCLE 15 IRRADIATION

FEET FROM MIDPLANE	NEUTRON FLUX (n/cm^2-sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	3.55E+09	3.21E+09	2.74E+09	2.25E+09
+5.5	8.65E+09	8.89E+09	6.69E+09	5.21E+09
+4.5	1.27E+10	1.35E+10	1.03E+10	8.36E+09
+3.5	1.48E+10	1.50E+10	1.13E+10	9.30E+09
+2.5	1.65E+10	1.61E+10	1.21E+10	1.03E+10
+1.5	1.63E+10	1.59E+10	1.15E+10	9.80E+09
+0.5	1.59E+10	1.61E+10	1.16E+10	9.85E+09
-0.5	1.65E+10	1.71E+10	1.26E+10	1.01E+10
-1.5	1.52E+10	1.66E+10	1.20E+10	9.93E+09
-2.5	1.45E+10	1.65E+10	1.27E+10	9.76E+09
-3.5	1.43E+10	1.54E+10	1.09E+10	9.31E+09
-4.5	1.28E+10	1.41E+10	9.53E+09	8.92E+09
-5.5	8.61E+09	9.02E+09	6.02E+09	5.59E+09
-6.5	3.17E+09	3.45E+09	2.62E+09	2.30E+09

TABLE 6.1-14

IRON DISPLACEMENT RATE AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 15 IRRADIATION

FEET FROM MIDPLANE	DISPLACEMENT RATE (dpa/sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	1.25E-12	1.11E-12	9.52E-13	7.87E-13
+5.5	3.05E-12	3.07E-12	2.32E-12	1.82E-12
+4.5	4.47E-12	4.67E-12	3.57E-12	2.92E-12
+3.5	5.24E-12	5.18E-12	3.93E-12	3.25E-12
+2.5	5.84E-12	5.56E-12	4.19E-12	3.59E-12
+1.5	5.76E-12	5.51E-12	3.99E-12	3.42E-12
+0.5	5.60E-12	5.56E-12	4.04E-12	3.44E-12
-0.5	5.84E-12	5.92E-12	4.36E-12	3.54E-12
-1.5	5.36E-12	5.74E-12	4.17E-12	3.47E-12
-2.5	5.12E-12	5.69E-12	4.42E-12	3.41E-12
-3.5	5.04E-12	5.32E-12	3.78E-12	3.25E-12
-4.5	4.51E-12	4.86E-12	3.31E-12	3.12E-12
-5.5	3.04E-12	3.12E-12	2.09E-12	1.95E-12
-6.5	1.12E-12	1.19E-12	9.11E-13	8.04E-13

FIGURE 6.1-1

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 0.0 DEGREF TRAVERSE IN THE REACTOR CAVITY
CYCLE 15 IRRADIATION

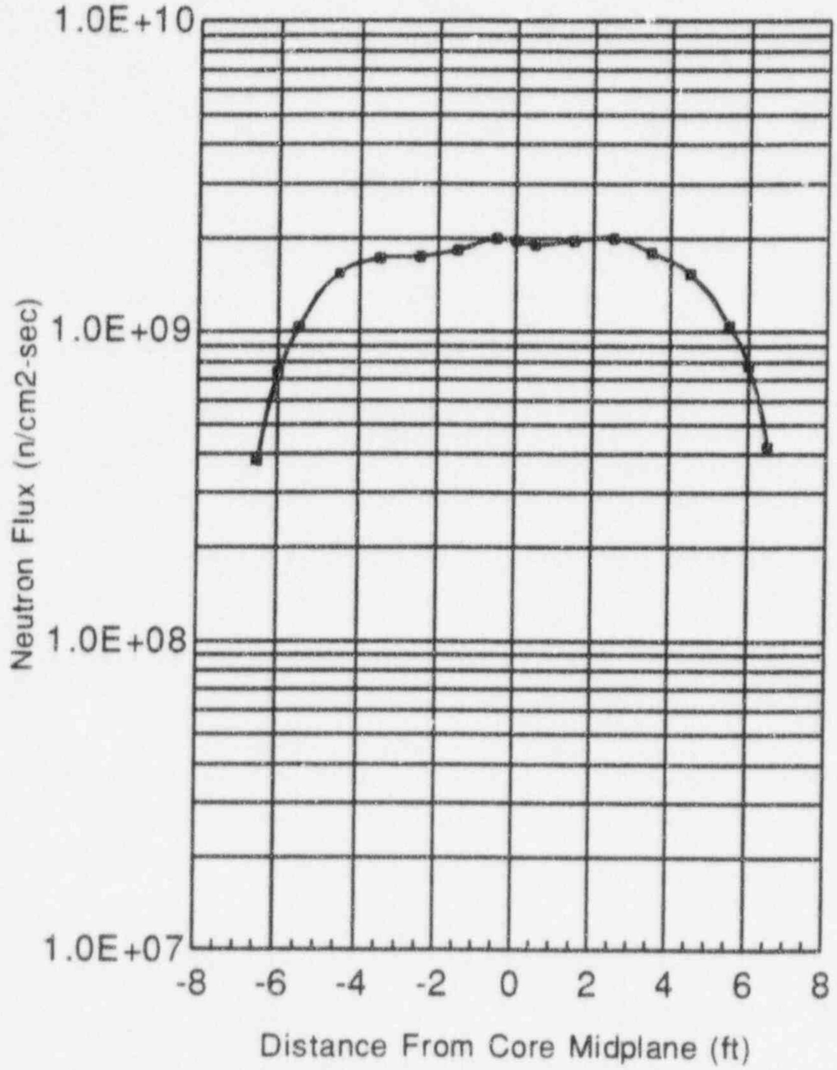


FIGURE 6.1-2

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 15.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 15 IRRADIATION

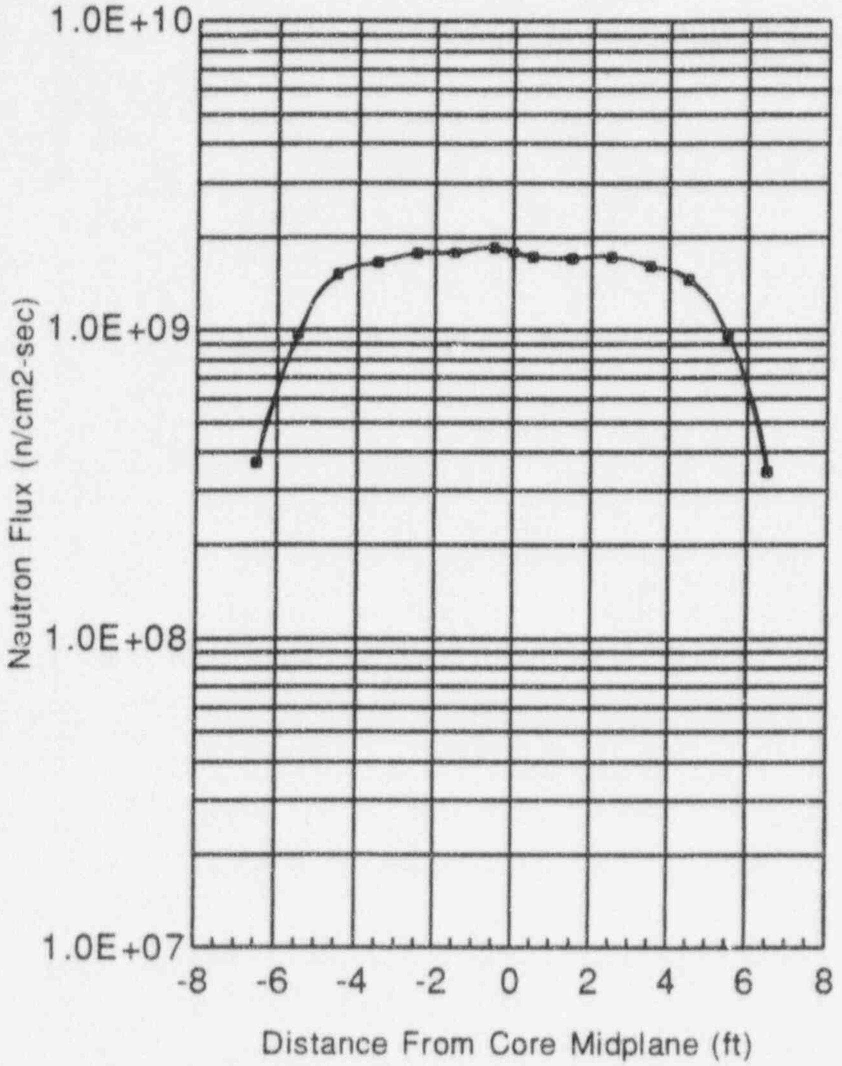


FIGURE 6.1-3

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 30.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 15 IRRADIATION

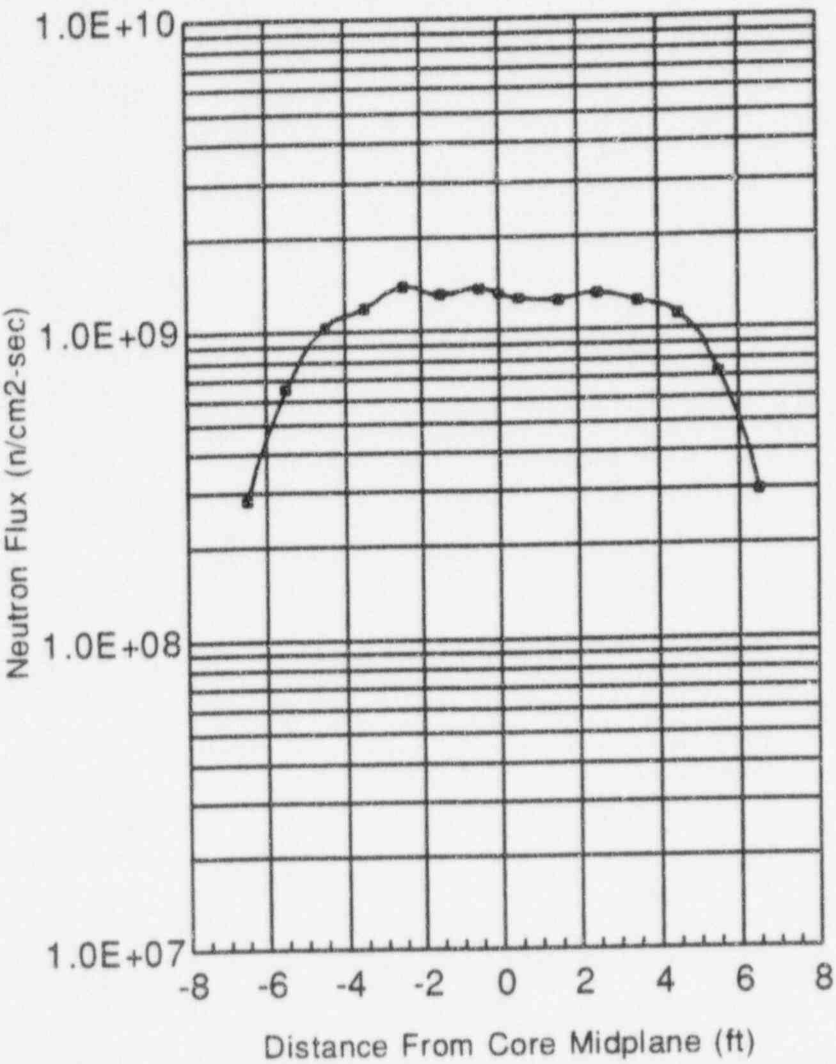
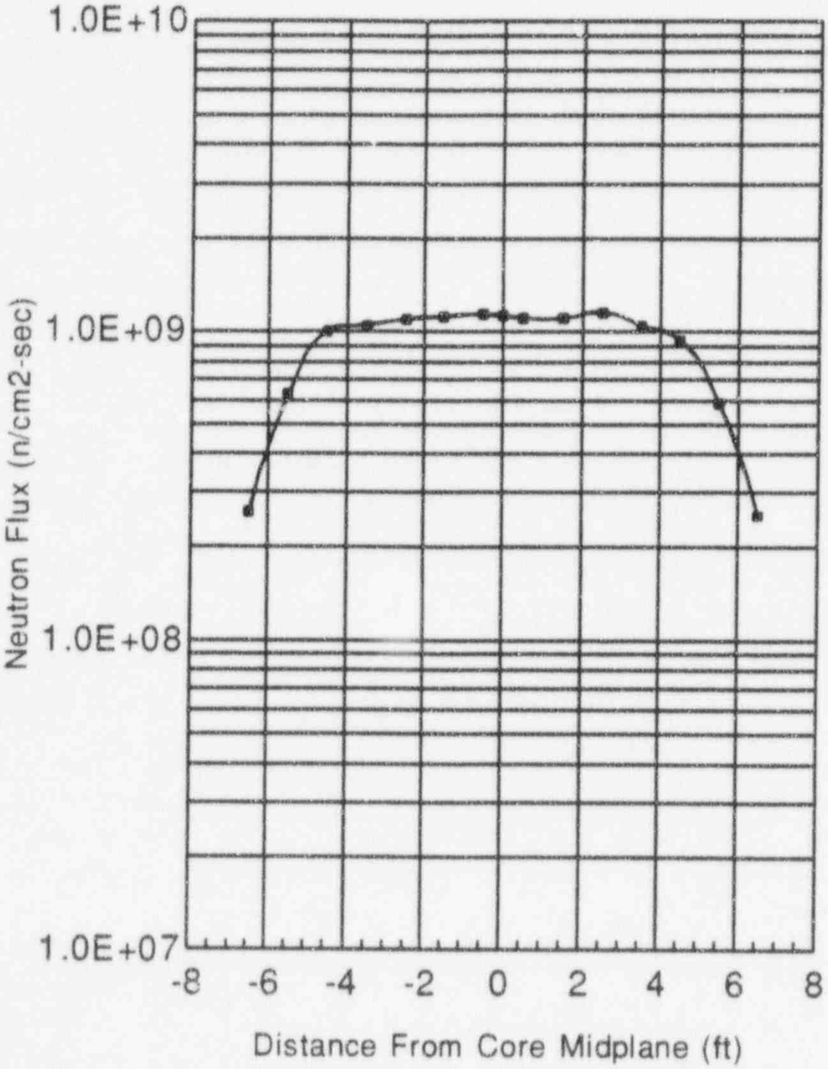


FIGURE 6.1-4

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION ALONG THE 45.0 DEGREE TRAVERSE IN THE REACTOR CAVITY CYCLE 15 IRRADIATION



6.2 - Cycle 16 Results

6.2.1 - Measured Reaction rates

During the Cycle 16 irradiation, six multiple foil sensor sets and four stainless steel gradient chains were deployed in the reactor cavity as depicted in Figures 2.1-1 and 2.1-2. The capsule identifications associated with each of the multiple foil sensor sets were as follows:

	CAPSULE IDENTIFICATION		
AZIMUTH (degrees)	CORE TOP	CORE MIDPLANE	CORE BOTTOM
0.0	M	N	O
15.0		P	
30.0		Q	
45.0		R	

The contents of each of these irradiation capsules are specified in Appendix C to this report.

The irradiation history of the Point Beach Unit 2 reactor during Cycle 16 is also listed in Appendix C. The irradiation history was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" for the applicable operating period. Based on this reactor operating history, the individual sensor characteristics, and the measured specific activities given in Appendix C, cycle average reaction rates referenced to a core power level of 1518 MWt were computed for each multiple foil sensor and gradient chain segment.

The computed reaction rates for the multiple foil sensor sets, including radiometric foils and solid state track recorders, irradiated during Cycle 16 are provided in Table 6.2-1. Corresponding reaction rate data from the four stainless steel gradient chains are recorded in Tables 6.2-2 through 6.2-4 for the Fe-54(n,p), Ni-58(n,p), and Co-59(n,γ) reactions, respectively.

In regard to the data listed in Table 6.2-1, the Fe-54(n,p) reaction rates represent an average of the bare and cadmium

covered measurements for each capsule. Likewise, the U-238(n,f) reaction rates were obtained by averaging the results of the radiometric foil and solid state track recorder data. In addition, the fission rate measurements include corrections for U-235 impurities in the U-238 sensors as well as corrections for photo-fission reactions in both the U-238 and Np-237 sensors.

6.2.2 - Results of the Least Squares Adjustment Procedure

The results of the application of the least squares adjustment procedure to the six sets of multiple foil measurements obtained from the Cycle 16 irradiation are provided in Tables 6.2-5 through 6.2-10. In these tables, the derived exposure experienced at each sensor set location along with data illustrating the fit of both the trial and adjusted spectra to the measurements are given. Also included in the tabulations are the 1σ uncertainties associated with each of the derived exposure rates.

In regard to the comparisons listed in Tables 6.2-5 through 6.2-10, it should be noted that the columns labeled "trial calc" were obtained by normalizing the neutron spectral data from Table 4.1-1 to the absolute calculated neutron flux (E > 1.0 MeV) averaged over the Cycle 16 irradiation period as discussed in Section 3.0. Thus, the comparisons illustrated in Tables 6.2-5 through 6.2-10 indicate the degree to which the calculated neutron energy spectra matched the measured data before and after adjustment. Absolute comparisons are discussed further in Section 7.0 of this report.

Complete traverses of fast neutron exposure rates in the reactor cavity were developed by combining the results of the least squares adjustment of the multiple foil data with the Fe-54(n,p) reaction rate measurements from the gradient chains. The gradient data were employed to establish relative axial distributions over the measurement range and these relative distributions were then normalized to the FERRET results from the midplane sensor sets to produce axial distributions of exposure rates in terms of $\phi(E > 1.0 \text{ MeV})$, $\phi(E > 0.1 \text{ MeV})$, and dpa/sec in the reactor cavity.

The resultant axial distributions of ϕ(E > 1.0 MeV), ϕ(E > 0.1 MeV), and dpa/sec from the gradient chain measurements are given in Tables 6.2-11, 6.2-12, and 6.2-13, respectively. The distributions of ϕ(E > 1.0 MeV) are depicted graphically in Figures 6.2-1 through 6.2-4. In these graphical presentations, results for axial locations of -6.0, 0.0, and +6.0 feet relative to the core midplane represent the explicit results of the FERRET evaluations summarized in Tables 6.2-5 through 6.2-10, while results at the remaining axial locations depict the normalized data from the gradient chains.

TABLE 6.2-1

SUMMARY OF REACTION RATES DERIVED FROM MULTIPLE FOIL SENSOR SETS
CYCLE 16 IRRADIATION

		REACTION RATE (rps/nucleus)					
		CAPSULE N	CAPSULE P	CAPSULE Q	CAPSULE R	CAPSULE M	CAPSULE O
Cu-63(n,α)	Cd	8.59E-19	7.50E-19	6.71E-19	6.61E-19	4.11E-19	3.69E-19
Ti-46(n,p)	Cd	1.30E-17	1.12E-17	9.81E-18	9.36E-18	6.53E-18	5.90E-18
Fe-54(n,p)	Cd	7.47E-17	6.35E-17	5.53E-17	5.25E-17	3.52E-17	3.50E-17
Ni-58(n,p)	Cd	1.07E-16	9.25E-17	7.85E-17	7.44E-17	5.52E-17	5.08E-17
U-238(n,f)	Cd	4.24E-16	3.94E-16	2.87E-16	2.90E-17	2.25E-16	1.89E-16
Np-237(n,f)	Cd	6.45E-15	5.66E-15	4.59E-15	4.47E-15	3.18E-15	3.14E-15
Co-59(n,γ)		8.74E-14	1.12E-13	1.00E-13	6.23E-14	3.68E-14	4.08E-14
Co-59(n,γ)	Cd	5.26E-14	6.71E-14	5.44E-14	4.14E-14	2.63E-14	2.65E-14
U-235(n,f)		8.27E-13	1.02E-12	9.98E-13	4.41E-13		3.66E-13
U-235(n,f)	Cd	2.66E-13	3.09E-13	2.32E-13	1.95E-13	7.72E-14	7.27E-14

Note: Cd indicates that the sensor was cadmium covered.

TABLE 6.2-2

Fe-54(n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS- CYCLE 16 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	1.93E-17	1.42E-17	1.19E-17	1.08E-17
+5.5	4.68E-17	3.52E-17	2.99E-17	2.57E-17
+4.5	7.39E-17	5.65E-17	4.34E-17	3.99E-17
+3.5	8.24E-17	6.76E-17	4.54E-17	4.19E-17
+2.5	8.77E-17	6.81E-17	5.01E-17	4.71E-17
+1.5	8.34E-17	6.44E-17	4.81E-17	4.67E-17
+0.5	7.50E-17	5.86E-17	5.04E-17	4.64E-17
-0.5	6.02E-17	5.23E-17	4.85E-17	4.57E-17
-1.5	5.91E-17	5.27E-17	4.71E-17	4.60E-17
-2.5	6.65E-17	5.15E-17	4.82E-17	4.40E-17
-3.5	6.92E-17	6.13E-17	4.37E-17	4.30E-17
-4.5	6.87E-17	5.86E-17	4.07E-17	4.06E-17
-5.5	4.66E-17	3.67E-17	2.69E-17	2.70E-17
-6.5	1.77E-17	1.57E-17	1.13E-17	1.07E-17

TABLE 6.2-3

Ni-58(n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLE 16 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	2.91E-17	2.11E-17	1.76E-17	1.54E-17
+5.5	7.10E-17	5.54E-17	4.33E-17	3.65E-17
+4.5	1.11E-16	8.63E-17	6.13E-17	5.75E-17
+3.5	1.18E-16	9.14E-17	6.66E-17	6.16E-17
+2.5	1.24E-16	9.52E-17	7.34E-17	6.60E-17
+1.5	1.15E-16	8.93E-17	7.04E-17	6.34E-17
+0.5	1.04E-16	8.43E-17	7.07E-17	6.34E-17
-0.5	8.69E-17	7.34E-17	7.10E-17	6.69E-17
-1.5	8.37E-17	7.60E-17	6.69E-17	6.28E-17
-2.5	9.19E-17	7.66E-17	6.75E-17	6.45E-17
-3.5	1.05E-16	8.43E-17	6.45E-17	6.13E-17
-4.5	9.81E-17	8.52E-17	6.01E-17	5.98E-17
-5.5	6.98E-17	5.42E-17	3.89E-17	3.98E-17
-6.5	2.87E-17	2.17E-17	1.60E-17	1.70E-17

TABLE 6.2-4

Co-59(n,γ) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLE 16 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	2.25E-14	2.25E-14	1.95E-14	1.67E-14
+5.5	5.14E-14	7.40E-14	5.53E-14	3.38E-14
+4.5	6.88E-14	1.06E-13	7.82E-14	4.68E-14
+3.5	7.98E-14	1.19E-13	8.97E-14	5.48E-14
+2.5	8.66E-14	1.25E-13	9.70E-14	5.89E-14
+1.5	8.40E-14	1.19E-13	9.59E-14	6.26E-14
+0.5	8.50E-14	1.13E-13	9.80E-14	6.21E-14
-0.5	8.34E-14	1.04E-13	9.54E-14	6.00E-14
-1.5	8.03E-14	9.96E-14	9.02E-14	6.05E-14
-2.5	7.87E-14	9.70E-14	8.97E-14	5.58E-14
-3.5	8.08E-14	9.91E-14	8.24E-14	5.27E-14
-4.5	6.62E-14	8.66E-14	6.47E-14	4.35E-14
-5.5	4.73E-14	6.36E-14	3.66E-14	3.24E-14
-6.5	3.48E-14	3.40E-14	2.59E-14	2.41E-14

TABLE 6.2-5

DERIVED EXPOSURE RATES FROM THE CAPSULE N DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.80E+09	1.53E+09	8%
ϕ(E > 0.1 MeV)	1.41E+10	1.31E+10	17%
ϕ(E < 0.414 eV)	3.38E+09	1.52E+09	25%
dpa/sec	5.08E-12	4.59E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	8.59E-19	8.91E-19	8.56E-19	0.96	1.00
Ti-46(n,p)	Cd	1.30E-17	1.33E-17	1.28E-17	0.98	1.02
Fe-54(n,p)	Cd	7.47E-17	8.66E-17	7.57E-17	0.86	0.99
Ni-58(n,p)	Cd	1.07E-16	1.24E-16	1.08E-16	0.86	0.99
U-238(n,f)	Cd	4.24E-16	5.15E-16	4.35E-16	0.82	0.98
Np-237(n,f)	Cd	6.45E-15	6.99E-15	6.28E-15	0.92	1.03
Co-59(n,γ)		8.74E-14	1.69E-13	8.97E-14	0.52	0.97
Co-59(n,γ)	Cd	5.26E-14	7.93E-14	5.22E-14	0.66	1.01
U-235(n,f)		8.27E-13	1.58E-12	7.95E-13	0.52	1.03
U-235(n,f)	Cd	2.66E-13	2.81E-13	2.61E-13	0.95	1.04

TABLE 6.2-6

DERIVED EXPOSURE RATES FROM THE CAPSULE P DOSIMETRY EVALUATION
15.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.54E+09	1.35E+09	8%
ϕ(E > 0.1 MeV)	1.33E+10	1.24E+10	17%
ϕ(E < 0.414 eV)	3.07E+09	1.93E+09	25%
dpa/sec	4.67E-12	4.29E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
15.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	7.50E-19	7.42E-19	7.45E-19	1.01	1.01
Ti-46(n,p)	Cd	1.12E-17	1.10E-17	1.10E-17	1.02	1.02
Fe-54(n,p)	Cd	6.35E-17	7.20E-17	6.50E-17	0.88	0.98
Ni-58(n,p)	Cd	9.25E-17	1.04E-16	9.32E-17	0.89	0.99
U-238(n,f)	Cd	3.94E-16	4.34E-16	3.84E-16	0.91	1.03
Np-237(n,f)	Cd	5.66E-15	6.28E-15	5.65E-15	0.90	1.00
Co-59(n,γ)		1.12E-13	1.61E-13	1.14E-13	0.70	0.98
Co-59(n,γ)	Cd	6.71E-14	8.09E-14	6.66E-14	0.83	1.01
U-235(n,f)		1.02E-12	1.45E-12	9.84E-13	0.70	1.04
U-235(n,f)	Cd	3.09E-13	2.85E-13	3.03E-13	1.08	1.02

TABLE 6.2-7

DERIVED EXPOSURE RATES FROM THE CAPSULE Q DOSIMETRY EVALUATION
30.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
$\phi(E > 1.0$ MeV)	1.21E+09	1.06E+09	8%
$\phi(E > 0.1$ MeV)	1.05E+10	9.77E+09	17%
$\phi(E < 0.414$ eV)	2.57E+09	1.93E+09	23%
dpa/sec	3.71E-12	3.40E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
30.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	6.71E-19	6.25E-19	6.68E-19	1.07	1.00
Ti-46(n,p)	Cd	9.81E-18	9.20E-18	9.68E-18	1.07	1.01
Fe-54(n,p)	Cd	5.53E-17	5.84E-17	5.56E-17	0.95	1.00
Ni-58(n,p)	Cd	7.85E-17	8.39E-17	7.87E-17	0.94	1.00
U-238(n,f)	Cd	2.87E-16	3.44E-16	3.04E-16	0.83	0.94
Np-237(n,f)	Cd	4.59E-15	4.94E-15	4.48E-15	0.93	1.03
Co-59(n,γ)		1.00E-13	1.35E-13	1.03E-13	0.74	0.97
Co-59(n,γ)	Cd	5.44E-14	6.79E-14	5.38E-14	0.80	1.01
U-235(n,f)		9.98E-13	1.21E-12	9.41E-13	0.83	1.06
U-235(n,f)	Cd	2.32E-13	2.38E-13	2.32E-13	0.98	1.00

TABLE 6.2-8

DERIVED EXPOSURE RATES FROM THE CAPSULE R DOSIMETRY EVALUATION
45.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
φ(E > 1.0 MeV)	1.06E+09	1.03E+09	8%
φ(E > 0.1 MeV)	8.70E+09	9.06E+09	17%
φ(E < 0.414 eV)	2.49E+09	8.11E+08	27%
dpa/sec	3.11E-12	3.17E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
45.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	6.61E-19	6.05E-19	6.54E-19	1.09	1.01
Ti-46(n,p)	Cd	9.36E-18	8.77E-18	9.27E-18	1.07	1.01
Fe-54(n,p)	Cd	5.25E-17	5.40E-17	5.32E-17	0.97	0.99
Ni-58(n,p)	Cd	7.44E-17	7.69E-17	7.50E-17	0.97	0.99
U-238(n,f)	Cd	2.90E-16	3.07E-16	2.95E-16	0.95	0.98
Np-237(n,f)	Cd	4.48E-15	4.20E-15	4.32E-15	1.07	1.04
Co-59(n,γ)		6.23E-14	1.22E-13	6.19E-14	0.51	1.01
Co-59(n,γ)	Cd	4.14E-14	5.62E-14	4.18E-14	0.74	0.99
U-235(n,f)		4.41E-13	1.16E-12	4.62E-13	0.38	0.96
U-235(n,f)	Cd	1.95E-13	1.98E-13	1.87E-13	0.99	1.04

TABLE 6.2-9

DERIVED EXPOSURE RATES FROM THE CAPSULE M DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE TOP

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	8.15E+08	7.70E+08	8%
ϕ(E > 0.1 MeV)	6.40E+09	6.07E+09	16%
ϕ(E < 0.414 eV)	1.53E+09	4.41E+08	30%
dpa/sec	2.30E-12	2.17E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE TOP

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	4.11E-19	4.03E-19	4.10E-19	1.02	1.00
Ti-46(n,p)	Cd	6.53E-18	6.03E-18	6.36E-18	1.08	1.03
Fe-54(n,p)	Cd	3.52E-17	3.92E-17	3.69E-17	0.90	0.95
Ni-58(n,p)	Cd	5.52E-17	5.63E-17	5.47E-17	0.98	1.01
U-238(n,f)	Cd	2.25E-16	2.33E-16	2.21E-16	0.97	1.02
Np-237(n,f)	Cd	3.18E-15	3.16E-15	3.08E-15	1.01	1.03
Co-59(n,γ)		3.68E-14	7.63E-14	3.73E-14	0.48	0.99
Co-59(n,γ)	Cd	2.63E-14	3.59E-14	2.60E-14	0.73	1.01
U-235(n,f)	Cd	7.72E-14	1.27E-13	8.09E-14	0.61	0.95

TABLE 6.2-10

DERIVED EXPOSURE RATES FROM THE CAPSULE O DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE BOTTOM

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	8.10E+08	7.21E+08	8%
ϕ(E > 0.1 MeV)	6.36E+09	5.81E+09	16%
ϕ(E < 0.414 eV)	1.52E+09	6.70E+08	23%
dpa/sec	2.29E-12	2.06E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE BOTTOM

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	3.69E-19	4.01E-19	3.71E-19	0.92	1.00
TI-46(n,p)	Cd	5.90E-18	6.00E-18	5.79E-18	0.98	1.02
Fe-54(n,p)	Cd	3.50E-17	3.90E-17	3.52E-17	0.90	0.99
Ni-58(n,p)	Cd	5.08E-17	5.60E-17	5.07E-17	0.91	1.00
U-238(n,f)	Cd	1.89E-16	2.32E-16	2.03E-16	0.82	0.93
Np-237(n,f)	Cd	3.14E-15	3.15E-15	2.97E-15	1.00	1.06
Co-59(n,γ)		4.08E-14	7.59E-14	4.33E-14	0.54	0.94
Co-59(n,γ)	Cd	2.65E-14	3.57E-14	2.55E-14	0.74	1.04
U-235(n,f)		3.66E-13	7.12E-13	3.32E-13	0.51	1.10
U-235(n,f)	Cd	7.27E-14	1.27E-13	7.86E-14	0.57	0.93

TABLE 6.2-11

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION WITHIN THE REACTOR CAVITY CYCLE 16 IRRADIATION

FEET FROM MIDPLANE	NEUTRON FLUX (n/cm^2-sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	4.36E+08	3.44E+08	2.55E+08	2.41E+08
+5.5	1.06E+09	8.56E+08	6.41E+08	5.74E+08
+4.5	1.67E+09	1.38E+09	9.29E+08	8.92E+08
+3.5	1.86E+09	1.65E+09	9.72E+08	9.38E+08
+2.5	1.98E+09	1.66E+09	1.07E+09	1.05E+09
+1.5	1.89E+09	1.57E+09	1.03E+09	1.04E+09
+0.5	1.70E+09	1.43E+09	1.08E+09	1.04E+09
-0.5	1.36E+09	1.27E+09	1.04E+09	1.02E+09
-1.5	1.34E+09	1.28E+09	1.01E+09	1.03E+09
-2.5	1.51E+09	1.25E+09	1.03E+09	9.86E+08
-3.5	1.57E+09	1.49E+09	9.36E+08	9.63E+08
-4.5	1.55E+09	1.43E+09	8.71E+08	9.08E+08
-5.5	1.06E+09	8.94E+08	5.77E+08	6.05E+08
-6.5	4.02E+08	3.82E+08	2.41E+08	2.40E+08

TABLE 6.2-12

FAST NEUTRON FLUX (E > 0.1 MeV) AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 16 IRRADIATION

FEET FROM MIDPLANE	NEUTRON FLUX (n/cm^2-sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	3.73E+08	3.16E+08	2.35E+09	2.12E+09
+5.5	9.08E+09	7.86E+09	5.90E+09	5.05E+09
+4.5	1.43E+10	1.26E+10	8.56E+09	7.85E+09
+3.5	1.60E+10	1.51E+10	8.96E+09	8.25E+09
+2.5	1.70E+10	1.52E+10	9.90E+09	9.26E+09
+1.5	1.62E+10	1.44E+10	9.50E+09	9.19E+09
+0.5	1.45E+10	1.31E+10	9.95E+09	9.13E+09
-0.5	1.17E+10	1.17E+10	9.59E+09	8.99E+09
-1.5	1.15E+10	1.18E+10	9.29E+09	9.05E+09
-2.5	1.29E+10	1.15E+10	9.52E+09	8.67E+09
-3.5	1.34E+10	1.37E+10	8.63E+09	8.47E+09
-4.5	1.33E+10	1.31E+10	8.03E+09	7.99E+09
-5.5	9.04E+09	8.21E+09	5.32E+09	5.32E+09
-6.5	3.44E+09	3.51E+09	2.22E+09	2.11E+09

TABLE 6.2-13

IRON DISPLACEMENT RATE AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 16 IRRADIATION

FEET FROM MIDPLANE	DISPLACEMENT RATE (dpa/sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	1.31E-12	1.09E-12	8.17E-13	7.42E-13
+5.5	3.18E-12	2.72E-12	2.05E-12	1.77E-12
+4.5	5.02E-12	4.37E-12	2.98E-12	2.75E-12
+3.5	5.59E-12	5.23E-12	3.12E-12	2.89E-12
+2.5	5.95E-12	5.27E-12	3.45E-12	3.24E-12
+1.5	5.67E-12	4.98E-12	3.31E-12	3.22E-12
+0.5	5.09E-12	4.54E-12	3.46E-12	3.19E-12
-0.5	4.09E-12	4.04E-12	3.34E-12	3.15E-12
-1.5	4.02E-12	4.07E-12	3.23E-12	3.17E-12
-2.5	4.52E-12	3.98E-12	3.31E-12	3.03E-12
-3.5	4.70E-12	4.74E-12	3.00E-12	2.96E-12
-4.5	4.66E-12	4.54E-12	2.80E-12	2.79E-12
-5.5	3.17E-12	2.84E-12	1.85E-12	1.86E-12
-6.5	1.20E-12	1.21E-12	7.73E-13	7.38E-13

FIGURE 6.2-1

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 0.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 16 IRRADIATION

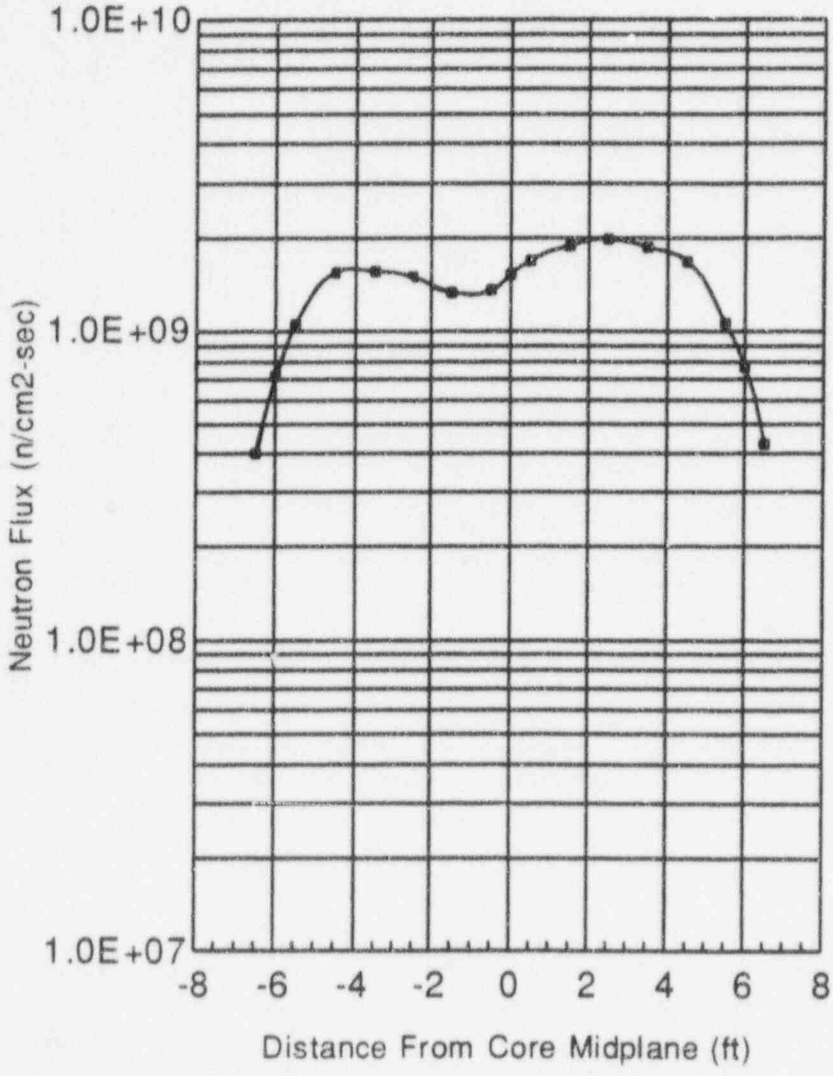


FIGURE 6.2-2

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 15.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 16 IRRADIATION

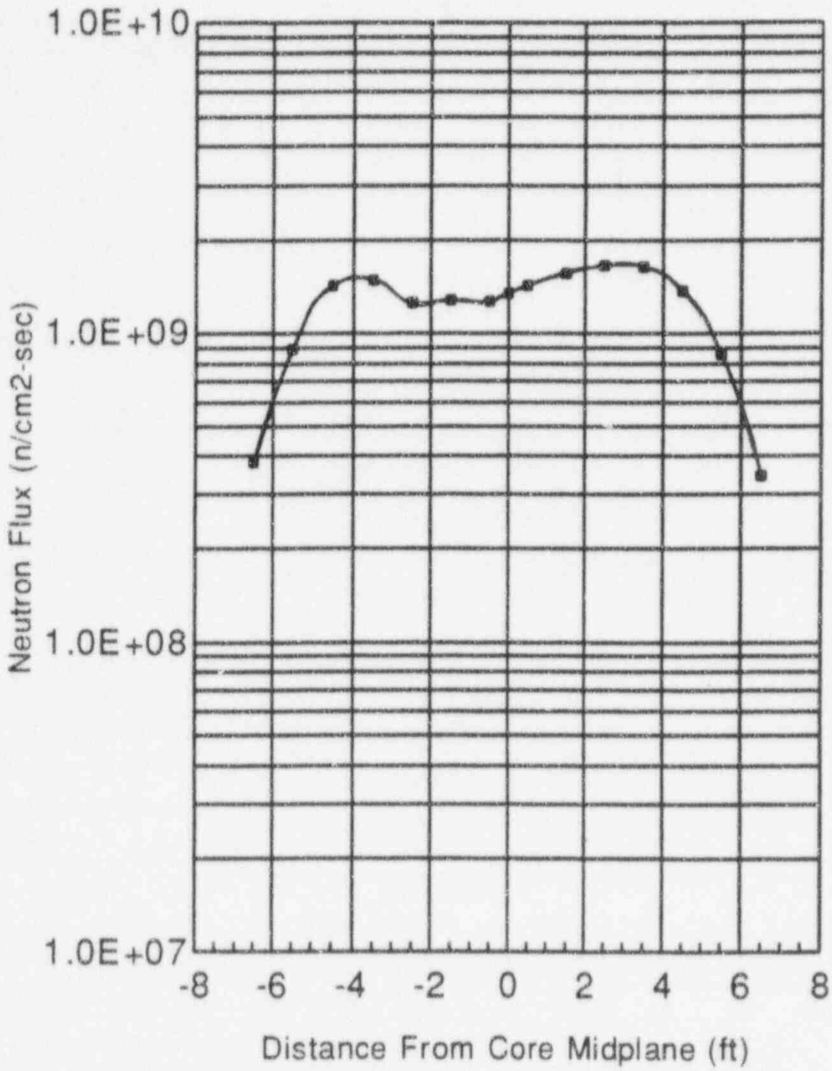


FIGURE 6.2-3

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 30.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 16 IRRADIATION

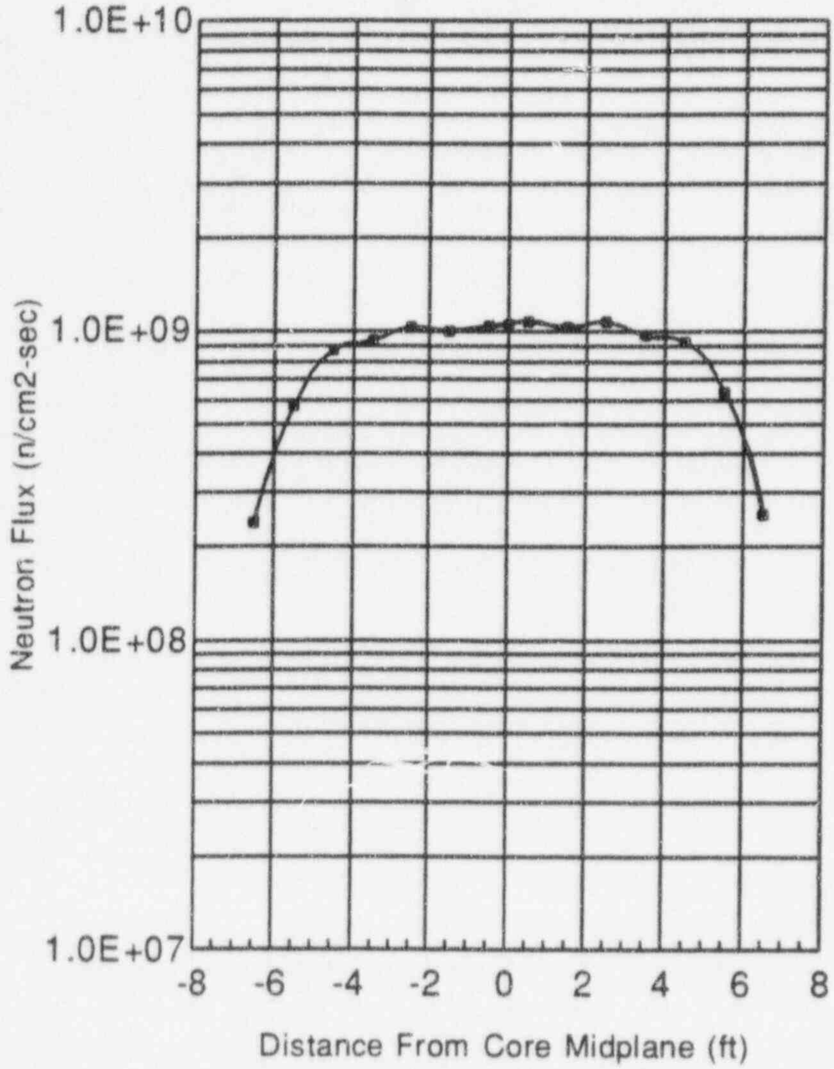
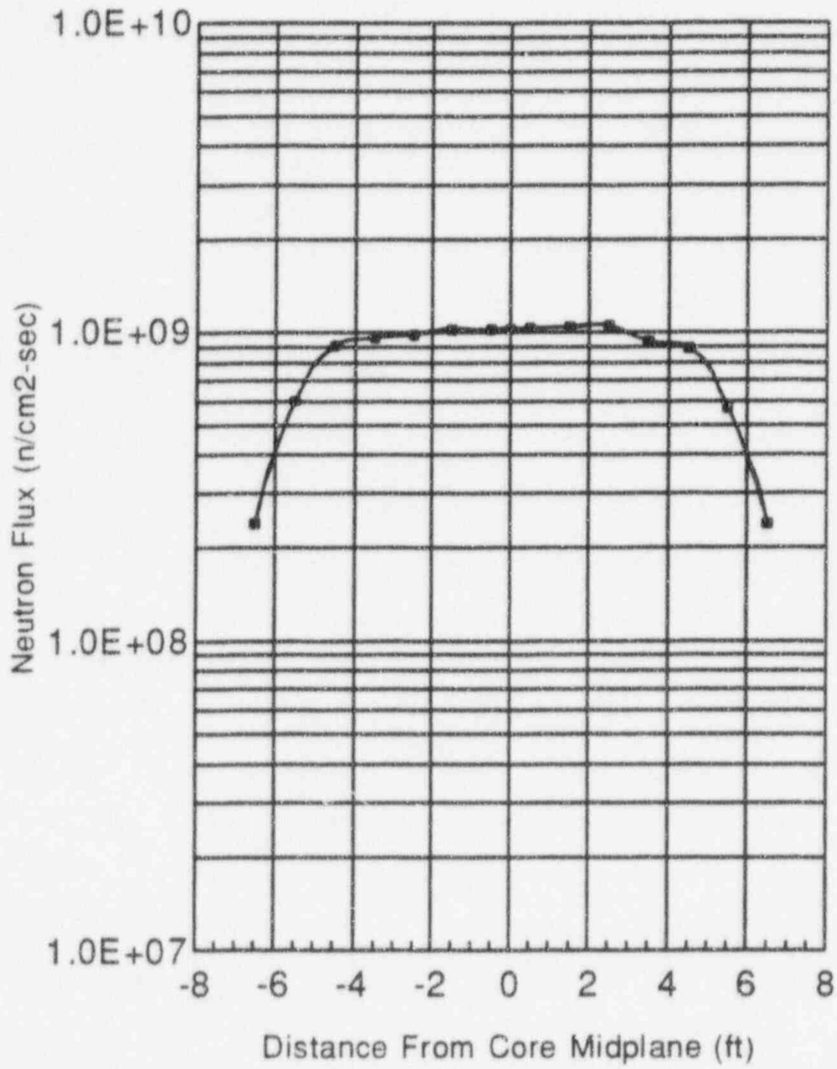


FIGURE 6.2-4

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 45.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 16 IRRADIATION



6.3 - Cycle 17 Results

6.3.1 - Measured Reaction rates

During the Cycle 17 irradiation, six multiple foil sensor sets and four stainless steel gradient chains were deployed in the reactor cavity as depicted in Figures 2.1-1 and 2.1-2. The capsule identifications associated with each of the multiple foil sensor sets were as follows:

	CAPSULE IDENTIFICATION		
AZIMUTH (degrees)	CORE TOP	CORE MIDPLANE	CORE BOTTOM
0.0	AA	BB	CC
15.0		DD	
30.0		EE	
45.0		FF	

The contents of each of these irradiation capsules are specified in Appendix D to this report.

The irradiation history of the Point Beach Unit 2 reactor during Cycle 17 is also listed in Appendix D. The irradiation history was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" for the applicable operating period. Based on this reactor operating history, the individual sensor characteristics, and the measured specific activities given in Appendix D, cycle average reaction rates referenced to a core power level of 1518 MWt were computed for each multiple foil sensor and gradient chain segment.

The computed reaction rates for the multiple foil sensor sets, including radiometric foils and solid state track recorders, irradiated during Cycle 17 are provided in Table 6.3-1. Corresponding reaction rate data from the four stainless steel gradient chains are recorded in Tables 6.3-2 through 6.3-4 for the Fe-54(n,p), Ni-58(n,p), and Co-59(n,γ) reactions, respectively.

In regard to the data listed in Table 6.3-1, the Fe-54(n,p) reaction rates represent an average of the bare and cadmium

covered measurements for each capsule. Likewise, the U-238(n,f) reaction rates were obtained by averaging the results of the radiometric foil and solid state track recorder data. In addition, the fission rate measurements include corrections for U-235 impurities in the U-238 sensors as well as corrections for photo-fission reactions in both the U-238 and Np-237 sensors.

6.3.2 - Results of the Least Squares Adjustment Procedure

The results of the application of the least squares adjustment procedure to the six sets of multiple foil measurements obtained from the Cycle 17 irradiation are provided in Tables 6.3-5 through 6.3-10. In these tables, the derived exposure experienced at each sensor set location along with data illustrating the fit of both the trial and adjusted spectra to the measurements are given. Also included in the tabulations are the 1σ uncertainties associated with each of the derived exposure rates.

In regard to the comparisons listed in Tables 6.3-5 through 6.3-10, it should be noted that the columns labeled "trial calc" were obtained by normalizing the neutron spectral data from Table 4.1-1 to the absolute calculated neutron flux (E > 1.0 MeV) averaged over the Cycle 17 irradiation period as discussed in Section 3.0. Thus, the comparisons illustrated in Tables 6.3-5 through 6.3-10 indicate the degree to which the calculated neutron energy spectra matched the measured data before and after adjustment. Absolute comparisons are discussed further in Section 7.0 of this report.

Complete traverses of fast neutron exposure rates in the reactor cavity were developed by combining the results of the least squares adjustment of the multiple foil data with the Fe-54(n,p) reaction rate measurements from the gradient chains. The gradient data were employed to establish relative axial distributions over the measurement range and these relative distributions were then normalized to the FERRET results from the midplane sensor sets to produce axial distributions of exposure rates in terms of ϕ(E > 1.0 MeV), ϕ(E > 0.1 MeV), and dpa/sec in the reactor cavity.

The resultant axial distributions of ϕ(E > 1.0 MeV), ϕ(E > 0.1 MeV), and dpa/sec from the gradient chain measurements are given in Tables 6.3-11, 6.3-12, and 6.3-13, respectively. The distributions of ϕ(E > 1.0 MeV) are depicted graphically in Figures 6.3-1 through 6.3-4. In these graphical presentations, results for axial locations of -6.0, 0.0, and +6.0 feet relative to the core midplane represent the explicit results of the FERRET evaluations summarized in Tables 6.3-5 through 6.3-10, while results at the remaining axial locations depict the normalized data from the gradient chains.

TABLE 6.3-1

SUMMARY OF REACTION RATES DERIVED FROM MULTIPLE FOIL SENSOR SETS
CYCLES 17 IRRADIATION

		REACTION RATE (rps/nucleus)					
		CAPSULE BB	CAPSULE DD	CAPSULE EE	CAPSULE FF	CAPSULE AA	CAPSULE CC
Cu-63(n,α)	Cd	9.04E-19	8.01E-19	6.79E-19	6.97E-19	4.23E-19	3.92E-19
Ti-46(n,p)	Cd	1.34E-17	1.18E-17	1.02E-17	1.00E-17	6.68E-18	6.19E-18
Fe-54(n,p)	Cd	7.82E-17	6.82E-17	5.66E-17	5.43E-17	3.57E-17	3.63E-17
Ni-58 (n,p)	Cd	1.10E-16	9.61E-17	8.01E-17	7.65E-17	5.67E-17	5.23E-17
U-238(n,f)	Cd	3.98E-16	3.93E-16	3.08E-16	2.78E-16	2.26E-16	1.85E-16
Np-237(n,f)	Cd	6.67E-15	5.39E-15	4.70E-15	3.90E-15		2.51E-15
Co-59(n,γ)		9.31E-14	1.17E-13	1.05E-13	6.50E-14	3.89E-14	4.25E-14
Co-59(n,γ)	Cd	5.52E-14	6.68E-14	5.76E-14	4.40E-14	2.79E-14	2.79E-14
U-235(n,f)		7.83E-13	1.10E-12	1.02E-12	4.52E-13	3.32E-13	3.95E-13
U-235(n,f)	Cd	2.63E-13	2.98E-13	2.66E-13	1.97E-13	9.87E-14	7.12E-14

Note: Cd indicates that the sensor was cadmium covered.

TABLE 6.3-2

Fe-54(n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLE 17 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	2.03E-17	1.52E-17	1.17E-17	1.09E-17
+5.5	5.05E-17	3.86E-17	2.97E-17	2.53E-17
+4.5	7.51E-17	5.25E-17	4.50E-17	3.95E-17
+3.5	8.39E-17	6.33E-17	5.30E-17	4.55E-17
+2.5	8.83E-17	7.02E-17	4.96E-17	5.05E-17
+1.5	8.84E-17	6.82E-17	5.05E-17	4.71E-17
+0.5	7.85E-17	6.43E-17	5.40E-17	4.59E-17
-0.5	6.77E-17	6.13E-17	5.35E-17	5.30E-17
-1.5	6.67E-17	5.54E-17	5.10E-17	4.91E-17
-2.5	6.57E-17	6.23E-17	5.20E-17	5.15E-17
-3.5	7.26E-17	6.13E-17	4.87E-17	4.74E-17
-4.5	7.06E-17	6.13E-17	4.62E-17	4.19E-17
-5.5	5.00E-17	4.07E-17	2.80E-17	2.91E-17
-6.5	2.08E-17	1.49E-17	1.13E-17	1.28E-17

TABLE 6.3-3

Ni-58(n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLES 17 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	3.14E-17	2.33E-17	1.87E-17	1.64E-17
+5.5	7.50E-17	5.99E-17	4.23E-17	3.68E-17
+4.5	1.14E-16	9.06E-17	6.49E-17	5.83E-17
+3.5	1.19E-16	9.49E-17	7.20E-17	6.53E-17
+2.5	1.28E-16	1.03E-16	7.57E-17	7.46E-17
+1.5	1.22E-16	9.39E-17	7.89E-17	7.12E-17
+0.5	1.09E-16	9.00E-17	7.30E-17	6.70E-17
-0.5	9.41E-16	8.29E-17	7.95E-17	7.24E-17
-1.5	9.10E-16	7.99E-17	7.48E-17	7.32E-17
-2.5	1.03E-16	8.35E-17	7.69E-17	7.10E-17
-3.5	1.05E-16	9.12E-17	7.06E-17	6.84E-17
-4.5	1.08E-16	8.44E-17	6.72E-17	6.05E-17
-5.5	7.22E-17	5.89E-17	4.43E-17	2.06E-17
-6.5	2.95E-17	2.35E-17	1.79E-17	1.86E-17

TABLE 6.3-4

Co-59(n,γ) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLE 17 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	2.11E-14	2.19E-14	1.81E-14	1.55E-14
+5.5	3.93E-14	7.05E-14	5.06E-14	3.13E-14
+4.5	5.32E-14	9.92E-14	7.31E-14	4.20E-14
+3.5	6.16E-14	1.12E-13	8.56E-14	5.03E-14
+2.5	6.68E-14	1.18E-13	9.19E-14	5.58E-14
+1.5	6.63E-14	1.15E-13	9.45E-14	5.90E-14
+0.5	6.68E-14	1.07E-13	9.45E-14	5.90E-14
-0.5	6.63E-14	8.45E-14	7.62E-14	4.92E-14
-1.5	6.42E-14	7.93E-14	7.41E-14	4.83E-14
-2.5	6.47E-14	7.88E-14	7.15E-14	4.62E-14
-3.5	6.37E-14	7.72E-14	6.58E-14	4.31E-14
-4.5	5.32E-14	6.68E-14	5.32E-14	3.51E-14
-5.5	3.64E-14	4.92E-14	2.88E-14	1.13E-14
-6.5	3.39E-14	2.63E-14	2.03E-14	1.93E-14

TABLE 6.3-5

DERIVED EXPOSURE RATES FROM THE CAPSULE BB DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.81E+09	1.52E+09	8%
ϕ(E > 0.1 MeV)	1.42E+10	1.32E+10	17%
ϕ(E < 0.414 eV)	3.40E+09	1.54E+09	15%
dpa/sec	5.11E-12	4.62E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	C/M TRIAL	C/M ADJUSTED
Cu-63(n,α)	Cd	9.04E-19	8.96E-19	9.00E-19	1.01	1.00
Ti-46(n,p)	Cd	1.34E-17	1.34E-17	1.32E-17	1.00	1.02
Fe-54(n,p)	Cd	7.82E-17	8.71E-17	7.83E-17	0.90	1.00
Ni-58(n,p)	Cd	1.10E-16	1.25E-16	1.11E-16	0.88	0.99
U-238(n,f)	Cd	3.97E-16	5.17E-16	4.33E-16	0.77	0.92
Np-237(n,f)	Cd	6.67E-15	7.03E-15	6.38E-15	0.95	1.05
Co-59(n,γ)		9.31E-14	1.70E-13	9.34E-14	0.55	1.00
Co-59(n,γ)	Cd	5.52E-14	7.97E-14	5.54E-14	0.69	1.00
U-235(n,f)		7.83E-13	1.59E-12	7.93E-13	0.49	0.99
U-235(n,f)	Cd	2.63E-13	2.83E-13	2.56E-13	0.93	1.03

TABLE 6.3-6

DERIVED EXPOSURE RATES FROM THE CAPSULE DD DOSIMETRY EVALUATION
15.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.55E+09	1.34E+09	8%
ϕ(E > 0.1 MeV)	1.34E+10	1.18E+10	17%
ϕ(E < 0.414 eV)	3.09E+09	2.13E+09	24%
dpa/sec	4.70E-12	4.15E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
15.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	8.01E-19	7.47E-19	7.96E-19	1.07	1.01
Ti-46(n,p)	Cd	1.18E-17	1.11E-17	1.17E-17	1.06	1.01
Fe-54(n,p)	Cd	6.82E-17	7.24E-17	6.89E-17	0.94	0.99
Ni-58(n,p)	Cd	9.61E-17	1.05E-16	9.71E-17	0.92	0.99
U-238(n,f)	Cd	3.93E-16	4.37E-16	3.90E-16	0.90	1.01
Np-237(n,f)	Cd	5.39E-15	6.32E-15	5.43E-15	0.85	0.99
Co-59(n,γ)		1.17E-13	1.62E-13	1.20E-13	0.72	0.98
Co-59(n,γ)	Cd	6.68E-14	8.14E-14	6.63E-14	0.82	1.01
U-235(n,f)		1.10E-12	1.46E-12	1.06E-12	0.75	1.04
U-235(n,f)	Cd	2.98E-13	2.86E-13	2.93E-13	1.04	1.02

TABLE 6.3-7

DERIVED EXPOSURE RATES FROM THE CAPSULE EE DOSIMETRY EVALUATION
30.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.24E+08	1.10E+09	8%
ϕ(E > 0.1 MeV)	1.08E+10	1.02E+10	17%
ϕ(E < 0.414 eV)	2.63E+09	1.99E+09	24%
dpa/sec	3.80E-12	3.53E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
30.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	6.79E-19	6.41E-19	6.78E-19	1.06	1.00
Ti-46(n,p)	Cd	1.02E-17	9.42E-18	1.00E-17	1.08	1.02
Fe-54(n,p)	Cd	5.66E-17	5.99E-17	5.72E-17	0.95	0.99
Ni-58(n,p)	Cd	8.01E-17	8.59E-17	8.07E-17	0.93	0.99
U-238(n,f)	Cd	3.08E-16	3.52E-16	3.17E-16	0.88	0.97
Np-237(n,f)	Cd	4.70E-15	5.07E-15	4.64E-15	0.93	1.01
Co-59(n,γ)		1.05E-13	1.38E-13	1.07E-13	0.76	0.98
Co-59(n,γ)	Cd	5.76E-14	6.96E-14	5.73E-14	0.83	1.01
U-235(n,f)		1.02E-12	1.24E-12	9.80E-13	0.82	1.01
U-235(n,f)	Cd	2.66E-13	2.44E-13	2.61E-13	1.09	1.04

TABLE 6.3-8

DERIVED EXPOSURE RATES FROM THE CAPSULE FF DOSIMETRY EVALUATION
45.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.10E+09	9.66E+08	8%
ϕ(E > 0.1 MeV)	9.03E+09	8.09E+09	17%
ϕ(E < 0.414 eV)	2.58E+09	8.25E+08	27%
dpa/sec	3.23E-12	2.88E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
45.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	6.97E-19	6.28E-19	6.92E-19	1.11	1.01
Ti-46(n,p)	Cd	1.00E-17	9.10E-18	9.87E-18	1.10	1.01
Fe-54(n,p)	Cd	5.43E-17	5.60E-17	5.48E-17	0.97	0.99
Ni-58(n,p)	Cd	7.65E-17	7.98E-17	7.68E-17	0.96	1.00
U-238(n,f)	Cd	2.78E-16	3.18E-16	2.87E-16	0.87	0.97
Np-237(n,f)	Cd	3.90E-15	4.36E-15	3.86E-15	0.89	1.01
Co-59(n,γ)		6.50E-14	1.27E-13	6.48E-14	0.51	1.00
Co-59(n,γ)	Cd	4.40E-14	5.83E-14	4.43E-14	0.76	0.99
U-235(n,f)		4.52E-13	1.21E-12	4.70E-13	0.37	0.96
U-235(n,f)	Cd	1.97E-13	2.06E-13	1.90E-13	0.96	1.04

TABLE 6.3-9

DERIVED EXPOSURE RATES FROM THE CAPSULE AA DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE TOP

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
$\phi(E > 1.0$ MeV)	8.26E+08	7.54E+08	10%
$\phi(E > 0.1$ MeV)	6.49E+09	5.91E+09	22%
$\phi(E < 0.414$ eV)	1.55E+09	5.69E+08	26%
dpa/sec	2.33E-12	2.12E-12	17%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE TOP

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	4.23E-19	4.09E-19	4.22E-19	1.03	1.00
Ti-46(n,p)	Cd	6.68E-18	6.12E-18	6.51E-18	1.09	1.03
Fe-54(n,p)	Cd	3.57E-17	3.98E-17	3.75E-17	0.90	0.95
Ni-58(n,p)	Cd	5.67E-17	5.71E-17	5.58E-17	0.99	1.02
U-238(n,f)	Cd	2.26E-16	2.36E-16	2.20E-16	0.96	1.03
Co-59(n,γ)		3.89E-14	7.74E-14	4.13E-14	0.50	0.94
Co-59(n,γ)	Cd	2.79E-14	3.64E-14	2.69E-14	0.77	1.04
U-235(n,f)		3.32E-13	7.27E-13	3.04E-13	0.46	1.09
U-235(n,f)	Cd	9.87E-14	1.29E-13	1.02E-13	0.77	0.97

TABLE 6.3-10

DERIVED EXPOSURE RATES FROM THE CAPSULE CC DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE BOTTOM

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	8.40E+08	6.66E+08	8%
ϕ(E > 0.1 MeV)	6.60E+09	4.89E+09	16%
ϕ(E < 0.414 eV)	1.58E+09	7.10E+08	22%
dpa/sec	2.37E-12	1.79E-12	12%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE BOTTOM

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	3.92E-19	4.16E-19	3.94E-19	0.94	1.00
Ti-46(n,p)	Cd	6.19E-18	6.22E-18	6.09E-18	1.00	1.02
Fe-54(n,p)	Cd	3.63E-17	4.04E-17	3.63E-17	0.90	1.00
Ni-58(n,p)	Cd	5.23E-17	5.81E-17	5.19E-17	0.90	1.01
U-238(n,f)	Cd	1.85E-16	2.40E-16	1.97E-16	0.77	0.94
Np-237(n,f)	Cd	2.51E-15	3.26E-15	2.49E-15	0.77	1.01
Co-59(n,γ)		4.25E-14	7.87E-14	4.57E-14	0.54	0.93
Co-59(n,γ)	Cd	2.79E-14	3.70E-14	2.67E-14	0.75	1.05
U-235(n,f)		3.95E-13	7.39E-13	3.50E-13	0.54	1.13
U-235(n,f)	Cd	7.12E-14	1.31E-13	7.73E-14	0.54	0.92

TABLE 6.3-11

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 17 IRRADIATION

FEET FROM MIDPLANE	NEUTRON FLUX (n/cm^2-sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	4.21E+08	3.25E+08	2.40E+08	2.13E+08
+5.5	1.05E+09	8.23E+08	6.09E+08	4.93E+08
+4.5	1.56E+09	1.12E+09	9.21E+08	7.71E+08
+3.5	1.74E+09	1.35E+09	1.08E+09	8.88E+08
+2.5	1.84E+09	1.50E+09	1.01E+09	9.87E+08
+1.5	1.75E+09	1.46E+09	1.03E+09	9.19E+08
+0.5	1.63E+09	1.37E+09	1.11E+09	8.97E+08
-0.5	1.41E+09	1.31E+09	1.09E+09	1.03E+09
-1.5	1.39E+09	1.18E+09	1.04E+09	9.58E+08
-2.5	1.37E+09	1.33E+09	1.06E+09	1.01E+09
-3.5	1.51E+09	1.31E+09	9.98E+08	9.27E+08
-4.5	1.47E+09	1.31E+09	9.46E+08	8.18E+08
-5.5	1.04E+09	8.69E+08	5.74E+08	5.68E+08
-6.5	4.31E+08	3.18E+08	2.31E+08	2.49E+08

TABLE 6.3-12

FAST NEUTRON FLUX (E > 0.1 MeV) AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 17 IRRADIATION

FEET FROM MIDPLANE	NEUTRON FLUX (n/cm^2-sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	3.66E+09	2.86E+09	2.23E+09	1.78E+09
+5.5	9.13E+09	7.25E+09	5.65E+09	4.13E+09
+4.5	1.36E+10	9.86E+09	8.54E+09	6.46E+09
+3.5	1.51E+10	1.19E+10	1.01E+10	7.44E+09
+2.5	1.59E+10	1.32E+10	9.41E+09	8.27E+09
+1.5	1.52E+10	1.28E+10	9.60E+09	7.70E+09
+0.5	1.42E+10	1.21E+10	1.02E+10	7.51E+09
-0.5	1.22E+10	1.15E+10	1.02E+10	8.67E+09
-1.5	1.20E+10	1.04E+10	9.69E+09	8.03E+09
-2.5	1.19E+10	1.17E+10	9.87E+09	8.43E+09
-3.5	1.31E+10	1.15E+10	9.25E+09	7.76E+09
-4.5	1.28E+10	1.15E+10	8.77E+09	6.85E+09
-5.5	9.04E+09	7.65E+09	5.32E+09	4.76E+09
-6.5	3.75E+09	2.80E+09	2.14E+09	2.09E+09

TABLE 6.3-13

IRON DISPLACEMENT RATE AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 17 IRRADIATION

FEET FROM MIDPLANE	DISPLACEMENT RATE (dpa/sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	1.28E-12	1.01E-12	7.70E-13	6.34E-13
+5.5	3.19E-12	2.55E-12	1.95E-12	1.47E-12
+4.5	4.74E-12	3.47E-12	2.96E-12	2.30E-12
+3.5	5.30E-12	4.18E-12	3.48E-12	2.65E-12
+2.5	5.58E-12	4.64E-12	3.26E-12	2.94E-12
+1.5	5.33E-12	4.51E-12	3.32E-12	2.74E-12
+0.5	4.96E-12	4.25E-12	3.55E-12	2.67E-12
-0.5	4.28E-12	4.05E-12	3.51E-12	3.09E-12
-1.5	4.22E-12	3.66E-12	3.35E-12	2.86E-12
-2.5	4.16E-12	4.12E-12	3.42E-12	3.00E-12
-3.5	4.59E-12	4.05E-12	3.20E-12	2.76E-12
-4.5	4.46E-12	4.05E-12	3.04E-12	2.44E-12
-5.5	3.16E-12	2.69E-12	1.84E-12	1.69E-12
-6.5	1.31E-12	9.85E-13	7.41E-13	7.43E-13

FIGURE 6.3-1

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 0.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 17 IRRADIATION

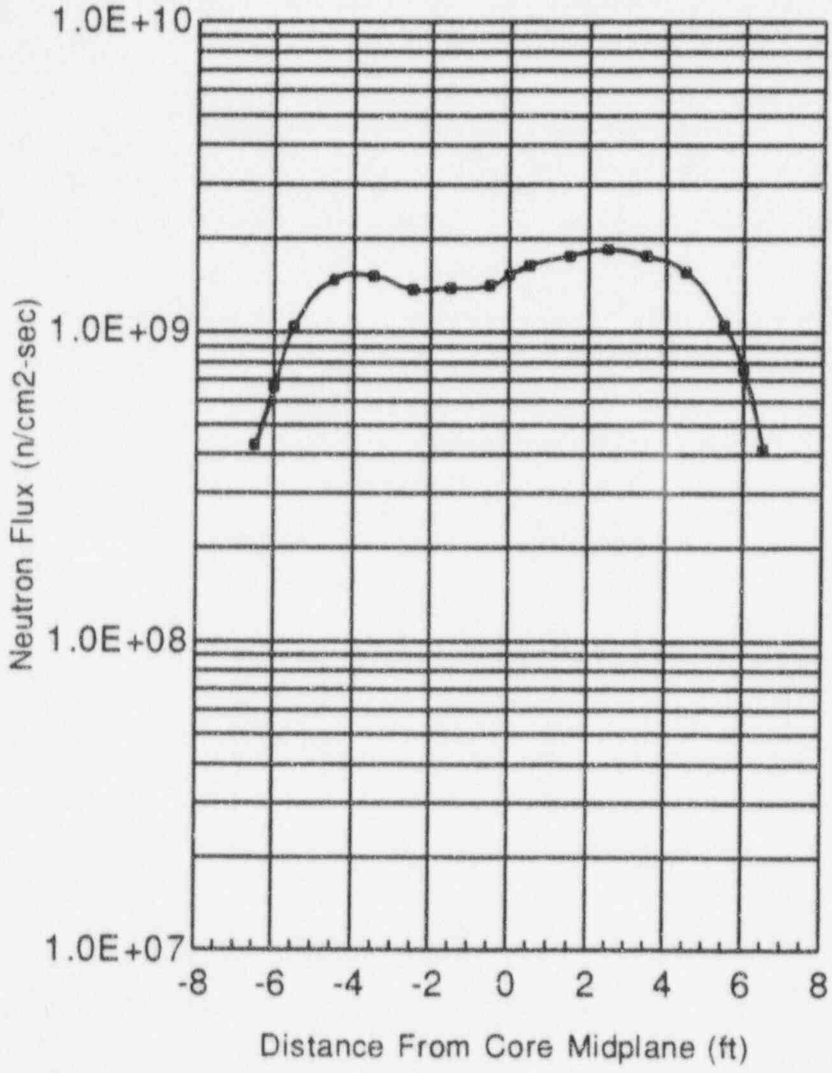


FIGURE 6.3-2

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 15.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 17 IRRADIATION

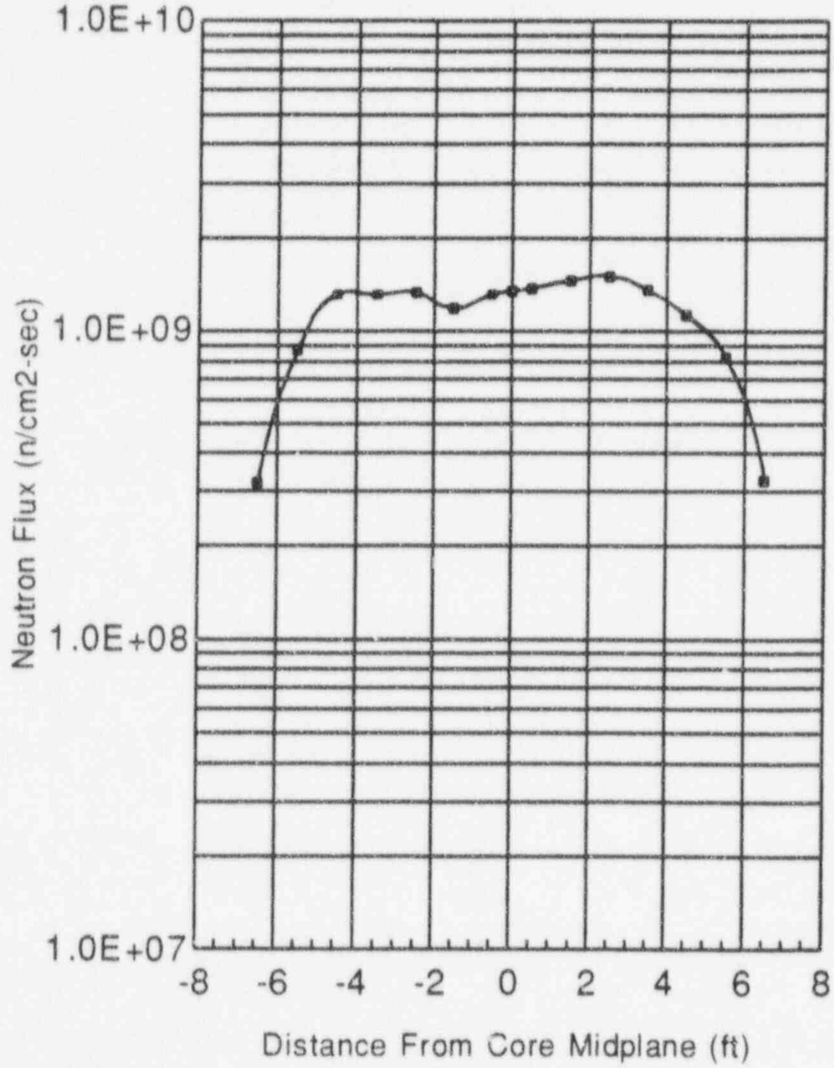


FIGURE 6.3-3

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 30.0DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 17 IRRADIATION

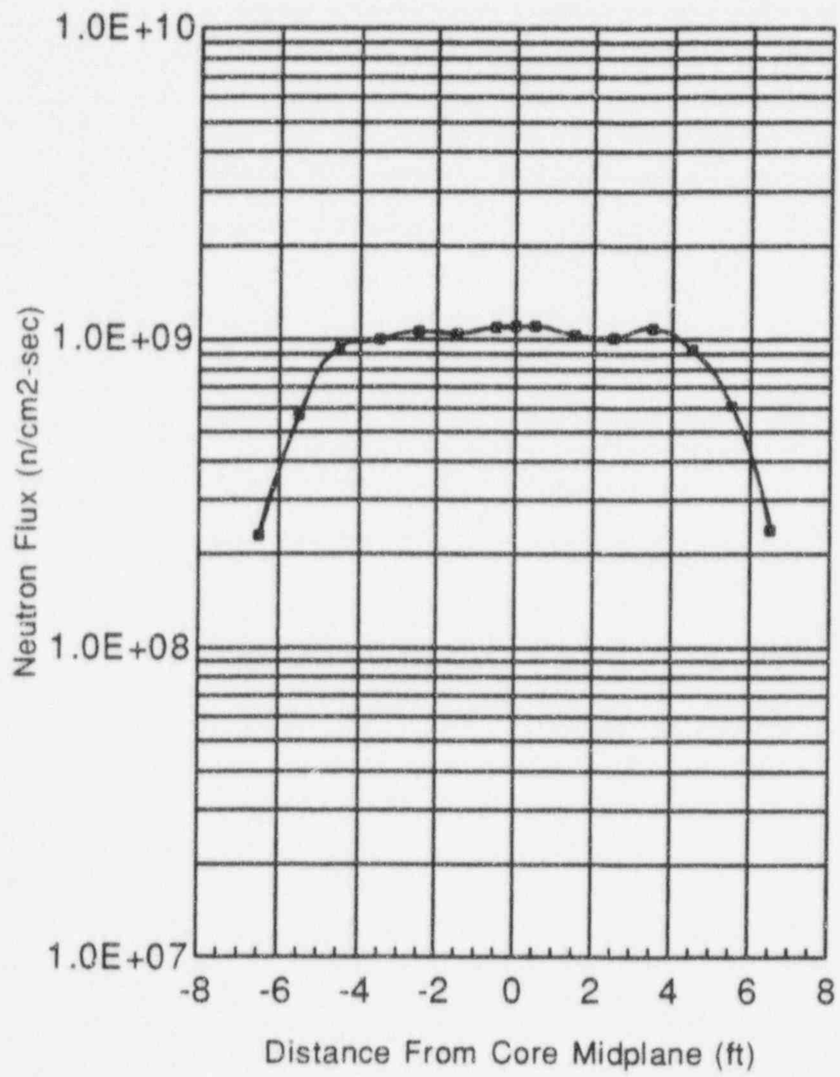
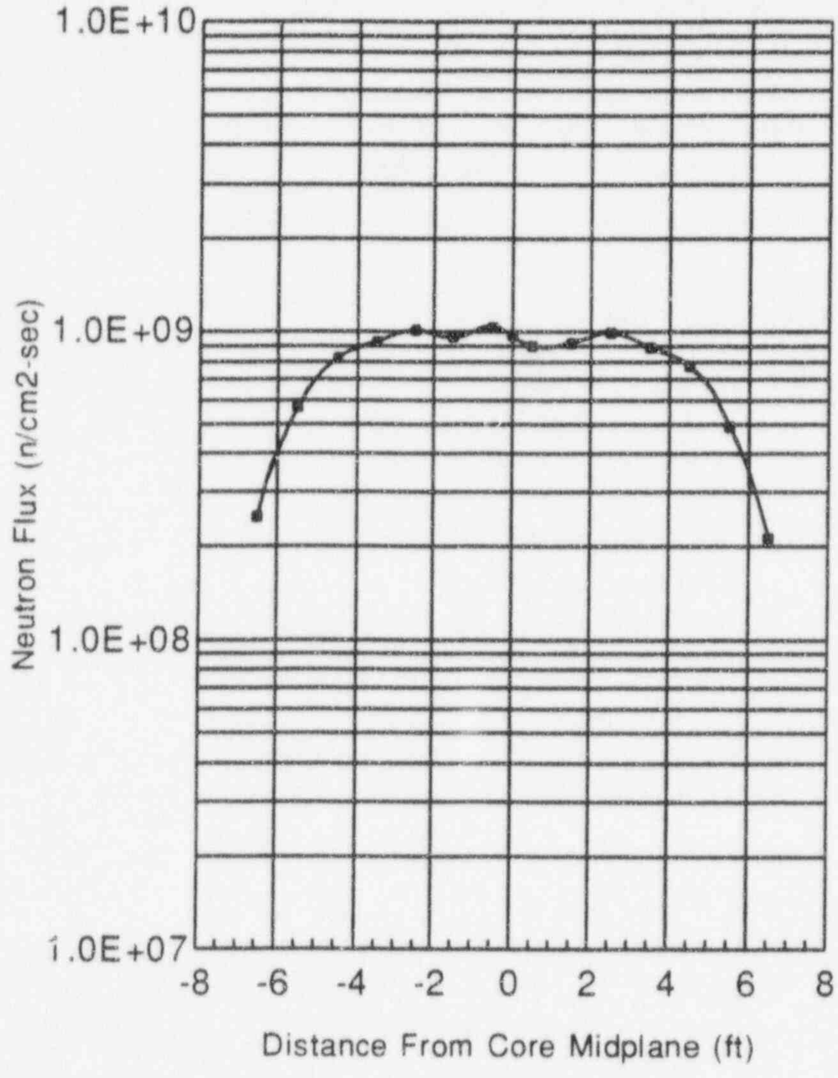


FIGURE 6.3-4

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION ALONG THE 45.0 DEGREE TRAVERSE IN THE REACTOR CAVITY CYCLE 17 IRRADIATION



6.4 - Cycles 18/20 Results

6.4.1 - Measured Reaction rates

During the Cycles 18/20 irradiation, six multiple foil sensor sets and four stainless steel gradient chains were deployed in the reactor cavity as depicted in Figures 2.1-1 and 2.1-2. The capsule identifications associated with each of the multiple foil sensor sets were as follows:

AZIMUTH (degrees)	CAPSULE IDENTIFICATION CORE TOP	CORE MIDPLANE	CORE BOTTOM
0.0	MM	NN	OO
15.0		PP	
30.0		QQ	
45.0		RR	

The contents of each of these irradiation capsules are specified in Appendix E to this report.

The irradiation history of the Point Beach Unit 2 reactor during Cycles 18/20 is also listed in Appendix E. The irradiation history was obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" for the applicable operating period. Based on this reactor operating history, the individual sensor characteristics, and the measured specific activities given in Appendix E, cycle average reaction rates referenced to a core power level of 1518 MWt were computed for each multiple foil sensor and gradient chain segment.

The computed reaction rates for the multiple foil sensor sets, including radiometric foils and solid state track recorders, irradiated during Cycles 18/20 are provided in Table 6.4-1. Corresponding reaction rate data from the four stainless steel gradient chains are recorded in Tables 6.4-2 through 6.4-4 for the Fe-54(n,p), Ni-58(n,p), and Co-59(n,γ) reactions, respectively.

In regard to the data listed in Table 6.4-1, the Fe-54(n,p) reaction rates represent an average of the bare and cadmium

covered measurements for each capsule. Likewise, the U-238(n,f) reaction rates were obtained by averaging the results of the radiometric foil and solid state track recorder data. In addition, the fission rate measurements include corrections for U-235 impurities in the U-238 sensors as well as corrections for photo-fission reactions in both the U-238 and Np-237 sensors.

6.4.2 - Results of the Least Squares Adjustment Procedure

The results of the application of the least squares adjustment procedure to the six sets of multiple foil measurements obtained from the Cycles 18/20 irradiation are provided in Tables 6.4-5 through 6.4-10. In these tables, the derived exposure experienced at each sensor set location along with data illustrating the fit of both the trial and adjusted spectra to the measurements are given. Also included in the tabulations are the 1σ uncertainties associated with each of the derived exposure rates.

In regard to the comparisons listed in Tables 6.4-5 through 6.4-10, it should be noted that the columns labeled "trial calc" were obtained by normalizing the neutron spectral data from Table 4.1-1 to the absolute calculated neutron flux (E > 1.0 MeV) averaged over the Cycles 18/20 irradiation period as discussed in Section 3.0. Thus, the comparisons illustrated in Tables 6.4-5 through 6.4-10 indicate the degree to which the calculated neutron energy spectra matched the measured data before and after adjustment. Absolute comparisons are discussed further in Section 7.0 of this report.

Complete traverses of fast neutron exposure rates in the reactor cavity were developed by combining the results of the least squares adjustment of the multiple foil data with the Fe-54(n,p) reaction rate measurements from the gradient chains. The gradient data were employed to establish relative axial distributions over the measurement range and these relative distributions were then normalized to the FERRET results from the midplane sensor sets to produce axial distributions of exposure rates in terms of ϕ(E > 1.0 MeV), ϕ(E > 0.1 MeV), and dpa/sec in the reactor cavity.

The resultant axial distributions of $\phi(E > 1.0 \text{ MeV})$, $\phi(E > 0.1 \text{ MeV})$, and dpa/sec from the gradient chain measurements are given in Tables 6.4-11, 6.4-12, and 6.4-13, respectively. The distributions of $\phi(E > 1.0 \text{ MeV})$ are depicted graphically in Figures 6.4-1 through 6.4-4. In these graphical presentations, results for axial locations of -6.0, 0.0, and +6.0 feet relative to the core midplane represent the explicit results of the FERRET evaluations summarized in Tables 6.4-5 through 6.4-10, while results at the remaining axial locations depict the normalized data from the gradient chains.

TABLE 6.4-1

SUMMARY OF REACTION RATES DERIVED FROM MULTIPLE FOIL SENSOR SETS
CYCLES 18/20 IRRADIATION

		REACTION RATE (rps/nucleus)					
		CAPSULE NN	CAPSULE PP	CAPSULE QQ	CAPSULE RR	CAPSULE MM	CAPSULE OO
Cu-63(n,α)	Cd	8.97E-19	7.95E-19	7.09E-19	6.97E-19	3.41E-19	3.14E-19
Ti-46(n,p)	Cd	1.37E-17	1.16E-17	1.04E-17	9.71E-18	5.56E-18	5.49E-18
Fe-54(n,p)	Cd	7.56E-17	6.47E-17	5.57E-17	5.25E-17	2.85E-17	3.08E-17
Ni-58 (n,p)	Cd	1.10E-16	9.34E-17	7.99E-17	7.57E-17	4.67E-17	4.53E-17
U-238(n,f)	Cd	4.59E-16	4.16E-16	3.37E-16	3.15E-16	1.86E-16	1.80E-16
Np-237(n,f)	Cd	5.76E-15	5.57E-15	4.73E-15	4.14E-15	2.80E-15	2.21E-15
Co-59(n,γ)		8.74E-14	1.09E-13	9.75E-14	6.19E-14	3.47E-14	3.66E-14
Co-59(n,γ)	Cd	5.34E-14	6.29E-14	5.43E-14	4.37E-14	2.50E-14	2.45E-14
U-235(n,f)		8.74E-13	1.18E-12	1.10E-12	5.36E-13	2.70E-13	3.03E-13
U-235(n,f)	Cd	2.19E-13	3.04E-13	2.47E-13	1.88E-13	6.83E-14	5.94E-14

Note: Cd indicates that the sensor was cadmium covered.

TABLE 6.4-2

Fe-54(n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLES 18/20 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	1.58E-17	1.19E-17	3.17E-17	9.24E-18
+5.5	4.04E-17	3.28E-17	2.40E-17	2.31E-17
+4.5	6.90E-17	5.95E-17	4.11E-17	3.75E-17
+3.5	7.81E-17	6.07E-17	4.95E-17	3.98E-17
+2.5	8.61E-17	7.02E-17	4.91E-17	4.81E-17
+1.5	8.50E-17	6.41E-17	4.72E-17	4.66E-17
+0.5	7.51E-17	6.34E-17	5.10E-17	4.77E-17
-0.5	6.21E-17	5.53E-17	5.06E-17	4.85E-17
-1.5	5.91E-17	5.46E-17	5.33E-17	4.58E-17
-2.5	6.52E-17	6.30E-17	4.76E-17	4.51E-17
-3.5	7.16E-17	6.34E-17	5.10E-17	4.43E-17
-4.5	6.44E-17	4.96E-17	3.63E-17	3.60E-17
-5.5	4.04E-17	3.24E-17	2.54E-17	2.31E-17
-6.5	1.33E-17	1.18E-17	9.29E-18	8.83E-18

TABLE 6.4-3

Ni-58(n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLES 18/20 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	2.25E-17	1.84E-17	4.17E-17	1.35E-17
+5.5	6.19E-17	4.77E-17	3.60E-17	3.13E-17
+4.5	9.87E-17	8.03E-17	5.95E-17	5.22E-17
+3.5	1.21E-16	9.50E-17	7.25E-17	6.16E-17
+2.5	1.28E-16	9.82E-17	7.25E-17	7.15E-17
+1.5	1.23E-16	9.56E-17	7.62E-17	6.22E-17
+0.5	1.13E-16	8.67E-17	7.75E-17	6.12E-17
-0.5	9.56E-17	8.86E-17	7.50E-17	7.03E-17
-1.5	8.41E-17	7.91E-17	7.50E-17	6.28E-17
-2.5	9.62E-17	8.35E-17	6.68E-17	7.03E-17
-3.5	1.07E-16	9.56E-17	6.06E-17	6.40E-17
-4.5	9.37E-17	7.65E-17	5.35E-17	5.43E-17
-5.5	6.07E-17	5.02E-17	3.33E-17	3.40E-17
-6.5	2.13E-17	2.05E-17	1.39E-17	1.42E-17

TABLE 6.4-4

Co-59(n,γ) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS - CYCLES 18/20 IRRADIATION

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	1.87E-14	1.95E-14	5.13E-14	1.39E-14
+5.5	4.25E-14	6.26E-14	4.56E-14	2.96E-14
+4.5	5.98E-14	9.34E-14	6.75E-14	3.99E-14
+3.5	6.95E-14	1.09E-13	8.23E-14	4.80E-14
+2.5	7.76E-14	1.14E-13	9.04E-14	5.49E-14
+1.5	7.80E-14	1.11E-13	9.16E-14	5.77E-14
+0.5	8.00E-14	1.04E-13	9.12E-14	5.73E-14
-0.5	8.02E-14	9.89E-14	8.80E-14	5.75E-14
-1.5	7.74E-14	9.34E-14	8.66E-14	5.67E-14
-2.5	7.62E-14	9.16E-14	8.29E-14	5.45E-14
-3.5	7.27E-14	8.82E-14	7.42E-14	4.92E-14
-4.5	5.94E-14	7.28E-14	5.76E-14	3.99E-14
-5.5	3.83E-14	5.07E-14	3.08E-14	2.75E-14
-6.5	2.88E-14	2.66E-14	2.11E-14	2.05E-14

TABLE 6.4-5

DERIVED EXPOSURE RATES FROM THE CAPSULE NN DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.78E+09	1.51E+09	8%
ϕ(E > 0.1 MeV)	1.40E+10	1.20E+10	16%
ϕ(E < 0.414 eV)	3.34E÷09	1.58E+09	24%
dpa/sec	5.02E-12	4.29E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	8.97E-19	8.82E-19	8.92E-19	1.02	1.01
Ti-46(n,p)	Cd	1.37E-17	1.32E-17	1.34E-17	1.04	1.02
Fe-54(n,p)	Cd	7.56E-17	8.57E-17	7.76E-17	0.88	0.97
Ni-58(n,p)	Cd	1.10E-16	1.23E-16	1.11E-17	0.89	0.99
U-238(n,f)	Cd	4.60E-16	5.09E-16	4.44E-16	0.90	1.04
Np-237(n,f)	Cd	5.76E-15	6.92E-15	5.79E-15	0.83	1.00
Co-59(n,γ)		8.74E-14	1.67E-13	9.20E-14	0.52	0.95
Co-59(n,γ)	Cd	5.34E-14	7.85E-14	5.21E-14	0.68	1.03
U-235(n,f)		8.74E-13	1.57E-12	7.96E-13	0.56	1.10
U-235(n,f)	Cd	2.19E-13	2.78E-13	2.23E-13	0.79	0.98

TABLE 6.4-6

DERIVED EXPOSURE RATES FROM THE CAPSULE PP DOSIMETRY EVALUATION
15.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
φ(E > 1.0 MeV)	1.54E+09	1.36E+09	8%
φ(E > 0.1 MeV)	1.33E+10	1.24E+10	17%
φ(E < 0.414 eV)	3.07E+09	2.11E+09	24%
dpa/sec	4.67E-12	4.31E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
15.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	7.95E-19	7.42E-19	7.86E-19	1.07	1.01
Ti-46(n,p)	Cd	1.16E-17	1.10E-17	1.14E-17	1.06	1.02
Fe-54(n,p)	Cd	6.47E-17	7.20E-17	6.66E-17	0.90	0.97
Ni-58(n,p)	Cd	9.34E-17	1.04E-16	9.47E-17	0.90	0.99
U-238(n,f)	Cd	4.16E-16	4.34E-16	3.93E-16	0.96	1.06
Np-237(n,f)	Cd	5.57E-15	6.28E-15	5.62E-15	0.89	0.99
Co-59(n,γ)		1.09E-13	1.61E-13	1.14E-13	0.68	0.96
Co-59(n,γ)	Cd	6.29E-14	8.09E-14	6.17E-14	0.78	1.02
U-235(n,f)		1.18E-12	1.45E-12	1.07E-12	0.81	1.10
U-235(n,f)	Cd	3.04E-13	2.85E-13	3.02E-13	1.07	1.01

TABLE 6.4-7

DERIVED EXPOSURE RATES FROM THE CAPSULE QQ DOSIMETRY EVALUATION
30.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
φ(E > 1.0 MeV)	1.26E+09	1.13E+09	8%
φ(E > 0.1 MeV)	1.10E+10	1.04E+10	17%
φ(E < 0.414 eV)	2.68E+09	1.97E+09	23%
dpa/sec	3.87E-12	3.62E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
30.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	7.09E-19	6.53E-19	7.02E-19	1.09	1.01
Ti-46(n,p)	Cd	1.04E-17	9.61E-18	1.02E-17	1.08	1.02
Fe-54(n,p)	Cd	5.57E-17	6.10E-17	5.72E-17	0.91	0.97
Ni-58(n,p)	Cd	7.99E-17	8.76E-17	8.09E-17	0.91	0.99
U-238(n,f)	Cd	3.37E-16	3.59E-16	3.26E-16	0.94	1.03
Np-237(n,f)	Cd	4.73E-15	5.16E-15	4.71E-15	0.92	1.00
Co-59(n,γ)		9.75E-14	1.41E-13	1.03E-13	0.69	0.95
Co-59(n,γ)	Cd	5.43E-14	7.09E-14	5.31E-14	0.77	1.02
U-235(n,f)		1.10E-12	1.26E-12	9.76E-13	0.87	1.13
U-235(n,f)	Cd	2.47E-13	2.49E-13	2.48E-13	0.99	1.00

TABLE 6.4-8

DERIVED EXPOSURE RATES FROM THE CAPSULE RR DOSIMETRY EVALUATION
45.0 DEGREE AZIMUTH - CORE MIDPLANE

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	1.13E+09	1.02E+09	8%
ϕ(E > 0.1 MeV)	9.28E+09	8.74E+09	17%
ϕ(E < 0.414 eV)	2.65E+09	9.01E+08	27%
dpa/sec	3.32E-12	3.09E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
45.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	6.97E-19	6.45E-19	6.86E-19	1.08	1.02
Ti-46(n,p)	Cd	9.71E-18	9.35E-18	9.60E-18	1.04	1.01
Fe-54(n,p)	Cd	5.25E-17	5.76E-17	5.40E-17	0.91	0.97
Ni-58(n,p)	Cd	7.57E-17	8.20E-17	7.64E-17	0.92	0.99
U-238(n,f)	Cd	3.15E-16	3.27E-16	3.01E-16	0.96	1.05
Np-237(n,f)	Cd	4.14E-15	4.48E-15	4.12E-15	0.92	1.01
Co-59(n,γ)		6.19E-14	1.30E-13	6.50E-14	0.48	0.95
Co-59(n,γ)	Cd	4.37E-14	5.99E-14	4.25E-14	0.73	1.03
U-235(n,f)		5.36E-13	1.24E-12	5.02E-13	0.43	1.07
U-235(n,f)	Cd	1.88E-13	2.11E-13	1.89E-13	0.89	1.00

TABLE 6.4-9

DERIVED EXPOSURE RATES FROM THE CAPSULE MM DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE TOP

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	6.60E+08	6.50E+08	8%
ϕ(E > 0.1 MeV)	5.18E+09	5.33E+09	16%
ϕ(E < 0.414 eV)	1.24E+09	4.85E+08	25%
dpa/sec	1.86E-12	1.88E-12	13%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE TOP

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	3.41E-19	3.26E-19	3.40E-19	1.05	1.00
Ti-46(n,p)	Cd	5.56E-18	4.88E-18	5.37E-18	1.14	1.04
Fe-54(n,p)	Cd	2.85E-17	3.17E-17	3.03E-17	0.90	0.94
Ni-58(n,p)	Cd	4.67E-17	4.56E-17	4.59E-17	1.02	1.02
U-238(n,f)	Cd	1.86E-16	1.89E-16	1.84E-16	0.98	1.01
Np-237(n,f)	Cd	2.80E-15	2.56E-15	2.68E-15	1.09	1.05
Co-59(n,γ)		3.47E-14	6.18E-14	3.67E-14	0.56	0.95
Co-59(n,γ)	Cd	2.50E-14	2.91E-14	2.40E-14	0.86	1.04
U-235(n,f)		2.70E-13	5.80E-13	2.50E-13	0.47	1.08
U-235(n,f)	Cd	6.83E-14	1.03E-13	7.33E-14	0.66	0.93

TABLE 6.4-10

DERIVED EXPOSURE RATES FROM THE CAPSULE OO DOSIMETRY EVALUATION
0.0 DEGREE AZIMUTH - CORE BOTTOM

	TRIAL VALUE	ADJUSTED VALUE	1σ UNCERTAINTY
ϕ(E > 1.0 MeV)	7.13E+08	6.07E+08	8%
ϕ(E > 0.1 MeV)	5.60E+09	4.32E+09	16%
ϕ(E < 0.414 eV)	1.34E+09	5.64E+08	23%
dpa/sec	2.01E-12	1.59E-12	12%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
0.0 DEGREE AZIMUTH - CORE MIDPLANE

		REACTION RATE (rps/nucleus)				
		MEASURED	TRIAL CALC.	ADJUSTED CALC.	M/C TRIAL	M/C ADJUSTED
Cu-63(n,α)	Cd	3.14E-19	3.53E-19	3.19E-19	0.89	0.98
Ti-46(n,p)	Cd	5.49E-18	5.28E-18	5.32E-18	1.04	1.03
Fe-54(n,p)	Cd	3.08E-17	3.43E-17	3.13E-17	0.90	0.98
Ni-58(n,p)	Cd	4.53E-17	4.93E-17	4.51E-17	0.92	1.00
U-238(n,f)	Cd	1.80E-16	2.04E-16	1.79E-16	0.88	1.01
Np-237(n,f)	Cd	2.21E-15	2.77E-15	2.22E-15	0.80	1.00
Co-59(n,γ)		3.66E-14	6.68E-14	3.87E-14	0.55	0.95
Co-59(n,γ)	Cd	2.45E-14	3.14E-14	2.36E-14	0.78	1.04
U-235(n,f)		3.03E-13	6.27E-13	2.78E-13	0.48	1.09
U-235(n,f)	Cd	5.94E-14	1.11E-13	6.44E-14	0.54	0.92

TABLE 6.4-11

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION WITHIN THE REACTOR CAVITY CYCLES 18/20 IRRADIATION

FEET FROM MIDPLANE	NEUTRON FLUX (n/cm^2-sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	3.48E+08	2.72E+08		1.96E+08
+5.5	8.89E+08	7.52E+08	5.33E+08	4.89E+08
+4.5	1.52E+09	1.36E+09	9.14E+08	7.95E+08
+3.5	1.72E+09	1.39E+09	1.10E+09	8.43E+08
+2.5	1.90E+09	1.61E+09	1.09E+09	1.02E+09
+1.5	1.87E+09	1.47E+09	1.05E+09	9.88E+08
+0.5	1.65E+09	1.45E+09	1.13E+09	1.01E+09
-0.5	1.37E+09	1.27E+09	1.13E+09	1.03E+09
-1.5	1.30E+09	1.25E+09	1.19E+09	9.72E+08
-2.5	1.43E+09	1.44E+09	1.06E+09	9.56E+08
-3.5	1.58E+09	1.45E+09	1.13E+09	9.40E+08
-4.5	1.42E+09	1.14E+09	8.08E+08	7.63E+08
-5.5	8.89E+08	7.43E+08	5.64E+08	4.90E+08
-6.5	2.93E+08	2.70E+08	2.07E+08	1.87E+08

TABLE 6.4-12

FAST NEUTRON FLUX (E > 0.1 MeV) AS A FUNCTION OF AXIAL POSITION WITHIN THE REACTOR CAVITY CYCLES 18/20 IRRADIATION

FEET FROM MIDPLANE	NEUTRON FLUX (n/cm^2-sec) 0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	2.77E+09	2.48E+09		1.68E+09
+5.5	7.07E+09	6.86E+09	4.90E+09	4.19E+09
+4.5	1.21E+10	1.24E+10	8.41E+09	6.81E+09
+3.5	1.37E+10	1.27E+10	1.01E+10	7.23E+09
+2.5	1.51E+10	1.47E+10	1.01E+10	8.74E+09
+1.5	1.49E+10	1.34E+10	9.66E+09	8.46E+09
+0.5	1.31E+10	1.32E+10	1.04E+10	8.67E+09
-0.5	1.09E+10	1.16E+10	1.04E+10	8.81E+09
-1.5	1.03E+10	1.14E+10	1.09E+10	8.33E+09
-2.5	1.14E+10	1.32E+10	9.74E+09	8.19E+09
-3.5	1.25E+10	1.32E+10	1.04E+10	8.05E+09
-4.5	1.13E+10	1.04E+10	7.43E+09	6.54E+09
-5.5	7.07E+09	6.78E+09	5.19E+09	4.20E+09
-6.5	2.33E+09	2.46E+09	1.90E+09	1.60E+09

TABLE 6.4-13

IRON DISPLACEMENT RATE AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLES 18/20 IRRADIATION

FEET FROM MIDPLANE	DISPLACEMENT RATE (dpa/sec)			
	0.0 DEG	15.0 DEG	30.0 DEG	45.0 DEG
+6.5	9.89E-13	8.62E-13		5.94E-13
+5.5	2.53E-12	2.38E-12	1.71E-12	1.48E-12
+4.5	4.31E-12	4.32E-12	2.93E-12	2.41E-12
+3.5	4.89E-12	4.41E-12	3.53E-12	2.55E-12
+2.5	5.39E-12	5.10E-12	3.50E-12	3.09E-12
+1.5	5.31E-12	4.66E-12	3.36E-12	2.99E-12
+0.5	4.70E-12	4.60E-12	3.63E-12	3.07E-12
-0.5	3.88E-12	4.02E-12	3.61E-12	3.11E-12
-1.5	3.69E-12	3.96E-12	3.80E-12	2.94E-12
-2.5	4.08E-12	4.57E-12	3.39E-12	2.90E-12
-3.5	4.48E-12	4.60E-12	3.63E-12	2.85E-12
-4.5	4.03E-12	3.60E-12	2.59E-12	2.31E-12
-5.5	2.53E-12	2.36E-12	1.81E-12	1.48E-12
-6.5	8.32E-13	8.56E-13	6.62E-13	5.67E-13

FIGURE 6.4-1

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 0.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLES 18/20 IRRADIATION

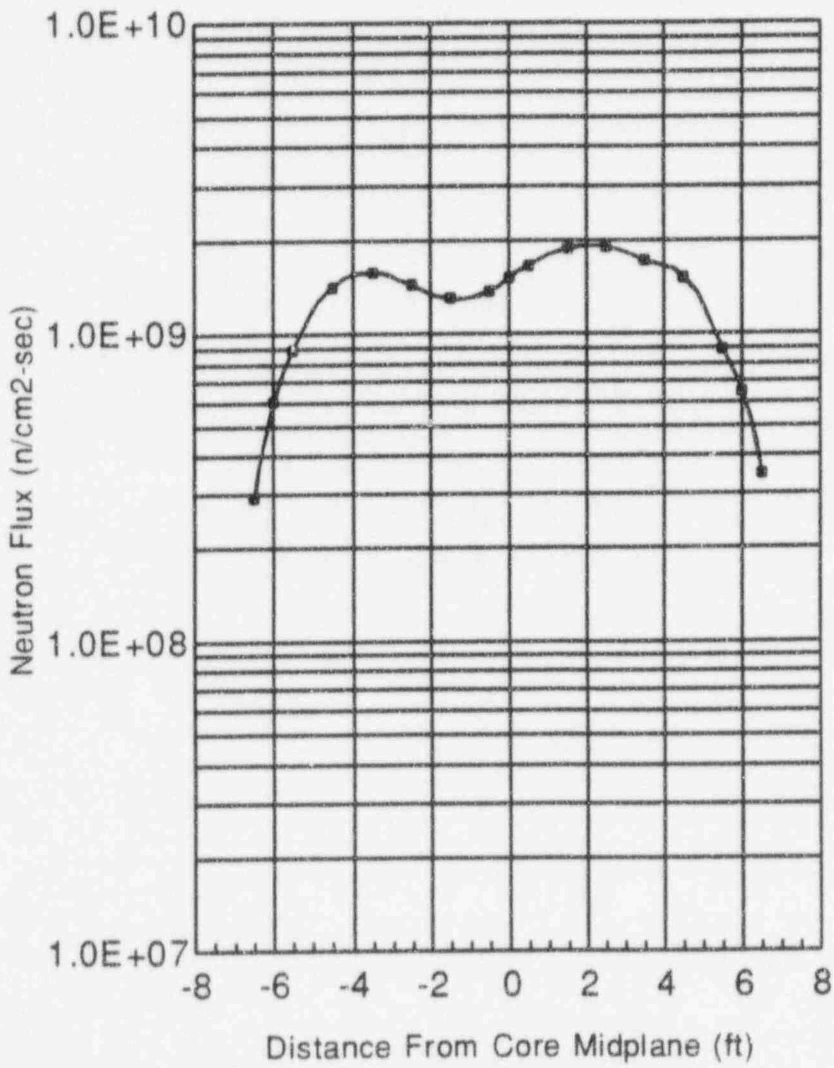


FIGURE 6.4-2

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 15.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLES 18/20 IRRADIATION

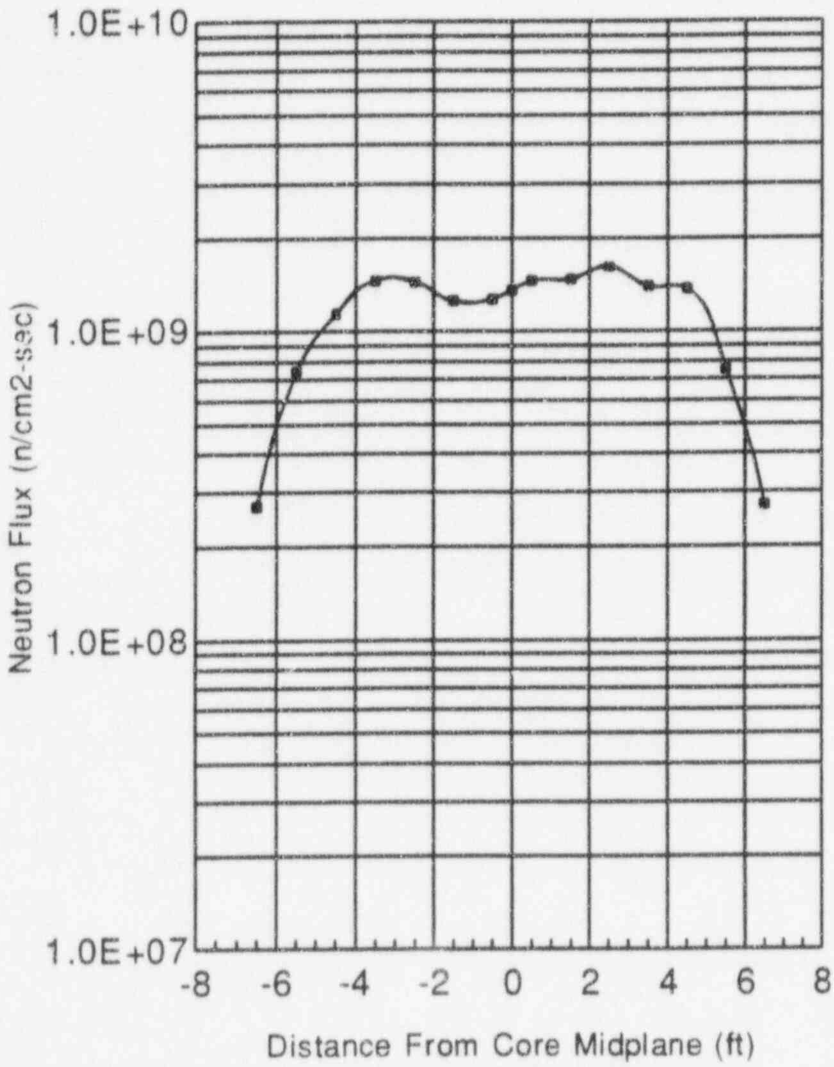


FIGURE 6.4-3

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 30.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLES 18/20 IRRADIATION

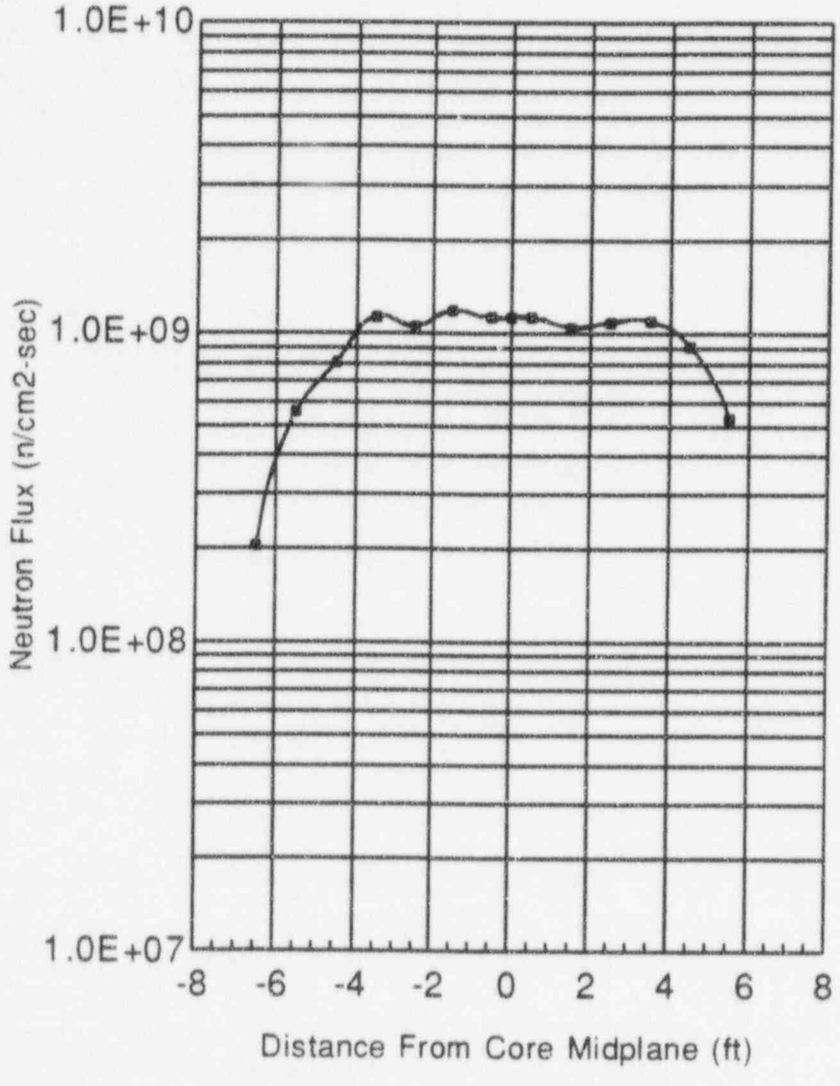
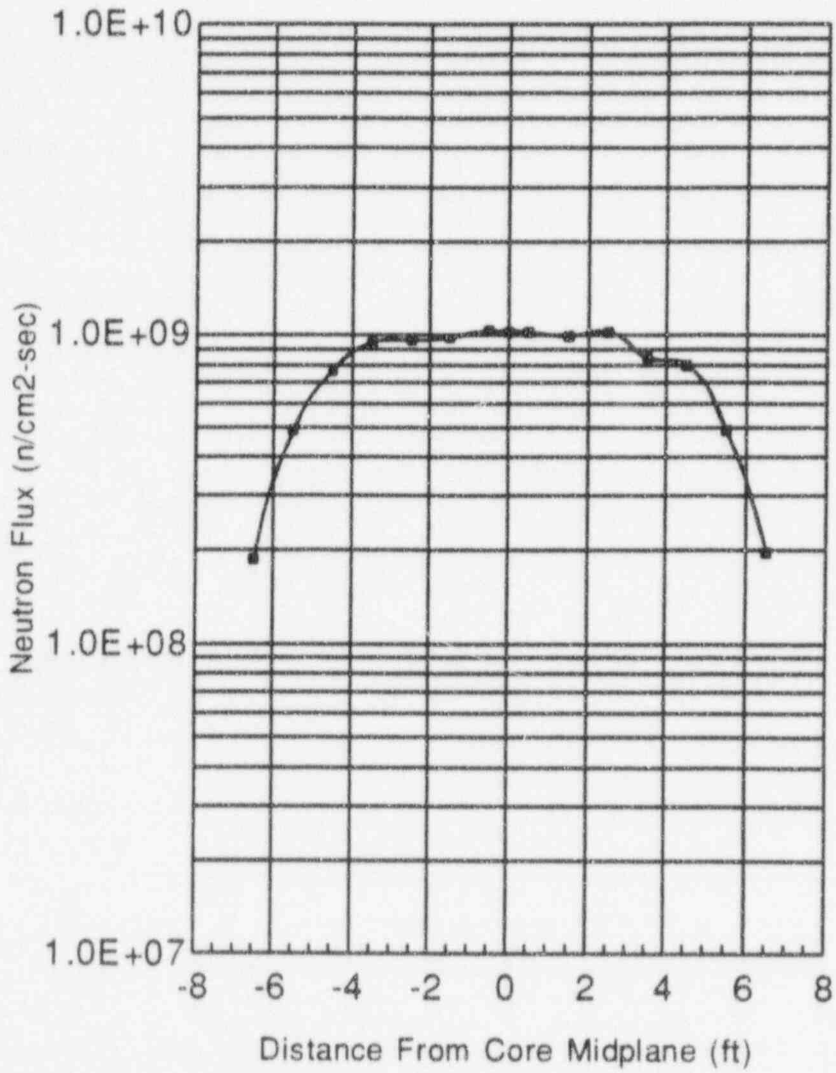


FIGURE 6.4-4

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 45.0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLES 18/20 IRRADIATION



SECTION 7.0

COMPARISON OF CALCULATIONS WITH MEASUREMENTS

As described in Section 3.3, the best estimate neutron exposure projections for the Point Beach Unit 2 pressure vessel were based on a combination of plant specific neutron transport calculations and plant specific measurements. Direct comparisons of the transport calculations with the Point Beach Unit 2 measurement data base were used to quantify the biases that may exist due to the transport methodology, reactor modeling, and/or reactor operating characteristics over the respective irradiation periods.

In this section, comparisons of the measurement results from surveillance capsule and reactor cavity dosimetry with corresponding analytical predictions at the measurement locations are presented. These comparisons are provided on two levels. In the first instance, predictions of fast neutron exposure rates in terms of ϕ(E > 1.0 MeV), ϕ(E > 0.1 MeV), and dpa/sec are compared with the results of the FERRET least squares adjustment procedure; while, in the second case, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. It is shown that these two levels of comparison yield consistent and similar results, indicating that the least squares adjustment methodology is producing accurate exposure results and that the measurement/calculation (M/C) comparisons yield an accurate plant specific bias factor that can be applied to neutron transport calculations performed for the Point Beach Unit 2 reactor to produce "best estimate" exposure projections for the pressure vessel wall.

7.1 Comparison of Least Squares Adjustment Results with Calculation

In Table 7.1-1, comparisons of measured and calculated exposure rates for the four surveillance capsule dosimetry sets withdrawn to date as well as for the four cycles of reactor cavity midplane dosimetry sets irradiated during Cycles 15, 16, 17, and 18/20 are given. In all cases, the calculated values were based on the fuel cycle specific exposure calculations averaged over the appropriate irradiation period.

An examination of Table 7.1-1 indicates that, considering all of the available core midplane data, the measured exposure rates were less than calculated values by factors of 0.921, 0.975, and 0.959 for $\phi(E > 1.0$ MeV), $\phi(E > 0.1$ MeV), and dpa/sec, respectively. The standard deviations associated with each of the 20 sample data sets were 0.074 (8.0%), 0.161 (16.5%), and 0.120 (12.5%), respectively.

7.2 Comparisons of Measured and Calculated Sensor Reaction Rates

In Table 7.2-1, measurement/calculation (M/C) ratios for each fast neutron sensor reaction rate from the surveillance capsule and reactor cavity irradiations are listed. This tabulation, provides a direct comparison, on an absolute basis, of calculation and measurement prior to the application of the least squares adjustment procedure as represented in the FERRET evaluations.

An examination of Table 7.2-1 shows consistent behavior for all reactions and all measurement points. The overall average M/C ratio for the entire data set has an associated 1σ standard deviation of 0.081 (8.3%). Furthermore, the average M/C bias of 0.974 observed in the reaction rate comparisons is in excellent agreement with the values of 0.921, 0.975, and 0.959 observed in the exposure rate comparisons shown in Table 7.1-1.

TABLE 7.1-1

COMPARISON OF MEASURED AND CALCULATED EXPOSURE RATES FROM SURVEILLANCE CAPSULE AND CAVITY DOSIMETRY IRRADIATIONS

	ϕ(E > 1.0 MeV) [n/cm^2-sec]		
	CALCULATED	MEASURED	M/C
INTERNAL CAPSULES			
V (13 DEGREES)	1.35E+11	1.44E+11	1.067
T (23 DEGREES)	7.91E+10	8.21E+10	1.037
R (13 DEGREES)	1.34E+11	1.39E+11	1.031
S (33 DEGREES)	6.65E+10	6.81E+10	1.025
0 DEGREE CAVITY			
Cycle 15	2.22E+09	1.95E+09	0.877
Cycle 16	1.80E+09	1.53E+09	0.847
Cycle 17	1.81E+09	1.52E+09	0.841
Cycle 18/20	1.78E+09	1.51E+09	0.848
15 DEGREE CAVITY			
Cycle 15	1.89E+09	1.78E+09	0.942
Cycle 16	1.54E+09	1.35E+09	0.875
Cycle 17	1.55E+09	1.34E+09	0.866
Cycle 18/20	1.54E+09	1.36E+09	0.885
30 DEGREE CAVITY			
Cycle 15	1.42E+09	1.31E+09	0.924
Cycle 16	1.21E+09	1.06E+09	0.873
Cycle 17	1.24E+09	1.10E+09	0.886
Cycle 18/20	1.26E+09	1.13E+09	0.890
45 DEGREE CAVITY			
Cycle 15	1.17E+09	1.12E+09	0.959
Cycle 16	1.06E+09	1.03E+09	0.967
Cycle 17	1.10E+09	9.66E+08	0.878
Cycle 18/20	1.13E+09	1.02E+09	0.906
AVERAGE M/C BIAS FACTOR (K)			0.921
STANDARD DEVIATION (1σ)			0.074

TABLE 7.1-1 (continued)

COMPARISON OF MEASURED AND CALCULATED EXPOSURE RATES FROM SURVEILLANCE CAPSULE AND CAVITY DOSIMETRY IRRADIATIONS

	ϕ(E > 0.1 MeV) [n/cm^2-sec] CALCULATED	MEASURED	M/C
INTERNAL CAPSULES			
V (13 DEGREES)	5.12E+11	5.71E+11	1.114
T (23 DEGREES)	2.72E+11	2.97E+11	1.091
R (13 DEGREES)	5.09E+11	5.61E+11	1.101
S (33 DEGREES)	2.35E+11	2.54E+11	1.084
0 DEGREE CAVITY			
Cycle 15	1.74E+10	1.62E+10	0.928
Cycle 16	1.41E+10	1.31E+10	0.925
Cycle 17	1.42E+10	1.32E+10	0.928
Cycle 18/20	1.40E+10	1.20E+10	0.854
15 DEGREE CAVITY			
Cycle 15	1.63E+10	1.66E+10	1.020
Cycle 16	1.33E+10	1.24E+10	0.934
Cycle 17	1.34E+10	1.18E+10	0.884
Cycle 18/20	1.33E+10	1.24E+10	0.934
30 DEGREE CAVITY			
Cycle 15	1.24E+10	1.21E+10	0.977
Cycle 16	1.05E+10	9.77E+09	0.928
Cycle 17	1.08E+10	1.02E+10	0.941
Cycle 18/20	1.10E+10	1.04E+10	0.946
45 DEGREE CAVITY			
Cycle 15	9.60E+09	9.99E+09	1.040
Cycle 16	8.70E+09	9.06E+09	1.041
Cycle 17	9.03E+09	8.09E+09	0.896
Cycle 18/20	9.28E+09	8.74E+09	0.942
AVERAGE M/C BIAS FACTOR (K)			0.975
STANDARD DEVIATION (1σ)			0.161

TABLE 7.1-1 (continued)

COMPARISON OF MEASURED AND CALCULATED EXPOSURE RATES FROM SURVEILLANCE CAPSULE AND CAVITY DOSIMETRY IRRADIATIONS

	Iron Displacements [dpa-sec]		
	CALCULATED	MEASURED	M/C
INTERNAL CAPSULES			
V (13 DEGREES)	2.41E-10	2.60E-10	1.081
T (23 DEGREES)	1.35E-10	1.43E-10	1.060
R (13 DEGREES)	2.39E-10	2.55E-10	1.065
S (33 DEGREES)	1.14E-10	1.20E-10	1.052
0 DEGREE CAVITY			
Cycle 15	6.26E-12	5.72E-12	0.914
Cycle 16	5.08E-12	4.59E-12	0.904
Cycle 17	5.11E-12	4.62E-12	0.906
Cycle 18/20	5.02E-12	4.29E-12	0.854
15 DEGREE CAVITY			
Cycle 15	5.73E-12	5.74E-12	1.002
Cycle 16	4.67E-12	4.29E-12	0.920
Cycle 17	4.70E-12	4.15E-12	0.883
Cycle 18/20	4.67E-12	4.31E-12	0.924
30 DEGREE CAVITY			
Cycle 15	4.35E-12	4.20E-12	0.965
Cycle 16	3.71E-12	3.40E-12	0.918
Cycle 17	3.80E-12	3.53E-12	0.931
Cycle 18/20	3.87E-12	3.62E-12	0.935
45 DEGREE CAVITY			
Cycle 15	3.43E-12	3.49E-12	1.015
Cycle 16	3.11E-12	3.17E-12	1.018
Cycle 17	3.23E-12	2.88E-12	0.893
Cycle 18/20	3.32E-12	3.09E-12	0.931
AVERAGE M/C BIAS FACTOR (K)			0.959
STANDARD DEVIATION (1σ)			0.120

TABLE 7.2-1

COMPARISON OF MEASURED AND CALCULATED NEUTRON SENSOR REACTION RATES
FROM SURVEILLANCE CAPSULE AND CAVITY DOSIMETRY IRRADIATIONS

	Cu63(n,α)	Ti46(n,p)	Fe54(n,p)	Ni58(n,p)	U238(n,f)	Np237(n,f)
INTERNAL CAPSULES						
V (13 DEGREES)	0.910		0.923		1.108	1.125
T (23 DEGREES)	0.947		0.984	0.950	1.007	1.129
R (13 DEGREES)	0.987		0.937	0.957	1.044	1.117
S (33 DEGREES)	1.070			0.883	1.113	1.089
0 DEGREE CAVITY						
Cycle 15	0.973	1.012	0.887	0.896	0.865	0.930
Cycle 16	0.964	0.977	0.863	0.863	0.823	0.923
Cycle 17	1.009	1.000	0.898	0.880	0.768	0.949
Cycle 18/20	1.017	1.038	0.882	0.894	0.904	0.832
15 DEGREE CAVITY						
Cycle 15	1.081	1.118	0.965	0.953	0.942	1.016
Cycle 16	1.011	1.018	0.882	0.889	0.908	0.901
Cycle 17	1.072	1.063	0.942	0.915	0.899	0.853
Cycle 18/20	1.071	1.055	0.899	0.898	0.959	0.887
30 DEGREE CAVITY						
Cycle 15	1.015	1.056	0.961	0.917	0.926	0.966
Cycle 16	1.074	1.066	0.947	0.936	0.834	0.929
Cycle 17	1.059	1.083	0.945	0.932	0.875	0.927
Cycle 18/20	1.086	1.082	0.913	0.912	0.939	0.917
45 DEGREE CAVITY						
Cycle 15	1.069	1.116	0.926	0.949	0.976	1.052
Cycle 16	1.093	1.067	0.972	0.967	0.945	1.067
Cycle 17	1.110	1.099	0.970	0.959	0.874	0.893
Cycle 18/20	1.081	1.039	0.911	0.923	0.963	0.924
AVERAGE	1.034	1.055	0.925	0.921	0.933	0.972
ST. DEV. (1σ)	0.055	0.042	0.032	0.033	0.086	0.091

OVERALL AVERAGE M/C RATIO	0.974
STANDARD DEVIATION (1σ)	0.081

SECTION 8.0

BEST ESTIMATE NEUTRON EXPOSURE OF PRESSURE VESSEL MATERIALS

In this section the measurement results provided in Sections 5.0 and 6.0 are combined with the results of the neutron transport calculations described in Section 4.0 to establish a mapping of the best estimate neutron exposure of the beltline region of the Point Beach Unit 2 reactor pressure vessel through the completion of Cycle 20. Based on the continued use of the Cycles 16-20 fuel loading patterns incorporating part length hafnium absorbers, projections of future vessel exposure to 32 and 48 effective full power years of operation are also provided. In addition to the spatial mapping over the beltline region, data pertinent to the maximum exposure experienced by the upper, intermediate, and lower shell forgings as well as the beltline circumferential welds are highlighted.

8.1 Exposure Distributions Within the Beltline Region

As described in Section 3.3 of this report, the best estimate vessel exposure was determined from the following relationship:

$$\Phi_{Best Est.} = K \, \Phi_{Calc.}$$

where: $\Phi_{Best\ Est.}$ = The best estimate fast neutron exposure at the location of interest.

K = The plant specific measurement/calculation (M/C) bias factor derived from all available surveillance capsule and reactor cavity dosimetry data.

$\Phi_{Calc.}$ = The absolute calculated fast neutron exposure at the location of interest.

From the data provided in Table 7.1-1, the plant specific bias factors (K) to be applied to the calculated exposure values given in Section 4.2 were as follows:

$\Phi(E > 1.0\ MeV)$	0.921 ± 0.074 (8.0%)
$\Phi(E > 0.1\ MeV)$	0.975 ± 0.161 (16.5%)
dpa	0.959 ± 0.120 (12.5%)

These bias factors were based on the results of the continuous monitoring program at Point Beach Unit 2 that has provide measured data from four internal surveillance capsules and sixteen reactor cavity sensor sets through the first 18.2 effective full power years of operation.

The uncertainties listed with the individual bias factors are at the 1σ level and are given on an absolute and percentage basis. Additional uncertainties associated with the evaluation of the best estimate vessel exposure are discussed in Section 8.2.

8.1.1 Exposure Accrued During Cycles 1 through 20

To assess the incremental exposure resulting from irradiation during Cycles 1 through 20, the bias factors listed in Section 8.1 were applied directly to the calculated values from Section 4.2 for the vessel clad/base metal interface to produce best estimate fluence levels characteristic of the midplane of the reactor core. The axial gradient chain measurements taken before and after implementation of part length hafnium were then employed to develop the complete axial traverse along the vessel wall. The best estimate results applicable to the vessel inner surface are incorporated into Tables 8.1-1 through 8.1-12 to establish the exposure accrued by the reactor vessel through the end of Cycles 14, 15 and 20, respectively.

Exposure distributions through the vessel wall, can be developed using these surface exposures and radial distribution functions from Section 4.0. This exposure information, applicable through the end of Cycle 20, was derived from an extensive set of measurements and assures that embrittlement gradients can be established with a minimum uncertainty. Further, as the monitoring program continues and additional data become available, the overall plant specific data base for Point Beach Unit 2 will expand resulting in reduced uncertainties and an improved accuracy in the assessment of vessel condition.

8.1.2 Projection of Future Vessel Exposure

At the end of Cycle 20, the Point Beach Unit 2 reactor had accrued 18.2 effective full power years (EFPY) of operation. In order to establish a framework for the assessment of future vessel condition, exposure projections to 32 and 48 EFPY are also included in Tables 8.1-1 through 8.1-12 in addition to the plant specific exposure assessments through the end of Cycle 20.

These extrapolations into the future were based on the assumption that the data averaged over the Cycles 16 through 20 irradiations were representative of all future fuel cycles. That is, that future fuel designs would incorporate the low leakage fuel management concept employed during Cycles 16 through 20. Examination of these projected exposure levels establishes the long term effectiveness of the low leakage fuel management incorporated to date and can be used as a guide in assessing strategies for future vessel exposure management. The validity of these projections for future operation will be confirmed via the continued cavity monitoring program.

TABLE 8.1-1

SUMMARY OF BEST ESTIMATE FAST NEUTRON (E > 1.0 MeV) EXPOSURE PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL
0 DEGREE AZIMUTHAL ANGLE

	Φ(E > 1.0 MeV) [n/cm^2]				
Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	3.15E+18	3.29E+18	4.02E+18	6.40E+18	9.15E+18
+5.5	7.67E+18	8.02E+18	9.84E+18	1.58E+19	2.26E+19
+4.5	1.12E+19	1.17E+19	1.46E+19	2.40E+19	3.49E+19
+3.5	1.32E+19	1.38E+19	1.70E+19	2.75E+19	3.97E+19
+2.5	1.47E+19	1.53E+19	1.88E+19	3.01E+19	4.32E+19
+1.5	1.45E+19	1.51E+19	1.85E+19	2.94E+19	4.20E+19
+0.5	1.41E+19	1.47E+19	1.77E+19	2.76E+19	3.90E+19
0.0	1.44E+19	1.50E+19	1.78E+19	2.68E+19	3.72E+19
-0.5	1.47E+19	1.53E+19	1.79E+19	2.60E+19	3.55E+19
-1.5	1.35E+19	1.41E+19	1.65E+19	2.45E+19	3.37E+19
-2.5	1.29E+19	1.34E+19	1.61E+19	2.46E+19	3.44E+19
-3.5	1.27E+19	1.32E+19	1.61E+19	2.52E+19	3.59E+19
-4.5	1.13E+19	1.18E+19	1.45E+19	2.33E+19	3.35E+19
-5.5	7.64E+18	7.99E+18	9.80E+18	1.57E+19	2.25E+19
-6.5	2.81E+18	2.94E+18	3.62E+18	5.85E+18	8.42E+18

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-2

SUMMARY OF BEST ESTIMATE FAST NEUTRON (E > 1.0 MeV) EXPOSURE PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL 15 DEGREE AZIMUTHAL ANGLE

	Φ(E > 1.0 MeV) [n/cm^2]				
Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	1.76E+18	1.85E+18	2.31E+18	3.79E+18	5.52E+18
+5.5	4.87E+18	5.11E+18	6.30E+18	1.01E+19	1.46E+19
+4.5	7.40E+18	7.78E+18	9.66E+18	1.58E+19	2.28E+19
+3.5	8.21E+18	8.63E+18	1.08E+19	1.77E+19	2.57E+19
+2.5	8.79E+18	9.24E+18	1.16E+19	1.91E+19	2.78E+19
+1.5	8.72E+18	9.16E+18	1.14E+19	1.85E+19	2.67E+19
+0.5	8.79E+18	9.24E+18	1.13E+19	1.80E+19	2.58E+19
0.0	9.09E+18	9.55E+18	1.15E+19	1.79E+19	2.54E+19
-0.5	9.38E+18	9.86E+18	1.17E+19	1.78E+19	2.49E+19
-1.5	9.08E+18	9.55E+18	1.14E+19	1.72E+19	2.40E+19
-2.5	9.01E+18	9.47E+18	1.14E+19	1.78E+19	2.52E+19
-3.5	8.43E+18	8.86E+18	1.09E+19	1.76E+19	2.54E+19
-4.5	7.69E+18	8.09E+18	9.97E+18	1.61E+19	2.32E+19
-5.5	4.94E+18	5.19E+18	6.41E+18	1.04E+19	1.50E+19
-6.5	1.89E+18	1.99E+18	2.46E+18	3.99E+18	5.76E+18

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-3

SUMMARY OF BEST ESTIMATE FAST NEUTRON (E > 1.0 MeV) EXPOSURE PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL 30 DEGREE AZIMUTHAL ANGLE

Φ(E > 1.0 MeV) [n/cm^2]

Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	1.50E+18	1.58E+18	1.95E+18	3.15E+18	4.55E+18
+5.5	3.60E+18	3.79E+18	4.68E+18	7.56E+18	1.09E+19
+4.5	5.56E+18	5.85E+18	7.23E+18	1.17E+19	1.69E+19
+3.5	6.29E+18	6.62E+18	8.17E+18	1.32E+19	1.91E+19
+2.5	6.43E+18	6.77E+18	8.36E+18	1.35E+19	1.95E+19
+1.5	6.25E+18	6.59E+18	8.13E+18	1.31E+19	1.90E+19
+0.5	6.57E+18	6.92E+18	8.55E+18	1.38E+19	1.99E+19
0.0	6.59E+18	6.94E+18	8.57E+18	1.39E+19	2.00E+19
-0.5	6.61E+18	6.97E+18	8.60E+18	1.39E+19	2.01E+19
-1.5	6.50E+18	6.85E+18	8.45E+18	1.37E+19	1.97E+19
-2.5	6.48E+18	6.83E+18	8.43E+18	1.36E+19	1.96E+19
-3.5	6.09E+18	6.42E+18	7.92E+18	1.28E+19	1.85E+19
-4.5	5.25E+18	5.53E+18	6.83E+18	1.10E+19	1.59E+19
-5.5	3.40E+18	3.58E+18	4.42E+18	7.15E+18	1.03E+19
-6.5	1.38E+18	1.45E+18	1.79E+18	2.90E+18	4.18E+18

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-4

SUMMARY OF BEST ESTIMATE FAST NEUTRON (E > 1.0 MeV) EXPOSURE PROJECTIONS
FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL
45 DEGREE AZIMUTHAL ANGLE

	Φ(E > 1.0 MeV) [n/cm^2]				
Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	1.24E+18	1.30E+18	1.62E+18	2.67E+18	3.88E+18
+5.5	2.94E+18	3.08E+18	3.85E+18	6.33E+18	9.21E+18
+4.5	4.66E+18	4.89E+18	6.10E+18	1.00E+19	1.46E+19
+3.5	5.10E+18	5.35E+18	6.67E+18	1.10E+19	1.60E+19
+2.5	5.79E+18	6.07E+18	7.58E+18	1.25E+19	1.81E+19
+1.5	5.56E+18	5.84E+18	7.29E+18	1.20E+19	1.74E+19
+0.5	5.57E+18	5.84E+18	7.29E+18	1.20E+19	1.74E+19
0.0	5.69E+18	5.96E+18	7.45E+18	1.23E+19	1.78E+19
-0.5	5.81E+18	6.09E+18	7.61E+18	1.25E+19	1.82E+19
-1.5	5.60E+18	5.87E+18	7.33E+18	1.21E+19	1.75E+19
-2.5	5.56E+18	5.83E+18	7.28E+18	1.20E+19	1.74E+19
-3.5	5.33E+18	5.59E+18	6.98E+18	1.15E+19	1.67E+19
-4.5	4.78E+18	5.02E+18	6.26E+18	1.03E+19	1.50E+19
-5.5	3.14E+18	3.30E+18	4.12E+18	6.78E+18	9.86E+18
-6.5	1.28E+18	1.35E+18	1.68E+18	2.77E+18	4.02E+18

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-5

SUMMARY OF BEST ESTIMATE FAST NEUTRON (E > 0.1 MeV) EXPOSURE PROJECTIONS
FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL
0 DEGREE AZIMUTHAL ANGLE

	Φ(E > 0.1 MeV) [n/cm^2]				
Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	9.13E+18	9.54E+18	1.17E+19	1.86E+19	2.66E+19
+5.5	2.23E+19	2.33E+19	2.85E+19	4.57E+19	6.56E+19
+4.5	3.26E+19	3.41E+19	4.24E+19	6.97E+19	1.01E+20
+3.5	3.82E+19	3.99E+19	4.93E+19	7.98E+19	1.15E+20
+2.5	4.26E+19	4.45E+19	5.46E+19	8.73E+19	1.25E+20
+1.5	4.20E+19	4.39E+19	5.36E+19	8.52E+19	1.22E+20
+0.5	4.08E+19	4.27E+19	5.14E+19	8.00E+19	1.13E+20
0.0	4.17E+19	4.36E+19	5.16E+19	7.78E+19	1.08E+20
-0.5	4.26E+19	4.45E+19	5.18E+19	7.55E+19	1.03E+20
-1.5	3.90E+19	4.08E+19	4.79E+19	7.10E+19	9.77E+19
-2.5	3.73E+19	3.90E+19	4.66E+19	7.12E+19	9.98E+19
-3.5	3.67E+19	3.84E+19	4.66E+19	7.32E+19	1.04E+20
-4.5	3.29E+19	3.44E+19	4.22E+19	6.76E+19	9.71E+19
-5.5	2.22E+19	2.32E+19	2.84E+19	4.55E+19	6.53E+19
-6.5	8.16E+18	8.53E+18	1.05E+19	1.70E+19	2.44E+19

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-6

SUMMARY OF BEST ESTIMATE FAST NEUTRON (E > 0.1 MeV) EXPOSURE PROJECTIONS
FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL
15 DEGREE AZIMUTHAL ANGLE

	Φ(E > 0.1 MeV) [n/cm^2]				
Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	5.42E+18	5.69E+18	7.11E+18	1.17E+19	1.70E+19
+5.5	1.50E+19	1.58E+19	1.94E+19	3.13E+19	4.50E+19
+4.5	2.28E+19	2.40E+19	2.97E+19	4.85E+19	7.03E+19
+3.5	2.53E+19	2.66E+19	3.31E+19	5.45E+19	7.92E+19
+2.5	2.71E+19	2.85E+19	3.56E+19	5.88E+19	8.57E+19
+1.5	2.69E+19	2.82E+19	3.50E+19	5.69E+19	8.22E+19
+0.5	2.71E+19	2.85E+19	3.49E+19	5.56E+19	7.96E+19
0.0	2.80E+19	2.94E+19	3.55E+19	5.53E+19	7.81E+19
-0.5	2.89E+19	3.04E+19	3.61E+19	5.49E+19	7.67E+19
-1.5	2.80E+19	2.94E+19	3.50E+19	5.31E+19	7.41E+19
-2.5	2.78E+19	2.92E+19	3.52E+19	5.49E+19	7.77E+19
-3.5	2.60E+19	2.73E+19	3.37E+19	5.44E+19	7.83E+19
-4.5	2.37E+19	2.49E+19	3.07E+19	4.96E+19	7.14E+19
-5.5	1.52E+19	1.60E+19	1.98E+19	3.20E+19	4.61E+19
-6.5	5.82E+18	6.12E+18	7.57E+18	1.23E+19	1.77E+19

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-7

SUMMARY OF BEST ESTIMATE FAST NEUTRON (E > 0.1 MeV) EXPOSURE PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL 30 DEGREE AZIMUTHAL ANGLE

	Φ(E > 0.1 MeV) [n/cm^2]				
Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	4.34E+18	4.57E+18	5.64E+18	9.12E+18	1.31E+19
+5.5	1.04E+19	1.09E+19	1.35E+19	2.18E+19	3.15E+19
+4.5	1.61E+19	1.69E+19	2.09E+19	3.36E+19	4.87E+19
+3.5	1.82E+19	1.91E+19	2.36E+19	3.82E+19	5.51E+19
+2.5	1.86E+19	1.96E+19	2.42E+19	3.91E+19	5.63E+19
+1.5	1.81E+19	1.90E+19	2.35E+19	3.80E+19	5.48E+19
+0.5	1.90E+19	2.00E+19	2.47E+19	4.00E+19	5.76E+19
0.0	1.91E+19	2.01E+19	2.48E+19	4.01E+19	5.78E+19
-0.5	1.91E+19	2.01E+19	2.49E+19	4.02E+19	5.80E+19
-1.5	1.88E+19	1.98E+19	2.44E+19	3.95E+19	5.70E+19
-2.5	1.87E+19	1.97E+19	2.44E+19	3.94E+19	5.68E+19
-3.5	1.76E+19	1.85E+19	2.29E+19	3.70E+19	5.34E+19
-4.5	1.52E+19	1.60E+19	1.97E+19	3.19E+19	4.60E+19
-5.5	9.82E+18	1.03E+19	1.28E+19	2.07E+19	2.98E+19
-6.5	3.99E+18	4.20E+18	5.18E+18	8.38E+18	1.21E+19

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-8

SUMMARY OF BEST ESTIMATE FAST NEUTRON (E > 0.1 MeV) EXPOSURE PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL 45 DEGREE AZIMUTHAL ANGLE

Φ(E > 0.1 MeV) [n/cm^2]

Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	3.44E+18	3.61E+18	4.51E+18	7.43E+18	1.08E+19
+5.5	8.18E+18	8.58E+18	1.07E+19	1.76E+19	2.57E+19
+4.5	1.30E+19	1.36E+19	1.70E+19	2.80E+19	4.07E+19
+3.5	1.42E+19	1.49E+19	1.86E+19	3.06E+19	4.45E+19
+2.5	1.61E+19	1.69E+19	2.11E+19	3.47E+19	5.05E+19
+1.5	1.55E+19	1.63E+19	2.03E+19	3.34E+19	4.86E+19
+0.5	1.55E+19	1.63E+19	2.03E+19	3.34E+19	4.86E+19
0.0	1.58E+19	1.66E+19	2.07E+19	3.41E+19	4.96E+19
-0.5	1.62E+19	1.70E+19	2.12E+19	3.49E+19	5.07E+19
-1.5	1.56E+19	1.63E+19	2.04E+19	3.36E+19	4.88E+19
-2.5	1.55E+19	1.62E+19	2.03E+19	3.34E+19	4.85E+19
-3.5	1.48E+19	1.56E+19	1.94E+19	3.20E+19	4.65E+19
-4.5	1.33E+19	1.40E+19	1.74E+19	2.87E+19	4.17E+19
-5.5	8.75E+18	9.18E+18	1.15E+19	1.89E+19	2.74E+19
-6.5	3.57E+18	3.75E+18	4.68E+18	7.70E+18	1.12E+19

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-9

SUMMARY OF BEST ESTIMATE IRON ATOM DISPLACEMENT [dpa] EXPOSURE PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL 0 DEGREE AZIMUTHAL ANGLE

	DISPLACEMENTS [dpa]				
Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	5.34E-03	5.58E-03	6.83E-03	1.09E-02	1.55E-02
+5.5	1.30E-02	1.36E-02	1.67E-02	2.68E-02	3.84E-02
+4.5	1.91E-02	1.99E-02	2.48E-02	4.08E-02	5.92E-02
+3.5	2.23E-02	2.33E-02	2.88E-02	4.67E-02	6.74E-02
+2.5	2.49E-02	2.60E-02	3.19E-02	5.11E-02	7.33E-02
+1.5	2.46E-02	2.57E-02	3.14E-02	4.99E-02	7.13E-02
+0.5	2.39E-02	2.50E-02	3.01E-02	4.68E-02	6.61E-02
0.0	2.44E-02	2.55E-02	3.02E-02	4.55E-02	6.32E-02
-0.5	2.49E-02	2.60E-02	3.03E-02	4.42E-02	6.03E-02
-1.5	2.28E-02	2.39E-02	2.80E-02	4.15E-02	5.72E-02
-2.5	2.18E-02	2.28E-02	2.72E-02	4.17E-02	5.84E-02
-3.5	2.15E-02	2.24E-02	2.72E-02	4.29E-02	6.09E-02
-4.5	1.92E-02	2.01E-02	2.47E-02	3.96E-02	5.68E-02
-5.5	1.30E-02	1.36E-02	1.66E-02	2.66E-02	3.82E-02
-6.5	4.78E-03	4.99E-03	6.15E-03	9.92E-03	1.43E-02

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-10

SUMMARY OF BEST ESTIMATE IRON ATOM DISPLACEMENT [dpa] EXPOSURE PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL 15 DEGREE AZIMUTHAL ANGLE

	DISPLACEMENTS [dpa]				
Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	3.08E-03	3.23E-03	4.03E-03	6.64E-03	9.65E-03
+5.5	8.51E-03	8.95E-03	1.10E-02	1.77E-02	2.55E-02
+4.5	1.29E-02	1.36E-02	1.69E-02	2.76E-02	3.99E-02
+3.5	1.44E-02	1.51E-02	1.88E-02	3.09E-02	4.50E-02
+2.5	1.54E-02	1.62E-02	2.02E-02	3.34E-02	4.87E-02
+1.5	1.53E-02	1.60E-02	1.99E-02	3.23E-02	4.67E-02
+0.5	1.54E-02	1.62E-02	1.98E-02	3.16E-02	4.52E-02
0.0	1.59E-02	1.67E-02	2.02E-02	3.14E-02	4.44E-02
-0.5	1.64E-02	1.72E-02	2.05E-02	3.12E-02	4.35E-02
-1.5	1.59E-02	1.67E-02	1.99E-02	3.01E-02	4.21E-02
-2.5	1.58E-02	1.66E-02	2.00E-02	3.12E-02	4.41E-02
-3.5	1.47E-02	1.55E-02	1.91E-02	3.09E-02	4.45E-02
-4.5	1.35E-02	1.41E-02	1.74E-02	2.82E-02	4.05E-02
-5.5	8.64E-03	9.08E-03	1.12E-02	1.82E-02	2.62E-02
-6.5	3.31E-03	3.48E-03	4.30E-03	6.98E-03	1.01E-02

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-11

SUMMARY OF BEST ESTIMATE IRON ATOM DISPLACEMENT [dpa] EXPOSURE PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL 30 DEGREE AZIMUTHAL ANGLE

DISPLACEMENTS [dpa]

Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	2.56E-03	2.70E-03	3.33E-03	5.39E-03	7.77E-03
+5.5	6.14E-03	6.47E-03	7.98E-03	1.29E-02	1.86E-02
+4.5	9.49E-03	1.00E-02	1.23E-02	2.00E-02	2.88E-02
+3.5	1.07E-02	1.13E-02	1.40E-02	2.26E-02	3.25E-02
+2.5	1.10E-02	1.16E-02	1.43E-02	2.31E-02	3.33E-02
+1.5	1.07E-02	1.12E-02	1.39E-02	2.25E-02	3.24E-02
+0.5	1.12E-02	1.18E-02	1.46E-02	2.36E-02	3.40E-02
0.0	1.13E-02	1.19E-02	1.46E-02	2.37E-02	3.41E-02
-0.5	1.13E-02	1.19E-02	1.47E-02	[illegible]	3.42E-02
-1.5	1.11E-02	1.17E-02	1.44E-02	[illegible]	3.37E-02
-2.5	1.11E-02	1.17E-02	1.44E-02	[illegible]	3.36E-02
-3.5	1.04E-02	1.10E-02	1.35E-02	[illegible]	3.15E-02
-4.5	8.96E-03	9.44E-03	1.17E-02	[illegible]	2.72E-02
-5.5	5.80E-03	6.11E-03	7.55E-03	[illegible]	1.76E-02
-6.5	2.36E-03	2.48E-03	3.06E-03	[illegible]	7.14E-03

Note: Axial location is provided relative to the axial midplane of the reactor core

TABLE 8.1-12

SUMMARY OF BEST ESTIMATE IRON ATOM DISPLACEMENT [dpa] EXPOSURE PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 2 REACTOR PRESSURE VESSEL 45 DEGREE AZIMUTHAL ANGLE

DISPLACEMENTS [dpa]

Z(ft)	EOC 14	EOC 15	EOC 20	32 EFPY	48 EFPY
+6.5	2.09E-03	2.19E-03	2.73E-03	4.50E-03	6.54E-03
+5.5	4.96E-03	5.20E-03	6.49E-03	1.07E-02	1.55E-02
+4.5	7.86E-03	8.25E-03	1.03E-02	1.60E-02	2.46E-02
+3.5	8.60E-03	9.02E-03	1.13E-02	1.85E-02	2.69E-02
+2.5	9.76E-03	1.02E-02	1.28E-02	2.10E-02	3.06E-02
+1.5	9.39E-03	9.85E-03	1.23E-02	2.02E-02	2.94E-02
+0.5	9.39E-03	9.85E-03	1.23E-02	2.02E-02	2.94E-02
0.0	9.59E-03	1.01E-02	1.26E-02	2.07E-02	3.01E-02
-0.5	9.80E-03	1.03E-02	1.28E-02	2.11E-02	3.07E-02
-1.5	9.44E-03	9.90E-03	1.24E-02	2.03E-02	2.96E-02
-2.5	9.38E-03	9.84E-03	1.23E-02	2.02E-02	2.94E-02
-3.5	8.99E-03	9.43E-03	1.18E-02	1.94E-02	2.82E-02
-4.5	8.06E-03	8.46E-03	1.06E-02	1.74E-02	2.53E-02
-5.5	5.30E-03	5.56E-03	6.94E-03	1.14E-02	1.66E-02
-6.5	2.27E-03	2.27E-03	2.83E-03	4.66E-03	6.78E-03

Note: Axial location is provided relative to the axial midplane of the reactor core

8.2 Exposure of Specific Beltline Materials

As shown in Figure 2.1-2, the beltline region of the Point Beach Unit 2 reactor pressure vessel is comprised of an intermediate shell forging (123V500), a lower shell forging (122W195), a circumferential weld (SA-1484) joining these two ring forgings, and a circumferential weld (Heat No. 21935) joining the intermediate shell forging to the nozzle shell course. The lower circumferential weld is centered 15.06 inches below the axial midplane of the active core; while the intermediate shell forging extends upward to an elevation 8.44 inches above the active fuel and the lower shell forging extends downward to an elevation 39.87 inches below the bottom of the active fuel. The maximum neutron exposure experienced by each of these beltline materials can be extracted from the data provided in Tables 8.1-1 through 8.1-12.

8.2-1 Circumferential Weld (SA-1484)

The current (End of Cycle 20) and projected maximum exposures of the beltline circumferential weld are listed in Table 8.2-1 and illustrated graphically in Figures 8.2-1 through 8.2-3. In this table and the accompanying figures, the weld exposure is expressed in terms of $\phi(E > 1.0\ \text{MeV})$, $\phi(E > 0.1\ \text{MeV})$, and dpa.

In developing the exposure profiles for the circumferential weld, it is noted that, although the flux reduction afforded by the Cycle 16-20 fuel loading patterns with part length hafnium absorbers has lessened the exposure rates within the 0-15 degree azimuthal sector, the maximum exposure point on the weld remains at the 0 degree azimuth throughout the service life of the unit. However, the magnitude of the projected exposures are significantly lower than would be the case had the flux reduction measures not been implemented.

8.2-2 Intermediate Shell Forging (123V500)

The current and projected maximum exposures of the intermediate shell forging are given in Table 8.2-2. Again, all three exposure parameters are provided. In the case of the intermediate forging, it can be noted from Table 8.1-1, that, due to the introduction of the part length absorbers and the corresponding reduction in exposure rates in the vicinity of the circumferential weld, the axial location of the maximum exposure at the 0 and 15 degree azimuthal angles shifts from an elevation near core midplane to an elevation approximately 2.5 ft. above core midplane as the lifetime of the unit increases. Corresponding variations at the 30 and 45 degree azimuths are less evident. Since the maximum exposure point for the intermediate shell forging is variable due to the flux reduction measures, these values are not illustrated graphically, but are presented only in tabular form.

8.2-3 Lower Shell Forging (122W195)

The current and and projected exposures for the lower shell forging are listed in Table 8.2-3. As in the case of the intermediate forging, all three exposure parameters are tabulated. In the case of the lower shell forging, the part length absorbers cause the maximum exposure location at 0 and 15 degrees to shift from the top of the forging to a position 3.5 feet below the active core midplane. However, the absorbers have a negligible impact at the 30 and 45 degree azimuths, resulting in the maximum exposure location remaining at the top of the forging adjacent to the circumferential weld. Again, due to this shift in the maximum exposure elevation, the data applicable to the lower shell forging are not illustrated graphically, but, rather, are presented only in tabular form.

8.2-4 Upper to Intermediate Shell Weld (Heat No. 21935) and Upper Shell Forging

The current and and projected exposures for the upper to intermediate shell circumferential weld and the adjacent upper shell forging are listed in Table 8.2-4. These materials located above the reactor core are seen to be unaffected by the introduction of the part length hafnium absorbers.

TABLE 8.2-1

MAXIMUM FAST NEUTRON EXPOSURE OF THE POINT BEACH UNIT 2 BELTLINE CIRCUMFERENTIAL WELD (SA-1484)

AZIMUTHAL ANGLE	$\Phi(E > 1.0$ MeV) $[n/cm^2]$ 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	1.69E+19	2.49E+19	3.42E+19
15°	1.19E+19	1.74E+19	2.42E+19
30°	8.49E+18	1.38E+19	1.98E+19
45°	7.40E+18	1.22E+19	1.77E+19

AZIMUTHAL ANGLE	$\Phi(E > 0.1$ MeV) $[n/cm^2]$ 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	4.89E+19	7.21E+19	9.90E+19
15°	3.53E+19	5.36E+19	7.48E+19
30°	2.45E+19	3.97E+19	5.73E+19
45°	2.06E+19	3.39E+19	4.93E+19

AZIMUTHAL ANGLE	IRON DISPLACEMENTS [dpa] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	2.86E-02	4.22E-02	5.80E-02
15°	2.01E-02	3.04E-02	4.25E-02
30°	1.45E-02	2.34E-02	3.38E-02
45°	1.25E-02	2.05E-02	2.99E-02

TABLE 8.2-2

MAXIMUM FAST NEUTRON EXPOSURE OF THE POINT BEACH UNIT 2
INTERMEDIATE SHELL FORGING (123V500)

AZIMUTHAL ANGLE	Φ(E > 1.0 MeV) [n/cm^2] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	1.88E+19	3.01E+19	4.32E+19
15°	1.16E+19	1.91E+19	2.78E+19
30°	8.60E+18	1.39E+19	2.01E+19
45°	7.61E+18	1.25E+19	1.82E+19

AZIMUTHAL ANGLE	Φ(E > 0.1 MeV) [n/cm^2] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	5.46E+19	8.73E+19	1.25E+20
15°	3.61E+19	5.88E+19	8.57E+19
30°	2.49E+19	4.02E+19	5.80E+19
45°	2.12E+19	3.49E+19	5.07E+19

AZIMUTHAL ANGLE	IRON DISPLACEMENTS [dpa] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	3.19E-02	5.11E-02	7.33E-02
15°	2.05E-02	3.34E-02	4.87E-02
30°	1.47E-02	2.37E-02	3.42E-02
45°	1.28E-02	2.11E-02	3.07E-02

TABLE 8.2-3

MAXIMUM FAST NEUTRON EXPOSURE OF THE POINT BEACH UNIT 2
LOWER SHELL FORGING (122W195)

AZIMUTHAL ANGLE	Φ(E > 1.0 MeV) [n/cm²] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	1.69E+19	2.52E+19	3.59E+19
15°	1.19E+19	1.78E+19	2.54E+19
30°	8.49E+18	1.38E+19	1.98E+19
45°	7.40E+18	1.22E+19	1.77E+19

AZIMUTHAL ANGLE	Φ(E > 0.1 MeV) [n/cm²] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	4.89E+19	7.32E+19	1.04E+20
15°	3.53E+19	5.49E+19	7.83E+19
30°	2.45E+19	3.97E+19	5.73E+19
45°	2.06E+19	3.39E+19	4.93E+19

AZIMUTHAL ANGLE	IRON DISPLACEMENTS [dpa] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	2.86E-02	4.29E-02	6.09E-02
15°	2.01E-02	3.12E-02	4.45E-02
30°	1.45E-02	2.34E-02	3.38E-02
45°	1.25E-02	2.05E-02	2.99E-02

TABLE 8.2-4

MAXIMUM FAST NEUTRON EXPOSURE OF THE POINT BEACH UNIT 2 UPPER/INTERMEDIATE SHELL CIRCUMFERENTIAL WELD (Heat NO. 21935) AND THE UPPER SHELL FORGING

AZIMUTHAL ANGLE	Φ(E > 1.0 MeV) [n/cm^2] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	3.45E+18	5.48E+18	7.84E+18
15°	2.00E+18	3.29E+18	4.78E+18
30°	1.69E+18	2.74E+18	3.95E+18
45°	1.39E+18	2.29E+18	3.33E+18

AZIMUTHAL ANGLE	Φ(E > 0.1 MeV) [n/cm^2] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	1.00E+19	1.59E+19	2.27E+19
15°	6.16E+18	1.01E+19	1.47E+19
30°	4.90E+18	7.92E+18	1.14E+19
45°	3.87E+18	6.37E+18	9.27E+18

AZIMUTHAL ANGLE	IRON DISPLACEMENTS [dpa] 18.2 EFPY	32.0 EFPY	48.0 EFPY
0°	5.85E-03	9.31E-03	1.33E-02
15°	3.50E-03	5.75E-03	8.36E-03
30°	2.89E-03	4.68E-03	6.75E-03
45°	2.34E-03	3.86E-03	5.61E-03

FIGURE 8.2-1

FAST NEUTRON FLUENCE (E > 1.0 MeV) AS A FUNCTION OF AZIMUTHAL ANGLE AT THE INNER RADIUS OF THE BELTLINE CIRCUMFERENTIAL WELD

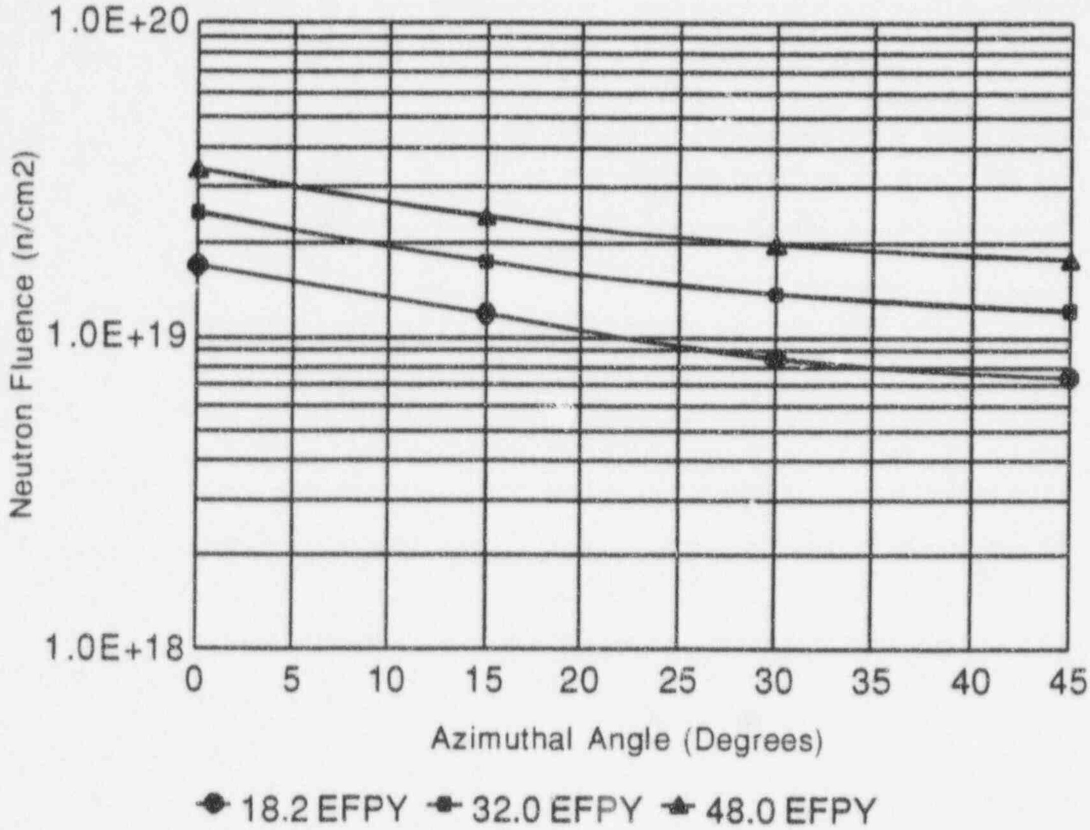


FIGURE 8.2-2

FAST NEUTRON FLUENCE (E > 0.1 MeV) AS A FUNCTION OF AZIMUTHAL ANGLE AT THE INNER RADIUS OF THE BELTLINE CIRCUMFERENTIAL WELD

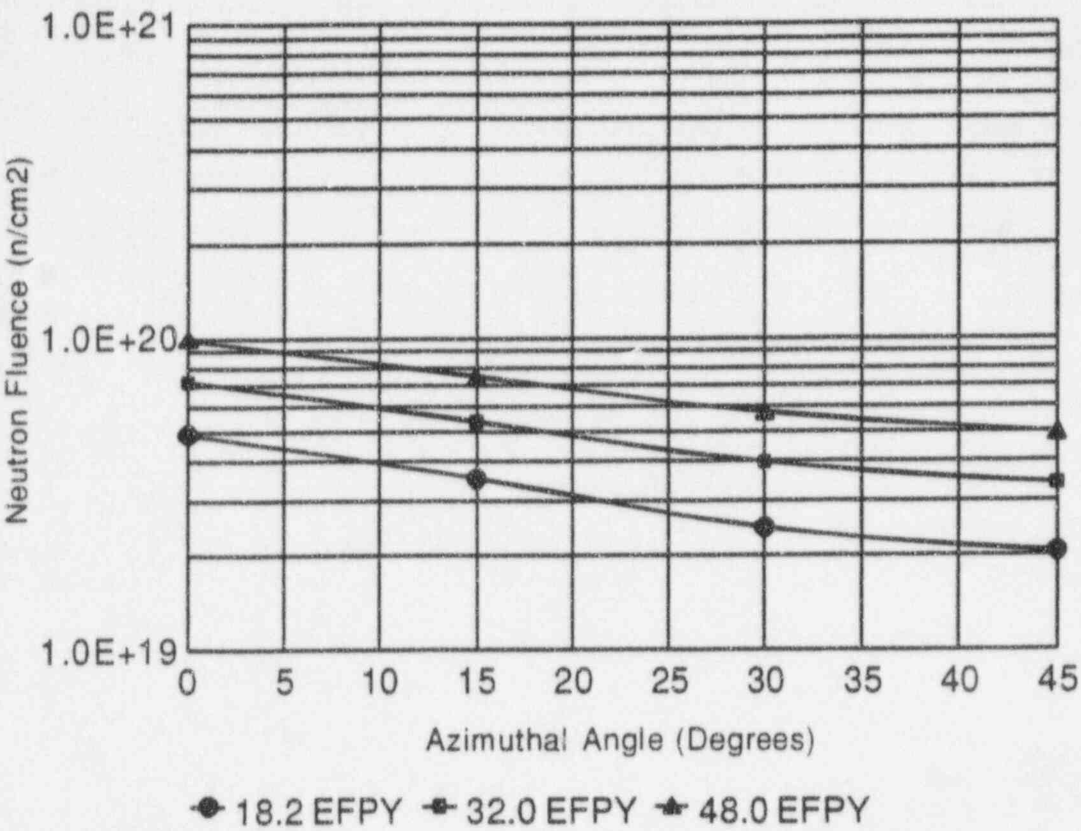
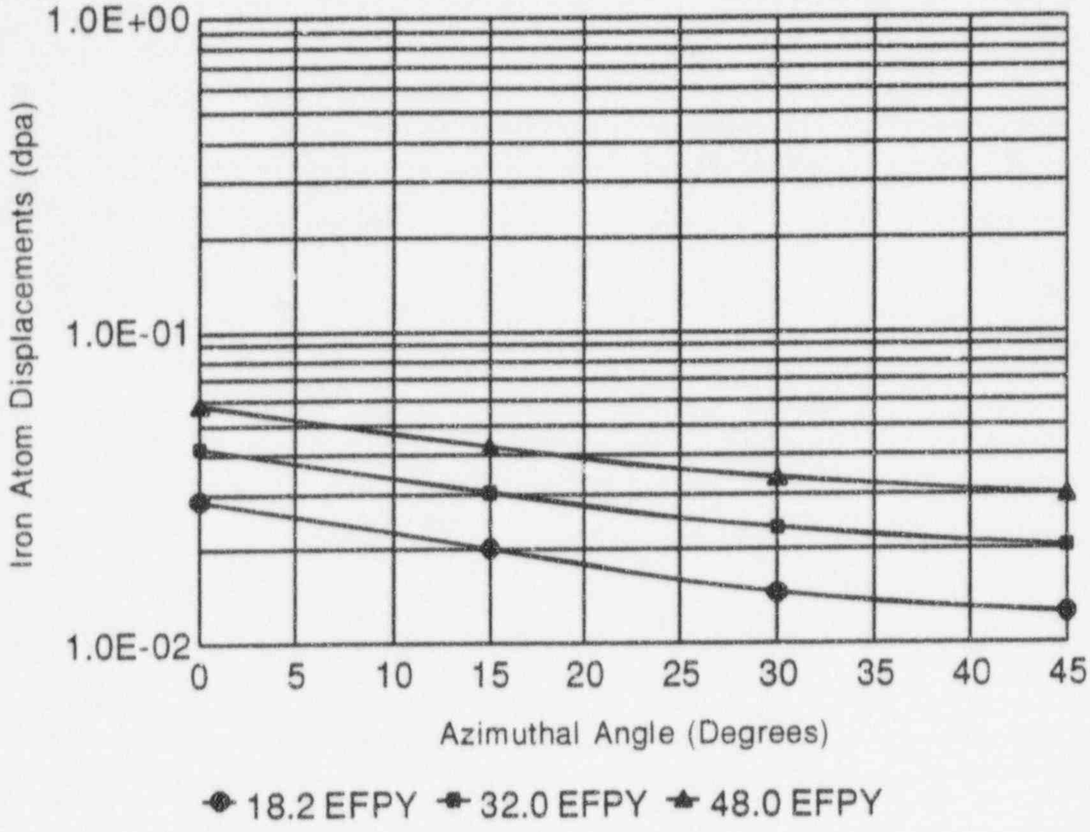


FIGURE 8.2-3

IRON ATOM DISPLACEMENTS [dpa] AS A FUNCTION OF AZIMUTHAL ANGLE AT THE INNER RADIUS OF THE BELTLINE CIRCUMFERENTIAL WELD



8.3 Uncertainties in Exposure Projections

The overall uncertainty in the best estimate exposure projections within the pressure vessel wall stem primarily from two sources;

1) the uncertainty in the bias factor (K) derived from the plant specific measurement data base; and

2) the analytical uncertainty associated with relating the results at the measurement locations to the desired results within the pressure vessel wall.

Uncertainty in the bias factor derives directly from the individual uncertainties in the measurement process, in the least squares adjustment procedure, and in the location of the surveillance capsule and cavity dosimetry sensor sets. The analytical uncertainty in the relationship between the exposure of the pressure vessel and the exposure at the measurement locations are based on the vessel thickness tolerance relative to the cavity data and on downcomer water density variations and vessel inner radius tolerance relative to the surveillance capsule data.

The 1σ uncertainties associated with the bias factors applicable to $\Phi(E > 1.0\ \text{MeV})$, $\Phi(E > 0.1\ \text{MeV})$, and dpa are given in Section 8.1 of this report. The additional information pertinent to the required analytical uncertainty for vessel locations has been obtained from benchmarking studies using the Westinghouse neutron transport methodology and from several comparisons of power reactor internal surveillance capsule dosimetry and reactor cavity dosimetry for which the irradiation history of all sensors was the same.

Based on these benchmarking evaluations the additional uncertainty associated with the tolerances in dosimetry positioning, vessel thickness, vessel inner radius and downcomer temperature was estimated to be approximately 6% for all exposure parameters. These uncertainty components were then combined as follows:

<u>1σ UNCERTAINTY</u>

	Φ(E > 1.0 MeV)	Φ(E > 0.1 MeV)	dpa
Bias Factor	8.0%	16.5%	12.5%
Analytical	6.0%	6.0%	6.0%
Combined	10.0%	17.6%	13.9%

Thus, the total uncertainty associated with the neutron exposure projections at the pressure vessel clad/base metal interface for Point Beach Unit 2 was estimated to be:

	1σ Uncertainty
Φ(E > 1.0 MeV)	10%
Φ(E > 0.1 MeV)	18%
dpa	14%

These uncertainty values are well within the 20% 1σ uncertainty in vessel fluence projections required by the PTS rule.

SECTION 9.0

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APPENDIX A

SPECIFIC ACTIVITIES AND IRRADIATION HISTORY OF SENSORS
FROM SURVEILLANCE CAPSULES V, T, R AND S

In this appendix, the irradiation history as extracted from NUREG-0020 and the measured specific activities of radiometric sensors irradiated in surveillance Capsules V, T, R and S are provided.

The irradiation history of capsules withdrawn to date was as follows:

CYCLE NO.	STARTUP	SHUTDOWN	COMMENT
1	05/30/72	10/17/74	CAPSULE V WITHDRAWN
2	12/20/74	02/26/76	
3	03/26/76	03/04/77	CAPSULE T WITHDRAWN
4	04/19/77	03/22/78	
5	04/17/78	03/23/79	CAPSULE R WITHDRAWN
6	04/19/79	04/11/80	
7	05/13/80	04/17/81	
8	05/21/81	04/16/82	
9	05/26/82	03/25/83	
10	07/05/83	09/28/84	
11	11/19/84	10/05/85	
12	11/24/85	09/27/86	
13	12/01/86	10/03/87	
14	11/17/87	10/08/88	
15	11/21/88	09/23/89	
16	11/24/89	10/06/90	CAPSULE S WITHDRAWN

REF. CORE POWER = 1518 MWt

The monthly thermal generation applicable to the Point Beach Unit 2 reactor is provided on pages A-2 and A-3. Pages A-4 through A-7 contain the measured specific activities ofsensors removed from Capsules T, R, and S.

MONTHLY THERMAL GENERATION DURING THE FIRST SIXTEEN FUEL CYCLES OF THE POINT BEACH UNIT 2 REACTOR

MONTH	THERMAL GENERATION (MW-hr)	MONTH	THERMAL GENERATION (MW-hr)	MONTH	THERMAL GENERATION (MW-hr)	MONTH	THERMAL GENERATION (MW-hr)
5/72-3/74	13508112	4/76	972825	5/78	1099610	6/80	1065238
4/74	1076568	5/76	956959	6/78	1044078	7/80	1114363
5/74	1111056	6/76	1007721	7/78	1040240	8/80	1124200
6/74	885000	7/76	1026604	8/78	1054425	9/80	1047595
7/74	954648	8/76	1022317	9/78	1059006	10/80	1112514
8/74	1111608	9/76	1005046	10/78	1104830	11/80	989299
9/74	1054224	10/76	1116240	11/78	1079867	12/80	1114432
10/74	557784	11/76	1064445	12/78	1074069	1/81	1115599
11/74	0	12/76	1102576	1/79	1116477	2/81	1008189
12/74	302016	1/77	1102848	2/79	1010047	3/81	1112552
1/75	1113456	2/77	1001354	3/79	746785	4/81	559549
2/75	881295	3/77	100292	4/79	585747	5/81	186873
3/75	1081779	4/77	214373	5/79	1071794	6/81	1047500
4/75	916898	5/77	1108075	6/79	1021650	7/81	1112509
5/75	880266	6/77	1066583	7/79	408680	8/81	1092410
6/75	914234	7/77	1072410	8/79	1114720	9/81	988920
7/75	1063799	8/77	973371	9/79	1032935	10/81	1088553
8/75	748416	9/77	1039145	10/79	1117434	11/81	1080909
9/75	997380	10/77	1109781	11/79	1045777	12/81	1073535
10/75	989176	11/77	1062668	12/79	1095273	1/82	1106250
11/75	974925	12/77	1106636	1/80	1111502	2/82	1005969
12/75	1114475	1/78	1066993	2/80	974189	3/82	1091347
1/76	1070693	2/78	1000903	3/80	613022	4/82	528202
2/76	929464	3/78	723452	4/80	328351	5/82	124874
3/76	115451	4/78	352782	5/80	539505	6/82	1074941

MONTHLY THERMAL GENERATION DURING THE FIRST SIXTEEN FUEL CYCLES
OF THE POINT BEACH UNIT 2 REACTOR

MONTH	THERMAL GENERATION (MW-hr)	MONTH	THERMAL GENERATION (MW-hr)	MONTH	THERMAL GENERATION (MW-hr)	MONTH	THERMAL GENERATION (MW-hr)
7/82	1117700	8/84	1126993	9/86	945202	10/88	229686
8/82	1052625	9/84	921284	10/86	0	11/88	229879
9/82	864026	10/84	0	11/86	10240	12/88	1113148
10/82	1094393	11/84	252959	12/86	1088521	1/89	1127540
11/82	1089345	12/84	1062541	1/87	1037881	2/89	1018557
12/82	1109078	1/85	1103110	2/87	1017595	3/89	1030496
1/83	1121499	2/85	1015875	3/87	1066066	4/89	1008555
2/83	1016200	3/85	1121631	4/87	1091705	5/89	1117762
3/83	869556	4/85	1089086	5/87	1031289	6/89	1089331
4/83	0	5/85	1125089	6/87	1090934	7/89	1126185
5/83	0	6/85	1081247	7/87	1120329	8/89	1078103
6/83	0	7/85	1104022	8/87	1033949	9/89	776161
7/83	883944	8/85	1125504	9/87	1092960	10/89	0
8/83	1075582	9/85	1090774	10/87	70587	11/89	141690
9/83	1087367	10/85	141568	11/87	382736	12/89	1123322
10/83	1115645	11/85	95661	12/87	1067563	1/90	1127322
11/83	1088861	12/85	1033331	1/88	1123573	2/90	1018430
12/83	1120529	1/86	1096374	2/88	1055151	3/90	1118045
1/84	1109752	2/86	1001623	3/88	1127373	4/90	1090485
2/84	1052908	3/86	1120825	4/88	949345	5/90	1127775
3/84	1125590	4/86	1037164	5/88	1127476	6/90	1080806
4/84	1068842	5/86	1110585	6/88	1091026	7/90	1128256
5/84	934923	6/86	1024233	7/88	1126029	8/90	1129766
6/84	1083029	7/86	1069310	8/88	1129392	9/90	1120491
7/84	1119374	8/86	1083054	9/88	1072861	10/90	162384

CHEMICAL ANALYSIS REPORT

WESTINGHOUSE ADVANCED REACTOR DIVISION

ANALYTICAL LABORATORIES
WALTZ MILL SITE

ANAL. SERV. REQUEST NO. 7317

ORIGINATOR	DEPT. & GRP.	EXT.	ROOM NO.	FACILITY
STAN ANDERSON				MONROEVILLE NUCLEAR CENTER

FINAL SIGNATURE	DATE RECEIVED	DATE REPORTED
C. O. Blackburn	3-15-78	4-7-78

METHOD	ANALYST	REFERENCE	METHOD	ANALYST	REFERENCE
Gamma Spectrometry	CAC	File			

RESULTS OF ANALYSIS

ORIGINATOR'S SAMPLE NO.	ANAL. SERV. LAB. NO.	DOSIMETER MATERIAL	Co-60	Co-58			
TOP-24	78-688	Al-Co	3.22×10^4		DPS / mg of wire MAR. 30, 1978		
" -26	78-689	Cu	91.1		↑		
TOP-MID-24	78-690	Al-Co	2.01×10^4				
" -25	78-691	Al-Co (Cd)	1.01×10^4				
" -26	78-692	Cu	82.3				
MID-24	78-693	Al-Co	2.09×10^4				
" -25	78-694	Al-Co (Cd)	1.07×10^4				
" -26	78-695	Ni		1.02×10^3			
BOTTOM MID -24	78-696	Al-Co	2.62×10^4				
" -25	78-697	Al-Co (Cd)	1.17×10^4				
" -26	78-698	Cu	95.0				
BOTTOM-24	78-699	Al-Co	2.48×10^4				
-25	78-700	Al-Co (Cd)	1.12×10^4				
-26	78-701	Cu	103		↓		
U-238 CAPSULE	78-702	369 DPS / mg U-238 } ON MAR. 30, 1978					
Np-237 "	78-703	2830 DPS / mg Np-237 }					

REMARKS:

CHARPY			Mn-54	
W-32	78-704	Fe	1.17×10^3	DPS/mg Fe MAR. 30, 1978
H-25	78-705	Fe	1.23×10^3	
E-47	78-706	Fe	1.44×10^3	
V-37	78-707	Fe	1.41×10^3	
R-30	78-708	Fe	1.11×10^3	
E-37	78-709	Fe	1.38×10^3	

POINT BEACH UNIT #2, "T" CAPSULE

CHECK COPIES FOR LEGIBILITY

CHEMICAL ANALYSIS REPORT
FORM 54741A

WESTINGHOUSE ADVANCED REACTOR DIVISION
ANALYTICAL LABORATORIES
WALTZ MILL SITE

ANAL. SERV. REQUEST NO. 8143

ORIGINATOR: S. L. ANDERSON | DEPT. & GRP.: | EXT.: | ROOM NO.: BAY 478 | FACILITY: MNC

APPROVAL SIGNATURE: C. A. Blackburn | DATE RECEIVED: 6-5-79 | DATE REPORTED: 6-22-79

METHOD	ANALYST	REFERENCE	METHOD	ANALYST	REFERENCE
Gamma Spectrometry	WTF	File			
	CAB				

RESULTS OF ANALYSIS

ORIGINATOR'S SAMPLE NO.	ANAL. SERV. LAB. NO.	Co-60	Co-58				
Top Co	79-1648	3.09 × 10⁴		dps/mg of wire		on 6-15-79	±[illegible]
Co(Cd)	1649	3.15 × 10⁴					
Cu	1650	196					
Top Mid Co	1651	5.62 × 10⁴					
Co(Cd)	1652	3.04 × 10⁴					
Cu	1653	177					
Mid. Co	1654	5.94 × 10⁴					
Co(Cd)	1655	2.93 × 10⁴					
Ni	1656	—	3.13 × 10⁴				
Bottom Mid Co	1657	*					
Co(Cd)	1658	3.49 × 10⁴					
Cu	1659	196					
Bottom Co	1660	7.88 × 10⁴					
Co(Cd)	1661	3.31 × 10⁴					
Cu	79-1662	208					

REMARKS:

U-238	79-1663	1.00 × 10³ dps/mg U-238 on 6-15-79 ±[illegible]
Np-237	79-1664	7.52 × 10³ dps/mg Np-237

POINT BEACH UNIT #2 "R" CAPSULE

* WIRE REPORTED NOT FOUND BY R&D HOT CELLS.

CHECK COPIES FOR LEGIBILITY

ORIGINATOR COPY

CHEMICAL ANALYSIS REPORT
FORM 5-741A

WESTINGHOUSE ADVANCED REACTOR DIVISION
ANALYTICAL LABORATORIES
WALTZ MILL SITE

ANAL. SERV. REQUEST NO. 8279

ORIGINATOR	DEPT. & SAP.	EXT.	ROOM NO.	FACILITY
S.L. ANDERSON		BAY 478		MNC

APPROVAL SIGNATURE	DATE RECEIVED	DATE REPORTED
C.A. Blackburn	8-27-79	8-30-79

METHOD	ANALYST	REFERENCE	METHOD	ANALYST	REFERENCE
Gamma Spectrometry	WTF Cal.	File			
Fe.	FRC				

RESULTS OF ANALYSIS

ORIGINATOR'S SAMPLE NO.	ANAL. SERV. LAB. NO.		Mn-54				
Charpy	Specimens		DPS/mg Fe	[illegible] Aug 29 1979		0830	± 10.7%
V-13	79-3450		3850				
E-23	3451		3560				
E-13	3452		3090				
H-9	3453		2900				
R-14	3454		3970				
W-16	79-3455		2870				

REMARKS:

POINT BEACH UNIT #2 "R" CAPSULE

(SEE REQUEST # 8143 FOR WIRES + FISSION CAPSULE RESULTS.)

CHECK COPIES FOR LEGIBILITY

ORIGINATOR COPY

Westinghouse Electric Corporation
Advanced Energy Systems - Analytical Laboratory
Waltz Mill Site

REPORT Request# 14340

TO: S.L.Anderson Energy Center - East(478)
E.Terek Energy Center - East(470)

Received: 5/7/91
Reported: 6/5/91

[RESULTS OF ANALYSIS]

Dosimetry: Point Beach Unit #2 Capsule S

Originator ID	Lab. Sample #	Dosimeter Material	Nuclide	(May 20, 1991) dpa/mg *		2 sigma
FISSION MONITORS						
U-238	91-1035	U-238	Cs-137	1.37E+03	+/-	2.3E+01
Np-237	91-1036	Np-237	Cs-137	8.55E+03	+/-	1.2E+02
TOP WIRES						
Co/Al(Cd)	91-1037	Co/Al	Co-60	1.93E+04	+/-	2.4E+02
Co/Al	91-1038	Co/Al	Co-60	4.13E+04	+/-	3.4E+02
Cu	91-1039	Cu	Co-60	1.83E+02	+/-	2.2E+00
TOP MID WIRES						
Co/Al(Cd)	91-1040	Co/Al	Co-60	1.86E+04	+/-	2.3E+02
Co/Al	91-1041	Co/Al	Co-60	3.39E+04	+/-	3.2E+02
Cu	91-1042	Cu	Co-60	1.53E+02	+/-	2.1E+00
MID WIRES						
Co/Al(Cd)	91-1043	Co/Al	Co-60	1.73E+04	+/-	2.2E+02
Co/Al	91-1044	Co/Al	Co-60	3.38E+04	+/-	2.8E+02
Ni	91-1045	Ni	Co-58	2.95E+03	+/-	3.8E+01
BOT MID WIRES						
Co/Al(Cd)	91-1046	Co/Al	Co-60	2.12E+04	+/-	2.2E+02
Co/Al	91-1047	Co/Al	Co-60	4.43E+04	+/-	3.3E+02
Cu	91-1048	Cu	Co-60	1.68E+02	+/-	2.1E+00
BOT WIRES						
Co/Al(Cd)	91-1049	Co/Al	Co-60	1.90E+04	+/-	2.1E+02
Co/Al	91-1050	Co/Al	Co-60	4.12E+04	+/-	3.2E+02
Cu	91-1051	Cu	Co-60	1.91E+02	+/-	2.7E+00

Remarks: * Results are in units of dpa/(mg of Dosimeter Material).

References: Request# 14340
Lab.Book#46 page 169,171.
Procedures: A-512,A-513,A-524.
Analyst: WIP, FRC

Approved: [signature]

Since surveillance Capsules T, R, and S were irradiated for multiple fuel cycles, the flux adjustment factors, C_j, defined in Section 3.0 were employed in the reaction rate calculations for the individual sensor sets.

The quantity C_j is defined as the calculated ratio of $\phi(E > 1.0 \text{ MeV})$ during the irradiation period j to the time weighted average $\phi(E > 1.0 \text{ MeV})$ over the entire irradiation period. The values of C_j used in the evaluation of the Point Beach Unit 2 surveillance capsules were as follows:

	FLUX ADJUSTMENT FACTOR C_j			
	CAPSULE V	CAPSULE T	CAPSULE R	CAPSULE S
CYCLE 1	1.000	0.973	1.008	1.080
CYCLE 2		1.029	1.022	1.179
CYCLE 3		1.010	1.008	1.168
CYCLE 4			0.955	1.116
CYCLE 5			1.008	1.119
CYCLE 6				1.155
CYCLE 7				0.971
CYCLE 8				0.926
CYCLE 9				0.938
CYCLE 10				0.961
CYCLE 11				0.949
CYCLE 12				0.905
CYCLE 13				0.859
CYCLE 14				0.934
CYCLE 15				0.856
CYCLE 16				0.753

APPENDIX B

MEASURED SPECIFIC ACTIVITY AND IRRADIATION HISTORY
OF REACTOR CAVITY SENSOR SETS - CYCLE 15

In this appendix, the irradiation history as extracted from NUREG-0020 and the measured specific activities of radiometric sensors irradiated in the reactor cavity during Cycle 15 are provided.

The irradiation history of Cycle 15 was as follows:

Cycle No.	STARTUP	SHUTDOWN	COMMENT
15	11/21/88	11/24/89	

Ref. Core Power = 1518 MWt

MONTH	THERMAL GENERATION (MW-Hr)
11/88	229879
12/88	1113148
1/89	1127540
2/89	1018557
3/89	1030496
4/89	1008555
5/89	1117762
6/89	1089331
7/89	1126185
8/89	1078103
9/89	776161
TOTAL	10715717

The irradiation capsule loading diagram and the measured specific activities of the radiometric monitors from the Cycle 15 irradiation are provided on pages B-2 through B-9. For the multiple foil sensor sets, the individual foil ID can be correlated with the capsule loading diagram provided in Section 6.1-1 in order to determine the location of the foil within the reactor cavity during irradiation.

CONTENTS OF MULTIPLE FOIL SENSOR SETS

CAPSULE ID and POSITION	BARE OR CADMIUM SHIELDED	RADIOMETRIC MONITOR ID Fe	Ni	Cu	Ti	Nb	Co	U-238	Np-237	SSTR PACKAGE
G-1	B	CG	--	--	--	--	BM	--	--	PB-6B
G-2	Cd	DG	AG	AG	G	M	DM	G	--	--
G-3	Cd	--	--	--	--	--	--	--	2	PB-6C
H-1	B	CH	--	--	--	--	BL	--	--	PB-16B
H-2	Cd	DH	AH	AH	H	L	DL	H	--	--
H-3	Cd	--	--	--	--	--	--	--	3	PB-16C
I-1	B	CI	--	--	--	--	BK	--	--	PB-7B
I-2	Cd	DI	AI	AI	I	K	DK	I	--	--
I-3	Cd	--	--	--	--	--	--	--	4	PB-7C
J-1	B	CJ	--	--	--	--	BI	--	--	PB-17B
J-2	Cd	DJ	AJ	AJ	J	I	DI	J	--	--
J-3	Cd	--	--	--	--	--	--	--	5	PB-17C
K-1	B	CK	--	--	--	--	BJ	--	--	PB-18B
K-2	Cd	DK	AK	AK	K	J	DJ	K	--	--
K-3	Cd	--	--	--	--	--	--	--	6	PB-18C
L-1	B	CL	--	--	--	--	BH	--	--	PB-19B
L-2	Cd	DL	AL	AL	L	H	DH	L	--	--
L-3	Cd	--	--	--	--	--	--	--	7	PB-19C
XX-1	B	CT	--	--	--	--	BA	--	--	PB-1B
XX-2	Cd	DT	AT	AT	T	A	DA	T	--	--
XX-3	Cd	--	--	--	--	--	--	--	1	PB-1C

Westinghouse Electric Corporation
Advanced Energy Systems - Analytical Laboratory
Waltz Mill Site

REPORT

Request# 13910

TO: A.H.Fero (W)NATD, Energy Center (East 4-17)

Received: 1/19/90
Reported: 2/20/90

[RESULTS OF ANALYSIS]
Point Beach Unit 2 Cycle 15 Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 2/1/90) dps/mg foil		2 sigma
AT	90-233	Ni	Co-58	10.5	+/-	0.1
AG	90-243	Ni	Co-58	94.9	+/-	1.1
AH	90-253	Ni	Co-58	245.4	+/-	3.1
AI	90-263	Ni	Co-58	93.0	+/-	1.4
AJ	90-273	Ni	Co-58	215.3	+/-	2.1
AK	90-283	Ni	Co-58	160.3	+/-	1.4
AL	90-293	Ni	Co-58	143.0	+/-	1.7
AT	90-234	Cu	Co-60	0.0233	+/-	0.0005
AG	90-244	Cu	Co-60	0.242	+/-	0.007
AH	90-254	Cu	Co-60	0.670	+/-	0.011
AI	90-264	Cu	Co-60	0.240	+/-	0.007
AJ	90-274	Cu	Co-60	0.618	+/-	0.011
AK	90-284	Cu	Co-60	0.467	+/-	0.010
AL	90-294	Cu	Co-60	0.448	+/-	0.007
T	90-235	Ti	Sc-46	0.190	+/-	0.011
G	90-245	Ti	Sc-46	2.00	+/-	0.05
H	90-255	Ti	Sc-46	5.05	+/-	0.15
I	90-265	Ti	Sc-46	1.98	+/-	0.07
J	90-275	Ti	Sc-46	4.64	+/-	0.12
K	90-285	Ti	Sc-46	3.49	+/-	0.11
L	90-295	Ti	Sc-46	3.30	+/-	0.08

Remarks:

AL File: 13910
References: Lab.Book#41p74. LB#35p213-216.
Procedures: A-524.
Analyst: WTF, WRM, CAB.

Approved: C.A. Blackburn 2-20-90

Westinghouse Electric Corporation
Advanced Energy Systems - Analytical Laboratory
Waltz Mill Site

REPORT

Request# 13910

TO: A.H.Fero (W)NATD, Energy Center (East 4-17)

Received: 1/19/90
Reported: 2/20/90

[RESULTS OF ANALYSIS]
Point Beach Unit 2 Cycle 15 Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 2/1/90) dps/mg foil		2 sigma
BA	90-237	AlCo	Co-60	65.1	+/-	1.1
DA	90-238	AlCo	Co-60	47.0	+/-	0.9
BM	90-247	AlCo	Co-60	178.8	+/-	4.6
DM	90-248	AlCo	Co-60	131.5	+/-	1.6
BL	90-257	AlCo	Co-60	525.6	+/-	8.3
DL	90-258	AlCo	Co-60	310.6	+/-	6.1
BK	90-267	AlCo	Co-60	209.8	+/-	3.6
DK	90-268	AlCo	Co-60	136.3	+/-	3.3
BI	90-277	AlCo	Co-60	657.6	+/-	8.9
DI	90-278	AlCo	Co-60	375.8	+/-	7.0
BJ	90-287	AlCo	Co-60	537.8	+/-	8.2
DJ	90-288	AlCo	Co-60	305.9	+/-	6.1
BH	90-297	AlCo	Co-60	339.9	+/-	6.6
DH	90-298	AlCo	Co-60	222.8	+/-	5.3
CT	90-231	Fe	Mn-54	0.757	+/-	0.016
DT	90-232	Fe	Mn-54	0.912	+/-	0.016
CG	90-241	Fe	Mn-54	6.72	+/-	0.12
DG	90-242	Fe	Mn-54	8.33	+/-	0.09
CH	90-251	Fe	Mn-54	20.81	+/-	0.37
DH	90-252	Fe	Mn-54	21.31	+/-	0.29
CI	90-261	Fe	Mn-54	8.39	+/-	0.19
DI	90-262	Fe	Mn-54	7.73	+/-	0.14
CJ	90-271	Fe	Mn-54	19.07	+/-	0.27
DJ	90-272	Fe	Mn-54	18.72	+/-	0.24
CK	90-281	Fe	Mn-54	15.15	+/-	0.13
DK	90-282	Fe	Mn-54	13.98	+/-	0.17
CL	90-291	Fe	Mn-54	12.41	+/-	0.22
DL	90-292	Fe	Mn-54	12.10	+/-	0.24

Remarks:

AL File: 13910
References: Lab.Book#41p74. LB#35p213-216.
Procedures: A-524.
Analyst: WTF, WRM, CAB.

Approved: C.A. Blackburn 2-20-90

Westinghouse Electric Corporation
Advanced Energy Systems - Analytical Laboratory
Waltz Mill Site

REPORT

Request# 13910

TO: A.H.Fero (W)NATD, Energy Center (East 4-17)

Received: 1/19/90
Reported: 2/20/90

[RESULTS OF ANALYSIS]
Point Beach Unit 2 Cycle 15 Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 2/1/90) dps/mg foil		2 sigma
T	90-239	U-238	Zr-95	1.40	+/-	0.01
G	90-249	U-238	Zr-95	7.74	+/-	0.07
H	90-259	U-238	Zr-95	19.59	+/-	0.13
I	90-269	U-238	Zr-95	7.90	+/-	0.07
J	90-279	U-238	Zr-95	17.76	+/-	0.13
K	90-289	U-238	Zr-95	13.32	+/-	0.11
L	90-299	U-238	Zr-95	11.69	+/-	0.08
T	90-239	U-238	Ru-103	0.544	+/-	0.005
G	90-249	U-238	Ru-103	3.780	+/-	0.040
H	90-259	U-238	Ru-103	9.308	+/-	0.075
I	90-269	U-238	Ru-103	3.801	+/-	0.038
J	90-279	U-238	Ru-103	8.259	+/-	0.074
K	90-289	U-238	Ru-103	6.082	+/-	0.063
L	90-299	U-238	Ru-103	5.451	+/-	0.048
T	90-239	U-238	Cs-137	0.114	+/-	0.003
G	90-249	U-238	Cs-137	0.680	+/-	0.019
H	90-259	U-238	Cs-137	1.843	+/-	0.036
I	90-269	U-238	Cs-137	0.677	+/-	0.021
J	90-279	U-238	Cs-137	1.683	+/-	0.036
K	90-289	U-238	Cs-137	1.241	+/-	0.030
L	90-299	U-238	Cs-137	1.102	+/-	0.021
1	90-240	Np	Cs-137	2.33	+/-	0.32
2	90-250	Np	Cs-137	10.59	+/-	0.53
3	90-260	Np	Cs-137	24.07	+/-	0.64
4	90-270	Np	Cs-137	9.69	+/-	0.58
5	90-280	Np	Cs-137	23.84	+/-	0.78
6	90-290	Np	Cs-137	16.37	+/-	0.59
7	90-300	Np	Cs-137	14.66	+/-	0.55

Remarks:

AL File: 13910
References: Lab.Book#41p74. LB#35p213-216.
Procedures: A-524.
Analyst: WTF, WRM, CAB.

Approved: C.A. Blackburn 2-20-90

Westinghouse Electric Corporation
Advanced Energy Systems - Analytical Laboratory
REPORT Waltz Mill Site Request# 13910

TO: A.H.Fero (W)NATD, Energy Center (East 4-17)

Received: 1/19/90
Reported: 2/20/90

[RESULTS OF ANALYSIS]

Point Beach Unit 2 Cycle 15 Reactor Cavity Dosimetry

Bead Chain Tag ID: S-2, 0 degree.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 2/1/90 Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
+6.5	90-301	3.11	+/-	0.11	5.20	+/-	0.12	42.4	+/-	0.3
+5.5	90-302	7.58	+/-	0.40	12.89	+/-	0.53	95.6	+/-	1.1
+4.5	90-303	11.07	+/-	0.53	18.96	+/-	0.63	122.7	+/-	1.1
+3.5	90-304	12.96	+/-	0.85	21.20	+/-	0.84	151.0	+/-	1.7
+2.5	90-305	14.50	+/-	0.63	23.24	+/-	0.72	168.4	+/-	1.4
+1.5	90-306	14.26	+/-	0.82	22.83	+/-	0.97	173.3	+/-	1.8
+0.5	90-307	13.88	+/-	0.69	21.38	+/-	0.65	177.5	+/-	1.4
-0.5	90-308	14.53	+/-	0.86	22.45	+/-	0.97	200.7	+/-	2.0
-1.5	90-309	13.29	+/-	0.60	20.72	+/-	0.65	187.0	+/-	1.4
-2.5	90-310	12.70	+/-	0.63	21.29	+/-	0.64	178.1	+/-	1.4
-3.5	90-311	12.50	+/-	0.60	20.34	+/-	0.64	163.6	+/-	1.3
-4.5	90-312	11.24	+/-	0.52	18.61	+/-	0.57	131.6	+/-	1.2
-5.5	90-313	7.55	+/-	0.42	12.54	+/-	0.44	87.4	+/-	1.0
-6.5	90-314	2.78	+/-	0.22	4.90	+/-	0.24	67.4	+/-	0.8

NOTE: For the gamma counts, 6 beads were cut from the chain at the designated points of "Feet from Midplane" (3 beads on each side of the point). 6 beads are about 1.4 inches long and weigh about 1.2 grams.

Remarks:

AL File: 13910
References: Lab.Book#41p74. LB#35p213-216.
Procedures: A-524.
Analyst: WTF, WRM, CAB.

Approved: C.A. Blackburn 2-20-90

Westinghouse Electric Corporation
Advanced Energy Systems - Analytical Laboratory
Waltz Mill Site

REPORT Request# 13910

TO: A.H.Fero (W)NATD, Energy Center (East 4-17)

Received: 1/19/90
Reported: 2/20/90

[RESULTS OF ANALYSIS]

Point Beach Unit 2 Cycle 15 Reactor Cavity Dosimetry

Bead Chain Tag ID: S-2, 15 degree.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 2/1/90: Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
+6.5	90-315	2.40	+/-	0.20	4.11	+/-	0.23	43.7	+/-	0.5
+5.5	90-316	6.64	+/-	0.51	10.19	+/-	0.53	141.4	+/-	1.3
+4.5	90-317	10.05	+/-	0.62	16.52	+/-	0.63	203.6	+/-	1.6
+3.5	90-318	11.15	+/-	0.98	17.72	+/-	0.94	236.1	+/-	2.3
+2.5	90-319	12.09	+/-	1.03	20.12	+/-	1.08	253.7	+/-	2.3
+1.5	90-320	11.87	+/-	0.92	19.71	+/-	0.99	253.9	+/-	2.3
+0.5	90-321	12.04	+/-	0.97	19.37	+/-	1.06	247.4	+/-	2.3
-0.5	90-322	12.84	+/-	1.03	20.33	+/-	1.08	253.9	+/-	2.3
-1.5	90-323	12.41	+/-	0.97	19.61	+/-	1.06	241.9	+/-	2.3
-2.5	90-324	12.26	+/-	0.92	18.61	+/-	1.05	227.6	+/-	2.2
-3.5	90-325	11.54	+/-	0.85	18.61	+/-	0.93	212.3	+/-	2.1
-4.5	90-326	10.52	+/-	0.87	17.06	+/-	0.94	176.8	+/-	1.9
-5.5	90-327	6.74	+/-	0.40	11.32	+/-	0.37	125.9	+/-	0.8
-6.5	90-328	2.58	+/-	0.21	4.50	+/-	0.22	68.2	+/-	0.5

NOTE: For the gamma counts, 6 beads were cut from the chain at the designated points of "Feet from Midplane" (3 beads on each side of the point). 6 beads are about 1.4 inches long and weigh about 1.2 grams.

Remarks:

AL File: 13910
References: Lab.Book#41p74. LB#35p213-216.
Procedures: A-524.
Analyst: WTF, WRM, CAB.

Approved: C. A. Blackburn 2-20-90

Westinghouse Electric Corporation
Advanced Energy Systems - Analytical Laboratory
Waltz Mill Site

REPORT

Request# 13910

TO: A.H.Fero (W)NATD, Energy Center (East 4-17)

Received: 1/19/90
Reported: 2/20/90

[RESULTS OF ANALYSIS]

Point Beach Unit 2 Cycle 15 Reactor Cavity Dosimetry

Bead Chain Tag ID: S-2, 30 degree.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 2/1/90 Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg	Co-60 2 sigma
+6.5	90-329	2.03	+/-	0.11	3.36	+/-	0.11	38.5	0.3
+5.5	90-330	4.95	+/-	0.33	8.58	+/-	0.37	108.2	0.8
+4.5	90-331	7.60	+/-	0.38	12.21	+/-	0.51	154.5	0.9
+3.5	90-332	8.37	+/-	0.67	13.71	+/-	0.54	183.4	1.2
+2.5	90-333	8.93	+/-	0.46	14.84	+/-	0.54	195.0	1.1
+1.5	90-334	8.51	+/-	0.78	14.97	+/-	0.98	204.0	2.0
+0.5	90-335	8.60	+/-	0.84	14.18	+/-	0.93	210.6	2.1
-0.5	90-336	9.30	+/-	0.65	14.69	+/-	0.70	206.6	1.6
-1.5	90-337	8.88	+/-	0.68	14.16	+/-	0.70	201.7	1.5
-2.5	90-338	9.42	+/-	0.66	14.13	+/-	0.61	192.8	1.5
-3.5	90-339	8.06	+/-	0.54	13.38	+/-	0.65	175.1	1.4
-4.5	90-340	7.05	+/-	0.35	11.97	+/-	0.43	138.9	0.9
-5.5	90-341	4.45	+/-	0.27	7.94	+/-	0.32	74.5	0.7
-6.5	90-342	1.94	+/-	0.20	3.31	+/-	0.26	51.9	0.6

NOTE: For the gamma counts, 6 beads were cut from the chain at the designated points of "Feet from Midplane" (3 beads on each side of the point). 6 beads are about 1.4 inches long and weigh about 1.2 grams.

Remarks:

AL File: 13910
References: Lab.Book#41p74. LB#35p213-216.
Procedures: A-524.
Analyst: WTF, WRM, CAB.

Approved: C.A. Blackburn 2-20-90

Westinghouse Electric Corporation
Advanced Energy Systems - Analytical Laboratory
Waltz Mill Site

REPORT

Request# 13910

TO: A.H.Fero (W)NATD, Energy Center (East 4-17)

Received: 1/19/90
Reported: 2/20/90

[RESULTS OF ANALYSIS]

Point Beach Unit 2 Cycle 15 Reactor Cavity Dosimetry

Bead Chain Tag ID: S-2, 45 degree.

Feet from Midplane	Lab Sample#	[<— dps/mg of chain @ 2/1/90 — Mn-54 dps/mg	2 sigma	Co-58 dps/mg	2 sigma	—>] Co-60 dps/mg	2 sigma
+6.5	90-343	1.79 +/-	0.14	3.09 +/-	0.19	32.4 +/-	0.3
+5.5	90-344	4.14 +/-	0.24	7.15 +/-	0.29	66.4 +/-	0.6
+4.5	90-345	6.64 +/-	0.43	10.80 +/-	0.49	92.1 +/-	1.0
+3.5	90-346	7.39 +/-	0.32	12.00 +/-	0.47	105.9 +/-	0.8
+2.5	90-347	8.16 +/-	0.38	12.88 +/-	0.43	119.2 +/-	0.9
+1.5	90-348	7.79 +/-	0.28	12.58 +/-	0.35	125.1 +/-	0.8
+0.5	90-349	7.83 +/-	0.35	12.51 +/-	0.37	124.9 +/-	0.8
-0.5	90-350	8.05 +/-	0.39	13.16 +/-	0.45	127.1 +/-	0.9
-1.5	90-351	7.89 +/-	0.40	12.74 +/-	0.45	123.5 +/-	0.9
-2.5	90-352	7.76 +/-	0.47	12.49 +/-	0.55	119.3 +/-	1.2
-3.5	90-353	7.40 +/-	0.50	11.63 +/-	0.56	109.2 +/-	1.1
-4.5	90-354	7.09 +/-	0.44	10.93 +/-	0.49	90.4 +/-	1.0
-5.5	90-355	4.44 +/-	0.22	7.55 +/-	0.25	64.3 +/-	0.5
-6.5	90-356	1.83 +/-	0.17	3.02 +/-	0.17	48.9 +/-	0.4

NOTE: For the gamma counts, 6 beads were cut from the chain at the designated points of "Feet from Midplane" (3 beads on each side of the point). 6 beads are about 1.4 inches long and weigh about 1.2 grams.

Remarks:

AL File: 13910
References: Lab.Book#41p74. LB#35p213-216.
Procedures: A-524.
Analyst: WTF, WRM, CAB.

Approved: C.A. Blackburn 2-20-90

APPENDIX C

MEASURED SPECIFIC ACTIVITY AND IRRADIATION HISTORY OF REACTOR CAVITY SENSOR SETS - CYCLE 16

In this appendix, the irradiation history as extracted from NUREG-0020 and the measured specific activities of radiometric sensors irradiated in the reactor cavity during Cycle 16 are provided.

The irradiation history of Cycle 16 was as follows:

Cycle No.	STARTUP	SHUTDOWN	COMMENT
16	11/24/89	10/06/90	Hf Absorbers Inserted

Ref. Core Power = 1518 MWt

MONTH	THERMAL GENERATION (MW-Hr)
11/89	141690
12/89	1123322
1/90	1127322
2/90	1018430
3/90	1118045
4/90	1090485
5/90	1127775
6/90	1080806
7/90	1128256
8/90	1129766
9/90	1120491
10/90	162384
TOTAL	11368772

The irradiation capsule loading diagram and the measured specific activities of the radiometric monitors from the Cycle 16 irradiation are provided on pages C-2 through C-8. For the multiple foil sensor sets, the individual foil ID can be correlated with the capsule loading diagram provided in Section 6.2-1 in order to determine the location of the foil within the reactor cavity during irradiation.

CONTENTS OF MULTIPLE FOIL SENSOR SETS
CYCLE 16 IRRADIATION

CAPSULE ID and POSITION	BARE OR CADMIUM SHIELDED	RADIOMETRIC MONITOR ID							
		Fe	Ni	Cu	Ti	Nb	Co	U-238	SSTR PACKAGE
M-1	B	CM	--	--	--	--	BG	--	PB-4B
M-2	Cd	DM	AM	AM	M	G	DG	M	--
M-3	Cd	--	--	--	--	--	--	--	PB-4C
N-1	B	CN	--	--	--	--	BF	--	PB-12B
N-2	Cd	DN	AN	AN	N	F	DF	N	--
N-3	Cd	--	--	--	--	--	--	--	PB-12C
O-1	B	CO	--	--	--	--	BE	--	PB-5B
O-2	Cd	DO	AO	AO	O	E	DE	O	--
O-3	Cd	--	--	--	--	--	--	--	PB-5C
P-1	B	CP	--	--	--	--	BD	--	PB-13B
P-2	Cd	DP	AP	AP	P	D	DD	P	--
P-3	Cd	--	--	--	--	--	--	--	PB-13C
Q-1	B	CS	--	--	--	--	BC	--	PB-14B
Q-2	Cd	DS	AS	AS	S	C	DC	S	--
Q-3	Cd	--	--	--	--	--	--	--	PB-14C
R-1	B	CR	--	--	--	--	BB	--	PB-15B
R-2	Cd	DT	AT	AT	T	A	DA	T	--
R-3	Cd	--	--	--	--	--	--	--	PB-15C

REPORT | Westinghouse Advanced Energy Systems | Request# 14245
| Analytical Laboratory - Waltz Mill Site |

Originator: S.L.Anderson (W)NTD, Energy Center (4-36)

Received: 1/14/91
Reported: 3/26/91

[RESULTS OF ANALYSIS]
Point Beach Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 12/12/90) dps/mg *		2 sigma
CM	91-254	Fe	Mn-54	9.09E+00	+/-	1.3E-01
DM	91-255	Fe	Mn-54	9.83E+00	+/-	1.4E-01
CN	91-263	Fe	Mn-54	2.04E+01	+/-	2.0E-01
DN	91-264	Fe	Mn-54	1.97E+01	+/-	1.9E-01
CO	91-272	Fe	Mn-54	9.93E+00	+/-	1.4E-01
DO	91-273	Fe	Mn-54	8.85E+00	+/-	1.3E-01
CP	91-281	Fe	Mn-54	1.75E+01	+/-	2.0E-01
DP	91-282	Fe	Mn-54	1.66E+01	+/-	1.7E-01
CS	91-290	Fe	Mn-54	1.47E+01	+/-	1.7E-01
DS	91-291	Fe	Mn-54	1.50E+01	+/-	1.7E-01
CR	91-299	Fe	Mn-54	1.42E+01	+/-	1.7E-01
DR	91-300	Fe	Mn-54	1.40E+01	+/-	1.6E-01
AM	91-257	Cu	Co-60	2.79E-01	+/-	4.7E-03
AN	91-266	Cu	Co-60	5.84E-01	+/-	8.4E-03
AO	91-275	Cu	Co-60	2.51E-01	+/-	5.6E-03
AP	91-284	Cu	Co-60	5.10E-01	+/-	6.4E-03
AS	91-293	Cu	Co-60	4.56E-01	+/-	7.5E-03
AR	91-302	Cu	Co-60	4.49E-01	+/-	7.3E-03
BG	91-260	AlCo	Co-60	1.85E+02	+/-	2.9E+00
DG	91-261	AlCo	Co-60	1.32E+02	+/-	2.4E+00
BF	91-269	AlCo	Co-60	4.39E+02	+/-	4.6E+00
DF	91-270	AlCo	Co-60	2.64E+02	+/-	3.5E+00
BE	91-278	AlCo	Co-60	2.05E+02	+/-	3.1E+00
DE	91-279	AlCo	Co-60	1.33E+02	+/-	2.5E+00
BD	91-287	AlCo	Co-60	5.63E+02	+/-	5.2E+00
DD	91-288	AlCo	Co-60	3.37E+02	+/-	3.3E+00
BC	91-296	AlCo	Co-60	5.03E+02	+/-	4.2E+00
DC	91-297	AlCo	Co-60	2.73E+02	+/-	3.0E+00
BB	91-305	AlCo	Co-60	3.13E+02	+/-	2.6E+00
DB	91-306	AlCo	Co-60	2.08E+02	+/-	2.0E+00

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14245
References: Lab.Book# 46 pages 103-104.
Procedures: A-524.
Analyst: WTF, FRC

Approved: [signature]

* REVISED REPORT

Westinghouse Advanced Energy Systems
Analytical Laboratory - Waltz Mill Site

Request# 14245

Originator: S.L.Anderson (W)NID, Energy Center (4-36)

Received: 1/14/91
Reported: 3/27/91

[RESULTS OF ANALYSIS]
Point Beach Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 12/12/90) dps/mg *		2 sigma
M	91-262	U-238	Cs-137	7.63E-01	+/-	2.7E-02
N	91-271	U-238	Cs-137	1.61E+00	+/-	8.7E-02
O	91-280	U-238	Cs-137	7.30E-01	+/-	3.9E-02
P	91-289	U-238	Cs-137	1.47E+00	+/-	6.2E-02
S	91-298	U-238	Cs-137	1.12E+00	+/-	2.2E-02
R	91-307	U-238	Cs-137	1.03E+00	+/-	2.0E-02
M	91-262	U-238	Ru-103	1.26E+01	+/-	2.3E-01
N	91-271	U-238	Ru-103	2.28E+01	+/-	5.2E-01
O	91-280	U-238	Ru-103	1.11E+01	+/-	3.2E-01
P	91-289	U-238	Ru-103	1.92E+01	+/-	4.2E-01
S	91-298	U-238	Ru-103	1.58E+01	+/-	2.0E-01
R	91-307	U-238	Ru-103	1.48E+01	+/-	1.9E-01
M	91-262	U-238	Zr-95	1.63E+01	+/-	2.1E-01
N	91-271	U-238	Zr-95	3.10E+01	+/-	4.1E-01
O	91-280	U-238	Zr-95	1.52E+01	+/-	2.3E-01
P	91-289	U-238	Zr-95	2.72E+01	+/-	3.8E-01
S	91-298	U-238	Zr-95	2.24E+01	+/-	2.0E-01
R	91-307	U-238	Zr-95	2.07E+01	+/-	1.9E-01
M	91-258	Ti	Sc-46	3.50E+00	+/-	8.1E-02
N	91-267	Ti	Sc-46	6.94E+00	+/-	9.6E-02
O	91-276	Ti	Sc-46	3.16E+00	+/-	6.3E-02
P	91-285	Ti	Sc-46	5.99E+00	+/-	8.8E-02
S	91-294	Ti	Sc-46	5.26E+00	+/-	8.3E-02
R	91-303	Ti	Sc-46	5.02E+00	+/-	7.9E-02
AM	91-256	Ni	Co-58	1.89E+02	+/-	3.7E+00
AN	91-265	Ni	Co-58	3.67E+02	+/-	5.1E+00
AO	91-274	Ni	Co-58	1.74E+02	+/-	3.5E+00
AP	91-283	Ni	Co-58	3.17E+02	+/-	4.8E+00
AS	91-292	Ni	Co-58	2.69E+02	+/-	4.4E+00
AR	91-301	Ni	Co-58	2.55E+02	+/-	4.1E+00

Remarks: * Results are in units of dps/(mg of Dosimeter Material).
* Sample #91-280,Zr-95 data corrected.

AL File: 14245
References: Lab.Book# 49 pages 32-37.
Procedures: A-524.
Analyst: WIF, TK

Approved: [signature]

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14220

Originator: S.L.Anderson (W)NTD, Energy Center (4-36)

Received: 1/14/91
Reported: 3/27/91

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Bead Chain Tag ID: 0 deg.

Feet from Midplane	Lab Sample#	Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
		[<— dps/mg of chain @ 12/12/90 —]								
+7.5	91-65-A	1.15E+00	+/-	1.5E-01	3.26E+00	+/-	2.7E-01	3.01E+01	+/-	2.4E-0[illegible]
+6.5	91-65-B	3.65E+00	+/-	1.9E-01	9.89E+00	+/-	4.2E-01	4.31E+01	+/-	2.8E-0[illegible]
+5.5	91-65-C	8.87E+00	+/-	3.1E-01	2.41E+01	+/-	7.6E-01	9.86E+01	+/-	4.3E-0[illegible]
+4.5	91-65-D	1.40E+01	+/-	5.6E-01	3.75E+01	+/-	1.2E+00	1.32E+02	+/-	7.0E-0[illegible]
+3.5	91-65-E	1.56E+01	+/-	8.0E-01	4.00E+01	+/-	1.8E+00	1.53E+02	+/-	1.1E+0[illegible]
+2.5	91-65-F	1.66E+01	+/-	7.2E-01	4.19E+01	+/-	1.8E+00	1.66E+02	+/-	9.9E-0[illegible]
+1.5	91-65-G	1.58E+01	+/-	7.9E-01	3.89E+01	+/-	1.6E+00	1.61E+02	+/-	9.2E-0[illegible]
+0.5	91-65-H	1.42E+01	+/-	7.2E-01	3.54E+01	+/-	1.7E+00	1.63E+02	+/-	9.8E-0[illegible]
-0.5	91-65-I	1.14E+01	+/-	6.5E-01	2.95E+01	+/-	1.5E+00	1.60E+02	+/-	9.7E-0[illegible]
-1.5	91-65-J	1.12E+01	+/-	6.6E-01	2.84E+01	+/-	1.4E+00	1.54E+02	+/-	9.6E-0[illegible]
-2.5	91-65-K	1.26E+01	+/-	7.9E-01	3.12E+01	+/-	1.6E+00	1.51E+02	+/-	8.5E-0[illegible]
-3.5	91-65-L	1.31E+01	+/-	6.9E-01	3.57E+01	+/-	1.5E+00	1.55E+02	+/-	9.6E-0[illegible]
-4.5	91-65-M	1.30E+01	+/-	7.2E-01	3.33E+01	+/-	1.5E+00	1.27E+02	+/-	7.8E-0[illegible]
-5.5	91-65-N	8.83E+00	+/-	5.5E-01	2.37E+01	+/-	1.2E+00	9.07E+01	+/-	7.3E-0[illegible]
-6.5	91-65-O	3.36E+00	+/-	5.0E-01	9.73E+00	+/-	9.9E-01	6.68E+01	+/-	5.7E-0[illegible]

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14220
References: Lab.Book# 49 pages 32-37.
Procedures: A-524.
Analyst: WTF, TK

Approved: [signature]

REPORT | Westinghouse Advanced Energy Systems | Request# 1422C
Analytical Laboratory - Waltz Mill Site

Originator: S.L.Anderson (W)NTD, Energy Center (4-36)

Received: 1/14/91
Reported: 3/27/91

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Bead Chain Tag ID: 15 deg.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 12/12/90: Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigm
+7.5	91-66-A	8.48E-01	+/-	1.7E-01	2.47E+00	+/-	3.9E-01	2.71E+01	+/-	2.9E-0
+6.5	91-66-B	2.68E+00	+/-	3.5E-01	7.15E+00	+/-	7.2E-01	4.32E+01	+/-	4.1E-0
+5.5	91-66-C	6.66E+00	+/-	5.5E-01	1.88E+01	+/-	1.3E+00	1.42E+02	+/-	9.1E-0
+4.5	91-66-D	1.07E+01	+/-	7.3E-01	2.93E+01	+/-	1.7E+00	2.03E+02	+/-	1.1E+0
+3.5	91-66-E	1.28E+01	+/-	1.4E+00	3.10E+01	+/-	2.6E+00	2.28E+02	+/-	1.7E+0
+2.5	91-66-F	1.29E+01	+/-	9.8E-01	3.23E+01	+/-	2.3E+00	2.40E+02	+/-	1.5E+0
+1.5	91-66-G	1.22E+01	+/-	1.2E+00	3.03E+01	+/-	2.6E+00	2.29E+02	+/-	1.6E+0
+0.5	91-66-H	1.11E+01	+/-	9.9E-01	2.86E+01	+/-	1.8E+00	2.17E+02	+/-	1.5E+0
-0.5	91-66-I	9.90E+00	+/-	1.1E+00	2.49E+01	+/-	2.4E+00	2.00E+02	+/-	1.5E+0
-1.5	91-66-J	9.97E+00	+/-	6.5E-01	2.58E+01	+/-	1.6E+00	1.91E+02	+/-	1.1E+0
-2.5	91-66-K	9.75E+00	+/-	9.6E-01	2.60E+01	+/-	2.2E+00	1.86E+02	+/-	1.2E+0
-3.5	91-66-L	1.16E+01	+/-	7.3E-01	2.86E+01	+/-	1.6E+00	1.90E+02	+/-	1.1E+0
-4.5	91-66-M	1.11E+01	+/-	6.8E-01	2.89E+01	+/-	1.6E+00	1.66E+02	+/-	1.0E+0
-5.5	91-66-N	6.95E+00	+/-	5.5E-01	1.84E+01	+/-	1.3E+00	1.22E+02	+/-	8.5E-0
-6.5	91-66-O	2.97E+00	+/-	4.2E-01	7.37E+00	+/-	8.1E-01	6.52E+01	+/-	5.1E-0

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14220
References: Lab.Book# 49 pages 32-37.
Procedures: A-524.
Analyst: WTF, TK

Approved: Bateman Nikki

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14220

Originator: S.L.Anderson (W)NTD, Energy Center (4-36)

Received: 1/14/91
Reported: 3/27/91

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Bead Chain Tag ID: 30 deg.

Feet from Midplane	Lab Sample#	[<——— dps/mg of chain @ 12/12/90 ———>] Mn-54 dps/mg	Mn-54 2 sigma	Co-58 dps/mg	Co-58 2 sigma	Co-60 dps/mg	Co-60 2 sigma
+7.5	91-67-A	7.58E-01 +/-	1.1E-01	2.13E+00 +/-	3.1E-01	2.29E+01 +/-	2.1E-01
+6.5	91-67-B	2.25E+00 +/-	2.2E-01	5.96E+00 +/-	5.3E-01	3.74E+01 +/-	2.9E-01
+5.5	91-67-C	5.66E+00 +/-	5.3E-01	1.47E+01 +/-	1.0E+00	1.06E+02 +/-	7.9E-01
+4.5	91-67-D	8.21E+00 +/-	6.1E-01	2.08E+01 +/-	1.4E+00	1.50E+02 +/-	9.4E-01
+3.5	91-67-E	8.59E+00 +/-	9.0E-01	2.26E+01 +/-	2.0E+00	1.72E+02 +/-	1.1E+00
+2.5	91-67-F	9.49E+00 +/-	6.7E-01	2.49E+01 +/-	1.5E+00	1.86E+02 +/-	1.1E+00
+1.5	91-67-G	9.11E+00 +/-	9.0E-01	2.39E+01 +/-	2.0E+00	1.84E+02 +/-	1.2E+00
+0.5	91-67-H	9.54E+00 +/-	6.4E-01	2.40E+01 +/-	1.5E+00	1.88E+02 +/-	1.1E+00
-0.5	91-67-I	9.19E+00 +/-	6.7E-01	2.41E+01 +/-	1.5E+00	1.83E+02 +/-	1.0E+00
-1.5	91-67-J	8.91E+00 +/-	9.2E-01	2.27E+01 +/-	2.1E+00	1.73E+02 +/-	1.1E+00
-2.5	91-67-K	9.13E+00 +/-	6.9E-01	2.29E+01 +/-	1.6E+00	1.72E+02 +/-	1.0E+00
-3.5	91-67-L	8.27E+00 +/-	6.3E-01	2.19E+01 +/-	1.4E+00	1.58E+02 +/-	9.8E-01
-4.5	91-67-M	7.70E+00 +/-	4.0E-01	2.04E+01 +/-	9.4E-01	1.24E+02 +/-	5.2E-01
-5.5	91-67-N	5.10E+00 +/-	2.3E-01	1.32E+01 +/-	5.2E-01	7.01E+01 +/-	3.6E-01
-6.5	91-67-O	2.13E+00 +/-	2.1E-01	5.42E+00 +/-	4.2E-01	4.97E+01 +/-	3.0E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14220
References: Lab.Book# 49 pages 32-37.
Procedures: A-524.
Analyst: WTF, TK

Approved: [signature]

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14220

Originator: S.L.Anderson (W)NID, Energy Center (4-36)

Received: 1/14/91
Reported: 3/27/91

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Bead Chain Tag ID: 45 deg.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 12/12/90 Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
+7.5	91-68-A	6.41E-01	+/-	9.2E-02	1.78E+00	+/-	1.9E-01	2.05E+01	+/-	1.3E-01
+6.5	91-68-B	2.04E+00	+/-	1.7E-01	5.23E+00	+/-	3.7E-01	3.20E+01	+/-	2.4E-01
+5.5	91-68-C	4.86E+00	+/-	3.4E-01	1.24E+01	+/-	7.1E-01	6.48E+01	+/-	3.9E-01
+4.5	91-68-D	7.55E+00	+/-	5.4E-01	1.95E+01	+/-	1.3E+00	8.97E+01	+/-	7.3E-01
+3.5	91-68-E	7.94E+00	+/-	5.9E-01	2.09E+01	+/-	1.3E+00	1.05E+02	+/-	7.9E-01
+2.5	91-68-F	8.91E+00	+/-	4.3E-01	2.24E+01	+/-	9.1E-01	1.13E+02	+/-	5.1E-01
+1.5	91-68-G	8.84E+00	+/-	5.9E-01	2.15E+01	+/-	1.2E+00	1.20E+02	+/-	8.5E-01
+0.5	91-68-H	8.78E+00	+/-	6.1E-01	2.15E+01	+/-	1.3E+00	1.19E+02	+/-	8.4E-01
-0.5	91-68-I	8.65E+00	+/-	6.7E-01	2.27E+01	+/-	1.5E+00	1.15E+02	+/-	7.4E-01
-1.5	91-68-J	8.71E+00	+/-	5.8E-01	2.13E+01	+/-	1.1E+00	1.16E+02	+/-	8.3E-01
-2.5	91-68-K	8.34E+00	+/-	6.3E-01	2.19E+01	+/-	1.3E+00	1.07E+02	+/-	7.2E-01
-3.5	91-68-L	8.15E+00	+/-	3.5E-01	2.08E+01	+/-	7.2E-01	1.01E+02	+/-	4.4E-01
-4.5	91-68-M	7.68E+00	+/-	5.0E-01	2.03E+01	+/-	1.4E+00	8.34E+01	+/-	6.3E-01
-5.5	91-68-N	5.12E+00	+/-	1.4E-01	1.35E+01	+/-	3.4E-01	6.22E+01	+/-	2.0E-01
-6.5	91-68-O	2.03E+00	+/-	2.0E-01	5.78E+00	+/-	4.5E-01	4.63E+01	+/-	2.7E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14220
References: Lab.Book# 49 pages 32-37.
Procedures: A-524.
Analyst: WTF, TK

Approved: Butram Mirabhi

APPENDIX D

MEASURED SPECIFIC ACTIVITY AND IRRADIATION HISTORY OF REACTOR CAVITY SENSOR SETS - CYCLE 17

In this appendix, the irradiation history as extracted from NUREG-0020 and the measured specific activities of radiometric sensors irradiated in the reactor cavity during Cycle 17 are provided.

The irradiation history of Cycle 17 was as follows:

Cycle No.	STARTUP	SHUTDOWN	COMMENT
17	11/17/90	09/27/91	Hf Absorbers Inserted

Ref. Core Power = 1518 MWt

MONTH	THERMAL GENERATION (MW-Hr)
11/89	405311
12/89	1120800
1/90	1123755
2/90	1013700
3/90	1110255
4/90	1087195
5/90	1124595
6/90	1080040
7/90	1100758
8/90	1087688
9/90	941710
TOTAL	11368772

The irradiation capsule loading diagram and the measured specific activities of the radiometric monitors from the Cycle 17 irradiation are provided on pages D-2 through D-9. For the multiple foil sensor sets, the individual foil ID can be correlated with the capsule loading diagram provided in Section 6.3-1 in order to determine the location of the foil within the reactor cavity during irradiation.

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14477

Originator: S. Anderson (W)NTD, Energy Center

Received: 10/18/91
Reported: 12/12/91

[RESULTS OF ANALYSIS]
POINT BEACH UNIT 2 CYCLE 17 REACTOR CAVITY DOSIMETRY

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 10/24/91) dps/mg *		2 sigma
BG	91-1822	Fe	Mn-54	9.82E+00	+/-	1.1E-01
AG	91-1823	Fe	Mn-54	1.08E+01	+/-	1.1E-01
BH	91-1831	Fe	Mn-54	2.30E+01	+/-	1.6E-01
AH	91-1832	Fe	Mn-54	2.22E+01	+/-	1.6E-01
AI	91-1840	Fe	Mn-54	1.11E+01	+/-	1.2E-01
BI	91-1841	Fe	Mn-54	9.91E+00	+/-	1.1E-01
BJ	91-1849	Fe	Mn-54	1.98E+01	+/-	1.7E-01
AJ	91-1850	Fe	Mn-54	1.96E+01	+/-	1.5E-01
BK	91-1858	Fe	Mn-54	1.64E+01	+/-	1.5E-01
AK	91-1859	Fe	Mn-54	1.63E+01	+/-	1.4E-01
BL	91-1867	Fe	Mn-54	1.58E+01	+/-	1.4E-01
AL	91-1868	Fe	Mn-54	1.56E+01	+/-	1.3E-01
G	91-1824	Ni	Co-58	2.83E+02	+/-	1.6E+00
H	91-1833	Ni	Co-58	5.51E+02	+/-	2.2E+00
I	91-1842	Ni	Co-58	2.61E+02	+/-	1.5E+00
J	91-1851	Ni	Co-58	4.80E+02	+/-	2.0E+00
K	91-1860	Ni	Co-58	4.00E+02	+/-	1.5E+00
L	91-1869	Ni	Co-58	3.82E+02	+/-	1.8E+00
BG	91-1828	AlCo	Co-60	1.95E+02	+/-	1.9E+00
AG	91-1829	AlCo	Co-60	1.40E+02	+/-	1.6E+00
BH	91-1837	AlCo	Co-60	4.67E+02	+/-	3.0E+00
AH	91-1838	AlCo	Co-60	2.77E+02	+/-	2.3E+00
AI	91-1846	AlCo	Co-60	2.13E+02	+/-	2.0E+00
BI	91-1847	AlCo	Co-60	1.40E+02	+/-	1.6E+00
BJ	91-1855	AlCo	Co-60	5.89E+02	+/-	3.3E+00
AJ	91-1856	AlCo	Co-60	3.35E+02	+/-	2.6E+00
BK	91-1864	AlCo	Co-60	5.25E+02	+/-	3.2E+00
AK	91-1865	AlCo	Co-60	2.89E+02	+/-	2.3E+00
BL	91-1873	AlCo	Co-60	3.26E+02	+/-	2.5E+00
AL	91-1874	AlCo	Co-60	2.21E+02	+/-	2.1E+00

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14477
References: Lab Book#46 pages 246-247
Procedures: A-524.
Analyst: WTF, TRK, MRK

Approved: Mark R. Kawchak

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14477

Originator: S. Anderson (W)NTD, Energy Center

Received: 10/18/91
Reported: 12/12/91

[RESULTS OF ANALYSIS]

POINT BEACH UNIT 2 CYCLE 17 REACTOR CAVITY DOSIMETRY

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 10/24/91) dps/mg *		2 sigma
G	91-1830	U-238	Cs-137	7.59E-01	+/-	2.21E-02
H	91-1839	U-238	Cs-137	1.49E+00	+/-	3.09E-02
I	91-1848	U-238	Cs-137	6.87E-01	+/-	2.34E-02
J	91-1857	U-238	Cs-137	1.37E+00	+/-	2.77E-02
K	91-1866	U-238	Cs-137	1.15E+00	+/-	2.52E-02
L	91-1875	U-238	Cs-137	1.01E+00	+/-	2.62E-02
G	91-1830	U-238	Ru-103	2.61E+01	+/-	1.09E-01
H	91-1839	U-238	Ru-103	4.67E+01	+/-	1.52E-01
I	91-1848	U-238	Ru-103	2.32E+01	+/-	1.18E-01
J	91-1857	U-238	Ru-103	4.22E+01	+/-	1.43E-01
K	91-1866	U-238	Ru-103	3.52E+01	+/-	1.32E-01
L	91-1875	U-238	Ru-103	3.19E+01	+/-	1.53E-01
G	91-1830	U-238	Zr-95	2.55E+01	+/-	1.25E-01
H	91-1839	U-238	Zr-95	4.70E+01	+/-	1.75E-01
I	91-1848	U-238	Zr-95	2.31E+01	+/-	1.35E-01
J	91-1857	U-238	Zr-95	4.38E+01	+/-	1.63E-01
K	91-1866	U-238	Zr-95	3.69E+01	+/-	1.50E-01
L	91-1875	U-238	Zr-95	3.23E+01	+/-	1.65E-01
G	91-1826	Ti	Sc-46	4.91E+00	+/-	4.79E-02
H	91-1835	Ti	Sc-46	9.84E+00	+/-	6.94E-02
I	91-1844	Ti	Sc-46	4.55E+00	+/-	4.62E-02
J	91-1853	Ti	Sc-46	8.68E+00	+/-	6.50E-02
K	91-1862	Ti	Sc-46	7.50E+00	+/-	6.02E-02
L	91-1871	Ti	Sc-46	7.36E+00	+/-	5.95E-02

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14477
References: Lab Book#46 pages 246-247
Procedures: A-524.
Analyst: WTF, TRK, MRK

Approved: Mark R. Kanzelf

REPORT | Westinghouse Advanced Energy Systems | Request# 14477
Analytical Laboratory - Waltz Mill Site

Originator: S. Anderson (W)NTD, Energy Center

Received: 10/18/91
Reported: 12/12/91

[RESULTS OF ANALYSIS]

POINT BEACH UNIT 2 CYCLE 17 REACTOR CAVITY DOSIMETRY

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 10/24/91) dps/mg *		2 sigma
G	91-1825	Cu	Co-60	2.87E-01	+/-	3.79E-03
H	91-1834	Cu	Co-60	6.14E-01	+/-	5.53E-03
I	91-1843	Cu	Co-60	2.66E-01	+/-	3.63E-03
J	91-1852	Cu	Co-60	5.44E-01	+/-	5.19E-03
K	91-1861	Cu	Co-60	4.61E-01	+/-	4.85E-03
L	91-1870	Cu	Co-60	4.73E-01	+/-	4.83E-03

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14477
References: Lab Book#46 pages 246-247
Procedures: A-524.
Analyst: WTF, TRK, MRK

Approved: Mark R. Kowalski

REPORT | Westinghouse Advanced Energy Systems / Analytical Laboratory - Waltz Mill Site | Request# 14477

Originator: S. Anderson (W)NATD, Energy Center
Radiation Engineering & Analysis
Westinghouse Electric Corporation

Received: 10/18/91
Reported: 2/18/92

[RESULTS OF ANALYSIS]

POINT BEACH UNIT 2 CYCLE 17 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: 0 deg.

Feet from Midplane	Lab Sample#	Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
		dps/mg of chain @ 10/24/91								
+7.5	91-1818A	1.25E+00	+/-	1.2E-01	4.79E+00	+/-	2.9E-01	2.80E+01	+/-	1.6E-01
+6.5	91-1818B	4.13E+00	+/-	2.4E-01	1.55E+01	+/-	5.9E-01	4.05E+01	+/-	2.7E-01
+5.5	91-1818C	1.03E+01	+/-	3.5E-01	3.71E+01	+/-	8.8E-01	7.52E+01	+/-	3.8E-01
+4.5	91-1818D	1.53E+01	+/-	9.9E-01	5.65E+01	+/-	2.2E+00	1.02E+02	+/-	1.0E+00
+3.5	91-1818E	1.71E+01	+/-	1.1E+00	5.90E+01	+/-	2.5E+00	1.18E+02	+/-	1.1E+00
+2.5	91-1818F	1.80E+01	+/-	1.1E+00	6.33E+01	+/-	2.6E+00	1.28E+02	+/-	1.1E+00
+1.5	91-1818G	1.72E+01	+/-	1.1E+00	6.01E+01	+/-	2.5E+00	1.27E+02	+/-	1.1E+00
+0.5	91-1818H	1.60E+01	+/-	1.0E+00	5.39E+01	+/-	2.5E+00	1.28E+02	+/-	1.1E+00
-0.5	91-1818I	1.38E+01	+/-	1.1E+00	4.65E+01	+/-	2.5E+00	1.27E+02	+/-	1.1E+00
-1.5	91-1818J	1.36E+01	+/-	1.1E+00	4.50E+01	+/-	2.6E+00	1.23E+02	+/-	1.1E+00
-2.5	91-1818K	1.34E+01	+/-	9.4E-01	5.07E+01	+/-	2.4E+00	1.24E+02	+/-	1.1E+00
-3.5	91-1818L	1.48E+01	+/-	9.8E-01	5.20E+01	+/-	2.3E+00	1.22E+02	+/-	1.1E+00
-4.5	91-1818M	1.44E+01	+/-	8.9E-01	5.34E+01	+/-	2.4E+00	1.02E+02	+/-	1.0E+00
-5.5	91-1818N	1.02E+01	+/-	4.5E-01	3.57E+01	+/-	1.0E+00	6.97E+01	+/-	4.7E-01
-6.5	91-1818O	4.23E+00	+/-	4.1E-01	1.46E+01	+/-	8.4E-01	6.49E+01	+/-	4.5E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14477
References: Lab Book#46 pages 246-247
Procedures: A-524.
Analyst: WTF, TRK, MRK

Approved: [signature]

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14477

Originator: S. Anderson (W)NATD, Energy Center
Radiation Engineering & Analysis
Westinghouse Electric Corporation

Received: 10/18/91
Reported: 2/18/92

[RESULTS OF ANALYSIS]

POINT BEACH UNIT 2 CYCLE 17 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: 15 deg.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 10/24/91 — Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
+7.5	91-1819A	1.11E+00	+/-	1.1E-01	4.14E+00	+/-	2.6E-01	2.56E+01	+/-	1.5E-01
+6.5	91-1819B	3.10E+00	+/-	2.3E-01	1.15E+01	+/-	5.5E-01	4.19E+01	+/-	2.8E-01
+5.5	91-1819C	7.86E+00	+/-	7.1E-01	2.96E+01	+/-	1.8E+00	1.35E+02	+/-	9.0E-01
+4.5	91-1819D	1.07E+01	+/-	7.8E-01	4.48E+01	+/-	2.2E+00	1.90E+02	+/-	1.1E+00
+3.5	91-1819E	1.29E+01	+/-	7.0E-01	4.69E+01	+/-	1.9E+00	2.15E+02	+/-	9.3E-01
+2.5	91-1819F	1.43E+01	+/-	1.3E+00	5.09E+01	+/-	3.0E+00	2.26E+02	+/-	1.5E+00
+1.5	91-1819G	1.39E+01	+/-	9.3E-01	4.64E+01	+/-	2.2E+00	2.20E+02	+/-	1.2E+00
+0.5	91-1819H	1.31E+01	+/-	8.9E-01	4.45E+01	+/-	2.2E+00	2.05E+02	+/-	1.1E+00
-0.5	91-1819I	1.25E+01	+/-	9.0E-01	4.10E+01	+/-	1.9E+00	1.62E+02	+/-	9.9E-01
-1.5	91-1819J	1.13E+01	+/-	7.4E-01	3.95E+01	+/-	2.0E+00	1.52E+02	+/-	9.5E-01
-2.5	91-1819K	1.27E+01	+/-	7.7E-01	4.13E+01	+/-	1.8E+00	1.51E+02	+/-	9.6E-01
-3.5	91-1819L	1.25E+01	+/-	8.4E-01	4.51E+01	+/-	2.0E+00	1.48E+02	+/-	9.5E-01
-4.5	91-1819M	1.25E+01	+/-	7.6E-01	4.17E+01	+/-	1.9E+00	1.28E+02	+/-	8.8E-01
-5.5	91-1819N	8.30E+00	+/-	6.7E-01	2.91E+01	+/-	1.5E+00	9.42E+01	+/-	7.5E-01
-6.5	91-1819O	3.04E+00	+/-	2.4E-01	1.16E+01	+/-	6.2E-01	5.03E+01	+/-	3.1E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14477
References: Lab Book#46 pages 246-247
Procedures: A-524.
Analyst: WTF, TRK, MRK

Approved: Mark Krutchel

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14477

Originator: S. Anderson (W)NATD, Energy Center
Radiation Engineering & Analysis
Westinghouse Electric Corporation

Received: 10/18/91
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[RESULTS OF ANALYSIS]

POINT BEACH UNIT 2 CYCLE 17 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: ~~20~~ 30 deg.

Feet from Midplane	Lab Sample#	Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
		dps/mg of chain @ 10/24/91								
+7.5	91-1820A	8.05E-01	+/-	1.1E-01	3.07E+00	+/-	2.7E-01	2.10E+01	+/-	1.4E-01
+6.5	91-1820B	2.39E+00	+/-	1.5E-01	9.26E+00	+/-	4.0E-01	3.47E+01	+/-	1.8E-01
+5.5	91-1820C	6.06E+00	+/-	5.7E-01	2.09E+01	+/-	1.4E+00	9.70E+01	+/-	7.7E-01
+4.5	91-1820D	9.17E+00	+/-	7.7E-01	3.21E+01	+/-	1.8E+00	1.40E+02	+/-	9.3E-01
+3.5	91-1820E	1.08E+01	+/-	8.0E-01	3.56E+01	+/-	1.8E+00	1.64E+02	+/-	9.1E-01
+2.5	91-1820F	1.01E+01	+/-	8.3E-01	3.74E+01	+/-	2.1E+00	1.76E+02	+/-	1.0E+00
+1.5	91-1820G	1.03E+01	+/-	7.9E-01	3.90E+01	+/-	2.3E+00	1.81E+02	+/-	1.0E+00
+0.5	91-1820H	1.10E+01	+/-	8.3E-01	3.61E+01	+/-	2.1E+00	1.81E+02	+/-	1.0E+00
-0.5	91-1820I	1.09E+01	+/-	8.7E-01	3.93E+01	+/-	2.1E+00	1.46E+02	+/-	9.4E-01
-1.5	91-1820J	1.04E+01	+/-	7.2E-01	3.70E+01	+/-	1.8E+00	1.42E+02	+/-	9.2E-01
-2.5	91-1820K	1.06E+01	+/-	8.2E-01	3.80E+01	+/-	2.0E+00	1.37E+02	+/-	9.1E-01
-3.5	91-1820L	9.93E+00	+/-	7.1E-01	3.49E+01	+/-	1.9E+00	1.26E+02	+/-	8.8E-01
-4.5	91-1820M	9.42E+00	+/-	6.7E-01	3.32E+01	+/-	1.7E+00	1.02E+02	+/-	7.8E-01
-5.5	91-1820N	5.71E+00	+/-	3.2E-01	2.19E+01	+/-	7.0E-01	5.51E+01	+/-	3.2E-01
-6.5	91-1820O	2.30E+00	+/-	2.1E-01	8.87E+00	+/-	5.1E-01	3.89E+01	+/-	2.7E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14477
References: Lab Book#46 pages 246-247
Procedures: A-524.
Analyst: WTF, TRK, MRK

Approved: [signature]

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14477

Originator: S. Anderson (W)NATD, Energy Center
Radiation Engineering & Analysis
Westinghouse Electric Corporation

Received: 10/18/91
Reported: 2/18/92

[RESULTS OF ANALYSIS]

POINT BEACH UNIT 2 CYCLE 17 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: 45 deg.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 10/24/91					
		Mn-54		Co-58		Co-60	
		dps/mg	2 sigma	dps/mg	2 sigma	dps/mg	2 sigma
+7.5	91-1821A	7.41E-01 +/-	9.0E-02	2.43E+00 +/-	1.9E-01	2.01E+01 +/-	1.2E-01
+6.5	91-1821B	2.22E+00 +/-	1.2E-01	8.10E+00 +/-	2.8E-01	2.97E+01 +/-	1.5E-01
+5.5	91-1821C	5.15E+00 +/-	2.9E-01	1.82E+01 +/-	7.0E-01	6.00E+01 +/-	3.4E-01
+4.5	91-1821D	8.05E+00 +/-	3.5E-01	2.88E+01 +/-	8.3E-01	8.04E+01 +/-	3.9E-01
+3.5	91-1821E	9.27E+00 +/-	5.8E-01	3.23E+01 +/-	1.7E+00	9.63E+01 +/-	7.6E-01
+2.5	91-1821F	1.03E+01 +/-	7.2E-01	3.69E+01 +/-	1.9E+00	1.07E+02 +/-	8.0E-01
+1.5	91-1821G	9.59E+00 +/-	6.5E-01	3.52E+01 +/-	1.8E+00	1.13E+02 +/-	8.3E-01
+0.5	91-1821H	9.36E+00 +/-	6.4E-01	3.31E+01 +/-	1.7E+00	1.13E+02 +/-	8.2E-01
-0.5	91-1821I	1.08E+01 +/-	6.5E-01	3.58E+01 +/-	1.6E+00	9.42E+01 +/-	7.5E-01
-1.5	91-1821J	1.00E+01 +/-	6.0E-01	3.62E+01 +/-	1.7E+00	9.26E+01 +/-	7.5E-01
-2.5	91-1821K	1.05E+01 +/-	7.5E-01	3.51E+01 +/-	1.5E+00	8.86E+01 +/-	7.3E-01
-3.5	91-1821L	9.67E+00 +/-	6.2E-01	3.38E+01 +/-	1.6E+00	8.25E+01 +/-	7.0E-01
-4.5	91-1821M	8.53E+00 +/-	3.5E-01	2.99E+01 +/-	8.5E-01	6.73E+01 +/-	3.6E-01
-5.5	91-1821N	5.93E+00 +/-	2.7E-01	1.02E+01 +/-	1.2E+00	2.17E+01 +/-	6.9E-01
-6.5	91-1821O	2.60E+00 +/-	1.5E-01	9.18E+00 +/-	3.8E-01	3.69E+01 +/-	1.9E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 14477
References: Lab Book#46 pages 246-247
Procedures: A-524.
Analyst: WTF, TRK, MRK

Approved: Mark Kurdiak

APPENDIX E

MEASURED SPECIFIC ACTIVITY AND IRRADIATION HISTORY OF REACTOR CAVITY SENSOR SETS - CYCLES 18 THROUGH 20

In this appendix, the irradiation history as extracted from NUREG-0020 and the measured specific activities of radiometric sensors irradiated in the reactor cavity during Cycles 18 through 20 are provided.

The irradiation history of Cycles 18 through 20 was as follows:

Cycle No.	STARTUP	SHUTDOWN	COMMENT
18	11/14/91	09/26/92	Hf Absorbers Inserted
19	11/18/92	09/25/93	Hf Absorbers Inserted
20	10/31/93	09/24/94	Hf Absorbers Inserted

Ref. Core Power = 1518 MWt

MONTH	THERMAL GENERATION (MW-Hr)	MONTH	THERMAL GENERATION (MW-Hr)	MONTH	THERMAL GENERATION (MW-Hr)
11/91	475898	11/92	393930	10/93	13664
12/91	1064971	12/92	1100982	11/93	1044717
1/92	1127958	1/93	1116955	12/93	1112456
2/92	1056750	2/93	1003337	1/94	1113719
3/92	1050400	3/93	1065780	2/94	992516
4/92	1084547	4/93	1073515	3/94	1115386
5/92	1126747	5/93	1121723	4/94	1082174
6/92	1126747	6/93	1071687	5/94	1114573
7/92	1124583	7/93	1116051	6/94	1092180
8/92	1100487	8/93	1118139	7/94	1126859
9/92	927929	9/93	866899	8/94	1129707
				9/94	821279
TOTAL	11267017	TOTAL	11048998	TOTAL	11759230

The irradiation capsule loading diagram and the measured specific activities of the radiometric monitors from the Cycles 18 through 20 irradiation are provided on pages E-2 through E-9. For the multiple foil sensor sets, the individual foil ID can be correlated with the capsule loading diagram provided in Section 6.4-1 in order to determine the location of the foil within the reactor cavity during irradiation.

CONTENTS OF MULTIPLE FOIL SENSOR SETS
CYCLES 18/20 IRRADIATION

Capsule I.D. and Position	Bare or Cadmium Shielded	Radiometric Monitor I.D.							SSTR Package
		Fe	Ni	Cu	Ti	Nb	Co	U-238	
MM-1	B	AA	-	-	-	-	BA	-	PB-39B
MM-2	Cd	AK	AA	AA	AK	AA	AA	AA	-
MM-3	Cd	-	-	-	-	-	-	-	PB-39C
NN-1	B	AB	-	-	-	-	BB	-	PB-41B
NN-2	Cd	AL	AB	AB	AL	AB	AB	AB	-
NN-3	Cd	-	-	-	-	-	-	-	PB-41C
OO-1	B	AC	-	-	-	-	BC	-	PB-40B
OO-2	Cd	AM	AC	AC	AM	AC	AC	AC	-
OO-3	Cd	-	-	-	-	-	-	-	PB-40C
PP-1	B	AD	-	-	-	-	BD	-	PB-42B
PP-2	Cd	AN	AD	AD	AN	AD	AD	AD	-
PP-3	Cd	-	-	-	-	-	-	-	PB-42C
QQ-1	B	AE	-	-	-	-	BE	-	PB-43B
QQ-2	Cd	AO	AE	AE	AO	AE	AE	AE	-
QQ-3	Cd	-	-	-	-	-	-	-	PB-43C
RR-1	B	AF	-	-	-	-	BF	-	PB-44B
RR-2	Cd	AP	AF	AF	AP	AF	AF	T	-
RR-3	Cd	-	-	-	-	-	-	-	PB-44C

REPORT

Westinghouse Electric Corporation
Chemistry Operations Technology & Analysis
Waltz Mill Site

Request# 15576

Originator: A.H.Fero (W)NTD, Energy Center (4-36)

Received: 12/12/94
Reported: 3/8/95

[RESULTS OF ANALYSIS]
Point Beach Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 2/20/95) dps/mg *		2 sigma
AA	95-016	Fe	Mn-54	1.00E+01	+/-	1.2E-01
AK	95-017	Fe	Mn-54	1.12E+01	+/-	1.2E-01
AB	95-025	Fe	Mn-54	2.85E+01	+/-	2.6E-01
AL	95-026	Fe	Mn-54	2.78E+01	+/-	2.4E-01
AC	95-034	Fe	Mn-54	1.21E+01	+/-	1.7E-01
AM	95-035	Fe	Mn-54	1.08E+01	+/-	1.5E-01
AD	95-043	Fe	Mn-54	2.38E+01	+/-	2.5E-01
AN	95-044	Fe	Mn-54	2.43E+01	+/-	2.6E-01
AE	95-052	Fe	Mn-54	2.08E+01	+/-	2.4E-01
AO	95-053	Fe	Mn-54	2.07E+01	+/-	2.2E-01
AF	95-061	Fe	Mn-54	1.95E+01	+/-	2.2E-01
AP	95-062	Fe	Mn-54	1.98E+01	+/-	2.1E-01
AA	95-019	Cu	Co-60	5.97E-01	+/-	1.8E-02
AB	95-028	Cu	Co-60	1.57E+00	+/-	2.9E-02
AC	95-037	Cu	Co-60	5.49E-01	+/-	1.7E-02
AD	95-046	Cu	Co-60	1.39E+00	+/-	2.8E-02
AE	95-055	Cu	Co-60	1.24E+00	+/-	2.7E-02
AF	95-064	Cu	Co-60	1.22E+00	+/-	2.7E-02
BA	95-022	AlCo	Co-60	4.49E+02	+/-	5.4E+00
AA	95-023	AlCo	Co-60	3.23E+02	+/-	4.6E+00
BB	95-031	AlCo	Co-60	1.13E+03	+/-	1.4E+01
AB	95-032	AlCo	Co-60	6.90E+02	+/-	1.1E+01
BC	95-040	AlCo	Co-60	4.73E+02	+/-	5.6E+00
AC	95-041	AlCo	Co-60	3.17E+02	+/-	4.5E+00
BD	95-049	AlCo	Co-60	1.41E+03	+/-	1.6E+01
AD	95-050	AlCo	Co-60	8.13E+02	+/-	1.2E+01
BE	95-058	AlCo	Co-60	1.26E+03	+/-	1.5E+01
AE	95-059	AlCo	Co-60	7.01E+02	+/-	1.1E+01
BF	95-067	AlCo	Co-60	8.01E+02	+/-	1.2E+01
AF	95-068	AlCo	Co-60	5.65E+02	+/-	9.6E+00

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 15576
Procedures: A-524.
Analyst: WTF, TRK

Approved:

REPORT

Westinghouse Electric Corporation
Chemistry Operations Technology & Analysis
Waltz Mill Site

Request# 15576

Originator: A.H.Fero (W)NTD, Energy Center (4-36)

Received: 12/12/94
Reported: 3/8/95

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 2/20/95) dps/mg *		2 sigma
AK	95-020	Ti	Sc-46	1.60E+00	+/-	4.9E-02
AL	95-029	Ti	Sc-46	3.94E+00	+/-	7.9E-02
AM	95-038	Ti	Sc-46	1.58E+00	+/-	4.9E-02
AN	95-047	Ti	Sc-46	3.34E+00	+/-	7.3E-02
AO	95-056	Ti	Sc-46	3.01E+00	+/-	6.9E-02
AP	95-065	Ti	Sc-46	2.86E+00	+/-	6.7E-02
AA	95-018	Ni	C0-58	7.41E+01	+/-	1.0E+00
AB	95-027	Ni	C0-58	1.74E+02	+/-	1.6E+00
AC	95-036	Ni	C0-58	7.18E+01	+/-	1.0E+00
AD	95-045	Ni	C0-58	1.48E+02	+/-	1.4E+00
AE	95-054	Ni	C0-58	1.28E+02	+/-	1.4E+00
AF	95-063	Ni	C0-58	1.23E+02	+/-	1.3E+00

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 15576
Procedures: A-524.
Analyst: WTF, TRK

Approved: [signature]

REPORT

Westinghouse Electric Corporation
Chemistry Operations Technology & Analysis
Waltz Mill Site

Request# 15576

Originator: A.H.Fero (W)NTD, Energy Center (4-36)

Received: 12/12/94
Reported: 3/8/95

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 2/20/95) dps/mg *		2 sigma
AA	95-024	U-238	Cs-137	1.79E+00	+/-	8.5E-02
AB	95-033	U-238	Cs-137	4.53E+00	+/-	1.1E-01
AC	95-042	U-238	Cs-137	1.73E+00	+/-	6.7E-02
AD	95-051	U-238	Cs-137	4.13E+00	+/-	9.2E-02
AE	95-060	U-238	Cs-137	3.37E+00	+/-	1.0E-01
T	95-069	U-238	Cs-137	3.14E+00	+/-	9.7E-02
AA	95-024	U-238	Ru-103	2.51E+00	+/-	1.1E-01
AB	95-033	U-238	Ru-103	5.63E+00	+/-	1.3E-01
AC	95-042	U-238	Ru-103	2.44E+00	+/-	9.2E-02
AD	95-051	U-238	Ru-103	4.95E+00	+/-	1.4E-01
AE	95-060	U-238	Ru-103	4.14E+00	+/-	1.4E-01
T	95-069	U-238	Ru-103	3.92E+00	+/-	1.2E-01
AA	95-024	U-238	Zr-95	5.78E+00	+/-	1.7E-01
AB	95-033	U-238	Zr-95	1.35E+01	+/-	2.3E-01
AC	95-042	U-238	Zr-95	5.66E+00	+/-	1.6E-01
AD	95-051	U-238	Zr-95	1.22E+01	+/-	2.3E-01
AE	95-060	U-238	Zr-95	9.89E+00	+/-	2.3E-01
T	95-069	U-238	Zr-95	9.58E+00	+/-	2.2E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 15576
Procedures: A-524.
Analyst: WTF, TRK

Approved: Mark Kunithal

REPORT

Westinghouse Electric Corporation
Chemistry Operations Technology & Analysis
Waltz Mill Site

Request# 15576

Originator: A.H.Fero (W)NTD, Energy Center (4-36)

Received: 12/12/94
Reported: 3/25/95

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Bead Chain Tag ID: 0 deg.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 2/20/95 Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
+7.5	95-012-A	1.08E+00	+/-	1.6E-01	1.05E+00	+/-	1.7E-01	6.32E+01	+/-	4.7E-01
+6.5	95-012-B	4.15E+00	+/-	2.9E-01	3.53E+00	+/-	3.3E-01	9.21E+01	+/-	7.1E-01
+5.5	95-012-C	1.06E+01	+/-	6.5E-01	9.72E+00	+/-	7.9E-01	2.10E+02	+/-	1.5E+00
+4.5	95-012-D	1.81E+01	+/-	8.6E-01	1.55E+01	+/-	9.3E-01	2.95E+02	+/-	2.0E+00
+3.5	95-012-E	2.05E+01	+/-	1.1E+00	1.90E+01	+/-	1.2E+00	3.43E+02	+/-	2.2E+00
+2.5	95-012-F	2.26E+01	+/-	1.1E+00	2.01E+01	+/-	1.2E+00	3.83E+02	+/-	2.3E+00
+1.5	95-012-G	2.23E+01	+/-	1.5E+00	1.93E+01	+/-	1.5E+00	3.85E+02	+/-	3.2E+00
+0.5	95-012-H	1.97E+01	+/-	1.1E+00	1.77E+01	+/-	1.2E+00	3.95E+02	+/-	2.7E+00
-0.5	95-012-I	1.63E+01	+/-	1.0E+00	1.50E+01	+/-	1.2E+00	3.96E+02	+/-	2.6E+00
-1.5	95-012-J	1.55E+01	+/-	1.0E+00	1.32E+01	+/-	1.1E+00	3.82E+02	+/-	2.5E+00
-2.5	95-012-K	1.71E+01	+/-	1.1E+00	1.51E+01	+/-	1.2E+00	3.76E+02	+/-	2.5E+00
-3.5	95-012-L	1.88E+01	+/-	1.1E+00	1.68E+01	+/-	1.1E+00	3.59E+02	+/-	2.2E+00
-4.5	95-012-M	1.69E+01	+/-	8.0E-01	1.47E+01	+/-	9.7E-01	2.93E+02	+/-	2.0E+00
-5.5	95-012-N	1.06E+01	+/-	4.5E-01	9.52E+00	+/-	4.5E-01	1.89E+02	+/-	1.0E+00
-6.5	95-012-O	3.49E+00	+/-	3.4E-01	3.35E+00	+/-	3.5E-01	1.42E+02	+/-	8.8E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 15576
Procedures: A-524
Analyst: WTF, TRK

Approved: Mark Kowalski

REPORT

Westinghouse Electric Corporation
Chemistry Operations Technology & Analysis
Waltz Mill Site

Request# 15576

Originator: A.H.Fero (W)NTD, Energy Center (4-36)

Received: 12/12/94
Reported: 3/25/95

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Bead Chain Tag ID: 15 deg.

Feet from Midplane	Lab Sample#	Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
		[<---- dps/mg of chain @ 2/20/95 ---->]								
+7.5	95-013-A	1.11E+00	+/-	1.5E-01	9.14E-01	+/-	1.7E-01	5.84E+01	+/-	4.1E-01
+6.5	95-013-B	3.11E+00	+/-	4.0E-01	2.88E+00	+/-	4.3E-01	9.63E+01	+/-	9.3E-01
+5.5	95-013-C	8.60E+00	+/-	8.3E-01	7.48E+00	+/-	9.0E-01	3.09E+02	+/-	2.1E+00
+4.5	95-013-D	1.56E+01	+/-	1.6E+00	1.26E+01	+/-	1.5E+00	4.61E+02	+/-	3.6E+00
+3.5	95-013-E	1.59E+01	+/-	1.2E+00	1.49E+01	+/-	1.4E+00	5.39E+02	+/-	3.0E+00
+2.5	95-013-F	1.84E+01	+/-	1.4E+00	1.54E+01	+/-	1.4E+00	5.64E+02	+/-	3.1E+00
+1.5	95-013-G	1.68E+01	+/-	1.7E+00	1.50E+01	+/-	1.8E+00	5.49E+02	+/-	3.9E+00
+0.5	95-013-H	1.66E+01	+/-	1.5E+00	1.36E+01	+/-	1.9E+00	5.11E+02	+/-	3.8E+00
-0.5	95-013-I	1.45E+01	+/-	1.5E+00	1.39E+01	+/-	2.2E+00	4.88E+02	+/-	3.7E+00
-1.5	95-013-J	1.43E+01	+/-	1.6E+00	1.24E+01	+/-	1.8E+00	4.61E+02	+/-	3.6E+00
-2.5	95-013-K	1.65E+01	+/-	1.7E+00	1.31E+01	+/-	1.8E+00	4.52E+02	+/-	3.5E+00
-3.5	95-013-L	1.66E+01	+/-	1.7E+00	1.50E+01	+/-	1.7E+00	4.35E+02	+/-	3.5E+00
-4.5	95-013-M	1.30E+01	+/-	9.7E-01	1.20E+01	+/-	1.2E+00	3.59E+02	+/-	2.3E+00
-5.5	95-013-N	8.50E+00	+/-	5.0E-01	7.88E+00	+/-	5.8E-01	2.50E+02	+/-	1.2E+00
-6.5	95-013-O	3.09E+00	+/-	3.3E-01	3.21E+00	+/-	4.3E-01	1.31E+02	+/-	8.5E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 15576
Procedures: A-524
Analyst: WTF, TRK

Approved: Mark Kandrak

REPORT

Westinghouse Electric Corporation
Chemistry Operations Technology & Analysis
Waltz Mill Site

Request# 15576

Originator: A.H.Fero (W)NTD, Energy Center (4-36)

Received: 12/12/94
Reported: 3/25/95

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Bead Chain Tag ID: 30 deg.

Feet from Midplane	Lab Sample#	[<---- dps/mg of chain @ 2/20/95 ---->] Mn-54 dps/mg	Mn-54 2 sigma	Co-58 dps/mg	Co-58 2 sigma	Co-60 dps/mg	Co-60 2 sigma
+7.5	95-014-A	2.90E+00 +/-	4.0E-01	2.50E+00 +/-	5.1E-01	1.53E+02 +/-	1.2E+00
+6.5	95-014-B	8.32E+00 +/-	6.8E-01	6.62E+00 +/-	6.7E-01	2.53E+02 +/-	1.5E+00
+5.5	95-014-C	6.29E+00 +/-	5.5E-01	5.72E+00 +/-	6.0E-01	2.25E+02 +/-	1.4E+00
+4.5	95-014-D	1.08E+01 +/-	8.7E-01	9.44E+00 +/-	1.2E+00	3.33E+02 +/-	2.2E+00
+3.5	95-014-E	1.30E+01 +/-	1.4E+00	1.15E+01 +/-	1.7E+00	4.06E+02 +/-	3.3E+00
+2.5	95-014-F	1.29E+01 +/-	1.4E+00	1.15E+01 +/-	1.4E+00	4.46E+02 +/-	3.5E+00
+1.5	95-014-G	1.24E+01 +/-	1.3E+00	1.21E+01 +/-	1.7E+00	4.52E+02 +/-	3.5E+00
+0.5	95-014-H	1.34E+01 +/-	1.4E+00	1.23E+01 +/-	1.6E+00	4.50E+02 +/-	3.5E+00
-0.5	95-014-I	1.33E+01 +/-	1.3E+00	1.19E+01 +/-	1.5E+00	4.34E+02 +/-	3.5E+00
-1.5	95-014-J	1.40E+01 +/-	1.4E+00	1.19E+01 +/-	1.6E+00	4.27E+02 +/-	3.4E+00
-2.5	95-014-K	1.25E+01 +/-	1.2E+00	1.06E+01 +/-	1.6E+00	4.09E+02 +/-	2.7E+00
-3.5	95-014-L	1.34E+01 +/-	1.3E+00	9.62E+00 +/-	1.3E+00	3.66E+02 +/-	2.3E+00
-4.5	95-014-M	9.54E+00 +/-	1.1E+00	8.49E+00 +/-	1.2E+00	2.84E+02 +/-	2.0E+00
-5.5	95-014-N	6.66E+00 +/-	6.3E-01	5.28E+00 +/-	6.7E+00	1.52E+02 +/-	1.2E+00
-6.5	95-014-O	2.44E+00 +/-	4.4E-01	2.21E+00 +/-	5.6E-01	1.04E+02 +/-	9.9E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 15576
Procedures: A-524.
Analyst: WTF, TRK

Approved: Mark Krudial

REPORT

Westinghouse Electric Corporation
Chemistry Operations Technology & Analysis
Waltz Mill Site

Request# 15576

Originator: A.H.Fero (W)NTD, Energy Center (4-36)

Received: 12/12/94
Reported: 3/25/95

[RESULTS OF ANALYSIS]

Point Beach Reactor Cavity Dosimetry

Bead Chain Tag ID: 45 deg.

Feet from Midplane	Lab Sample#	Mn-54 dps/mg		Mn-54 2 sigma	Co-58 dps/mg		Co-58 2 sigma	Co-60 dps/mg		Co-60 2 sigma
		[<---------------- dps/mg of chain @ 2/20/95 ---------------->]								
+7.5	95-015-A	8.68E-01	+/-	1.3E-01	8.88E-01	+/-	1.8E-01	4.59E+01	+/-	3.6E-01
+6.5	95-015-B	2.44E+00	+/-	1.8E-01	2.17E+00	+/-	2.2E-01	6.87E+01	+/-	4.5E-01
+5.5	95-015-C	6.09E+00	+/-	5.1E-01	5.03E+00	+/-	5.8E-01	1.46E+02	+/-	1.2E+00
+4.5	95-015-D	9.90E+00	+/-	5.8E-01	8.40E+00	+/-	7.3E-01	1.97E+02	+/-	1.4E+00
+3.5	95-015-E	1.05E+01	+/-	6.6E-01	9.91E+00	+/-	7.6E-01	2.37E+02	+/-	1.5E+00
+2.5	95-015-F	1.27E+01	+/-	8.2E-01	1.15E+01	+/-	1.7E+00	2.71E+02	+/-	1.9E+00
+1.5	95-015-G	1.23E+01	+/-	8.8E-01	1.00E+01	+/-	1.1E+00	2.85E+02	+/-	2.0E+00
+0.5	95-015-H	1.26E+01	+/-	9.3E-01	9.85E+00	+/-	1.0E+00	2.83E+02	+/-	2.0E+00
-0.5	95-015-I	1.28E+01	+/-	8.8E-01	1.13E+01	+/-	1.1E+00	2.84E+02	+/-	2.0E+00
-1.5	95-015-J	1.21E+01	+/-	8.3E-01	1.01E+01	+/-	8.9E-01	2.80E+02	+/-	2.0E+00
-2.5	95-015-K	1.19E+01	+/-	9.2E-01	1.13E+01	+/-	1.3E+00	2.69E+02	+/-	2.0E+00
-3.5	95-015-L	1.17E+01	+/-	7.1E-01	1.03E+01	+/-	9.0E-01	2.43E+02	+/-	1.5E+00
-4.5	95-015-M	9.50E+00	+/-	6.4E-01	8.74E+00	+/-	6.9E-01	1.97E+02	+/-	1.4E+00
-5.5	95-015-N	6.10E+00	+/-	4.9E-01	5.47E+00	+/-	5.7E-01	1.36E+02	+/-	1.1E+00
-6.5	95-015-O	2.33E+00	+/-	3.0E-01	2.29E+00	+/-	3.4E-01	1.01E+02	+/-	7.5E-01

Remarks: * Results are in units of dps/(mg of Dosimeter Material).

AL File: 15576
Procedures: A-524
Analyst: WTF, TRK

Approved: Mark Kowalczyk