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WESTINGHOUSE CLASS 3

REACTOR CAVITY NEUTRON MEASUREMENT PROGRAM
FOR
WISCONSIN ELECTRIC POWER COMPANY
POINT BEACH UNIT 1

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EXECUTIVE SUMMARY

At the conclusion of Fuel Cycle 16, a reactor cavity measurement program was instituted at Point Beach Unit 1 to provide continuous monitoring of the beltline region of the reactor pressure vessel. 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.

Prior to Cycle 17, the reactor was operating with a conventional low leakage fuel management strategy. At the onset of Cycle 17, additional neutron flux reduction in the beltline region of the reactor was achieved by the introduction of part length hafnium absorbers in selected fuel assemblies. This flux reduction approach is intended to reduce the maximum exposure experienced by the beltline circumferential weld and the lower shell longitudinal weld located along the 15 degree azimuth.

Based on the continued use of the flux reduction currently in place, the current (EOC 19) maximum fast neutron exposure as well as the projected fast neutron exposure of the vessel beltline materials at 32 and 48 effective full power years of operation is summarized as follows:

MAXIMUM VESSEL FLUENCE (E > 1.0 MeV)
[n/cm2]

	16.1 EFPY	32 EFPY	48 EFPY
Circumferential Weld	1.59E+19	2.43E+19	3.26E+19
Int. Shell Plate	1.62E+19	2.78E+19	3.97E+19
Lower Shell Plate	1.59E+19	2.43E+19	3.26E+19
Int. Longitudinal Weld	1.01E+19	1.78E+19	2.57E+19
Lower Longitudinal Weld	9.63E+18	1.63E+19	2.26E+19
Upper/Int. Shell Circ. Weld	1.54E+18	3.17E+18	4.83E+18
Lower Shell/Head Circ. Weld	< 1.00E+17	< 1.00E+17	< 1.00E+17

As further data are accumulated from subsequent irradiations, the neutron environment in the vicinity of the Unit 1 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 the three-dimensional and potentially non-symmetric flux reduction measures to be accurately accounted for in a manner that would be difficult using analysis alone.

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SECTION 1.0

OVERVIEW OF THE PROGRAM

The Reactor Cavity Neutron Measurement Program [1] initiated at Point Beach Unit 1 at the start of Fuel Cycle 17 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 previously withdrawn internal surveillance capsules and with the results of neutron transport calculations, the reactor cavity neutron dosimetry provides the determination of the neutron exposure of the pressure vessel and the projection of 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; and,

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.

Within the nuclear industry it has been common practice to base estimates of the fast neutron exposure of pressure vessels either directly on the results of neutron transport calculations or on the analytical results normalized to measurements obtained from internal surveillance capsules. However, there are potential drawbacks associated with both of these approaches to exposure assessment.

In performing neutron transport calculations for pressurized water reactors, several design and operational variables have an impact on the magnitude of the analytical prediction of exposure rates within the pressure vessel wall as well as on the uncertainties associated with that prediction. Of particular note in this regard are cycle to cycle variations in core power distributions (particularly with implementation of low leakage loading patterns), variations of water temperature in the downcomer regions of the reactor internals, and deviations in as-built versus design dimensions for the reactor internals and pressure vessel. The manner in which these important variables are treated in the analysis may lead to an increased uncertainty in the exposure evaluations for the pressure vessel; and, these increased uncertainties may well result in the use of overly conservative estimates of vessel embrittlement in the assessment of pressure temperature limitations as well as of the expected service life of the component.

The reactor vessel materials surveillance program [2] consisting of several surveillance capsules attached to the thermal shield in the downcomer region near the pressure vessel wall has been in service since the initial startup of the reactor. The neutron dosimetry contained in these capsules provides measurement capability to determine the fast neutron exposure of the materials test specimens also located within the capsules, but at the same time produces measured data only at a single location within the reactor geometry. Therefore, the surveillance capsule dosimetry, by itself, cannot provide information regarding the azimuthal, radial, and axial gradients of neutron exposure within the pressure vessel. Furthermore, data from internal surveillance capsules are, by design, obtained at rather infrequent intervals; and surveillance measurement locations may not be in proximity to critical areas on the pressure vessel. These limitations place a heavy reliance on analytical results to project exposure levels to the vessel wall as well as to provide predictions of vessel exposure for time periods beyond the last scheduled capsule withdrawal.

With the addition of supplementary passive neutron sensors in the reactor cavity annulus between the reactor vessel wall and the biological shield, the deficiencies in both surveillance dosimetry and analytical prediction can be mitigated and the uncertainties associated with exposure estimates for the pressure vessel can be minimized. With state of the art neutron sensors deployed to establish the absolute magnitude of the azimuthal and axial exposure rate distributions in the reactor cavity, the burden placed on the neutron transport calculation is reduced to the determination of relative neutron energy spectra for sensor set interpretation and relative spatial distributions for extrapolation of the measurement results to positions at the inner radius and through the thickness of the pressure vessel wall. Studies have shown that the operational and design variables cited above that have a strong impact on the calculated magnitude of exposure rates have only a minor effect on both the interpretation of cavity dosimetry and on the extrapolation of measurement results to key vessel locations. It is possible, therefore, to employ cavity measurements and a set of reference neutron transport calculations to produce vessel exposure projections with a reduced uncertainty over that inherent in an approach based on analysis alone. Furthermore, since the cavity neutron measurements are not directly tied to the materials surveillance program, measurement intervals can be chosen to easily provide integral vessel exposure over plant lifetime.

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

Because of this potential shift away from threshold fluence toward an energy dependent damage function for data correlation, ASTM Standard Practice E853, "Analysis and Interpretation of Light Water Reactor Surveillance Results", recommends reporting displacements per iron atom (dpa) along with fluence (E > 1.0 MeV) to provide a data base for future reference. The energy dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693, "Characterizing Neutron Exposures in Ferritic Steels in Terms of Displacements per Atom". The application of the dpa parameter to the assessment of embrittlement gradients has already been promulgated in Revision 2 to Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials".

With the aforementioned views in mind, the Reactor Cavity Measurement Program was established to meet the following objectives:

1 - Determine azimuthal and axial gradients of fast neutron exposure over the beltline region of the reactor pressure vessel.

2 - Provide measurement capability sufficient to allow the determination of pressure vessel exposure in terms of both fluence (E > 1.0 MeV) and iron displacements per atom.

3 - Establish a methodology for the projection of exposure gradients through the thickness of the pressure vessel wall.

4 - 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 19. 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 Cycle 19 exposure of the vessel itself.

Results of exposure evaluations from surveillance capsule dosimetry withdrawn at the end of Fuel Cycles 1, 3, 5, and 11 as well as cavity dosimetry results from Cycles 17 and 18 are incorporated to provide the integrated exposure of the pressure vessel from plant startup through the end of Cycle 19. 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 provided based on the measured flux reduction achieved by the insertion of part length hafnium absorbers in selected fuel assemblies at the onset of Cycle 17. Current evaluations and future projections are provided for the beltline circumferential welds as well as for the upper, intermediate and lower shell longitudinal welds and shell plates that comprise the highly irradiated portions of the reactor vessel.

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.

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 as well as opposite the top and bottom of the active fuel. 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 ± 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, attached 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 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

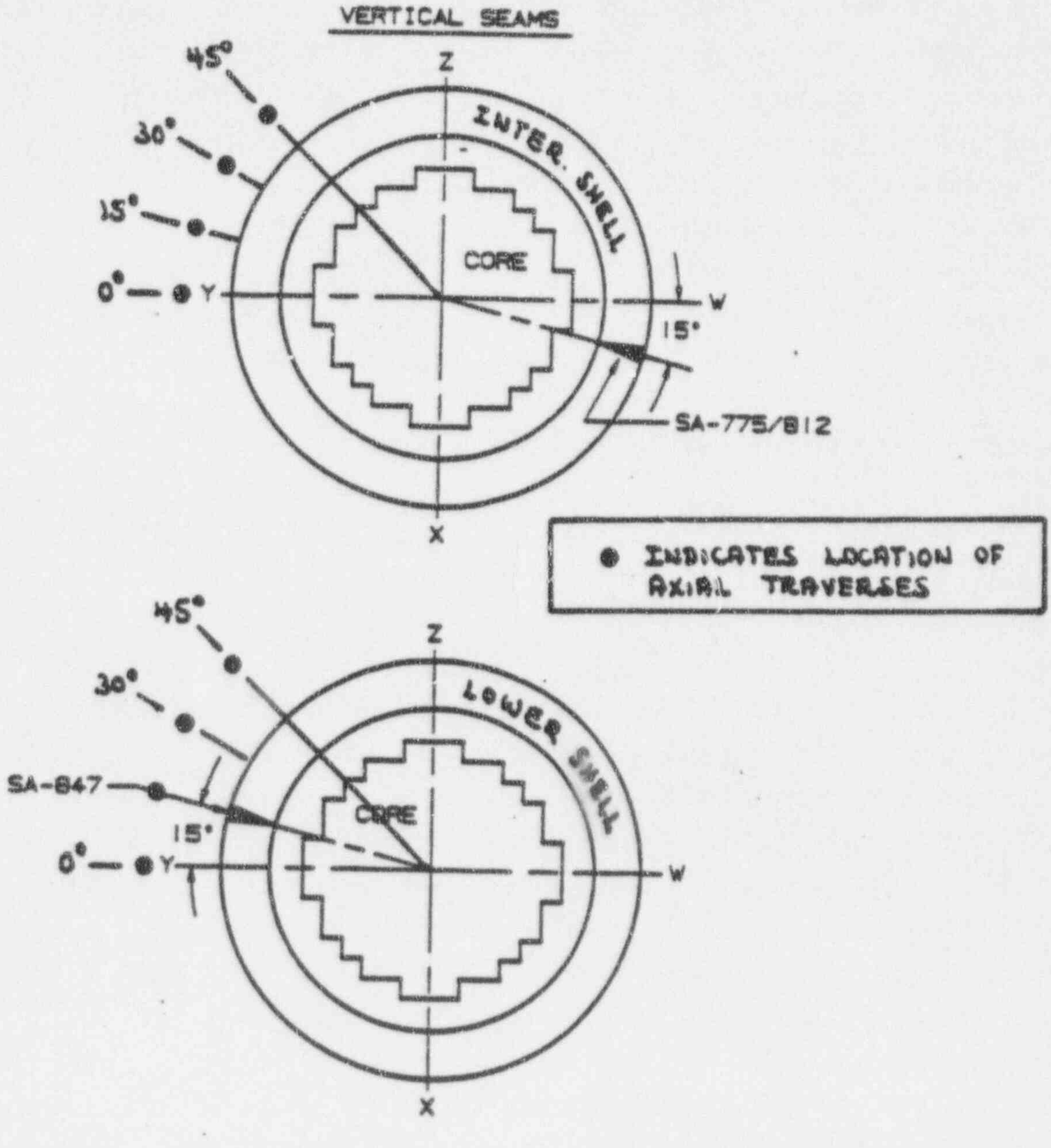


FIGURE 2.1-2

AXIAL LOCATION OF MULTIPLE FOIL SENSOR SETS

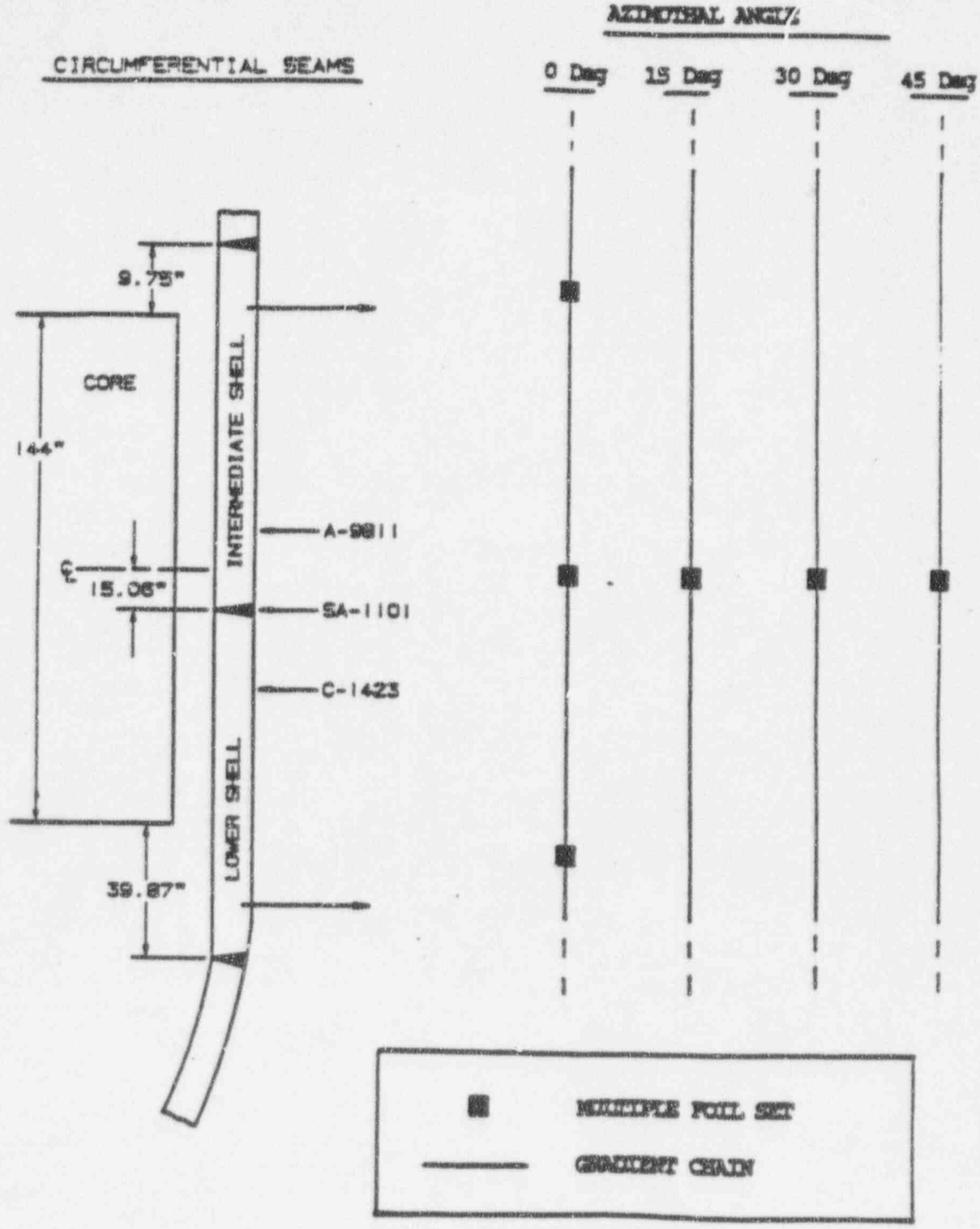
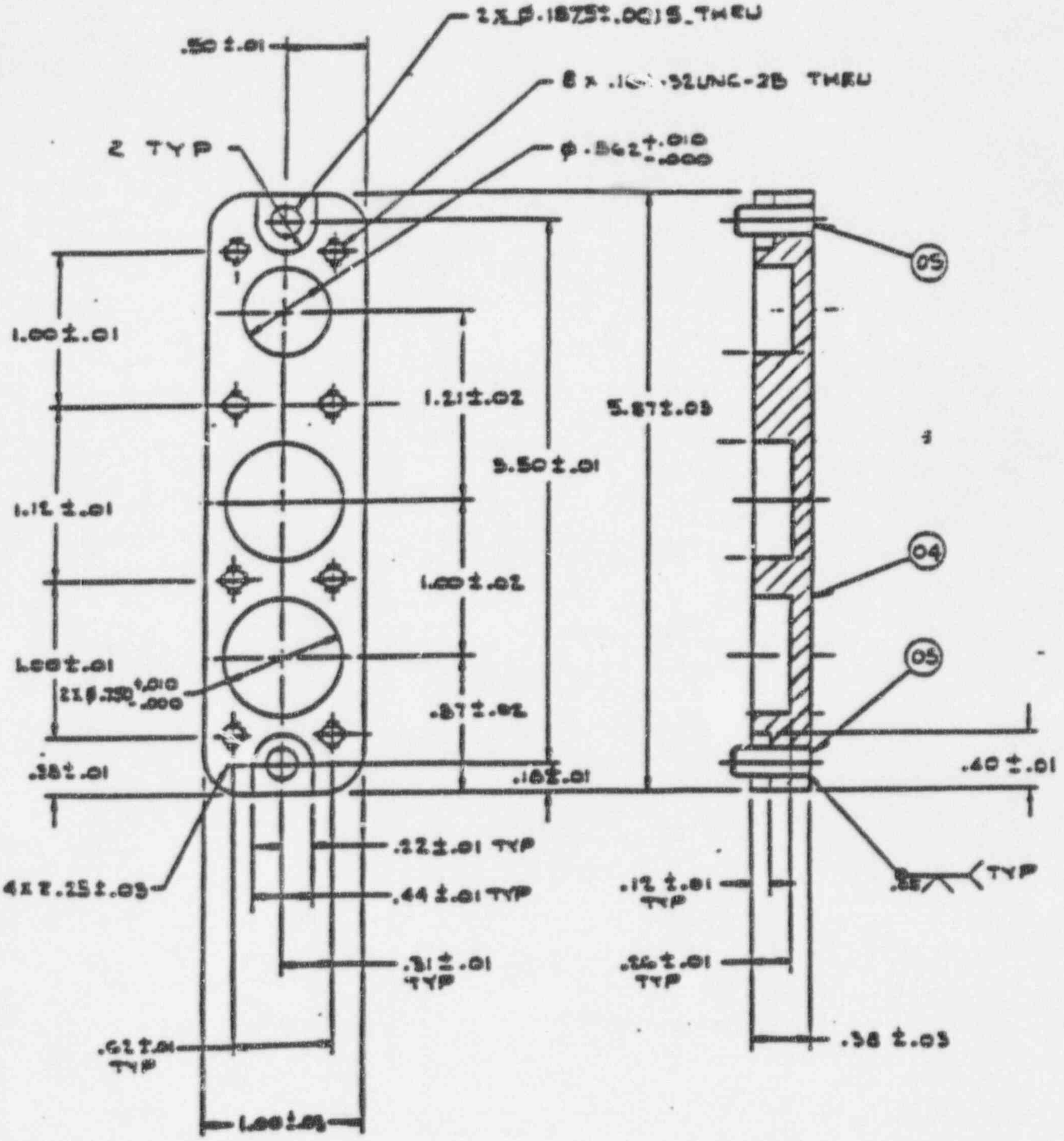


FIGURE 2.1-3

IRRADIATION CAPSULE FOR CAVITY SENSOR SETS



2.2 - Description of Surveillance Capsule Dosimetry

Over the course of the first 11 fuel cycles at Point Beach Unit 1, 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 startup of the reactor and the initiation of the Reactor Cavity Measurement Program. The specific withdrawal dates of these four capsules were as follows:

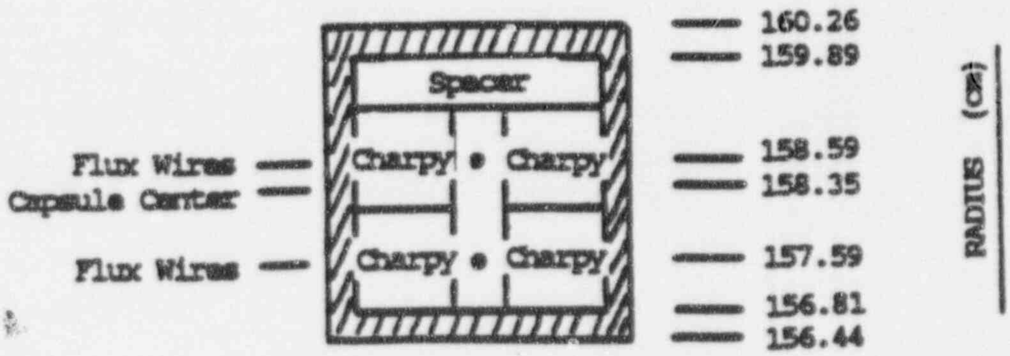
Capsule V	End of Cycle 1	10/72
Capsule S	End of Cycle 3	11/75
Capsule R	End of Cycle 5	10/77
Capsule T	End of Cycle 11	10/83

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, S, R, and T are 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

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 and gamma rays 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 - Extrapolate dosimetry results to key locations at the inner radius and through the thickness of the 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.

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 and gamma ray 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 and 3-2 was carried out in R,θ geometry using the DOT two-dimensional discrete ordinates code [3] and the SAILOR cross-section library [4]. The SAILOR library is a 67 group coupled neutron-gamma ray ENDFB-IV based data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P3 expansion of the cross-sections and the angular discretization was modeled with an S8 order of angular quadrature. The reference forward calculation was normalized to a core midplane power density characteristic of operation at a thermal power level of 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 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 best estimate 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 S8 order of angular quadrature and the P3 cross-section approximation from the SAILOR library. Adjoint source locations were chosen at each of the azimuthal locations containing cavity dosimetry (0, 15, 30, and 45 degrees) as well as at the corresponding azimuths on the pressure vessel inner radius. In addition, adjoint calculations were carried out for sources positioned at the geometric center of surveillance 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, ϕ (E > 1.0 MeV).

The importance functions generated from these eleven 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:

$$\phi(R,\theta) = \int_R \int_\theta \int_E I(R,\theta,E)\ S(R,\theta,E)\ R\ dR\ d\theta\ dE$$

where: $\phi(R,\theta)$ = Flux (E > 1.0 MeV) at radius R and azimuthal angle θ.

$I(R,\theta,E)$ = 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, $I(R,\theta,E)$, permitted 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 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,

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

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

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

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

$$\phi\ (E > 0.1\ \text{MeV}) = [\phi\ (E > 1.0\ \text{MeV})]_{adjoint}\ R_2$$

FIGURE 3.1-1

REACTOR GEOMETRY SHOWING A 45° R,θ SECTOR

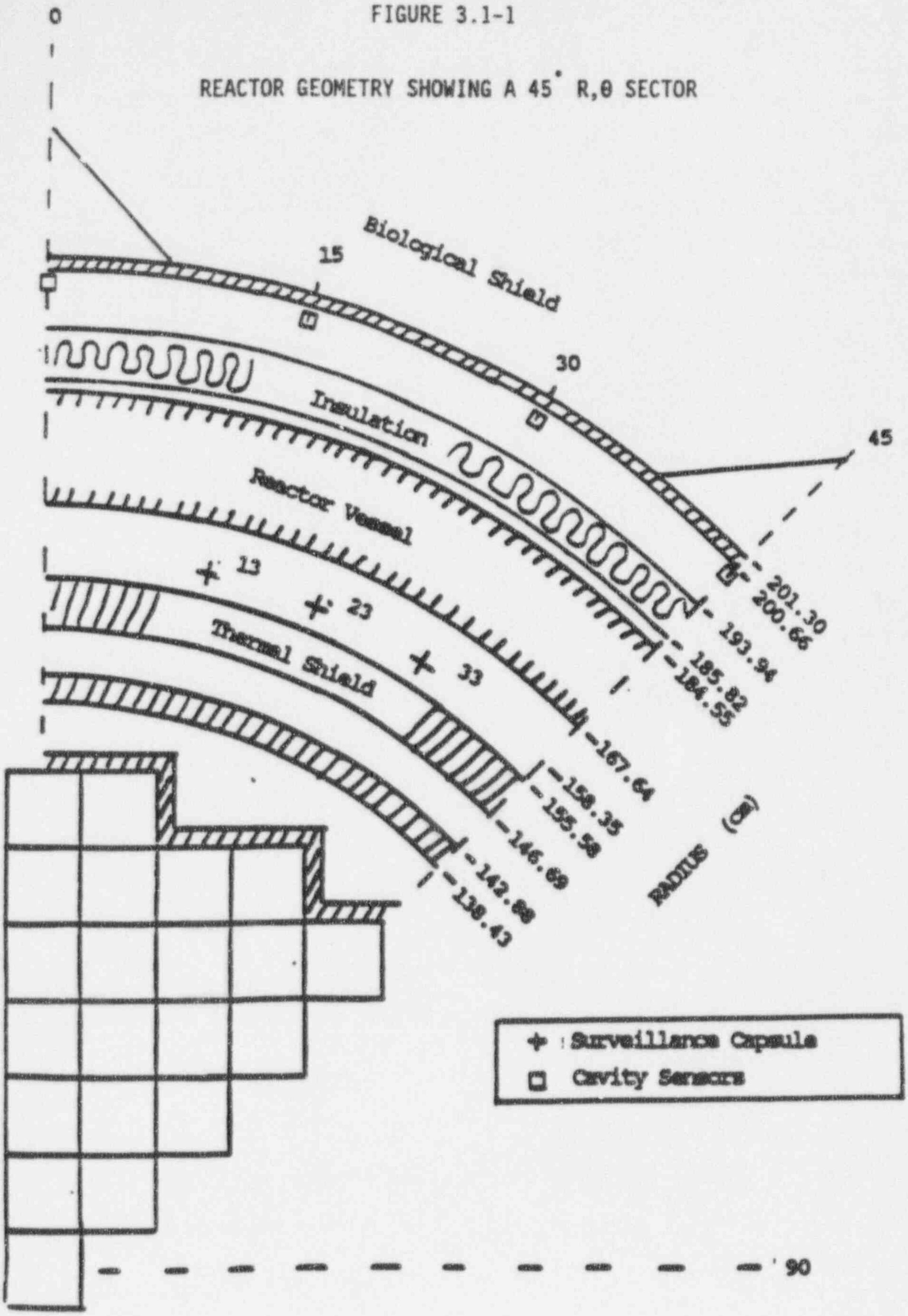
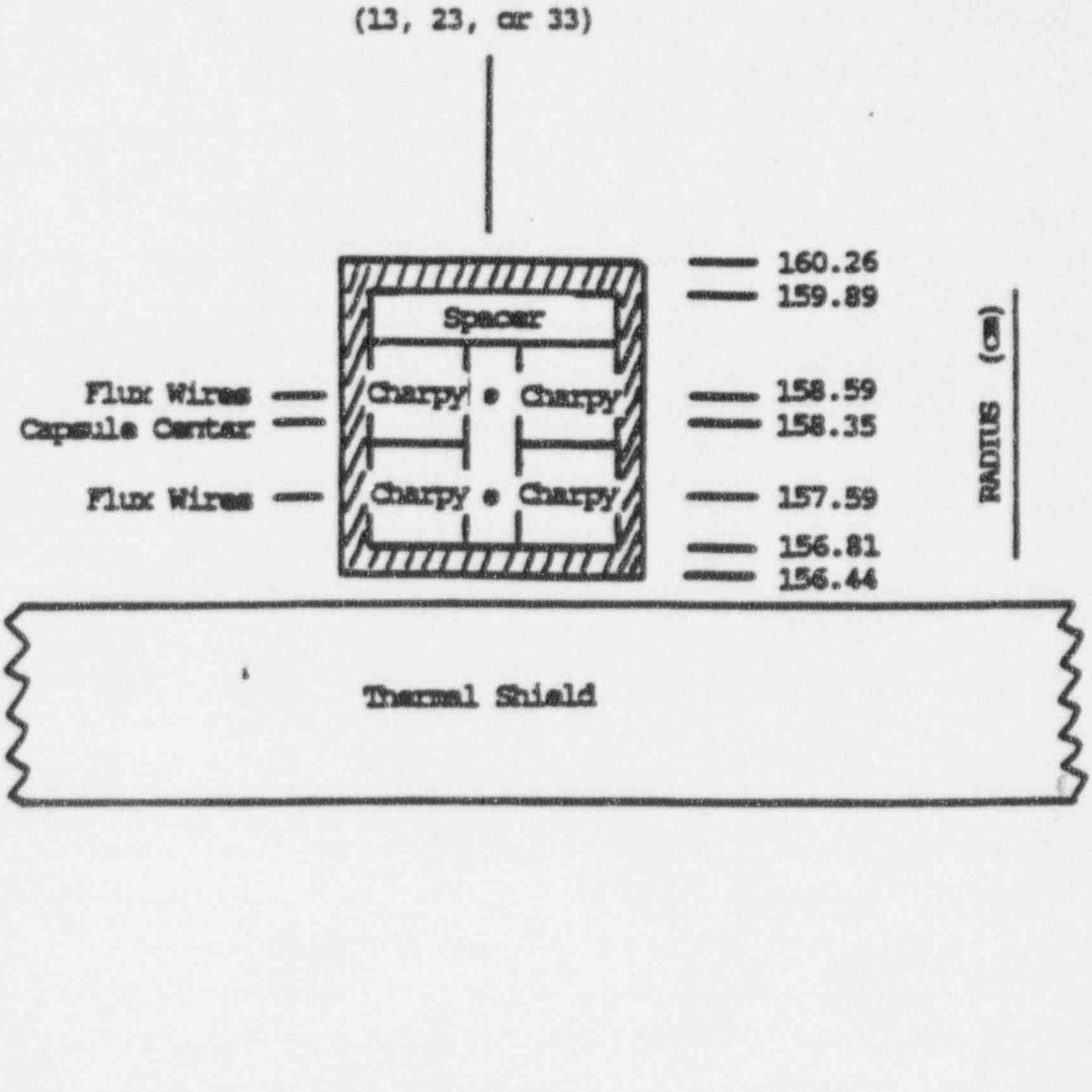


FIGURE 3.1-2

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; and,
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 some cases, the specific activities pertaining to individual internal surveillance capsules were determined from prior analysis by a radiochemical laboratory other than Westinghouse. In those cases, 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 [5 through 15]. 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 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 bases 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:

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

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 ϕ (E > 1.0 MeV) during irradiation period j to the time weighted average ϕ (E > 1.0 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:

$$R = \frac{A}{\sum_j [P_j/P_{ref}]\, t_j}$$

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. Likewise, corrections were made to both U-238 and Np-237 sensors to account for gamma ray induced fission reactions occuring 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 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 procedure required correcting the measured reaction rates by the application of analytically determined spatial gradients. For the Point Beach Unit 1 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 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 [16]. The FERRET approach used the measured reaction rate data and the calculated neutron energy spectrum at the sensor set locations as input and proceeded to adjust a priori (calculated) group fluxes to produce a best fit (in a least squares sense) to the reaction rate data. The exposure parameters along with associated uncertainties were then obtained from the adjusted spectra.

In the FERRET evaluations, a log-normal least-squares algorithm weights both the a priori 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 cross section σ_{ig}. (In this case, FERRET also adjusts the cross-sections.) The log-normal approach automatically accounts for the physical constraint of positive fluxes, even with large assigned uncertainties.

In the FERRET analysis of the dosimetry data, the continuous quantities (i.e., fluxes and cross-sections) were approximated in 53 groups. The calculated fluxes from the reference forward calculation were expanded into the FERRET group structure using the SAND-II code [17]. This procedure was carried out by first expanding the a priori spectrum into the SAND-II 620 group structure using a SPLINE interpolation procedure for interpolation in regions where group boundaries do not coincide. The 620-point spectrum was then easily collapsed to the group scheme used in FERRET.

The cross-sections were also collapsed into the 53 energy-group structure using SAND II with calculated spectra (as expanded to 620 groups) as weighting functions. The cross sections were taken from the ENDF/B-V dosimetry file. Uncertainty estimates and 53 x 53 covariance matrices were constructed for each cross section. Correlations between cross sections were neglected due to data and code limitations, but this omission does not significantly impact the results of the adjustment.

For each set of data or a priori values, the inverse of the corresponding relative covariance matrix M is used as a statistical weight. In some cases, as for the cross sections, a multigroup covariance matrix is used. More often, a simple parameterized form is employed:

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

where R_n specifies an overall fractional normalization uncertainty (i.e., complete correlation) for the corresponding set of values. The fractional uncertainties R_g specify additional random uncertainties for group g that are correlated with a correlation matrix:

$$P_{gg'} = (1-\theta)\, \delta_{gg'} + \theta\, e^{\left[\frac{-(g-g')^2}{2\gamma^2}\right]}$$

The first term specifies purely random uncertainties while the second term describes short-range correlations over a range γ (θ specifies the strength of the latter term).

For the a priori calculated fluxes, a short-range correlation of $\gamma = 6$ groups was used. This choice implies that neighboring groups are strongly correlated when θ is close to 1. Strong long-range correlations (or anticorrelations) were justified based on information presented by R.E. Maerker[18]. Maerker's results are closely duplicated when $\gamma = 6$. 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 calculation were normalized to the measured Fe-54 (n,p) Mn-54 reaction rates to remove any constant calculation to measurement bias and, thus, to permit the adjustment to take place on a relative basis. 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 uncertair ' - in the measured reaction rates and the input (a priori) spectra u .n 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.

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 sensor locations at 0, 15, 30, and 45 degrees relative to the core cardinal axes are listed. These data represent the a priori 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 are provided along with the reference exposure rates in terms of ϕ (E > 1.0 MeV), ϕ (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 ϕ (E > 0.1 MeV) and dpa/sec for each irradiation period. In addition, the ratios of U238(γ,f)/U238(n,f) and Np237(γ,f)/Np237(n,f) were used to make photo-fission corrections to measured reaction rates in the U238 and Np237 fission monitors prior to use in the FERRET adjustment procedure.

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 ϕ (E > 1.0 MeV), ϕ (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 1 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. These factors were used in a multiplicative fashion to relate measured reaction rates to the corresponding value at the geometric center of the capsules.

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 4.1-10.

In Table 4.1-6, the calculated azimuthal distribution of fast neutron flux (E > 1.0 MeV) is listed for the center of the vessel cladding, at the pressure vessel clad/base metal interface, and at the center of the first mesh interval in the base metal. The interface information (base metal inner radius) was obtained by averaging the two sets of data obtained directly from the reference forward calculation. In this detailed tabulation, calculated flux levels are given for each of the 51 azimuthal mesh intervals included in the analytical model model.

In Table 4.1-7, the calculated azimuthal distribution of exposure rates in terms of ϕ (E > 1.0 MeV), ϕ (E > 0.1 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 4.1-7 are the exposure rate ratios [ϕ (E > 0.1 MeV)]/[ϕ (E > 1.0 MeV)] and [dpa/sec]/[ϕ (E > 1.0 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 ϕ (E > 1.0 MeV), ϕ (E > 0.1 Mev), and dpa/sec is given in Tables 4.1-8, 4.1-9, and 4.1-10, 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 were obtained by normalizing the calculated or projected exposure at the vessel inner radius to the gradient data given in Tables 4.1-8 through 4.1-10.

TABLE 4.1-1

CALCULATED REFERENCE NEUTRON ENERGY SPECTRA AT CAVITY SENSOR SET LOCATIONS

1518 MWt; $F_a = 1.2$

LOWER ENERGY	AZIMUTHAL ANGLE				LOWER ENERGY	AZIMUTHAL ANGLE			
(Mev)	0 DEG	15 DEG	30 DEG	45 DEG	(Mev)	0 DEG	15 DEG	30 DEG	45 DEG
1.42E+1	9.07E+5	7.07E+5	5.54E+5	5.30E+5	2.97E-1	3.68E+9	3.13E+9	2.20E+9	1.72E+9
1.22E+1	3.25E+6	2.53E+6	2.00E+6	1.89E+6	1.83E-1	3.87E+9	3.47E+9	2.48E+9	1.86E+9
1.00E+1	9.58E+6	7.25E+6	5.61E+6	5.24E+6	1.11E-1	4.67E+9	4.14E+9	2.97E+9	2.27E+9
8.61E+0	1.59E+7	1.19E+7	9.14E+6	8.45E+6	6.74E-2	3.87E+9	3.45E+9	2.49E+9	1.92E+9
7.41E+0	2.33E+7	1.73E+7	1.31E+7	1.20E+7	4.09E-2	2.20E+9	2.02E+9	1.47E+9	1.11E+9
6.07E+0	4.56E+7	3.37E+7	2.53E+7	2.28E+7	3.18E-2	6.53E+8	6.11E+8	4.47E+8	3.34E+8
4.97E+0	5.56E+7	4.11E+7	3.05E+7	2.69E+7	2.61E-2	3.79E+8	3.58E+8	2.64E+8	1.97E+8
3.68E+0	1.04E+8	7.70E+7	5.58E+7	4.83E+7	2.42E-2	1.39E+9	1.18E+9	8.48E+8	6.84E+8
3.01E+0	9.63E+7	7.14E+7	5.10E+7	4.36E+7	2.19E-2	9.09E+8	8.06E+8	5.76E+8	4.45E+8
2.73E+0	8.52E+7	6.31E+7	4.23E+7	3.79E+7	1.50E-2	1.78E+9	1.66E+9	1.19E+9	8.77E+8
2.47E+0	1.05E+8	7.89E+7	5.56E+7	4.67E+7	7.10E-3	1.67E+9	1.60E+9	1.17E+9	8.67E+8
2.37E+0	5.44E+7	4.12E+7	2.89E+7	2.42E+7	3.36E-3	2.33E+9	2.16E+9	1.59E+9	1.22E+9
2.35E+0	1.87E+7	1.38E+7	9.56E+6	8.07E+6	1.59E-3	1.98E+9	1.84E+9	1.36E+9	1.05E+9
2.23E+0	9.34E+7	6.97E+7	4.85E+7	4.09E+7	4.54E-4	2.70E+9	2.51E+9	1.87E+9	1.45E+9
1.92E+0	2.28E+8	1.71E+8	1.20E+8	1.01E+8	2.14E-4	1.47E+9	1.37E+9	1.03E+9	7.96E+8
1.65E+0	3.56E+8	2.69E+8	1.87E+8	1.56E+8	1.01E-4	1.50E+9	1.39E+9	1.04E+9	8.18E+8
1.35E+0	5.64E+8	4.37E+8	3.04E+8	2.49E+8	3.73E-5	1.96E+9	1.80E+9	1.35E+9	1.07E+9
1.00E+0	1.34E+9	1.06E+9	7.34E+8	5.95E+8	1.07E-5	2.29E+9	2.10E+9	1.58E+9	1.26E+9
8.21E-1	1.25E+9	1.00E+9	6.95E+8	5.57E+8	5.04E-6	1.27E+9	1.16E+9	8.71E+8	6.95E+8
7.43E-1	5.98E+8	5.06E+8	3.55E+8	2.72E+8	1.86E-6	1.46E+9	1.34E+9	1.01E+9	8.05E+8
6.08E-1	2.96E+9	2.41E+9	1.67E+9	1.34E+9	8.76E-7	9.28E+8	8.52E+8	6.42E+8	5.15E+8
4.98E-1	2.25E+9	1.92E+9	1.35E+9	1.04E+9	4.14E-7	7.66E+8	7.04E+8	5.30E+8	4.28E+8
3.69E-1	2.96E+9	2.76E+9	1.77E+9	1.38E+9	1.00E-7	1.35E+9	1.21E+9	9.04E+8	7.49E+8
					0.00	3.68E+9	2.90E+9	2.14E+9	2.04E+9

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

TABLE 4.1-2

CALCULATED NEUTRON SENSOR REACTION RATES AND EXPOSURE RATES
AT THE CAVITY SENSOR SET LOCATIONS
1518 MWt; $F_a = 1.20$

	AZIMUTHAL ANGLE			
	0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
	Reaction Rate (rps/nucleus)			
Cu63(n,α)	1.84E-18	1.39E-18	1.05E-18	9.72E-19
Ti46(n,p)	2.79E-17	2.10E-17	1.56E-17	1.41E-17
Fe54(n,p)	1.63E-16	1.22E-16	8.88E-17	7.81E-17
Ni58(n,p)	2.28E-16	1.72E-16	1.24E-16	1.08E-16
U238(n,f) (Cd)	9.55E-16	7.31E-16	5.12E-16	4.32E-16
Np237(n,f) (Cd)	1.41E-14	1.15E-14	8.01E-15	6.42E-15
Co59(n,γ)	2.60E-13	2.30E-13	1.68E-13	1.44E-13
Co59(n,γ) (Cd)	1.46E-13	1.36E-13	1.01E-13	7.97E-14
U238(γ,f)	3.88E-17	3.13E-17	2.32E-17	2.12E-17
Np237(γ,f)	1.10E-16	8.84E-17	6.54E-17	5.98E-17
	Neutron Flux (n/cm^2)			
φ (E > 1.0 MeV)	3.20E+09	2.53E+09	1.73E+09	1.43E+09
φ (E > 0.1 MeV)	2.54E+10	2.17E+10	1.52E+10	1.19E+10
	dpa/sec			
Displacement rate	9.71E-12	8.14E-12	5.70E-12	4.52E-12
	Exposure Rate Ratios			
φ(E > 0.1)/φ(E > 1.0)	7.94	8.58	8.79	8.32
[dpa/sec]/φ(E > 1.0)	3.03E-21	3.22E-21	3.29E-21	3.16E-21
U238(γ,f)/U238(n,f)	0.0406	0.0428	0.0453	0.0491
Np237(γ,f)/Np237(n,f)	0.00780	0.00769	0.00816	0.00931

TABLE 4.1-3

CALCULATED REFERENCE NEUTRON ENERGY SPECTRA AT SURVEILLANCE CAPSULE LOCATIONS

1518 MWt; F_a = 1.2

LOWER ENERGY (Mev)	AZIMUTHAL ANGLE 13 DEG	23 DEG	33 DEG	LOWER ENERGY (Mev)	AZIMUTHAL ANGLE 13 DEG	23 DEG	33 DEG
1.42E+1	2.31E+07	1.89E+07	1.66E+07	2.97E-1	4.86E+10	2.53E+10	2.34E+10
1.22E+1	8.46E+07	6.90E+07	6.05E+07	1.83E-1	6.29E+10	3.28E+10	3.04E+10
1.00E+1	3.02E+08	2.38E+08	2.08E+08	1.11E-1	6.11E+10	3.15E+10	2.92E+10
8.61E+0	5.52E+08	4.34E+08	3.79E+08	6.74E-2	4.91E+10	2.52E+10	2.34E+10
7.41E+0	9.48E+08	7.19E+08	6.30E+08	4.09E-2	3.71E+10	1.90E+10	1.77E+10
6.07E+0	2.14E+09	1.60E+09	1.40E+09	3.18E-2	1.28E+10	6.56E+09	6.09E+09
4.97E+0	3.03E+09	2.19E+09	1.93E+09	2.61E-2	7.27E+09	3.74E+09	3.47E+09
3.68E+0	7.83E+09	4.30E+09	3.84E+09	2.42E-2	1.38E+10	7.01E+09	6.52E+09
3.01E+0	5.90E+09	3.76E+09	3.40E+09	2.19E-2	8.52E+09	4.31E+09	4.01E+09
2.73E+0	4.83E+09	3.05E+09	2.76E+09	1.50E-2	2.17E+10	1.11E+10	1.03E+10
2.47E+0	5.75E+09	3.57E+09	3.26E+09	7.10E-3	3.39E+10	1.74E+10	1.61E+10
2.37E+0	2.86E+09	1.78E+09	1.62E+09	3.36E-3	4.51E+10	2.31E+10	2.14E+10
2.35E+0	8.75E+08	5.44E+08	4.95E+08	1.59E-3	4.00E+10	2.04E+10	1.89E+10
2.23E+0	4.31E+09	2.66E+09	2.43E+09	4.54E-4	6.16E+10	3.12E+10	2.90E+10
1.92E+0	1.12E+10	6.84E+09	6.25E+09	2.14E-4	3.55E+10	1.80E+10	1.67E+10
1.65E+0	1.49E+10	8.86E+09	8.13E+09	1.01E-4	3.75E+10	1.90E+10	1.76E+10
1.35E+0	2.25E+10	1.31E+10	1.21E+10	3.73E-5	5.02E+10	2.53E+10	2.35E+10
1.00E+0	4.64E+10	2.62E+10	2.42E+10	1.07E-5	6.10E+10	3.07E+10	2.86E+10
8.21E-1	3.49E+10	1.92E+10	1.78E+10	5.04E-6	3.52E+10	1.76E+10	1.64E+10
7.43E-1	1.69E+10	9.27E+09	8.60E+09	1.86E-6	4.31E+10	2.16E+10	2.01E+10
6.08E-1	5.52E+10	2.94E+10	2.72E+10	8.76E-7	2.90E+10	1.46E+10	1.36E+10
4.98E-1	4.10E+10	2.30E+10	2.13E+10	4.14E-7	2.52E+10	1.27E+10	1.18E+10
3.69E-1	5.17E+10	2.72E+10	2.52E+10	1.00E-7	5.48E+10	2.79E+10	2.57E+10
				0.00	1.25E+11	6.52E+10	5.91E+10

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

TABLE 4.1-4

CALCULATED NEUTRON SENSOR REACTION RATES AND EXPOSURE RATES AT THE CENTER OF THE SURVEILLANCE CAPSULES

	AZIMUTHAL ANGLE		
	13 DEGREES	23 DEGREES	33 DEGREES
	Reaction Rate (rps/nucleus)		
Cu63(n,α)	6.82E-17	5.25E-17	4.60E-17
Fe54(n,p)	8.40E-15	5.80E-15	5.17E-15
Ni58(n,p)	1.16E-14	7.82E-15	7.00E-15
U238(n,f) (Cd)	4.42E-14	2.78E-14	2.52E-14
Np237(n,f) (Cd)	4.04E-13	2.31E-13	2.12E-13
Co59(n,γ)	8.81E-12	4.53E-12	4.16E-12
Co59(n,γ) (Cd)	3.76E-12	1.90E-12	1.77E-12
U238(γ,f)	2.23E-15	1.29E-15	1.19E-15
Np237(γ,f)	6.23E-15	3.60E-15	3.32E-15
	Neutron Flux (n/cm^2)		
ϕ (E > 1.0 MeV)	1.33E+11	7.99E+10	7.31E+10
ϕ (E > 0.1 MeV)	5.08E+11	2.78E+11	2.56E+11
	dpa/sec		
Displacement rate	2.46E-10	1.41E-10	1.29E-10
	Exposure Rate Ratios		
ϕ(E > 0.1)/ϕ(E > 1.0)	3.82	3.48	3.50
[dpa/sec]/ϕ(E > 1.0)	1.85E-21	1.76E-21	1.76E-21
U238(γ,f)/U238(n,f)	0.0505	0.0464	0.0472
Np237(γ,f)/Np237(n,f)	0.0154	0.0156	0.0157

TABLE 4.1-5

RADIAL GRADIENT CORRECTIONS FOR SENSORS CONTAINED IN POINT BEACH UNIT 1 INTERNAL SURVEILLANCE CAPSULES

	RADIAL LOCATION (cm)		
	157.59	158.35	158.59
		13 DEGREE CAPSULE	
Cu63(n,α)	0.866	1.000	1.040
Fe54(n,p)	0.856	1.000	1.045
Ni58(n,p)	0.857	1.000	1.046
U238(n,f) (Cd)	0.856	1.000	1.049
Np237(n,f) (Cd)	0.862	1.000	1.050
Co59(n,γ)	0.950	1.000	0.977
Co59(n,γ) (Cd)	0.860	1.000	1.047
		23 DEGREE CAPSULE	
Cu63(n,α)	0.865	1.000	1.040
Fe54(n,p)	0.856	1.000	1.044
Ni58(n,p)	0.856	1.000	1.045
U238(n,f) (Cd)	0.858	1.000	1.048
Np237(n,f) (Cd)	0.866	1.000	1.050
Co59(n,γ)	0.963	1.000	0.972
Co59(n,γ) (Cd)	0.873	1.000	1.042
		33 DEGREE CAPSULE	
Cu63(n,α)	0.867	1.000	1.040
Fe54(n,p)	0.856	1.000	1.045
Ni58(n,p)	0.856	1.000	1.046
U238(n,f) (Cd)	0.856	1.000	1.048
Np237(n,f) (Cd)	0.863	1.000	1.050
Co59(n,γ)	0.956	1.000	0.975
Co59(n,γ) (Cd)	0.868	1.000	1.045

TABLE 4.1-6

AZIMUTHAL VARIATION OF FAST NEUTRON FLUX (E > 1.0 MeV)
AT THE PRESSURE VESSEL INNER RADIUS

THETA	RADIUS (cm)			THETA	RADIUS (cm)		
(Deg)	167.84	168.04	168.27	(Deg)	167.84	168.04	168.27
0.25	4.47E+10	4.41E+10	4.35E+10	23.00	1.97E+10	1.95E+10	1.93E+10
0.75	4.46E+10	4.40E+10	4.34E+10	23.43	1.97E+10	1.94E+10	1.92E+10
2.00	4.42E+10	4.37E+10	4.31E+10	23.72	1.98E+10	1.95E+10	1.91E+10
4.00	4.32E+10	4.27E+10	4.22E+10	23.90	1.97E+10	1.94E+10	1.91E+10
6.00	4.18E+10	4.13E+10	4.07E+10	24.11	1.96E+10	1.93E+10	1.90E+10
8.00	3.96E+10	3.91E+10	3.86E+10	24.61	1.94E+10	1.92E+09	1.90E+10
10.00	3.66E+10	3.62E+10	3.57E+10	26.00	1.93E+10	1.91E+09	1.89E+10
11.39	3.41E+10	3.37E+10	3.32E+10	28.00	1.91E+10	1.89E+09	1.86E+10
11.84	3.32E+10	3.27E+10	3.22E+10	30.00	1.86E+10	1.84E+09	1.82E+10
12.06	3.29E+10	3.24E+10	3.19E+10	31.39	1.81E+10	1.79E+10	1.76E+10
12.28	3.24E+10	3.19E+10	3.14E+10	31.89	1.79E+10	1.77E+10	1.74E+10
12.57	3.16E+10	3.12E+10	3.08E+10	32.10	1.79E+10	1.76E+10	1.73E+10
13.00	3.03E+10	3.00E+10	2.96E+10	32.28	1.79E+10	1.76E+10	1.73E+10
13.44	2.94E+10	2.91E+10	2.87E+10	32.57	1.77E+10	1.75E+10	1.72E+10
13.72	2.90E+10	2.85E+10	2.80E+10	33.00	1.74E+10	1.72E+10	1.70E+10
13.90	2.86E+10	2.82E+10	2.77E+10	33.44	1.73E+10	1.71E+10	1.69E+10
14.12	2.81E+10	2.77E+10	2.73E+10	33.72	1.74E+10	1.71E+10	1.68E+10
14.61	2.71E+10	2.68E+10	2.65E+10	33.90	1.73E+10	1.70E+10	1.67E+10
16.00	2.50E+10	2.47E+10	2.44E+10	34.12	1.72E+10	1.70E+10	1.67E+10
18.00	2.29E+10	2.26E+10	2.23E+10	34.61	1.71E+10	1.69E+10	1.67E+10
20.00	2.12E+10	2.10E+10	2.07E+10	36.00	1.69E+10	1.67E+10	1.65E+10
21.39	2.06E+10	2.04E+10	2.01E+10	38.00	1.66E+10	1.64E+10	1.62E+10
21.88	2.04E+10	2.01E+10	1.98E+10	40.00	1.63E+10	1.61E+10	1.59E+10
22.10	2.03E+10	2.00E+10	1.97E+10	42.00	1.60E+10	1.58E+10	1.56E+10
22.28	2.03E+10	2.00E+10	1.96E+10	44.00	1.58E+10	1.56E+10	1.54E+10
22.56	2.00E+10	1.98E+10	1.95E+10				

Note: The vessel clad/base metal interface is located at a radius of 168.04 cm.

TABLE 4.1-7

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

THETA	Flux (n/cm2-sec)			(E > 0.1)	dpa/sec
(Deg)	(E > 1.0)	(E > 0.1)	dpa/sec	(E > 1.0)	(E > 1.0)
0.25	4.41E+10	1.20E+11	7.26E-11	2.72	1.65E-21
4.00	4.27E+10	1.16E+11	7.04E-11	2.72	1.65E-21
10.00	3.62E+10	1.01E+11	6.04E-11	2.79	1.67E-21
14.61	2.68E+10	7.83E+10	4.57E-11	2.92	1.71E-21
20.00	2.10E+10	5.84E+10	3.53E-11	2.78	1.68E-21
24.61	1.92E+10	5.32E+10	3.22E-11	2.77	1.68E-21
30.00	1.84E+10	4.96E+10	3.05E-11	2.70	1.66E-21
34.61	1.69E+10	4.66E+10	2.82E-11	2.76	1.67E-21
40.00	1.61E+10	4.19E+10	2.64E-11	2.60	1.64E-21
44.00	1.56E+10	4.06E+10	2.56E-11	2.60	1.64E-21

TABLE 4.1-8

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

RADIUS (cm)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
168.04	1.000	1.000	1.000	1.000
168.27	0.986	0.988	0.987	0.988
168.88	0.972	0.976	0.975	0.975
169.75	0.922	0.929	0.927	0.928
170.89	0.841	0.850	0.847	0.850
172.17	0.734	0.744	0.740	0.745
173.49	0.621	0.631	0.627	0.633
174.90	0.519	0.530	0.525	0.531
176.30	0.426	0.437	0.431	0.438
177.50	0.348	0.359	0.353	0.359
178.91	0.291	0.302	0.296	0.302
180.42	0.235	0.247	0.242	0.246
181.51	0.187	0.197	0.192	0.196
182.60	0.156	0.167	0.163	0.165
183.90	0.129	0.141	0.137	0.139
184.55	0.101	0.113	0.111	0.112

Note: Base Metal Inner Radius = 168.04 cm.
1/4 T Location = 172.17 cm.
1/2 T Location = 176.30 cm.
3/4 T Location = 180.42 cm.
Base Metal Outer Radius = 184.55 cm.

TABLE 4.1-9

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

RADIUS	AZIMUTHAL ANGLE			
(cm)	0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
168.04	1.000	1.000	1.000	1.000
168.27	1.005	1.006	1.007	1.007
168.88	1.009	1.013	1.014	1.014
169.75	1.006	1.012	1.014	1.015
170.89	0.981	0.992	0.996	0.998
172.17	0.934	0.949	0.955	0.959
173.49	0.872	0.891	0.898	0.903
174.90	0.803	0.827	0.834	0.840
176.30	0.729	0.757	0.764	0.769
177.50	0.657	0.687	0.695	0.700
178.91	0.595	0.629	0.638	0.641
180.42	0.524	0.562	0.572	0.572
181.51	0.452	0.492	0.503	0.500
182.60	0.400	0.442	0.454	0.450
183.90	0.347	0.393	0.407	0.400
184.55	0.282	0.331	0.350	0.343

Note: Base Metal Inner Radius = 168.04 cm.
1/4 T Location = 172.17 cm.
1/2 T Location = 176.30 cm.
3/4 T Location = 180.42 cm.
Base Metal Outer Radius = 184.55 cm.

TABLE 4.1-10

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

RADIUS (cm)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
168.04	1.000	1.000	1.000	1.000
168.27	0.988	0.990	0.990	0.990
168.88	0.977	0.981	0.980	0.980
169.75	0.937	0.945	0.943	0.943
170.89	0.874	0.886	0.882	0.883
172.17	0.790	0.806	0.801	0.803
173.49	0.701	0.720	0.713	0.715
174.90	0.617	0.639	0.630	0.632
176.30	0.536	0.561	0.552	0.553
177.50	0.465	0.491	0.482	0.482
178.91	0.409	0.437	0.429	0.428
180.42	0.350	0.380	0.372	0.370
181.51	0.294	0.324	0.318	0.314
182.60	0.256	0.287	0.282	0.277
183.90	0.219	0.251	0.249	0.243
184.55	0.177	0.210	0.211	0.205

Note: Base Metal Inner Radius = 168.04 cm.
1/4 T Location = 172.17 cm.
1/^ T Location = 176.30 cm.
3/4 T Location = 180.42 cm.
Base Metal Outer Radius = 184.55 cm.

4.2 - Fuel Cycle Specific Adjoint Calculations

Results of the fuel cycle specific adjoint transport calculations for the first 19 cycles of operation at Point Beach Unit 1 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 17, 18, and 19 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 19 were taken from the fuel cycle design reports applicable to Point Beach Unit 1 [19 through 37]. 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 19 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

ϕ (E > 1.0 MeV) [n/cm2-sec]

CYCLE No	13 DEGREES	23 DEGREES	33 DEGREES
1	1.10E+11	6.45E+10	5.86E+10
2	1.06E+11	6.40E+10	5.81E+10
3	1.10E+11	6.72E+10	6.14E+10
4	1.18E+11	7.15E+10	6.66E+10
5	1.13E+11	6.98E+10	6.52E+10
6	1.06E+11	6.76E+10	6.47E+10
7	1.09E+11	6.62E+10	6.07E+10
8	8.79E+10	5.72E+10	5.36E+10
9	7.77E+10	5.26E+10	4.98E+10
10	7.47E+10	5.32E+10	5.13E+10
11	8.70E+10	5.61E+10	4.93E+10
12	7.32E+10	5.08E+10	4.55E+10
13	7.97E+10	5.67E+10	4.99E+10
14	7.76E+10	5.63E+10	4.97E+10
15	7.76E+10	5.49E+10	4.75E+10
16	7.83E+10	5.61E+10	4.87E+10
17	6.48E+10	4.62E+10	4.42E+10
18	6.07E+10	4.32E+10	4.00E+10
19	6.03E+10	4.58E+10	4.35E+10

TABLE 4.2-2

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

CYCLE No	CYCLE LENGTH (EFPS)	Φ (E > 1.0 MeV) [n/cm2] 13 DEGREES	23 DEGREES	33 DEGREES
1	4.69E+07	5.16E+18	3.03E+18	2.75E+18
2	2.89E+07	8.22E+18	4.87E+18	4.43E+18
3	3.82E+07	1.24E+19	7.44E+18	6.77E+18
4	2.20E+07	1.50E+19	9.01E+18	8.24E+18
5	2.48E+07	1.78E+19	1.07E+19	9.86E+18
6	2.57E+07	2.05E+19	1.25E+19	1.15E+19
7	2.75E+07	2.35E+19	1.43E+19	1.32E+19
8	2.03E+07	2.53E+19	1.55E+19	1.43E+19
9	1.88E+07	2.68E+19	1.65E+19	1.52E+19
10	2.04E+07	2.83E+19	1.75E+19	1.63E+19
11	1.94E+07	3.00E+19	1.86E+19	1.72E+19
12	3.04E+07	3.22E+19	2.02E+19	1.86E+19
13	2.49E+07	3.42E+19	2.16E+19	1.98E+19
14	2.66E+07	3.63E+19	2.31E+19	2.12E+19
15	2.65E+07	3.83E+19	2.45E+19	2.24E+19
16	2.69E+07	4.04E+19	2.60E+19	2.37E+19
17	2.62E+07	4.21E+19	2.73E+19	2.49E+19
18	2.62E+07	4.37E+19	2.84E+19	2.59E+19
19	2.71E+07	4.53E+19	2.96E+19	2.71E+19

TABLE 4.2-3

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

ϕ (E > 1.0 MeV) [n/cm2-sec]

CYCLE No	0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	3.69E+10	2.16E+10	1.48E+10	1.29E+10
2	3.52E+10	2.10E+10	1.47E+10	1.27E+10
3	3.66E+10	2.18E+10	1.55E+10	1.33E+10
4	3.94E+10	2.32E+10	1.67E+10	1.45E+10
5	3.69E+10	2.22E+10	1.63E+10	1.43E+10
6	3.51E+10	2.11E+10	1.61E+10	1.43E+10
7	3.65E+10	2.15E+10	1.53E+10	1.31E+10
8	3.01E+10	1.78E+10	1.34E+10	1.27E+10
9	2.61E+10	1.59E+10	1.25E+10	1.19E+10
10	2.38E+10	1.54E+10	1.28E+10	1.25E+10
11	2.98E+10	1.77E+10	1.28E+10	1.08E+10
12	2.31E+10	1.51E+10	1.18E+10	1.01E+10
13	2.46E+10	1.66E+10	1.30E+10	1.09E+10
14	2.38E+10	1.62E+10	1.29E+10	1.08E+10
15	2.40E+10	1.61E+10	1.24E+10	1.02E+10
16	2.43E+10	1.63E+10	1.28E+10	1.04E+10
17	2.06E+10	1.35E+10	1.11E+10	1.06E+10
18	1.92E+10	1.26E+10	1.02E+10	9.29E+09
19	1.85E+10	1.28E+10	1.10E+10	1.02E+10

TABLE 4.2-4

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

CYCLE No	CYCLE LENGTH (EFPS)	Φ (E > 1.0 MeV) [n/cm2] 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	4.69E+07	1.73E+18	1.01E+18	6.94E+17	6.05E+17
2	2.89E+07	2.75E+18	1.62E+18	1.12E+18	9.72E+17
3	3.82E+07	4.15E+18	2.45E+18	1.71E+18	1.48E+18
4	2.20E+07	5.01E+18	2.96E+18	2.08E+18	1.80E+18
5	2.48E+07	5.93E+18	3.51E+18	2.48E+18	2.15E+18
6	2.57E+07	6.83E+18	4.06E+18	2.90E+18	2.52E+18
7	2.75E+07	7.83E+18	4.65E+18	3.32E+18	2.88E+18
8	2.03E+07	8.44E+18	5.01E+18	3.59E+18	3.14E+18
9	1.88E+07	8.94E+18	5.31E+18	3.82E+18	3.36E+18
10	2.04E+07	9.42E+18	5.62E+18	4.09E+18	3.62E+18
11	1.94E+07	1.00E+19	5.96E+18	4.33E+18	3.83E+18
12	3.04E+07	1.07E+19	6.42E+18	4.69E+18	4.13E+18
13	2.49E+07	1.13E+19	6.84E+18	5.02E+18	4.41E+18
14	2.66E+07	1.19E+19	7.27E+18	5.36E+18	4.69E+18
15	2.65E+07	1.26E+19	7.69E+18	5.69E+18	4.96E+18
16	2.69E+07	1.32E+19	8.13E+18	6.03E+18	5.24E+18
17	2.62E+07	1.38E+19	8.49E+18	6.32E+18	5.52E+18
18	2.62E+07	1.43E+19	8.82E+18	6.59E+18	5.76E+18
19	2.71E+07	1.48E+19	9.17E+18	6.89E+18	6.04E+18

TABLE 4.2-5

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

ϕ (E > 1.0 MeV) [n/cm2-sec]

CYCLE No	0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	2.67E+09	2.04E+09	1.40E+09	1.17E+09
2	2.57E+09	1.98E+09	1.38E+09	1.16E+09
3	2.67E+09	2.06E+09	1.45E+09	1.21E+09
4	2.86E+09	2.20E+09	1.57E+09	1.31E+09
5	2.70E+09	2.11E+09	1.52E+09	1.29E+09
6	2.56E+09	2.01E+09	1.50E+09	1.28E+09
7	2.65E+09	2.03E+09	1.43E+09	1.19E+09
8	2.19E+09	1.70E+09	1.27E+09	1.11E+09
9	1.91E+09	1.51E+09	1.17E+09	1.03E+09
10	1.78E+09	1.47E+09	1.19E+09	1.07E+09
11	2.16E+09	1.68E+09	1.20E+09	9.89E+08
12	1.73E+09	1.42E+09	1.08E+09	9.15E+08
13	1.86E+09	1.55E+09	1.19E+09	9.91E+08
14	1.80E+09	1.51E+09	1.18E+09	9.82E+08
15	1.81E+09	1.50E+09	1.14E+09	9.40E+08
16	1.83E+09	1.52E+09	1.17E+09	9.58E+08
17	1.54E+09	1.28E+09	1.03E+09	9.18E+08
18	1.44E+09	1.19E+09	9.43E+08	8.21E+08
19	1.41E+09	1.20E+09	1.01E+09	8.91E+08

TABLE 4.2-6

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

CYCLE No	CYCLE LENGTH (EFPS)	Φ (E > 1.0 MeV) [n/cm2] 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	4.69E+07	1.25E+17	9.57E+16	6.57E+16	5.49E+16
2	2.89E+07	1.99E+17	1.53E+17	1.06E+17	8.84E+16
3	3.82E+07	3.01E+17	2.32E+17	1.61E+17	1.35E+17
4	2.20E+07	3.64E+17	2.80E+17	1.95E+17	1.63E+17
5	2.48E+07	4.31E+17	3.32E+17	2.33E+17	1.95E+17
6	2.57E+07	4.97E+17	3.84E+17	2.72E+17	2.28E+17
7	2.75E+07	5.70E+17	4.40E+17	3.11E+17	2.61E+17
8	2.03E+07	6.14E+17	4.74E+17	3.37E+17	2.84E+17
9	1.88E+07	6.50E+17	5.03E+17	3.59E+17	3.03E+17
10	2.04E+07	6.87E+17	5.33E+17	3.83E+17	3.25E+17
11	1.94E+07	7.29E+17	5.65E+17	4.06E+17	3.44E+17
12	3.04E+07	7.81E+17	6.08E+17	4.39E+17	3.72E+17
13	2.49E+07	8.28E+17	6.47E+17	4.69E+17	3.96E+17
14	2.66E+07	8.75E+17	6.87E+17	5.00E+17	4.23E+17
15	2.65E+07	9.23E+17	7.27E+17	5.30E+17	4.47E+17
16	2.69E+07	9.73E+17	7.68E+17	5.62E+17	4.73E+17
17	2.62E+07	1.01E+18	8.01E+17	5.89E+17	4.97E+17
18	2.62E+07	1.05E+18	8.33E+17	6.14E+17	5.19E+17
19	2.71E+07	1.09E+18	8.66E+17	6.41E+17	5.43E+17

TABLE 4.2-7

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

ϕ (E > 0.1 MeV) [n/cm2-sec]

CYCLE No	13 DEGREES	23 DEGREES	33 DEGREES
1	4.20E+11	2.24E+11	2.05E+11
2	4.05E+11	2.23E+11	2.03E+11
3	4.20E+11	2.34E+11	2.15E+11
4	4.51E+11	2.49E+11	2.33E+11
5	4.32E+11	2.43E+11	2.28E+11
6	4.05E+11	2.35E+11	2.26E+11
7	4.16E+11	2.30E+11	2.12E+11
8	3.36E+11	1.99E+11	1.88E+11
9	2.97E+11	1.83E+11	1.74E+11
10	2.85E+11	1.85E+11	1.80E+11
11	3.32E+11	1.95E+11	1.73E+11
12	2.80E+11	1.77E+11	1.59E+11
13	3.04E+11	1.97E+11	1.75E+11
14	2.96E+11	1.96E+11	1.74E+11
15	2.96E+11	1.91E+11	1.66E+11
16	2.99E+11	1.95E+11	1.70E+11
17	2.48E+11	1.61E+11	1.55E+11
18	2.32E+11	1.50E+11	1.40E+11
19	2.31E+11	1.59E+11	1.52E+11

TABLE 4.2-8

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

CYCLE No	CYCLE LENGTH (EFPS)	Φ (E > 0.1 MeV) [n/cm2] 13 DEGREES	23 DEGREES	33 DEGREES
1	4.69E+07	1.97E+19	1.05E+19	9.62E+18
2	2.89E+07	3.14E+19	1.70E+19	1.55E+19
3	3.82E+07	4.75E+19	2.59E+19	2.37E+19
4	2.20E+07	5.74E+19	3.14E+19	2.88E+19
5	2.48E+07	6.81E+19	3.74E+19	3.45E+19
6	2.57E+07	7.85E+19	4.34E+19	4.03E+19
7	2.75E+07	8.99E+19	4.98E+19	4.62E+19
8	2.03E+07	9.68E+19	5.38E+19	5.00E+19
9	1.88E+07	1.02E+20	5.73E+19	5.32E+19
10	2.04E+07	1.08E+20	6.10E+19	5.69E+19
11	1.94E+07	1.15E+20	6.48E+19	6.03E+19
12	3.04E+07	1.23E+20	7.02E+19	6.51E+19
13	2.49E+07	1.31E+20	7.51E+19	6.94E+19
14	2.66E+07	1.39E+20	8.03E+19	7.41E+19
15	2.65E+07	1.46E+20	8.54E+19	7.85E+19
16	2.69E+07	1.54E+20	9.06E+19	8.31E+19
17	2.62E+07	1.61E+20	9.48E+19	8.71E+19
18	2.62E+07	1.67E+20	9.88E+19	9.08E+19
19	2.71E+07	1.73E+20	1.03E+20	9.49E+19

TABLE 4.2-9

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

ϕ (E > 0.1 MeV) [n/cm2-sec]

CYCLE No	0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	1.00E+11	6.29E+10	4.00E+10	3.35E+10
2	9.57E+10	6.11E+10	3.97E+10	3.30E+10
3	9.96E+10	6.34E+10	4.19E+10	3.46E+10
4	1.07E+11	6.75E+10	4.51E+10	3.77E+10
5	1.00E+11	6.46E+10	4.40E+10	3.72E+10
6	9.55E+10	6.14E+10	4.35E+10	3.72E+10
7	9.93E+10	6.26E+10	4.13E+10	3.41E+10
8	8.19E+10	5.18E+10	3.62E+10	3.30E+10
9	7.10E+10	4.63E+10	3.38E+10	3.09E+10
10	6.47E+10	4.48E+10	3.46E+10	3.25E+10
11	8.11E+10	5.15E+10	3.46E+10	2.81E+10
12	6.28E+10	4.39E+10	3.19E+10	2.63E+10
13	6.69E+10	4.83E+10	3.51E+10	2.83E+10
14	6.47E+10	4.71E+10	3.48E+10	2.81E+10
15	6.53E+10	4.69E+10	3.35E+10	2.65E+10
16	6.61E+10	4.74E+10	3.46E+10	2.70E+10
17	5.60E+10	3.93E+10	3.00E+10	2.76E+10
18	5.22E+10	3.67E+10	2.75E+10	2.42E+10
19	5.04E+10	3.71E+10	2.97E+10	2.64E+10

TABLE 4.2-10

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

CYCLE No	CYCLE LENGTH (EFPS)	Φ (E > 0.1 MeV) [n/cm2] 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	4.69E+07	4.71E+18	2.95E+18	1.87E+18	1.57E+18
2	2.89E+07	7.47E+18	4.71E+18	3.02E+18	2.53E+18
3	3.82E+07	1.13E+19	7.14E+18	4.62E+18	3.85E+18
4	2.20E+07	1.36E+19	8.62E+18	5.61E+18	4.68E+18
5	2.48E+07	1.61E+19	1.02E+19	6.70E+18	5.60E+18
6	2.57E+07	1.86E+19	1.18E+19	7.82E+18	6.56E+18
7	2.75E+07	2.13E+19	1.35E+19	8.96E+18	7.49E+18
8	2.03E+07	2.30E+19	1.46E+19	9.69E+18	8.16E+18
9	1.88E+07	2.43E+19	1.54E+19	1.03E+19	8.74E+18
10	2.04E+07	2.56E+19	1.64E+19	1.10E+19	9.41E+18
11	1.94E+07	2.72E+19	1.74E+19	1.17E+19	9.95E+18
12	3.04E+07	2.91E+19	1.87E+19	1.27E+19	1.07E+19
13	2.49E+07	3.08E+19	1.99E+19	1.35E+19	1.15E+19
14	2.66E+07	3.25E+19	2.12E+19	1.45E+19	1.22E+19
15	2.65E+07	3.42E+19	2.24E+19	1.54E+19	1.29E+19
16	2.69E+07	3.60E+19	2.37E+19	1.63E+19	1.36E+19
17	2.62E+07	3.75E+19	2.47E+19	1.71E+19	1.44E+19
18	2.62E+07	3.88E+19	2.57E+19	1.78E+19	1.50E+19
19	2.71E+07	4.02E+19	2.67E+19	1.86E+19	1.57E+19

TABLE 4.2-11

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

ϕ (E > 0.1 MeV) [n/cm2-sec]

CYCLE No	0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	2.12E+10	1.75E+10	1.23E+10	9.73E+09
2	2.04E+10	1.70E+10	1.21E+10	9.65E+09
3	2.12E+10	1.77E+10	1.27E+10	1.01E+10
4	2.27E+10	1.89E+10	1.38E+10	1.09E+10
5	2.14E+10	1.81E+10	1.34E+10	1.07E+10
6	2.03E+10	1.72E+10	1.32E+10	1.06E+10
7	2.10E+10	1.74E+10	1.26E+10	9.90E+09
8	1.74E+10	1.46E+10	1.12E+10	9.24E+09
9	1.52E+10	1.30E+10	1.03E+10	8.57E+09
10	1.41E+10	1.26E+10	1.05E+10	8.90E+09
11	1.72E+10	1.44E+10	1.05E+10	8.23E+09
12	1.37E+10	1.22E+10	9.49E+09	7.61E+09
13	1.48E+10	1.33E+10	1.05E+10	8.25E+09
14	1.43E+10	1.30E+10	1.04E+10	8.17E+09
15	1.44E+10	1.29E+10	1.00E+10	7.82E+09
16	1.45E+10	1.30E+10	1.03E+10	7.97E+09
17	1.22E+10	1.10E+10	9.05E+09	7.64E+09
18	1.14E+10	1.02E+10	8.29E+09	6.83E+09
19	1.12E+10	1.03E+10	8.85E+09	7.41E+09

TABLE 4.2-12

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

CYCLE No	CYCLE LENGTH (EFPS)	Φ (E > 0.1 MeV) [n/cm2] 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	4.69E+07	9.94E+17	8.21E+17	5.77E+17	4.57E+17
2	2.89E+07	1.58E+18	1.31E+18	9.28E+17	7.35E+17
3	3.82E+07	2.39E+18	1.99E+18	1.41E+18	1.12E+18
4	2.20E+07	2.89E+18	2.40E+18	1.72E+18	1.36E+18
5	2.48E+07	3.43E+18	2.85E+18	2.05E+18	1.63E+18
6	2.57E+07	3.95E+18	3.29E+18	2.39E+18	1.90E+18
7	2.75E+07	4.53E+18	3.77E+18	2.73E+18	2.17E+18
8	2.03E+07	4.88E+18	4.07E+18	2.96E+18	2.36E+18
9	1.88E+07	5.16E+18	4.31E+18	3.15E+18	2.52E+18
10	2.04E+07	5.45E+18	4.57E+18	3.37E+18	2.70E+18
11	1.94E+07	5.79E+18	4.85E+18	3.57E+18	2.86E+18
12	3.04E+07	6.20E+18	5.22E+18	3.86E+18	3.09E+18
13	2.49E+07	6.57E+18	5.55E+18	4.12E+18	3.30E+18
14	2.66E+07	6.95E+18	5.90E+18	4.40E+18	3.52E+18
15	2.65E+07	7.33E+18	6.24E+18	4.66E+18	3.72E+18
16	2.69E+07	7.72E+18	6.59E+18	4.94E+18	3.94E+18
17	2.62E+07	8.04E+18	6.88E+18	5.18E+18	4.14E+18
18	2.62E+07	8.34E+18	7.14E+18	5.39E+18	4.32E+18
19	2.71E+07	8.64E+18	7.42E+18	5.63E+18	4.52E+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 DEGREES	23 DEGREES	33 DEGREES
1	2.03E-10	1.14E-10	1.03E-10
2	1.96E-10	1.13E-10	1.02E-10
3	2.03E-10	1.18E-10	1.08E-10
4	2.18E-10	1.26E-10	1.17E-10
5	2.09E-10	1.23E-10	1.15E-10
6	1.96E-10	1.19E-10	1.14E-10
7	2.02E-10	1.17E-10	1.07E-10
8	1.63E-10	1.01E-10	9.43E-11
9	1.44E-10	9.26E-11	8.76E-11
10	1.38E-10	9.36E-11	9.03E-11
11	1.61E-10	9.87E-11	8.68E-11
12	1.35E-10	8.94E-11	8.01E-11
13	1.47E-10	9.98E-11	8.78E-11
14	1.44E-10	9.91E-11	8.75E-11
15	1.44E-10	9.66E-11	8.36E-11
16	1.45E-10	9.87E-11	8.57E-11
17	1.20E-10	8.13E-11	7.78E-11
18	1.12E-10	7.60E-11	7.04E-11
19	1.12E-10	8.06E-11	7.65E-11

TABLE 4.2-14

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

CYCLE No	CYCLE LENGTH (EFPS)	Displacements [dpa] 13 DEGREES	23 DEGREES	33 DEGREES
1	4.69E+07	9.54E-03	5.32E-03	4.84E-03
2	2.89E+07	1.52E-02	8.58E-03	7.79E-03
3	3.82E+07	2.30E-02	1.31E-02	1.19E-02
4	2.20E+07	2.78E-02	1.59E-02	1.45E-02
5	2.48E+07	3.30E-02	1.89E-02	1.73E-02
6	2.57E+07	3.80E-02	2.20E-02	2.03E-02
7	2.75E+07	4.36E-02	2.52E-02	2.32E-02
8	2.03E+07	4.69E-02	2.72E-02	2.51E-02
9	1.88E+07	4.96E-02	2.90E-02	2.68E-02
10	2.04E+07	5.24E-02	3.09E-02	2.86E-02
11	1.94E+07	5.55E-02	3.28E-02	3.03E-02
12	3.04E+07	5.96E-02	3.55E-02	3.27E-02
13	2.49E+07	6.33E-02	3.80E-02	3.49E-02
14	2.66E+07	6.71E-02	4.06E-02	3.72E-02
15	2.65E+07	7.09E-02	4.32E-02	3.95E-02
16	2.69E+07	7.48E-02	4.58E-02	4.18E-02
17	2.62E+07	7.80E-02	4.80E-02	4.38E-02
18	2.62E+07	8.09E-02	5.00E-02	4.56E-02
19	2.71E+07	8.39E-02	5.22E-02	4.77E-02

TABLE 4.2-15

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

	Displacement Rate [dpa/sec]			
CYCLE No	0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	6.09E-11	3.69E-11	2.46E-11	2.12E-11
2	5.81E-11	3.59E-11	2.44E-11	2.08E-11
3	6.04E-11	3.73E-11	2.57E-11	2.18E-11
4	6.50E-11	3.97E-11	2.77E-11	2.38E-11
5	6.09E-11	3.80E-11	2.71E-11	2.35E-11
6	5.79E-11	3.61E-11	2.67E-11	2.35E-11
7	6.02E-11	3.68E-11	2.54E-11	2.15E-11
8	4.97E-11	3.04E-11	2.22E-11	2.08E-11
9	4.31E-11	2.72E-11	2.08E-11	1.95E-11
10	3.93E-11	2.63E-11	2.12E-11	2.05E-11
11	4.92E-11	3.03E-11	2.12E-11	1.77E-11
12	3.81E-11	2.58E-11	1.96E-11	1.66E-11
13	4.06E-11	2.84E-11	2.16E-11	1.79E-11
14	3.93E-11	2.77E-11	2.14E-11	1.77E-11
15	3.96E-11	2.75E-11	2.06E-11	1.67E-11
16	4.01E-11	2.79E-11	2.12E-11	1.71E-11
17	3.40E-11	2.31E-11	1.84E-11	1.74E-11
18	3.17E-11	2.15E-11	1.69E-11	1.52E-11
19	3.06E-11	2.18E-11	1.83E-11	1.66E-11

TABLE 4.2-16

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

CYCLE No	CYCLE LENGTH (EFPS)	Displacements [dpa] 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	4.69E+07	2.86E-03	1.73E-03	1.15E-03	9.92E-04
2	2.89E+07	4.53E-03	2.77E-03	1.86E-03	1.59E-03
3	3.82E+07	6.84E-03	4.19E-03	2.84E-03	2.43E-03
4	2.20E+07	8.27E-03	5.07E-03	3.45E-03	2.95E-03
5	2.48E+07	9.78E-C3	6.01E-03	4.12E-03	3.53E-03
6	2.57E+07	1.13E-02	6.94E-03	4.81E-03	4.13E-03
7	2.75E+07	1.29E-02	7.95E-03	5.51E-03	4.73E-03
8	2.03E+07	1.39E-02	8.56E-03	5.96E-03	5.15E-03
9	1.88E+07	1.47E-02	9.08E-03	6.35E-03	5.52E-03
10	2.04E+07	1.55E-02	9.61E-03	6.78E-03	5.93E-03
11	1.94E+07	1.65E-02	1.02E-02	7.19E-03	6.28E-03
12	3.04E+07	1.77E-02	1.10E-02	7.79E-03	6.78E-03
13	2.49E+07	1.87E-02	1.17E-02	8.33E-03	7.23E-03
14	2.66E+07	1.97E-02	1.24E-02	8.90E-03	7.70E-03
15	2.65E+07	2.08E-02	1.32E-02	9.44E-03	8.14E-03
16	2.69E+07	2.18E-02	1.39E-02	1.00E-02	8.60E-03
17	2.62E+07	2.27E-02	1.45E-02	1.05E-02	9.05E-03
18	2.62E+07	2.36E-02	1.51E-02	1.09E-02	9.45E-03
19	2.71E+07	2.44E-02	1.57E-02	1.14E-02	9.90E-03

TABLE 4.2-17

CALCULATED IRON ATOM DISPLACEMENT RATE AT THE CAVITY SENSOR SET LOCATIONS

	Displacement rate [dpa/sec]			
CYCLE No	0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	8.09E-12	6.57E-12	4.61E-12	3.70E-12
2	7.79E-12	6.38E-12	4.54E-12	3.67E-12
3	8.09E-12	6.63E-12	4.77E-12	3.82E-12
4	8.67E-12	7.08E-12	5.17E-12	4.14E-12
5	8.18E-12	6.79E-12	5.00E-12	4.08E-12
6	7.76E-12	6.47E-12	4.94E-12	4.04E-12
7	8.03E-12	6.54E-12	4.70E-12	3.76E-12
8	6.64E-12	5.47E-12	4.18E-12	3.51E-12
9	5.79E-12	4.86E-12	3.85E-12	3.25E-12
10	5.39E-12	4.73E-12	3.92E-12	3.38E-12
11	6.54E-12	5.41E-12	3.95E-12	3.13E-12
12	5.24E-12	4.57E-12	3.55E-12	2.89E-12
13	5.64E-12	4.99E-12	3.92E-12	3.13E-12
14	5.45E-12	4.86E-12	3.88E-12	3.10E-12
15	5.48E-12	4.83E-12	3.75E-12	2.97E-12
16	5.54E-12	4.89E-12	3.85E-12	3.03E-12
17	4.67E-12	4.12E-12	3.39E-12	2.90E-12
18	4.36E-12	3.83E-12	3.10E-12	2.59E-12
19	4.28E-12	3.87E-12	3.31E-12	2.82E-12

TABLE 4.2-18

CALCULATED IRON ATOM DISPLACEMENTS AT THE CAVITY SENSOR SET LOCATIONS

CYCLE No	CYCLE LENGTH (EFPS)	Displacements [dpa] 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
1	4.69E+07	3.79E-04	3.08E-04	2.16E-04	1.73E-04
2	2.89E+07	6.04E-04	4.92E-04	3.47E-04	2.79E-04
3	3.82E+07	9.14E-04	7.46E-04	5.29E-04	4.25E-04
4	2.20E+07	1.10E-03	9.02E-04	6.43E-04	5.16E-04
5	2.48E+07	1.31E-03	1.07E-03	7.67E-04	6.18E-04
6	2.57E+07	1.51E-03	1.24E-03	8.94E-04	7.22E-04
7	2.75E+07	1.73E-03	1.42E-03	1.02E-03	8.25E-04
8	2.03E+07	1.86E-03	1.53E-03	1.11E-03	8.96E-04
9	1.88E+07	1.97E-03	1.62E-03	1.18E-03	9.57E-04
10	2.04E+07	2.08E-03	1.72E-03	1.26E-03	1.03E-03
11	1.94E+07	2.21E-03	1.82E-03	1.34E-03	1.09E-03
12	3.04E+07	2.37E-03	1.96E-03	1.44E-03	1.17E-03
13	2.49E+07	2.51E-03	2.08E-03	1.54E-03	1.25E-03
14	2.66E+07	2.65E-03	2.21E-03	1.65E-03	1.34E-03
15	2.65E+07	2.80E-03	2.34E-03	1.75E-03	1.41E-03
16	2.69E+07	2.95E-03	2.47E-03	1.85E-03	1.50E-03
17	2.62E+07	3.07E-03	2.58E-03	1.94E-03	1.57E-03
18	2.62E+07	3.18E-03	2.68E-03	2.02E-03	1.64E-03
19	2.71E+07	3.30E-03	2.78E-03	2.11E-03	1.72E-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 1 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 DEGREES	END OF CYCLE 1	4.69E+07
S	33 DEGREES	END OF CYCLE 3	1.14E+08
R	13 DEGREES	END OF CYCLE 5	1.61E+08
T	23 DEGREES	END OF CYCLE 11	2.93E+08

5.1 - Measured Reaction Rates

With the exception of Capsule V, radiometric counting of each of these data sets was accomplished by Westinghouse [38, through 40] using the procedures discussed in Section 3.0 of this report. 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 [41]. However, in this case, the measured specific activities were not published.

The irradiation history of the Point Beach Unit 1 reactor during the first 11 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 S, R, and T), 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 4.0 of this report.

Based on the irradiation history and associated flux level adjustment factors, the individual sensor characteristics, 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 S, R, and T. 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 41. 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, for 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 include adjustments for γ,f reactions occuring over 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 Tables 5.2-1 through 5.2-4. In these tables, the derived exposure experienced by each capsule along with data illustrating the fit of both the a priori 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 5.2-1 through 5.2-4, it should be noted that the columns labeled "a priori calc" were obtained by normalizing the neutron spectral data from Table 4.1-3 to the measured

Fe-54 (n,p) reaction rates from each sensor set as discussed in Section 3.0. Thus, the comparisons illustrated in Tables 5.2-1 through 5.2-4 indicate only the degree to which the relative neutron energy spectra matched the measured sensor data before and after adjustment. These data are not meant to provide an absolute comparison of calculation and measurement. Absolute comparisons are discussed 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 RATE (rps/nucleus)			
REACTION	CAPSULE V	CAPSULE S	CAPSULE R	CAPSULE T
*Cu63(nα)Co60	8.22E-17	4.41E-17	7.06E-17	5.31E-17
*Fe54(np)Mn54	6.39E-15	4.66E-15	8.49E-15	5.01E-15
*Ni58(np)Co58	7.90E-15	5.56E-15	1.17E-14	
U238(nf)Cs137	3.44E-14	2.40E-14	4.98E-14	2.82E-14
Np237(nf)Cs137	3.93E-13	2.16E-13	4.22E-13	2.44E-13
*Co59(nγ)Co60		2.02E-12	7.86E-12	4.09E-12
Co59(nγ)Co60		8.13E-13	4.37E-12	1.88E-12

* - Bare foil, all others were cadmium covered

TABLE 5.2-1

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

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.03E+11	1.07E+11	8%
ϕ (E > 0.1 Mev)	3.93E+11	4.88E+11	16%
ϕ (E < 0.414 ev)	1.36E+11	1.56E+11	82%
ϕ (Total)	1.01E+12	1.24E+12	22%
dpa/sec	1.85E-10	2.13E-10	11%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
SURVEILLANCE CAPSULE V

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	8.22E-17	7.16E-17	8.00E-17	0.87	0.97
Fe-54 (n,p)	6.39E-15	6.72E-15	6.44E-15	1.05	1.01
Ni-58 (n,p)	7.90E-15	9.04E-15	8.23E-15	1.14	1.04
U-238 (n,f) (Cd)	3.44E-14	3.37E-14	3.36E-14	0.98	0.98
Np-237 (n,f) (Cd)	3.93E-13	3.02E-13	3.65E-13	0.77	0.93

TABLE 5.2-2

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

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	6.60E+10	7.27E+10	8%
ϕ (E > 0.1 Mev)	2.35E+11	2.58E+11	15%
ϕ (E < 0.414 ev)	7.58E+10	5.16E+11	20%
u (total)	5.81E+11	5.22E+11	14%
dpa/sec	1.15E-10	1.24E-10	10%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
SURVEILLANCE CAPSULE S

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	4.41E-17	5.65E-17	4.48E-17	1.28	1.02
Fe-54 (n,p)	4.66E-15	4.60E-15	4.49E-15	0.99	0.96
Ni-58 (n,p)	5.56E-15	6.15E-15	5.72E-15	1.11	1.03
U-238 (n,f) (Cd)	2.40E-14	2.23E-14	2.36E-14	0.93	1.98
Np-237 (n,f) (Cd)	2.16E-13	1.87E-13	2.11E-13	0.87	0.98
Co-59 (n,γ)	2.02E-12	3.26E-12	2.02E-12	1.61	1.00
Co-59 (n,γ) (Cd)	8.13E-13	1.44E-12	8.17E-12	1.78	1.00

TABLE 5.2-3

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

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.36E+11	1.48E+11	8%
ϕ (E > 0.1 Mev)	5.23E+11	5.61E+11	15%
ϕ (E < 0.414 ev)	1.81E+11	1.47E+11	22%
ϕ (Total)	1.35E+12	1.38E+12	14%
dpa/sec	2.46E-10	2.62E-10	10%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
SURVEILLANCE CAPSULE R

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	7.06E-17	9.51E-17	7.22E-17	1.35	1.02
Fe-54 (n,p)	8.49E-15	8.93E-15	8.46E-15	1.05	1.00
Ni-58 (n,p)	1.17E-14	1.20E-14	1.16E-14	1.03	0.99
U-238 (n,f) (Cd)	4.98E-14	4.47E-14	4.72E-14	0.90	0.95
Np-237 (n,f) (Cd)	4.22E-13	4.01E-13	4.32E-13	0.95	1.02
Co-59 (n,γ)	7.86E-12	7.78E-12	7.87E-12	0.99	1.00
Co-59 (n,γ) (Cd)	4.37E-12	3.46E-12	4.35E-12	0.79	1.00

TABLE 5.2-4

DERIVED EXPOSURE RATES FROM SURVEILLANCE CAPSULE T DOSIMETRY
WITHDRAWN AT THE END OF FUEL CYCLE 11

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	6.92E+10	8.25E+10	8%
ϕ (E > 0.1 Mev)	2.44E+11	3.01E+11	15%
ϕ (E < 0.414 ev)	7.98E+10	9.11E+10	20%
ϕ (Total)	6.03E+11	7.10E+11	14%
dpa/sec	1.20E-10	1.43E-10	10%

COMPARISON OF MEASURED AND CALCULATED SENSOR REACTION RATES
SURVEILLANCE CAPSULE T

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	5.31E-17	6.18E-17	5.37E-17	1.16	1.01
Fe-54 (n,p)	5.01E-15	4.95E-15	5.03E-15	0.99	1.00
U-238 (n,f) (Cd)	2.82E-14	2.36E-14	2.69E-14	0.84	0.95
Np-237 (n,f) (Cd)	2.44E-13	1.95E-13	2.41E-13	0.80	0.99
Co-59 (n,γ)	4.09E-12	3.41E-12	4.09E-12	0.83	1.00
Co-59 (n,γ) (Cd)	1.88E-12	1.49E-12	1.88E-12	0.79	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 1 the program was initiated at the beginning of Fuel Cycle 17; and, to date, has included measurement evaluations at the conclusion of Cycles 17, 18, and 19. The evaluation of each set of data was accomplished using a consistent approach based on the methodology discussed in Section 3.0, resulting in an accurate data base defining the exposure of the reactor vessel wall.

6.1 - Cycle 17 Results

6.1.1 - Measured Reaction rates

During the Cycle 17 irradiation, six multiple foil sensor sets and four stainless steel gradient chains were deployed in the reactor 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 [1]:

	CAPSULE IDENTIFICATION		
AZIMUTH (degrees)	CORE TOP	CORE MIDPLANE	CORE BOTTOM
0	A	B	C
15		D	
30		E	
45		F	

The contents of each of these irradiation capsules is specified in Reference 1 and, for completeness, is also included in Appendix B to this report.

The irradiation history of the Point Beach Unit 1 reactor during Cycle 17 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 17 are provided in Table 6.1-1. Corresponding reaction rate data from the the four stainless steel gradient 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) and Np-237 (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 and the effects of γ,f reactions in the U-238 sensors as well for the effects of γ,f reactions in the Np-237 monitors.

6.1.2 - Results of the Least Squares Adjustment Procedure

The results of the application of the least squares adjustment procedure to the six sets of multiple oil measurements obtained from the Cycle 17 irradiation are provided in Tables 6.1-5 through 6.1-10. In these tables, the derived exposure experienced at each sensor set location along with data illustrating the fit of both the a priori 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 6.1-10, it should be noted that the columns labeled "a priori calc" were obtained by normalizing the neutron spectral data from Table 4.1-1 to the measured Fe-54 (n,p) reaction rates from each sensor set as discussed in Section 3.0. Thus, the comparisons illustrated in Tables 6.1-5 through 6.1-10 indicate only the degree to which the relative neutron energy spectra matched the measured data before and after adjustment. These data are not meant to provide an absolute comparison of calculation and measurement. Absolute comparisons are discussed 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) and Ni-58 (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 are given in Tables 6.1-11, 6.1-12, and 6-13, respectively. The distributions of ϕ (E > 1.0 MeV) are depicted graphically in Figures 6.1-1 through 6.1-4. In these graphical presentations, the solid symbols at -6.0, 0.0, and +6.0 feet represent the explicit results of the FERRET evaluations, while the remaining open symbols depict the normalized data from the gradient chains. Note that data at -6.0 and +6.0 feet are present only along the 0 degree traverse. For the 15, 30, and 45 degree traverses, multiple foil measurements were made only on the core midplane.

TABLE 6.1-1

SUMMARY OF REACTION RATES DERIVED FROM MULTIPLE FOIL SENSOR SETS IRRADIATED DURING CYCLE 17

	REACTION RATE (rps/nucleus)					
REACTION	CAPSULE B	CAPSULE D	CAPSULE E	CAPSULE F	CAPSULE A	CAPSULE B
Cu63(n,α)	8.64E-19	7.83E-19	7.12E-19	7.03E-19	3.99E-19	3.70E-19
Ti46(n,p)	1.28E-17	1.14E-17	1.01E-17	9.99E-18	6.36E-18	4.31E-18
Fe54(n,p)	7.44E-17	6.44E-17	5.68E-17	5.47E-17	3.35E-17	2.60E-17
Ni58(n,p)	1.02E-16	9.05E-17	7.99E-17	7.79E-17	5.32E-17	3.60E-17
U238(n,f)	3.89E-16	3.38E-16	2.96E-16	3.14E-16	2.18E-16	1.27E-16
Np237(n,f)	6.26E-15	5.28E-15	4.62E-15	4.35E-15	3.15E-15	1.86E-15
*Co59(n,γ)	9.30E-14	1.16E-13	9.99E-14	7.08E-14	3.45E-14	3.16E-14
Co59(n,γ)	5.15E-14	5.86E-14	5.44E-14	4.38E-14	2.55E-14	2.01E-14
*U235(nf)	1.00E-12	1.31E-12	1.14E-12	7.30E-13	2.95E-13	3.49E-13
U235(nf)	2.51E-13	3.01E-13	2.54E-13	2.37E-13	8.71E-14	5.35E-14

* - Bare foil, all others were cadmium covered

TABLE 6.1-2

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

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0 DEG	15 DEG	30 DEG	45 DEG
+6.5	1.79E-17	1.44E-17	1.33E-17	1.22E-17
+5.5	4.55E-17	3.82E-17	3.17E-17	2.87E-17
+4.5	7.16E-17	5.76E-17	4.49E-17	4.46E-17
+3.5	7.91E-17	6.52E-17	4.99E-17	4.61E-17
+2.5	8.32E-17	7.16E-17	5.45E-17	5.27E-17
+1.5	8.20E-17	6.81E-17	5.25E-17	5.15E-17
+0.5	6.92E-17	6.28E-17	5.54E-17	5.12E-17
-0.5	6.34E-17	5.93E-17	5.30E-17	5.13E-17
-1.5	5.49E-17	5.07E-17	5.24E-17	5.25E-17
-2.5	5.33E-17	5.01E-17	5.08E-17	4.98E-17
-3.5	5.09E-17	4.82E-17	5.08E-17	4.71E-17
-4.5	4.68E-17	4.60E-17	4.27E-17	4.28E-17
-5.5	3.28E-17	3.12E-17	2.88E-17	2.86E-17
-6.5	1.41E-17	1.22E-17	1.12E-17	1.11E-17

TABLE 6.1-3

Ni-58 (n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS IRRADIATED DURING CYCLE 17

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0 DEG	15 DEG	30 DEG	45 DEG
+6.5	2.81E-17	2.14E-17	1.90E-17	1.85E-17
+5.5	7.05E-17	5.57E-17	4.55E-17	4.22E-17
+4.5	1.07E-16	8.36E-17	7.05E-17	6.39E-17
+3.5	1.16E-16	9.39E-17	7.25E-17	6.84E-17
+2.5	1.22E-16	9.92E-17	7.75E-17	7.38E-17
+1.5	1.16E-16	9.71E-17	7.71E-17	7.50E-17
+0.5	9.96E-17	8.77E-17	7.46E-17	7.13E-17
-0.5	8.69E-17	8.28E-17	7.54E-17	7.30E-17
-1.5	7.95E-17	7.46E-17	7.30E-17	7.01E-17
-2.5	7.58E-17	7.13E-17	7.21E-17	7.13E-17
-3.5	7.50E-17	6.84E-17	6.76E-17	6.52E-17
-4.5	6.64E-17	6.31E-17	6.39E-17	6.11E-17
-5.5	5.00E-17	4.51E-17	4.01E-17	4.14E-17
-6.5	2.10E-17	1.84E-17	1.68E-17	1.68E-17

TABLE 6.1-4

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

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0 DEG	15 DEG	30 DEG	45 DEG
+6.5	2.27E-14	2.20E-14	2.04E-14	1.87E-14
+5.5	4.90E-14	7.07E-14	5.82E-14	3.74E-14
+4.5	6.91E-14	1.06E-13	8.21E-14	5.19E-14
+3.5	8.10E-14	1.19E-13	9.52E-14	6.09E-14
+2.5	8.76E-14	1.33E-13	1.03E-13	6.74E-14
+1.5	8.70E-14	1.29E-13	1.03E-13	6.96E-14
+0.5	8.81E-14	1.20E-13	1.03E-13	6.80E-14
-0.5	8.92E-14	1.13E-13	1.00E-13	6.96E-14
-1.5	8.32E-14	1.04E-13	9.90E-14	6.85E-14
-2.5	7.78E-14	9.74E-14	9.25E-14	6.47E-14
-3.5	7.02E-14	8.87E-14	8.65E-14	6.04E-14
-4.5	5.60E-14	7.67E-14	6.80E-14	4.93E-14
-5.5	3.57E-14	5.55E-14	3.67E-14	3.54E-14
-6.5	2.68E-14	3.03E-14	2.56E-14	2.74E-14

TABLE 6.1-5

DERIVED EXPOSURE RATES FROM THE CAPSULE B DOSIMETRY EVALUATION
0 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.49E+09	1.50E+09	8%
ϕ (E > 0.1 Mev)	1.20E+10	1.15E+10	16%
ϕ (E < 0.414 ev)	2.27E+09	1.77E+09	23%
ϕ (Total)	3.25E+10	3.13E+10	15%
dpa/sec	4.99E-12	4.84E-12	14%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	8.64E-19	1.11E-18	8.97E-19	1.28	1.04
Ti-46 (n,p)	1.28E-17	1.34E-17	1.26E-17	1.04	0.99
Fe-54 (n,p)	7.44E-17	7.35E-17	7.25E-17	0.99	0.97
Ni-58 (n,p)	1.02E-16	1.01E-16	1.01E-16	0.99	0.99
U-238 (n,f) (Cd)	3.89E-16	4.30E-16	4.25E-16	1.11	1.09
Np-237 (n,f) (Cd)	6.26E-15	6.38E-15	6.27E-15	1.02	1.00
Co-59 (n,γ)	9.30E-14	1.22E-13	9.69E-14	1.31	1.04
Co-59 (n,γ) (Cd)	5.15E-14	6.08E-14	5.06E-14	1.18	0.98
U-235 (n,f)	1.00E-12	1.11E-12	9.14E-13	1.11	0.91
U-235 (n,f) (Cd)	2.51E-13	2.33E-13	2.49E-13	0.93	0.99

TABLE 6.1-6

DERIVED EXPOSURE RATES FROM THE CAPSULE D DOSIMETRY EVALUATION
15 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.34E+09	1.28E+09	8%
ϕ (E > 0.1 Mev)	1.20E+10	1.06E+10	16%
ϕ (E < 0.414 ev)	2.17E+09	2.39E+09	22%
ϕ (Total)	2.91E+10	2.86E+10	13%
dpa/sec	4.25E-12	3.84E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	7.83E-19	9.70E-19	8.12E-18	1.24	1.04
Ti-46 (n,p)	1.14E-17	1.16E-17	1.12E-17	1.02	0.99
Fe-54 (n,p)	6.44E-17	6.38E-17	6.32E-17	0.99	0.98
Ni-58 (n,p)	9.05E-17	8.77E-17	8.87E-17	0.97	0.98
U-238 (n,f) (Cd)	3.38E-16	3.80E-16	3.66E-16	1.12	1.08
Np-237 (n,f) (Cd)	5.28E-15	6.04E-15	5.43E-15	1.14	1.03
Co-59 (n,γ)	1.16E-13	1.22E-13	1.20E-13	1.06	1.04
Co-59 (n,γ) (Cd)	5.86E-14	6.54E-14	5.79E-14	1.12	0.99
U-235 (n,f)	1.31E-12	1.05E-12	1.20E-12	0.80	0.91
U-235 (n,f) (Cd)	3.01E-13	2.40E-13	2.93E-13	0.80	0.97

TABLE 6.1-7

DERIVED EXPOSURE RATES FROM THE CAPSULE E DOSIMETRY EVALUATION
30 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.12E+09	1.11E+09	8%
ϕ (E > 0.1 Mev)	1.01E+10	9.20E+09	16%
ϕ (E < 0.414 ev)	1.93E+09	1.99E+09	22%
ϕ (Total)	2.53E+10	2.49E+10	13%
dpa/sec	3.56E-12	3.32E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	7.12E-19	8.91E-19	7.38E-19	1.25	1.04
Ti-46 (n,p)	1.01E-17	1.05E-17	1.00E-17	1.04	0.99
Fe-54 (n,p)	5.68E-17	5.58E-17	5.57E-17	0.98	0.98
Ni-58 (n,p)	7.99E-17	7.65E-17	7.82E-17	0.96	0.98
U-238 (n,f) (Cd)	2.96E-16	3.22E-16	3.19E-16	1.09	1.08
Np-237 (n,f) (Cd)	4.62E-15	5.06E-15	4.73E-15	1.10	1.02
Co-59 (n,γ)	9.99E-14	1.10E-13	1.05E-13	1.10	1.06
Co-59 (n,γ) (Cd)	5.44E-14	5.89E-14	5.32E-14	1.08	0.98
U-235 (n,f)	1.14E-12	9.35E-13	1.01E-12	0.82	0.88
U-235 (n,f) (Cd)	2.54E-13	2.15E-13	2.52E-13	0.85	0.99

TABLE 6.1-8

DERIVED EXPOSURE RATES FROM THE CAPSULE F DOSIMETRY EVALUATION
45 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.02E+09	1.10E+09	8%
ϕ (E > 0.1 Mev)	8.59E+09	8.50E+09	16%
ϕ (E < 0.414 ev)	1.94E+09	1.22E+09	25%
ϕ (Total)	2.18E+10	2.17E+10	13%
dpa/sec	3.09E-12	3.11E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	7.03E-19	8.96E-19	7.28E-19	1.27	1.04
Ti-46 (n,p)	9.99E-18	1.04E-17	9.86E-18	1.04	0.99
Fe-54 (n,p)	5.47E-17	5.37E-17	5.43E-17	0.98	0.99
Ni-58 (n,p)	7.79E-17	7.31E-17	7.64E-17	0.94	0.98
U-238 (n,f) (Cd)	3.14E-16	2.98E-16	3.18E-16	0.95	1.01
Np-237 (n,f) (Cd)	4.35E-15	4.44E-15	4.49E-15	1.02	1.03
Co-59 (n,γ)	7.08E-14	1.03E-13	7.40E-14	1.45	1.04
Co-59 (n,γ) (Cd)	4.38E-14	5.09E-14	4.30E-14	1.16	0.98
U-235 (n,f)	7.30E-13	9.31E-13	6.70E-13	1.27	0.92
U-235 (n,f) (Cd)	2.37E-13	1.86E-13	2.31E-13	0.79	0.97

TABLE 6.1-9

DERIVED EXPOSURE RATES FROM THE CAPSULE A DOSIMETRY EVALUATION
0 DEGREE AZIMUTH - CORE TOP

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	6.72E+08	7.73E+08	8%
ϕ (E > 0.1 Mev)	5.41E+09	5.68E+09	16%
ϕ (E < 0.414 ev)	1.03E+09	4.78E+08	25%
ϕ (Total)	1.47E+10	1.36E+10	15%
dpa/sec	2.26E-12	2.37E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	3.99E-19	5.02E-19	4.15E-19	1.26	1.04
Ti-46 (n,p)	6.36E-18	6.04E-18	6.17E-18	0.95	0.97
Fe-54 (n,p)	3.35E-17	3.32E-17	3.44E-17	0.99	1.03
Ni-58 (n,p)	5.32E-17	4.56E-17	5.12E-17	0.86	0.96
U-238 (n,f) (Cd)	2.18E-16	1.94E-16	2.19E-16	0.89	1.01
Np-237 (n,f) (Cd)	3.15E-15	2.88E-15	3.17E-15	0.91	1.01
Co-59 (n,γ)	3.45E-14	5.49E-14	3.70E-14	1.59	1.07
Co-59 (n,γ) (Cd)	2.55E-14	2.75E-14	2.44E-14	1.08	0.96
U-235 (n,f)	2.95E-13	5.00E-13	2.65E-13	1.69	0.90
U-235 (n,f) (Cd)	8.71E-14	1.05E-13	9.12E-14	1.21	1.05

TABLE 6.1-10

DERIVED EXPOSURE RATES FROM THE CAPSULE C DOSIMETRY EVALUATION
0 DEGREE AZIMUTH - CORE BOTTOM

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	5.22E+08	4.69E+08	8%
ϕ (E > 0.1 Mev)	4.20E+09	3.38E+09	16%
ϕ (E < 0.414 ev)	7.98E+08	5.70E+08	20%
ϕ (Total)	1.14E+10	8.85E+09	15%
dpa/sec	1.75E-12	1.44E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	3.70E-19	3.90E-19	3.71E-19	1.05	1.00
Ti-46 (n,p)	4.31E-18	4.69E-18	4.39E-18	1.09	1.02
Fe-54 (n,p)	2.60E-17	2.58E-17	2.54E-17	0.99	0.98
Ni-58 (n,p)	3.60E-17	3.54E-17	3.53E-17	0.98	0.98
U-238 (n,f) (Cd)	1.27E-16	1.51E-16	1.39E-16	1.19	1.09
Np-237 (n,f) (Cd)	1.86E-15	2.24E-15	1.87E-15	1.20	1.01
Co-59 (n,γ)	3.16E-14	4.27E-14	3.50E-14	1.35	1.11
Co-59 (n,γ) (Cd)	2.01E-14	2.13E-14	1.90E-14	1.06	0.94
U-235 (n,f)	3.49E-13	3.88E-13	2.89E-13	1.11	0.83
U-235 (n,f) (Cd)	5.35E-14	8.17E-12	5.82E-14	1.53	1.09

TABLE 6.1-11

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

HEIGHT (ft)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
+6.5	4.28E+08	3.12E+08	2.77E+08	2.72E+08
+6.0	7.73E+08			
+5.5	1.08E+09	8.18E+08	6.61E+08	6.30E+08
+4.5	1.67E+09	1.23E+09	9.77E+08	9.66E+08
+3.5	1.83E+09	1.39E+09	1.05E+09	1.02E+09
+2.5	1.92E+09	1.50E+09	1.13E+09	1.13E+09
+1.5	1.86E+09	1.44E+09	1.11E+09	1.12E+09
+0.5	1.58E+09	1.32E+09	1.12E+09	1.09E+09
0.0	1.50E+09	1.28E+09	1.11E+09	1.10E+09
-0.5	1.42E+09	1.24E+09	1.10E+09	1.11E+09
-1.5	1.26E+09	1.09E+09	1.08E+09	1.10E+09
-2.5	1.21E+09	1.06E+09	1.05E+09	1.08E+09
-3.5	1.18E+09	1.02E+09	1.02E+09	1.00E+09
-4.5	1.06E+09	9.56E+08	9.10E+08	9.25E+08
-5.5	7.73E+08	6.66E+08	5.92E+08	6.22E+08
-6.0	4.69E+08			
-6.5	3.28E+08	2.66E+08	2.39E+08	2.47E+08

TABLE 6.1-12

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

HEIGHT (ft)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
+6.5	3.28E+09	2.58E+09	2.29E+09	2.10E+09
+6.0	5.68E+09			
+5.5	8.29E+09	6.78E+09	5.48E+09	4.87E+09
+4.5	1.28E+10	1.02E+10	8.10E+09	7.46E+09
+3.5	1.40E+10	1.15E+10	8.68E+09	7.85E+09
+2.5	1.47E+10	1.24E+10	9.38E+09	8.72E+09
+1.5	1.43E+10	1.19E+10	9.18E+09	8.69E+09
+0.5	1.21E+10	1.09E+10	9.28E+09	8.45E+09
0.0	1.15E+10	1.06E+10	9.20E+09	8.50E+09
-0.5	1.09E+10	1.03E+10	9.12E+09	8.55E+09
-1.5	9.66E+09	9.04E+09	8.92E+09	8.48E+09
-2.5	9.30E+09	8.78E+09	8.73E+09	8.33E+09
-3.5	9.04E+09	8.44E+09	8.46E+09	7.75E+09
-4.5	8.15E+09	7.92E+09	7.54E+09	7.15E+09
-5.5	5.92E+09	5.51E+09	4.90E+09	4.81E+09
-6.0	3.38E+09			
-6.5	2.52E+09	2.20E+09	1.98E+09	1.91E+09

TABLE 6.1-13

IRON ATOM DISPLACEMENT RATE (dpa/sec) AS A FUNCTION OF AXIAL POSITION WITHIN THE REACTOR CAVITY CYCLE 17 IRRADIATION

HEIGHT (ft)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
+6.5	1.38E-12	9.35E-13	8.28E-13	7.69E-13
+6.0	2.37E-12			
+5.5	3.49E-12	2.46E-12	1.98E-12	1.78E-12
+4.5	5.39E-12	3.69E-12	2.92E-12	2.73E-12
+3.5	5.90E-12	4.16E-12	3.13E-12	2.87E-12
+2.5	6.20E-12	4.49E-12	3.38E-12	3.19E-12
+1.5	6.00E-12	4.33E-12	3.31E-12	3.18E-12
+0.5	5.11E-12	3.95E-12	3.35E-12	3.09E-12
0.0	4.84E-12	3.84E-12	3.32E-12	3.11E-12
-0.5	4.57E-12	3.73E-12	3.29E-12	3.13E-12
-1.5	4.07E-12	3.27E-12	3.22E-12	3.10E-12
-2.5	3.91E-12	3.18E-12	3.15E-12	3.05E-12
-3.5	3.80E-12	3.06E-12	3.05E-12	2.83E-12
-4.5	3.43E-12	2.87E-12	2.72E-12	2.62E-12
-5.5	2.49E-12	2.00E-12	1.77E-12	1.76E-12
-6.0	1.44E-12			
-6.5	1.06E-12	7.98E-13	7.14E-13	6.99E-13

FIGURE 6.1-1

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 17 IRRADIATION

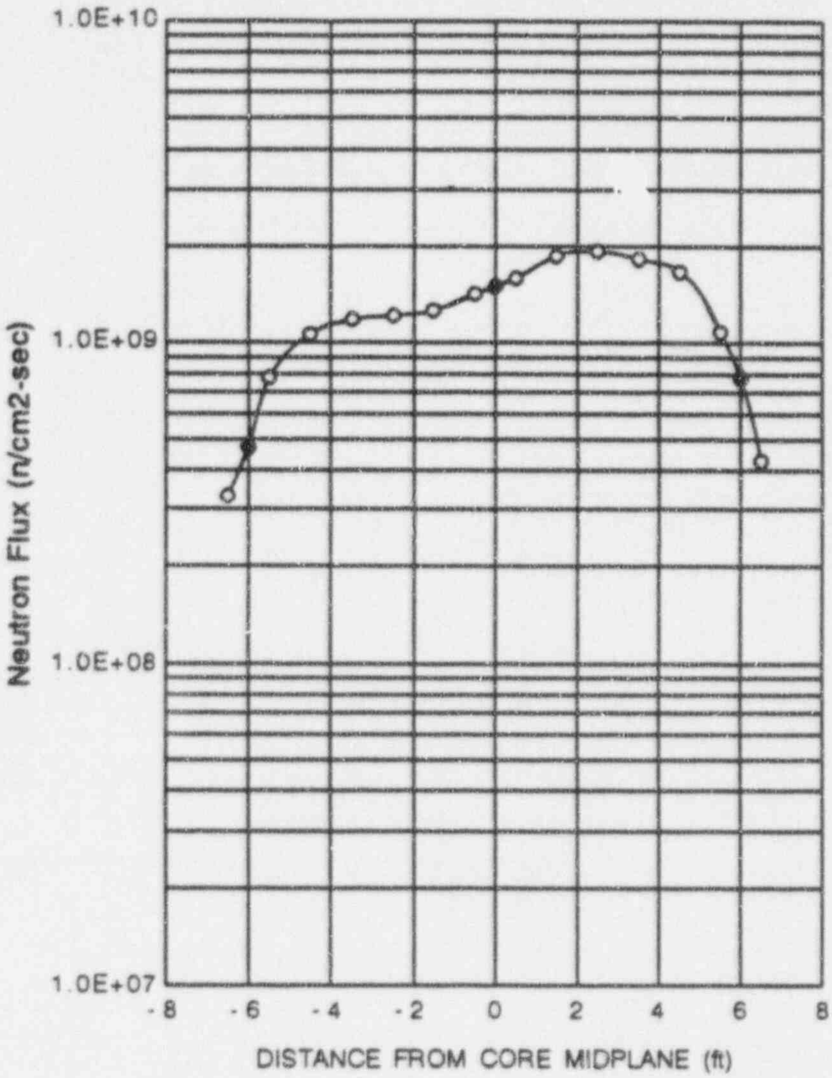


FIGURE 6.1-2

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 15 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 17 IRRADIATION

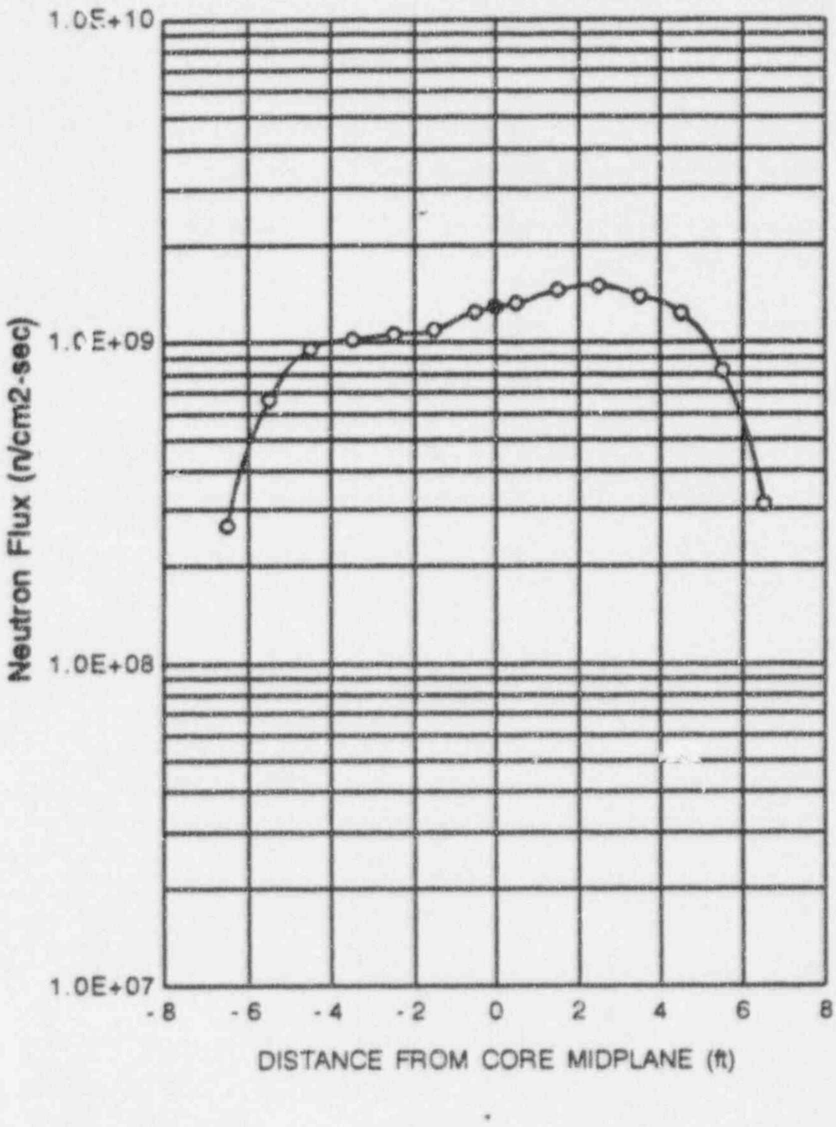


FIGURE 6.1-3

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION ALONG THE 30 DEGREE TRAVERSE IN THE REACTOR CAVITY CYCLE 17 IRRADIATION

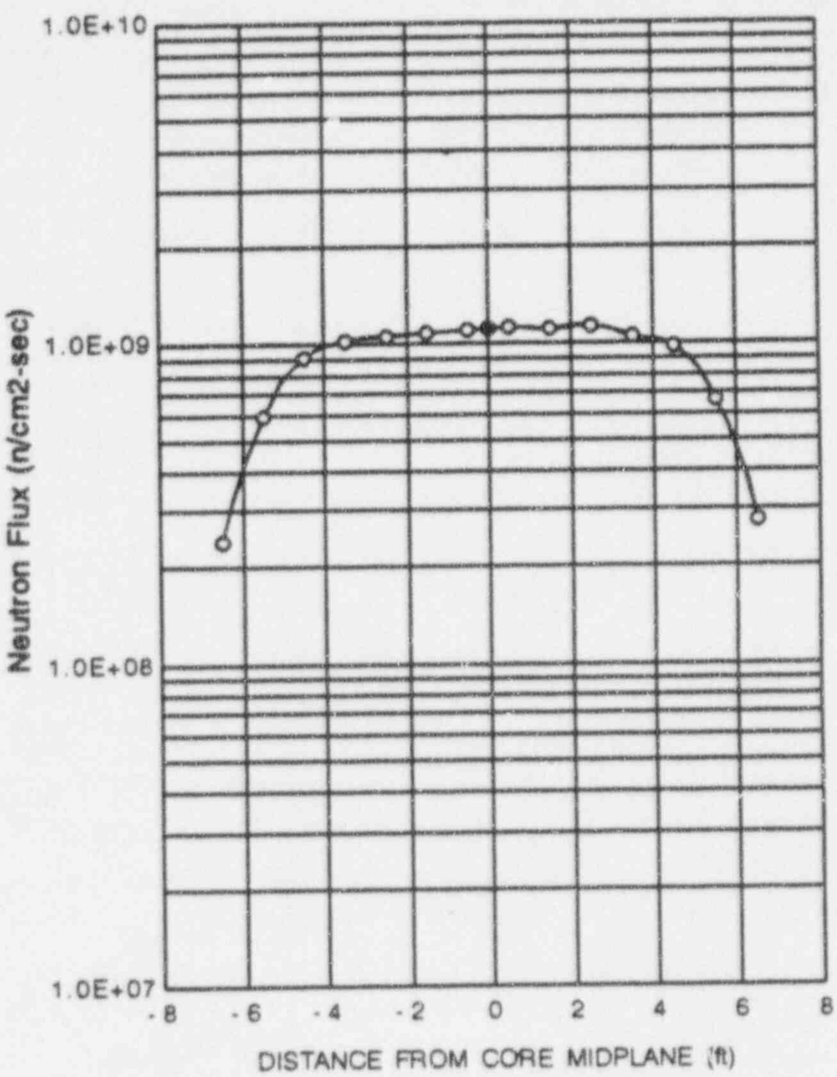
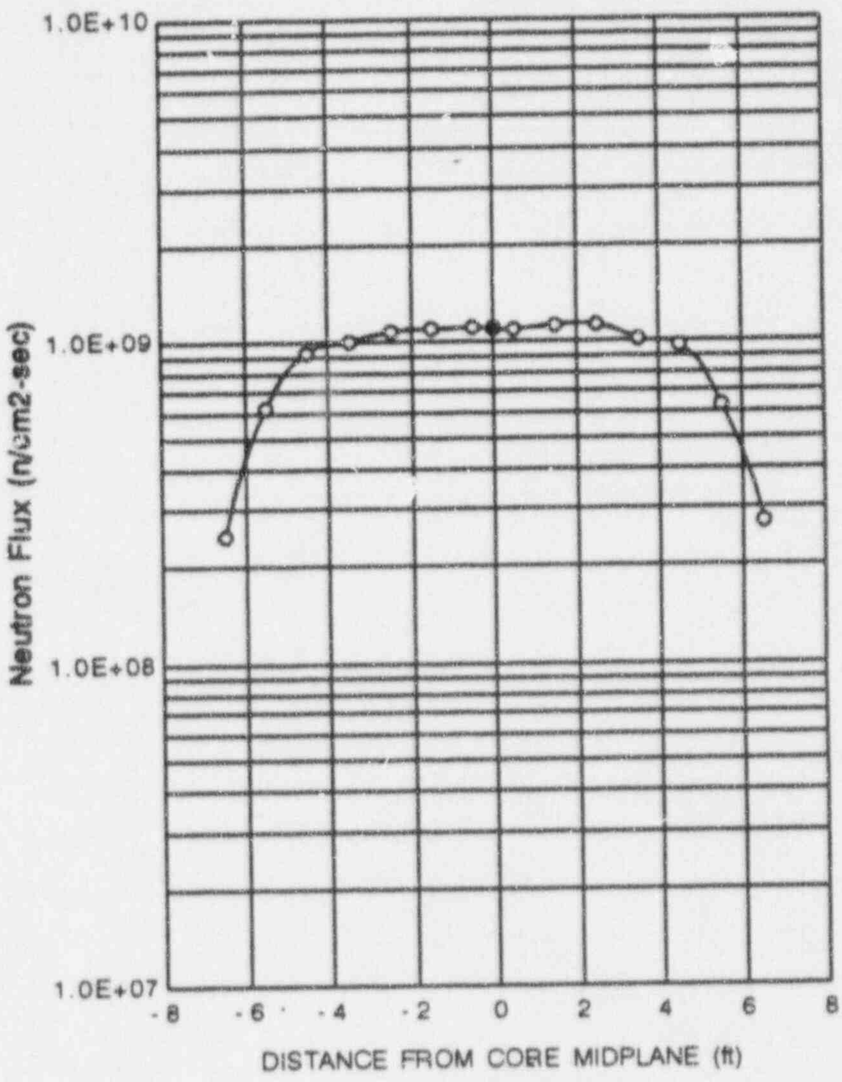


FIGURE 6.1-4

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 45 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 17 IRRADIATION



6.2 - Cycle 18 Results

6.2.1 - Measured Reaction rates

During the Cycle 18 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 [1]:

	CAPSULE IDENTIFICATION		
AZIMUTH (degrees)	CORE TOP	CORE MIDPLANE	CORE BOTTOM
0	S	T	U
15		V	
30		W	
45		X	

The contents of each of these irradiation capsules is specified in Reference 1 and, for completeness, is also included in Appendix C to this report.

The irradiation history of the Point Beach Unit 1 reactor during Cycle 18 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 18 are provided in Table 6.2-1. Corresponding reaction rate data from the the four stainless steel gradient 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 and the effects of γ,f reactions in the U-238 sensors as well for the effects of γ,f reactions in the Np-237 monitors.

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 18 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 a priori 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 "a priori calc" were obtained by normalizing the neutron spectral data from Table 4.1-1 to the measured Fe-54 (n,p) reaction rates from each sensor set as discussed in Section 3.0. Thus, the comparisons illustrated in Tables 6.2-5 through 6.2-10 indicate only the degree to which the relative neutron energy spectra matched the measured data before and after adjustment. These data are not meant to provide an absolute comparison of calculation and measurement. Absolute comparisons are discussed 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) and Ni-58 (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 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, the solid symbols at -6.0, 0.0, and +6.0 feet represent the explicit results of the FERRET evaluations, while the remaining open symbols depict the normalized data from the gradient chains. Note that data at -6.0 and +6.0 feet are present only along the 0 degree traverse. For the 15, 30, and 45 degree traverses, multiple foil measurements were made only at the core midplane.

TABLE 6.2-1

SUMMARY OF REACTION RATES DERIVED FROM MULTIPLE FOIL SENSOR SETS IRRADIATED DURING CYCLE 18

	REACTION RATE (rps/nucleus)					
REACTION	CAPSULE T	CAPSULE V	CAPSULE W	CAPSULE X	CAPSULE S	CAPSULE U
Cu63(n,α)	7.70E-19	7.37E-19	6.78E-19	6.46E-19	5.21E-19	2.02E-19
Ti46(n,p)	1.15E-17	1.12E-17	9.79E-18	9.36E-18	8.06E-18	3.21E-18
Fe54(n,p)	6.76E-17	6.29E-17	5.48E-17	5.14E-17	4.41E-17	1.87E-17
Ni58(n,p)	9.57E-16	8.94E-17	8.02E-17	7.34E-17	6.90E-17	2.72E-17
U238(n,f)	3.57E-16	3.46E-16	2.84E-16	2.69E-16	2.70E-16	1.01E-16
Np237(n,f)	5.07E-15	5.07E-15	4.88E-15	3.96E-15	3.52E-15	1.58E-15
*Co59(n,γ)	8.99E-14	1.13E-13	9.84E-14	6.57E-14	4.21E-14	2.94E-14
Co59(n,γ)	4.92E-14	5.61E-14	5.01E-14	4.09E-14	2.96E-14	1.77E-14
*U235(n,f)	8.20E-13	1.46E-12	1.09E-12	8.54E-13	3.44E-13	3.18E-13
U235(n,f)	2.01E-13	2.88E-13	2.48E-13	2.25E-13	8.70E-14	4.42E-14

* - Bare foil, all others were cadmium covered

TABLE 6.2-2

Fe-54 (n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS IRRADIATED DURING CYCLE 18

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0 DEG	15 DEG	30 DEG	45 DEG
+6.5	2.62E-17	1.69E-17	1.15E-17	1.38E-17
+5.5	5.22E-17	3.90E-17	2.90E-17	3.01E-17
+4.5	7.56E-17	5.80E-17	4.36E-17	4.37E-17
+3.5	7.68E-17	6.00E-17	4.62E-17	4.31E-17
+2.5	7.93E-17	6.41E-17	5.19E-17	4.34E-17
+1.5	7.74E-17	6.41E-17	5.49E-17	4.86E-17
+0.5	6.78E-17	5.54E-17	5.27E-17	4.79E-17
-0.5	5.97E-17	5.67E-17	5.08E-17	4.87E-17
-1.5	5.58E-17	5.21E-17	5.29E-17	4.76E-17
-2.5	5.39E-17	5.14E-17	4.51E-17	4.64E-17
-3.5	4.99E-17	4.69E-17	4.89E-17	4.16E-17
-4.5	4.06E-17	3.98E-17	4.24E-17	3.59E-17
-5.5	2.71E-17	2.76E-17	2.61E-17	2.23E-17
-6.5	9.14E-18	1.03E-17	1.17E-17	8.89E-18

TABLE 6.2-3

Ni-58 (n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS IRRADIATED DURING CYCLE 18

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0 DEG	15 DEG	30 DEG	45 DEG
+6.5	3.77E-17	2.51E-17	1.75E-17	2.07E-17
+5.5	8.35E-17	6.09E-17	4.25E-17	4.47E-17
+4.5	1.11E-16	8.49E-17	6.43E-17	5.94E-17
+3.5	1.12E-16	8.79E-17	7.07E-17	6.53E-17
+2.5	1.12E-16	9.43E-17	7.31E-17	6.68E-17
+1.5	1.08E-16	8.54E-17	7.27E-17	6.87E-17
+0.5	9.52E-17	8.15E-17	6.92E-17	6.92E-17
-0.5	8.39E-17	8.00E-17	7.31E-17	6.63E-17
-1.5	8.35E-17	7.81E-17	6.92E-17	6.92E-17
-2.5	8.05E-17	7.12E-17	7.02E-17	6.92E-17
-3.5	7.46E-17	6.97E-17	6.68E-17	6.38E-17
-4.5	6.09E-17	6.19E-17	5.99E-17	5.30E-17
-5.5	3.94E-17	4.19E-17	3.90E-17	3.47E-17
-6.5	1.48E-17	1.63E-17	1.72E-17	1.21E-17

TABLE 6.2-4

Co-59 (n,γ) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS IRRADIATED DURING CYCLE 18

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0 DEG	15 DEG	30 DEG	45 DEG
+6.5	2.62E-14	2.05E-14	1.73E-14	1.76E-14
+5.5	4.75E-14	6.44E-14	4.78E-14	3.41E-14
+4.5	6.55E-14	8.96E-14	6.83E-14	4.49E-14
+3.5	6.94E-14	1.03E-13	7.92E-14	5.14E-14
+2.5	7.26E-14	1.09E-13	8.46E-14	5.63E-14
+1.5	7.21E-14	1.05E-13	8.52E-14	5.57E-14
+0.5	7.75E-14	9.67E-14	8.36E-14	5.57E-14
-0.5	7.48E-14	9.50E-14	8.30E-14	5.63E-14
-1.5	7.04E-14	8.74E-14	8.08E-14	5.52E-14
-2.5	6.61E-14	8.25E-14	7.59E-14	5.25E-14
-3.5	5.84E-14	7.59E-14	7.04E-14	4.76E-14
-4.5	4.08E-14	6.39E-14	5.57E-14	3.80E-14
-5.5	2.78E-14	4.45E-14	3.02E-14	2.80E-14
-6.5	2.28E-14	2.29E-14	2.14E-14	2.26E-14

TABLE 6.2-5

DERIVED EXPOSURE RATES FROM THE CAPSULE T DOSIMETRY EVALUATION
0 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.35E+09	1.32E+09	8%
ϕ (E > 0.1 Mev)	1.09E+10	9.46E+09	16%
ϕ (E < 0.414 ev)	2.07E+09	1.60E+09	22%
ϕ (Total)	2.96E+10	2.57E+10	14%
dpa/sec	4.54E-12	4.04E-12	14%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	7.70E-19	1.01E-18	8.01E-19	1.31	1.04
Ti-46 (n,p)	1.15E-17	1.22E-17	1.14E-17	1.06	0.99
Fe-54 (n,p)	6.76E-17	6.69E-17	6.62E-17	0.99	0.98
Ni-58 (n,p)	9.57E-17	9.18E-17	9.34E-17	0.96	0.98
U-238 (n,f) (Cd)	3.57E-16	3.91E-16	3.84E-16	1.10	1.08
Np-237 (n,f) (Cd)	5.07E-15	5.80E-15	5.22E-15	1.14	1.03
Co-59 (n,γ)	8.99E-14	1.11E-13	9.10E-14	1.23	1.01
Co-59 (n,γ) (Cd)	4.92E-14	5.53E-14	4.89E-14	1.12	0.99
U-235 (n,f)	8.20E-13	1.01E-12	8.00E-13	1.23	0.98
U-235 (n,f) (Cd)	2.01E-13	2.12E-13	1.99E-13	1.05	1.00

TABLE 6.2-6

DERIVED EXPOSURE RATES FROM THE CAPSULE V DOSIMETRY EVALUATION
15 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.31E+09	1.27E+09	8%
ϕ (E > 0.1 Mev)	1.17E+10	1.03E+10	16%
ϕ (E < 0.414 ev)	2.12E+09	2.48E+09	21%
ϕ (Total)	2.85E+10	2.78E+10	13%
dpa/sec	4.15E-12	3.75E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	7.37E-19	9.48E-19	7.69E-19	1.29	1.04
Ti-46 (n,p)	1.12E-17	1.13E-17	1.10E-17	1.01	0.98
Fe-54 (n,p)	6.29E-17	6.23E-17	6.20E-17	0.99	0.99
Ni-58 (n,p)	8.94E-17	8.57E-17	8.75E-17	0.96	0.98
U-238 (n,f) (Cd)	3.46E-16	3.71E-16	3.65E-16	1.07	1.06
Np-237 (n,f) (Cd)	5.07E-15	5.90E-15	5.28E-15	1.16	1.04
Co-59 (n,γ)	1.13E-13	1.20E-13	1.21E-13	1.06	1.07
Co-59 (n,γ) (Cd)	5.61E-14	6.39E-14	5.48E-14	1.14	0.98
U-235 (n,f)	1.46E-12	1.03E-12	1.25E-12	0.70	0.85
U-235 (n,f) (Cd)	2.88E-13	2.34E-13	2.84E-13	0.81	0.99

TABLE 6.2-7

DERIVED EXPOSURE RATES FROM THE CAPSULE W DOSIMETRY EVALUATION
30 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.08E+09	1.12E+09	8%
ϕ (E > 0.1 Mev)	9.71E+09	9.51E+09	16%
ϕ (E < 0.414 ev)	1.86E+09	2.01E+09	22%
ϕ (Total)	2.44E+10	2.49E+10	13%
dpa/sec	3.44E-12	3.40E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	6.78E-19	8.59E-19	7.04E-19	1.27	1.04
Ti-46 (n,p)	9.79E-18	1.01E-17	9.67E-18	1.03	0.99
Fe-54 (n,p)	5.48E-17	5.38E-17	5.41E-17	0.98	0.99
Ni-58 (n,p)	8.02E-17	7.37E-17	7.76E-17	0.92	0.97
U-238 (n,f) (Cd)	2.84E-16	3.11E-16	3.15E-16	1.09	1.11
Np-237 (n,f) (Cd)	4.88E-15	4.88E-15	4.90E-15	1.00	1.01
Co-59 (n,γ)	9.84E-14	1.06E-13	1.02E-13	1.07	1.04
Co-59 (n,γ) (Cd)	5.01E-14	5.67E-14	4.95E-14	1.13	0.99
U-235 (n,f)	1.09E-12	9.02E-13	1.00E-12	0.83	0.92
U-235 (n,f) (Cd)	2.48E-13	2.08E-13	2.43E-13	0.84	0.98

TABLE 6.2-8

DERIVED EXPOSURE RATES FROM THE CAPSULE X DOSIMETRY EVALUATION
45 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	9.58E+08	9.89E+08	8%
ϕ (E > 0.1 Mev)	8.07E+09	7.66E+09	16%
ϕ (E < 0.414 ev)	1.82E+09	1.27E+09	23%
ϕ (Total)	2.05E+10	2.03E+10	13%
dpa/sec	2.90E-12	2.81E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	6.46E-19	8.42E-19	6.73E-19	1.30	1.04
Ti-46 (n,p)	9.36E-18	9.74E-18	9.23E-18	1.04	0.99
Fe-54 (n,p)	5.14E-17	5.04E-17	5.06E-17	0.98	0.98
Ni-58 (n,p)	7.34E-17	6.87E-17	7.15E-17	0.94	0.97
U-238 (n,f) (Cd)	2.69E-16	2.81E-16	2.87E-16	1.04	1.07
Np-237 (n,f) (Cd)	3.96E-15	4.18E-15	4.07E-15	1.05	1.03
Co-59 (n,γ)	6.57E-14	9.66E-14	7.22E-14	1.47	1.10
Co-59 (n,γ) (Cd)	4.09E-14	4.79E-14	3.93E-14	1.17	0.96
U-235 (n,f)	8.54E-13	8.75E-13	7.01E-13	1.02	0.82
U-235 (n,f) (Cd)	2.25E-13	1.75E-13	2.26E-13	0.78	1.01

TABLE 6.2-9

DERIVED EXPOSURE RATES FROM THE CAPSULE S DOSIMETRY EVALUATION
0 DEGREE AZIMUTH - CORE TOP

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	8.84E+08	9.39E+08	8%
ϕ (E > 0.1 Mev)	7.11E+09	6.33E+09	16%
ϕ (E < 0.414 ev)	1.35E+09	5.99E+08	24%
ϕ (Total)	1.93E+10	1.49E+10	15%
dpa/sec	2.97E-12	2.69E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	5.21E-19	6.60E-19	5.40E-19	1.27	1.04
Ti-46 (n,p)	8.06E-18	7.94E-18	7.89E-18	0.99	0.98
Fe-54 (n,p)	4.41E-17	4.37E-17	4.48E-17	0.99	1.02
Ni-58 (n,p)	6.90E-17	5.99E-17	6.62E-17	0.87	0.96
U-238 (n,f) (Cd)	2.70E-16	2.56E-16	2.73E-16	0.95	1.01
Np-237 (n,f) (Cd)	3.52E-15	3.79E-15	3.60E-15	1.08	1.02
Co-59 (n,γ)	4.21E-14	7.23E-14	4.45E-14	1.72	1.06
Co-59 (n,γ) (Cd)	2.96E-14	3.61E-14	2.85E-14	1.22	0.96
U-235 (n,f)	3.44E-13	6.57E-13	3.16E-13	1.91	0.92
U-235 (n,f) (Cd)	8.70E-14	1.38E-13	9.29E-14	1.59	1.07

TABLE 6.2-10

DERIVED EXPOSURE RATES FROM THE CAPSULE U DOSIMETRY EVALUATION
0 DEGREE AZIMUTH - CORE BOTTOM

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	3.75E+08	3.89E+08	8%
ϕ (E > 0.1 Mev)	3.02E+09	2.78E+09	16%
ϕ (E < 0.414 ev)	5.73E+08	5.33E+08	18%
ϕ (Total)	8.19E+09	7.16E+09	14%
dpa/sec	1.26E-12	1.17E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	2.02E-19	2.80E-19	2.12E-19	1.39	1.05
Ti-46 (n,p)	3.21E-18	3.37E-18	3.15E-18	1.05	0.98
Fe-54 (n,p)	1.87E-17	1.85E-17	1.84E-17	0.99	0.98
Ni-58 (n,p)	2.72E-17	2.54E-17	2.64E-17	0.93	0.97
U-238 (n,f) (Cd)	1.01E-16	1.08E-16	1.11E-16	1.07	1.09
Np-237 (n,f) (Cd)	1.58E-15	1.61E-15	1.58E-15	1.02	1.00
Co-59 (n,γ)	2.94E-14	3.07E-14	3.21E-14	1.04	1.09
Co-59 (n,γ) (Cd)	1.77E-14	1.53E-14	1.68E-14	0.87	0.95
U-235 (n,f)	3.18E-13	2.79E-13	2.69E-13	0.88	0.84
U-235 (n,f) (Cd)	4.42E-14	5.87E-14	4.74E-14	1.33	1.07

TABLE 6.2-11

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

	AZIMUTHAL ANGLE			
HEIGHT (ft)	0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
+6.5	5.49E+08	3.89E+08	2.62E+08	2.92E+08
+6.0	9.39E+08			
+5.5	1.15E+09	9.21E+08	6.48E+08	6.34E+08
+4.5	1.60E+09	1.32E+09	9.77E+08	8.81E+08
+3.5	1.62E+09	1.37E+09	1.06E+09	9.17E+08
+2.5	1.65E+09	1.47E+09	1.14E+09	9.31E+08
+1.5	1.60E+09	1.40E+09	1.17E+09	9.99E+08
+0.5	1.40E+09	1.27E+09	1.12E+09	9.95E+08
0.0	1.32E+09	1.27E+09	1.12E+09	9.89E+08
-0.5	1.24E+09	1.27E+09	1.12E+09	9.83E+08
-1.5	1.19E+09	1.20E+09	1.12E+09	9.92E+08
-2.5	1.15E+09	1.14E+09	1.04E+09	9.80E+08
-3.5	1.07E+09	1.08E+09	1.05E+09	8.91E+08
-4.5	8.69E+08	9.37E+08	9.30E+08	7.54E+08
-5.5	5.71E+08	6.42E+08	5.89E+08	4.81E+08
-6.0	3.89E+08			
-6.5	2.03E+08	2.45E+08	2.62E+08	1.79E+08

TABLE 6.2-12

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

HEIGHT (ft)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
+6.5	3.93E+09	3.15E+09	2.22E+09	2.26E+09
+6.0	6.33E+09			
+5.5	8.27E+09	7.47E+09	5.50E+09	4.91E+09
+4.5	1.15E+10	1.07E+10	8.30E+09	6.83E+09
+3.5	1.16E+10	1.11E+10	8.96E+09	7.10E+09
+2.5	1.18E+10	1.19E+10	9.65E+09	7.21E+09
+1.5	1.14E+10	1.13E+10	9.91E+09	7.74E+09
+0.5	1.01E+10	1.03E+10	9.47E+09	7.71E+09
0.0	9.46E+09	1.03E+10	9.51E+09	7.66E+09
-0.5	8.86E+09	1.03E+10	9.55E+09	7.61E+09
-1.5	8.55E+09	9.77E+09	9.49E+09	7.68E+09
-2.5	8.25E+09	9.26E+09	8.83E+09	7.59E+09
-3.5	7.64E+09	8.75E+09	8.96E+09	6.90E+09
-4.5	6.23E+09	7.60E+09	7.90E+09	5.84E+09
-5.5	4.09E+09	5.21E+09	5.00E+09	3.73E+09
-6.0	2.78E+09			
-6.5	1.46E+09	1.99E+09	2.22E+09	1.39E+09

TABLE 6.2-13

IRON ATOM DISPLACEMENT RATE (dpa/sec) AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 18 IRRADIATION

HEIGHT (ft)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
+6.5	1.68E-12	1.15E-12	7.95E-13	8.30E-13
+6.0	2.69E-12			
+5.5	3.53E-12	2.72E-12	1.97E-12	1.80E-12
+4.5	4.90E-12	3.91E-12	2.97E-12	2.50E-12
+3.5	4.96E-12	4.05E-12	3.20E-12	2.61E-12
+2.5	5.04E-12	4.33E-12	3.45E-12	2.65E-12
+1.5	4.89E-12	4.13E-12	3.54E-12	2.84E-12
+0.5	4.30E-12	3.75E-12	3.39E-12	2.83E-12
0.0	4.04E-12	3.75E-12	3.40E-12	2.81E-12
-0.5	3.78E-12	3.75E-12	3.41E-12	2.79E-12
-1.5	3.65E-12	3.56E-12	3.39E-12	2.82E-12
-2.5	3.52E-12	3.37E-12	3.16E-12	2.78E-12
-3.5	3.26E-12	3.19E-12	3.20E-12	2.53E-12
-4.5	2.66E-12	2.77E-12	2.82E-12	2.14E-12
-5.5	1.75E-12	1.90E-12	1.79E-12	1.37E-12
-6.0	1.17E-12			
-6.5	6.23E-13	7.23E-13	7.95E-13	5.10E-13

FIGURE 6.2-1

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 18 IRRADIATION

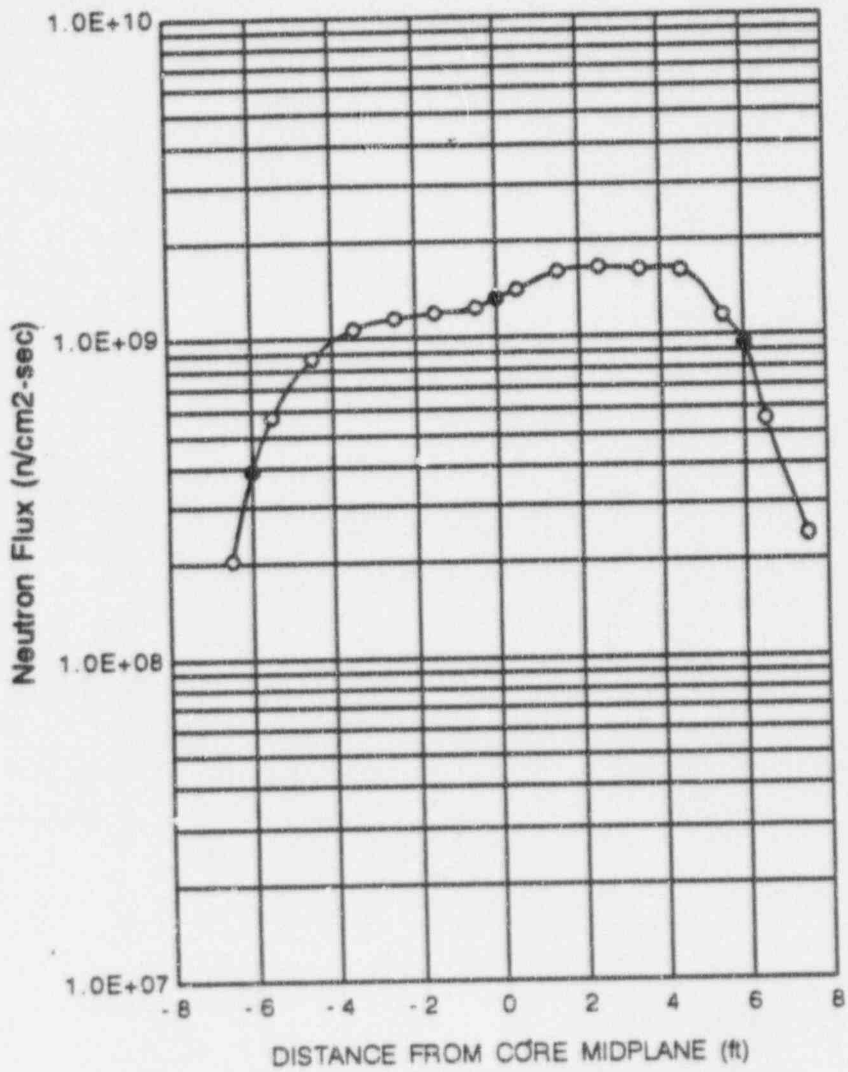


FIGURE 6.2-2

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 15 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 18 IRRADIATION

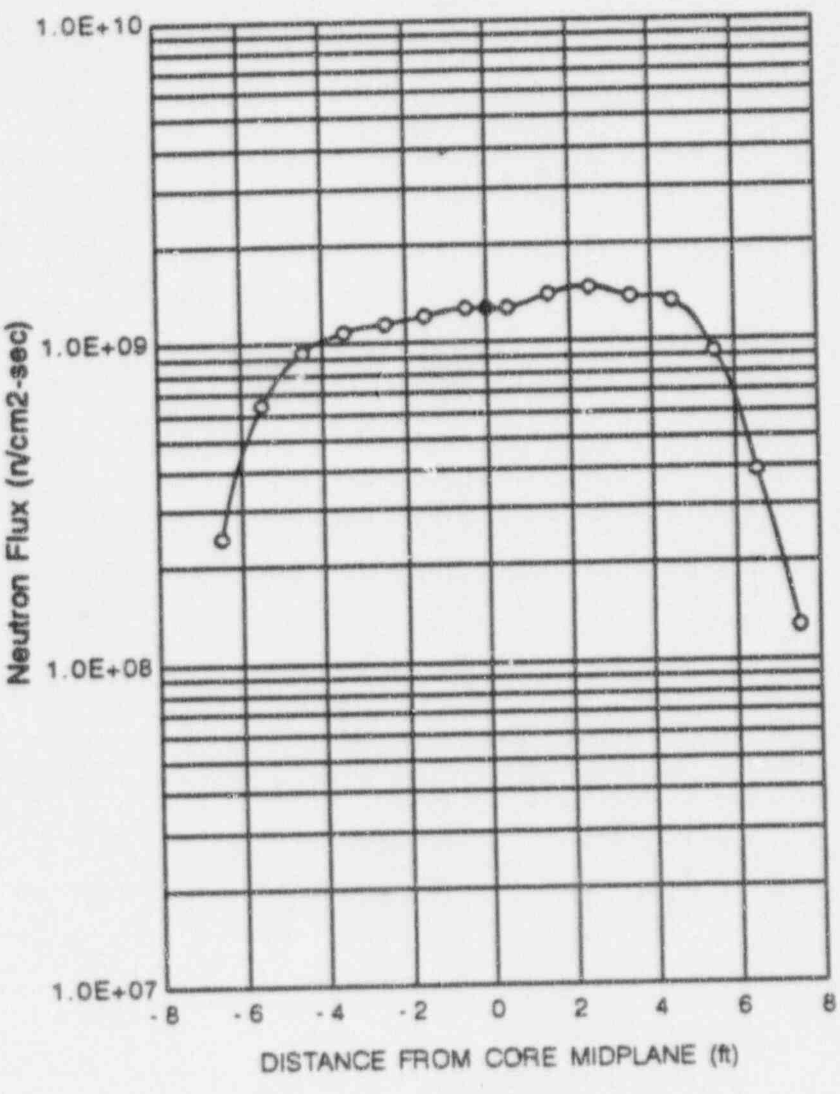


FIGURE 6.2-3

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 30 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 18 IRRADIATION

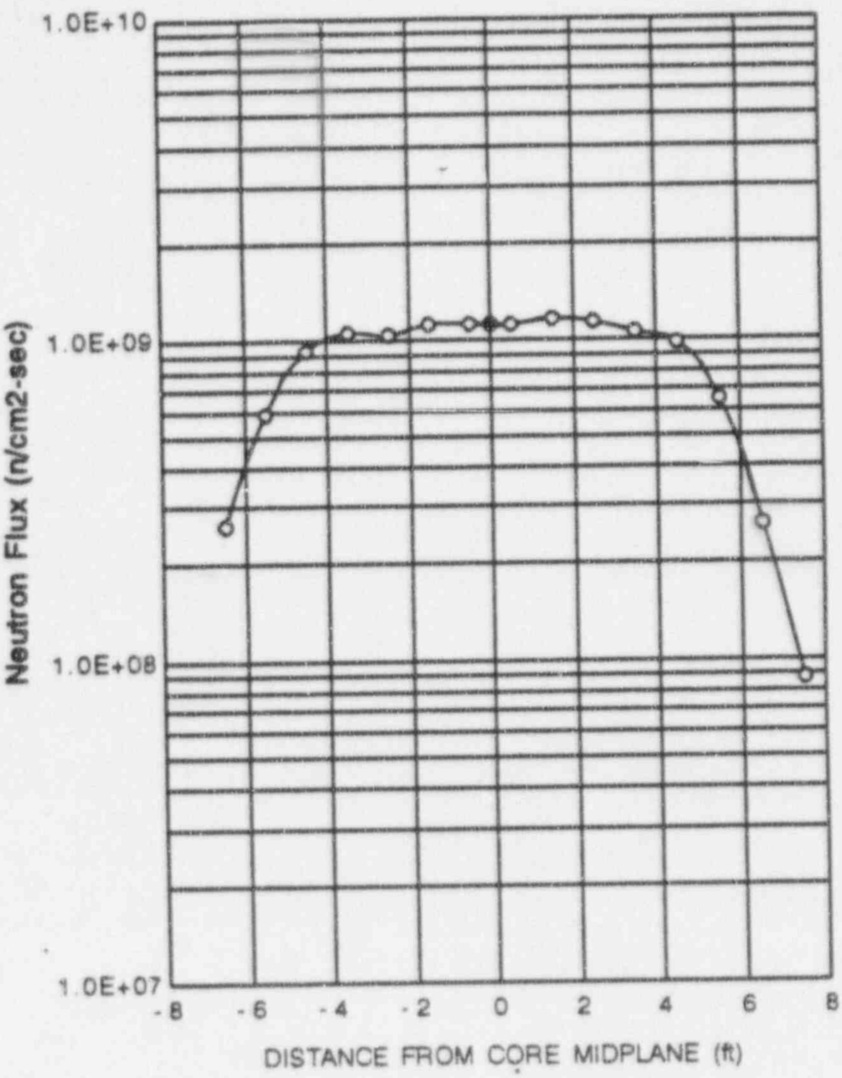
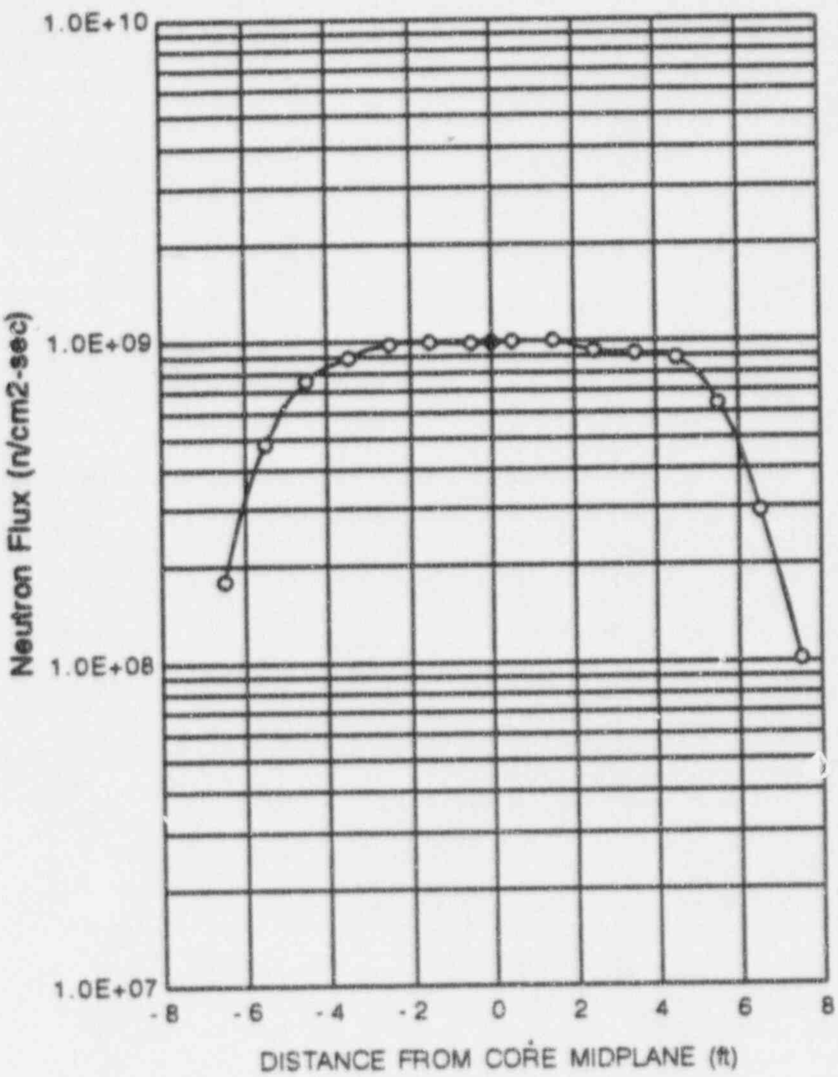


FIGURE 6.2-4

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 45 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 18 IRRADIATION



6.3 - Cycle 19 Results

6.3.1 - Measured Reaction rates

During the Cycle 19 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 [1]:

	CAPSULE IDENTIFICATION		
AZIMUTH (degrees)	CORE TOP	CORE MIDPLANE	CORE BOTTOM
0	GG	HH	II
15		JJ	
30		KK	
45		LL	

The contents of each of these irradiation capsules is specified in Reference 1 and, for completeness, is also included in Appendix D to this report.

The irradiation history of the Point Beach Unit 1 reactor during Cycle 19 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 19 are provided in Table 6.3-1. Corresponding reaction rate data from the the four stainless steel gradient 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 and the effects of γ,f reactions in the U-238 sensors as well for the effects of γ,f reactions in the Np-237 monitors.

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 19 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 a priori 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 "a priori calc" were obtained by normalizing the neutron spectral data from Table 4.1-1 to the measured Fe-54 (n,p) reaction rates from each sensor set as discussed in Section 3.0. Thus, the comparisons illustrated in Tables 6.3-5 through 6.3-10 indicate only the degree to which the relative neutron energy spectra matched the measured data before and after adjustment. These data are not meant to provide an absolute comparison of calculation and measurement. Absolute comparisons are discussed 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) and Ni-58 (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 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, the solid symbols at -6.0, 0.0, and +6.0 feet represent the explicit results of the FERRET evaluations, while the remaining open symbols depict the normalized data from the gradient chains. Note that at -6.0 and +6.0 feet, data are present only along the 0 degree traverse. For the 15, 30, and 45 degree traverses, multiple foil measurements were made only on the core midplane.

TABLE 6.3-1

SUMMARY OF REACTION RATES DERIVED FROM MULTIPLE FOIL SENSOR SETS IRRADIATED DURING CYCLE 19

	REACTION RATE (rps/nucleus)					
REACTION	CAPSULE HH	CAPSULE JJ	CAPSULE KK	CAPSULE LL	CAPSULE GG	CAPSULE II
Cu63(n,α)	7.83E-19	7.73E-19	7.52E-19	7.11E-19	3.95E-19	2.05E-19
Ti46(n,p)	1.15E-17	1.12E-17	1.06E-17	1.02E-17	6.43E-18	3.28E-18
Fe54(n,p)	6.62E-17	6.39E-17	5.93E-17	5.61E-17	3.43E-17	1.91E-17
Ni58(n,p)	9.29E-16	8.85E-17	8.34E-17	7.78E-17	5.26E-17	2.67E-17
U238(n,f)	3.84E-16	3.39E-16	3.39E-16	2.78E-16	2.54E-16	9.59E-17
Np237(n,f)	5.61E-15	5.31E-15	4.76E-15	4.79E-15	3.09E-15	1.54E-15
*Co59(n,γ)	8.67E-14	1.11E-13	9.77E-14	6.81E-14	3.51E-14	2.68E-14
Co59(n,γ)	4.90E-14	5.56E-14	5.16E-14	4.25E-14	2.48E-14	1.64E-14
*U235(n,f)	1.15E-12	1.41E-12	1.23E-12	8.56E-13	2.88E-13	3.06E-13
U235(n,f)	2.90E-13	3.21E-13	2.27E-13	2.12E-13	8.15E-14	4.98E-14

* - Bare foil, all others were cadmium covered

TABLE 6.3-2

Fe-54 (n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS IRRADIATED DURING CYCLE 19

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus) 0 DEG	15 DEG	30 DEG	45 DEG
+6.5	1.79E-17	1.38E-17	1.23E-17	1.25E-17
+5.5	4.43E-17	3.49E-17	3.17E-17	2.91E-17
+4.5	6.83E-17	5.71E-17	4.66E-17	4.17E-17
+3.5	7.41E-17	5.56E-17	4.63E-17	4.57E-17
+2.5	7.83E-17	6.77E-17	5.05E-17	5.17E-17
+1.5	7.20E-17	6.30E-17	5.34E-17	5.19E-17
+0.5	6.56E-17	5.98E-17	5.50E-17	5.12E-17
-0.5	5.82E-17	5.45E-17	5.14E-17	5.20E-17
-1.5	5.40E-17	5.19E-17	5.45E-17	5.01E-17
-2.5	5.14E-17	5.07E-17	4.65E-17	4.97E-17
-3.5	4.78E-17	4.62E-17	4.55E-17	4.17E-17
-4.5	4.08E-17	3.97E-17	3.92E-17	3.40E-17
-5.5	2.57E-17	2.29E-17	2.25E-17	2.01E-17
-6.5	9.63E-18	8.15E-18	9.31E-18	7.20E-18

TABLE 6.3-3

Ni-58 (n,p) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS IRRADIATED DURING CYCLE 19

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0 DEG	15 DEG	30 DEG	45 DEG
+6.5	2.68E-17	2.19E-17	1.86E-17	1.84E-17
+5.5	6.72E-17	5.58E-17	4.38E-17	4.23E-17
+4.5	9.81E-17	7.92E-17	6.45E-17	6.18E-17
+3.5	1.04E-16	8.40E-17	7.38E-17	6.75E-17
+2.5	1.13E-16	9.24E-17	7.65E-17	7.35E-17
+1.5	1.08E-16	8.67E-17	7.98E-17	7.38E-17
+0.5	9.57E-17	8.28E-17	7.62E-17	7.35E-17
-0.5	8.52E-17	8.13E-17	7.68E-17	7.20E-17
-1.5	8.04E-17	7.56E-17	7.50E-17	6.93E-17
-2.5	7.56E-17	7.26E-17	7.17E-17	7.02E-17
-3.5	7.32E-17	6.78E-17	6.66E-17	6.30E-17
-4.5	6.33E-17	5.97E-17	5.79E-17	5.13E-17
-5.5	3.93E-17	3.54E-17	3.33E-17	3.09E-17
-6.5	1.43E-17	1.31E-17	1.36E-17	1.02E-17

TABLE 6.3-4

Co-59 (n,γ) REACTION RATES DERIVED FROM THE STAINLESS STEEL GRADIENT CHAINS IRRADIATED DURING CYCLE 19

FEET FROM MIDPLANE	REACTION RATE (rps/nucleus)			
	0 DEG	15 DEG	30 DEG	45 DEG
+6.5	1.58E-14	2.01E-14	1.55E-14	1.54E-14
+5.5	4.30E-14	6.45E-14	4.34E-14	2.89E-14
+4.5	5.93E-14	8.94E-14	5.98E-14	3.74E-14
+3.5	6.86E-14	9.98E-14	7.02E-14	4.53E-14
+2.5	7.43E-14	1.13E-13	7.54E-14	4.95E-14
+1.5	7.38E-14	1.09E-13	7.59E-14	5.00E-14
+0.5	7.85E-14	1.02E-13	7.54E-14	5.02E-14
-0.5	6.34E-14	9.93E-14	7.48E-14	5.07E-14
-1.5	5.98E-14	9.25E-14	7.22E-14	4.93E-14
-2.5	5.61E-14	8.68E-14	6.65E-14	4.68E-14
-3.5	4.90E-14	7.85E-14	6.08E-14	3.99E-14
-4.5	3.63E-14	6.50E-14	4.33E-14	3.08E-14
-5.5	2.33E-14	4.22E-14	2.41E-14	2.18E-14
-6.5	2.19E-14	2.14E-14	1.69E-14	1.76E-14

TABLE 6.3-5

DERIVED EXPOSURE RATES FROM THE CAPSULE HH DOSIMETRY EVALUATION
0 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.32E+09	1.39E+09	8%
ϕ (E > 0.1 Mev)	1.07E+10	1.07E+10	16%
ϕ (E < 0.414 ev)	2.02E+09	1.78E+09	23%
ϕ (Total)	2.89E+10	3.06E+10	14%
dpa/sec	4.45E-12	4.51E-12	14%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	7.83E-19	9.89E-19	8.11E-19	1.26	1.04
Ti-46 (n,p)	1.15E-17	1.19E-17	1.13E-17	1.03	0.99
Fe-54 (n,p)	6.62E-17	6.54E-17	6.54E-17	0.99	0.99
Ni-58 (n,p)	9.29E-17	8.98E-17	9.16E-17	0.97	0.99
U-238 (n,f) (Cd)	3.84E-16	3.83E-16	3.96E-16	1.00	1.03
Np-237 (n,f) (Cd)	5.61E-15	5.68E-15	5.75E-15	1.01	1.03
Co-59 (n,γ)	8.67E-14	1.08E-13	9.38E-14	1.25	1.08
Co-59 (n,γ) (Cd)	4.90E-14	5.41E-14	4.75E-14	1.10	0.97
U-235 (n,f)	1.15E-12	9.84E-13	9.60E-13	0.86	0.83
U-235 (n,f) (Cd)	2.90E-13	2.07E-13	2.87E-13	0.71	0.99

TABLE 6.3-6

DERIVED EXPOSURE RATES FROM THE CAPSULE JJ DOSIMETRY EVALUATION
15 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.32E+09	1.28E+09	8%
ϕ (E > 0.1 Mev)	1.18E+10	1.07E+10	16%
ϕ (E < 0.414 ev)	2.13E+09	2.41E+09	22%
ϕ (Total)	2.87E+10	2.92E+10	13%
dpa/sec	4.18E-12	3.86E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	7.73E-19	9.55E-19	8.01E-19	1.24	1.04
Ti-46 (n,p)	1.12E-17	1.14E-17	1.10E-17	1.02	0.99
Fe-54 (n,p)	6.39E-17	6.28E-17	6.25E-17	0.98	0.98
Ni-58 (n,p)	8.85E-17	8.64E-17	8.71E-17	0.98	0.98
U-238 (n,f) (Cd)	3.39E-16	3.74E-16	3.64E-16	1.10	1.08
Np-237 (n,f) (Cd)	5.31E-15	5.95E-15	5.46E-15	1.12	1.03
Co-59 (n,γ)	1.11E-13	1.21E-13	1.17E-13	1.09	1.06
Co-59 (n,γ) (Cd)	5.56E-14	6.44E-14	5.47E-14	1.16	0.98
U-235 (n,f)	1.41E-12	1.04E-12	1.23E-12	0.73	0.87
U-235 (n,f) (Cd)	3.21E-13	2.36E-13	3.12E-13	0.74	0.97

TABLE 6.3-7

DERIVED EXPOSURE RATES FROM THE CAPSULE KK DOSIMETRY EVALUATION
30 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.17E+09	1.20E+09	8%
ϕ (E > 0.1 Mev)	1.05E+10	9.62E+09	16%
ϕ (E < 0.414 ev)	2.01E+09	2.07E+09	21%
ϕ (Total)	2.64E+10	2.48E+10	13%
dpa/sec	3.72E-12	3.49E-12	13%

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

REACTION	REACTION RATE (rps/nucleus) MEASURED	A PRIORI CALC.	ADJUSTED CALC.	C/M A PRIORI	ADJUSTED
Cu-63 (n,α)	7.52E-19	9.30E-19	7.76E-19	1.24	1.03
Ti-46 (n,p)	1.06E-17	1.10E-17	1.05E-17	1.03	0.99
Fe-54 (n,p)	5.93E-17	5.83E-17	5.86E-17	0.98	0.99
Ni-58 (n,p)	8.34E-17	7.98E-17	8.21E-17	0.96	0.98
U-238 (n,f) (Cd)	3.39E-16	3.36E-16	3.44E-16	0.99	1.02
Np-237 (n,f) (Cd)	4.76E-15	5.29E-15	4.94E-15	1.11	1.04
Co-59 (n,γ)	9.77E-14	1.14E-13	1.05E-13	1.17	1.08
Co-59 (n,γ) (Cd)	5.16E-14	6.14E-14	5.00E-14	1.19	0.97
U-235 (n,f)	1.23E-12	9.76E-13	1.04E-12	0.79	0.84
U-235 (n,f) (Cd)	2.27E-13	2.25E-13	2.30E-13	0.99	1.01

TABLE 6.3-8

DERIVED EXPOSURE RATES FROM THE CAPSULE LL DOSIMETRY EVALUATION
45 DEGREE AZIMUTH - CORE MIDPLANE

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	1.05E+08	1.09E+09	8%
ϕ (E > 0.1 Mev)	8.81E+09	8.91E+09	16%
ϕ (E < 0.414 ev)	1.99E+09	1.31E+09	23%
ϕ (Total)	2.24E+10	2.21E+10	13%
dpa/sec	3.17E-12	3.21E-12	13%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	7.11E-19	9.19E-19	7.39E-19	1.29	1.04
Ti-46 (n,p)	1.02E-17	1.06E-17	1.01E-17	1.04	0.99
Fe-54 (n,p)	5.61E-17	5.50E-17	5.48E-17	0.98	0.98
Ni-58 (n,p)	7.78E-17	7.50E-17	7.64E-17	0.96	0.98
U-238 (n,f) (Cd)	2.78E-16	3.06E-16	3.10E-16	1.10	1.11
Np-237 (n,f) (Cd)	4.79E-15	4.56E-15	4.74E-15	0.95	1.99
Co-59 (n,γ)	6.81E-14	1.05E-13	7.48E-14	1.55	1.10
Co-59 (n,γ) (Cd)	4.25E-14	5.22E-14	4.07E-14	1.23	0.96
U-235 (n,f)	8.56E-13	9.55E-13	7.07E-13	1.12	0.83
U-235 (n,f) (Cd)	2.12E-13	1.91E-13	2.16E-13	0.90	1.02

TABLE 6.3-9

DERIVED EXPOSURE RATES FROM THE CAPSULE GG DOSIMETRY EVALUATION
0 DEGREE AZIMUTH - CORE TOP

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	6.85E+08	8.17E+08	8%
ϕ (E > 0.1 Mev)	5.51E+09	5.72E+09	16%
ϕ (E < 0.414 ev)	1.05E+09	4.93E+08	24%
ϕ (Total)	1.50E+10	1.33E+10	15%
dpa/sec	2.30E-12	2.39E-12	13%

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

REACTION	REACTION RATE (rps/nucleus) MEASURED	A PRIORI CALC.	ADJUSTED CALC.	C/M A PRIORI	ADJUSTED
Cu-63 (n,α)	3.95E-19	5.11E-19	4.11E-19	1.29	1.04
Ti-46 (n,p)	6.43E-18	6.15E-18	6.22E-18	0.96	0.97
Fe-54 (n,p)	3.43E-17	3.38E-17	3.53E-17	0.99	1.03
Ni-58 (n,p)	5.26E-17	4.64E-17	5.15E-17	0.88	0.98
U-238 (n,f) (Cd)	2.54E-16	1.98E-16	2.34E-16	0.78	0.92
Np-237 (n,f) (Cd)	3.09E-15	2.94E-15	3.19E-15	0.95	1.03
Co-59 (n,γ)	3.51E-14	5.60E-14	3.70E-14	1.60	1.05
Co-59 (n,γ) (Cd)	2.48E-14	2.80E-14	2.39E-14	1.13	0.97
U-235 (n,f)	2.88E-13	5.09E-13	2.66E-13	1.77	0.92
U-235 (n,f) (Cd)	8.15E-14	1.07E-13	8.53E-14	1.32	1.05

TABLE 6.3-10

DERIVED EXPOSURE RATES FROM THE CAPSULE II DOSIMETRY EVALUATION
0 DEGREE AZIMUTH - CORE BOTTOM

PARAMETER	A PRIORI VALUE	ADJUSTED VALUE	UNCERTAINTY
ϕ (E > 1.0 Mev)	3.82E+08	3.78E+08	8%
ϕ (E > 0.1 Mev)	3.07E+09	2.75E+09	16%
ϕ (E < 0.414 ev)	5.83E+08	4.99E+08	20%
ϕ (Total)	8.33E+09	7.32E+09	14%
dpa/sec	1.28E-12	1.16E-12	14%

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

	REACTION RATE (rps/nucleus)			C/M	
REACTION	MEASURED	A PRIORI CALC.	ADJUSTED CALC.	A PRIORI	ADJUSTED
Cu-63 (n,α)	2.05E-19	2.85E-19	2.15E-19	1.39	1.05
Ti-46 (n,p)	3.28E-18	3.43E-18	3.21E-18	1.05	0.98
Fe-54 (n,p)	1.91E-17	1.89E-17	1.86E-17	0.99	0.97
Ni-58 (n,p)	2.67E-17	2.59E-17	2.61E-17	0.97	0.98
U-238 (n,f) (Cd)	9.59E-17	1.10E-16	1.08E-16	1.15	1.13
Np-237 (n,f) (Cd)	1.54E-15	1.64E-15	1.54E-15	1.06	1.00
Co-59 (n,γ)	2.68E-14	3.12E-14	2.95E-14	1.16	1.10
Co-59 (n,γ) (Cd)	1.64E-14	1.56E-14	1.56E-14	0.95	0.95
U-235 (n,f)	3.06E-13	2.84E-13	2.54E-13	0.93	0.83
U-235 (n,f) (Cd)	4.98E-14	5.97E-14	5.25E-14	1.20	1.05

TABLE 6.3-11

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 19 IRRADIATION

HEIGHT (ft)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
+6.5	4.07E+08	3.25E+08	2.85E+08	2.70E+08
+6.0	8.17E+08			
+5.5	1.01E+09	8.26E+08	7.01E+08	6.24E+08
+4.5	1.52E+09	1.26E+09	1.03E+09	9.03E+08
+3.5	1.63E+09	1.28E+09	1.10E+09	9.88E+08
+2.5	1.75E+09	1.48E+09	1.17E+09	1.10E+09
+1.5	1.64E+09	1.38E+09	1.23E+09	1.10E+09
+0.5	1.47E+09	1.32E+09	1.22E+09	1.09E+09
0.0	1.39E+09	1.28E+09	1.20E+09	1.09E+09
-0.5	1.31E+09	1.24E+09	1.18E+09	1.09E+09
-1.5	1.22E+09	1.17E+09	1.20E+09	1.05E+09
-2.5	1.16E+09	1.13E+09	1.09E+09	1.05E+09
-3.5	1.10E+09	1.05E+09	1.04E+09	9.12E+08
-4.5	9.44E+08	9.10E+08	8.96E+08	7.43E+08
-5.5	5.90E+08	5.33E+08	5.15E+08	4.44E+08
-6.0	3.78E+08			
-6.5	2.18E+08	1.93E+08	2.12E+08	1.52E+08

TABLE 6.2-12

FAST NEUTRON FLUX (E > 0.1 MeV) AS A FUNCTION
OF AXIAL POSITION WITHIN THE REACTOR CAVITY
CYCLE 19 IRRADIATION

HEIGHT (ft)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
+6.5	3.13E+09	2.72E+09	2.28E+09	2.21E+09
+6.0	5.72E+09			
+5.5	7.80E+09	6.90E+09	5.62E+09	5.10E+09
+4.5	1.17E+10	1.05E+10	8.27E+09	7.38E+09
+3.5	1.26E+10	1.07E+10	8.82E+09	8.08E+09
+2.5	1.35E+10	1.24E+10	9.37E+09	8.96E+09
+1.5	1.26E+10	1.16E+10	9.84E+09	9.00E+09
+0.5	1.13E+10	1.10E+10	9.77E+09	8.92E+09
0.0	1.07E+10	1.07E+10	9.62E+09	8.91E+09
-0.5	1.01E+10	1.04E+10	9.47E+09	8.90E+09
-1.5	9.42E+09	9.79E+09	9.65E+09	8.57E+09
-2.5	8.91E+09	9.48E+09	8.71E+09	8.59E+09
-3.5	8.46E+09	8.75E+09	8.30E+09	7.46E+09
-4.5	7.27E+09	7.61E+09	7.18E+09	6.07E+09
-5.5	4.55E+09	4.45E+09	4.13E+09	3.63E+09
-6.0	2.75E+09			
-6.5	1.68E+09	1.62E+09	1.70E+09	1.25E+09

TABLE 6.2-13

IRON ATOM DISPLACEMENT RATE (dpa/sec) AS A FUNCTION OF AXIAL POSITION WITHIN THE REACTOR CAVITY CYCLE 19 IRRADIATION

HEIGHT (ft)	AZIMUTHAL ANGLE 0 DEGREES	15 DEGREES	30 DEGREES	45 DEGREES
+6.5	1.32E-12	9.81E-13	8.28E-13	7.95E-13
+6.0	2.39E-12			
+5.5	3.29E-12	2.49E-12	2.04E-12	1.84E-12
+4.5	4.93E-12	3.79E-12	3.00E-12	2.66E-12
+3.5	5.29E-12	3.85E-12	3.20E-12	2.91E-12
+2.5	5.67E-12	4.46E-12	3.40E-12	3.23E-12
+1.5	5.32E-12	4.17E-12	3.57E-12	3.24E-12
+0.5	4.78E-12	3.97E-12	3.54E-12	3.21E-12
0.0	4.51E-12	3.86E-12	3.49E-12	3.21E-12
-0.5	4.24E-12	3.75E-12	3.44E-12	3.21E-12
-1.5	3.97E-12	3.53E-12	3.50E-12	3.09E-12
-2.5	3.76E-12	3.42E-12	3.16E-12	3.09E-12
-3.5	3.57E-12	3.15E-12	3.01E-12	2.69E-12
-4.5	3.06E-12	2.74E-12	2.61E-12	2.19E-12
-5.5	1.92E-12	1.61E-12	1.50E-12	1.31E-12
-6.0	1.16E-12			
-6.5	7.07E-13	5.83E-13	6.16E-13	4.49E-13

FIGURE 6.3-1

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 0 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 19 IRRADIATION

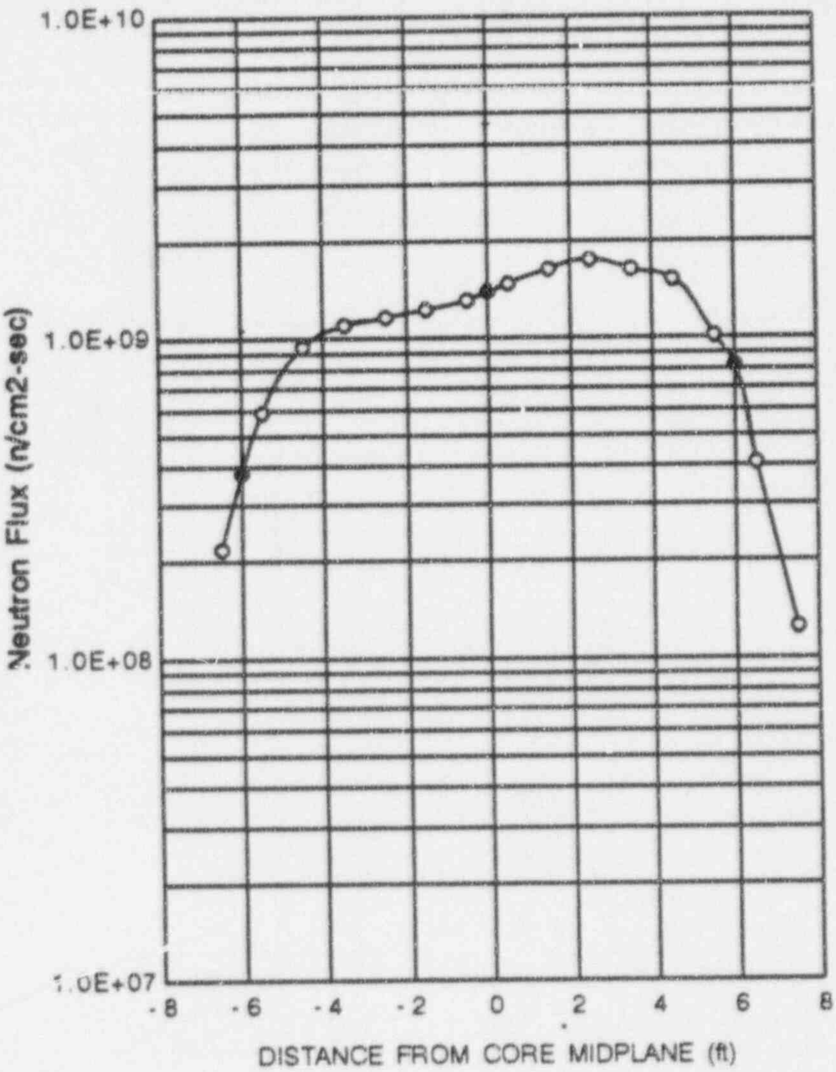


FIGURE 6.3-2

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 15 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 19 IRRADIATION

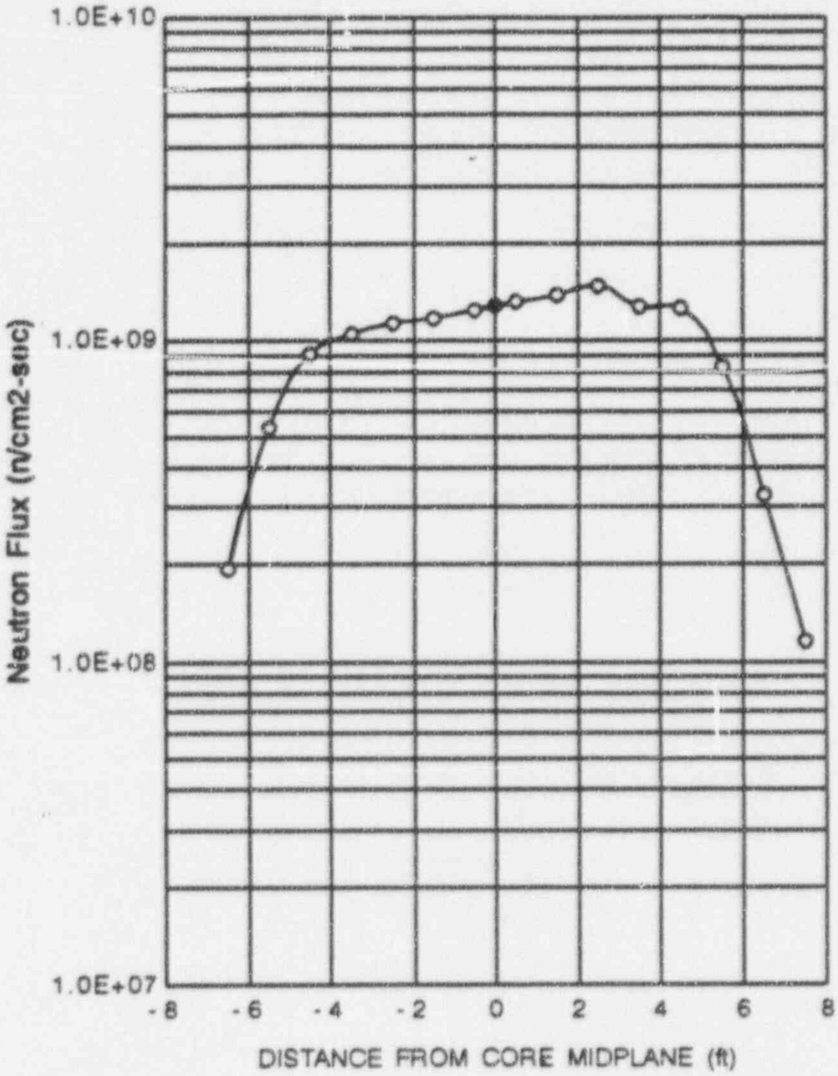


FIGURE 6.3-3

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION
ALONG THE 30 DEGREE TRAVERSE IN THE REACTOR CAVITY
CYCLE 19 IRRADIATION

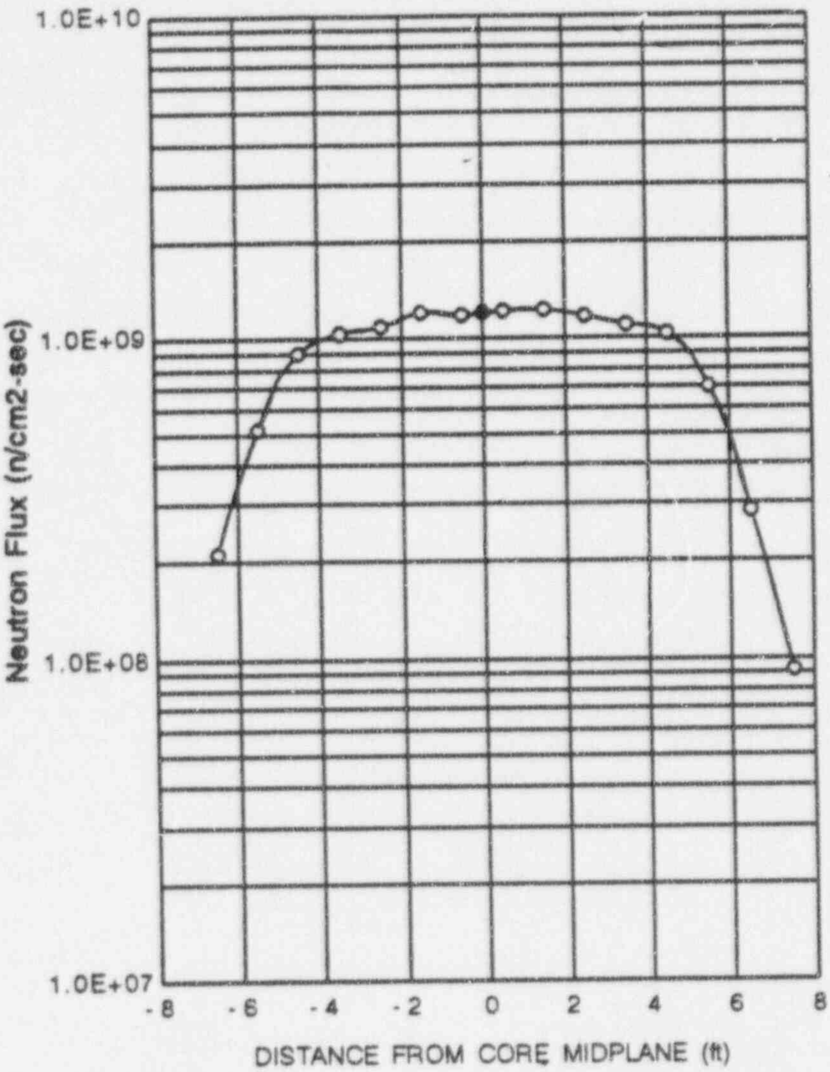
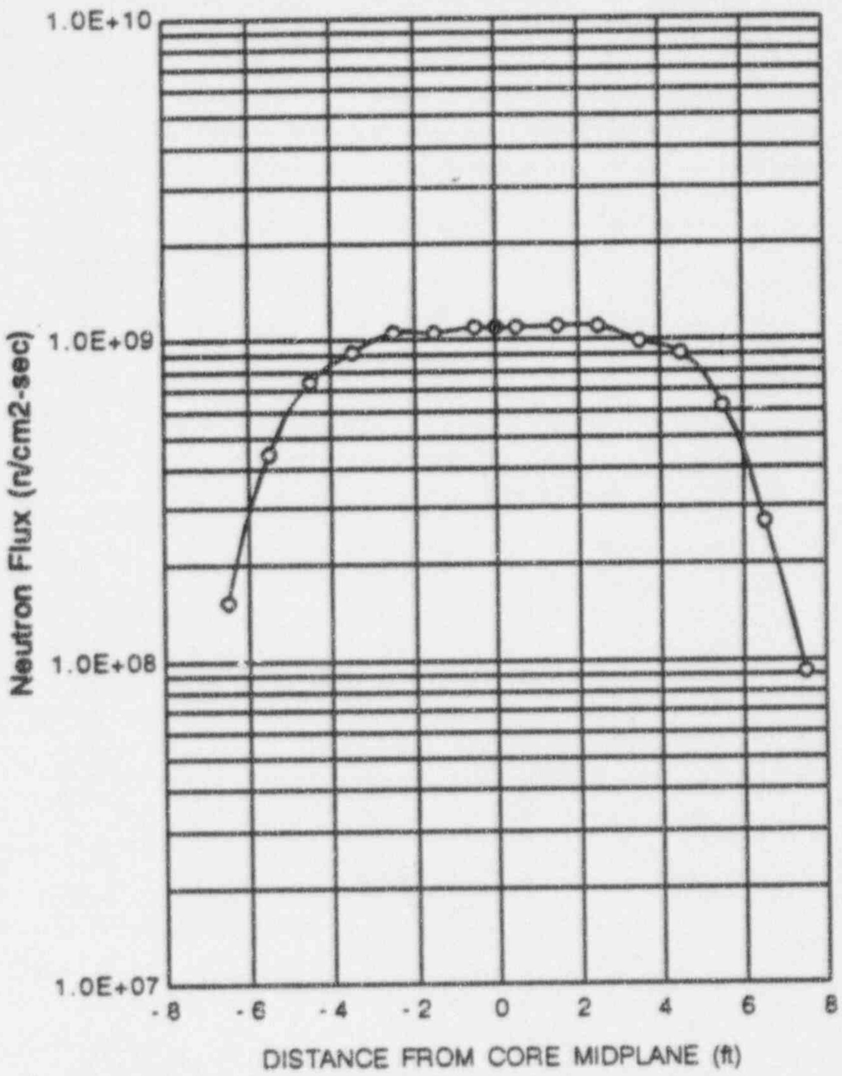


FIGURE 6.3-4

FAST NEUTRON FLUX (E > 1.0 MeV) AS A FUNCTION OF AXIAL POSITION ALONG THE 45 DEGREE TRAVERSE IN THE REACTOR CAVITY CYCLE 19 IRRADIATION



SECTION 7.0

COMPARISON OF CALCULATIONS WITH MEASUREMENTS

In order to develop accurate neutron exposure profiles at the inner diameter and through the thickness of the pressure vessel wall, the measurement results provided in Sections 5.0 and 6.0 must be combined with analytically determined spatial gradients provided in Section 4.0. In essence, this approach accepts the measurement results as the best available exposure rate information for the irradiation period in question and assumes that the analytically determined radial distribution functions provide accurate representations of the spatial gradients that exist among the measurement locations and the points of interest within the pressure vessel wall. This approach is analagous to the common practice of normalizing a cycle specific forward neutron transport calculation to available measurements from either surveillance capsule or reactor cavity dosimetry programs.

An indication of the acceptability of this method of exposure determination can be gained by an absolute comparison of the results of neutron transport calculations with all measured results applicable to a given reactor. These comparisons quantify the biases that may exist due to the transport methodology, reactor modeling, and/or reactor operating characteristics over the respective irradiation periods; and, furthermore, demonstrate the degree of consistency among the measurements obtained from different geometric locations and varying irradiation intervals.

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 calculation/measurement comparisons are yielding accurate bias factors that can be applied to neutron transport calculations performed for the Point Beach Unit 1 reactor.

7.1 Comparison of Least Squares Adjustment Results with Calculation

In Table 7.1-1, comparisons of calculated and measured exposure rates for the four surveillance capsule dosimetry sets and for the three cycles of reactor midplane dosimetry sets irradiated during Cycles 17, 18, and 19 are given. In all cases, the calculated values were based on the fuel cycle specific exposure calculations averaged over the appropriate irradiation period. That is, the Capsule V values apply to Cycle 1, the Capsules S, R, and T values represent averages over Cycles 1 through 3, 1 through 5, and 1 through 11, respectively; and, the cavity measurements directly apply to Cycles 17, 18, and 19.

An examination of Table 7.1-1 indicates that, considering all of the available core midplane data, the calculated exposure rates underpredicted measurements by factors of 0.904, 0.928, and 0.941 for ϕ (E > 1.0 MeV), ϕ (E > 0.1 MeV), and dpa/sec, respectively. The standard deviations associated with each of the 12 sample data sets were 0.10, 0.12, and 0.09, respectively.

The data comparisons provided in Table 7.1-1 also indicate a bias between the surveillance capsule and reactor cavity dosimetry comparisons; where the agreement between calculation and measurement is best at the cavity sensor locations and somewhat worse at the surveillance capsules. The reasons for this apparent bias cannot be acertained at this time, but may be due to the complexities in modeling the Point Beach Unit 1 surveillance capsules and support structure that exhibit a variable geometry over the capsule height. In contrast, the sensor holders used in the cavity irradiations were designed to provide free field measurements by minimizing perturbations in the neutron field and, thus, simplifying the neutron transport calculations. In any event, the inclusion of both sets of measurements in the data base listed in Table 7.1-1 results in an acceptable determination of calculation to measurement bias factors with standard deviations of better than 13% in all cases.

7.2 Comparisons of Measured and Calculated Sensor Reaction Rates

In Table 7.2-1, calculation/measurement 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 standard deviations observed for the six fast neutron reactions range from 10% to 14% on an individual reaction basis; whereas, the overall average C/M ratio for the entire data set has an associated 1σ standard deviation of approximately 12.0%. Furthermore, the average C/M bias of 0.943 observed in the reaction rate comparisons is in excellent agreement with the values of 0.904, 0.928, and 0.941 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

FAST NEUTRON FLUX (E > 1.0 MeV)

	ϕ [n/cm2-sec} MEASURED	CALCULATED	C/M
13 DEGREE CAPSULE (V)	1.07E+11	1.10E+11	1.028
33 DEGREE CAPSULE (S)	7.27E+10	5.94E+10	0.817
13 DEGREE CAPSULE (R)	1.48E+11	1.11E+10	0.750
23 DEGREE CAPSULE (T)	8.25E+10	6.35E+10	0.770
0 DEGREE CAVITY (CY17)	1.50E+09	1.54E+09	1.027
15 DEGREE CAVITY (CY17)	1.28E+09	1.28E+09	1.000
30 DEGREE CAVITY (CY17)	1.11E+09	1.03E+09	0.928
45 DEGREE CAVITY (CY17)	1.10E+09	9.18E+08	0.835
0 DEGREE CAVITY (CY18)	1.32E+09	1.44E+09	1.091
15 DEGREE CAVITY (CY18)	1.27E+09	1.19E+09	0.937
30 DEGREE CAVITY (CY18)	1.12E+09	9.43E+08	0.842
45 DEGREE CAVITY (CY18)	9.89E+08	8.21E+08	0.830
0 DEGREE CAVITY (CY19)	1.39E+09	1.41E+09	1.014
15 DEGREE CAVITY (CY19)	1.28E+09	1.20E+09	0.938
30 DEGREE CAVITY (CY19)	1.20E+09	1.01E+09	0.842
45 DEGREE CAVITY (CY19)	1.09E+09	8.91E+08	0.817
AVERAGE C/M RATIO			0.904
1σ VARIATION			0.10

TABLE 7.1-1 (Continued)

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

FAST NEUTRON FLUX (E > 0.1 MeV)

	ϕ [n/cm2-sec]		
	MEASURED	CALCULATED	C/M
13 DEGREE CAPSULE (V)	4.88E+11	4.20E+11	0.861
33 DEGREE CAPSULE (T)	2.58E+11	2.08E+11	0.806
13 DEGREE CAPSULE (R)	5.61E+11	4.23E+11	0.754
23 DEGREE CAPSULE (S)	3.01E+11	2.21E+11	0.734
0 DEGREE CAVITY (CY17)	1.15E+10	1.22E+10	1.061
15 DEGREE CAVITY (CY17)	1.06E+10	1.10E+10	1.038
30 DEGREE CAVITY (CY17)	9.20E+09	9.05E+09	0.984
45 DEGREE CAVITY (CY17)	8.50E+09	7.64E+09	0.899
0 DEGREE CAVITY (CY18)	9.46E+09	1.14E+10	1.205
15 DEGREE CAVITY (CY18)	1.03E+10	1.02E+09	0.990
30 DEGREE CAVITY (CY18)	9.51E+09	8.29E+09	0.872
45 DEGREE CAVITY (CY18)	7.66E+09	6.83E+09	0.892
0 DEGREE CAVITY (CY19)	1.07E+10	1.12E+10	1.047
15 DEGREE CAVITY (CY19)	1.07E+10	1.03E+10	0.963
30 DEGREE CAVITY (CY19)	9.62E+09	8.85E+09	0.920
45 DEGREE CAVITY (CY19)	8.91E+09	7.41E+09	0.832
AVERAGE C/M RATIO			0.928
1σ VARIATION			0.12

TABLE 7.1-1 (Continued)

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

IRON ATOM DISPLACEMENT RATE

	[dpa/sec]		
	MEASURED	CALCULATED	C/M
13 DEGREE CAPSULE (V)	2.13E-10	2.03E-10	0.953
23 DEGREE CAPSULE (S)	1.24E-10	1.04E-10	0.839
13 DEGREE CAPSULE (R)	2.62E-10	2.05E-10	0.782
33 DEGREE CAPSULE (T)	1.43E-10	1.12E-10	0.783
0 DEGREE CAVITY (CY17)	4.84E-12	4.67E-12	0.965
15 DEGREE CAVITY (CY17)	3.84E-12	4.12E-12	1.073
30 DEGREE CAVITY (CY17)	3.32E-12	3.39E-12	1.021
45 DEGREE CAVITY (CY17)	3.11E-12	2.90E-12	0.932
0 DEGREE CAVITY (CY18)	4.04E-12	4.36E-12	1.079
15 DEGREE CAVITY (CY18)	3.75E-12	3.83E-12	1.021
30 DEGREE CAVITY (CY18)	3.40E-12	3.10E-12	0.912
45 DEGREE CAVITY (CY18)	2.81E-12	2.59E-12	0.922
0 DEGREE CAVITY (CY19)	4.51E-12	4.28E-12	0.949
15 DEGREE CAVITY (CY19)	3.86E-12	3.87E-12	1.003
30 DEGREE CAVITY (CY19)	3.49E-12	3.31E-12	0.948
45 DEGREE CAVITY (CY19)	3.21E-12	2.82E-12	0.879
AVERAGE C/M RATIO			0.941
1σ VARIATION			0.09

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)
CAPSULES						
V	0.686		1.088	1.214	1.064	0.850
T	0.848		0.901	1.023	0.854	0.796
R	0.806		0.826	0.827	0.741	0.799
S	0.785		0.920		0.784	0.754
CY17 CAVITY						
0 DEGREE	1.025	1.049	1.054	1.076	1.184	1.084
15 DEGREE	0.898	0.932	0.958	0.962	1.096	1.102
30 DEGREE	0.878	0.920	0.931	0.924	1.032	1.032
45 DEGREE	0.888	0.906	0.917	0.890	0.896	0.947
CY18 CAVITY						
0 DEGREE	1.075	1.092	1.085	1.072	1.206	1.251
15 DEGREE	0.887	0.882	0.912	0.905	0.997	1.067
30 DEGREE	0.844	0.869	0.883	0.843	0.987	0.895
45 DEGREE	0.864	0.865	0.872	0.845	0.925	0.931
CY19 CAVITY						
0 DEGREE	1.035	1.069	1.085	1.081	1.109	1.107
15 DEGREE	0.853	0.889	0.906	0.922	1.024	1.027
30 DEGREE	0.815	0.859	0.874	0.868	0.884	0.982
45 DEGREE	0.852	0.861	0.867	0.865	0.977	0.835
AVERAGE C/M	0.877	0.933	0.942	0.954	0.985	0.966
1σ	0.10	0.09	0.09	0.11	0.13	0.14

TOTAL C/M RATIO	0.943
1σ VARIATION	0.11

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 1 reactor pressure vessel through the completion of Cycle 19. Based on the continued use of the Cycles 17/18/19 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 and lower shell plates and the beltline circumferential and longitudinal welds are highlighted.

8.1 Exposure Distributions Within the Beltline Region

In essence, an approach using analytically determined gradient information to extrapolate measurement results to locations of interest within the pressure vessel is based on the assertion that the measured values of exposure rates in the reactor cavity represent the best available neutron flux data for the irradiation period in question and, further, on the assumption that the analytically determined radial distribution functions provide accurate representations of the spatial gradients that exist among the measurement locations and points of interest within the pressure vessel wall. This method is analagous to the common practice of normalizing a cycle specific forward neutron transport calculation to available measurements from either surveillance capsule or reactor cavity dosimetry programs.

This approach provides accurate assessments of vessel exposure with associated uncertainties for periods of operation during which continuous monitoring has occured. In the case of Point Beach Unit 1, the cavity dosimetry program providing a complete spatial mapping of a sector of the beltline region of the pressure vessel was installed at the start of Cycle 17. Additional monitoring was limited to the four scheduled surveillance capsule withdrawals described in preceding sections of this report. The dosimetry data from these capsules provide measurement information at a single point within the reactor geometry for the four extended irradiation periods, but cannot be used to establish a verification of the exposure of the vessel at azimuthal locations far removed from the measurement point. Therefore, in order to establish a baseline exposure of the pressure vessel applicable to the onset of the reactor cavity measurement program, all available core midplane measured data were combined with fuel cycle specific transport calculations to provide best estimate exposures for the first 16 cycles of operation. The reactor cavity measurements were then used directly to provide the continuous monitoring capability for Cycles 17 and beyond.

8.1.1 Baseline Exposure at the End of Cycle 16

In Table 7.1-1, comparisons of calculated and measured exposure rates for the four surveillance capsule dosimetry sets and for the twelve cavity dosimetry sets that were located on the core midplane are given. From Table 7.1-1, it was noted that, considering all of the midplane data, the calculated exposure values underpredicted measurement by factors of 0.904, 0.928, and 0.941 for ϕ (E > 1.0 MeV), ϕ (E > 0.1 MeV), and dpa/sec, respectively. The corresponding 1σ standard deviations in these averages of the sixteen sample data sets were 0.10, 0.12, and 0.09.

In developing the best estimate baseline exposure for the Point Beach Unit 1 reactor pressure vessel these ratios were employed as bias factors to scale the cycle specific neutron transport calculations documented in Section 4.0 of this report. In particular, the following bias factors were employed to establish the baseline exposures of the vessel wall:

	M/C BIAS
Φ (E > 1.0 MeV)	1.106
Φ (E > 0.1 MeV)	1.078
dpa	1.063

The end of Cycle 16 best estimate exposures at the pressure vessel clad/base metal interface are provided in Tables 8.1-1 through 8.1-4 for Φ (E > 1.0 MeV), in Tables 8.1-5 through 8.1-8 for Φ (E > 0.1 MeV), and in Tables 8.1-9 through 8.1-12 for dpa. In these data tables, exposures are presented as a function of axial position for four azimuthal locations around the circumference of the vessel. From these tabulations, the locations of maximum exposure of the various materials comprising the beltline region can easily be determined. Exposure distributions through the vessel wall can be developed by normalizing the surface exposures from Tables 8.1-1 through 8.1-12 to the appropriate radial distribution functions given in Section 4.0 of this report.

8.1.2 Exposure Accrued During Cycles 17, 18, and 19

To assess the incremental exposure resulting from the Cycles 17, 18, and 19 irradiations, the measured results from the reactor cavity multiple foil sensor sets were directly extrapolated to the vessel clad/base metal interface using the analytically derived gradient data from Section 4.0 of this report. The axial gradient chain measurements were, of course, employed to develop the axial traverse along the vessel wall. The extrapolated results applicable to the vessel inner surface are also incorporated into Tables 8.1-1 through 8.1-12 to establish the best estimate exposure accrued by the reactor vessel through the end of Cycles 17, 18, and 19, respectively.

Again, as noted above, 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 19, 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 1 will expand resulting in reduced uncertainties and an improved accuracy in the assesment of vessel condition.

8.1.3 Projection of Future Vessel Exposure

At the end of Cycle 19, the Point Beach Unit 1 reactor had accrued 16.1 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 included in Tables 8.1-1 through 8.1-12 in addition to the plant specific exposure assessments through the end of Cycle 18.

These temporal extrapolations into the future were based on the assumption that the average of the measured data from the Cycles 17, 18, and 19 irradiations were representative of all future fuel cycles. That is, that future fuel designs would incorporate the low leakage fuel management concept including part length hafnium absorbers designed to provide flux reduction measures at the maximum exposure locations along the beltline circumferential weld. Examination of these projected exposure levels establishes the long term effectiveness of flux reduction measures 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 1 REACTOR PRESSURE VESSEL - 0 DEGREE AZIMUTHAL ANGLE

HEIGHT	Φ (E > 1.0 MeV) [n/cm2]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	4.41E+18	4.68E+18	5.01E+18	5.30E+18	1.09E+19	1.66E+19
+5.5	8.37E+18	8.75E+18	9.15E+18	9.51E+18	1.67E+19	2.40E+19
+4.5	1.22E+19	1.28E+19	1.33E+19	1.39E+19	2.45E+19	3.52E+19
+3.5	1.37E+19	1.44E+19	1.49E+19	1.55E+19	2.68E+19	3.81E+19
+2.5	1.42E+19	1.48E+19	1.54E+19	1.60E+19	2.78E+19	3.97E+19
+1.5	1.44E+19	1.50E+19	1.56E+19	1.62E+19	2.75E+19	3.89E+19
+0.5	1.46E+19	1.52E+19	1.56E+19	1.62E+19	2.61E+19	3.60E+19
0.0	1.46E+19	1.51E+19	1.56E+19	1.61E+19	2.54E+19	3.48E+19
-0.5	1.46E+19	1.51E+19	1.55E+19	1.60E+19	2.48E+19	3.36E+19
-1.5	1.46E+19	1.50E+19	1.55E+19	1.59E+19	2.41E+19	3.23E+19
-2.5	1.45E+19	1.49E+19	1.53E+19	1.57E+19	2.36E+19	3.14E+19
-3.5	1.42E+19	1.46E+19	1.49E+19	1.53E+19	2.28E+19	3.02E+19
-4.5	1.29E+19	1.33E+19	1.36E+19	1.40E+19	2.03E+19	2.68E+19
-5.5	9.27E+18	9.54E+18	9.74E+18	9.95E+18	1.42E+19	1.86E+19
-6.0	5.13E+18	5.30E+18	5.43E+18	5.57E+18	8.31E+18	1.11E+19

Note: Height is provided relative to the axial midplane of the active 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 1 REACTOR PRESSURE VESSEL - 15 DEGREE AZIMUTHAL ANGLE

HEIGHT (ft)	Φ (E > 1.0 MeV) [n/cm2] EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	2.72E+18	2.88E+18	3.09E+18	3.28E+18	6.83E+18	1.04E+19
+5.5	5.17E+18	5.39E+18	5.65E+18	5.89E+18	1.04E+19	1.50E+19
+4.5	7.45E+18	7.79E+18	8.16E+18	8.53E+18	1.53E+19	2.21E+19
+3.5	8.43E+18	8.81E+18	9.19E+18	9.56E+18	1.67E+19	2.39E+19
+2.5	8.68E+18	9.10E+18	9.50E+18	9.93E+18	1.78E+19	2.57E+19
+1.5	8.86E+18	9.26E+18	9.65E+18	1.00E+19	1.75E+19	2.51E+19
+0.5	8.96E+18	9.32E+18	9.67E+18	1.01E+19	1.70E+19	2.39E+19
0.0	8.99E+18	9.35E+18	9.70E+18	1.01E+19	1.69E+19	2.37E+19
-0.5	8.99E+18	9.34E+18	9.69E+18	1.00E+19	1.67E+19	2.34E+19
-1.5	8.98E+18	9.28E+18	9.62E+18	9.95E+18	1.61E+19	2.23E+19
-2.5	8.90E+18	9.20E+18	9.51E+18	9.84E+18	1.58E+19	2.17E+19
-3.5	8.67E+18	8.95E+18	9.25E+18	9.55E+18	1.51E+19	2.07E+19
-4.5	7.99E+18	8.25E+18	8.51E+18	8.77E+18	1.37E+19	1.87E+19
-5.5	5.70E+18	5.88E+18	6.06E+18	6.21E+18	9.47E+18	1.27E+19
-6.0	3.15E+18	3.26E+18	3.39E+18	3.48E+18	5.58E+18	7.68E+18

Note: Height is provided relative to the axial midplane of the active 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 1 REACTOR PRESSURE VESSEL - 30 DEGREE AZIMUTHAL ANGLE

HEIGHT	Φ (E > 1.0 MeV) [n/cm2]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	2.02E+18	2.16E+18	2.31E+18	2.47E+18	5.31E+18	8.17E+18
+5.5	3.84E+18	4.02E+18	4.21E+18	4.42E+18	8.06E+18	1.17E+19
+4.5	5.54E+18	5.82E+18	6.09E+18	6.40E+18	1.18E+19	1.73E+19
+3.5	6.26E+18	6.56E+18	6.85E+18	7.18E+18	1.30E+19	1.88E+19
+2.5	6.45E+18	6.77E+18	7.09E+18	7.43E+18	1.37E+19	1.99E+19
+1.5	6.57E+18	6.88E+18	7.21E+18	7.58E+18	1.39E+19	2.03E+19
+0.5	6.65E+18	6.96E+18	7.28E+18	7.64E+18	1.39E+19	2.02E+19
0.0	6.67E+18	6.98E+18	7.30E+18	7.65E+18	1.39E+19	2.01E+19
-0.5	6.67E+18	6.98E+18	7.30E+18	7.65E+18	1.38E+19	2.00E+19
-1.5	6.67E+18	6.97E+18	7.29E+18	7.64E+18	1.38E+19	2.00E+19
-2.5	6.61E+18	6.91E+18	7.21E+18	7.53E+18	1.33E+19	1.91E+19
-3.5	6.44E+18	6.73E+18	7.02E+18	7.33E+18	1.30E+19	1.86E+19
-4.5	5.93E+18	6.18E+18	6.45E+18	6.71E+18	1.17E+19	1.67E+19
-5.5	4.24E+18	4.40E+18	4.57E+18	4.72E+18	7.79E+18	1.09E+19
-6.0	2.34E+18	2.45E+18	2.56E+18	2.66E+18	4.63E+18	6.61E+18

Note: Height is provided relative to the axial midplane of the active 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 1 REACTOR PRESSURE VESSEL - 45 DEGREE AZIMUTHAL ANGLE

HEIGHT	Φ (E > 1.0 MeV) [n/cm2]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	1.76E+18	1.89E+18	2.05E+18	2.20E+18	5.02E+18	7.84E+18
+5.5	3.34E+18	3.53E+18	3.72E+18	3.91E+18	7.53E+18	1.12E+19
+4.5	4.82E+18	5.11E+18	5.38E+18	5.66E+18	1.09E+19	1.62E+19
+3.5	5.43E+18	5.74E+18	6.01E+18	6.32E+18	1.19E+19	1.75E+19
+2.5	5.60E+18	5.94E+18	6.21E+18	6.55E+18	1.26E+19	1.87E+19
+1.5	5.72E+18	6.06E+18	6.35E+18	6.70E+18	1.29E+19	1.91E+19
+0.5	5.77E+18	6.10E+18	6.40E+18	6.74E+18	1.28E+19	1.90E+19
0.0	5.80E+18	6.13E+18	6.42E+18	6.76E+18	1.28E+19	1.90E+19
-0.5	5.81E+18	6.14E+18	6.43E+18	6.77E+18	1.29E+19	1.90E+19
-1.5	5.80E+18	6.13E+18	6.42E+18	6.75E+18	1.28E+19	1.88E+19
-2.5	5.75E+18	6.08E+18	6.37E+18	6.69E+18	1.26E+19	1.86E+19
-3.5	5.61E+18	5.91E+18	6.17E+18	6.46E+18	1.18E+19	1.72E+19
-4.5	5.17E+18	5.44E+18	5.67E+18	5.90E+18	1.05E+19	1.52E+19
-5.5	3.69E+18	3.88E+18	4.02E+18	4.16E+18	7.12E+18	1.01E+19
-6.0	2.05E+18	2.16E+18	2.26E+18	2.35E+18	4.24E+18	6.14E+18

Note: Height is provided relative to the axial midplane of the active 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 1 REACTOR PRESSURE VESSEL - 0 DEGREE AZIMUTHAL ANGLE

HEIGHT	Φ (E > 0.1 MeV) [n/cm2]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	1.18E+19	1.24E+19	1.32E+19	1.39E+19	2.74E+19	4.10E+19
+5.5	2.22E+19	2.32E+19	2.42E+19	2.51E+19	4.37E+19	6.24E+19
+4.5	3.21E+19	3.37E+19	3.50E+19	3.65E+19	6.39E+19	9.15E+19
+3.5	3.63E+19	3.80E+19	3.94E+19	4.09E+19	7.00E+19	9.93E+19
+2.5	3.74E+19	3.92E+19	4.06E+19	4.22E+19	7.27E+19	1.03E+20
+1.5	3.83E+19	4.00E+19	4.14E+19	4.29E+19	7.21E+19	1.01E+20
+0.5	3.87E+19	4.02E+19	4.14E+19	4.27E+19	6.83E+19	9.40E+19
0.0	3.88E+19	4.02E+19	4.13E+19	4.26E+19	6.68E+19	9.11E+19
-0.5	3.88E+19	4.01E+19	4.12E+19	4.24E+19	6.51E+19	8.80E+19
-1.5	3.87E+19	3.99E+19	4.09E+19	4.20E+19	6.31E+19	8.43E+19
-2.5	3.84E+19	3.95E+19	4.05E+19	4.16E+19	6.17E+19	8.20E+19
-3.5	3.74E+19	3.85E+19	3.94E+19	4.04E+19	5.96E+19	7.89E+19
-4.5	3.44E+19	3.54E+19	3.61E+19	3.70E+19	5.35E+19	7.01E+19
-5.5	2.46E+19	2.53E+19	2.58E+19	2.63E+19	3.74E+19	4.86E+19
-6.0	1.36E+19	1.40E+19	1.43E+19	1.47E+19	2.14E+19	2.83E+19

Note: Height is provided relative to the axial midplane of the active 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 1 REACTOR PRESSURE VESSEL - 15 DEGREE AZIMUTHAL ANGLE

HEIGHT	Φ (E > 0.1 MeV) [n/cm2]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	7.72E+18	8.15E+18	8.69E+18	9.19E+18	1.85E+19	2.78E+19
+5.5	1.47E+19	1.53E+19	1.60E+19	1.67E+19	2.94E+19	4.22E+19
+4.5	2.11E+19	2.21E+19	2.31E+19	2.41E+19	4.30E+19	6.20E+19
+3.5	2.39E+19	2.50E+19	2.61E+19	2.71E+19	4.71E+19	6.72E+19
+2.5	2.47E+19	2.58E+19	2.70E+19	2.82E+19	5.02E+19	7.24E+19
+1.5	2.51E+19	2.62E+19	2.73E+19	2.84E+19	4.94E+19	7.04E+19
+0.5	2.54E+19	2.65E+19	2.74E+19	2.85E+19	4.79E+19	6.73E+19
0.0	2.55E+19	2.65E+19	2.75E+19	2.86E+19	4.76E+19	6.67E+19
-0.5	2.55E+19	2.65E+19	2.75E+19	2.85E+19	4.71E+19	6.59E+19
-1.5	2.54E+19	2.63E+19	2.72E+19	2.82E+19	4.54E+19	6.26E+19
-2.5	2.52E+19	2.60E+19	2.69E+19	2.78E+19	4.44E+19	6.10E+19
-3.5	2.46E+19	2.54E+19	2.62E+19	2.70E+19	4.26E+19	5.83E+19
-4.5	2.26E+19	2.34E+19	2.41E+19	2.48E+19	3.87E+19	5.27E+19
-5.5	1.62E+19	1.67E+19	1.72E+19	1.76E+19	2.67E+19	3.59E+19
-6.0	8.97E+18	9.26E+18	9.60E+18	9.86E+18	1.55E+19	2.11E+19

Note: Height is provided relative to the axial midplane of the active 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 1 REACTOR PRESSURE VESSEL - 30 DEGREE AZIMUTHAL ANGLE

HEIGHT	Φ (E > 0.1 MeV) [n/cm2]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	5.34E+18	5.66E+18	6.03E+18	6.40E+18	1.31E+19	1.99E+19
+5.5	1.01E+19	1.06E+19	1.11E+19	1.16E+19	2.08E+19	3.01E+19
+4.5	1.46E+19	1.53E+19	1.60E+19	1.67E+19	3.05E+19	4.43E+19
+3.5	1.65E+19	1.72E+19	1.80E+19	1.88E+19	3.36E+19	4.84E+19
+2.5	1.69E+19	1.77E+19	1.86E+19	1.94E+19	3.53E+19	5.12E+19
+1.5	1.72E+19	1.80E+19	1.89E+19	1.98E+19	3.59E+19	5.21E+19
+0.5	1.75E+19	1.83E+19	1.91E+19	2.00E+19	3.59E+19	5.19E+19
0.0	1.76E+19	1.84E+19	1.92E+19	2.01E+19	3.59E+19	5.17E+19
-0.5	1.76E+19	1.84E+19	1.92E+19	2.01E+19	3.57E+19	5.15E+19
-1.5	1.76E+19	1.83E+19	1.92E+19	2.00E+19	3.57E+19	5.14E+19
-2.5	1.74E+19	1.81E+19	1.89E+19	1.97E+19	3.43E+19	4.90E+19
-3.5	1.69E+19	1.77E+19	1.84E+19	1.92E+19	3.35E+19	4.79E+19
-4.5	1.56E+19	1.63E+19	1.70E+19	1.76E+19	3.02E+19	4.29E+19
-5.5	1.12E+19	1.16E+19	1.21E+19	1.24E+19	2.03E+19	2.81E+19
-6.0	6.19E+18	6.43E+18	6.73E+18	6.95E+18	1.18E+19	1.66E+19

Note: Height is provided relative to the axial midplane of the active 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 1 REACTOR PRESSURE VESSEL - 45 DEGREE AZIMUTHAL ANGLE

HEIGHT	Φ (E > 0.1 MeV) [n/cm2]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	4.45E+18	4.77E+18	5.12E+18	5.48E+18	1.20E+19	1.85E+19
+5.5	8.45E+18	8.91E+18	9.37E+18	9.86E+18	1.88E+19	2.77E+19
+4.5	1.22E+19	1.29E+19	1.35E+19	1.42E+19	2.72E+19	4.02E+19
+3.5	1.38E+19	1.45E+19	1.52E+19	1.60E+19	2.98E+19	4.36E+19
+2.5	1.42E+19	1.51E+19	1.57E+19	1.66E+19	3.15E+19	4.65E+19
+1.5	1.44E+19	1.53E+19	1.60E+19	1.69E+19	3.21E+19	4.74E+19
+0.5	1.47E+19	1.55E+19	1.62E+19	1.70E+19	3.20E+19	4.71E+19
0.0	1.47E+19	1.55E+19	1.62E+19	1.70E+19	3.20E+19	4.71E+19
-0.5	1.47E+19	1.55E+19	1.62E+19	1.70E+19	3.20E+19	4.71E+19
-1.5	1.47E+19	1.55E+19	1.62E+19	1.70E+19	3.18E+19	4.67E+19
-2.5	1.46E+19	1.53E+19	1.60E+19	1.69E+19	3.15E+19	4.63E+19
-3.5	1.42E+19	1.50E+19	1.56E+19	1.63E+19	2.95E+19	4.28E+19
-4.5	1.30E+19	1.37E+19	1.43E+19	1.48E+19	2.62E+19	3.77E+19
-5.5	9.37E+18	9.82E+18	1.02E+19	1.05E+19	1.78E+19	2.51E+19
-6.0	5.19E+18	5.44E+18	5.68E+18	5.89E+18	1.04E+19	1.48E+19

Note: Height is provided relative to the axial midplane of the active core.

TABLE 8.1-9

SUMMARY OF BEST ESTIMATE IRON ATOM DISPLACEMENT
PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 1
REACTOR PRESSURE VESSEL - 0 DEGREE AZIMUTHAL ANGLE

HEIGHT	IRON ATOM DISPLACEMENTS [dpa]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	7.01E-03	7.46E-03	7.97E-03	8.43E-03	1.74E-02	2.65E-02
+5.5	1.33E-02	1.40E-02	1.46E-02	1.53E-02	2.77E-02	4.03E-02
+4.5	1.92E-02	2.03E-02	2.12E-02	2.22E-02	4.06E-02	5.91E-02
+3.5	2.18E-02	2.29E-02	2.39E-02	2.49E-02	4.44E-02	6.41E-02
+2.5	2.24E-02	2.36E-02	2.46E-02	2.57E-02	4.61E-02	6.67E-02
+1.5	2.29E-02	2.40E-02	2.49E-02	2.60E-02	4.56E-02	6.53E-02
+0.5	2.32E-02	2.41E-02	2.50E-02	2.59E-02	4.31E-02	6.03E-02
0.0	2.33E-02	2.42E-02	2.50E-02	2.58E-02	4.21E-02	5.84E-02
-0.5	2.32E-02	2.40E-02	2.48E-02	2.56E-02	4.08E-02	5.62E-02
-1.5	2.32E-02	2.39E-02	2.46E-02	2.54E-02	3.96E-02	5.38E-02
-2.5	2.30E-02	2.37E-02	2.44E-02	2.51E-02	3.87E-02	5.23E-02
-3.5	2.23E-02	2.30E-02	2.37E-02	2.44E-02	3.72E-02	5.02E-02
-4.5	2.06E-02	2.13E-02	2.18E-02	2.24E-02	3.35E-02	4.46E-02
-5.5	1.47E-02	1.51E-02	1.55E-02	1.58E-02	2.33E-02	3.08E-02
-6.0	8.14E-03	8.42E-03	8.64E-03	8.86E-03	1.34E-02	1.80E-02

Note: Height is provided relative to the axial midplane of the active core.

TABLE 8.1-10

SUMMARY OF BEST ESTIMATE IRON ATOM DISPLACEMENT
PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 1
REACTOR PRESSURE VESSEL - 15 DEGREE AZIMUTHAL ANGLE

HEIGHT	IRON ATOM DISPLACEMENTS [dpa]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	4.45E-03	4.70E-03	5.00E-03	5.28E-03	1.05E-02	1.57E-02
+5.5	8.43E-03	8.79E-03	9.19E-03	9.57E-03	1.68E-02	2.40E-02
+4.5	1.22E-02	1.28E-02	1.33E-02	1.39E-02	2.46E-02	3.54E-02
+3.5	1.38E-02	1.44E-02	1.50E-02	1.56E-02	2.70E-02	3.84E-02
+2.5	1.41E-02	1.48E-02	1.54E-02	1.61E-02	2.86E-02	4.12E-02
+1.5	1.45E-02	1.51E-02	1.57E-02	1.63E-02	2.82E-02	4.01E-02
+0.5	1.47E-02	1.52E-02	1.58E-02	1.64E-02	2.74E-02	3.84E-02
0.0	1.47E-02	1.52E-02	1.58E-02	1.64E-02	2.71E-02	3.80E-02
-0.5	1.47E-02	1.52E-02	1.58E-02	1.63E-02	2.69E-02	3.75E-02
-1.5	1.47E-02	1.52E-02	1.57E-02	1.62E-02	2.60E-02	3.58E-02
-2.5	1.46E-02	1.50E-02	1.55E-02	1.60E-02	2.54E-02	3.49E-02
-3.5	1.41E-02	1.46E-02	1.51E-02	1.55E-02	2.44E-02	3.33E-02
-4.5	1.31E-02	1.35E-02	1.39E-02	1.43E-02	2.22E-02	3.01E-02
-5.5	9.31E-03	9.61E-03	9.88E-03	1.01E-02	1.53E-02	2.05E-02
-6.0	5.17E-03	5.34E-03	5.52E-03	5.67E-03	8.86E-03	1.21E-02

Note: Height is provided relative to the axial midplane of the active core.

TABLE 8.1-11

SUMMARY OF BEST ESTIMATE IRON ATOM DISPLACEMENT
PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 1
REACTOR PRESSURE VESSEL - 30 DEGREE AZIMUTHAL ANGLE

HEIGHT	IRON ATOM DISPLACEMENTS [dpa]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	3.23E-03	3.42E-03	3.64E-03	3.86E-03	7.82E-03	1.18E-02
+5.5	6.12E-03	6.40E-03	6.69E-03	6.99E-03	1.25E-02	1.80E-02
+4.5	8.84E-03	9.26E-03	9.68E-03	1.01E-02	1.83E-02	2.65E-02
+3.5	9.98E-03	1.04E-02	1.09E-02	1.14E-02	2.01E-02	2.89E-02
+2.5	1.03E-02	1.08E-02	1.13E-02	1.18E-02	2.11E-02	3.06E-02
+1.5	1.05E-02	1.10E-02	1.15E-02	1.20E-02	2.16E-02	3.12E-02
+0.5	1.06E-02	1.11E-02	1.16E-02	1.21E-02	2.15E-02	3.10E-02
0.0	1.06E-02	1.11E-02	1.16E-02	1.21E-02	2.14E-02	3.08E-02
-0.5	1.06E-02	1.11E-02	1.16E-02	1.21E-02	2.14E-02	3.07E-02
-1.5	1.06E-02	1.11E-02	1.16E-02	1.21E-02	2.14E-02	3.07E-02
-2.5	1.05E-02	1.10E-02	1.14E-02	1.19E-02	2.06E-02	2.93E-02
-3.5	1.03E-02	1.07E-02	1.12E-02	1.16E-02	2.01E-02	2.86E-02
-4.5	9.46E-03	9.85E-03	1.03E-02	1.06E-02	1.81E-02	2.56E-02
-5.5	6.76E-03	7.01E-03	7.27E-03	7.49E-03	1.21E-02	1.68E-02
-6.0	3.75E-03	3.90E-03	4.07E-03	4.20E-03	7.06E-03	9.93E-03

Note: Height is provided relative to the axial midplane of the active core.

TABLE 8.1-12

SUMMARY OF BEST ESTIMATE IRON ATOM DISPLACEMENT
PROJECTIONS FOR THE BELTLINE REGION OF THE POINT BEACH UNIT 1
REACTOR PRESSURE VESSEL - 45 DEGREE AZIMUTHAL ANGLE

HEIGHT	IRON ATOM DISPLACEMENTS [dpa]					
(ft)	EOC 16	EOC 17	EOC 18	EOC 19	32 EFPY	48 EFPY
+6.0	2.77E-03	2.96E-03	3.18E-03	3.39E-03	7.27E-03	1.12E-02
+5.5	5.26E-03	5.54E-03	5.82E-03	6.11E-03	1.15E-02	1.69E-02
+4.5	7.59E-03	8.02E-03	8.40E-03	8.83E-03	1.67E-02	2.45E-02
+3.5	8.57E-03	9.02E-03	9.42E-03	9.88E-03	1.82E-02	2.66E-02
+2.5	8.82E-03	9.32E-03	9.73E-03	1.02E-02	1.92E-02	2.83E-02
+1.5	9.00E-03	9.50E-03	9.94E-03	1.05E-02	1.96E-02	2.89E-02
+0.5	9.11E-03	9.60E-03	1.00E-02	1.05E-02	1.96E-02	2.87E-02
0.0	9.14E-03	9.63E-03	1.01E-02	1.06E-02	1.96E-02	2.87E-02
-0.5	9.14E-03	9.63E-03	1.01E-02	1.06E-02	1.96E-02	2.87E-02
-1.5	9.13E-03	9.62E-03	1.01E-02	1.05E-02	1.95E-02	2.85E-02
-2.5	9.07E-03	9.55E-03	9.97E-03	1.05E-02	1.93E-02	2.82E-02
-3.5	8.83E-03	9.28E-03	9.67E-03	1.01E-02	1.81E-02	2.61E-02
-4.5	8.14E-03	8.55E-03	8.88E-03	9.23E-03	1.61E-02	2.30E-02
-5.5	5.83E-03	6.10E-03	6.31E-03	6.52E-03	1.09E-02	1.53E-02
-6.0	3.13E-03	3.29E-03	3.43E-03	3.55E-03	6.25E-03	8.96E-03

Note: Height is provided realative to the axial
midplane of the active core.

8.2 Exposure of Specific Beltline Materials

As shown in Figure 2.1-2, the beltline region of the Point Beach Unit 1 reactor pressure vessel is comprised of an intermediate shell plate (A-9811), a lower shell plate (C-1423), an intermediate shell longitudinal weld (SA-775/812), a lower shell longitudinal weld (SA-847), and a circumferential weld (SA-1101) joining the two shell plates. The circumferential weld is centered 15.06 inches below the axial midplane of the active core; while the intermediate shell plate extends upward to an elevation 9.75 inches above the active fuel and the lower shell plate extends downward to an elevation 39.87 inches below the bottom of the active fuel. The intermediate and lower shell longitudinal welds are both located at an azimuthal angle of 15 degrees and exhibit the same axial extent as the respective shell plates. 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. Also considered part of the beltline region is the circumferential weld located at the top of the intermediate shell 9.75 inches above the active fuel. The corresponding circumferential weld at the bottom of the lower shell, 39.87 inches below the active fuel, experiences neutron radiation levels low enough that the lifetime exposure of the weld will remain below 1.0E+17 n/cm^2. Therefore, this weldment is not considered to be part of the beltline region.

8.2-1 Circumferential Weld (SA-1101)

The current (End of Cycle 19) 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 Φ (E > 1.0 MeV), $\dot{\Phi}$ (E > 0.1 MeV), and dpa.

In developing the exposure profiles for the circumferential weld, it is noted that, although the flux reduction afforded by the Cycles 17/18/19 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 Plate (A-9811)

The current and projected maximum exposures of the intermediate shell plate are given in Table 8.2-2. Again, all three exposure parameters are provided. In the case of the intermediate plate, it can be noted from Table 8.1-1, that, due to the introduction of the part length absorbers and the corresponding redution in exposure rates in the vicinity of the circumferential weld, the axial location of the maximum exposure at 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 plate 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 Plate (C-1423)

The current and projected exposures for the lower shell plate are listed in Table 8.2-3. As in the case of the intermediate plate, all three exposure parameters are tabulated. In the case of the lower shell plate, the part length absorbers cause the maximum exposure location at all azimuthal angles to remain at the top of the plate. Therefore, the maximum exposure of the lower shell plate is identical to that of the circumferential weld. As a result the graphical illustrations of exposure depicted in Figures 8.2-1 through 8.2-3 apply equally to both the circumferential weld and the lower shell plate.

8.2-4 Longitudinal Welds (SA-775/812 and SA-847)

The current and projected exposures for the intermediate and lower shell longitudinal welds are listed in Table 8.2-4. These two welds are located at the 15 degree azimuth relative to the core cardinal axes. The maximum exposure of each of the welds occurs at an axial elevation identical to that of the respective shell plates. That is, for the intermediate shell weld, the maximum exposure point shifts upward over the course of the irradiation; while, for the lower shell weld, the location of maximum exposure remains at the interface with the circumferential weld.

8.2-5 Upper/Intermediate Shell Circumferential Weld

The current and projected exposures for the circumferential weld joining the upper and intermediate shell courses are listed in Table 8.2-5. As in the case of the other beltline materials all three exposure parameters are tabulated. Due to its location above the top of the reactor core, the inclusion of the part length hafnium absorbers in the fuel cycle design has a negligible impact on the fast neutron exposure experienced by the weld.

TABLE 8.2-1

MAXIMUM FAST NEUTRON EXPOSURE OF POINT BEACH UNIT 1
BELTLINE CIRCUMFERENTIAL WELD (SA-1101)

AZIMUTHAL ANGLE	Φ (E > 1.0 MeV) [n/cm2] EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	1.59E+19	2.43E+19	3.26E+19
15 DEGREES	9.63E+18	1.63E+19	2.26E+19
30 DEGREES	7.64E+18	1.38E+19	2.00E+19
45 DEGREES	6.75E+18	1.28E+19	1.89E+19

AZIMUTHAL ANGLE	Φ (E > 0.1 MeV) [n/cm2] EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	4.21E+19	6.36E+19	8.52E+19
15 DEGREES	2.83E+19	4.58E+19	6.34E+19
30 DEGREES	2.00E+19	3.57E+19	5.14E+19
45 DEGREES	1.70E+19	3.18E+19	4.68E+19

AZIMUTHAL ANGLE	IRON ATOM DISPLACEMENTS [dpa] EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	2.55E-02	3.99E-02	5.44E-02
15 DEGREES	1.62E-02	2.62E-02	3.62E-02
30 DEGREES	1.21E-02	2.14E-02	3.07E-02
45 DEGREES	1.05E-02	1.95E-02	2.86E-02

TABLE 8.2-2

MAXIMUM FAST NEUTRON EXPOSURE OF POINT BEACH UNIT 1
INTERMEDIATE SHELL PLATE (A-9811)

	Φ (E > 1.0 MeV) [n/cm2]		
AZIMUTHAL ANGLE	EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	1.62E+19	2.78E+19	3.97E+19
15 DEGREES	1.01E+19	1.78E+19	2.57E+19
30 DEGREES	7.65E+18	1.39E+19	2.03E+19
45 DEGREES	6.77E+18	1.29E+19	1.91E+19

	Φ (E > 0.1 MeV) [n/cm2]		
AZIMUTHAL ANGLE	EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	4.29E+19	7.27E+19	1.03E+20
15 DEGREES	2.86E+19	5.02E+19	7.24E+19
30 DEGREES	2.01E+19	3.59E+19	5.21E+19
45 DEGREES	1.70E+19	3.21E+19	4.74E+19

	IRON ATOM DISPLACEMENTS [dpa]		
AZIMUTHAL ANGLE	EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	2.60E-02	4.61E-02	6.67E-02
15 DEGREES	1.64E-02	2.86E-02	4.12E-02
30 DEGREES	1.21E-02	2.16E-02	3.12E-02
45 DEGREES	1.06E-02	1.96E-02	2.89E-02

TABLE 8.2-3

MAXIMUM FAST NEUTRON EXPOSURE OF POINT BEACH UNIT 1
LOWER SHELL PLATE (C-1423)

	Φ (E > 1.0 MeV) [n/cm2]		
AZIMUTHAL ANGLE	EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	1.59E+19	2.43E+19	3.26E+19
15 DEGREES	9.63E+18	1.63E+19	2.26E+19
30 DEGREES	7.64E+18	1.38E+19	2.00E+19
45 DEGREES	6.75E+18	1.28E+19	1.89E+19

	Φ (E > 0.1 MeV) [n/cm2]		
AZIMUTHAL ANGLE	EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	4.21E+19	6.36E+19	8.52E+19
15 DEGREES	2.83E+19	4.58E+19	6.34E+19
30 DEGREES	2.00E+19	3.57E+19	5.14E+19
45 DEGREES	1.70E+19	3.18E+19	4.68E+19

	IRON ATOM DISPLACEMENTS [dpa]		
AZIMUTHAL ANGLE	EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	2.55E-02	3.99E-02	5.44E-02
15 DEGREES	1.62E-02	2.62E-02	3.62E-02
30 DEGREES	1.21E-02	2.14E-02	3.07E-02
45 DEGREES	1.05E-02	1.95E-02	2.86E-02

TABLE 8.2-4

MAXIMUM FAST NEUTRON EXPOSURE OF POINT BEACH UNIT 1
BELTLINE LONGITUDINAL WELDS

	Φ (E > 1.0 MeV) [n/cm2]		
AZIMUTHAL ANGLE	EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
INT. SHELL (SA-775/812)	1.01E+19	1.78E+19	2.57E+19
LOWER SHELL (SA-847)	9.63E+18	1.63E+19	2.26E+19

	Φ (E > 0.1 MeV) [n/cm2]		
AZIMUTHAL ANGLE	EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
INT. SHELL (SA-775/812)	2.86E+19	5.02E+19	7.24E+19
LOWER SHELL (SA-847)	2.83E+19	4.58E+19	6.34E+19

	IRON ATOM DISPLACEMENTS [dpa]		
AZIMUTHAL ANGLE	EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
INT. SHELL (SA-775/812)	1.64E-02	2.86E-02	4.12E-02
LOWER SHELL (SA-847)	1.62E-02	2.62E-02	3.62E-02

TABLE 8.2-5

MAXIMUM FAST NEUTRON EXPOSURE OF POINT BEACH UNIT 1
UPPER/INTERMEDIATE SHELL CIRCUMFERENTIAL WELD

AZIMUTHAL ANGLE	Φ (E > 1.0 MeV) [n/cm2] EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	1.54E+18	3.17E+18	4.83E+18
15 DEGREES	9.53E+17	1.99E+18	3.02E+18
30 DEGREES	7.18E+17	1.54E+18	2.38E+18
45 DEGREES	6.40E+17	1.46E+18	2.28E+18

AZIMUTHAL ANGLE	Φ (E > 0.1 MeV) [n/cm2] EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	4.04E+18	7.97E+18	1.19E+19
15 DEGREES	2.67E+18	5.38E+18	8.08E+18
30 DEGREES	1.86E+18	3.81E+18	5.78E+18
45 DEGREES	1.59E+18	3.49E+18	5.38E+18

AZIMUTHAL ANGLE	IRON ATOM DISPLACEMENTS [dpa] EOC 19 16.1 EFPY	32.0 EFPY	48.0 EFPY
0 DEGREES	2.45E-03	5.06E-03	7.70E-03
15 DEGREES	1.53E-03	3.05E-03	4.56E-03
30 DEGREES	1.12E-03	2.27E-03	3.43E-03
45 DEGREES	9.85E-04	2.11E-03	3.26E-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

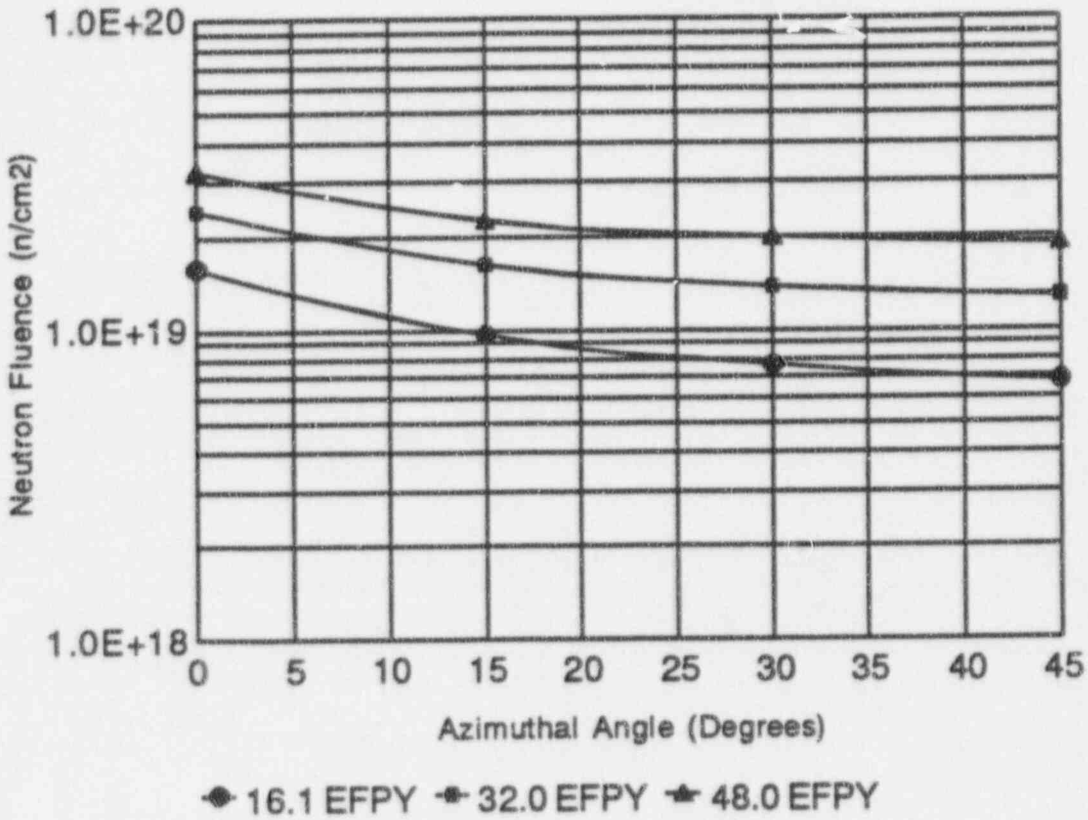


FIGURE 8.2-2

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

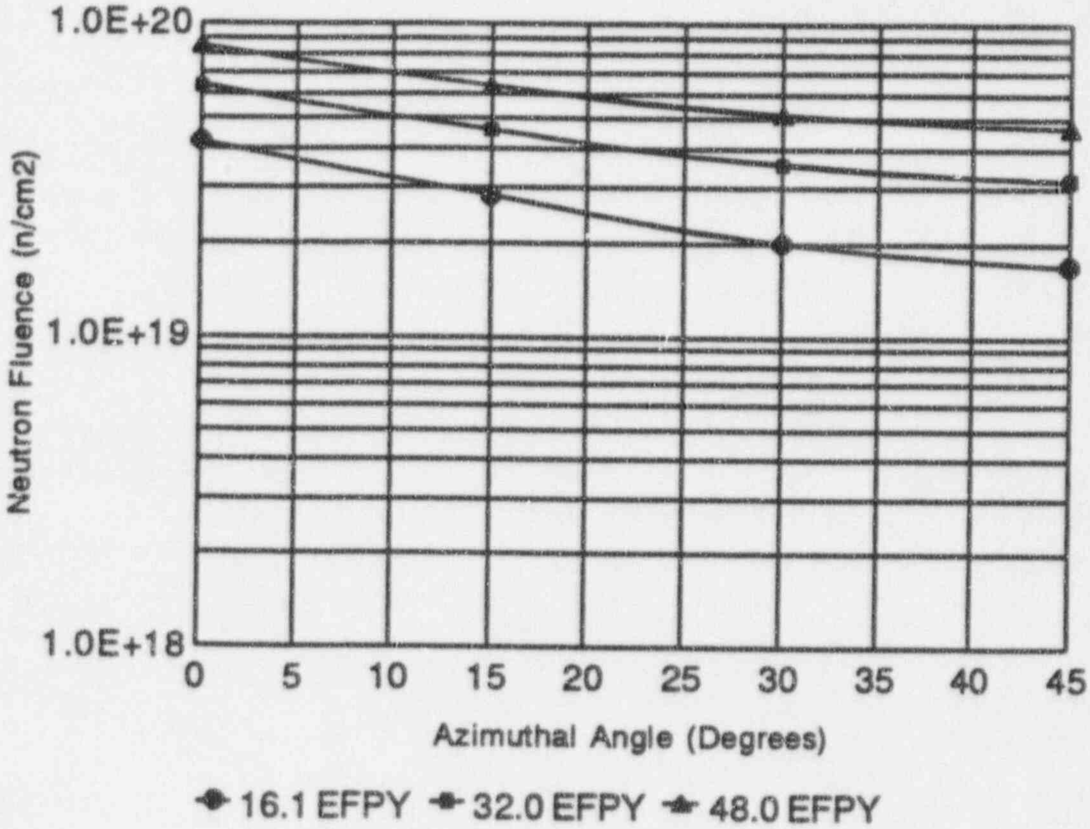
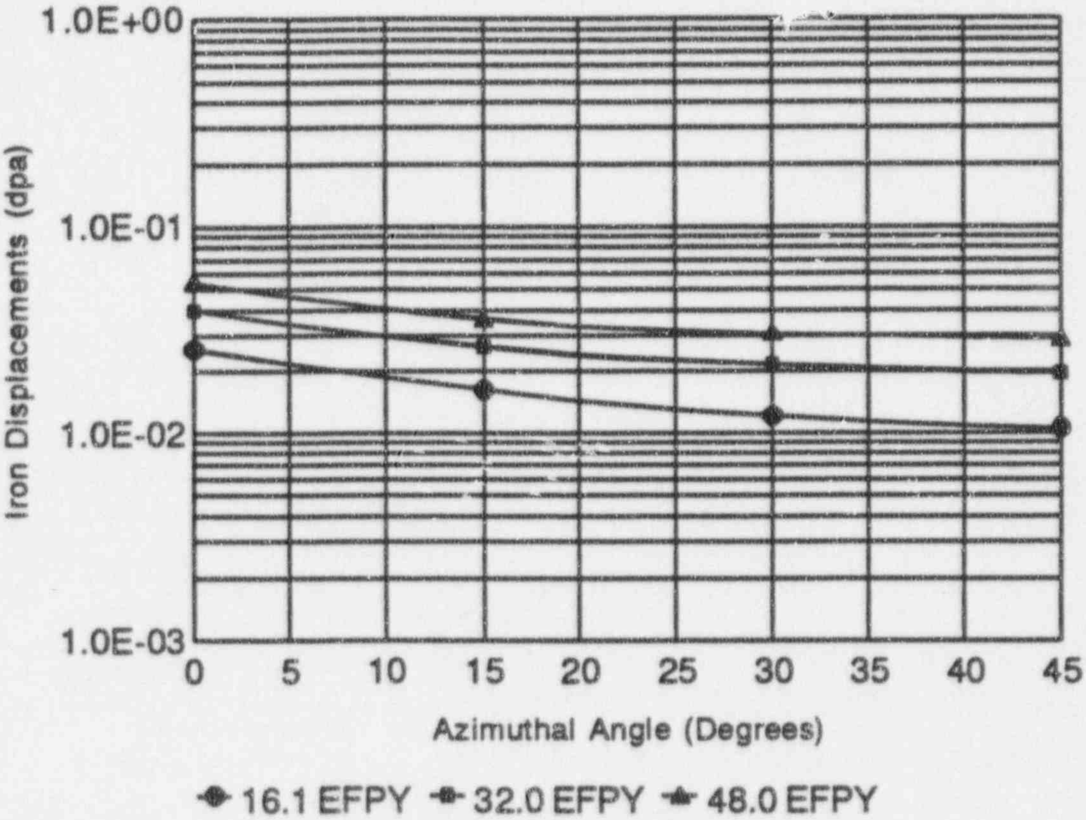


FIGURE 8.2-3

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 uncertainties associated with the exposure rates and integrated exposures determined for Point Beach Unit 1 stem from two basic sources; the accuracy of the neutron flux measurements at the sensor set locations and the accuracy of the radial gradient projections derived from the use of the transport code. Based on the least squares adjustment approach used in the FERRET analyses the 1 sigma uncertainties in the measured data were as follows:

	1σ UNCERTAINTY	
	CAPSULE	CAVITY
Flux (E > 1.0 MeV)	8%	8%
Flux (E > 0.1 MeV)	16%	16%
dpa/sec	11%	13%

These values represent uncertainties derived from the reaction rate measurements and from the least squares fit of the output spectrum to the measured data. As additional data is obtained from the ongoing measurement program, the knowledge of the neutron spectra at the measurement locations will increase and the uncertainties in the measured exposure parameters will be reduced somewhat.

Since the ultimate goal of the cavity measurement program is the evaluation of the exposure of the vessel itself, an additional uncertainty associated with the ability to translate results from the measurement locations to the points of interest within the vessel must be included along with the measurement uncertainties listed above. Information pertinent to this extrapolation uncertainty 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 uncertainty or bias associated with the calculated slope through the steel vessel was estimated to be approximately 5% for all exposure parameters. Thus, the total uncertainty associated with projections at the clad/base metal interface is estimated to be as follows for each exposure parameter of interest.

1σ UNCERTAINTY

	VESSEL IR
Flux (E > 1.0 MeV)	13%
Flux (E > 0.1 MeV)	21%
dpa/sec	17%

Use of these values represents the bounding 1σ uncertainties for vessel exposure, since with penetration into the vessel wall the extrapolation uncertainty lessens until at the outer surface the overall uncertainty reverts simply to the measurement uncertainty. Again, as more data are accumulated from both reactor cavity and surveillance capsule dosimetry sets, the extrapolation uncertainty will also be reduced resulting in higher levels of accuracy in the vessel exposure projections.

SECTION 9.0

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APPENDIX A

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

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

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

CYCLE NO.	STARTUP	SHUTDOWN	COMMENT
1	11/02/70	09/28/72	CAPSULE V WITHDRAWN
2	03/11/73	04/06/74	
3	06/06/74	11/16/75	CAPSULE S WITHDRAWN
4	01/07/76	10/01/76	
5	11/23/76	10/04/77	CAPSULE R WITHDRAWN
6	11/04/77	09/20/78	
7	10/14/78	10/05/79	
8	12/01/79	11/29/80	
9	12/29/80	10/09/81	
10	12/09/81	10/22/82	
11	12/08/82	10/01/83	CAPSULE T WITHDRAWN

REF. CORE POWER = 1518 MWt

The monthly thermal generation applicable to the Point Beach Unit 1 reactor is provided on pages A-3 and A-4. Pages A-5 through A-8 contain the measured specific activities of sensors removed from Capsules S, R, and T.

Since surveillance Capsules S, R, and T 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 ϕ (E > 1.0 MeV) during the irradiation period j to the time weighted average ϕ (E > 1.0 MeV) over the entire irradiation period. The values of C_j used in the evaluation of the Point Beach Unit 1 surveillance capsules were as follows:

C_j

	CAPSULE S	CAPSULE R	CAPSULE T
CYCLE 1	0.931	0.991	0.989
CYCLE 2	0.978	0.955	0.982
CYCLE 3	1.034	0.991	1.031
CYCLE 4		1.063	1.097
CYCLE 5		1.018	1.071
CYCLE 6			1.037
CYCLE 7			1.015
CYCLE 8			0.877
CYCLE 9			0.807
CYCLE 10			0.816
CYCLE 11			0.860

MONTHLY THERMAL GENERATION DURING THE FIRST 11 FUEL CYCLES OF THE POINT BEACH UNIT 1 REACTOR

MONTH	THERMAL GENERATION (MW-Hr)	MONTH	THERMAL GENERATION (MW-Hr)	MONTH	THERMAL GENERATION (MW-Hr)	MONTH	THERMAL GENERATION (MW-Hr)
11/70	178200	12/72	0	1/75	499488	2/77	943434
12/70	453840	1/73	0	2/75	896398	3/77	1099786
1/71	721680	2/73	0	3/75	0	4/77	1063777
2/71	655872	3/73	634632	4/75	818168	5/77	1069044
3/71	927024	4/73	779040	5/75	1101929	6/77	731782
4/71	438480	5/73	810960	6/75	1017790	7/77	1051139
5/71	828072	6/73	888480	7/75	1107172	8/77	1117757
6/71	886320	7/73	1069872	8/75	1107373	9/77	1007699
7/71	931488	8/73	999192	9/75	1047241	10/77	124440
8/71	1020024	9/73	1008000	10/75	1100361	11/77	876867
9/71	890640	10/73	947856	11/75	547037	12/77	1087979
10/71	965712	11/73	816480	12/75	0	1/78	1051834
11/71	770400	12/73	941904	1/76	724911	2/78	736854
12/71	979104	1/74	1011096	2/76	1007835	3/78	1115694
1/72	1034904	2/74	972384	3/76	1096079	4/78	1056265
2/72	968136	3/74	1101120	4/76	1055030	5/78	933674
3/72	1034904	4/74	187128	5/76	1038077	6/78	1054582
4/72	1001520	5/74	0	6/76	1056957	7/78	1117338
5/72	1034904	6/74	674280	7/76	1094830	8/78	1116325
6/72	1001520	7/74	1079520	8/76	1100256	9/78	693385
7/72	1022256	8/74	1030200	9/76	1076231	10/78	543459
8/72	1103352	9/74	992832	10/76	26520	11/78	1072516
9/72	932400	10/74	1036944	11/76	136694	12/78	1096664
10/72	0	11/74	938184	12/76	1001759	1/79	1105062
11/72	0	12/74	1102512	1/77	1105105	2/79	1003316

MONTHLY THERMAL GENERATION DURING THE FIRST 11 FUEL CYCLES
OF THE POINT BEACH UNIT 1 REACTOR

MONTH	THERMAL GENERATION (MW-Hr)	MONTH	THERMAL GENERATION (MW-Hr)	MONTH	THERMAL GENERATION (MW-Hr)	MONTH	THERMAL GENERATION (MW-Hr)
3/79	824123	5/80	901815	7/81	467103	9/82	852120
4/79	1064749	6/80	870132	8/81	988811	10/82	595125
5/79	1108609	7/80	722969	9/81	976689	11/82	0
6/79	1067584	8/80	650041	10/81	257443	12/82	600074
7/79	1116197	9/80	871153	11/81	0	1/83	869385
8/79	420979	10/80	901880	12/81	601995	2/83	778703
9/79	1026915	11/80	738200	1/82	952450	3/83	799717
10/79	141578	12/80	16350	2/82	847563	4/83	840721
11/79	0	1/81	876078	3/82	792690	5/83	869323
12/79	507068	2/81	815372	4/82	510547	6/83	846076
1/80	885427	3/81	885413	5/82	867029	7/83	872089
2/80	811187	4/81	874796	6/82	837113	8/83	866073
3/80	0	5/81	902360	7/82	873382	9/83	842740
4/80	716495	6/81	868336	8/82	868056	10/83	1496

CHEMICAL ANALYSIS REPORT
FORM 54741A

WESTINGHOUSE ADVANCED REACTOR DIVISION
ANALYTICAL LABORATORIES
WALTZ MILL SITE

ANAL. SERV. REQUEST NO. 5977

ORIGINATOR	DEPT. & GRP.	EXT.	ROOM NO.	FACILITY
S. L. ANDERSON	VASP-125			MNC

APPROVAL SIGNATURE	DATE RECEIVED	DATE REPORTED
C. A. Blackburn	MAR. 17, 1976	MAR. 25, 1976

METHOD	ANALYST	REFERENCE	METHOD	ANALYST	REFERENCE
Gamma Spectrometry	CAB.	F.I.F			

RESULTS OF ANALYSIS

ORIGINATOR'S SAMPLE NO.	ANAL. SERV. LAB. NO.	DOSIMETER MATERIAL	^{60}Co (a)	^{58}Co (b)	^{54}Mn (c)	^{137}Cs (d)	
Bot-Mid 24,25	76-806	Co	2.40×10^4				
"	-807	Co *	0.976×10^4				
Bot-Mid 26	-808	Cu	1.04×10^2				
Mid 24,25	-809	Co	2.10×10^4				
"	-810	Co *	0.791×10^4				
Mid 27	-811	Ni		1.11×10^4			
Top-Mid 24,25	-812	Co	2.50×10^4				
"	-813	Co *	0.904×10^4				
Top-Mid 26	-814	Cu	0.915×10^2				
Top 24,25	-815	Co	2.28×10^4				
"	-816	Co *	0.874×10^4				
Top 26	-817	Cu	1.05×10^2				
CHARPY-A-37	-818	Fe			1.64×10^3		
" C-37	-819	Fe			1.77×10^3		
" R-26	-820	Fe			1.75×10^3		
" R-27	-821	Fe			1.36×10^3		
" WH-32	-822	Fe			1.94×10^3		
" WW-31	-823	Fe			1.52×10^3		
Np CAPSULE	-824	^{237}Np				2.72×10^3	
U CAPSULE	76-825	^{238}U				3.23×10^2	

REMARKS:

* CADMIUM COVERED WIRE.

(a) DPS/mg of wire ± 10% @ 12:00 MAR. 23, 1976

(b) DPS/mg of wire ± 10% @ 12:00 MAR. 23, 1976

(c) DPS/mg of Fe ± 10% @ 12:00 MAR. 23, 1976

(d) { Np — DPS/mg of Np-237 ± 10% MAR. 25, 1976
 { U — DPS/mg of U-238 ± 10% MAR. 25, 1976

WEP Point Beach #1 "S" CAPSULE

CHECK ~~ORIGINAL~~ FOR LEGIBILITY

ORIGINAL ... DV

CHEMICAL ANALYSIS REPORT FORM 54741A	WESTINGHOUSE ADVANCED REACTOR DIVISION ANALYTICAL LABORATORIES WALTZ MILL SITE	ANAL. SERV. REQUEST NO. 7338
ORIGINATOR: S.L. ANDERSON	DEPT. & GRP.:	EXT. x 5165	ROOM NO.: MNC.	FACILITY:
APPROVAL SIGNATURE: C.A. Blackburn	DATE RECEIVED 4-6-78	DATE REPORTED 4-25-78

METHOD	ANALYST	REFERENCE	METHOD	ANALYST	REFERENCE
Gamma Spectrometry	WTF/CAB	FILE			

RESULTS OF ANALYSIS

ORIGINATOR'S SAMPLE NO.	ANAL. SERV. LAB. NO.	DOSIMETER MATERIAL	DPS/mg wire 4-17-78 Co-60	Co-58	CHARPY SPECIMEN	ANAL. SERV. LAB. No.	DPS/mg 4-17-78 Mn-54
TOP -24	78-795	Co	5.77 x 10⁴	--	C-13	78-813	3.87 x
" 25	-796	Co(Cd)	2.32 x 10⁴		A-13	-814	3.57 x
" 26	-797	Cu	2.04 x 10²		R-10	-815	4.38 x
TOP MID-24	-798	Co	5.56 x 10⁴		WH-16	-816	3.82 x
" 25	-799	Co (Cd)	2.27 x 10⁴		WW-15	-817	3.61 x
" 26	-800	Cu	1.79 x 10²		R-11	78-818	3.59 x
MID -24	-801	Co	5.24 x 10⁴				
" -25	-802	Co (Cd)	2.46 x 10⁴				
" -26	-803	Ni	—	1.17 x 10⁴			
BOTTOM MID -24	-804	Co	5.62 x 10⁴				
" -25	-805	Co(Cd)	2.85 x 10⁴				
" -26	-806	Cu	2.14 x 10²				
BOTTOM -24	-807	Co	5.00 x 10⁴				
-25	-808	Co(Cd)	2.67 x 10⁴				
-26	78-809	Cu	2.19 x 10²				

REMARKS:

FISSION DOSIMETERS.

U-237	78-810	9.92×10^{2} DPS/mg U-238 } 4-25-78
Np-237	78-811	7.47×10^{3} DPS/mg Np-237 }

POINT BEACH UNIT #1 "R" CAPSULE

ALL RESULTS ARE ± 10%

CHECK COPY FOR LEGIBILITY

CHEMICAL ANALYSIS REPORT

[illegible] ANALYTICAL LABORATORIES WALTZ MILL SITE

ANAL. SERV. REQUEST NO. 7338

FACILITY: MNC

ORIGINATOR: S.L. ANDERSON

DATE RECEIVED: 4-6-78

DATE REPORTED: 5-9-78

APPROVAL SIGNATURE: C. A. Blackburn

METHOD	ANALYST	REFERENCE	METHOD	ANALYST	REFERENCE
Correction to the REPORT UNDER THIS REQUEST Number REPORTED ON 4-25-78.					

RESULTS OF ANALYSIS

ORIGINATOR'S SAMPLE NO.	ANAL. SERV. LAB. NO.	Mn-54 dps/mgFe on Apr. 19, 1978 ± 10%
CHARPY SPECIMEN		
C-13	78-813	2.84×10^3
A-13	-814	2.54
R-10	-815	3.11
WH-16	-816	2.90
WW-15	-817	2.60
R-11	78-818	2.54×10^3

REMARKS:

POINT BEACH UNIT # 1 "R" CAPSULE

CHECK [illegible] LEGIBILITY

CHEMICAL ANALYSIS REPORT
FORM 54741 A

ANALYTICAL LABORATORIES
WALTZ MILL SITE

ANAL. SERV. REQUEST NO. 11508

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

APPROVAL SIGNATURE	DATE RECEIVED	DATE REPORTED
C. A. Blackburn	6-4-84	6-28-84

METHOD	ANALYST	REFERENCE	METHOD	ANALYST	REFERENCE
A-512	WTF	LAB. BOOK	#10 page 29		
A-514	CAB & RMCK		#7 page 38		
A-513	WTF		#5 page 35		

RESULTS OF ANALYSES

ORIGINATOR'S SAMPLE NO.	ANAL. SERV. LAB. NO.	DOSIMETER I.D.	(dps / mg Fe) Mn-54	on 6-20-84	(dps / mg of wire) Co-60	on 6-20-84	
CHARPY	84-1656	A-25	918				
"	-1657	R-19	830				
"	-1658	WW-23	874				
"	-1659	C-25	1066				
"	-1660	R-18	1020				
"	-1661	WH-24	1180				
WIRE TOP	-1664	CoAl			36,000		
"	-1665	CoAl (Cd)			12,000		
"	-1666	Cu			185		
MID	-1667	CoAl			31,400		
"	-1668	CoAl (Cd)			11,500		
"	-1669	Cu			184		
BOT MID	-1670	CoAl (Cd)			13,100		
"	-1671	Cu			178		
BOT	-1672	CoAl			34,800		
"	-1673	CoAl (Cd)			13,400		

REMARKS:

U-238	84-1662	Cs-137 = 926 dps / mg of U-238 on 6-20-84
Np-237	84-1663	Cs-137 = 7102 dps / mg of Np-237 on 6-20-84

POINT BEACH UNIT #1 "T" CAPSULE

BOT MID CoAl WIRE NOT FOUND.

BOT Cu WIRE HIGH IN AG-110 m AND NOT COPPER COLOR.

CHECK-COPIES FOR LEGIBILITY

ORIGINATOR COPY

APPENDIX B

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

In this appendix, the irradiation history of the fuel cycle and the measured specific activities of the radiometric sensors are provided.

The irradiation history of Cycle 17 was as follows:

CYCLE STARTUP	05/15/89
CYCLE SHUTDOWN	03/30/90
REF. CORE POWER	1518 MWt

MONTH	THERMAL GENERATION (MW-Hr)
5/89	485668
6/89	1089329
7/89	1118652
8/89	985053
9/89	951439
10/89	1092473
11/89	1081457
12/89	1124510
1/90	1129394
2/90	1007834
3/90	959825
TOTAL	11025634

The irradiation capsule loading diagram and the measured specific activities from the Cycle 17 irradiation are provided on pages B-3 through B-10. 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	SSTR PACKAGE
A-1	B	CA	--	--	--	--	BT	--	PB-2B
A-2	Cd	DA	AA	AA	A	T	DT	A	--
A-3	Cd	--	--	--	--	--	--	--	PB-2C
B-1	B	CB	--	--	--	--	BS	--	PB-8B
B-2	Cd	DB	AB	AB	B	S	DS	B	--
B-3	Cd	--	--	--	--	--	--	--	PB-8C
C-1	B	CC	--	--	--	--	BR	--	PB-3B
C-2	Cd	DC	AC	AC	C	R	DR	C	--
C-3	Cd	--	--	--	--	--	--	--	PB-3C
D-1	B	CD	--	--	--	--	BP	--	PB-9B
D-2	Cd	DD	AD	AD	D	P	DP	D	--
D-3	Cd	--	--	--	--	--	--	--	PB-9C
E-1	B	CE	--	--	--	--	BO	--	PB-10B
E-2	Cd	DE	AE	AE	E	O	DO	E	--
E-3	Cd	--	--	--	--	--	--	--	PB-10C
F-1	B	CF	--	--	--	--	BN	--	PB-11B
F-2	Cd	DF	AF	AF	F	N	DN	F	--
F-3	Cd	--	--	--	--	--	--	--	PB-11C

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

REPORT — Request# 14063

TO: S.L.Anderson (W)NATD, Energy Center (East 4-17)

Received: 6/11/90
Reported: 8/1/90

[RESULTS OF ANALYSIS]
Point Beach Unit 1 Cycle 17 Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 7/5/90) dps/mg foil		2 sigma
BT	90-1081	AlCo	Co-60	165.7	+/-	1.8
DT	90-1082	AlCo	Co-60	123.2	+/-	2.1
BS	90-1083	AlCo	Co-60	448.3	+/-	3.9
DS	90-1084	AlCo	Co-60	247.6	+/-	4.2
BR	90-1085	AlCo	Co-60	152.2	+/-	2.3
DR	90-1086	AlCo	Co-60	96.9	+/-	1.9
BP	90-1087	AlCo	Co-60	560.4	+/-	6.0
DP	90-1088	AlCo	Co-60	281.9	+/-	4.0
BO	90-1089	AlCo	Co-60	480.8	+/-	5.3
DO	90-1090	AlCo	Co-60	261.7	+/-	4.0
BN	90-1091	AlCo	Co-60	341.1	+/-	4.5
DN	90-1092	AlCo	Co-60	211.3	+/-	3.5
CA	90-1093	Fe	Mn-54	7.80	+/-	0.12
DA	90-1094	Fe	Mn-54	8.55	+/-	0.09
CB	90-1095	Fe	Mn-54	18.51	+/-	0.28
DB	90-1096	Fe	Mn-54	17.81	+/-	0.26
CC	90-1097	Fe	Mn-54	6.75	+/-	0.08
DC	90-1098	Fe	Mn-54	5.90	+/-	0.08
CD	90-1099	Fe	Mn-54	16.02	+/-	0.29
DD	90-1100	Fe	Mn-54	15.37	+/-	0.34
CE	90-1101	Fe	Mn-54	13.83	+/-	0.18
DE	90-1102	Fe	Mn-54	13.89	+/-	0.18
CF	90-1103	Fe	Mn-54	13.29	+/-	0.18
DF	90-1104	Fe	Mn-54	13.44	+/-	0.17

Remarks:

AL File: 14063
References: Lab.Book#35 page 257,261-264. LB#37p93,94.
Procedures: A-524.
Analyst: WTF, CAB.

Approved: C.A. Blackburn 8-1-90

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

REPORT Request# 14063

TO: S.L.Anderson (W)NATD, Energy Center (East 4-17)

Received: 6/11/90
Reported: 8/1/90

[RESULTS OF ANALYSIS]
Point Beach Unit 1 Cycle 17 Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 7/5/90) dps/mg foil		2 sigma
AA	90-1063	Ni	Co-58	130.9	+/-	1.4
AB	90-1064	Ni	Co-58	251.6	+/-	2.5
AC	90-1065	Ni	Co-58	88.6	+/-	1.2
AD	90-1066	Ni	Co-58	222.5	+/-	2.5
AE	90-1067	Ni	Co-58	196.6	+/-	2.3
AF	90-1068	Ni	Co-58	191.9	+/-	2.2
AA	90-1069	Cu	Co-60	0.260	+/-	0.007
AB	90-1070	Cu	Co-60	0.563	+/-	0.017
AC	90-1071	Cu	Co-60	0.187	+/-	0.006
AD	90-1072	Cu	Co-60	0.510	+/-	0.008
AE	90-1073	Cu	Co-60	0.464	+/-	0.009
AF	90-1074	Cu	Co-60	0.458	+/-	0.009
A	90-1075	Ti	Sc-46	2.61	+/-	0.04
B	90-1076	Ti	Sc-46	5.25	+/-	0.07
C	90-1077	Ti	Sc-46	1.77	+/-	0.03
D	90-1078	Ti	Sc-46	4.66	+/-	0.07
E	90-1079	Ti	Sc-46	4.13	+/-	0.07
F	90-1080	Ti	Sc-46	4.10	+/-	0.07

Remarks:

AL File: 14063
References: Lab.Book#35 page 257,261-264. LB#37p93,94.
Procedures: A-524.
Analyst: WTF, CAB.

Approved: C.A. Blackburn 8-1-90

Westinghouse Electric Corporation
Advanced Energy Systems - Analytical Laboratory
REPORT Waltz Mill Site Request# 14063

TO: S.L.Anderson (W)NATD, Energy Center (East 4-17)

Received: 6/11/90
Reported: 8/1/90

[RESULTS OF ANALYSIS]
Point Beach Unit 1 Cycle 17 Reactor Cavity Dosimetry

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 7/5/90) dps/mg foil		2 sigma
A	90-1105	U-238	Zr-95	10.96	+/-	0.08
B	90-1106	U-238	Zr-95	19.84	+/-	0.13
C	90-1107	U-238	Zr-95	7.38	+/-	0.07
D	90-1108	U-238	Zr-95	18.03	+/-	0.13
E	90-1109	U-238	Zr-95	15.85	+/-	0.12
F	90-1110	U-238	Zr-95	15.49	+/-	0.10
A	90-1105	U-238	Ru-103	6.87	+/-	0.05
B	90-1106	U-238	Ru-103	11.92	+/-	0.08
C	90-1107	U-238	Ru-103	4.38	+/-	0.04
D	90-1108	U-238	Ru-103	10.36	+/-	0.08
E	90-1109	U-238	Ru-103	9.21	+/-	0.07
F	90-1110	U-238	Ru-103	9.06	+/-	0.06
A	90-1105	U-238	Cs-137	0.691	+/-	0.020
B	90-1106	U-238	Cs-137	1.382	+/-	0.035
C	90-1107	U-238	Cs-137	0.477	+/-	0.016
D	90-1108	U-238	Cs-137	1.267	+/-	0.033
E	90-1109	U-238	Cs-137	1.101	+/-	0.030
F	90-1110	U-238	Cs-137	1.071	+/-	0.022

Remarks:

AL File: 14063
References: Lab.Book#35 page 257,261-264. LB#37p93,94.
Procedures: A-524.
Analyst: WTF, CAB.

Approved: C. A. Blackburn 8-1-90

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

REPORT

Request# 14063

TO: S.L.Anderson (W)NATD, Energy Center (East 4-17)

Received: 6/11/90
Reported: 8/7/90

[RESULTS OF ANALYSIS]

Point Beach Unit 1 Cycle 17 Reactor Cavity Dosimetry

Bead Chain Tag ID: S-1 0 degree.

Feet from Midplane	Lab Sample#	[<----- dps/mg of chain @ 7/5/90 Mn-54 dps/mg	2 sigma	Co-58 dps/mg	2 sigma	Co-60 dps/mg ----->]	2 sigma
+6.5	90-1111	3.07 +/-	0.13	6.86 +/-	0.17	41.7 +/-	0.3
+5.5	90-1112	7.82 +/-	0.26	17.16 +/-	0.35	90.1 +/-	0.5
+4.5	90-1113	12.26 +/-	0.39	26.00 +/-	0.57	127.2 +/-	0.9
+3.5	90-1114	13.64 +/-	0.43	28.27 +/-	0.58	148.6 +/-	0.9
+2.5	90-1115	14.32 +/-	0.41	29.79 +/-	0.62	161.0 +/-	1.0
+1.5	90-1116	14.06 +/-	0.45	28.39 +/-	0.60	160.1 +/-	0.9
+0.5	90-1117	11.90 +/-	0.43	24.30 +/-	0.60	161.7 +/-	0.9
-0.5	90-1118	10.86 +/-	0.44	21.21 +/-	0.59	164.3 +/-	1.0
-1.5	90-1119	9.44 +/-	0.38	19.39 +/-	0.55	152.8 +/-	0.9
-2.5	90-1120	9.17 +/-	0.37	18.52 +/-	0.51	142.6 +/-	0.9
-3.5	90-1121	8.75 +/-	0.38	18.27 +/-	0.56	128.8 +/-	0.8
-4.5	90-1122	8.04 +/-	0.32	16.17 +/-	0.45	102.6 +/-	0.7
-5.5	90-1123	5.63 +/-	0.26	12.16 +/-	0.43	65.7 +/-	0.6
-6.5	90-1124	2.42 +/-	0.14	5.12 +/-	0.18	49.3 +/-	0.3

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: 14063
References: Lab.Book#35 page 257,261-264. LB#37p93,94.
Procedures: A-524.
Analyst: WTF, CAB.

Approved: C. A. Blackburn 8-7-90

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

REPORT   Request# 14063

TO: S.L.Anderson (W)NATD, Energy Center (East 4-17)

Received: 6/11/90
Reported: 8/7/90

[RESULTS OF ANALYSIS]

Point Beach Unit 1 Cycle 17 Reactor Cavity Dosimetry

Bead Chain Tag ID: S-1 15 degree.

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 @ 7/5/90 —>]								
+6.5	90-1125	2.48	+/-	0.17	5.22	+/-	0.21	40.4	+/-	0.4
+5.5	90-1126	6.57	+/-	0.30	13.59	+/-	0.40	130.1	+/-	0.7
+4.5	90-1127	9.90	+/-	0.58	20.42	+/-	0.76	195.1	+/-	1.4
+3.5	90-1128	11.24	+/-	0.48	22.92	+/-	0.68	219.1	+/-	1.1
+2.5	90-1129	12.33	+/-	1.18	24.21	+/-	1.17	243.9	+/-	2.4
+1.5	90-1130	11.71	+/-	0.99	23.69	+/-	1.22	237.9	+/-	2.3
+0.5	90-1131	10.84	+/-	0.90	21.36	+/-	1.16	219.9	+/-	2.1
-0.5	90-1132	10.21	+/-	0.92	20.20	+/-	1.26	207.5	+/-	2.2
-1.5	90-1133	8.72	+/-	0.90	18.23	+/-	1.09	191.0	+/-	2.1
-2.5	90-1134	8.61	+/-	0.91	17.41	+/-	1.02	179.2	+/-	2.0
-3.5	90-1135	8.29	+/-	0.38	16.74	+/-	0.58	162.9	+/-	1.0
-4.5	90-1136	7.90	+/-	0.49	15.36	+/-	0.62	141.0	+/-	1.2
-5.5	90-1137	5.37	+/-	0.33	10.98	+/-	0.44	102.3	+/-	0.8
-6.5	90-1138	2.10	+/-	0.18	4.49	+/-	0.24	55.7	+/-	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: 14063
References: Lab.Book#35 page 257,261-264. LB#37p93,94.
Procedures: A-524.
Analyst: WTF, CAB.

Approved: O.A. Blackburn 8-7-90

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

REPORT Request# 14063

TO: S.L.Anderson (W)NATD, Energy Center (East 4-17)

Received: 6/11/90
Reported: 8/3/90

[RESULTS OF ANALYSIS]

Point Beach Unit 1 Cycle 17 Reactor Cavity Dosimetry

Bead Chain Tag ID: S-1 30 degree.

Feet from Midplane	Lab Sample#	[<— dpsmg of chain @ 7/5/90 —>] 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	90-1139	2.29 +/-	0.17	4.63 +/-	0.22	37.5 +/-	0.3
+5.5	90-1140	5.45 +/-	0.26	11.11 +/-	0.34	107.3 +/-	0.6
+4.5	90-1141	7.71 +/-	0.40	17.16 +/-	0.55	151.2 +/-	0.9
+3.5	90-1142	8.58 +/-	0.42	17.69 +/-	0.56	175.3 +/-	1.0
+2.5	90-1143	9.36 +/-	0.41	18.92 +/-	0.52	189.8 +/-	1.0
+1.5	90-1144	9.03 +/-	0.48	18.75 +/-	0.64	189.8 +/-	1.1
+0.5	90-1145	9.53 +/-	0.60	18.17 +/-	0.76	190.4 +/-	1.4
-0.5	90-1146	9.11 +/-	0.47	18.41 +/-	0.63	183.8 +/-	1.1
-1.5	90-1147	9.00 +/-	0.53	17.78 +/-	0.71	181.8 +/-	1.3
-2.5	90-1148	8.74 +/-	0.54	17.55 +/-	0.77	169.6 +/-	1.3
-3.5	90-1149	8.73 +/-	0.58	16.51 +/-	0.70	158.7 +/-	1.2
-4.5	90-1150	7.34 +/-	0.36	15.55 +/-	0.63	125.2 +/-	0.9
-5.5	90-1151	4.95 +/-	0.15	9.79 +/-	0.21	67.5 +/-	0.4
-6.5	90-1152	1.92 +/-	0.11	4.09 +/-	0.15	47.1 +/-	0.3

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: 14063
References: Lab.Book#35 page 257,261-264. LB#37p93,94.
Procedures: A-524.
Analyst: WTF, CAB.

Approved: C.A. Blackburn 8-3-90

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

REPORT | Request# 14063

TO: S.L.Anderson (W)NATD, Energy Center (East 4-17)

Received: 6/11/90
Reported: 8/3/90

[RESULTS OF ANALYSIS]

Point Beach Unit 1 Cycle 17 Reactor Cavity Dosimetry

Bead Chain Tag ID: S-1 45 degree.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 7/5/90					
		Mn-54		Co-58		Co-60	
		dps/mg	2 sigma	dps/mg	2 sigma	dps/mg	2 sigma
+6.5	90-1153	2.10 +/-	0.13	4.52 +/-	0.19	34.3 +/-	0.3
+5.5	90-1154	4.94 +/-	0.22	10.31 +/-	0.28	68.7 +/-	0.5
+4.5	90-1155	7.66 +/-	0.27	15.57 +/-	0.34	95.5 +/-	0.6
+3.5	90-1156	7.92 +/-	0.28	16.74 +/-	0.35	111.9 +/-	0.6
+2.5	90-1157	9.06 +/-	0.28	18.02 +/-	0.36	123.5 +/-	0.7
+1.5	90-1158	8.85 +/-	0.38	18.26 +/-	0.63	128.4 +/-	0.9
+0.5	90-1159	8.80 +/-	0.28	17.36 +/-	0.38	124.6 +/-	0.6
-0.5	90-1160	8.81 +/-	0.39	17.82 +/-	0.49	128.0 +/-	0.9
-1.5	90-1161	9.07 +/-	0.41	17.13 +/-	0.48	125.6 +/-	0.9
-2.5	90-1162	8.56 +/-	0.42	17.35 +/-	0.49	119.4 +/-	0.8
-3.5	90-1163	8.10 +/-	0.35	15.94 +/-	0.47	110.6 +/-	0.8
-4.5	90-1164	7.35 +/-	0.34	14.86 +/-	0.45	90.7 +/-	0.7
-5.5	90-1165	4.92 +/-	0.16	10.09 +/-	0.21	65.1 +/-	0.3
-6.5	90-1166	1.90 +/-	0.12	4.10 +/-	0.16	50.3 +/-	0.3

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: 14063
References: Lab.Book#35 page 257,261-264. LB#37p93,94.
Procedures: A-524.
Analyst: WTF, CAB.

Approved: C.A. Blackburn 8-3-90

APPENDIX C

MEASURED SPECIFIC ACTIVITY AND IRRADIATION HISTORY OF CYCLE 18 REACTOR CAVITY SENSOR SETS

In this appendix, the irradiation history of the fuel cycle and the measured specific activities of the radiometric sensors are provided.

The irradiation history of Cycle 18 was as follows:

CYCLE STARTUP	05/17/90
CYCLE SHUTDOWN	04/07/91
REF. CORE POWER	1518 MWt

MONTH	THERMAL GENERATION (MW-Hr)
5/90	419623
6/90	1063672
7/90	770935
8/90	1063585
9/90	1049622
10/90	1103418
11/90	1072594
12/90	1109270
1/91	1109675
2/91	1003357
3/91	1112312
4/91	177264
TOTAL	11055327

The irradiation capsule loading diagram and the measured specific activities from the Cycle 18 irradiation are provided on pages C-3 through C-10. 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 18 IRRADIATION

CAPSULE ID and POSITION	BARE OR CADMIUM SHIELDED	RADIOMETRIC MONITOR ID							SSTR PACKAGE
		Fe	Ni	Cu	Ti	Nb	Co	U-238	
S-1	B	BA	--	--	--	--	BA	--	PB-21B
S-2	Cd	AA	AA	A	A	A	DA	A	--
S-3	Cd	--	--	--	--	--	--	--	PB-21C
T-1	B	BB	--	--	--	--	BB	--	PB-23B
T-2	Cd	AB	AB	B	B	B	AB	B	--
T-3	Cd	--	--	--	--	--	--	--	PB-23C
U-1	B	BC	--	--	--	--	BC	--	PB-22B
U-2	Cd	AC	AC	C	C	C	AC	C	--
U-3	Cd	--	--	--	--	--	--	--	PB-22C
V-1	B	BD	--	--	--	--	BD	--	PB-24B
V-2	Cd	AD	AD	D	D	D	AD	D	--
V-3	Cd	--	--	--	--	--	--	--	PB-24C
W-1	B	BE	--	--	--	--	BE	--	PB-25B
W-2	Cd	AE	AE	E	E	E	AE	E	--
W-3	Cd	--	--	--	--	--	--	--	PB-25C
X-1	B	BF	--	--	--	--	BF	--	PB-26B
X-2	Cd	AF	AF	F	F	F	AF	F	--
X-3	Cd	--	--	--	--	--	--	--	PB-26C

REPORT | Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site | Request# 14344

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

Received: 5/20/91
Reported: 8/28/91

[RESULTS OF ANALYSIS]
POINT BEACH UNIT 1 CYCLE 18 REACTOR CAVITY DOSIMETRY

Capsule/ Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 8/01/91) dps/mg *		2 sigma
S/AA	91-1066	Fe	Mn-54	1.07E+01	+/-	1.0E-01
S/BA	91-1067	Fe	Mn-54	9.97E+00	+/-	1.0E-01
T/AC	91-1075	Fe	Mn-54	4.13E+00	+/-	6.4E-02
T/BC	91-1076	Fe	Mn-54	4.65E+00	+/-	7.1E-02
U/AB	91-1084	Fe	Mn-54	1.56E+01	+/-	1.3E-01
U/BB	91-1085	Fe	Mn-54	1.61E+01	+/-	1.3E-01
W/AE	91-1093	Fe	Mn-54	1.29E+01	+/-	1.1E-01
W/BE	91-1094	Fe	Mn-54	1.28E+01	+/-	1.1E-01
X/AF	91-1102	Fe	Mn-54	1.20E+01	+/-	1.1E-01
X/BF	91-1103	Fe	Mn-54	1.21E+01	+/-	1.1E-01
V/AD	91-1111	Fe	Mn-54	1.46E+01	+/-	1.2E-01
V/BD	91-1112	Fe	Mn-54	1.49E+01	+/-	1.3E-01
S/A	91-1068	Ni	Co-58	1.42E+02	+/-	1.5E+00
T/C	91-1077	Ni	Co-58	5.59E+01	+/-	8.3E-01
U/B	91-1086	Ni	Co-58	1.97E+02	+/-	1.7E+00
W/E	91-1095	Ni	Co-58	1.65E+02	+/-	1.6E+00
X/F	91-1104	Ni	Co-58	1.51E+02	+/-	1.5E+00
V/D	91-1113	Ni	Co-58	1.84E+02	+/-	1.7E+00
S/AA	91-1072	AlCo	Co-60	1.42E+02	+/-	2.3E+00
S/BA	91-1073	AlCo	Co-60	2.02E+02	+/-	2.7E+00
T/AC	91-1081	AlCo	Co-60	8.49E+01	+/-	1.8E+00
T/BC	91-1082	AlCo	Co-60	1.41E+02	+/-	2.3E+00
U/AB	91-1090	AlCo	Co-60	2.36E+02	+/-	2.8E+00
U/BB	91-1091	AlCo	Co-60	4.31E+02	+/-	4.0E+00
W/AE	91-1099	AlCo	Co-60	2.40E+02	+/-	3.0E+00
W/BE	91-1100	AlCo	Co-60	4.72E+02	+/-	4.1E+00
X/AF	91-1108	AlCo	Co-60	1.96E+02	+/-	2.7E+00
X/BF	91-1109	AlCo	Co-60	3.15E+02	+/-	3.4E+00
V/AD	91-1117	AlCo	Co-60	2.69E+02	+/-	3.1E+00
V/BD	91-1118	AlCo	Co-60	5.40E+02	+/-	4.4E+00

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

AL File: 14344
References: Lab Book#49 pages 108-112; Lab Book#46 page 172
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: Mark Kanghak

REPORT | Westinghouse Advanced Energy Systems | Request# 14344
| Analytical Laboratory - Waltz Mill Site |

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

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[RESULTS OF ANALYSIS]

POINT BEACH UNIT 1 CYCLE 18 REACTOR CAVITY DOSIMETRY

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 8/01/91) dps/mg *		2 sigma
S/A	91-1074	U-238	Cs-137	8.68E-01	+/-	2.04E-02
T/C	91-1083	U-238	Cs-137	3.85E-01	+/-	1.31E-02
U/B	91-1092	U-238	Cs-137	1.30E+00	+/-	2.30E-02
W/E	91-1101	U-238	Cs-137	1.05E+00	+/-	2.10E-02
X/F	91-1110	U-238	Cs-137	9.83E-01	+/-	2.16E-02
V/D	91-1119	U-238	Cs-137	1.26E+00	+/-	2.30E-02
S/A	91-1074	U-238	Ru-103	5.95E+00	+/-	6.10E-02
T/C	91-1083	U-238	Ru-103	2.64E+00	+/-	3.75E-02
U/B	91-1092	U-238	Ru-103	8.34E+00	+/-	6.28E-02
W/E	91-1101	U-238	Ru-103	6.78E+00	+/-	6.21E-02
X/F	91-1110	U-238	Ru-103	6.25E+00	+/-	6.46E-02
V/D	91-1119	U-238	Ru-103	7.91E+00	+/-	6.75E-02
S/A	91-1074	U-238	Zr-95	1.09E+01	+/-	8.82E-02
T/C	91-1083	U-238	Zr-95	4.96E+00	+/-	5.59E-02
U/B	91-1092	U-238	Zr-95	1.58E+01	+/-	1.01E-01
W/E	91-1101	U-238	Zr-95	1.32E+01	+/-	9.41E-02
X/F	91-1110	U-238	Zr-95	1.19E+01	+/-	9.91E-02
V/D	91-1119	U-238	Zr-95	1.52E+01	+/-	1.00E-01
S/A	91-1070	Ti	Sc-46	2.84E+00	+/-	5.21E-02
T/C	91-1079	Ti	Sc-46	1.13E+00	+/-	3.18E-02
U/B	91-1088	Ti	Sc-46	4.05E+00	+/-	5.92E-02
W/E	91-1097	Ti	Sc-46	3.45E+00	+/-	5.43E-02
X/F	91-1106	Ti	Sc-46	3.30E+00	+/-	5.41E-02
V/D	91-1115	Ti	Sc-46	3.93E+00	+/-	5.82E-02

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

AL File: 14344
References: Lab Book#49 pages 108-112; Lab Book#46 page 172
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: T. Mark Kawchak

REPORT | Westinghouse Advanced Energy Systems | Request# 14344
| Analytical Laboratory - Waltz Mill Site |

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[RESULTS OF ANALYSIS]
POINT BEACH UNIT 1 CYCLE 18 REACTOR CAVITY DOSIMETRY

Capsule/ Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 8/01/91) dps/mg *		2 sigma
S/A	91-1069	Cu	Co-60	3.38E-01	+/-	1.79E-02
T/C	91-1078	Cu	Co-60	1.31E-01	+/-	1.13E-02
U/B	91-1087	Cu	Co-60	5.00E-01	+/-	2.13E-02
W/E	91-1096	Cu	Co-60	4.40E-01	+/-	2.05E-02
X/F	91-1105	Cu	Co-60	4.19E-01	+/-	1.98E-02
V/D	91-1114	Cu	Co-60	4.78E-01	+/-	2.10E-02

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

AL File: 14344
References: Lab Book#49 pages 108-112; Lab Book#46 page 172
Procedures: A-524.
Analyst: WTP, TRK, MRK, FRC

Approved: Mark Kawchak

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[RESULTS OF ANALYSIS]

POINT BEACH UNIT 1 CYCLE 18 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: 0 deg.

Feet from Midplane	Lab Sample#	[<------ dps/mg of chain @ 8/01/91					------>]
		Mn-54		Co-58		Co-60	
		dps/mg	2 sigma	dps/mg	2 sigma	dps/mg	2 sigma
+7.5	91-1062A	1.86E+00 +/-	9.6E-02	3.40E+00 +/-	2.0E-01	2.89E+01 +/-	2.2E-01
+6.5	91-1062B	4.33E+00 +/-	2.1E-01	7.67E+00 +/-	2.4E-01	4.80E+01 +/-	4.1E-01
+5.5	91-1062C	8.63E+00 +/-	4.5E-01	1.70E+01 +/-	6.6E-01	8.70E+01 +/-	9.8E-01
+4.5	91-1062D	1.25E+01 +/-	5.7E-01	2.27E+01 +/-	7.9E-01	1.20E+02 +/-	1.2E+00
+3.5	91-1062E	1.27E+01 +/-	5.3E-01	2.28E+01 +/-	7.6E-01	1.27E+02 +/-	1.2E+00
+2.5	91-1062F	1.31E+01 +/-	5.6E-01	2.29E+01 +/-	6.9E-01	1.33E+02 +/-	1.2E+00
+1.5	91-1062G	1.28E+01 +/-	5.4E-01	2.20E+01 +/-	7.5E-01	1.32E+02 +/-	1.2E+00
+0.5	91-1062H	1.12E+01 +/-	5.3E-01	1.94E+01 +/-	7.4E-01	1.42E+02 +/-	1.3E+00
-0.5	91-1062I	9.87E+00 +/-	6.2E-01	1.71E+01 +/-	7.8E-01	1.37E+02 +/-	1.2E+00
-1.5	91-1062J	9.22E+00 +/-	5.3E-01	1.70E+01 +/-	7.5E-01	1.29E+02 +/-	1.2E+00
-2.5	91-1062K	8.91E+00 +/-	4.9E-01	1.64E+01 +/-	7.1E-01	1.21E+02 +/-	1.2E+00
-3.5	91-1062L	8.25E+00 +/-	5.3E-01	1.52E+01 +/-	6.1E-01	1.07E+02 +/-	1.1E+00
-4.5	91-1062M	6.71E+00 +/-	3.9E-01	1.24E+01 +/-	5.6E-01	7.47E+01 +/-	9.1E-01
-5.5	91-1062N	4.48E+00 +/-	1.9E-01	8.03E+00 +/-	2.6E-01	5.09E+01 +/-	4.2E-01
-6.5	91-1062O	1.51E+00 +/-	1.4E-01	3.01E+00 +/-	2.0E-01	4.17E+01 +/-	3.8E-01

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

AL File: 14344
References: Lab Book#49 pages 108-112; Lab Book#46 pages 172
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: Mark Kurshak

REPORT | Westinghouse Advanced Energy Systems | Request# 14344
Analytical Laboratory - Waltz Mill Site

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[RESULTS OF ANALYSIS]

POINT BEACH UNIT 1 CYCLE 18 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: 30 deg.

Feet from Midplane	Lab Sample#	dps/mg of chain @ 8/01/91: 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	91-1063A	6.44E-01	+/-	6.6E-02	1.17E+00	+/-	6.8E-02	1.84E+01	+/-	1.5E-01
+6.5	91-1063B	1.90E+00	+/-	1.3E-01	3.57E+00	+/-	1.7E-01	3.16E+01	+/-	3.3E-01
+5.5	91-1063C	4.80E+00	+/-	3.7E-01	8.66E+00	+/-	2.3E-01	8.75E+01	+/-	9.7E-01
+4.5	91-1063D	7.20E+00	+/-	4.9E-01	1.31E+01	+/-	5.6E-01	1.25E+02	+/-	1.2E+00
+3.5	91-1063E	7.64E+00	+/-	5.0E-01	1.44E+01	+/-	6.2E-01	1.45E+02	+/-	1.3E+00
+2.5	91-1063F	8.57E+00	+/-	5.7E-01	1.49E+01	+/-	6.5E-01	1.55E+02	+/-	1.3E+00
+1.5	91-1063G	9.08E+00	+/-	5.9E-01	1.48E+01	+/-	6.8E-01	1.56E+02	+/-	1.3E+00
+0.5	91-1063H	8.71E+00	+/-	6.2E-01	1.41E+01	+/-	6.1E-01	1.53E+02	+/-	1.3E+00
-0.5	91-1063I	8.39E+00	+/-	5.4E-01	1.49E+01	+/-	6.4E-01	1.52E+02	+/-	1.3E+00
-1.5	91-1063J	8.75E+00	+/-	6.7E-01	1.41E+01	+/-	3.2E-01	1.48E+02	+/-	1.3E+00
-2.5	91-1063K	7.46E+00	+/-	5.4E-01	1.43E+01	+/-	6.4E-01	1.39E+02	+/-	1.2E+00
-3.5	91-1063L	8.08E+00	+/-	6.1E-01	1.36E+01	+/-	3.3E-01	1.29E+02	+/-	1.2E+00
-4.5	91-1063M	7.01E+00	+/-	5.2E-01	1.22E+01	+/-	5.3E-01	1.02E+02	+/-	1.1E+00
-5.5	91-1063N	4.31E+00	+/-	2.1E-01	7.94E+00	+/-	2.3E-01	5.53E+01	+/-	2.2E-01
-6.5	91-1063O	1.94E+00	+/-	8.7E-02	3.51E+00	+/-	9.6E-02	3.92E+01	+/-	3.7E-01

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

AL File: 14344
References: Lab Book#49 pages 108-112; Lab Book#46 page 172
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: Mark Kawchak

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14344

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[RESULTS OF ANALYSIS]

POINT BEACH UNIT 1 CYCLE 18 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: 45 degrees

Feet from Midplane	Lab Sample#	dps/mg of chain @ 8/01/91 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	91-1064A	8.41E-01 +/-	1.3E-01	1.37E+00 +/-	1.3E-01	2.08E+01 +/-	2.7E-01
+6.5	91-1064B	2.28E+00 +/-	1.5E-01	4.21E+00 +/-	2.0E-01	3.23E+01 +/-	3.3E-01
+5.5	91-1064C	4.98E+00 +/-	2.0E-01	9.10E+00 +/-	2.9E-01	6.24E+01 +/-	4.7E-01
+4.5	91-1064D	7.22E+00 +/-	4.5E-01	1.21E+01 +/-	5.2E-01	8.22E+01 +/-	9.4E-01
+3.5	91-1064E	7.13E+00 +/-	3.9E-01	1.33E+01 +/-	6.1E-01	9.41E+01 +/-	1.0E+00
+2.5	91-1064F	7.17E+00 +/-	5.0E-01	1.36E+01 +/-	7.0E-01	1.03E+02 +/-	1.2E+00
+1.5	91-1064G	8.03E+00 +/-	6.0E-01	1.40E+01 +/-	7.3E-01	1.02E+02 +/-	1.2E+00
+0.5	91-1064H	7.91E+00 +/-	5.2E-01	1.41E+01 +/-	7.2E-01	1.02E+02 +/-	1.1E+00
-0.5	91-1064I	8.05E+00 +/-	6.1E-01	1.35E+01 +/-	6.9E-01	1.03E+02 +/-	1.2E+00
-1.5	91-1064J	7.87E+00 +/-	5.2E-01	1.41E+01 +/-	7.2E-01	1.01E+02 +/-	1.2E+00
-2.5	91-1064K	7.66E+00 +/-	5.4E-01	1.41E+01 +/-	7.6E-01	9.61E+01 +/-	1.2E+00
-3.5	91-1064L	6.88E+00 +/-	5.6E-01	1.30E+01 +/-	6.8E-01	8.72E+01 +/-	1.1E+00
-4.5	91-1064M	5.93E+00 +/-	4.4E-01	1.08E+01 +/-	5.3E-01	6.96E+01 +/-	8.8E-01
-5.5	91-1064N	3.69E+00 +/-	3.1E-01	7.07E+00 +/-	4.4E-01	5.12E+01 +/-	7.5E-01
-6.5	91-1064O	1.47E+00 +/-	1.6E-01	2.46E+00 +/-	1.9E-01	4.13E+01 +/-	3.8E-01

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

AL File: 14344
References: Lab Book#49 pages 108-112; Lab Book#46 pages 172
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: Mark Kawchak

REPORT	Westinghouse Advanced Energy Systems Analytical Laboratory - Waltz Mill Site	Request# 14344

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

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[RESULTS OF ANALYSIS]

POINT BEACH UNIT 1 CYCLE 18 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: V

Feet from Midplane	Lab Sample#	[<---- dps/mg of chain @ 8/01/91 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	91-1065A	9.36E-01	+/-	1.1E-01	1.62E+00	+/-	1.4E-01	2.31E+01	+/-	1.4E-01
+6.5	91-1065B	2.79E+00	+/-	2.3E-01	5.11E+00	+/-	2.7E-01	3.76E+01	+/-	4.7E-01
+5.5	91-1065C	6.44E+00	+/-	6.4E-01	1.24E+01	+/-	4.7E-01	1.18E+02	+/-	1.5E+00
+4.5	91-1065D	9.58E+00	+/-	7.4E-01	1.73E+01	+/-	1.0E+00	1.64E+02	+/-	1.7E+00
+3.5	91-1065E	9.92E+00	+/-	8.3E-01	1.79E+01	+/-	9.8E-01	1.88E+02	+/-	1.9E+00
+2.5	91-1065F	1.06E+01	+/-	8.2E-01	1.92E+01	+/-	1.0E+00	2.00E+02	+/-	1.9E+00
+1.5	91-1065G	1.06E+01	+/-	7.4E-01	1.74E+01	+/-	9.1E-01	1.92E+02	+/-	1.7E+00
+0.5	91-1065H	9.16E+00	+/-	7.3E-01	1.66E+01	+/-	9.7E-01	1.77E+02	+/-	1.8E+00
-0.5	91-1065I	9.37E+00	+/-	7.6E-01	1.63E+01	+/-	9.5E-01	1.74E+02	+/-	1.8E+00
-1.5	91-1065J	8.61E+00	+/-	6.9E-01	1.59E+01	+/-	9.2E-01	1.60E+02	+/-	1.7E+00
-2.5	91-1065K	8.50E+00	+/-	7.2E-01	1.45E+01	+/-	9.1E-01	1.51E+02	+/-	1.7E+00
-3.5	91-1065L	7.75E+00	+/-	6.1E-01	1.42E+01	+/-	9.0E-01	1.39E+02	+/-	1.6E+00
-4.5	91-1065M	6.57E+00	+/-	6.0E-01	1.26E+01	+/-	7.9E-01	1.17E+02	+/-	1.5E+00
-5.5	91-1065N	4.56E+00	+/-	2.8E-01	8.53E+00	+/-	3.8E-01	8.15E+01	+/-	6.7E-01
-6.5	91-1065O	1.70E+00	+/-	1.0E-01	3.31E+00	+/-	1.4E-01	4.20E+01	+/-	2.6E-01

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

AL File: 14344
References: Lab Book#49 pages 108-112; Lab Book#46 pages 172
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: [signature]

APPENDIX D

MEASURED SPECIFIC ACTIVITY AND IRRADIATION HISTORY OF CYCLE 19 REACTOR CAVITY SENSOR SETS

In this appendix, the irradiation history of the fuel cycle and the measured specific activities of the radiometric sensors are provided.

The irradiation history of Cycle 19 was as follows:

CYCLE STARTUP	05/21/91
CYCLE SHUTDOWN	04/12/92
REF. CORE POWER	1518 MWt

MONTH	THERMAL GENERATION (MW-Hr)
5/91	312554
6/91	1027546
7/91	1126841
8/91	1128204
9/91	1084952
10/91	1058783
11/91	957418
12/91	1091753
1/92	1126601
2/92	1054684
3/92	1093321
4/92	366568
TOTAL	11429225

The irradiation capsule loading diagram and the measured specific activities from the Cycle 19 irradiation are provided on pages D-3 through D-10. 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.

CONTENTS OF MULTIPLE FOIL SENSOR SETS
CYCLE 19 IRRADIATION

CAPSULE ID and POSITION	BARE OR CADMIUM SHIELDED	RADIOMETRIC MONITOR ID Fe	Ni	Cu	Ti	Nb	Co	U-238	SSTR PACKAGE
GG-1	B	BM	--	--	--	--	BM	--	PB-33B
GG-2	Cd	AM	M	M	M	M	AM	M	--
GG-3	Cd	--	--	--	--	--	--	--	PB-33C
HH-1	B	BN	--	--	--	--	BN	--	PB-35B
HH-2	Cd	AN	N	N	N	N	AN	N	--
HH-3	Cd	--	--	--	--	--	--	--	PB-35C
II-1	B	BO	--	--	--	--	BO	--	PB-34B
II-2	Cd	AO	O	O	O	O	AO	O	--
II-3	Cd	--	--	--	--	--	--	--	PB-34C
JJ-1	B	BP	--	--	--	--	BP	--	PB-36B
JJ-2	Cd	AP	P	P	P	P	AP	BC	--
JJ-3	Cd	--	--	--	--	--	--	--	PB-36C
KK-1	B	BR	--	--	--	--	BR	--	PB-37B
KK-2	Cd	AR	R	R	R	R	AR	BD	--
KK-3	Cd	--	--	--	--	--	--	--	PB-37C
LL-1	B	BS	--	--	--	--	BS	--	PB-38B
LL-2	Cd	AS	S	S	S	S	AS	BE	--
LL-3	Cd	--	--	--	--	--	--	--	PB-38C

REPORT — Westinghouse Advanced Programs
Analytical Laboratory - Waltz Mill Site — Request# 14667

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

Received: 5/5/92
Reported: 7/10/92

[RESULTS OF ANALYSIS]

POINT BEACH UNIT 1 CYCLE 19 REACTOR CAVITY DOSIMETRY

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 6/17/92) dps/mg *		2 sigma
M	92-1288	U-238	Cs-137	7.36E-01	+/-	7.74E-02
N	92-1297	U-238	Cs-137	1.36E+00	+/-	1.03E-01
O	92-1306	U-238	Cs-137	3.73E-01	+/-	2.10E-02
P	92-1315	U-238	Cs-137	1.34E+00	+/-	5.64E-02
R	92-1324	U-238	Cs-137	1.33E+00	+/-	1.10E-01
S	92-1333	U-238	Cs-137	1.16E+00	+/-	6.06E-02
M	92-1288	U-238	Ru-103	1.19E+01	+/-	2.41E-01
N	92-1297	U-238	Ru-103	1.97E+01	+/-	3.35E-01
O	92-1306	U-238	Ru-103	6.11E+00	+/-	6.58E-02
P	92-1315	U-238	Ru-103	1.85E+01	+/-	3.96E-01
R	92-1324	U-238	Ru-103	1.72E+01	+/-	3.55E-01
S	92-1333	U-238	Ru-103	1.57E+01	+/-	2.86E-01
M	92-1288	U-238	Zr-95	1.52E+01	+/-	3.70E-01
N	92-1297	U-238	Zr-95	2.72E+01	+/-	4.76E-01
O	92-1306	U-238	Zr-95	8.27E+00	+/-	8.91E-02
P	92-1315	U-238	Zr-95	2.59E+01	+/-	5.43E-01
R	92-1324	U-238	Zr-95	2.34E+01	+/-	5.27E-01
S	92-1333	U-238	Zr-95	2.16E+01	+/-	4.41E-01
M	92-1284	Ti	Sc-46	3.45E+00	+/-	1.09E-01
N	92-1293	Ti	Sc-46	6.19E+00	+/-	1.12E-01
O	92-1302	Ti	Sc-46	1.76E+00	+/-	5.56E-02
P	92-1311	Ti	Sc-46	5.98E+00	+/-	1.46E-01
R	92-1320	Ti	Sc-46	5.67E+00	+/-	1.42E-01
S	92-1329	Ti	Sc-46	5.44E+00	+/-	1.39E-01

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

AL File: 14667
References: Lab Book#56 pages 59-60
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: Mark Kurchak

REPORT | Westinghouse Advanced Programs Analytical Laboratory - Waltz Mill Site | Request# 14667

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

Received: 5/5/92
Reported: 7/10/92

[RESULTS OF ANALYSIS]
POINT BEACH UNIT 1 CYCLE 19 REACTOR CAVITY DOSIMETRY

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 6/17/92) dps/mg *		2 sigma
BM	92-1280	Fe	Mn-54	8.90E+00	+/-	1.1E-01
AM	92-1281	Fe	Mn-54	9.47E+00	+/-	1.1E-01
BN	92-1289	Fe	Mn-54	1.79E+01	+/-	1.8E-01
AN	92-1290	Fe	Mn-54	1.76E+01	+/-	1.4E-01
BO	92-1298	Fe	Mn-54	5.42E+00	+/-	9.3E-02
AO	92-1299	Fe	Mn-54	4.79E+00	+/-	7.6E-02
BP	92-1307	Fe	Mn-54	1.72E+01	+/-	1.8E-01
AP	92-1308	Fe	Mn-54	1.71E+01	+/-	1.5E-01
BR	92-1316	Fe	Mn-54	1.60E+01	+/-	1.7E-01
AR	92-1317	Fe	Mn-54	1.58E+01	+/-	1.4E-01
BS	92-1325	Fe	Mn-54	1.51E+01	+/-	1.5E-01
AS	92-1326	Fe	Mn-54	1.50E+01	+/-	1.3E-01
M	92-1282	Ni	Co-58	1.77E+02	+/-	1.3E+00
N	92-1291	Ni	Co-58	3.13E+02	+/-	2.6E+00
O	92-1300	Ni	Co-58	8.98E+01	+/-	1.4E+00
P	92-1309	Ni	Co-58	2.98E+02	+/-	2.5E+00
R	92-1318	Ni	Co-58	2.81E+02	+/-	2.4E+00
S	92-1327	Ni	Co-58	2.62E+02	+/-	2.4E+00
BM	91-1286	AlCo	Co-60	1.77E+02	+/-	1.7E+00
AM	91-1287	AlCo	Co-60	1.25E+02	+/-	1.4E+00
BN	91-1295	AlCo	Co-60	4.37E+02	+/-	2.7E+00
AN	91-1296	AlCo	Co-60	2.47E+02	+/-	2.0E+00
BO	91-1304	AlCo	Co-60	1.35E+02	+/-	1.9E+00
AO	91-1305	AlCo	Co-60	8.27E+01	+/-	1.2E+00
BP	91-1313	AlCo	Co-60	5.60E+02	+/-	4.3E+00
AP	91-1314	AlCo	Co-60	2.80E+02	+/-	3.0E+00
BR	91-1322	AlCo	Co-60	4.92E+02	+/-	3.9E+00
AR	91-1323	AlCo	Co-60	2.60E+02	+/-	2.8E+00
BS	91-1331	AlCo	Co-60	3.43E+02	+/-	3.1E+00
AS	91-1332	AlCo	Co-60	2.14E+02	+/-	2.6E+00

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

AL File: 14667
References: Lab Book#56 pages 59-60
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: [signature]

REPORT

Westinghouse Advanced Programs
Analytical Laboratory - Waltz Mill Site

Request# 1466,

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

Received: 5/5/92
Reported: 7/10/92

[RESULTS OF ANALYSIS]

POINT BEACH UNIT 1 CYCLE 19 REACTOR CAVITY DOSIMETRY

Foil ID	Lab Sample#	Dosimeter Material	Nuclide	(@ 6/17/92) dps/mg *		2 sigma
M	92-1283	Cu	Co-60	2.69E-01	+/-	5.28E-03
N	92-1292	Cu	Co-60	5.34E-01	+/-	9.23E-03
O	92-1301	Cu	Co-60	1.40E-01	+/-	4.68E-03
P	92-1310	Cu	Co-60	5.27E-01	+/-	9.07E-03
R	92-1319	Cu	Co-60	5.13E-01	+/-	7.21E-03
S	92-1328	Cu	Co-60	4.85E-01	+/-	8.05E-03

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

AL File: 14667
References: Lab Book#56 pages 59-60
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: Mark Kunzleh

REPORT — Westinghouse Advanced Programs
Analytical Laboratory - Waltz Mill Site — Request# 14667

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

Received: 5/5/92
Reported: 8/20/92

[RESULTS OF ANALYSIS]

POINT BEACH UNIT ~~2~~ 1 CYCLE ~~17~~ 19 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: S-6 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 @ ~~7~~ 6/17/92					
+7.5	92-1276A	1.02E+00	+/- 8.5E-02	2.78E+00	+/- 1.1E-01	1.87E+01	+/- 1.3E-01
+6.5	92-1276B	3.38E+00	+/- 3.5E-01	8.92E+00	+/- 5.2E-01	3.04E+01	+/- 4.8E-01
+5.5	92-1276C	8.37E+00	+/- 1.0E+00	2.24E+01	+/- 1.5E+00	8.27E+01	+/- 1.4E+00
+4.5	92-1276D	1.29E+01	+/- 1.1E+00	3.27E+01	+/- 1.9E+00	1.14E+02	+/- 8.3E-01
+3.5	92-1276E	1.40E+01	+/- 1.3E+00	3.46E+01	+/- 2.0E+00	1.32E+02	+/- 1.8E+00
+2.5	92-1276F	1.48E+01	+/- 1.4E+00	3.77E+01	+/- 2.2E+00	1.43E+02	+/- 1.9E+00
+1.5	92-1276G	1.36E+01	+/- 1.4E+00	3.61E+01	+/- 2.1E+00	1.42E+02	+/- 1.9E+00
+0.5	92-1276H	1.24E+01	+/- 1.4E+00	3.19E+01	+/- 2.0E+00	1.51E+02	+/- 1.9E+00
-0.5	92-1276I	1.10E+01	+/- 1.2E+00	2.84E+01	+/- 2.0E+00	1.22E+02	+/- 1.7E+00
-1.5	92-1276J	1.02E+01	+/- 1.3E+00	2.68E+01	+/- 1.8E+00	1.15E+02	+/- 1.7E+00
-2.5	92-1276K	9.72E+00	+/- 1.1E+00	2.52E+01	+/- 1.8E+00	1.08E+02	+/- 1.6E+00
-3.5	92-1276L	9.04E+00	+/- 1.1E+00	2.44E+01	+/- 1.7E+00	9.43E+01	+/- 1.5E+00
-4.5	92-1276M	7.72E+00	+/- 9.7E-01	2.11E+01	+/- 1.5E+00	6.99E+01	+/- 1.3E+00
-5.5	92-1276N	4.86E+00	+/- 4.1E-01	1.31E+01	+/- 6.7E-01	4.49E+01	+/- 5.9E-01
-6.5	92-1276O	1.82E+00	+/- 4.1E-01	4.75E+00	+/- 4.9E-01	4.21E+01	+/- 5.6E-01

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

AL File: 14667
References: Lab Book#56 pages 59-60; Lab Book #59 Pages 3-7
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: [signature]

REPORT

Westinghouse Advanced Programs
Analytical Laboratory - Waltz Mill Site

Request# 14667

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

Received: 5/5/92
Reported: 8/20/92

[RESULTS OF ANALYSIS]

POINT BEACH UNIT ~~2~~ 1 CYCLE ~~17~~ 19 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: S-6 15 deg.

Feet from Midplane	Lab Sample#	dps/mg of chain @ ~~7~~ 6/17/92: 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	92-1277A	9.53E-01	+/-	1.4E-01	2.54E+00	+/-	1.8E-01	2.33E+01	+/-	2.1E-01
+6.5	92-1277B	2.61E+00	+/-	1.7E-01	7.30E+00	+/-	2.9E-01	3.87E+01	+/-	2.7E-01
+5.5	92-1277C	6.60E+00	+/-	5.8E-01	1.86E+01	+/-	9.0E-01	1.24E+02	+/-	8.7E-01
+4.5	92-1277D	1.08E+01	+/-	7.2E-01	2.64E+01	+/-	1.0E+00	1.72E+02	+/-	[illegible]
+3.5	92-1277E	1.05E+01	+/-	5.0E-01	2.80E+01	+/-	8.6E-01	1.92E+02	+/-	[illegible]
+2.5	92-1277F	1.28E+01	+/-	8.5E-01	3.08E+01	+/-	1.3E+00	2.18E+02	+/-	1.2E+00
+1.5	92-1277G	1.19E+01	+/-	7.5E-01	2.89E+01	+/-	1.2E+00	2.09E+02	+/-	1.2E+00
+0.5	92-1277H	1.13E+01	+/-	8.4E-01	2.76E+01	+/-	1.1E+00	1.97E+02	+/-	1.1E+00
-0.5	92-1277I	1.03E+01	+/-	8.9E-01	2.71E+01	+/-	1.2E+00	1.91E+02	+/-	1.1E+00
-1.5	92-1277J	9.80E+00	+/-	7.6E-01	2.52E+01	+/-	1.2E+00	1.78E+02	+/-	1.1E+00
-2.5	92-1277K	9.59E+00	+/-	7.8E-01	2.42E+01	+/-	1.1E+00	1.67E+02	+/-	1.0E+00
-3.5	92-1277L	8.73E+00	+/-	7.1E-01	2.26E+01	+/-	1.0E+00	1.51E+02	+/-	9.7E-01
-4.5	92-1277M	7.51E+00	+/-	6.4E-01	1.99E+01	+/-	8.8E-01	1.25E+02	+/-	8.8E-01
-5.5	92-1277N	4.32E+00	+/-	4.6E-01	1.18E+01	+/-	6.9E-01	8.12E+01	+/-	7.0E-01
-6.5	92-1277O	1.54E+00	+/-	2.4E-01	4.36E+00	+/-	3.4E-01	4.12E+01	+/-	3.5E-01

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

AL File: 14667
References: Lab Book#56 pages 59-60; Lab Book #59 Pages 3-7
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: Mark Kawchak

REPORT

Westinghouse Advanced Programs
Analytical Laboratory - Waltz Mill Site

Request# 14667

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

Received: 5/5/92
Reported: 8/20/92

[RESULTS OF ANALYSIS]

POINT BEACH UNIT ~~2~~ 1 CYCLE ~~17~~ 19 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: S-6 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 @ ~~7~~/6/17/92 --->]								
+7.5	92-1278A	7.21E-01	+/-	7.1E-02	2.09E+00	+/-	1.2E-01	1.76E+01	+/-	1.2E-01
+6.5	92-1278B	2.33E+00	+/-	1.2E-01	6.19E+00	+/-	1.8E-01	2.98E+01	+/-	1.5E-01
+5.5	92-1278C	5.99E+00	+/-	5.4E-01	1.46E+01	+/-	7.9E-01	8.34E+01	+/-	7.1E-01
+4.5	92-1278D	8.81E+00	+/-	5.8E-01	2.15E+01	+/-	9.2E-01	1.15E+02	+/-	8.6E-01
+3.5	92-1278E	8.75E+00	+/-	6.2E-01	2.46E+01	+/-	1.1E+00	1.35E+02	+/-	9.2E-01
+2.5	92-1278F	9.54E+00	+/-	7.4E-01	2.55E+01	+/-	1.1E+00	1.45E+02	+/-	9.7E-01
+1.5	92-1278G	1.01E+01	+/-	7.4E-01	2.60E+01	+/-	1.1E+00	1.46E+02	+/-	9.8E-01
+0.5	92-1278H	1.04E+01	+/-	7.4E-01	2.54E+01	+/-	1.1E+00	1.45E+02	+/-	9.5E-01
-0.5	92-1278I	9.72E+00	+/-	7.1E-01	2.56E+01	+/-	1.1E+00	1.44E+02	+/-	9.7E-01
-1.5	92-1278J	1.03E+01	+/-	7.8E-01	2.50E+01	+/-	1.1E+00	1.39E+02	+/-	9.5E-01
-2.5	92-1278K	8.79E+00	+/-	7.2E-01	2.39E+01	+/-	1.1E+00	1.28E+02	+/-	9.0E-01
-3.5	92-1278L	8.59E+00	+/-	6.9E-01	2.22E+01	+/-	1.0E+00	1.17E+02	+/-	8.7E-01
-4.5	92-1278M	7.41E+00	+/-	4.0E-01	1.93E+01	+/-	6.6E-01	8.33E+01	+/-	5.1E-01
-5.5	92-1278N	4.25E+00	+/-	3.0E-01	1.11E+01	+/-	4.9E-01	4.64E+01	+/-	3.7E-01
-6.5	92-1278O	1.76E+00	+/-	2.5E-01	4.54E+00	+/-	3.2E-01	3.25E+01	+/-	3.1E-01

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

AL File: 14667
References: Lab Book#56 pages 59-60; Lab Book #59 Pages 3-7
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: Mark Kavchak

REPORT

Westinghouse Advanced Programs
Analytical Laboratory - Waltz Mill Site

Request# 14667

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

Received: 5/5/92
Reported: 8/20/92

[RESULTS OF ANALYSIS]

POINT BEACH UNIT ~~2~~ 1 CYCLE ~~17~~ 19 REACTOR CAVITY DOSIMETRY

Bead Chain Tag ID: S-6 45 deg.

Feet from Midplane	Lab Sample#	[<---- dps/mg of chain @ ~~7~~ 6/17/92 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	92-1279A	7.88E-01 +/-	5.2E-01	2.14E+00 +/-	8.1E-02	1.66E+01 +/-	4.8E-02
+6.5	92-1279B	2.36E+00 +/-	1.4E-01	6.12E+00 +/-	2.3E-01	2.97E+01 +/-	2.1E-01
+5.5	92-1279C	5.49E+00 +/-	3.4E-01	1.41E+01 +/-	4.8E-01	5.56E+01 +/-	4.1E-01
+4.5	92-1279D	7.89E+00 +/-	2.4E-01	2.06E+01 +/-	3.4E-01	7.20E+01 +/-	3.3E-01
+3.5	92-1279E	8.64E+00 +/-	4.0E-01	2.25E+01 +/-	7.0E-01	8.71E+01 +/-	5.2E-01
+2.5	92-1279F	9.77E+00 +/-	4.9E-01	2.45E+01 +/-	7.2E-01	9.52E+01 +/-	5.4E-01
+1.5	92-1279G	9.81E+00 +/-	4.7E-01	2.46E+01 +/-	7.7E-01	9.62E+01 +/-	5.5E-01
+0.5	92-1279H	9.67E+00 +/-	4.7E-01	2.45E+01 +/-	7.8E-01	9.66E+01 +/-	5.5E-01
-0.5	92-1279I	9.83E+00 +/-	5.4E-01	2.40E+01 +/-	7.3E-01	9.75E+01 +/-	5.6E-01
-1.5	92-1279J	9.46E+00 +/-	4.5E-01	2.31E+01 +/-	7.3E-01	9.49E+01 +/-	5.4E-01
-2.5	92-1279K	9.40E+00 +/-	4.9E-01	2.34E+01 +/-	7.1E-01	9.01E+01 +/-	5.3E-01
-3.5	92-1279L	7.88E+00 +/-	2.5E-01	2.10E+01 +/-	3.8E-01	7.67E+01 +/-	3.5E-01
-4.5	92-1279M	6.43E+00 +/-	2.1E-01	1.71E+01 +/-	3.2E-01	5.93E+01 +/-	3.0E+03
-5.5	92-1279N	3.79E+00 +/-	1.8E-01	1.03E+01 +/-	2.7E-01	4.19E+01 +/-	2.5E-01
-6.5	92-1279O	1.36E+00 +/-	1.4E-01	3.39E+00 +/-	2.1E-01	3.39E+01 +/-	2.3E-01

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

AL File: 14667
References: Lab Book#56 pages 59-60; Lab Book #59 Pages 3-7
Procedures: A-524.
Analyst: WTF, TRK, MRK, FRC

Approved: Mark Kanchock