WESTINGHOUSE PROPRIETARY CLASS 3

CUSTOMER DESIGNATED DISTRIBUTION

WCAP-10962

ZION UNITS 1 AND 2 REACTOR VESSEL FLUENCE AND RT_{PTS} EVALUATIONS

E. L. Furchi
S. L. Anderson
K. R. Balkey
V. A. Perone
M. A. Weaver

Work Performed for Commonwealth Edison Company

December 1985

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

APPROVED: F.L. Lau
F. L. Lau, Manager
Radiation and Systems
Analysis

APPROVED: C.W. Hirst
C. W. Hirst, Manager
Reactor Coolant System
Components Licensing

WESTINGHOUSE ELECTRIC CORPORATION
NUCLEAR ENERGY SYSTEMS
P. O. BOX 355
PITTSBURGH, PENNSYLVANIA 15230

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES (Continued)

<u>LIST OF FIGURES</u>

SECTION I

INTRODUCTION

The purpose of this report is to determine the reference temperature for pressurized thermal shock (RT_{PTS}) values for the Zion Units 1 and 2 reactor vessels to address the Pressurized Thermal Shock (PTS) Rule. Section I discusses the Rule and provides the methodology for calculating RT_{PTS}. Section II presents the results of the neutron exposure evaluation assessing the effects that past and present core management strategies have had on neutron fluence levels in the reactor vessel. Section III provides the reactor vessel beltline region material properties for both units. Section IV provides the RT_{PTS} calculations for the present, projected end-of-license, and end-of-life fluence values.

I.1 THE PRESSURIZED THERMAL SHOCK RULE

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

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

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

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

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

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

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

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

* Requires NRC approval for operation beyond the Screening Criterion.

For applicants of operating licenses, values of the projected RT_{PTS} are to be provided in the Final Safety Analysis Report. This requirement is added as part of 10CFR Part 50.34.

In the Rule, the NRC provides guidance regarding the calculation of the toughness state of the reactor vessel materials - the "reference temperature for nil ductility transition" (RT_{NDT}). For purposes of the Rule, RT_{NDT} is now defined as "the reference temperature for pressurized thermal shock" (RT_{PTS}) and calculated as prescribed by 10 CFR 50.61(b) of the Rule. Each USNRC licensed PWR must submit a projection of RT_{PTS} values from the time of the submittal to the license expiration date. This assessment must be submitted within 6 months after the effective date of the Rule, on January 23,

1986, with updates whenever changes occur affecting projected values. The calculation must be made for each weld and plate, or forging, in the reactor vessel beltline. The purpose of this report is to provide the RT_{PTS} values for Zion Units 1 and 2.

I.2 THE CALCULATION OF RT_{PTS}

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

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

Equation 1:

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

Equation 2:

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

where

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

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

generic mean value of I was used. In Equation 2, M=0°F if a measured value of I was used, and M=34°F if the generic mean value of I was used.

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

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

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

SECTION II

NEUTRON EXPOSURE EVALUATION

This section presents the results from the application of the Westinghouse derived adjoint flux program to the Zion Units 1 and 2 reactor vessels for Commonwealth Edison Company. The use of adjoint importance functions provides a cost effective tool to assess the effects that past and present core management strategies have had an neutron fluence levels in the reactor vessel. The use of adjoint importance functions provides a cost effective tool to assess the effects that past and present core management strategies have had on neutron fluence levels in the reactor vessel. Both of the Zion plants have recently operated using low leakage core management schemes.

II.1 METHOD OF ANALYSIS

A plan view of the Zion Units 1 and 2 reactor geometry at the core midplane is shown in Figure II.1-1. Since the reactor exhibits 1/8th core symmetry only a 0°-45° sector is depicted. Eight irradiation capsules attached to the thermal shield are included in the design to constitute the reactor vessel surveillance program. Two capsules are located symmetrically in each quadrant at azimuthal positions of 4° and 40° from the reactor core cardinal axes as shown in Figure II.1-1.

In performing the fast neutron exposure evaluations for the reactor geometry shown in Figure II.1-1, two sets of transport calculations were carried out. The first, a single computation in the conventional forward mode, was utilized to provide baseline data derived from a design basis core power distribution against which cycle by cycle plant specific calculations can be compared. The second set of calculations consisted of series of adjoint analyses relating the response of interest (neutron flux > 1.0 MeV) at several selected locations within the reactor geometry to the power distributions in the reactor core. These adjoint importance functions, when combined with cycle specific core power distributions, yield the plant specific exposure data for each operating fuel cycle.

The forward transport calculation was carried out in R,Θ geometry using the DOT discrete ordinates code [2] and the SAILOR cross-section library [3]. The SAILOR library is a 47 group, ENDF-BIV based data set produced specifically for light water reactor applications. Anisotropic scattering is treated with a P_3 expansion of the cross-sections.

The design basis core power distribution utilized in the forward analysis was derived from statistical studies of long-term operation of Westinghouse 4-loop plants. Inherent in the development of this design basis core power distribution is the use of an out-in fuel management strategy; i.e., fresh fuel on the core periphery. Furthermore, for the peripheral fuel assemblies, a 2σ uncertainty derived from the statistical evaluation of plant to plant and cycle to cycle variations in peripheral power was used. Since it is unlikely that a single reactor would have a power distribution at the nominal $+2\sigma$ level for a large number of fuel cycles, the use of this design basis distribution is expected to yield somewhat conservative results. This is especially true in cases where low leakage fuel management has been employed.

The adjoint analyses were also carried out using the P_3 cross-section approximation from the SAILOR library. Adjoint source locations were chosen at the center of each of the surveillance capsules as well as at positions along the inner diameter of the pressure vessel. Again, these calculations were run in R,Θ geometry to provide power distribution importance functions for the exposure parameters of interest (neutron flux > 1.0 MeV). Having the adjoint importance functions and appropriate core power distributions, the response of interest is calculated as:

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

where:

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

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

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

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

Core power distributions for use in the plant specific fluence evaluations for Zion Units 1 and 2 were taken from the design of each operating cycle for the two reactors. The specific power distribution data used in the analysis is provided in Appendix A of this report. The data listed in Appendix A represents cycle averaged relative assembly powers. Therefore, the adjoint results were in terms of fuel cycle averaged neutron flux which when multiplied by the fuel cycle length yields the incremental fast neutron fluence.

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

II.2 FAST NEUTRON FLUENCE RESULTS

Calculated fast neutron ($E > 1.0$ MeV) exposure results for Zion Units 1 and 2 are presented in Tables II.2-1 through II.2-12 and in Figures II.2-1 through II.2-6. Data is presented at several azimuthal locations on the inner radius of the pressure vessel as well as at the center of each surveillance capsule.

In Tables II.2-1 through II.2-4 plant specific maximum neutron flux and fluence levels at 0°, 15°, 30°, and 45° on the pressure vessel inner radius are listed for the first 7 completed fuel cycles of Zion Unit 1. Also presented are the design basis fluence levels predicted using the generic 4-loop core power distribution at the nominal + 2σ level. Similar data for the center of surveillance capsules located at 4° and 40° are given in Tables II.2-5 and II.2-6, respectively.

In addition to the calculated data given for the surveillance capsule locations, measured fluence data from previously withdrawn surveillance capsules are also presented for comparison with analytical results. In the case of Unit 1, capsules were removed from the 40° location at the end of cycles 1, 4 and 6.

Cycle-specific and design basis fast neutron flux and fluence data at the inner radius of the pressure vessel are given in Tables II.2-7 through II.2-10 for the first 7 completed fuel cycles of Zion Unit 2. As in the case of Unit 1, data are presented for the 0°, 15°, 30°, and 45° azimuthal angles. Evaluations of cycle-specific and design basis fluence levels at the two surveillance capsule locations are given in Tables II.2-11 and II.2-12.

For Unit 2, surveillance capsules were removed from the 40° position following cycles 1 and 4. Dosimetry evaluations from these two capsule withdrawals are also listed in Table II.2-12.

Several observations regarding the data presented in Tables II.2-1 through II.2-12 are worthy of note. These observations may be summarized as follows:

1. For both units, calculated cycle-specific fast neutron (E > 1.0 MeV) fluence levels at the surveillance capsule center are in excellent agreement with measured data. The maximum difference between the cycle-specific calculations and the measurements is less than 7%. Differences of this magnitude are well within the uncertainty of the experimental results.

2. For Unit 1, low leakage fuel management introduced following cycle 6 has reduced the peak flux on the pressure vessel by about 35%.

3. For Unit 2, low leakage fuel management introduced following cycle 5 has reduced the peak flux on the pressure vessel by about 25%. This reduction has been maintained over the last 2 operating cycles.

4. For both of the Zion reactors the maximum neutron flux incident on the pressure vessel (45° azimuthal position) during the fuel cycles using out-in fuel management (cycles 1 through 6 for Unit 1 and 1 through 5 for Unit 2) was, on the average, approximately 15% less than predictions based on the design basis core power distributions.

Graphical presentations of the plant specific fast neutron fluence at key locations on the pressure vessel are shown in Figures II.2-1 and II.2-2 as a function of full power operating time for Zion Units 1 and 2, respectively. For both Units 1 and 2, pressure vessel data is presented for the 45° location on the circumferential weld as well as for the 0° longitudinal welds (see Section III.1).

In regard to Figure II.2-1 and II.2-2, the solid portions of the fluence curves are based directly on the cycle specific evaluations presented in this report. The dashed portions of these curves, however, involve a projection into the future. Since both Zion Units are committed to a consistent form of low leakage fuel management, the average neutron flux at the key locations over the low leakage fuel cycles was used for all temporal projections. In particular, the neutron flux average over cycle 7 was used to project future fluence levels for Unit 1, while the neutron flux average over cycles 6 and 7 was employed for Unit 2.

The fluence projections in Figures II.2-1 and II.2-2 have been carried out to 32 effective full power years. However, since RT_{PTS} data corresponding to the license expiration date must be supplied to the NRC in response to the Pressurized Thermal Shock Rule (10CFR50.61(b)(1)), the fluences corresponding

to the license expiration date are indicated in Figures II.2-1 and II.2-2. An 80% capacity factor was assumed for operation beyond Cycle 7 (including the seventh refueling outage) for each Zion Unit.

It should be noted that implementation of a more severe low leakage pattern would act to reduce the projections of fluence at key locations. On the other hand, relaxation of the current low leakage patterns or a return to out-in fuel management would increase those projections. In any event the RT_{PTS} assessment must be updated per 10CFR50.61(b)(1) whenever, among other things, changes in core loadings significantly impact the fluence and RT_{PTS} projections.

In Figures II.2-3 and II.2-4, the azimuthal variation of maximum fast neutron ($E > 1.0$ MeV) fluence at the inner radius of the pressure vessel is presented as a function of azimuthal angle for Units 1 and 2, respectively. Data are presented for both current and projected end-of-life conditions. In Figure II.2-5, the relative radial variation of fast neutron flux and fluence within the pressure vessel wall is presented. Similar data showing the relative axial variation of fast neutron flux and fluence over the beltline region of the pressure vessel is shown in Figure II.2-6. A three-dimensional description of the fast neutron exposure of the pressure vessel wall can be constructed using the data given in Figure II.2-3 through II.2-6 along with the relation

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

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

$\phi\ (\theta)$ = Fast neutron fluence at azimuthal location θ on the pressure vessel inner radius from Figure II.2-3 or II.2-4

F (R) = Relative fast neutron flux at depth R into the pressure vessel from Figure II.2-5

G (Z) = Relative fast neutron flux at axial position Z from Figure II.2-6

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

TABLE II.2-1

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE

VESSEL INNER RADIUS - 0° AZIMUTHAL ANGLE[(a)]

ZION UNIT 1

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis[(b)]
1	1.2	7.46×10^9	2.73×10^{17}	3.37×10^{17}
2	2.1	8.46×10^9	5.13×10^{17}	5.99×10^{17}
3	2.7	8.63×10^9	6.99×10^{17}	7.98×10^{17}
4	3.5	8.10×10^9	8.98×10^{17}	1.02×10^{18}
5	4.3	8.82×10^9	1.11×10^{18}	1.24×10^{18}
6	5.0	8.52×10^9	1.31×10^{18}	1.46×10^{18}
7	5.9	8.66×10^9	1.54×10^{18}	1.71×10^{18}
Cy 8 → EOL[(c)]	25.8	8.66×10^9	6.98×10^{18}	7.50×10^{18}
EOL 32.0 EFPY	32.0	8.66×10^9	8.68×10^{18}	9.31×10^{18}

a) Applicable to the longitudinal welds at 0°, 90°, 180°, 270° in the peak axial flux.

b) Design basis fast neutron flux = 9.22×10^9 n/cm^2-sec at 3391 MW_{th}.

c) Current neutron fluences are defined as of the beginning of Cycle 8 (February 9, 1984).
The data pertaining to Cy 8 and beyond represent a projection to the license expiration date (December 26, 2008) using the Cycle 7 average flux and an 80% capacity factor.

TABLE II.2-2

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE

VESSEL INNER RADIUS - 15° AZIMUTHAL ANGLE

ZION UNIT 1

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis[a]
1	1.2	1.16×10^{10}	4.82×10^{17}	5.27×10^{17}
2	2.1	1.32×10^{10}	8.57×10^{17}	9.36×10^{17}
3	2.7	1.35×10^{10}	1.15×10^{18}	1.25×10^{18}
4	3.5	1.29×10^{10}	1.46×10^{18}	1.60×10^{18}
5	4.3	1.45×10^{10}	1.81×10^{18}	1.94×10^{18}
6	5.0	1.41×10^{10}	2.15×10^{18}	2.29×10^{18}
7	5.9	1.19×10^{10}	2.47×10^{18}	2.67×10^{18}
Cy 8 → EOL[b]	25.8	1.19×10^{10}	9.95×10^{18}	1.17×10^{19}
EOL 32.0 EFPY	32.0	1.19×10^{10}	1.23×10^{19}	1.45×10^{19}

a) Design basis fast neutron flux = 1.44×10^{10} n/cm^2-sec at 3391 MW_{th}.

b) Current neutron fluences are defined as of the beginning of Cycle 8 (February 9, 1984)
The data pertaining to Cy 8 and beyond represent a projection to the license expiration date (December 26, 2008) using the Cycle 7 average flux and an 80% capacity factor.

TABLE II.2-3

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE VESSEL INNER RADIUS - 30° AZIMUTHAL ANGLE ZION UNIT 1

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis[(a)]
1	1.2	1.46×10^{10}	5.34×10^{17}	6.55×10^{17}
2	2.1	1.60×10^{10}	9.89×10^{17}	1.16×10^{18}
3	2.7	1.69×10^{10}	1.35×10^{18}	1.55×10^{18}
4	3.5	1.68×10^{10}	1.77×10^{18}	1.99×10^{18}
5	4.3	1.81×10^{10}	2.19×10^{18}	2.41×10^{18}
6	5.0	1.79×10^{10}	2.62×10^{18}	2.84×10^{18}
7	5.9	1.14×10^{10}	2.93×10^{18}	3.32×10^{18}
Cy 8 → EOL[(b)]	25.8	1.14×10^{10}	1.01×10^{19}	1.46×10^{19}
EOL 32.0 EFPY	32.0	1.14×10^{10}	1.23×10^{19}	1.81×10^{19}

a) Design basis fast neutron flux = 1.79×10^{10} n/cm^2-sec at 3391 MW_{th}.

b) Current neutron fluences are defined as of the beginning of Cycle 8 (February 9, 1984).
The data pertaining to Cy 8 and beyond represent a projection to the license expiration date (December 26, 2008) using the Cycle 7 average flux and an 80% capacity factor.

TABLE II.2-4

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

ZION UNIT 1

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis[b]
1	1.2	2.18×10^{10}	7.98×10^{17}	1.01×10^{18}
2	2.1	2.35×10^{10}	1.47×10^{18}	1.80×10^{18}
3	2.7	2.64×10^{10}	2.03×10^{18}	2.40×10^{18}
4	3.5	2.63×10^{10}	2.68×10^{18}	3.08×10^{18}
5	4.3	2.68×10^{10}	3.32×10^{18}	3.73×10^{18}
6	5.0	2.81×10^{10}	3.99×10^{18}	4.40×10^{18}
7	5.9	1.63×10^{10}	4.43×10^{18}	5.14×10^{18}
Cy 8 → EOL[c]	25.8	1.63×10^{10}	1.47×10^{19}	2.25×10^{19}
EOL → 32.0 EFPY	32.0	1.63×10^{10}	1.79×10^{19}	2.80×10^{19}

a) Maximum fast neutron flux incident upon the intermediate and lower shell plates and the intermediate to lower shell circumferential weld.

b) Design basis fast neutron flux = 2.77×10^{10} n/cm^2-sec at 3391 MW_{th}.

c) Current neutron fluences are defined as of the beginning of Cycle 8 (February 9, 1984).
The data pertaining to Cy 8 and beyond represent a projection to the license expiration date (December 26, 2008) using the Cycle 7 average flux and an 80% capacity factor.

TABLE II.2-5

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE 4°
SURVEILLANCE CAPSULE CENTER - ZION UNIT 1

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis
1	1.2	2.28×10^{10}	8.35×10^{17}	1.03×10^{18}
2	2.1	2.59×10^{10}	1.57×10^{18}	1.83×10^{18}
3	2.7	2.64×10^{10}	2.14×10^{18}	2.43×10^{18}
4	3.5	2.48×10^{10}	2.75×10^{18}	3.13×10^{18}
5	4.3	2.70×10^{10}	3.40×10^{18}	3.80×10^{18}
6	5.0	2.61×10^{10}	4.01×10^{18}	4.48×10^{18}
7	5.9	2.65×10^{10}	4.71×10^{18}	5.23×10^{18}

Note: Design Basis $\phi = 2.82 \times 10^{10}$ n/cm^2-sec at 3391 MW_{th}.

TABLE II.2-6

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE 40°
SURVEILLANCE CAPSULE CENTER - ZION UNIT 1

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis	Capsule Data
1	1.2	6.90×10^{10}	2.53×10^{18}	3.21×10^{18}	2.73×10^{18}
2	2.1	7.44×10^{10}	4.65×10^{18}	5.70×10^{18}	
3	2.7	8.36×10^{10}	6.43×10^{18}	7.59×10^{18}	
4	3.5	8.33×10^{10}	8.49×10^{18}	9.74×10^{18}	8.64×10^{18}
5	4.3	8.49×10^{10}	1.05×10^{19}	1.18×10^{19}	
6	5.0	8.90×10^{10}	1.26×10^{19}	1.39×10^{19}	1.24×10^{19}
7	5.9	5.16×10^{10}	1.40×10^{19}	1.63×10^{19}	

Note: Design Basis $\phi = 8.77 \times 10^{10}$ n/cm^2-sec at 3391 MW_{th}.

TABLE II.2-7

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

ZION UNIT 2

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis[b]
1	1.2	7.33×10^9	2.84×10^{17}	3.58×10^{17}
2	2.0	8.99×10^9	5.00×10^{17}	5.79×10^{17}
3	2.8	7.88×10^9	6.94×10^{17}	8.06×10^{17}
4	3.4	6.65×10^9	8.28×10^{17}	9.92×10^{17}
5	4.3	8.18×10^9	1.05×10^{18}	1.25×10^{18}
6	5.0	8.19×10^9	1.24×10^{18}	1.46×10^{18}
7	5.7	7.49×10^9	1.41×10^{18}	1.67×10^{18}
Cy 8 → EOL[c]	25.3	7.85×10^9	6.26×10^{18}	7.36×10^{18}
EOL → 32.0 EFPY	32.0	7.85×10^9	7.92×10^{18}	9.31×10^{18}

a) Applicable to the longitudinal welds at 0°, 90°, 180°, 270° in the peak axial flux.

b) Design basis fast neutron flux = 9.22×10^9 n/cm^2-sec at 3391 MW_{th}.

c) Current neutron fluences defined as of the beginning of Cycle 8 (July 9, 1984).
The data pertaining to Cy 8 and beyond represent a projection to the license expiration date (December 26, 2008) using the average of the Cycle 6 and 7 fluxes and an 80% capacity factor.

TABLE II.2-8

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE

VESSEL INNER RADIUS - 15° AZIMUTHAL ANGLE

ZION UNIT 2

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis(a)
1	1.2	1.15×10^{10}	4.46×10^{17}	5.59×10^{17}
2	2.0	1.39×10^{10}	7.80×10^{17}	9.04×10^{17}
3	2.8	1.23×10^{10}	1.08×10^{18}	1.26×10^{18}
4	3.4	1.19×10^{10}	1.32×10^{18}	1.55×10^{18}
5	4.3	1.27×10^{10}	1.67×10^{18}	1.95×10^{18}
6	5.0	1.17×10^{10}	1.95×10^{18}	2.28×10^{18}
7	5.7	1.11×10^{10}	2.19×10^{18}	2.60×10^{18}
Cy 8 → EOL(b)	25.3	1.14×10^{10}	9.23×10^{18}	1.15×10^{19}
EOL 32.0 EFPY	32.0	1.14×10^{10}	1.16×10^{19}	1.45×10^{19}

a) Design basis fast neutron flux = 1.44×10^{10} n/cm^2-sec at 3391 MW_{th}.

b) Current neutron fluences defined as of the beginning of Cycle 8 (July 9, 1984).
The data pertaining to Cy 8 and beyond represent a projection to the license expiration date (December 26, 2008) using the average of the Cycle 6 and 7 fluxes and an 80% capacity factor.

TABLE II.2-9

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE PRESSURE
VESSEL INNER RADIUS - 30° AZIMUTHAL ANGLE
ZION UNIT 2

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis[a]
1	1.2	1.42×10^{10}	5.51×10^{17}	6.95×10^{17}
2	2.0	1.63×10^{10}	9.42×10^{17}	1.12×10^{18}
3	2.8	1.58×10^{10}	1.33×10^{18}	1.56×10^{18}
4	3.4	1.64×10^{10}	1.66×10^{18}	1.93×10^{18}
5	4.3	1.61×10^{10}	2.10×10^{18}	2.42×10^{18}
6	5.0	1.24×10^{10}	2.40×10^{18}	2.84×10^{18}
7	5.7	1.26×10^{10}	2.67×10^{18}	3.23×10^{18}
Cy 8 → EOL[b]	25.3	1.25×10^{10}	1.04×10^{19}	1.43×10^{19}
EOL 32.0 EFPY	32.0	1.25×10^{10}	1.30×10^{19}	1.81×10^{19}

a) Design basis fast neutron flux = 1.79×10^{10} n/cm^2-sec at 3391 MW_{th}.

b) Current neutron fluences defined as of the beginning of Cycle 8 (July 9, 1984).
The data pertaining to Cy 8 and beyond represent a projection to the license expiration date (December 26, 2008) using the average of the Cycle 6 and 7 fluxes and an 80% capacity factor.

TABLE II.2-10

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

ZION UNIT 2

	Elapsed Irradiation	Cycle Avg.	Beltline Region Cumulative Fluence (n/cm²)	
Cycle No.	Time (EFPY)	Flux (n/cm²-sec)	Cycle Specific	Design Basis[(b)]
1	1.2	2.09×10^{10}	8.11×10^{17}	1.07×10^{18}
2	2.0	2.37×10^{10}	1.38×10^{18}	1.74×10^{18}
3	2.8	2.50×10^{10}	1.99×10^{18}	2.42×10^{18}
4	3.4	2.70×10^{10}	2.54×10^{18}	2.98×10^{18}
5	4.3	2.64×10^{10}	3.27×10^{18}	3.74×10^{18}
6	5.0	1.72×10^{10}	3.67×10^{18}	4.39×10^{18}
7	5.7	1.82×10^{10}	4.07×10^{18}	5.00×10^{18}
Cy 8 → EOL[(c)]	25.3	1.76×10^{10}	1.49×10^{19}	2.21×10^{19}
EOL 32.0 EFPY	32.0	1.76×10^{10}	1.87×10^{19}	2.80×10^{19}

a) Maximum fast neutron flux incident upon the intermediate and lower shell plates and the intermediate to lower shell circumferential weld.

b) Design basis fast neutron flux = 2.77×10^{10} n/cm^2-sec at 3391 MW_{th}.

c) Current neutron fluences defined as of the beginning of Cycle 8 (July 9,1984).
The data pertaining to Cy 8 and beyond represent a projection to the license expiration date (December 26, 2008) using the average of the Cycle 6 and 7 fluxes and an 80% capacity factor.

TABLE II.2-11

FAST NEUTRON (E > 1.0 MeV) EXPC   AT THE 4°

SURVEILLANCE CAPSULE CENTER - ZION UNIT 2

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis
1	1.2	2.24×10^{10}	8.69×10^{17}	1.09×10^{18}
2	2.0	2.75×10^{10}	1.53×10^{18}	1.77×10^{18}
3	2.8	2.41×10^{10}	2.12×10^{18}	2.46×10^{18}
4	3.4	2.03×10^{10}	2.53×10^{18}	3.03×10^{18}
5	4.3	2.50×10^{10}	3.21×10^{18}	3.81×10^{18}
6	5.0	2.50×10^{10}	3.79×10^{18}	4.47×10^{18}
7	5.7	2.29×10^{10}	4.31×10^{18}	5.09×10^{18}

Note: Design Basis $\phi = 2.82 \times 10^{10}$ n/cm^2-sec at 3391 MW_{th}.

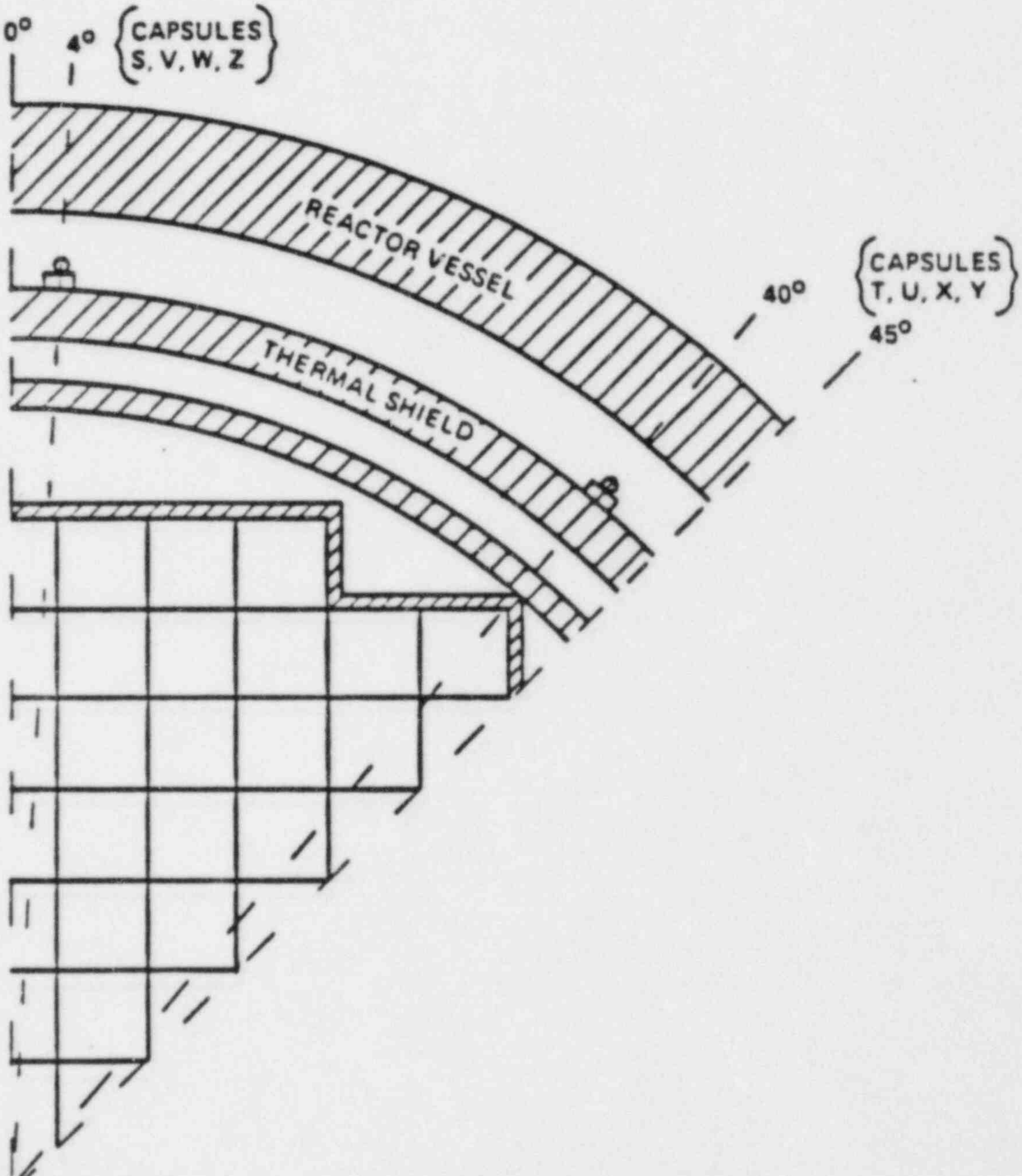
TABLE II.2-12

FAST NEUTRON (E > 1.0 MeV) EXPOSURE AT THE 40°
SURVEILLANCE CAPSULE CENTER - ZION UNIT 2

Cycle No.	Elapsed Irradiation Time (EFPY)	Cycle Avg. Flux (n/cm^2-sec)	Beltline Region Cumulative Fluence (n/cm^2) Cycle Specific	Design Basis	Capsule Data
1	1.2	6.62×10^{10}	2.57×10^{18}	3.40×10^{18}	2.70×10^{18}
2	2.0	7.50×10^{10}	4.37×10^{18}	5.51×10^{18}	
3	2.8	7.92×10^{10}	6.30×10^{18}	7.66×10^{18}	
4	3.4	8.55×10^{10}	8.04×10^{18}	9.44×10^{18}	8.49×10^{18}
5	4.3	8.36×10^{10}	1.04×10^{19}	1.18×10^{19}	
6	5.0	5.45×10^{10}	1.16×10^{19}	1.39×10^{19}	
7	5.7	5.76×10^{10}	1.29×10^{19}	1.58×10^{19}	

Note: Design Basis $\phi = 8.77 \times 10^{10}$ n/cm^2-sec at 3391 MW_{th}.

Figure II. 1-1
ZION REACTOR GEOMETRY



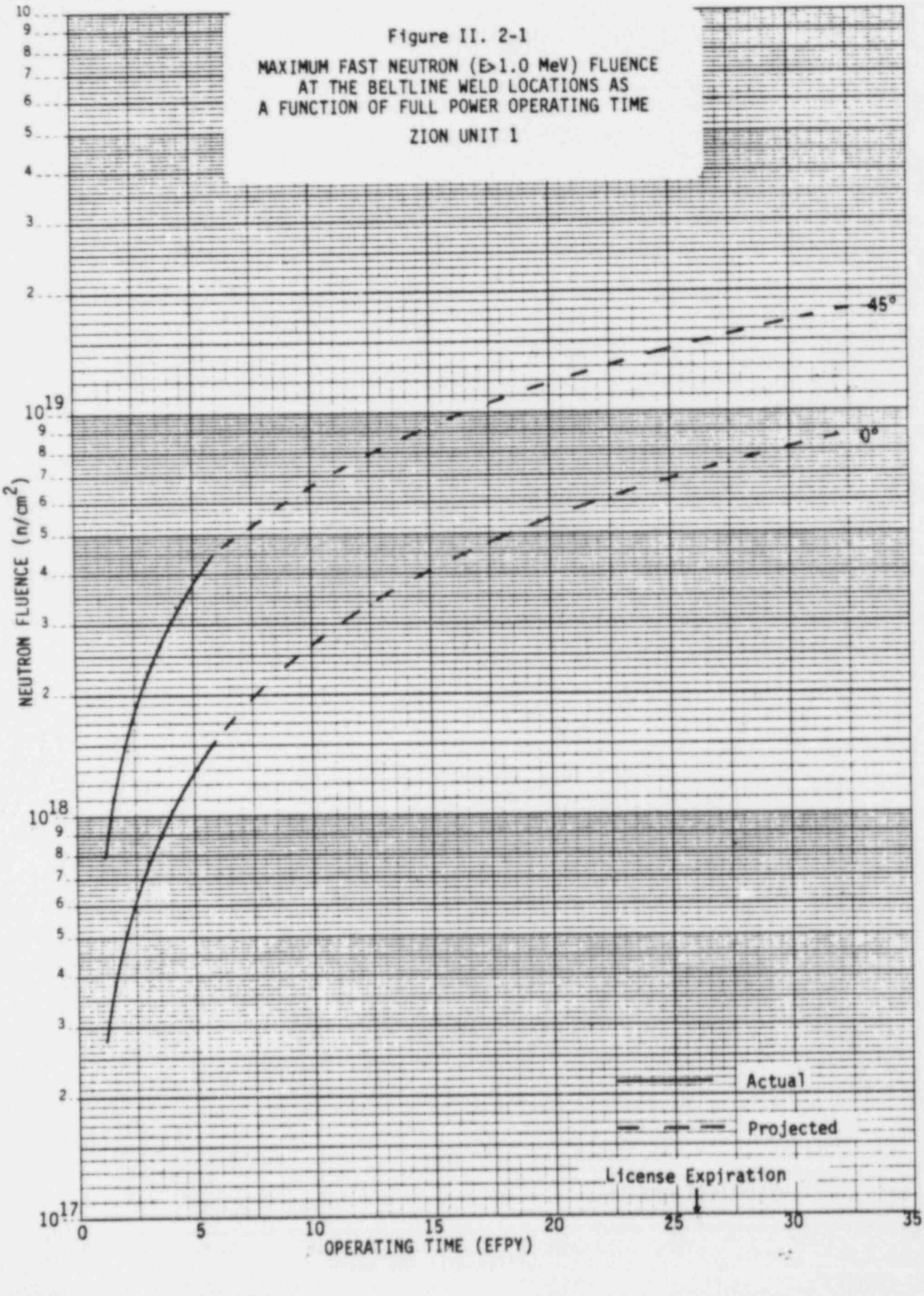


Figure II. 2-1

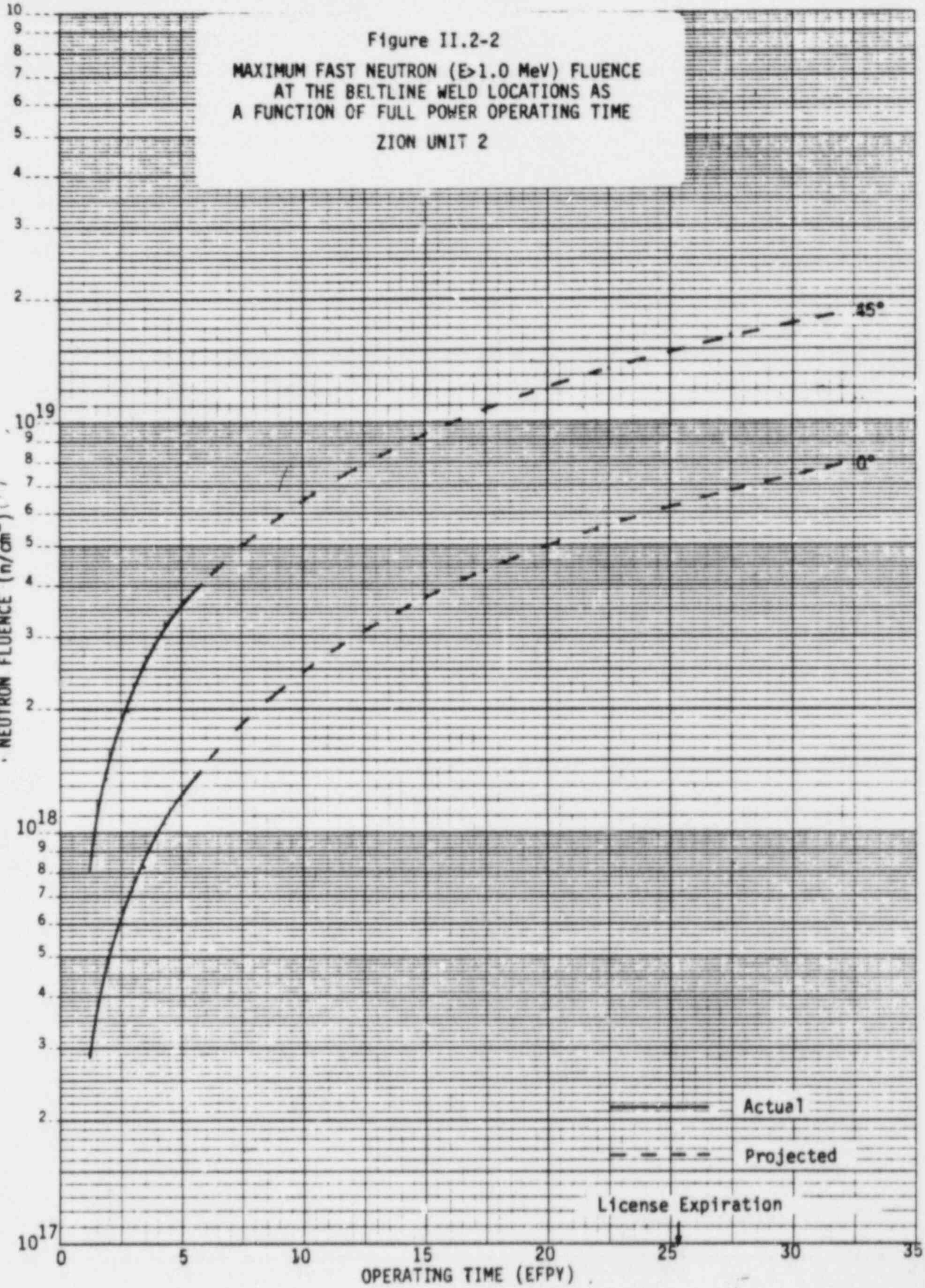
MAXIMUM FAST NEUTRON (E>1.0 MeV) FLUENCE AT THE BELTLINE WELD LOCATIONS AS A FUNCTION OF FULL POWER OPERATING TIME

ZION UNIT 1

Figure II.2-2

MAXIMUM FAST NEUTRON (E>1.0 MeV) FLUENCE AT THE BELTLINE WELD LOCATIONS AS A FUNCTION OF FULL POWER OPERATING TIME

ZION UNIT 2



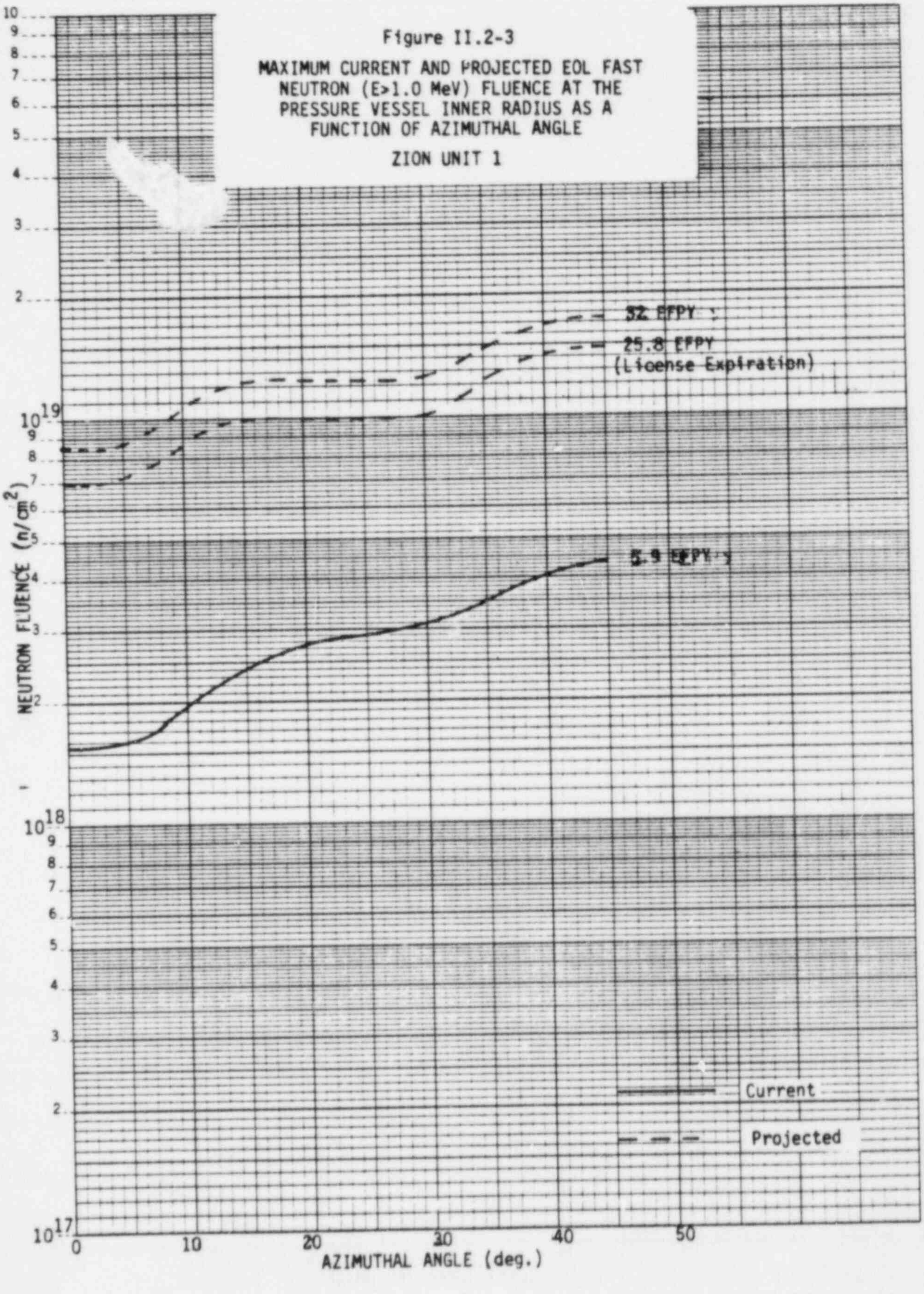


Figure II.2-3
MAXIMUM CURRENT AND PROJECTED EOL FAST NEUTRON (E>1.0 MeV) FLUENCE AT THE PRESSURE VESSEL INNER RADIUS AS A FUNCTION OF AZIMUTHAL ANGLE
ZION UNIT 1
32 EFPY
25.8 EFPY
(License Expiration)
5.9 EFPY
NEUTRON FLUENCE (n/cm²)
AZIMUTHAL ANGLE (deg.)
Current
Projected
0
10
20
30
40
50

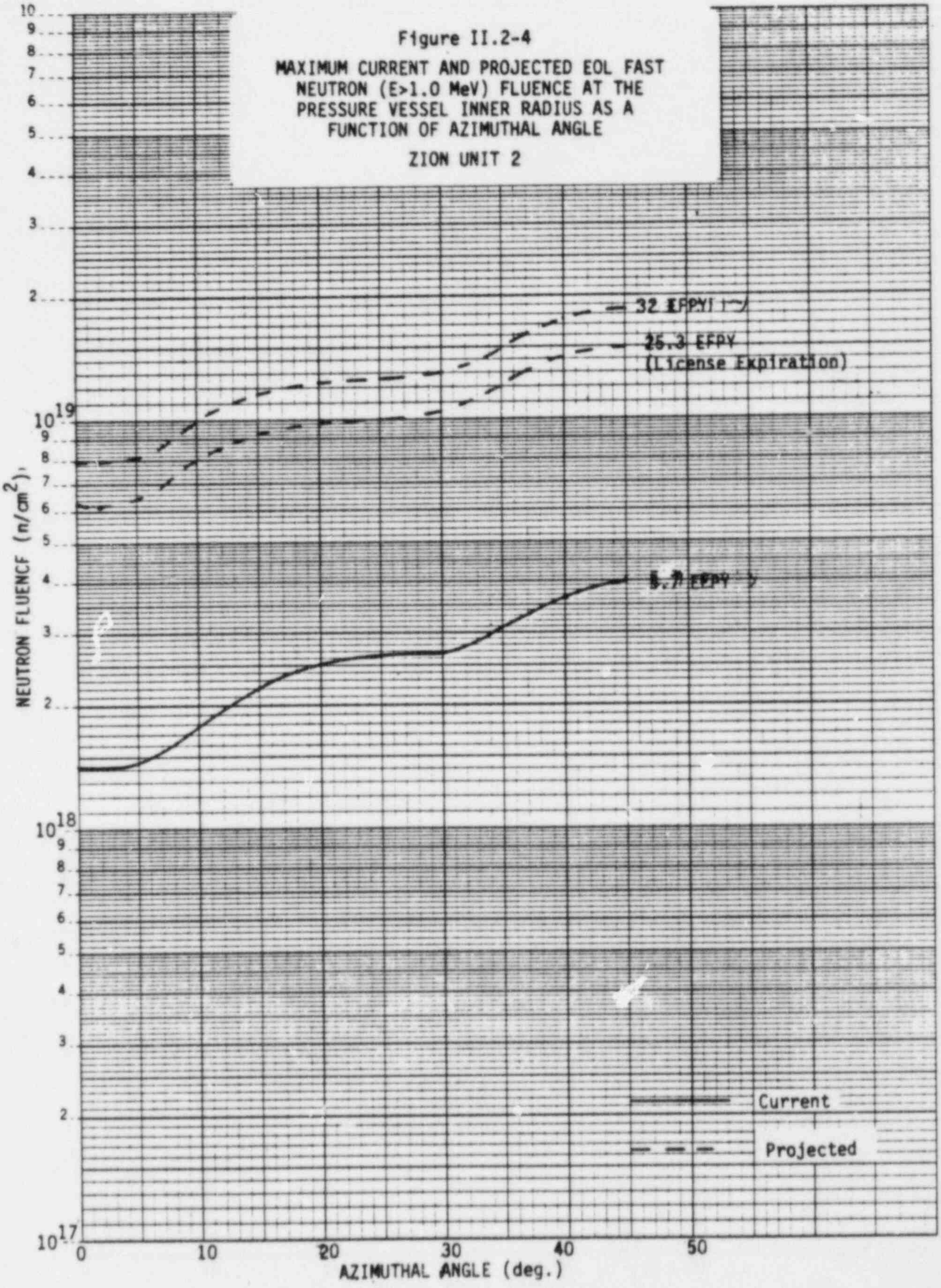


Figure II.2-4

MAXIMUM CURRENT AND PROJECTED EOL FAST NEUTRON (E>1.0 MeV) FLUENCE AT THE PRESSURE VESSEL INNER RADIUS AS A FUNCTION OF AZIMUTHAL ANGLE

ZION UNIT 2

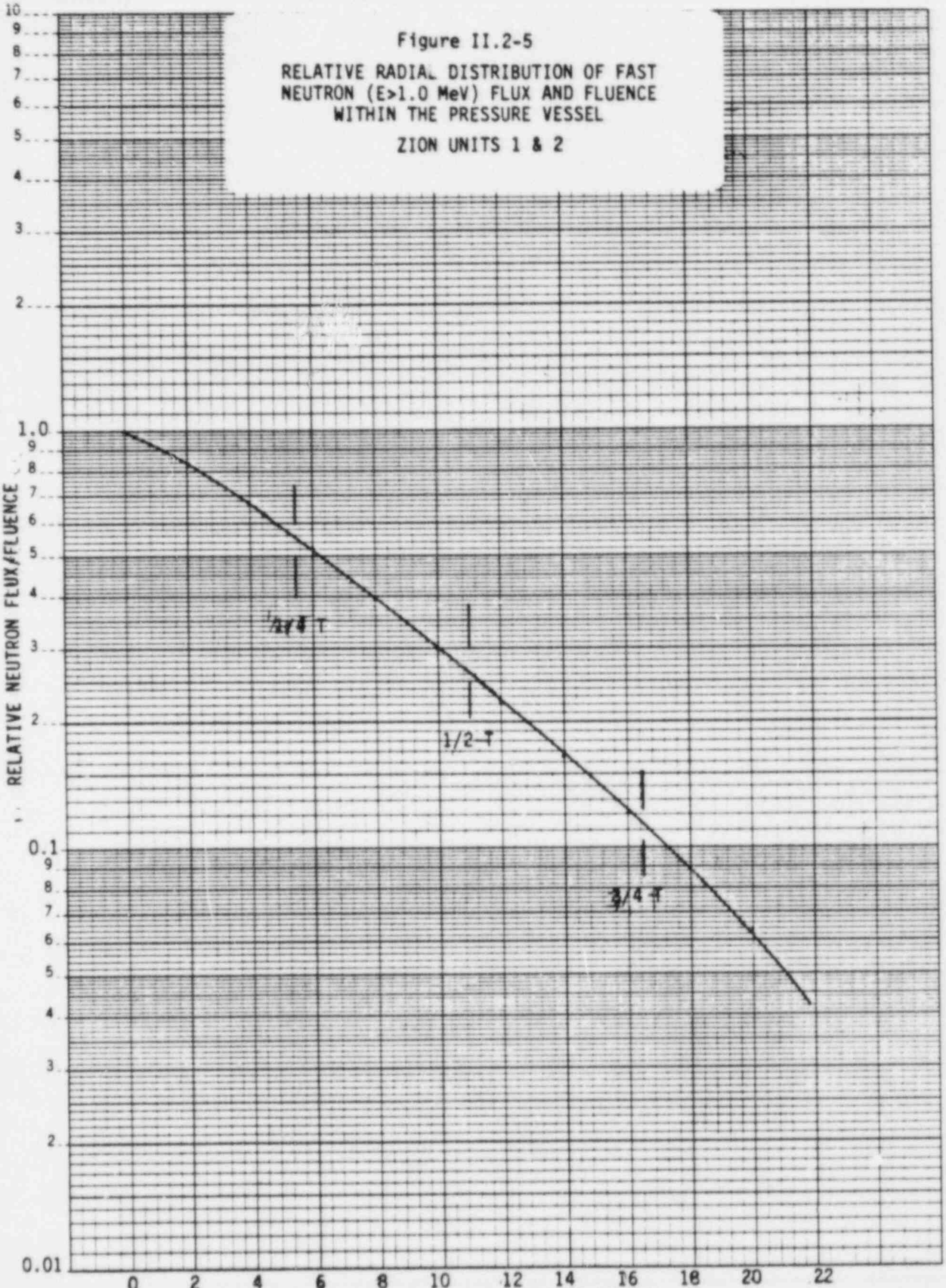


Figure II.2-5

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

ZION UNITS 1 & 2

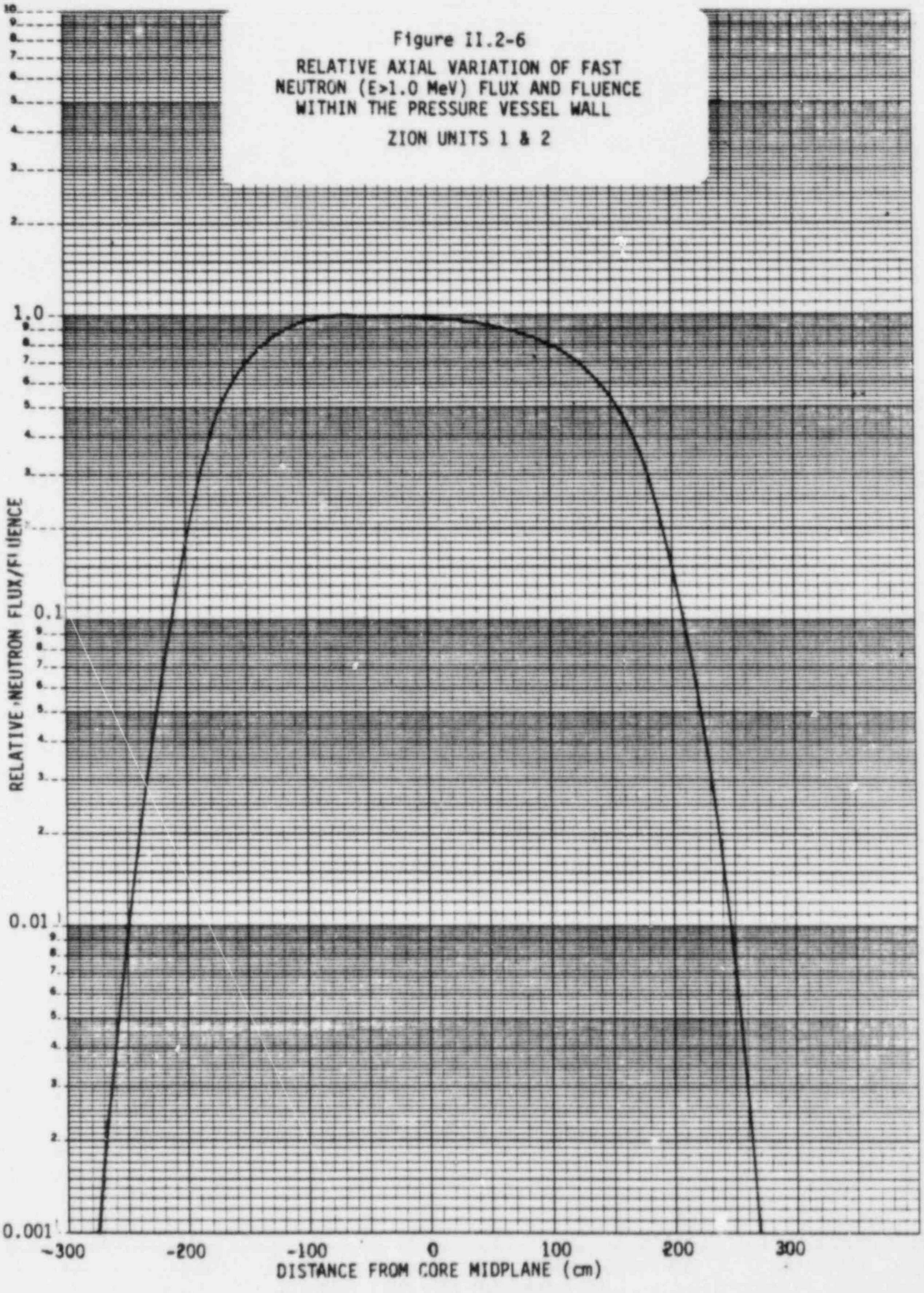


Figure II.2-6
RELATIVE AXIAL VARIATION OF FAST NEUTRON (E>1.0 MeV) FLUX AND FLUENCE WITHIN THE PRESSURE VESSEL WALL
ZION UNITS 1 & 2
RELATIVE NEUTRON FLUX/FLUENCE
1.0
0.1
0.01
0.001
-300
-200
-100
0
100
200
300
DISTANCE FROM CORE MIDPLANE (cm)

SECTION III

MATERIAL PROPERTIES

For the RT_{PTS} calculation, the best estimate copper and nickel chemical composition of the reactor vessel beltline material is necessary. The material properties for the Zion Units 1 and 2 beltline region will be presented in this section.

III.1 IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIALS

The beltline region is defined by the Rule [1] to be "the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron irradiation damage to be considered in the selection of the most limiting material with regard to radiation damage." Figures III.1-1 and III.1-2 identify and indicate the location of all beltline region materials for the Zion Units 1 and 2 reactor vessels.

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

Material property values for the shell plates, which have been docketed with the NRC in Reference 5, were derived from vessel fabrication test certificate results. The property data for the welds have also been docketed with the NRC in Reference 5, however, the weld properties cannot be used in the same direct manner as the properties for the plates.

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

To address the variation in chemistry, Babcock & Wilcox (B&W) performed a reactor vessel beltline weld chemistry study of eight B&W vessels, including Zion Units 1 and 2, and reported the results in BAW-1799 [6] for the Westinghouse Owners Group (WOG). The scope of the work included collecting

existing sources of chemistry data, performing extensive chemical analysis on the available archive reactor vessel weldments, and developing predictive methods with the aid of statistical analyses to establish the chemistry of the reactor vessel beltline weldments in question.

In addition to the B&W report BAW-1799, the WOG Reactor Vessel Beltline Region Weld Metal Data Base was used. The WOG data base, which was developed in 1984 and is continually being updated, contains information from weld qualification records, surveillance capsule reports, the B&W report BAW-1799, and the Materials Properties Council (MPC) and the NRC Mender MATSURV data bases.

For each of the welds in the Zion Units 1 and 2 beltline region, a material data search was performed using the WOG data base. Searches were performed for materials having the identical weld wire heat number as those in the Zion vessels, but any combination of wire and flux was allowed. For all of the data found for a particular wire, the copper, nickel, phosphorous and silicon values were averaged and the standard deviations were calculated. Although phosphorous and silicon are not needed for the PTS Rule, they are provided for the sake of completeness. The information obtained from the data base searches is found in Appendix B.

When the results of the material data base searches were evaluated, it was found that a large scatter existed in the measured as-deposited copper values for the data obtained for weld wire heat 72105, which is associated with the WF-70 weld seams, and weld wire 71249, which is associated with the Unit 2 girth weld. Preliminary RT_{PTS} calculations showed that the submerged arc welds fabricated with filler wire heat number 72105 were limiting in regards to PTS for both Zion Units 1 and 2 reactor vessels. Since the chemical composition of these welds significantly impacts the PTS concern for the Zion vessels, a statistical materials program, which is discussed in the next section, was developed to address the large scatter in copper content. The chemical composition values for heat 71249 have already been addressed via evaluations of reactor vessel materials data for Turkey Point Units 3 and 4 (see Reference [7]).

III.3 MATERIAL CHEMISTRY STUCY FOR WELD WIRE HEAT 72105

Filler wire heat number 72105 was used with Linde 80 flux lot number 8669 in making the beltline welds of both reactor vessels and with Linde 80 flux lot number 8773 in making the Zion reactor vessel surveillance capsules. The following is a summary of the results from the statistical materials program undertaken to address the large scatter found in the measured as-deposited copper values for filler wire heat number 72105.

III.3.1 STATISTICAL EVALUATION

III.3.1.1 Statistical Analysis Results

As part of the Westinghouse Statistical Materials Program, all available data on the chemical composition of heat 72105 was gathered. B&W previously recommended in report BAW-1799 that mean chemistry values of 0.35 wt% and 0.59% for copper and nickel content be respectively defined for weld wire heat number 72105. These values were also used to define RT_{NDT} values for the respective Zion 1 and 2 reactor vessel welds in the NRC Policy Issue on PTS SECY-82-465 [11]. However, 30 more data points than those used to substantiate the B&W recommended values were obtained via the WOG data base development. These additional chemical measurements, which were obtained from several sources, change the previous recommendations given in BAW-1799.

From a total population of 87 data points, which include 57 data points from B&W report BAW-1799 [6], the mean copper value obtained was 0.32% with a coefficient of variation (standard deviation divided by the mean) of 21%. (A high coefficient of variation indicates that there is a significant amount of scatter in the data). The mean copper value obtained from only the B&W archive data in BAW-1799 is 0.35% with a coefficient of variation of 17%.

The total population of 87 data points contains measurements for heat 72105 in combination with three different flux lot numbers. The data follows for the three different lots of flux.

TABLE III.3-1

SUMMARY OF RESULTS FROM WESTINGHOUSE STATISTICAL

MATERIALS STUDY FOR HEAT 72105*

FLUX LOT NUMBER	NUMBER OF DATA POINTS	MEAN Cu (WT%)	COEFFICIENT OF VARIATION
8688	2	0.255%	-
8669	17	0.405%	16%
8773	68	0.295%	14.5%

* All data is provided in Appendix B of this report.

The 68 data points for flux lot 8773 were obtained using four different chemical analyses techniques, four different laboratories, and both irradiated and unirradiated samples. Except for two data points, the data for flux lot 8669 were obtained by using emission spectrometry analyses on a weldment from a nozzle bore dropout taken from some vessel other than Zion Units 1 or 2. The other two points were obtained from a weld metal qualification test and a retest of this specimen.

It appears from the data found in BAW-1799 Appendix B [6] (see Figure III.3-1), that two very distinct populations of data exist for the same filler wire even though all of the measurements were done by B&W using emission spectrographic analysis of unirradiated specimens. Such a large discrepancy should not exist between different flux lots of welds made from the same heat of filler wire, since the flux does not contain an appreciable amount of copper. The principal source of copper is from the filler wire coating, and the flux neither extracts copper from the weld deposit nor significantly contributes copper to the weld deposit. In the next two sections, the impact of the flux is evaluated to determine whether the flux populations are truly different or if a bias exists in one of the populations.

III.3.1.2 Filler Wire Examination

To examine the potential for a biased set of data, the copper content in the filler wire, which is the principal source of copper in the weldment, was evaluated. In Appendix A of the B&W report, an examination of the filler wire heat number 72105 was performed. In this analysis, the filler wire was stripped of its copper coating. The copper quantity in both the coating and the bare metal was measured to determine the principle source of copper in the as-deposited weld metal, which is usually the surface coating. The quantity of copper content present on each wire sample was measured in compliance with ASTM D-168-77, Method D-Atomic Absorption Spectrophotometry. B&W looked at five different heats of wire, one of which was 72105. They used samples that were 8-10 inches long obtained from two or more spools of filler wire. As shown in Tables A-3, A-4, and A-5 in BAW-1799, B&W obtained a copper concentration of 0.230 wt% for the coating and a copper concentration of 0.075 wt% for the bare filler wire for heat 72105. Thus, the total copper concentration found for wire heat number 72105 in this analysis was 0.30 wt%. This number is in agreement with the mean copper value of 0.295% determined for all available measurements with flux lot number 8773 from the Westinghouse statistical materials study (see Table III.3-1). The 0.30 wt% value is also in agreement with chemical measurements of weld metal qualification material made with flux lots 8688 and 8669 (average measured values equal 0.255 wt% for flux 8688 and 0.305 wt% for flux 8669 - see Appendix B of this report).

III.3.1.3 Impact of Flux Lot for Other Filler Wires

In order to determine whether or not other welds made from the same filler wire with more than one flux result in distinct populations, the archive data in Appendix B of BAW-1799 were examined. Table III.3-2 summarizes the result of the calculated means, standard deviations, and coefficient of variations of the populations examined. The same phenomenon of two distinctly different means with small coefficients of variation, as found for the 72105 data, only exists for one other heat, number 71249. However, the NRC has already reviewed the data for heat 71249 and has accepted the mean value from all the data for the copper weight percentage [7].

From other data in the B&W report, flux 8669 did not result in the same high copper values as found for the nozzle bore dropout weldment when used with other filler wires (i.e., heat 72442), and discrepancies in the various flux populations do not exist (see Table III.3-2).

From the above observations, there are two possible explanations for the high copper reported in the nozzle dropout. Either the chemical analysis is in error or the particular spool(s) used for the nozzle dropout contained much higher copper than all the other 72105 spools.

III.3.2 CALCULATION OF ADJUSTED RT_{NDT}

Even if the data from the nozzle bore dropout with flux lot 8669 is biased, it may be difficult to defend eliminating this data from the total population because this lot of flux was used in making welds in the two Zion vessels.

The methodology for calculating the RT_{PTS} values prescribed in the PTS Rule is just one of many ways for calculating RT_{NDT}. Since the methodology was issued for calculating RT_{PTS} values, another trend curve, Regulatory Guide 1.99 Revision 2, is under development [8]. This trend curve represents the latest methodology for calculating RT_{NDT}. Therefore, in order to put the difference in copper values for filler wire 72105 into perspective, the difference in terms of the adjusted RT_{NDT}, as prescribed in the proposed Regulatory Guide 1.99 Rev. 2 (see Table III.3-3) was calculated.

For the limiting welds of the Zion reactor vessels, using 0.56% Ni and copper mean values of 0.32% and 0.40% (the maximum copper value provided in Table III.3-3), and predicted fluence values representative of 32 effective full power years, the difference in the mean shift of RT_{NDT} is 34°F. This is well within the margin added to the mean shift of RT_{NDT}, which is 56°F per the provision of the proposed Regulatory Guide 1.99 Revision 2. Thus, averaging all available data proves to be a viable approach for weld heat number 72105 for the Zion Units 1 and 2 reactor vessels.

III.3.3 SUMMARY OF MATERIAL CHEMISTRY STUDY FOR WELD HEAT NUMBER 72105

Based on the above logic, the copper content for weld heat number 72105 is 0.32 wt%, a number that is the average of all data with a standard deviation of 0.067 wt%. The mean nickel content is 0.56 wt% with a standard deviation of 0.06 wt%. These numbers are used to calculate the RT_{PTS} values found in Section IV.

III.4 SUMMARY OF PLANT-SPECIFIC MATERIAL PROPERTIES

A summary of the pertinent chemical and mechanical properties of the beltline region plate and weld materials of the Zion Units 1 and 2 reactor vessels are respectively given in Tables III.4-1 and III.4-2 along with the references for this information. Although phosphorus is no longer used in the calculation of RT_{NDT} with respect to the PTS rule [1], it is given for reference since it is currently used in the Regulatory Guide 1.99, Revision 1 trend curve [10].

The initial RT_{NDT} value of 0°F, which is shown for all of the Zion Units 1 and 2 reactor vessel beltline weldments, is the generic mean value defined in the PTS rule [1] for welds made with Linde 80 flux.

The data in Tables III.4-1 and III.4-2 are used to evaluate the RT_{PTS} values for the Zion Unit 1 and 2 reactor vessels.

TABLE III.3-2

SUMMARY OF ARCHIVE DATA FROM BAW 1799 APPENDIX B

Filler Wire	Flux Lot #	Number of Data Points	Mean Cu μ (wt%)	Standard Deviation σ (wt%)	Coefficient of Variation σ/μ
72105*	8669	15	0.42	0.048	11.4%
	8773	36	0.32	0.028	9.0%
299L44	8650	48	0.352	.024	6.9%
	8596	11	0.375	.010	2.8%
71249	8445	9	0.181	.027	14.7%
	8738	26	0.286	.020	7.1%
61782	8436	12	0.204	.045	21.9%
	8457	29	0.270	.043	16.0%
406L44	8688	21	0.318	.012	3.7%
	8773	8	0.275	.012	4.3%
72442*	8669	13	0.222	.066	29.6%
	8579	18	0.262	.028	10.7%

*One weld contained these two filler wires with flux lot number 8669.

TABLE III.3-3

CHEMISTRY FACTOR (CF) FOR WELDS PER PROPOSED REG. GUIDE 1.99, Rev. 2

$$\text{MEAN SHIFT } RT_{NDT} = [CF]f^{(0.28\ -\ .10\ \log f)}$$

Copper, Wt. %	Nickel, Wt. % 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	15[illegible]	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

TABLE III.4-1

ZION UNIT 1 REACTOR VESSEL BELTLINE REGION

MATERIAL PROPERTIES

	Cu (Wt.%)	Ni (Wt.%)	P (Wt.%)	I (°F)	Source
Intermediate Shell Plate 8-144-2:	0.12	0.49	0.010	10 [a]	Ref. [5]
Intermediate Shell Plate 8-144-1:	0.12	0.49	0.010	5	Ref. [5]
Lower Shell Plate 9-144-1:	0.13	0.48	0.013	-4	Ref. [5]
Lower Shell Plate 9-144-2:	0.15	0.50	0.010	20	Ref. [5]
Circumferential Weld - Intermed. to Lower Shell WF-70, Heat 72105, Linde 80 Flux 8669:	0.32	0.56	0.017	0 [b]	WOG Material Data Base, Statistical Materials Study Section III.3
Longitudinal Welds - Intermed. & Lower Shell WF-4/WF-8, Heat No. 8T1762, Linde 80 Flux 8597/8632:	0.29	0.55	0.013	0 [b]	Ref. [6]

Notes:

(a) The initial RT_{NDT} value for this plate is estimated according to Branch Position MTEB 5-2 [9]

(b) The initial RT_{NDT} values for the welds are the generic mean value defined by the PTS rule [1] for Linde 80 welds.

TABLE III.4-2

ZION UNIT 2 REACTOR VESSEL BELTLINE REGION
MATERIAL PROPERTIES

	Cu (Wt.%)	Ni (Wt.%)	P (Wt.%)	I (°F)	Source
Intermediate Shell Plate 8-152-1:	0.12	0.51	0.010	22[(a)]	Ref. [5]
Intermediate Shell Plate 8-152-2:	0.12	0.53	0.010	22	Ref. [5]
Lower Shell Plate 9-152-1:	0.12	0.54	0.010	10[(a)]	Ref. [5]
Lower Shell Plate 9-152-2:	0.14	0.52	0.008	2[(a)]	Ref. [5]
Circumferential Weld - Intermed. to Lower Shell SA-1769, Heat 71249, Linde 80 Flux 8738:	0.26	0.60	0.019	0[(b)]	WOG Material Data Base, Ref. [6] and Ref. [7][(c)]
Longitudinal Welds - Intermed. Shell WF-29, Heat 72102, Linde 80 Flux 8650:	0.23	0.63	.019	0[(b)]	Ref. [6]
Longitudinal Welds - Lower shell WF-70, Heat 72105, Linde 80 Flux 8669:	0.32	0.56	0.017	0[(b)]	WOG Material Data Base, Statistical Materials Study - Section III.3

Notes:

(a) The initial RT_{NDT} value for these plates are estimated according to Branch Position MTEB 5-2 [9]

(b) The initial RT_{NDT} values for the welds are the generic mean value defined by the PTS rule [1] for Linde 80 welds.

(c) Agreement exists between References [6] and [7] and the WOG Material Data Base for heat 71249.

FIGURE III.1-1

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL
FOR THE ZION UNIT NO. 1 REACTOR VESSEL

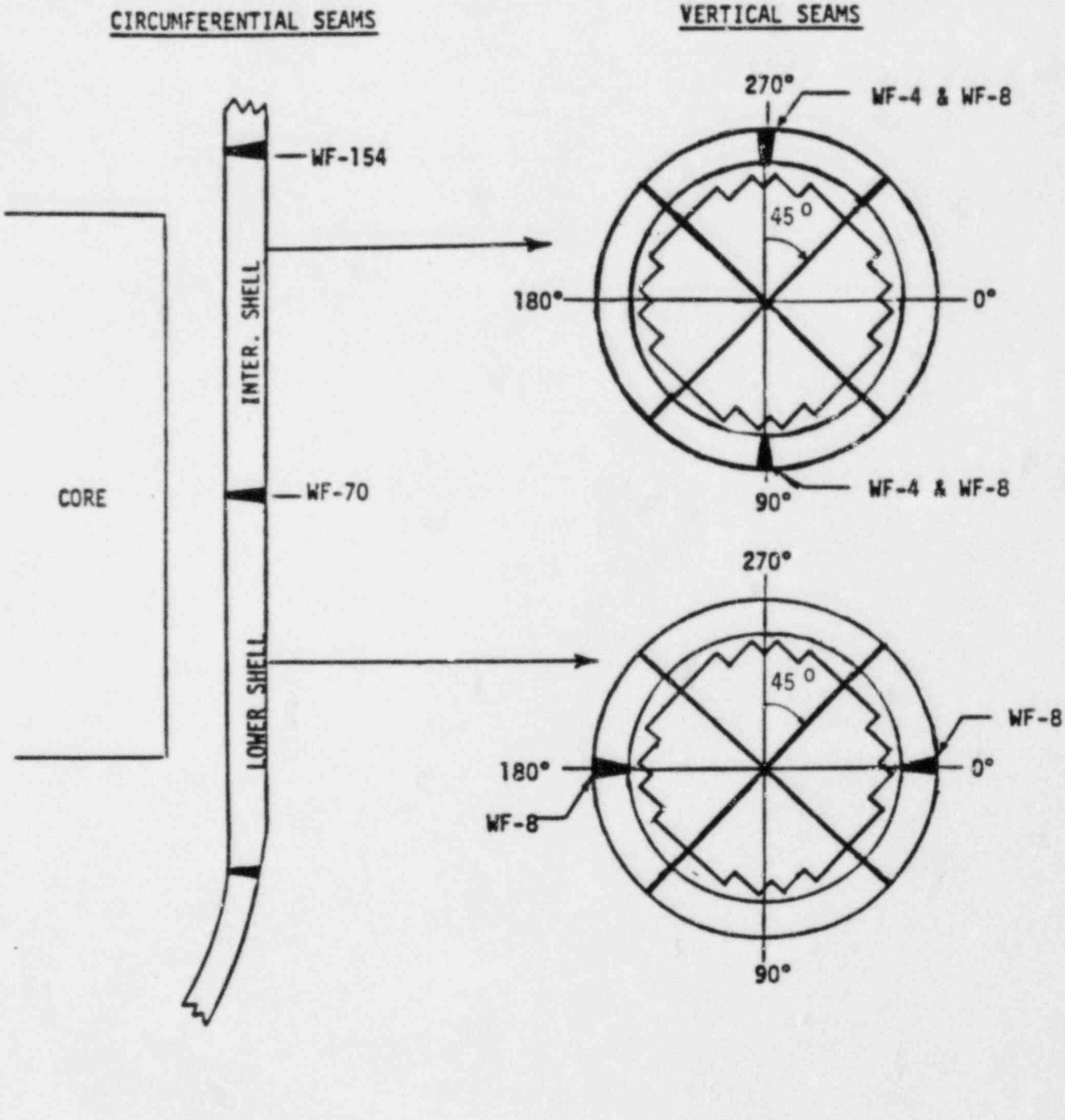


FIGURE III.1-2

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL
FOR THE ZION UNIT NO. 2 REACTOR VESSEL

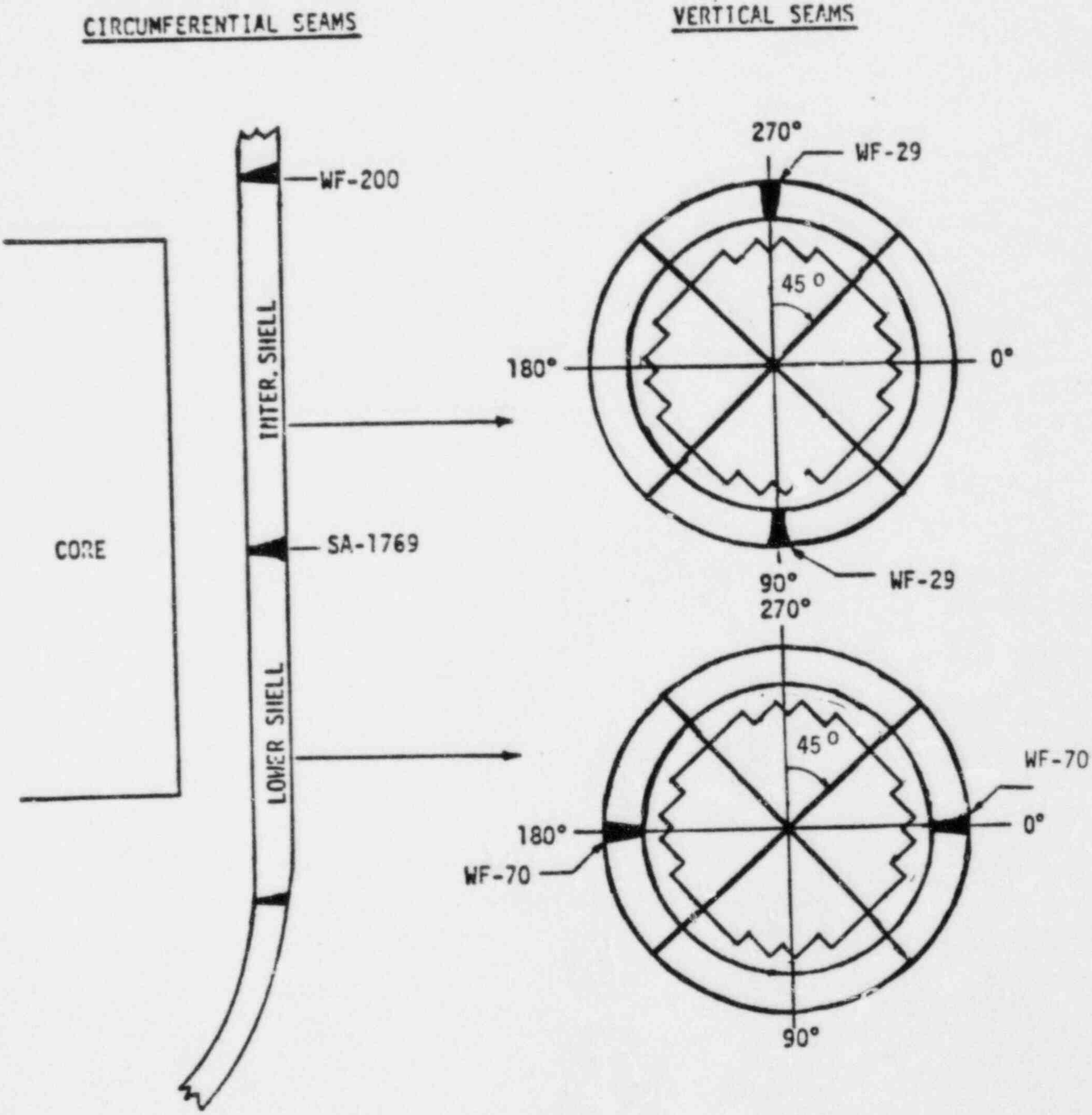
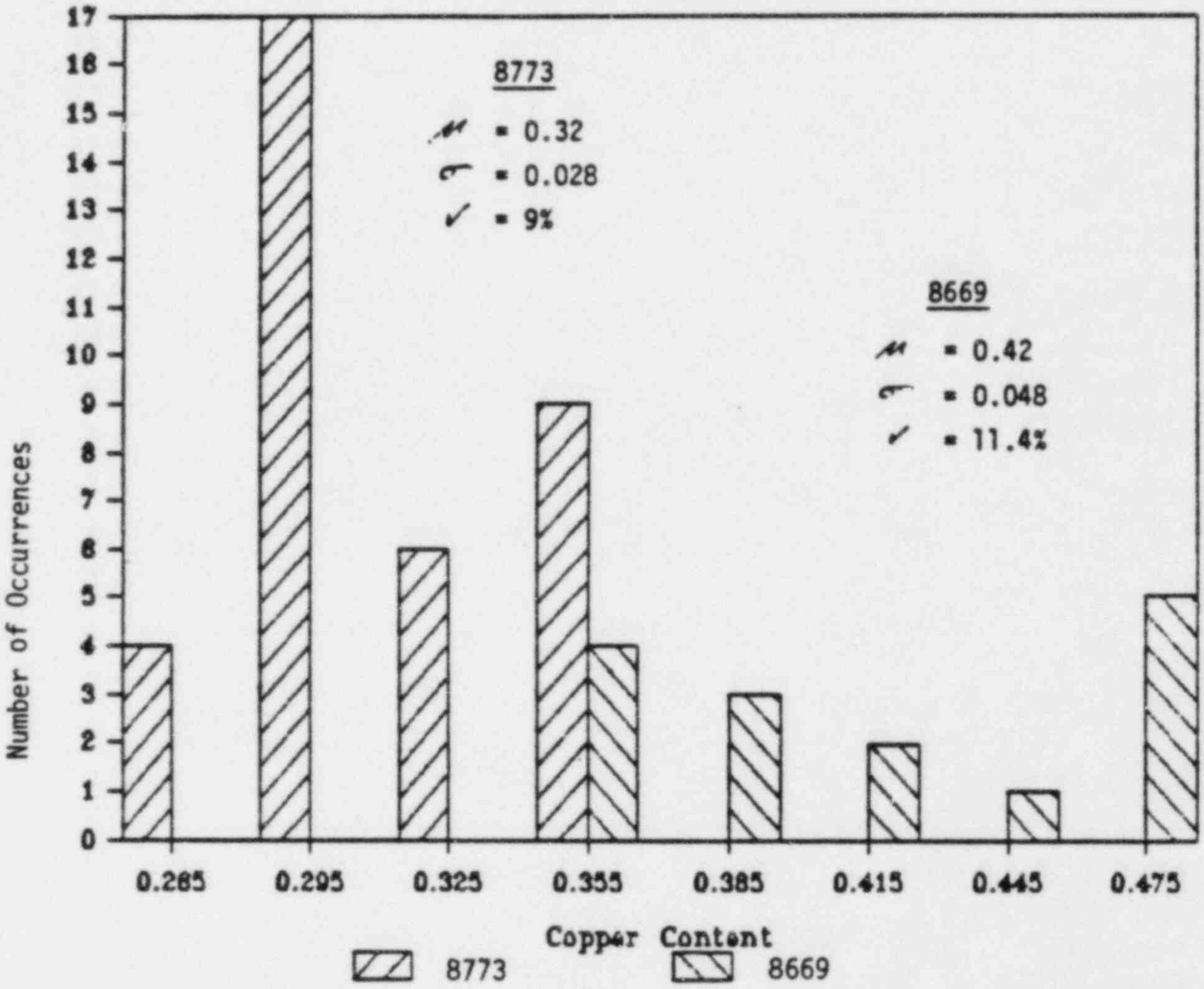


FIGURE III.3-1 B&W ARCHIVE DATA*

FLUX 8773 AND 8669 COMPARISON

Copper Histogram



* Data is found in BAW 1799 [6] Appendix B

SECTION IV

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

Using the methodology prescribed in Section I.2, the results of the fast neutron exposure provided in Section II, and the material properties discussed in Section III, the RT_{PTS} values for Zion Units 1 and 2 can now be determined.

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

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Zion Units 1 and 2 reactor vessels as a function of several fluence values and pertinent vessel lifetimes. The tabulated results from the total evaluation are presented in Appendix C for all beltline region materials for both units.

Figures IV.1-1 and IV.1-2 present the RT_{PTS} values for the limiting longitudinal weld, circumferential weld and shell plate of the Zion Units 1 and 2 vessels in terms of RT_{PTS} versus fluence* curves. The curves in these figures can be used:

- o to provide guidance to evaluate fuel reload options in relation to the NRC RT_{PTS} Screening Criterion for PTS (i.e., RT_{PTS} values can be readily projected for any options under consideration, provided fluence is known), and
- o to show the current (5.9 EFPY for Zion 1 and 5.7 EFPY for Zion 2), end-of-license (25.8 EFPY for Zion 1 and 25.3 EFPY for Zion 2) and end-of-life (32 EFPY for both Zion Units) RT_{PTS} values using actual and projected fluence.

---

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

Table IV.3-1 and IV.3-2 provide a summary of the RT_{PTS} values for all beltline region materials for the lifetime of interest.

IV.2 DISCUSSION OF RESULTS

As shown in Figures IV.1-1 and IV.1-2, the welds are the governing locations for both reactor vessels relative to PTS. All the RT_{PTS} values remain below the NRC screening values for PTS using the projected fluence values through end-of-life (32 EFPY).

TABLE IV.3-1

RT_{PTS} VALUES FOR ZION UNIT 1

Location	Vessel Material	RT_{PTS} Values (°F) Present (5.9 EFPY)	End-of-License (25.8 EFPY)	End-of-Life (32. EFPY)
1	Intermediate shell plate 8-144-2	112	132	136
2	Intermediate shell plate 8-144-1	107	127	131
3	Lower shell plate 9-144-1	103	125	129
4	Lower shell plate 9-144-2	138	164	169
5	Intermediate to lower shell circumferential weld WF-70	222	284	297
6	Intermediate shell longitudinal welds WF-4/WF-8	169	224	235

TABLE IV.3-2

RT_{PTS} VALUES FOR ZION UNIT 2

Location	Vessel Material	RT_{PTS} Values (°F) Present (5.7 EFPY)	End-of-License (25.3 EFPY)	End-of-Life (32 EFPY)
1	Intermediate shell plate 8-152-1	123	146	150
2	Intermediate shell plate 8-152-2	124	146	151
3	Lower shell plate 9-152-1	112	135	140
4	Lower shell plate 9-152-2	114	141	146
5	Intermediate to lower shell circumferential weld SA-1769	190	245	256
6	Intermediate shell longitudinal welds WF-29	147	190	199
7	Lower shell longitudinal welds WF-70	179	238	250

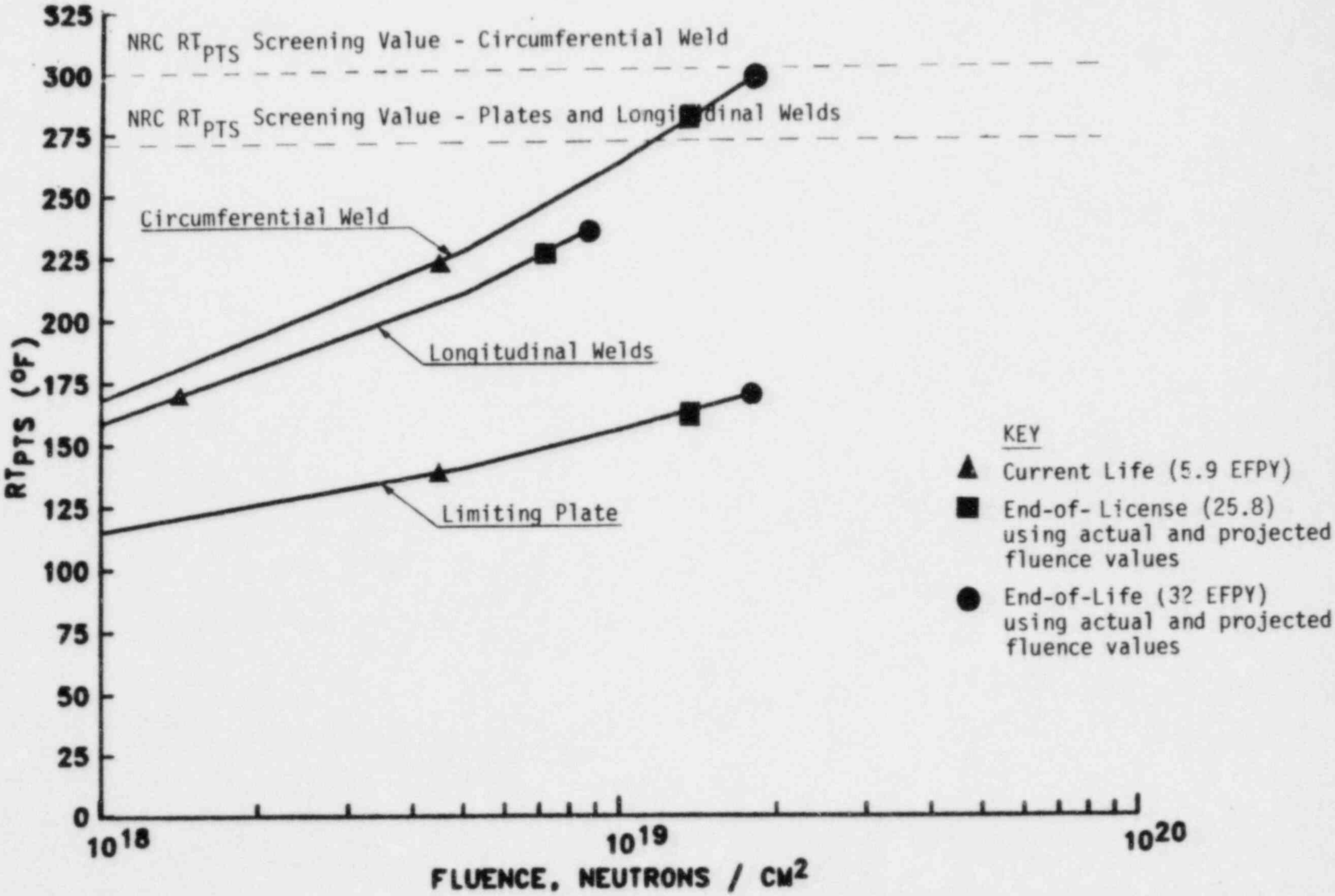


Figure IV.1.1 Zion Unit 1 - RT$_{PTS}$ Curves per PTS Rule Methodology [1]

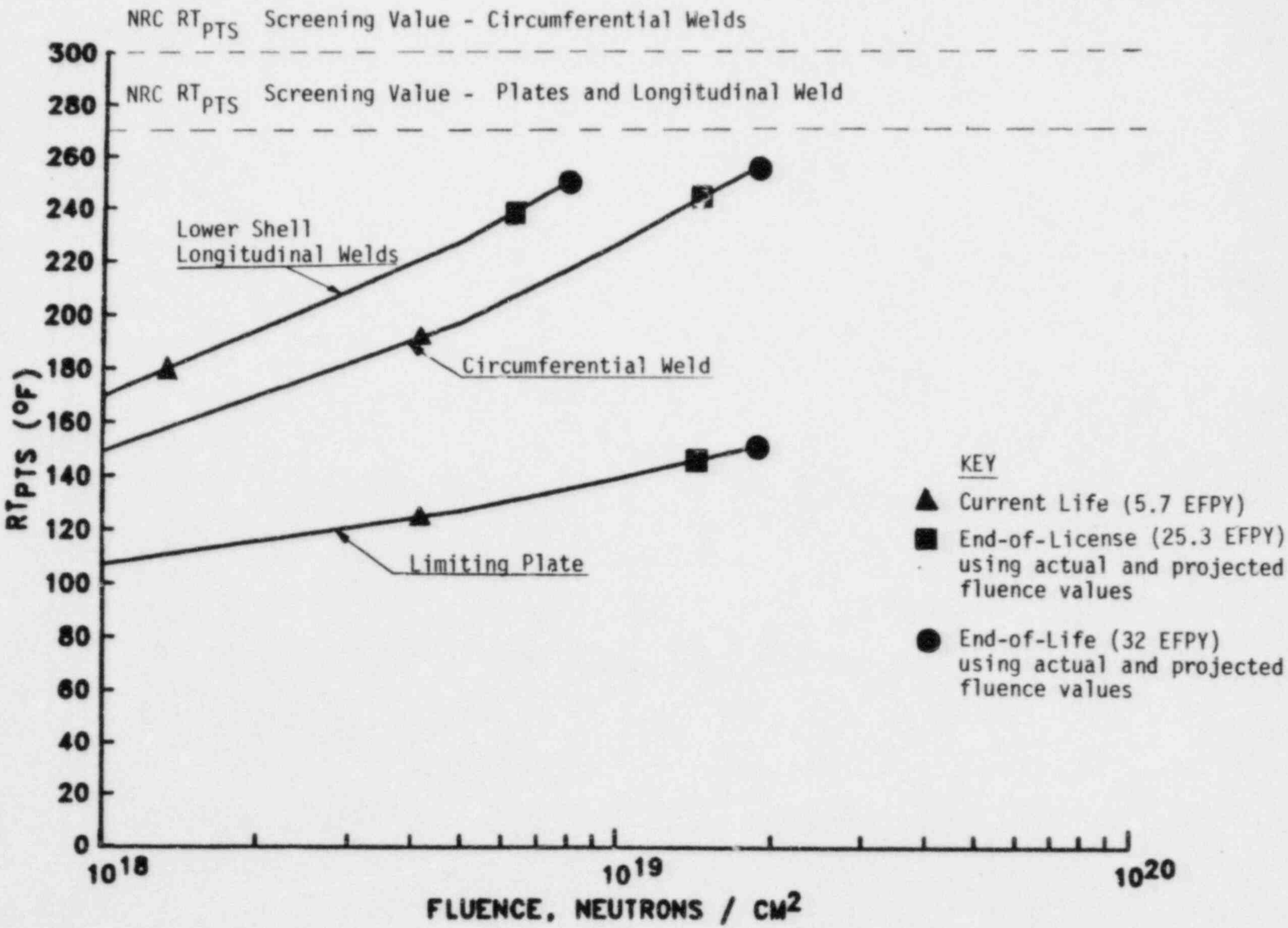


Figure IV. 2.1 Zion Unit 2 - RT_{PTS} Curve per PTS Rule Methodology [1]

SECTION V

CONCLUSIONS AND RECOMMENDATIONS

Calculations have been completed in order to determine RT_{PTS} values for the Zion Units 1 and 2 reactor vessels to meet the requirements of the NRC Rule for Pressurized Thermal Shock [1]. This work entailed a neutron exposure evaluation and a reactor vessel material study.

Detailed fast neutron exposure evaluations using plant specific cycle by cycle core power distributions and state-of-the-art neutron transport methodology have been completed for the Zion Units 1 and 2 pressure vessels. Explicit calculations were performed for the first seven operating cycles of both units. For both units, projection of the fast neutron exposure beyond the current operating cycle was based on continued implementation of low leakage fuel management similar to that employed during cycle 7 for Unit 1 and cycles 6 and 7 for Unit 2.

In regard to the low leakage fuel management already in place at the Zion Units, the plant specific evaluations have demonstrated that for the low leakage case the average fast neutron flux at the 45° azimuthal position has been reduced by about 35% at Unit 1 and 25% at Unit 2 relative to that existing prior to implementation of low leakage. In particular, the following data applies at the 45° location.

	ϕ (n/cm^2-sec)	
	Unit 1	Unit 2
Out-In Pattern	2.51×10^{10}	2.42×10^{10}
Low Leakage Pattern	1.63×10^{10}	1.76×10^{10}

This location represents the maximum fast neutron flux incident on the reactor pressure vessel. At other locations on the vessel, as well as at the surveillance capsules, the impact of low leakage will differ from the data presented above.

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

The fast neutron fluence values from the plant specific calculations have been compared directly with measured fluence levels derived from neutron dosimetry contained in the three surveillance capsules withdrawn from Zion Unit 1 and the two surveillance capusles withdrawn from Zion Unit 2. For Unit 1, the ratio of calculated to measured fluence values ranges from 0.93 to 1.02 for the three capsule data points. The corresponding ratio for Unit 2 is 0.95 for both capsules removed from that reactor. This excellent agreement between calculation and measurement supports the use of this analytical approach to perform a plant specific evaluations for the Zion reactors.

Material property values for the Zion Units 1 and 2 reactor vessel beltline region components were determined. The pertinent chemical and mechanical properties for the shell plates remain the same as those that have been docketed with the NRC in Reference 5. The weld material properties are consistent with those recommended by B&W in report BAW-1799 [6] with the exception of those weldments made with weld wire heat number 72105. For this weld material, a material chemistry study was completed by Westinghouse that included more data than that reported in BAW-1799 in order to define mean chemistry values of copper and nickel (0.32% and 0.56% respectively) for these limiting weld locations.

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Zion Units 1 and 2 reactor vessels as a function of several fluence values and pertinent vessel lifetimes. For both reactor vessels, all the RT_{PTS} values remain below the NRC screening values for PTS using the projected fluence exposure through 32 EFPY. The most

limiting values at end-of-license (25.8 EFPY for Zion 1 and 25.3 EFPY for Zion 2) are 284°F for the circumferential weld for Unit 1 and 238°F for the longitudinal welds in the lower shell of Unit 2.

This report is provided to enable Commonwealth Edison Company to comply with the initial 6 months submittal requirements of the USNRC PTS Rule.

SECTION VI

REFERENCES

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

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

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

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

5. Commonwealth Edison letter from D. E. O'Brien to Mr. A. Schwenur of the NRC, "Zion Station Units 1 and 2 NRC Docket Nos. 50-295 and 50-304," September 7, 1977.

6. B&W Owners Group Report, BAW-1799, "B&W 177-FA Reactor Vessel Beltline Weld Chemistry Study", July 1983.

7. NRC Letter Docket Nos. 50-250 and 50-251, "Evaluation of Reactor Vessel Materials Data for Turkey Point Plant Units 3 and 4 Reactor Vessels", from S. A. Varga to J. W. Williams, Jr., of Florida Power and Light Company, April 26, 1984.

8. Advisory Committee for Reactor Safeguards (ACRS) Metal Components Subcommittee Meeting, "Draft Regulatory Guide 1.99, Revision 2, Radiation Damage to Reactor Vessel Materials", Washington, D. C., September 4-5, 1985.

9. NUREG-0800 - U.S. NRC Standard Review Plan, Branch Technical Position 5-2, Revision 1, July 1981.

10. "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Regulatory Guide 1.99 - Revision 1, U.S. Nuclear Regulatory Commission, Washington, April 1977.

11. NRC Policy Issue - "Pressurized Thermal Shock," SECY-82-465, November 23, 1982.

12. Commonwealth Edison Letter from H. E. Bliss to S. Anderson (Westinghouse), "ISOTOTE and 2D Calculated Edits of the Assembly Average Burnup", June 22,1984.

APPENDIX A

POWER DISTRIBUTIONS

Core power distributions used in the plant specific fast neutron exposure analysis of the Zion Units 1 and 2 pressure vessels were derived from the following fuel cycle design reports and verified by comparison with burnup data supplied by Commonwealth Edison [12].

Fuel Cycle	Unit 1	Unit 2
1	WCAP-7675-R1	WCAP-7675
2	WCAP-8616	WCAP-8881
3	WCAP-9114	WCAP-9246
4	WCAP-9356	WCAP-9458
5	WCAP-9568	WCAP-9687
6	WCAP-9859	WCAP-9959
7	WCAP-10047	WCAP-10282

A schematic diagram of the core configuration applicable to Zion Units 1 and 2 is shown in Figure A.1-1. Cycle averaged relative assembly powers for each operating fuel cycle of Zion Units 1 and 2 are listed in Tables A.1-1 and A.1-2, respectively.

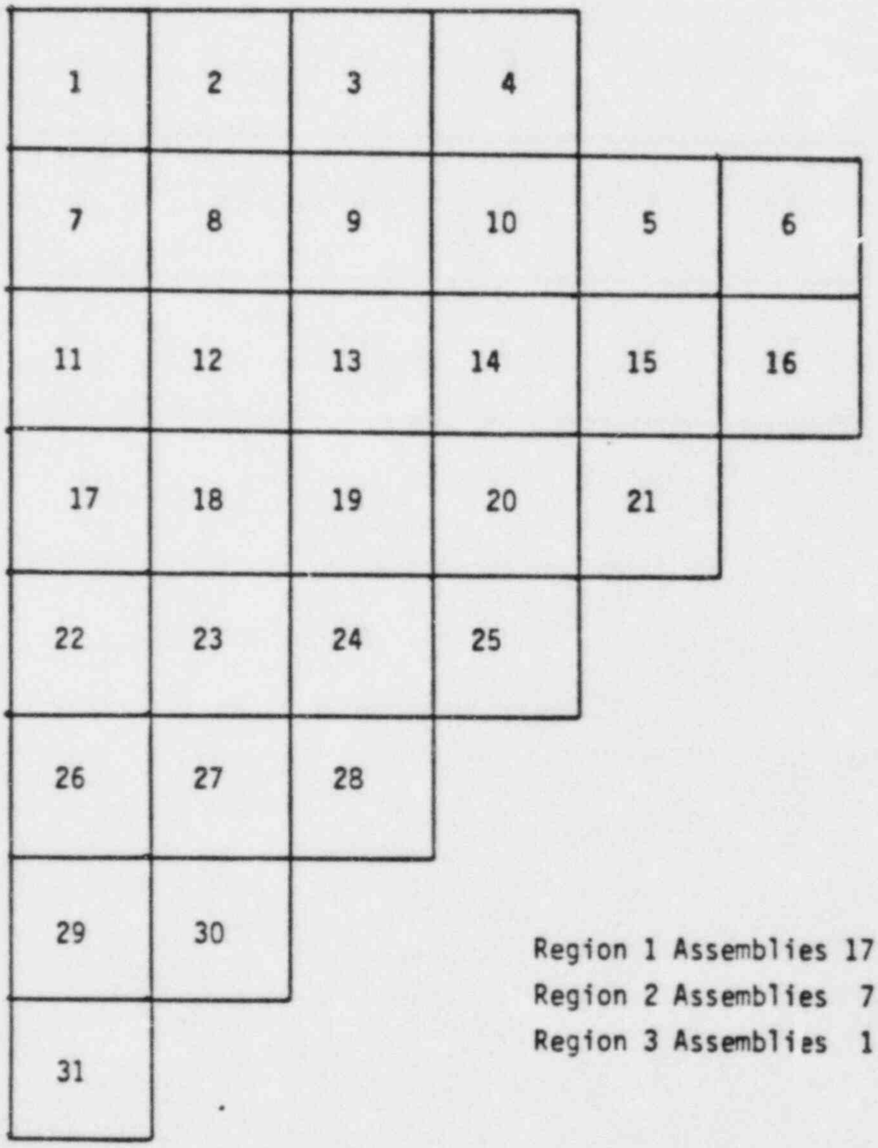
On Figure A.1-1 and in Tables A.1-1 and A.1-2 an identification number is assigned to each fuel assembly location; and three regions consisting of subsets of fuel assemblies are defined. In performing the adjoint evaluations, the relative power in assemblies comprising Region 3 has been adjusted to account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

surveillance capsules or at the pressure vessel. Therefore, power distribution data for assemblies in Region 1 are not listed in Tables A.1-1 and A.1-2.

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

Figure A.1-1

Zion Units 1 and 2 Core Description
for Power Distribution Maps



Region 1 Assemblies 17 - 31
Region 2 Assemblies 7 - 16
Region 3 Assemblies 1 - 6

TABLE A.1-1

CORE POWER DISTRIBUTIONS USED IN THE PLANT SPECIFIC FLUENCE ANALYSIS

ZION UNIT 1

Assembly	Fuel Cycle 1	2	3	4	5	6	7
1	0.81	0.98	0.98	0.91	1.02	0.96	1.00
2	0.89	0.98	1.01	0.95	1.01	0.99	1.03
3	0.76	0.92	0.95	0.89	1.01	1.01	0.95
4	0.66	0.73	0.75	0.73	0.86	0.80	0.47
5	0.93	0.99	1.03	1.06	1.10	1.10	0.64
6	0.56	0.63	0.71	0.71	0.74	0.77	0.39
7	1.04	0.97	0.87	0.83	0.92	0.88	0.79
8	1.05	1.16	1.23	1.17	1.16	0.91	1.13
9	1.01	0.96	0.99	0.96	1.01	1.19	1.13
10	1.02	1.18	1.14	1.16	1.19	1.11	1.05
11	1.17	0.94	0.82	0.86	0.82	1.08	0.89
12	1.12	0.95	0.87	0.89	0.92	1.15	1.01
13	1.15	0.95	0.93	0.90	1.29	1.13	1.23
14	1.08	1.11	1.19	1.01	1.02	1.02	1.08
15	0.97	0.99	0.99	1.19	1.08	1.16	1.13
16	1.04	0.90	1.12	1.01	0.84	0.99	1.04

TABLE A.1-2

CORE POWER DISTRIBUTIONS USED IN THE PLANT SPECIFIC FLUENCE ANALYSIS

ZION UNIT 2

Assembly	Fuel Cycle 1	2	3	4	5	6	7
1	0.80	1.04	0.89	0.59	0.95	0.88	0.72
2	0.87	1.05	0.92	0.81	0.95	0.99	0.92
3	0.75	0.98	0.85	0.83	0.90	0.91	0.85
4	0.65	0.77	0.68	0.70	0.70	0.47	0.46
5	0.92	0.99	0.98	1.01	0.98	0.91	0.96
6	0.53	0.64	0.67	0.73	0.72	0.41	0.43
7	1.02	0.96	0.87	0.77	0.88	0.89	1.19
8	1.03	1.20	1.18	1.16	1.11	1.12	1.17
9	1.00	0.98	0.94	0.92	0.89	1.16	0.90
10	1.00	1.20	1.14	1.14	1.05	1.09	1.12
11	1.17	0.87	0.93	0.98	1.05	1.09	1.21
12	1.12	1.03	1.00	0.98	0.97	0.99	1.07
13	1.15	0.94	1.10	1.01	1.19	1.18	1.24
14	1.08	0.96	1.12	0.97	0.92	1.15	1.01
15	0.97	0.88	1.01	1.03	0.88	0.91	1.04
16	1.02	0.87	1.06	1.16	1.11	0.97	1.05

APPENDIX B

WELD CHEMISTRY

Tables B.1-1 through B.1-4 provide the weld data output from the WOG Material Data Base. Given are the searches of all available data for the wire heat in the Zion Units 1 and 2 reactor vessels beltline region. The pertinent material chemical compositions are given, along with the wire/flux identification. The mean chemistry values and the population standard deviation are then calculated. The mean values of copper and nickel are used in the RT_{PTS} analysis.

Weld Chemistry Data Source and Plant:

AN1	-	Arkansas Nuclear 1
BAW-1799	-	Babcock & Wilcox Report Number
B&W	-	Babcock & Wilcox
COM	-	Zion 2
CR3	-	Crystal River 3
Cu	-	Weight % of Copper
CWE	-	Zion 1
ESA	-	Emission Spectrographic Analysis
FLA	-	Turkey Point 4
FPL	-	Turkey Point 3
MATSURV	-	NRC Mender MATSURV Data Base
MPC	-	Materials Properties Council Data Base
Ni	-	Weight % of Nickel
OC1	-	Oconee 1
P	-	Weight % of Phosphorous
RGE	-	Robert Emmett Ginna
RS1	-	Rancho Seco 1
SC	-	Surveillance Capsule
Si	-	Weight % of Silicon
TMI	-	Three Mile Island 1
VIR	-	Surry 1
WEP	-	Point Beach 1
WMQR	-	Weld Metal Qualification Retest
WQ	-	Weld Qualification

TABLE B.1-1 ZION UNIT 1 INTERMEDIATE AND LOWER SHELL LONGITUDINAL WELDS CHEMISTRY FROM WOG MATERIALS DATA BASE - WIRE HEAT NUMBER 8T1762

ID	WIRE HEAT	WIRE TYPE	FLUX TYPE	FLUX LOT	WELDCHEM DATA SOURCE	Cu	Ni	P	Si	PLANT	DESCRIPTION
0216	8T1762	MN-MO-NI	LINDE 80	8632	BW,WQ	0.200	0.610	0.009	0.530	CR3	INTER SHELL LONG
										CWE	INTER SHELL LONG
										CWE	LOWER SHELL LONG
										TM1	INTER SHELL LONG
										VIR	LOWER SHELL LONG
0270	8T1762	MN-MO-NI	LINDE 80	8553	BW,WQ	0.180	0.610	0.017	0.430	OC1	LOWER SHELL LONG
										WEP	NOZZLE TO INTER SHELL
0278	8T1762	MN-MO-NI	LINDE 80	8650	BW,WQ	0.105	0.450	0.004	0.390	AN1	INTER SHELL LONG
										AN1	LOWER SHELL LONG
										CR3	INTER SHELL LONG
0335	8T1762	MN-MO-NI	LINDE 80	8597	BW,WQ	0.170	0.530	0.017	0.510	CWE	INTER SHELL LONG
										VIR	LOWER SHELL LONG
0336	8T1762	MN-MO-NI	LINDE 80	8553	BAW-1799,WQ	0.160	0.600	0.017	0.430	OC1	LOWER SHELL LONG
										WEP	NOZZLE TO INTER SHELL
0337	8T1762	MN-MO-NI	LINDE 80	8596	BAW-1799,WQ	0.220	0.600	0.015	0.430	CR3	LOWER SHELL LONG
0772	8T1762	MN-MO-NI	LINDE 80	8578	BAW-1799,WQ	0.220	0.430	0.017	0.460		
mean						0.179286*	0.547143**	0.013714	0.454286		
std.dev.						0.040252	0.078467	0.005187	0.049618		

*Since these values are limited to the weld metal qualification test reports, report BAW-1799 recommends that a mean value of copper equal to 0.29 wt% with a standard deviation equal to 0.07 wt% be applied because a bias exists in the retest of similar type welds.

**BAW-1799 recommends that a mean value of nickel equal to 0.55 wt% be applied.

TABLE B.1-2 ZION UNIT 2 LOWER SHELL LONGITUDINAL WELD CHEMISTRY FROM WOG MATERIAL DATA BASE - WIRE HEAT NUMBER 72102

ID	WIRE HEAT	WIRE TYPE	FLUX TYPE	FLUX LOT	WELDCHEM DATA SOURCE	Cu	Ni	P	Si	PLANT	DESCRIPTION
0217	72102	MN-MO-NI	LINDE 80	8650	BW,WQ	0.160	0.270	0.017	0.420	COM RS1 RS1	INTER SHELL LONG INTER SHELL LONG LOWER SHELL LONG
0245	72102	MN-MO-NI	LINDE 80	8479	BW,WQ	0.210	0.530	0.022	0.470		
0764	72102	MN-MO-NI	LINDE 80	8650	BAW-1799,WQR	0.210	0.630	0.015	0.420	COM RS1 RS1	INTER SHELL LONG INTER SHELL LONG LOWER SHELL LONG
						*	**				
mean						0.193333	0.476667	0.018000	0.436667		
std.dev.						0.028868	0.185831	0.003606	0.028868		

*BAW-1799 recommends that a mean value of copper equal to 0.23 wt% with a standard deviation equal to 0.07 wt% be applied based upon retest of weld metal qualification test samples for similar welds.

**BAW-1799 recommends that a mean value of nickel equal to 0.63 wt% be applied.

TABLE B.1-3 ZION UNIT 2 BELTLINE CIRCUMFERENTIAL WELD CHEMISTRY FROM WOG MATERIALS DATA BASE - WIRE HEAT NUMBER 71249

ID	WIRE HEAT	WIRE TYPE	FLUX TYPE	FLUX LOT	WELDCHEM DATA SOURCE	Cu	Ni	P	Si	PLANT	DESCRIPTION
0219	71249	MN-MO-NI	LINDE 80	8738	BW,WQ	0.190	0.660	0.021	0.450	COM	INTER TO LOWER SHELL
										CR3	NOZZLE TO INTER SHELL
0223	71249	MN-MO-NI	LINDE 80	8445	BW,WQ	0.210	0.570	0.021	0.520	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0241	71249	MN-MO-NI	LINDE 80	8669	BW,WQ	0.210	0.550	0.012	0.410		
0273	71249	MN-MO-NI	LINDE 80	8457	BW,WQ	0.230	0.550	0.020	0.510	FLA	SURVEILLANCE WELD
0296	71249	MN-MO-NI	LINDE 80	8445	FPL,SC	0.310	0.570	0.011	0.660	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0297	71249	MN-MO-NI	LINDE 80	8457	FLA,SC	0.300	0.600	0.014	0.500	FLA	SURVEILLANCE WELD
0454	71249	MN-MO-NI	LINDE 80	8445	BAW-1799,ESA	0.160	0.550	0.019	0.540	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0455	71249	MN-MO-NI	LINDE 80	8445	BAW-1799,ESA	0.150	0.540	0.018	0.550	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0456	71249	MN-MO-NI	LINDE 80	8445	BAW-1799,ESA	0.180	0.550	0.019	0.540	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0457	71249	MN-MO-NI	LINDE 80	8445	BAW-1799,ESA	0.190	0.540	0.019	0.610	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0458	71249	MN-MO-NI	LINDE 80	8445	BAW-1799,ESA	0.150	0.550	0.020	0.600	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0459	71249	MN-MO-NI	LINDE 80	8445	BAW-1799,ESA	0.170	0.540	0.019	0.620	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0460	71249	MN-MO-NI	LINDE 80	8445	BAW-1799,ESA	0.200	0.540	0.020	0.630	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0461	71249	MN-MO-NI	LINDE 80	8445	BAW-1799,ESA	0.200	0.540	0.019	0.630	FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL

TABLE B.1-3 (cont'd)

0462	71249	MN-MO-NI	LINDE 80	8445	BAW-1799,ESA	0.230	0.520	0.017	0.620	WEP FLA FPL FPL RGE WEP	INTER TO LOWER SHELL INTER TO LOWER SHELL INTER TO LOWER SHELL SURVEILLANCE WELD NOZZLE TO INTER SHELL INTER TO LOWER SHELL
0463	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.310	0.640	0.021	0.550	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0464	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.270	0.640	0.021	0.550	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0465	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.270	0.640	0.020	0.560	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0466	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.280	0.640	0.021	0.560	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0467	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.280	0.640	0.021	0.550	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0468	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.310	0.640	0.020	0.560	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0469	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.280	0.640	0.020	0.550	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0470	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.280	0.630	0.021	0.550	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0471	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.270	0.630	0.020	0.550	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0472	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.260	0.620	0.019	0.550	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0473	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.240	0.620	0.020	0.560	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0474	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.280	0.620	0.020	0.570	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0475	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.280	0.620	0.020	0.570	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0476	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.270	0.620	0.019	0.560	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0477	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.290	0.630	0.020	0.570	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0478	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.300	0.630	0.021	0.580	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0479	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.340	0.630	0.020	0.580	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0480	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.280	0.620	0.020	0.570	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0481	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.290	0.620	0.019	0.540	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0482	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.310	0.620	0.020	0.560	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0483	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.280	0.620	0.020	0.570	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0484	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.280	0.630	0.019	0.580	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0485	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.270	0.630	0.018	0.550	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0486	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.310	0.630	0.018	0.560	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0487	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.310	0.630	0.018	0.560	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0488	71249	MN-MO-NI	LINDE 80	8738	BAW-1799,ESA	0.290	0.630	0.018	0.560	COM CR3	INTER TO LOWER SHELL NOZZLE TO INTER SHELL
0773	71249	MN-MO-NI	LINDE 80	8492	BAW-1799,WO	0.200	0.570	0.021	0.430		
0775	71249	MN-MO-NI	LINDE 80	8445	FFL,SC	0.350				FLA FPL FPL	INTER TO LOWER SHELL INTER TO LOWER SHELL SURVEILLANCE WELD

TABLE B.1-3 (cont'd)

										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0776	71249	MN-MO-NI	LINDE 80	8445	FPL,SC	0.340				FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
0777	71249	MN-MO-NI	LINDE 80	8445	FPL,SC	0.320				FLA	INTER TO LOWER SHELL
										FPL	INTER TO LOWER SHELL
										FPL	SURVEILLANCE WELD
										RGE	NOZZLE TO INTER SHELL
										WEP	INTER TO LOWER SHELL
mean						0.260444	0.602381	0.019143	0.556905		
std.dev.						0.053511	0.040353	0.002193	0.047805		

TABLE B.1-4 ZION UNIT 1 BELTLINE CIRCUMFERENTIAL WELD AND ZION UNIT 2 LOWER SHELL LONGITUDINAL WELD CHEMISTRY FROM WOG MATERIALS DATA BASE - WIRE HEAT NUMBER 72105

ID	WIRE HEAT	WIRE TYPE	FLUX TYPE	FLUX LOT	WELDCHEM DATA SOURCE	Cu	Ni	P	Si	PLANT	DESCRIPTION
0218	72105	MN-MO-NI	LINDE 80	8669	BW,WQ	0.270	0.460	0.014	0.480	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0274	72105	MN-MO-NI	LINDE 80	8773	BW,WQ	0.300	0.480	0.020	0.560	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0284	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.350	0.570	0.020	0.680	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0295	72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.280	[illegible]	0.017	0.470	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0403	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.430	0.590	0.021	0.650	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0404	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.420	0.590	0.020	0.600	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0405	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.400	0.590	0.020	0.600	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0406	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.390	0.590	0.019	0.560	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0407	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.350	0.580	0.018	0.530	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL

TABLE B.1-4 (cont'd)

										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0408	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.350	0.580	0.018	0.540	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0409	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.390	0.580	0.018	0.550	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0410	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.370	0.590	0.018	0.540	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0411	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.360	0.590	0.018	0.530	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0412	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.480	0.590	0.018	0.530	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0413	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.470	0.600	0.018	0.520	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0414	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.470	0.610	0.017	0.490	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0415	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.490	0.610	0.018	0.490	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0416	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.470	0.610	0.017	0.480	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL

TABLE B.1-4 (cont'd)

										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0417	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,ESA	0.440	0.600	0.018	0.490	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TM1	NOZZLE TO INTER SHELL
0418	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.350	0.590	0.023	0.690	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0419	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.360	0.580	0.022	0.640	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0420	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.350	0.580	0.021	0.630	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0421	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.360	0.580	0.023	0.690	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0422	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.360	0.580	0.021	0.600	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0423	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.360	0.570	0.022	0.640	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0424	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.370	0.590	0.018	0.540	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0425	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.350	0.610	0.017	0.530	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0426	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.370	0.600	0.019	0.560	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0427	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.330	0.620	0.019	0.540	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD

TABLE B.1-4 (cont'd)

0428	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.320	0.590	0.015	0.590	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0429	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.320	0.590	0.014	0.580	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0430	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.320	0.590	0.015	0.560	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0431	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.310	0.590	0.015	0.570	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0432	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.300	0.580	0.016	0.560	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0433	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.310	0.580	0.016	0.570	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0434	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.320	0.580	0.016	0.590	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0435	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.310	0.580	0.017	0.680	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0436	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.310	0.580	0.017	0.590	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0437	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.300	0.580	0.017	0.590	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0438	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.310	0.580	17	0.590	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0439	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.300	0.580	0.017	0.610	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0440	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.280	0.590	0.016	0.570	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD

TABLE B.1-4 (cont'd)

0441	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.280	0.590	0.016	0.550	OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0442	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.300	0.580	0.016	0.560	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0443	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.300	0.590	0.016	0.560	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0444	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.270	0.580	0.024	0.770	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0445	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.300	0.580	0.023	0.690	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0446	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.310	0.580	0.022	0.660	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0447	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.320	0.580	0.022	0.660	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0448	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.290	0.570	0.016	0.690	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0449	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.300	0.580	0.017	0.590	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0450	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.290	0.570	0.016	0.570	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0451	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.290	0.580	0.017	0.600	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
0452	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.300	0.580	0.016	0.600	CWE OC2 OC3 COM CR3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD
										CWE OC2 OC3	SURVEILLANCE WELD SURVEILLANCE WELD SURVEILLANCE WELD

TABLE B.1-4 (cont'd)

0453	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,ESA	0.280	0.580	0.017	0.600	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0658	72105	MN-MO-NI	LINDE 80	8773	MPC,DB,OC2,SC	0.360	0.580	0.022	0.650	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0659	72105	MN-MO-NI	LINDE 80	8773	MPC,DB,OC3,SC	0.300	0.580	0.017	0.610	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0664	72105	MN-MO-NI	LINDE 80	8773	MPC,DB,CR3,SC	0.390	0.100	0.021	1.000	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0678	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.216	0.530	0.017	0.619	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0679	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.270	0.570			COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0680	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.218	0.545	0.018	0.661	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0681	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.250	0.490			COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0682	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.260	0.560			COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0683	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.260	0.540			COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0684	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.240	0.550			COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0685	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.260	0.530			COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0686	72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.280	0.560			COM	SURVEILLANCE WELD
										CR3	

TABLE B.1-4 (cont'd)

0687 72105	MN-MO-NI	LINDE 80	8773	CWE,SC	0.250	0.540			CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0688 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.190	0.520			CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0689 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.230	0.520			CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0690 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.260	0.530	0.024	0.520	CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0691 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.230	0.540			CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0692 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.250	0.530			CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0693 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.310	0.520	0.024	0.270	CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0694 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.210	0.480			CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0695 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.280	0.550	0.026	0.490	CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0696 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.230	0.470			CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0697 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.220	0.520			CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
0698 72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.200	0.560			CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD
									OC3	SURVEILLANCE WELD
									COM	SURVEILLANCE WELD
									CR3	
									CWE	SURVEILLANCE WELD
									OC2	SURVEILLANCE WELD

TABLE B.1-4 (cont'd)

										OC3	SURVEILLANCE WELD
0714	72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.270	0.530			COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0715	72105	MN-MO-NI	LINDE 80	8773	COM,SC	0.260	0.540			COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0744	72105	MN-MO-NI	LINDE 80	8773	MPC,DB,OC2	0.340	0.600	0.013		COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0752	72105	MN-MO-NI	LINDE 80	8669	BAW-1799,WQR	0.340	0.580	0.019	0.510	COM	LOWER SHELL LONG
										CR3	INTER TO LOWER SHELL
										CWE	INTER TO LOWER SHELL
										FLA	NOZZLE TO INTER SHELL
										OC3	INTER TO LOWER SHELL
										RS1	LOWER SHELL LONG
										TMI	NOZZLE TO INTER SHELL
0753	72105	MN-MO-NI	LINDE 80	8688	BAW-1799,WQR	0.300	0.610	0.017	0.580		
0754	72105	MN-MO-NI	LINDE 80	8773	BAW-1799,WQR	0.400	0.590	0.021	0.570	COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
0771	72105	MN-MO-NI	LINDE 80	8688	BAW-1799,WQ	0.210	0.590	0.018	0.590		
0774	72105	MN-MO-NI	LINDE 80	8773	MPC,DB,OC3	0.290	0.580	0.017		COM	SURVEILLANCE WELD
										CR3	
										CWE	SURVEILLANCE WELD
										OC2	SURVEILLANCE WELD
										OC3	SURVEILLANCE WELD
mean						0.315563	0.563506	0.018464	0.584627		
std.dev.						0.067005	0.060015	0.002747	0.088673		

APPENDIX C

RT_{PTS} VALUES OF ZION UNITS 1 AND 2 REACTOR VESSEL BELTLINE REGION MATERIALS

C.1 ZION UNIT 1

Tables C.1-1 through C.1-6 provide the RT_{PTS} values, as a function of both constant fluence and constant EFPY (assuming the projected fluences values), for all beltline region materials of the Zion Unit 1 reactor vessel. The RT_{PTS} values are calculated in accordance with the PTS rule, which is Reference [1] in the main body of this report. The vessel location numbers in the following tables correspond to the vessel materials identified below and in Table III.4-1 of the main report.

Location	Vessel Material
1	Intermediate shell plate 8-144-2
2	Intermediate shell plate 8-144-1
3	Lower shell plate 9-144-1
4	Lower shell plate 9-144-2
5	Intermediate to lower shell circumferential weld WF-70
6	Intermediate and lower shells longitudinal welds WF-4/WF-8

C.2 ZION UNIT 2

Tables C.2-1 through C.2-6 provide the RT_{PTS} values, as a function of both constant fluence and constant EFPY (assuming the projected fluence values), for all beltline region materials of the Zion Unit 2 reactor vessel. The RT_{PTS} values are calculated in accordance with the proposed PTS rule, which is Reference [1] in the main body of this report. The vessel location numbers in the following tables correspond to the vessel materials identified below and in Table III.4-2 of the main report.

Location	Vessel Material
1	Intermediate shell plate 8-152-1
2	Intermediate shell plate 8-152-1
3	Lower shell plate 8-152-1
4	Lower shell plate 9-152-2
5	Intermediate to lower shell circumferential weld SA-1769
6	Intermediate shell longitudinal weld WF-29
7	Lower shell longitudinal weld WF-70

TABLE C.1-1

RT_{PTS} VALUES FOR ZION UNIT 1 REACTOR VESSEL BELTLINE REGION MATERIALS
@ FLUENCE = 1.0×10^{18} n/cm^2

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	CWE	.12	.49	.010	10.	ACTUAL	B.M.	.10E+19	94.
2	CWE	.12	.49	.010	5.	ACTUAL	B.M.	.10E+19	89.
3	CWE	.13	.48	.013	-4.	ACTUAL	B.M.	.10E+19	83.
4	CWE	.15	.50	.010	20.	ACTUAL	B.M.	.10E+19	115.
5	CWE	.32	.56	.017	0.	GENERIC	C.W.	.10E+19	168.
6	CWE	.29	.55	.013	0.	GENERIC	L.W.	.10E+19	157.

TABLE C.1-2

RT_{PTS} VALUES FOR ZION UNIT 1 REACTOR VESSEL BELTLINE REGION MATERIALS @ FLUENCE = 5.0×10^{18} n/cm^2

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	CWE	.12	.48	.010	10.	ACTUAL	B.M.	.50E+19	114.
2	CWE	.12	.49	.010	5.	ACTUAL	B.M.	.50E+19	109.
3	CWE	.13	.48	.013	-4.	ACTUAL	B.M.	.50E+19	104.
4	CWE	.15	.50	.010	20.	ACTUAL	B.M.	.50E+19	140.
5	CWE	.32	.56	.017	0.	GENERIC	C.W.	.50E+19	227.
5	CWE	.28	.55	.013	0.	GENERIC	L.W.	.50E+19	210.

TABLE C.1-3

RT_{PTS} VALUES FOR ZION UNIT 1 REACTOR VESSEL BELTLINE MATERIALS
@ FLUENCE = 1.0×10^{19} n/cm^2

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	CWE	.12	.49	.010	10.	ACTUAL	B.M.	1.10E+20	125.
2	CWE	.12	.49	.010	5.	ACTUAL	B.M.	1.10E+20	120.
3	CWE	.13	.48	.013	-4.	ACTUAL	B.M.	1.10E+20	117.
4	CWE	.15	.50	.010	20.	ACTUAL	B.M.	1.10E+20	155.
5	CWE	.32	.56	.017	0.	GENERIC	C.W.	1.10E+20	262.

TABLE C.1-4

RT_{PTS} VALUES FOR ZION UNIT 1 REACTOR VESSEL BELTLINE REGION MATERIALS
@CURRENT (5.9 EFPY) FLUENCE

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	CWE	.12	.49	.010	10.	ACTUAL	B.M.	.44E+19	112.
2	CWE	.12	.49	.010	5.	ACTUAL	B.M.	.44E+19	107.
3	CWE	.13	.48	.013	-4.	ACTUAL	B.M.	.44E+19	103.
4	CWE	.15	.50	.010	20.	ACTUAL	B.M.	.44E+19	138.
5	CWE	.32	.56	.017	0.	GENERIC	C.W.	.44E+19	222.
6	CWE	.29	.55	.013	0.	GENERIC	L.W.	.15E+19	169.

TABLE C.1-5

RT_{PTS} VALUES FOR ZION UNIT 1 REACTOR VESSEL BELTLINE REGION MATERIALS

@END OF LICENSE (25.8 EFPY) - PROJECTED FLUENCE VALUE

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	CWE	.12	.49	.010	10.	ACTUAL	B.M.	.15E+20	132.
2	CWE	.12	.49	.010	5.	ACTUAL	B.M.	.15E+20	127.
3	CWE	.13	.48	.013	-4.	ACTUAL	B.M.	.15E+20	125.
4	CWE	.15	.50	.010	20.	ACTUAL	B.M.	.15E+20	164.
5	CWE	.32	.56	.017	0.	GENERIC	C.W.	.15E+20	284.
6	CWE	.29	.55	.013	0.	GENERIC	L.W.	.70E+19	224.

TABLE C.1-6

RT_{PTS} VALUES FOR ZION UNIT 1 REACTOR VESSEL BELTLINE REGION MATERIALS @ 32 EFPY - PROJECTED FLUENCE VALUES

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	CWE	.12	.49	.010	10.	ACTUAL	B.M.	.18E+20	136.
2	CWE	.12	.49	.010	5.	ACTUAL	B.M.	.18E+20	131.
3	CWE	.13	.48	.013	-4.	ACTUAL	B.M.	.18E+20	129.
4	CWE	.15	.50	.010	20.	ACTUAL	B.M.	.18E+20	169.
5	CWE	.32	.56	.017	0.	GENERIC	C.W.	.18E+20	297.
6	CWE	.29	.55	.013	0.	GENERIC	L.W.	.87E+19	235.

TABLE C.2-1

RT_{PTS} VALUES FOR ZION UNIT 2 REACTOR VESSEL BELTLINE REGION MATERIALS
@FLUENCE = 1.0×10^{18} n/cm^2

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	COM	.12	.51	.010	22.	ACTUAL	B.M.	.10E+19	106.
2	COM	.12	.53	.010	22.	ACTUAL	B.M.	.10E+19	107.
3	COM	.12	.54	.010	10.	ACTUAL	B.M.	.10E+19	95.
4	COM	.14	.52	.008	2.	ACTUAL	B.M.	.10E+19	94.
5	COM	.26	.60	.019	0.	GENERIC	C.W.	.10E+19	149.
6	COM	.23	.63	.019	0.	GENERIC	L.W.	.10E+19	139.
7	COM	.32	.56	.017	0.	GENERIC	L.W.	.10E+19	168.

TABLE C.2-2

RT_{PTS} VALUES FOR ZION UNIT 2 REACTOR VESSEL BELTLINE REGION MATERIALS

@FLUENCE = 5.0×10^{18} n/cm^2

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	COM	.12	.51	.010	22.	ACTUAL	B.M.	.50E+19	126.
2	COM	.12	.53	.010	22.	ACTUAL	B.M.	.50E+19	127.
3	COM	.12	.54	.010	10.	ACTUAL	B.M.	.50E+19	115.
4	COM	.14	.52	.008	2.	ACTUAL	B.M.	.50E+19	117.
5	COM	.26	.60	.018	0.	GENERIC	C.W.	.50E+19	197.
6	COM	.23	.63	.018	0.	GENERIC	L.W.	.50E+19	182.
7	COM	.32	.56	.017	0.	GENERIC	L.W	.50E+19	227.

TABLE C.2-3

RT_{PTS} VALUES FOR ZION UNIT 2 REACTOR VESSEL BELTLINE MATERIALS
@FLUENCE = 1.0×10^{19} n/cm^2

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	COM	.12	.51	.010	22.	ACTUAL	B.M.	.10E+20	138.
2	COM	.12	.53	.010	22.	ACTUAL	B.M.	.10E+20	139.
3	COM	.12	.54	.010	10.	ACTUAL	B.M.	.10E+20	127.
4	COM	.14	.52	.008	2.	ACTUAL	B.M.	.10E+20	131.
5	COM	.26	.60	.019	0.	GENERIC	C.W.	.10E+20	226.

TABLE C.2-4

RT_{PTS} VALUES FOR ZION UNIT 2 REACTOR VESSEL BELTLINE REGION MATERIALS
@CURRENT (5.7 EFPY) FLUENCE

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	COM	.12	.51	.010	22.	ACTUAL	B.M.	.41E+19	123.
2	COM	.12	.53	.010	22.	ACTUAL	B.M.	.41E+19	124.
3	COM	.12	.54	.010	10.	ACTUAL	B.M.	.41E+19	112.
4	COM	.14	.52	.008	2.	ACTUAL	B.M.	.41E+19	114.
5	COM	.26	.60	.019	0.	GENERIC	C.W.	.41E+19	190.
6	COM	.23	.63	.019	0.	GENERIC	L.W.	.14E+19	147.
7	COM	.32	.56	.017	0.	GENERIC	L.W.	.14E+19	179.

TABLE C.2-5

RT_{PTS} VALUES FOR ZION UNIT 2 REACTOR VESSEL BELTLINE REGION MATERIALS
@END OF LICENSE (25.3 EFPY) - PROJECTED FLUENCE VALUE

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	COM	.12	.51	.010	22.	ACTUAL	B.M.	.15E+20	146.
2	COM	.12	.53	.010	22.	ACTUAL	B.M.	.15E+20	146.
3	COM	.12	.54	.010	10.	ACTUAL	B.M.	.15E+20	135.
4	COM	.14	.52	.008	2.	ACTUAL	B.M.	.15E+20	141.
5	COM	.26	.60	.019	0.	GENERIC	C.W.	.15E+20	245.
6	COM	.23	.63	.019	0.	GENERIC	L.W.	.63E+19	190.
7	COM	.32	.56	.017	0.	GENERIC	L.W.	.63E+19	238.

TABLE C.2-6

RT_{PTS} VALUES FOR ZION UNIT 2 REACTOR VESSEL BELTLINE REGION MATERIALS
@32 EFPY - PROJECTED FLUENCE VALUES

LOC	PLANT	CU	NI	P	RTNDTI	VALUE	TYPE	FLUENCE	RTPTS
1	COM	.12	.51	.010	22.	ACTUAL	B.M.	.19E+20	150.
2	COM	.12	.83	.010	22.	ACTUAL	B.M.	. 3E+20	151.
3	COM	.12	.84	.010	10.	ACTUAL	B.M.	.19E+20	140.
4	COM	.14	.52	.008	2.	ACTUAL	B.M.	.19E+20	146.
5	COM	.26	.60	.018	0.	GENERIC	C.W.	.19E+20	256.
6	COM	.23	.63	.018	0.	GENERIC	L.W.	.80E+19	198.
7	COM	.32	.56	.017	0.	GENERIC	L.W.	.80E+19	250.