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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261 / RENEWED LICENSE NO. DPR-23

SUPPLEMENT TO RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO REVISE REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS

Dear Sir/Madam:

By letter dated November 2, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15307A069) Duke Energy Progress, Inc. (DEP) submitted a license amendment request (LAR) for H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2). This LAR would revise the reactor coolant system (RCS) pressure and temperature (P-T) limits in the Technical Specifications (TSs) of the HBRSEP2. The proposed revision would extend the HBRSEP2 P-T limits applicability from the current 35 effective full power years (EFPY) up to 50 EFPY. The 50 EFPY P-T limits are based on the P-T limit curves developed in Westinghouse report, WCAP-15827, Revision 0, "H. B. Robinson Unit 2, Heatup and Cooldown Limit Curves for Normal Operation," March 2003, which was included as Attachment 4 to the submittal.

The Nuclear Regulatory Commission (NRC) staff determined that additional information is needed to complete its LAR review. A draft of that information request was received by DEP via electronic mail message dated March 2, 2016, which provided four (4) requests for additional information (RAIs).

An RAI clarification call was held on March 15, 2016 between NRC staff and DEP. DEP requested a revised RAI response date for RAI No. 3 to accommodate outstanding vendor deliverables necessary for DEP's response. The DEP responses to RAIs 1, 2, and 4 were provided to NRC via letter dated March 31, 2016. The enclosure to this letter provides Westinghouse report, WCAP-18100-NP, which is specifically referenced in the DEP response for RAI No. 2.

Please address any comments or questions regarding this matter to Mr. Scott Connelly, Acting Manager – Nuclear Regulatory Affairs at (843) 857-1569.

There are no new regulatory commitments made in this letter.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 9, 2016.

Sincerely,

R. Michael Glover

R. Michael Glover
Site Vice President

RMG/jmw

Enclosure

cc: Region Administrator, NRC, Region II
Mr. Dennis Galvin, NRC Project Manager, NRR
NRC Resident Inspector, HBRSEP2
Ms. S. E. Jenkins, Manager, Infectious and Radioactive Waste Management Section (SC)

U. S. Nuclear Regulatory Commission
Enclosure to Serial: RNP-RA/16-0031
63 Pages (including this cover page)

SUPPLEMENT TO RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO REVISE REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS

WCAP-18100-NP
Revision 0

March 2016

Ex-Vessel Neutron Dosimetry Program for H. B. Robinson Unit 2 Cycles 16 through 29

Westinghouse

WCAP-18100-NP
Revision 0

Ex-Vessel Neutron Dosimetry Program for H. B. Robinson Unit 2 Cycles 16 through 29

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March 2016

Reviewer: Greg A. Fischer*
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Approved: Laurent P. Houssay*, Manager
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*Electronically approved records are authenticated in the electronic document management system.

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RECORD OF REVISIONS

Revision	Description	Completed
0-A	Original Draft Issue	February 2016
0	Original Issue; there were no comments on Revision 0-A by Duke Energy; therefore, Revision 0 is the same as Revision 0-A of this report	March 2016

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EXECUTIVE SUMMARY

This report describes a neutron fluence analysis performed for the H. B. Robinson Unit 2 pressure vessel beltline and extended beltline regions based on the guidance specified in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." In this analysis, maximum fast neutron exposures expressed in terms of fast neutron fluence (E > 1.0 MeV) and iron displacements per atom (dpa) were established for the beltline and extended beltline regions of the pressure vessel.

In the beginning of Cycles 10 and 11, temporary ex-vessel neutron dosimetry was installed at three azimuthal locations at an axial elevation above the active fuel and three corresponding azimuthal locations at an axial elevation approximately one foot below the core midplane. The permanent ex-vessel neutron dosimetry installation that occurred at the beginning Cycle 12 placed multiple foil sensor sets such that spectra evaluations could be made at three azimuthal locations at an axial elevation opposite the lower circumferential weld and three corresponding azimuthal locations adjacent to the upper circumferential weld. The use of the ex-vessel neutron dosimetry program coupled with available surveillance capsule measurements provides a plant-specific database that enables the evaluation of the vessel exposure and the uncertainty associated with that exposure over the service life of the unit.

In this report, ex-vessel neutron dosimetry is evaluated at the conclusion of Cycles 10, 11, 13, 15, and 29. In addition, the neutron dosimetry sensor sets from the four internal surveillance capsules withdrawn and evaluated from H. B. Robinson Unit 2 were re-analyzed using the current dosimetry evaluation methodology. The dosimetry re-evaluations were used to validate the calculational models that were applied in the plant-specific neutron transport analyses.

Based on the plant-specific neutron transport calculations for the first 29 operating cycles (approximately 33.18 effective full-power years (EFPY)) at H. B. Robinson Unit 2, the reactor pressure vessel beltline materials accrued the following maximum fast neutron exposure:

Cycle	Total Time (EFPY)	Fluence (n/cm^2)			
		0°	15°	30°	45°
29	33.18	3.97E+19	2.07E+19	1.30E+19	8.90E+18

Cycle	Total Time (EFPY)	dpa			
		0°	15°	30°	45°
29	33.18	6.53E-02	3.43E-02	2.13E-02	1.45E-02

The comparisons of calculations with the H. B. Robinson Unit 2 core midplane measurement database demonstrated that the plant-specific calculations meet the uncertainty requirements specified in Regulatory Guide 1.190.

Based on the core power distributions and associated reactor operating characteristics of Cycles 27 through 29, including the current core power level of 2339 MWt, projections of the maximum neutron exposure of the reactor pressure vessel beltline materials for reactor operation through 80 EFPY were calculated. The maximum exposures obtained from these projections are as follows:

Total Time (EFPY)	Fluence (n/cm^2)			
	0°	15°	30°	45°
35.00	4.16E+19	2.16E+19	1.35E+19	9.29E+18
40.00	4.67E+19	2.41E+19	1.49E+19	1.04E+19
45.00	5.18E+19	2.65E+19	1.62E+19	1.14E+19
50.00	5.69E+19	2.90E+19	1.76E+19	1.25E+19
55.00	6.20E+19	3.14E+19	1.90E+19	1.36E+19
60.00	6.71E+19	3.39E+19	2.03E+19	1.46E+19
65.00	7.22E+19	3.64E+19	2.17E+19	1.57E+19
70.00	7.74E+19	3.88E+19	2.31E+19	1.68E+19
75.00	8.25E+19	4.13E+19	2.44E+19	1.78E+19
80.00	8.76E+19	4.37E+19	2.58E+19	1.89E+19

Total Time (EFPY)	dpa			
	0°	15°	30°	45°
35.00	6.83E-02	3.58E-02	2.21E-02	1.51E-02
40.00	7.67E-02	3.99E-02	2.43E-02	1.68E-02
45.00	8.51E-02	4.40E-02	2.65E-02	1.86E-02
50.00	9.35E-02	4.80E-02	2.88E-02	2.03E-02
55.00	1.02E-01	5.21E-02	3.10E-02	2.21E-02
60.00	1.10E-01	5.62E-02	3.32E-02	2.38E-02
65.00	1.19E-01	6.03E-02	3.54E-02	2.55E-02
70.00	1.27E-01	6.44E-02	3.77E-02	2.73E-02
75.00	1.36E-01	6.85E-02	3.99E-02	2.90E-02
80.00	1.44E-01	7.25E-02	4.21E-02	3.08E-02

Currently, utilities with ex-vessel neutron dosimetry systems are evaluating their dosimetry approximately every five fuel cycles in order to confirm/adjust their reactor vessel fluence projections. It is also recommended that the ex-vessel neutron dosimetry be evaluated prior to and following major changes in plant operation such as:

- Changing from low-leakage core loading patterns back to out-in fuel management, or changing to a mixed oxide (MOX) core.

- Significant cycle length increases (such as changing from 18-month cycles to 24-month cycles).
- Adding or removing features that alter the neutron flux shape in the peripheral fuel assemblies (such as part-length hafnium absorbers).
- Uprating plant power (or other changes that significantly alter reactor coolant temperature and density).
- Significantly altering the reactor internals geometry (such as removal of the thermal shield or the addition of titanium hydride neutron shield panels).

1 INTRODUCTION

In the assessment of the state of embrittlement of light water reactor (LWR) pressure vessels, an accurate evaluation of the neutron exposure of each of the materials comprising the beltline region of the vessel is required. In Appendix G to 10 CFR 50 (Reference 1), the beltline region is defined as:

"… 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 reactor core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."

In the case of H. B. Robinson Unit 2, the beltline region that is opposite to active fuel region is comprised of nine shell plates (three forming the lower shell course, three forming the intermediate shell course, and three forming the upper shell course); nine longitudinal welds (three in the lower shell course, three in the intermediate shell course, and three in the upper shell course) and two circumferential welds (one joining the lower shell course to the intermediate shell course and one joining the intermediate shell course to upper shell course). Each of these 20 materials must be considered in the overall embrittlement assessments of the beltline region. In addition, Section 2.2 of NUREG-15111, "Reactor Pressure Vessel Status Report," dated December 1994, states, "The NRC staff considered materials with a projected neutron fluence of greater than 1.0E+17 neutrons per square centimeter (n/cm^2) at end-of-license to experience sufficient neutron damage to be included in the beltline." Therefore, the beltline definition in 10 CFR Part 50, Appendix G is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1.0E+17 n/cm^2 (E > 1.0 MeV), and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period (Reference 2). Consequently, fluences for vessel materials well above and below the active fuel should be calculated; the additional materials to be considered are the outlet nozzle to upper shell welds, inlet nozzle to upper shell welds, and the lower shell to lower vessel head circumferential weld. Thus, plant-specific exposure assessments must include evaluations as a function of axial and azimuthal location over the entire beltline region. In this report, pressure vessel materials that are above and below the active fuel height region are referred to as "extended beltline."

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 3), describes state-of-the-art calculation and measurement procedures that are acceptable to the Nuclear Regulatory Commission (NRC) staff for determining reactor pressure vessel (RPV) fluence. Also included in Regulatory Guide 1.190 is a discussion of the steps required to qualify and validate the methodology used to determine the neutron exposure of the RPV wall. One important step in the validation process is the comparison of plant-specific neutron calculations with available measurements.

This report describes a neutron fluence analysis performed for the H. B. Robinson Unit 2 RPV beltline and extended beltline regions based on the guidance specified in Regulatory Guide 1.190. In this assessment, neutron exposures expressed in terms of fast neutron fluence (E > 1.0 MeV) and iron displacements per atom (dpa) were established for each of the materials comprising the beltline and extended beltline regions of the RPV.

The exposure experienced by the RPV was determined on a fuel-cycle-specific basis for the first 29 cycles (approximately 33.18 effective full-power years (EFPY)). Following completion of plant-specific exposure assessments encompassing fuel Cycles 1 through 29, a projection of neutron exposure was performed that was based on the cycle core designs of Cycles 27 through 29. Projections of the neutron exposure of the RPV beltline materials were performed up to 80 EFPY. Projections beyond the end of Cycle 29 assumed the current reactor power level of 2339 MWt.

H. B. Robinson Unit 2 has a plant-specific measurement database for use in validating the results of neutron transport calculations. In addition to the dosimetry included in the in-vessel capsules irradiated as an integral part of the Reactor Vessel Materials Surveillance Program (Reference 4), an Ex-Vessel Neutron Dosimetry (EVND) system designed to provide supplementary dosimetry information, was installed prior to Cycle 10. To date, four in-vessel capsules (Capsules S, V, T, and X) were installed and withdrawn as part of the Reactor Vessel Materials Surveillance Program. Capsules S, V, T, and X were withdrawn for analysis at the conclusion of Cycles 1, 3, 8, and 20, respectively. EVND sets were installed at the beginning of Cycles 10, 11, 12, 13, 14, and 16. Note that Cycle 12 EVND was not recovered from the plant. The last EVND set was installed prior to Cycle 16 and withdrawn at the conclusion of Cycle 29.

Using dosimetry evaluation methodologies that follow the guidance of Regulatory Guide 1.190, the last EVND set was analyzed. In addition, the neutron dosimetry sensor sets from previous dosimetry were re-analyzed using the current dosimetry evaluation methodology. In-vessel dosimetry evaluations, along with the ex-vessel neutron dosimetry, were used to validate the calculational models that were applied in the plant-specific neutron transport analysis.

In subsequent sections of this report, the methodologies used to perform neutron transport calculations and dosimetry evaluations are described. The results of the plant-specific transport calculations are given for the beltline and extended beltline regions of the H. B. Robinson Unit 2 RPV, and comparisons of calculations and measurements demonstrating that the transport calculations meet the requirements of Regulatory Guide 1.190 are provided.

2 NEUTRON TRANSPORT CALCULATIONS

The exposure of the H. B. Robinson Unit 2 RPV was developed based on a series of fuel-cycle-specific neutron transport calculations validated by comparison with plant-specific measurements. Measurement data were obtained from both in-vessel and ex-vessel capsule irradiations. In this section, the neutron transport methodology is discussed, and the calculated results applicable to the in-vessel surveillance capsules and the RPV beltline and extended beltline materials are presented.

2.1 METHOD OF ANALYSIS

In performing the fast neutron exposure evaluations for the H. B. Robinson Unit 2 reactor, plant-specific forward transport calculations were carried out using the following one-dimensional/two-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = \phi(r,\theta) \times \frac{\phi(r,z)}{\phi(r)}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in r,θ geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the r,θ two-dimensional calculation.

The synthesis procedure for H. B. Robinson Unit 2 was complicated by the use of partial length shield assemblies (PLSA) beginning in fuel Cycle 10. These assemblies were designed to provide a reduction in exposure to the lower circumferential weld. The PLSA are located in the three outer row positions near each of the core cardinal axes where the fluence maximum is located. The fuel pins in the PLSA contain lower enrichment fuel pellets in the top 8.5 feet of the core region and contain stainless steel rods in the lower 3.5 feet instead of fuel. The PLSA assemblies are irradiated at these locations for a total of six cycles.

To perform the synthesis with the asymmetry introduced by the PLSA, the core source is divided into two axial regions and two sets of calculations are performed. For the top region, the fluence rate is given by:

$$\phi_t(r,\theta,z) = \phi_t(r,\theta) \times \frac{\phi_t(r,z)}{\phi_t(r)}$$

In this region, the axial source is set to zero in the lower 3.5 feet of the core. In the PLSA region, the fluence rate is given by:

$$\phi_p(r,\theta,z) = \phi_p(r,\theta) \times \frac{\phi_p(r,z)}{\phi_p(r)}$$

In the PLSA region, the axial source is set to zero in the upper 8.5 feet of the core. The total fluence rate is given by the sum of the two parts:

$$\phi(r,\theta,z) = \phi_t(r,\theta,z) + \phi_p(r,\theta,z)$$

For this analysis, all of the transport calculations were carried out using a two-dimensional discrete ordinates code, DORT (Reference 5), and the BUGLE-96 cross-section library (Reference 6). The BUGLE-96 library provides a coupled 47-neutron and 20-gamma-ray group cross-section data set produced specifically for LWR applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions were treated on a fuel-cycle-specific basis.

Plan views of the r,θ model of the H. B. Robinson Unit 2 reactor geometry at the core midplane are shown in Figures 2-1 and 2-2. In Figures 2-1 and 2-2, a single octant is depicted showing an octant with and without surveillance capsules, respectively. In addition to the core, reactor internals, RPV, and primary biological shield, the models developed for these octant geometries also included explicit representations of the RPV cladding, reflective insulation, and reactor cavity liner plate.

From a neutronic standpoint, the inclusion of the surveillance capsules and associated support structure in the analytical model is significant. Since the presence of the capsules and structure has a marked impact on the magnitude of the neutron fluence rate as well as on the relative neutron and gamma-ray spectra at dosimetry locations within the capsules, a meaningful evaluation of the radiation environment internal to the capsules can be made only when these perturbation effects are properly accounted for in the analysis.

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

In developing the r,θ analytical models of the reactor geometry shown in Figures 2-1 and 2-2, nominal design dimensions were employed for the various structural components. Water temperatures and, hence, coolant density in the reactor core and downcomer regions of the reactor were taken to be representative of full power operating conditions. The reactor core was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc. The r,θ geometric mesh description of the reactor model shown in Figures 2-1 and 2-2 consisted of 161 radial by 107 azimuthal intervals. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion used in the r,θ calculations was 0.001.

A section view of the r,z model of the H. B. Robinson Unit 2 reactor is shown in Figure 2-3. The model extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an approximately 23 foot axial span from an elevation 6 feet below to 5 feet above the active fuel region. The axial extent of the model was chosen to permit the determination of the exposure of extended beltline vessel materials.

As in the case of the r,θ models, nominal design dimensions and full-power coolant densities were employed in the calculations. In this case, the homogenous core region was treated as an equivalent

cylinder with a volume equal to that of the active core zone. The stainless steel former plates located between the core baffle and core barrel regions were also explicitly included in the model. The r,z geometric mesh description of the reactor model shown in Figure 2-3 consisted of 158 radial by 222 axial intervals. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion used in the r,z calculations was 0.001.

The one-dimensional radial model used in the synthesis procedure consisted of the same 158 radial mesh intervals included in the r,z model. Thus, radial synthesis factors could be determined on a meshwise basis throughout the entire geometry.

The core power distributions used in the plant specific transport analysis for the H. B. Robinson Unit 2 reactor were generated based on fuel assembly-specific initial enrichments, beginning-of-cycle burnups, end-of-cycle burnups, axial power distributions, assembly uranium loading, and assembly pin-by-pin power distributions.

For each fuel cycle of operation, the fuel assembly specific enrichment and burnup data were used to generate the spatially dependent neutron source throughout the reactor core. This source description included the spatial variation of isotope dependent (U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242) fission spectra, neutron emission rate per fission, and energy release per fission based on the burnup history of individual fuel assemblies. These fuel assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin cartesian coordinates to the r,θ, r,z, and r spatial mesh arrays used in the DORT discrete ordinates calculations.

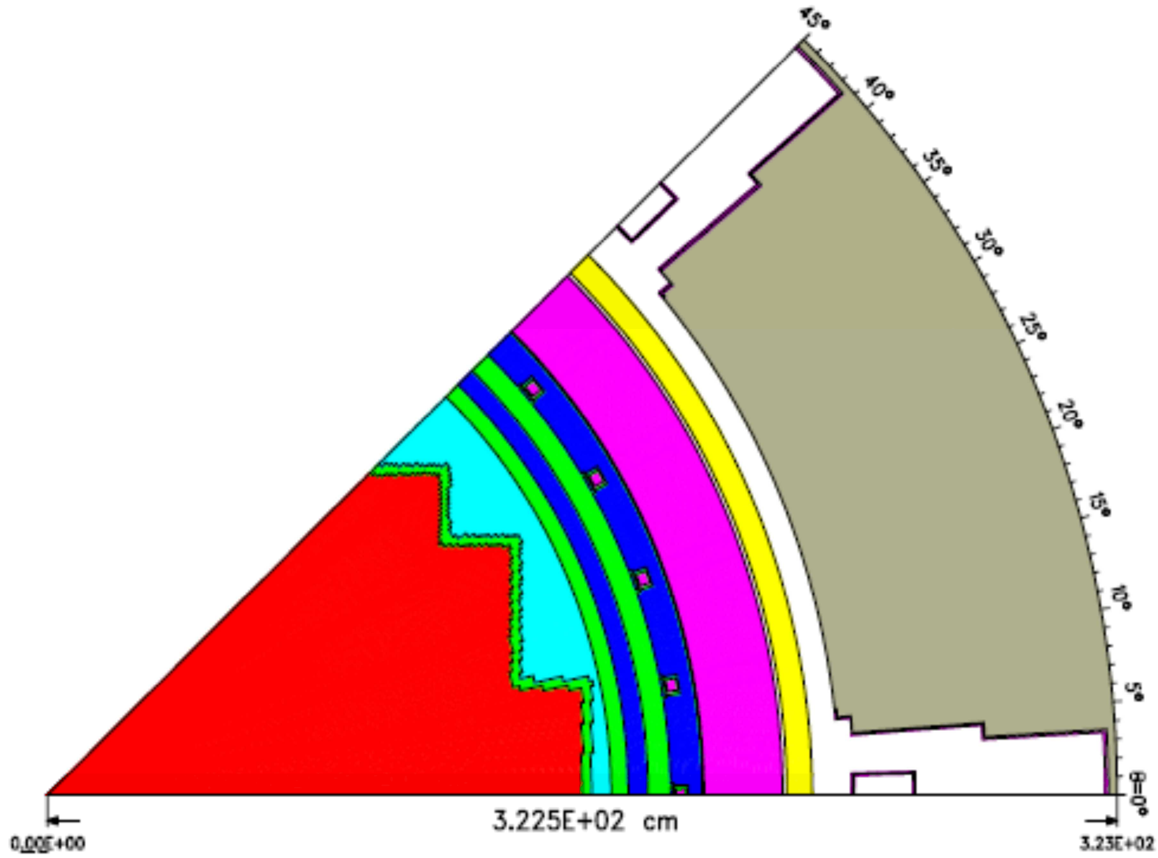

Figure 2-1. H. B. Robinson Unit 2 Reactor r,θ Plan View at Core Midplane with Surveillance Capsules

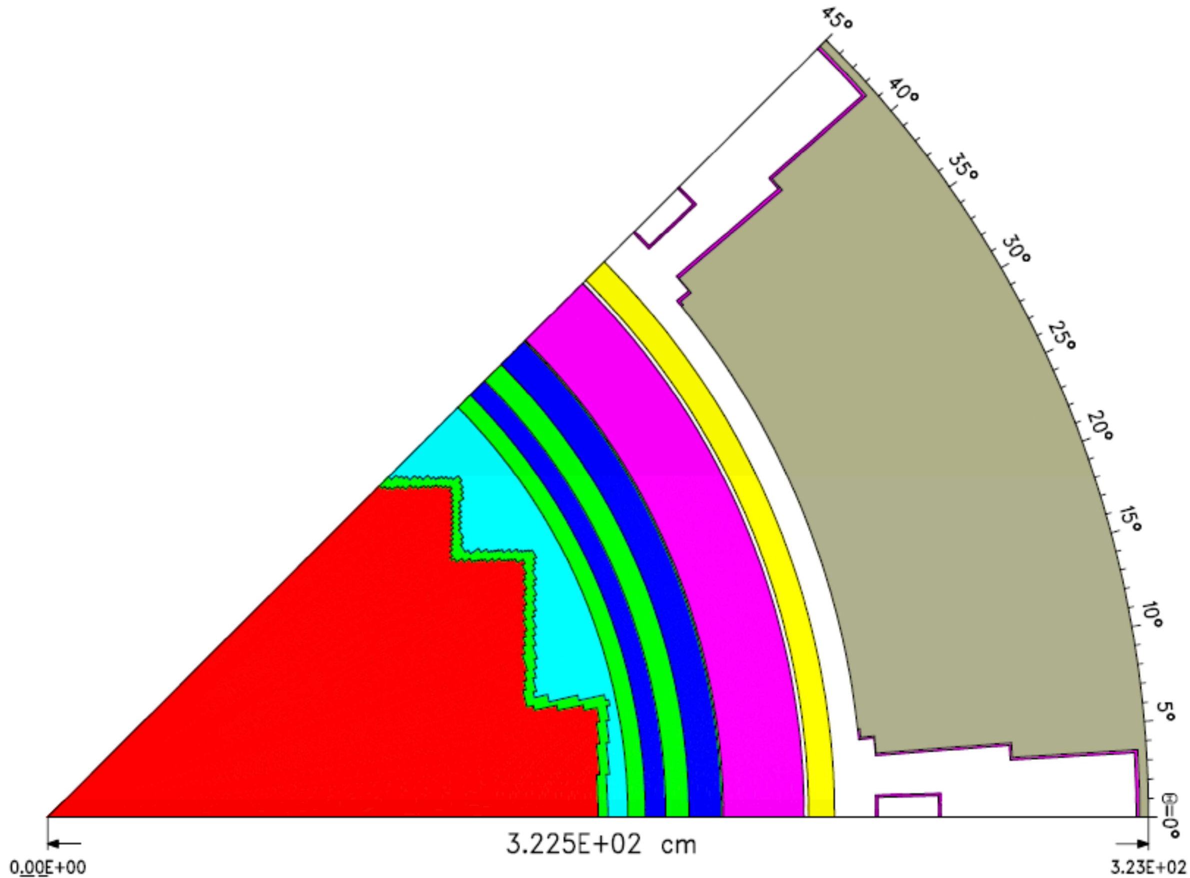

Figure 2-2. H. B. Robinson Unit 2 Reactor r,θ Plan View at Core Midplane without Surveillance Capsules

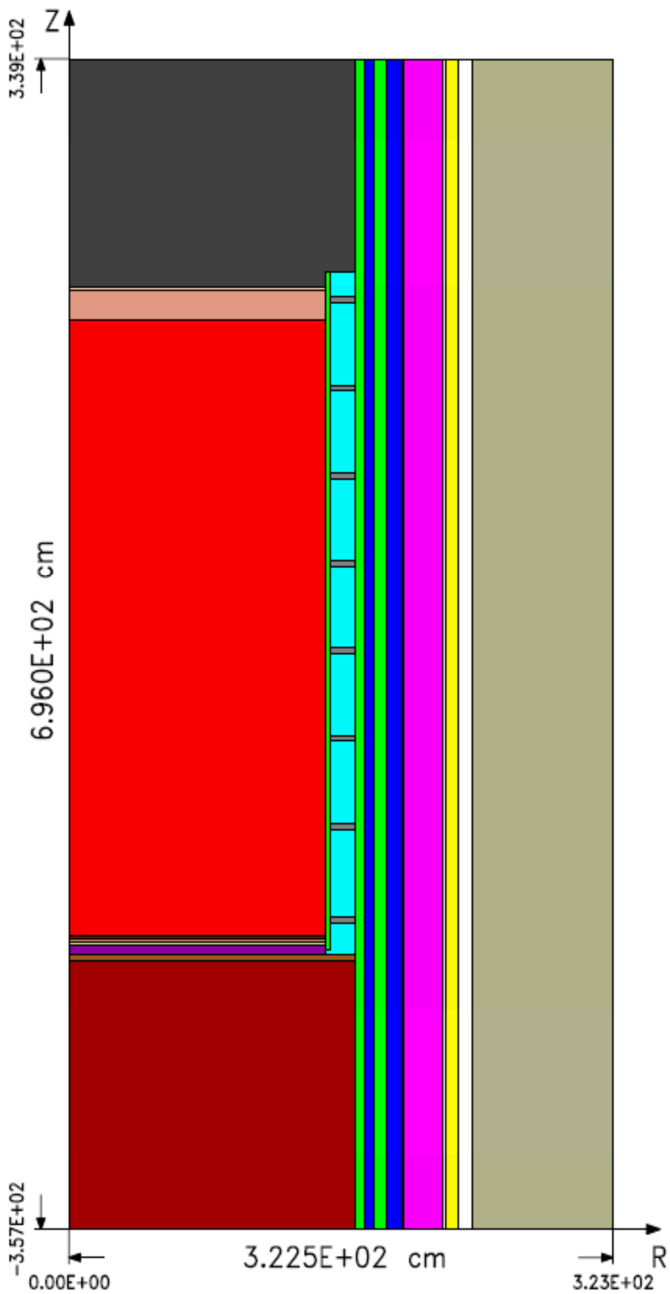

Figure 2-3. H. B. Robinson Unit 2 Reactor r,z Section View

2.2 CALCULATED NEUTRON EXPOSURE OF RPV BELTLINE MATERIALS

The calculated plant- and fuel-cycle-specific fast neutron fluence (E > 1.0 MeV) and integrated iron displacements per atom experienced by the materials comprising the beltline region of the H. B. Robinson Unit 2 RPV are given in Tables 2-1 and 2-2, respectively, for plant operation through the conclusion of Cycle 29 as well as future projections. As presented, these data represent the maximum neutron exposures at the RPV clad/base metal interface at azimuthal angles of 0°, 15°, 30°, and 45° relative to the core major axes. The data listed in Tables 2-1 and 2-2 represent a summary of the calculated neutron exposure of the RPV for all fuel cycles completed to date (Cycles 1 through 29) and future projections. The maximum neutron exposure occurs at the 0° azimuthal location.

Neutron exposure projections beyond the conclusion of Cycle 29 were based on the spatial core power distributions and associated plant operating characteristics of Cycles 27 through 29, including the current power level of 2339 MWt. Exposure projections are given for operating periods extending to 80 EFPY.

Table 2-1. Calculated Fast Neutron Fluence (E > 1.0 MeV) at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)			
			0°	15°	30°	45°
1	1.34	1.34	2.52E+18	1.15E+18	6.53E+17	4.48E+17
2	0.85	2.19	4.08E+18	1.89E+18	1.09E+18	7.34E+17
3	1.14	3.33	5.84E+18	2.73E+18	1.61E+18	1.10E+18
4	0.83	4.16	7.55E+18	3.50E+18	2.04E+18	1.39E+18
5	0.85	5.00	9.18E+18	4.25E+18	2.46E+18	1.67E+18
6	0.88	5.88	1.08E+19	5.00E+18	2.88E+18	1.95E+18
7	0.81	6.69	1.23E+19	5.72E+18	3.31E+18	2.22E+18
8	0.84	7.53	1.38E+19	6.44E+18	3.72E+18	2.49E+18
9	0.86	8.38	1.46E+19	6.89E+18	4.09E+18	2.74E+18
10	0.87	9.26	1.56E+19	7.50E+18	4.57E+18	3.02E+18
11	0.92	10.17	1.66E+19	8.15E+18	5.09E+18	3.30E+18
12	0.98	11.15	1.76E+19	8.87E+18	5.68E+18	3.60E+18
13	0.98	12.14	1.84E+19	9.52E+18	6.25E+18	3.91E+18
14	1.00	13.13	1.93E+19	1.02E+19	6.86E+18	4.22E+18
15	1.07	14.21	2.02E+19	1.10E+19	7.51E+18	4.55E+18
16	1.08	15.28	2.17E+19	1.17E+19	7.91E+18	4.86E+18
17	1.18	16.47	2.32E+19	1.24E+19	8.31E+18	5.15E+18
18	1.35	17.81	2.48E+19	1.32E+19	8.77E+18	5.46E+18
19	1.41	19.23	2.63E+19	1.40E+19	9.27E+18	5.83E+18
20	1.42	20.65	2.76E+19	1.47E+19	9.75E+18	6.21E+18
21	1.40	22.05	2.88E+19	1.54E+19	1.02E+19	6.58E+18
22	1.44	23.49	3.06E+19	1.62E+19	1.06E+19	6.85E+18
23	1.31	24.80	3.20E+19	1.69E+19	1.09E+19	7.12E+18
24	1.45	26.24	3.34E+19	1.75E+19	1.13E+19	7.42E+18
25	1.37	27.62	3.46E+19	1.81E+19	1.16E+19	7.70E+18
26	1.34	28.95	3.54E+19	1.86E+19	1.19E+19	7.99E+18
27	1.35	30.31	3.64E+19	1.92E+19	1.22E+19	8.27E+18
28	1.46	31.77	3.83E+19	2.00E+19	1.27E+19	8.59E+18
29	1.41	33.18	3.97E+19	2.07E+19	1.30E+19	8.90E+18
Future		35.00	4.16E+19	2.16E+19	1.35E+19	9.29E+18
Future		40.00	4.67E+19	2.41E+19	1.49E+19	1.04E+19
Future		45.00	5.18E+19	2.65E+19	1.62E+19	1.14E+19
Future		50.00	5.69E+19	2.90E+19	1.76E+19	1.25E+19
Future		55.00	6.20E+19	3.14E+19	1.90E+19	1.36E+19
Future		60.00	6.71E+19	3.39E+19	2.03E+19	1.46E+19
Future		65.00	7.22E+19	3.64E+19	2.17E+19	1.57E+19
Future		70.00	7.74E+19	3.88E+19	2.31E+19	1.68E+19
Future		75.00	8.25E+19	4.13E+19	2.44E+19	1.78E+19
Future		80.00	8.76E+19	4.37E+19	2.58E+19	1.89E+19

Table 2-2. Calculated dpa at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	dpa			
			0°	15°	30°	45°
1	1.34	1.34	4.14E-03	1.91E-03	1.07E-03	7.30E-04
2	0.85	2.19	6.71E-03	3.15E-03	1.78E-03	1.20E-03
3	1.14	3.33	9.62E-03	4.54E-03	2.64E-03	1.79E-03
4	0.83	4.16	1.24E-02	5.82E-03	3.33E-03	2.27E-03
5	0.85	5.00	1.51E-02	7.08E-03	4.02E-03	2.72E-03
6	0.88	5.88	1.78E-02	8.32E-03	4.72E-03	3.18E-03
7	0.81	6.69	2.03E-02	9.53E-03	5.41E-03	3.63E-03
8	0.84	7.53	2.28E-02	1.07E-02	6.08E-03	4.06E-03
9	0.86	8.38	2.41E-02	1.15E-02	6.68E-03	4.47E-03
10	0.87	9.26	2.57E-02	1.25E-02	7.46E-03	4.93E-03
11	0.92	10.17	2.73E-02	1.35E-02	8.32E-03	5.39E-03
12	0.98	11.15	2.89E-02	1.47E-02	9.28E-03	5.88E-03
13	0.98	12.14	3.03E-02	1.58E-02	1.02E-02	6.38E-03
14	1.00	13.13	3.17E-02	1.69E-02	1.12E-02	6.89E-03
15	1.07	14.21	3.32E-02	1.82E-02	1.23E-02	7.43E-03
16	1.08	15.28	3.57E-02	1.94E-02	1.29E-02	7.93E-03
17	1.18	16.47	3.83E-02	2.06E-02	1.36E-02	8.41E-03
18	1.35	17.81	4.08E-02	2.19E-02	1.43E-02	8.91E-03
19	1.41	19.23	4.32E-02	2.32E-02	1.51E-02	9.52E-03
20	1.42	20.65	4.54E-02	2.44E-02	1.59E-02	1.02E-02
21	1.40	22.05	4.74E-02	2.56E-02	1.67E-02	1.08E-02
22	1.44	23.49	5.04E-02	2.69E-02	1.73E-02	1.12E-02
23	1.31	24.80	5.27E-02	2.80E-02	1.78E-02	1.16E-02
24	1.45	26.24	5.49E-02	2.91E-02	1.84E-02	1.21E-02
25	1.37	27.62	5.68E-02	3.01E-02	1.89E-02	1.25E-02
26	1.34	28.95	5.82E-02	3.09E-02	1.95E-02	1.30E-02
27	1.35	30.31	5.99E-02	3.18E-02	2.00E-02	1.35E-02
28	1.46	31.77	6.30E-02	3.32E-02	2.07E-02	1.40E-02
29	1.41	33.18	6.53E-02	3.43E-02	2.13E-02	1.45E-02
Future		35.00	6.83E-02	3.58E-02	2.21E-02	1.51E-02
Future		40.00	7.67E-02	3.99E-02	2.43E-02	1.68E-02
Future		45.00	8.51E-02	4.40E-02	2.65E-02	1.86E-02
Future		50.00	9.35E-02	4.80E-02	2.88E-02	2.03E-02
Future		55.00	1.02E-01	5.21E-02	3.10E-02	2.21E-02
Future		60.00	1.10E-01	5.62E-02	3.32E-02	2.38E-02
Future		65.00	1.19E-01	6.03E-02	3.54E-02	2.55E-02
Future		70.00	1.27E-01	6.44E-02	3.77E-02	2.73E-02
Future		75.00	1.36E-01	6.85E-02	3.99E-02	2.90E-02
Future		80.00	1.44E-01	7.25E-02	4.21E-02	3.08E-02

2.3 CALCULATED NEUTRON EXPOSURE OF IN-VESSEL SURVEILLANCE CAPSULES

Eight irradiation capsules attached to the thermal shield were originally included in the H. B. Robinson Unit 2 reactor design to constitute the reactor pressure vessel surveillance program. The capsules were located at azimuthal angles of 30°, 150° (30° from the core cardinal axis), 40°, 50°, 230° (40° from the core cardinal axis), 280° (10° from the core cardinal axis), 270° (0° from the core cardinal axis), and 290° (20° from the core cardinal axis). The irradiation history of each of these eight in-vessel surveillance capsules is given in Table 2-3.

Table 2-3. Irradiation History of In-Vessel Surveillance Capsules

Capsule	Azimuthal Location (With Respect to Core Cardinal Axes)(1)	Status
S	10º	Withdrawn EOC 1
V	20º	Withdrawn EOC 3
T	0º	Withdrawn EOC 8
X(2)	40º, 0º	Withdrawn EOC 20
U(3)	30º, 10º	In Reactor
Y(4)	30º, 20°	In Reactor
W(5)	40º, 0°	In Reactor
Z(6)	40º	Withdrawn EOC 1

Notes:
1. Capsule relocation information was obtained from Reference 7 and H. B. Robinson Unit 2 Updated Final Safety Analysis Report (FSAR), Rev. 25, Section 5.3.
2. Capsule X was relocated from 50° (40° with respect to core cardinal axes) to 270° (0° with respect to core cardinal axes) at EOC 8.
3. Capsule U was relocated from 30° to 280° (10° with respect to core cardinal axes) at EOC 8.
4. Capsule Y was relocated from 150° (30° with respect to core cardinal axes) to 290° (20° with respect to core cardinal axes) at EOC 27.
5. Capsule W was relocated from 40° to 270° (0° with respect to core cardinal axes) at EOC 27.
6. Capsule Z was not analyzed.

Results of the discrete ordinates analyses pertinent to each of these in-vessel surveillance capsules are provided in Tables 2-4 through 2-7. In Tables 2-4 and 2-5, the calculated fuel-cycle-specific fast neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate (dpa/s), respectively, are provided at the geometric center of the five capsule positions (0°, corresponding to the 270° location, 10° corresponding to the 280° location, 20° corresponding to the 290° location, 30° corresponding to the 30° and 150° locations, and 40° corresponding to the 40°, 50°, and 230° locations). In Tables 2-6 and 2-7, fast neutron

fluence (E > 1.0 MeV) and iron displacements per atom, respectively, are provided at the geometric center of the capsules.

In Table 2-8, lead factors associated with each of the H. B. Robinson Unit 2 in-vessel surveillance capsules are listed. The lead factor applicable to any given operating period is defined as the ratio of the fast neutron (E > 1.0 MeV) fluence accrued by the test specimens contained in the surveillance capsule to the corresponding fast neutron fluence (E > 1.0 MeV) accrued at the location of maximum exposure at the reactor pressure vessel clad/base metal interface. From data provided in Section 2.2 of this report, it is noted that the maximum exposure of the H. B. Robinson Unit 2 reactor pressure vessel after the first 29 cycles of operation occurs at the 0° azimuth. The lead factors for capsules that have been withdrawn from the reactor represent the conditions at the time of withdrawal. For the capsules that are in the reactor, lead factors were calculated at the end of the last completed fuel cycle (Cycle 29).

Table 2-4. Calculated Fast Neutron Fluence Rate (E > 1.0 MeV) at the Geometric Center of the Surveillance Capsules

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)				
			0°	10°	20°	30°	40°
1	1.34	1.34	1.64E+11	1.14E+11	5.10E+10	3.92E+10	2.72E+10
2	0.85	2.19	1.63E+11	1.17E+11	5.40E+10	4.18E+10	2.81E+10
3	1.14	3.33	1.39E+11	9.63E+10	4.73E+10	3.80E+10	2.66E+10
4	0.83	4.16	1.84E+11	1.29E+11	5.50E+10	4.20E+10	2.92E+10
5	0.85	5.00	1.69E+11	1.20E+11	5.32E+10	4.05E+10	2.70E+10
6	0.88	5.88	1.63E+11	1.15E+11	5.16E+10	3.96E+10	2.65E+10
7	0.81	6.69	1.64E+11	1.17E+11	5.54E+10	4.29E+10	2.82E+10
8	0.84	7.53	1.67E+11	1.18E+11	5.31E+10	4.07E+10	2.69E+10
9	0.86	8.38	8.34E+10	6.10E+10	4.09E+10	3.67E+10	2.52E+10
10	0.87	9.26	1.04E+11	7.85E+10	5.53E+10	4.63E+10	2.74E+10
11	0.92	10.17	9.51E+10	7.54E+10	5.76E+10	4.80E+10	2.64E+10
12	0.98	11.15	8.93E+10	7.31E+10	6.01E+10	4.94E+10	2.63E+10
13	0.98	12.14	7.68E+10	6.45E+10	5.64E+10	4.80E+10	2.65E+10
14	1.00	13.13	7.61E+10	6.59E+10	6.06E+10	5.09E+10	2.70E+10
15	1.07	14.21	7.17E+10	6.32E+10	6.01E+10	5.03E+10	2.62E+10
16	1.08	15.28	1.29E+11	9.08E+10	4.03E+10	3.05E+10	2.38E+10
17	1.18	16.47	1.13E+11	8.16E+10	3.85E+10	2.79E+10	2.03E+10
18	1.35	17.81	1.01E+11	7.31E+10	3.69E+10	2.78E+10	1.90E+10
19	1.41	19.23	9.18E+10	6.73E+10	3.56E+10	2.84E+10	2.17E+10
20	1.42	20.65	8.01E+10	5.97E+10	3.34E+10	2.75E+10	2.19E+10
21	1.40	22.05	7.47E+10	5.63E+10	3.33E+10	2.73E+10	2.12E+10
22	1.44	23.49	1.13E+11	7.81E+10	2.94E+10	1.93E+10	1.52E+10
23	1.31	24.80	9.38E+10	6.65E+10	2.83E+10	1.97E+10	1.61E+10
24	1.45	26.24	8.04E+10	5.79E+10	2.72E+10	2.02E+10	1.63E+10
25	1.37	27.62	7.54E+10	5.51E+10	2.68E+10	1.94E+10	1.55E+10
26	1.34	28.95	5.44E+10	4.23E+10	2.45E+10	1.92E+10	1.57E+10
27	1.35	30.31	6.47E+10	4.80E+10	2.57E+10	1.96E+10	1.62E+10
28	1.46	31.77	1.13E+11	7.84E+10	3.25E+10	2.36E+10	1.73E+10
29	1.41	33.18	8.83E+10	6.35E+10	2.81E+10	2.05E+10	1.61E+10
Future			8.86E+10	6.33E+10	2.87E+10	2.12E+10	1.65E+10

Table 2-5. Calculated Iron Displacements per Atom per Second at the Geometric Center of the Surveillance Capsules

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	dpa/s				
			0°	10°	20°	30°	40°
1	1.34	1.34	2.78E-10	1.94E-10	8.37E-11	6.44E-11	4.41E-11
2	0.85	2.19	2.76E-10	2.00E-10	8.86E-11	6.86E-11	4.57E-11
3	1.14	3.33	2.36E-10	1.65E-10	7.75E-11	6.25E-11	4.32E-11
4	0.83	4.16	3.12E-10	2.20E-10	9.03E-11	6.90E-11	4.73E-11
5	0.85	5.00	2.87E-10	2.05E-10	8.74E-11	6.65E-11	4.39E-11
6	0.88	5.88	2.76E-10	1.96E-10	8.47E-11	6.50E-11	4.30E-11
7	0.81	6.69	2.78E-10	2.01E-10	9.09E-11	7.05E-11	4.58E-11
8	0.84	7.53	2.82E-10	2.01E-10	8.70E-11	6.68E-11	4.37E-11
9	0.86	8.38	1.40E-10	1.03E-10	6.63E-11	5.99E-11	4.06E-11
10	0.87	9.26	1.75E-10	1.33E-10	9.03E-11	7.60E-11	4.45E-11
11	0.92	10.17	1.60E-10	1.28E-10	9.40E-11	7.89E-11	4.28E-11
12	0.98	11.15	1.50E-10	1.24E-10	9.80E-11	8.12E-11	4.28E-11
13	0.98	12.14	1.29E-10	1.09E-10	9.20E-11	7.90E-11	4.30E-11
14	1.00	13.13	1.28E-10	1.12E-10	9.89E-11	8.38E-11	4.38E-11
15	1.07	14.21	1.21E-10	1.07E-10	9.80E-11	8.28E-11	4.25E-11
16	1.08	15.28	2.17E-10	1.55E-10	6.60E-11	4.99E-11	3.85E-11
17	1.18	16.47	1.91E-10	1.39E-10	6.30E-11	4.57E-11	3.29E-11
18	1.35	17.81	1.70E-10	1.24E-10	6.03E-11	4.56E-11	3.08E-11
19	1.41	19.23	1.55E-10	1.14E-10	5.81E-11	4.64E-11	3.51E-11
20	1.42	20.65	1.35E-10	1.01E-10	5.45E-11	4.50E-11	3.55E-11
21	1.40	22.05	1.25E-10	9.52E-11	5.44E-11	4.47E-11	3.44E-11
22	1.44	23.49	1.91E-10	1.33E-10	4.82E-11	3.16E-11	2.46E-11
23	1.31	24.80	1.58E-10	1.13E-10	4.62E-11	3.22E-11	2.61E-11
24	1.45	26.24	1.35E-10	9.84E-11	4.44E-11	3.30E-11	2.64E-11
25	1.37	27.62	1.27E-10	9.34E-11	4.38E-11	3.16E-11	2.51E-11
26	1.34	28.95	9.15E-11	7.16E-11	3.99E-11	3.14E-11	2.54E-11
27	1.35	30.31	1.09E-10	8.13E-11	4.19E-11	3.20E-11	2.61E-11
28	1.46	31.77	1.91E-10	1.34E-10	5.32E-11	3.86E-11	2.80E-11
29	1.41	33.18	1.49E-10	1.08E-10	4.60E-11	3.35E-11	2.61E-11
Future			1.49E-10	1.08E-10	4.70E-11	3.47E-11	2.68E-11

Table 2-6. Calculated Fast Neutron Fluence (E > 1.0 MeV) at the Geometric Center of the Surveillance Capsules

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)				
			0°	10°	20°	30°	40°
1	1.34	1.34	6.92E+18	4.79E+18	2.15E+18	1.65E+18	1.15E+18
2	0.85	2.19	1.13E+19	7.93E+18	3.60E+18	2.77E+18	1.90E+18
3	1.14	3.33	1.63E+19	1.14E+19	5.30E+18	4.14E+18	2.86E+18
4	0.83	4.16	2.11E+19	1.48E+19	6.74E+18	5.24E+18	3.62E+18
5	0.85	5.00	2.56E+19	1.80E+19	8.16E+18	6.32E+18	4.34E+18
6	0.88	5.88	3.01E+19	2.11E+19	9.59E+18	7.42E+18	5.08E+18
7	0.81	6.69	3.43E+19	2.41E+19	1.10E+19	8.52E+18	5.80E+18
8	0.84	7.53	3.87E+19	2.73E+19	1.24E+19	9.59E+18	6.51E+18
9	0.86	8.38	4.10E+19	2.89E+19	1.35E+19	1.06E+19	7.19E+18
10	0.87	9.26	4.38E+19	3.11E+19	1.50E+19	1.19E+19	7.94E+18
11	0.92	10.17	4.66E+19	3.32E+19	1.67E+19	1.32E+19	8.71E+18
12	0.98	11.15	4.93E+19	3.55E+19	1.86E+19	1.48E+19	9.52E+18
13	0.98	12.14	5.17E+19	3.75E+19	2.03E+19	1.63E+19	1.04E+19
14	1.00	13.13	5.41E+19	3.96E+19	2.22E+19	1.79E+19	1.12E+19
15	1.07	14.21	5.65E+19	4.17E+19	2.43E+19	1.96E+19	1.21E+19
16	1.08	15.28	6.09E+19	4.48E+19	2.56E+19	2.06E+19	1.29E+19
17	1.18	16.47	6.51E+19	4.79E+19	2.71E+19	2.16E+19	1.37E+19
18	1.35	17.81	6.94E+19	5.10E+19	2.86E+19	2.28E+19	1.45E+19
19	1.41	19.23	7.35E+19	5.40E+19	3.02E+19	2.41E+19	1.54E+19
20	1.42	20.65	7.71E+19	5.66E+19	3.17E+19	2.53E+19	1.64E+19
21	1.40	22.05	8.04E+19	5.91E+19	3.32E+19	2.65E+19	1.73E+19
22	1.44	23.49	8.55E+19	6.27E+19	3.45E+19	2.74E+19	1.80E+19
23	1.31	24.80	8.94E+19	6.54E+19	3.57E+19	2.82E+19	1.87E+19
24	1.45	26.24	9.31E+19	6.81E+19	3.69E+19	2.92E+19	1.95E+19
25	1.37	27.62	9.63E+19	7.04E+19	3.81E+19	3.00E+19	2.01E+19
26	1.34	28.95	9.86E+19	7.22E+19	3.91E+19	3.08E+19	2.08E+19
27	1.35	30.31	1.01E+20	7.43E+19	4.02E+19	3.16E+19	2.15E+19
28	1.46	31.77	1.07E+20	7.79E+19	4.17E+19	3.27E+19	2.23E+19
29	1.41	33.18	1.11E+20	8.07E+19	4.30E+19	3.36E+19	2.30E+19
Future		35.00	1.16E+20	8.44E+19	4.46E+19	3.48E+19	2.39E+19
Future		40.00	1.30E+20	9.43E+19	4.91E+19	3.82E+19	2.65E+19
Future		45.00	1.44E+20	1.04E+20	5.37E+19	4.15E+19	2.92E+19
Future		50.00	1.58E+20	1.14E+20	5.82E+19	4.49E+19	3.18E+19
Future		55.00	1.72E+20	1.24E+20	6.27E+19	4.82E+19	3.44E+19
Future		60.00	1.85E+20	1.34E+20	6.73E+19	5.16E+19	3.70E+19
Future		65.00	1.99E+20	1.44E+20	7.18E+19	5.49E+19	3.96E+19
Future		70.00	2.13E+20	1.54E+20	7.64E+19	5.83E+19	4.22E+19
Future		75.00	2.27E+20	1.64E+20	8.09E+19	6.16E+19	4.48E+19
Future		80.00	2.41E+20	1.74E+20	8.54E+19	6.50E+19	4.74E+19

Table 2-7. Calculated Iron Displacements per Atom at the Geometric Center of the Surveillance Capsules

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	dpa				
			0°	10°	20°	30°	40°
1	1.34	1.34	1.17E-02	8.19E-03	3.53E-03	2.71E-03	1.86E-03
2	0.85	2.19	1.92E-02	1.36E-02	5.91E-03	4.56E-03	3.09E-03
3	1.14	3.33	2.76E-02	1.95E-02	8.70E-03	6.81E-03	4.64E-03
4	0.83	4.16	3.58E-02	2.52E-02	1.11E-02	8.61E-03	5.88E-03
5	0.85	5.00	4.35E-02	3.07E-02	1.34E-02	1.04E-02	7.05E-03
6	0.88	5.88	5.11E-02	3.62E-02	1.58E-02	1.22E-02	8.24E-03
7	0.81	6.69	5.82E-02	4.13E-02	1.81E-02	1.40E-02	9.41E-03
8	0.84	7.53	6.56E-02	4.66E-02	2.04E-02	1.58E-02	1.06E-02
9	0.86	8.38	6.94E-02	4.94E-02	2.22E-02	1.74E-02	1.17E-02
10	0.87	9.26	7.42E-02	5.30E-02	2.46E-02	1.95E-02	1.29E-02
11	0.92	10.17	7.89E-02	5.67E-02	2.74E-02	2.17E-02	1.41E-02
12	0.98	11.15	8.35E-02	6.06E-02	3.04E-02	2.43E-02	1.55E-02
13	0.98	12.14	8.75E-02	6.39E-02	3.33E-02	2.67E-02	1.68E-02
14	1.00	13.13	9.16E-02	6.74E-02	3.64E-02	2.94E-02	1.82E-02
15	1.07	14.21	9.56E-02	7.11E-02	3.97E-02	3.22E-02	1.96E-02
16	1.08	15.28	1.03E-01	7.63E-02	4.19E-02	3.39E-02	2.09E-02
17	1.18	16.47	1.10E-01	8.15E-02	4.43E-02	3.56E-02	2.21E-02
18	1.35	17.81	1.17E-01	8.68E-02	4.68E-02	3.75E-02	2.34E-02
19	1.41	19.23	1.24E-01	9.19E-02	4.94E-02	3.96E-02	2.50E-02
20	1.42	20.65	1.30E-01	9.64E-02	5.19E-02	4.16E-02	2.66E-02
21	1.40	22.05	1.36E-01	1.01E-01	5.43E-02	4.36E-02	2.81E-02
22	1.44	23.49	1.45E-01	1.07E-01	5.65E-02	4.50E-02	2.92E-02
23	1.31	24.80	1.51E-01	1.11E-01	5.84E-02	4.63E-02	3.03E-02
24	1.45	26.24	1.57E-01	1.16E-01	6.04E-02	4.78E-02	3.15E-02
25	1.37	27.62	1.63E-01	1.20E-01	6.23E-02	4.92E-02	3.26E-02
26	1.34	28.95	1.67E-01	1.23E-01	6.40E-02	5.05E-02	3.37E-02
27	1.35	30.31	1.71E-01	1.26E-01	6.58E-02	5.19E-02	3.48E-02
28	1.46	31.77	1.80E-01	1.33E-01	6.82E-02	5.37E-02	3.61E-02
29	1.41	33.18	1.87E-01	1.37E-01	7.03E-02	5.52E-02	3.73E-02
Future		35.00	1.95E-01	1.43E-01	7.30E-02	5.72E-02	3.88E-02
Future		40.00	2.19E-01	1.60E-01	8.04E-02	6.26E-02	4.30E-02
Future		45.00	2.42E-01	1.77E-01	8.78E-02	6.81E-02	4.72E-02
Future		50.00	2.66E-01	1.94E-01	9.52E-02	7.36E-02	5.15E-02
Future		55.00	2.90E-01	2.11E-01	1.03E-01	7.91E-02	5.57E-02
Future		60.00	3.13E-01	2.28E-01	1.10E-01	8.46E-02	5.99E-02
Future		65.00	3.37E-01	2.45E-01	1.17E-01	9.00E-02	6.41E-02
Future		70.00	3.60E-01	2.62E-01	1.25E-01	9.55E-02	6.83E-02
Future		75.00	3.84E-01	2.79E-01	1.32E-01	1.01E-01	7.26E-02
Future		80.00	4.08E-01	2.96E-01	1.40E-01	1.06E-01	7.68E-02

Table 2-8. Lead Factors for H. B. Robinson Unit 2 Surveillance Capsules

Capsule	Azimuthal Location (°)	Irradiation Cycle(s)	Capsule Fast Neutron Fluence (E > 1.0 MeV) (n/cm^2)	Lead Factor
S	10	1	4.79E+18	1.90
V	20	1-3	5.30E+18	0.91
T	0	1-8	3.87E+19	2.80
X	40,0	1-20	4.49E+19	1.63
Z(1)	40	1	1.15E+18	0.46
U(2)	30, 10	1-29	6.31E+19	1.59
Y(2)	30, 20	1-29	3.44E+19	0.87
W(2)	40, 0	1-29	3.06E+19	0.77

Notes:

1. Capsule Z was not analyzed.
2. Capsule is in the reactor.

2.4 PRESSURE VESSEL MATERIAL EXPOSURES

Pressure vessel material maximum exposures in terms of fast neutron fluence (E > 1.0 MeV) and dpa are given in Tables 2-9 through 2-15.

Table 2-9. Outlet Nozzle to Upper Shell Welds – Lowest Extent - Exposure

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (E > 1.0 MeV) (n/cm^2)			dpa		
			10°	250°	130°	10°	250°	130°
1	1.34	1.34	5.76E+15	2.84E+15	1.54E+15	1.19E-05	5.82E-06	3.13E-06
2	0.85	2.19	1.08E+16	5.40E+15	2.91E+15	2.23E-05	1.10E-05	5.89E-06
3	1.14	3.33	1.52E+16	7.73E+15	4.25E+15	3.13E-05	1.58E-05	8.62E-06
4	0.83	4.16	2.07E+16	1.03E+16	5.65E+15	4.26E-05	2.11E-05	1.14E-05
5	0.85	5.00	2.58E+16	1.28E+16	6.95E+15	5.31E-05	2.62E-05	1.41E-05
6	0.88	5.88	3.10E+16	1.54E+16	8.30E+15	6.36E-05	3.14E-05	1.68E-05
7	0.81	6.69	3.58E+16	1.79E+16	9.58E+15	7.33E-05	3.64E-05	1.93E-05
8	0.84	7.53	4.02E+16	2.01E+16	1.07E+16	8.25E-05	4.09E-05	2.17E-05
9	0.86	8.38	4.24E+16	2.17E+16	1.17E+16	8.72E-05	4.42E-05	2.38E-05
10	0.87	9.26	4.46E+16	2.33E+16	1.26E+16	9.17E-05	4.75E-05	2.55E-05
11	0.92	10.17	4.66E+16	2.49E+16	1.34E+16	9.59E-05	5.09E-05	2.71E-05
12	0.98	11.15	4.83E+16	2.63E+16	1.40E+16	9.96E-05	5.40E-05	2.86E-05
13	0.98	12.14	4.99E+16	2.78E+16	1.48E+16	1.03E-04	5.71E-05	3.01E-05
14	1.00	13.13	5.15E+16	2.93E+16	1.55E+16	1.07E-04	6.04E-05	3.17E-05
15	1.07	14.21	5.32E+16	3.10E+16	1.63E+16	1.10E-04	6.40E-05	3.34E-05
16	1.08	15.28	5.57E+16	3.22E+16	1.70E+16	1.16E-04	6.66E-05	3.49E-05
17	1.18	16.47	5.85E+16	3.37E+16	1.78E+16	1.22E-04	6.97E-05	3.66E-05
18	1.35	17.81	6.14E+16	3.53E+16	1.86E+16	1.28E-04	7.30E-05	3.83E-05
19	1.41	19.23	6.44E+16	3.70E+16	1.97E+16	1.34E-04	7.67E-05	4.06E-05
20	1.42	20.65	6.73E+16	3.88E+16	2.09E+16	1.40E-04	8.03E-05	4.30E-05
21	1.40	22.05	6.99E+16	4.04E+16	2.20E+16	1.46E-04	8.38E-05	4.52E-05
22	1.44	23.49	7.38E+16	4.21E+16	2.28E+16	1.54E-04	8.72E-05	4.70E-05
23	1.31	24.80	7.70E+16	4.36E+16	2.37E+16	1.61E-04	9.03E-05	4.88E-05
24	1.45	26.24	8.04E+16	4.53E+16	2.47E+16	1.67E-04	9.38E-05	5.09E-05
25	1.37	27.62	8.33E+16	4.68E+16	2.56E+16	1.73E-04	9.69E-05	5.28E-05
26	1.34	28.95	8.56E+16	4.83E+16	2.66E+16	1.78E-04	1.00E-04	5.47E-05
27	1.35	30.31	8.84E+16	4.99E+16	2.76E+16	1.84E-04	1.03E-04	5.68E-05
28	1.46	31.77	9.23E+16	5.17E+16	2.86E+16	1.92E-04	1.07E-04	5.88E-05
29	1.41	33.18	9.56E+16	5.33E+16	2.95E+16	1.99E-04	1.10E-04	6.07E-05
Future		35.00	9.99E+16	5.54E+16	3.08E+16	2.08E-04	1.15E-04	6.33E-05
Future		40.00	1.12E+17	6.13E+16	3.43E+16	2.32E-04	1.27E-04	7.04E-05
Future		45.00	1.23E+17	6.72E+16	3.78E+16	2.57E-04	1.39E-04	7.75E-05
Future		50.00	1.35E+17	7.31E+16	4.12E+16	2.81E-04	1.51E-04	8.46E-05
Future		55.00	1.47E+17	7.90E+16	4.47E+16	3.06E-04	1.64E-04	9.17E-05
Future		60.00	1.59E+17	8.50E+16	4.82E+16	3.30E-04	1.76E-04	9.88E-05
Future		65.00	1.70E+17	9.09E+16	5.16E+16	3.54E-04	1.88E-04	1.06E-04
Future		70.00	1.82E+17	9.68E+16	5.51E+16	3.79E-04	2.00E-04	1.13E-04
Future		75.00	1.94E+17	1.03E+17	5.86E+16	4.03E-04	2.12E-04	1.20E-04
Future		80.00	2.05E+17	1.09E+17	6.21E+16	4.28E-04	2.25E-04	1.27E-04

Table 2-10. Inlet Nozzle to Upper Shell Welds – Lowest Extent - Exposure

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (E > 1.0 MeV) (n/cm^2)			dpa		
			10°	200°	320°	10°	200°	320°
1	1.34	1.34	4.85E+15	2.39E+15	1.30E+15	1.02E-05	4.98E-06	2.68E-06
2	0.85	2.19	9.10E+15	4.54E+15	2.45E+15	1.91E-05	9.45E-06	5.04E-06
3	1.14	3.33	1.28E+16	6.51E+15	3.57E+15	2.68E-05	1.36E-05	7.38E-06
4	0.83	4.16	1.74E+16	8.69E+15	4.76E+15	3.65E-05	1.81E-05	9.80E-06
5	0.85	5.00	2.17E+16	1.08E+16	5.85E+15	4.55E-05	2.25E-05	1.20E-05
6	0.88	5.88	2.61E+16	1.30E+16	6.98E+15	5.45E-05	2.69E-05	1.44E-05
7	0.81	6.69	3.01E+16	1.50E+16	8.07E+15	6.28E-05	3.11E-05	1.66E-05
8	0.84	7.53	3.38E+16	1.69E+16	9.02E+15	7.06E-05	3.50E-05	1.85E-05
9	0.86	8.38	3.57E+16	1.82E+16	9.87E+15	7.46E-05	3.78E-05	2.03E-05
10	0.87	9.26	3.75E+16	1.96E+16	1.06E+16	7.85E-05	4.07E-05	2.18E-05
11	0.92	10.17	3.92E+16	2.09E+16	1.12E+16	8.21E-05	4.35E-05	2.32E-05
12	0.98	11.15	4.06E+16	2.22E+16	1.18E+16	8.53E-05	4.63E-05	2.45E-05
13	0.98	12.14	4.20E+16	2.34E+16	1.24E+16	8.83E-05	4.89E-05	2.58E-05
14	1.00	13.13	4.33E+16	2.47E+16	1.30E+16	9.13E-05	5.18E-05	2.72E-05
15	1.07	14.21	4.48E+16	2.61E+16	1.37E+16	9.45E-05	5.48E-05	2.86E-05
16	1.08	15.28	4.69E+16	2.71E+16	1.43E+16	9.91E-05	5.71E-05	2.99E-05
17	1.18	16.47	4.92E+16	2.83E+16	1.50E+16	1.04E-04	5.97E-05	3.13E-05
18	1.35	17.81	5.17E+16	2.97E+16	1.57E+16	1.10E-04	6.26E-05	3.29E-05
19	1.41	19.23	5.43E+16	3.12E+16	1.66E+16	1.15E-04	6.57E-05	3.48E-05
20	1.42	20.65	5.67E+16	3.26E+16	1.76E+16	1.20E-04	6.89E-05	3.69E-05
21	1.40	22.05	5.89E+16	3.40E+16	1.85E+16	1.25E-04	7.18E-05	3.88E-05
22	1.44	23.49	6.22E+16	3.54E+16	1.92E+16	1.32E-04	7.48E-05	4.03E-05
23	1.31	24.80	6.49E+16	3.67E+16	2.00E+16	1.38E-04	7.74E-05	4.18E-05
24	1.45	26.24	6.77E+16	3.81E+16	2.08E+16	1.44E-04	8.05E-05	4.37E-05
25	1.37	27.62	7.02E+16	3.94E+16	2.16E+16	1.49E-04	8.32E-05	4.53E-05
26	1.34	28.95	7.22E+16	4.07E+16	2.24E+16	1.53E-04	8.58E-05	4.70E-05
27	1.35	30.31	7.45E+16	4.20E+16	2.33E+16	1.58E-04	8.86E-05	4.87E-05
28	1.46	31.77	7.78E+16	4.36E+16	2.41E+16	1.65E-04	9.18E-05	5.05E-05
29	1.41	33.18	8.06E+16	4.49E+16	2.49E+16	1.71E-04	9.47E-05	5.21E-05
Future		35.00	8.42E+16	4.67E+16	2.60E+16	1.78E-04	9.85E-05	5.43E-05
Future		40.00	9.41E+16	5.17E+16	2.89E+16	1.99E-04	1.09E-04	6.04E-05
Future		45.00	1.04E+17	5.67E+16	3.18E+16	2.20E-04	1.19E-04	6.66E-05
Future		50.00	1.14E+17	6.17E+16	3.48E+16	2.41E-04	1.30E-04	7.27E-05
Future		55.00	1.24E+17	6.67E+16	3.77E+16	2.62E-04	1.40E-04	7.88E-05
Future		60.00	1.34E+17	7.17E+16	4.06E+16	2.83E-04	1.51E-04	8.49E-05
Future		65.00	1.44E+17	7.67E+16	4.36E+16	3.04E-04	1.61E-04	9.10E-05
Future		70.00	1.54E+17	8.16E+16	4.65E+16	3.25E-04	1.72E-04	9.71E-05
Future		75.00	1.64E+17	8.66E+16	4.95E+16	3.46E-04	1.82E-04	1.03E-04
Future		80.00	1.73E+17	9.16E+16	5.24E+16	3.67E-04	1.93E-04	1.09E-04

Table 2-11. Upper Shell, Intermediate Shell, and Lower Shell Plates - Exposure

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (E > 1.0 MeV) (n/cm^2)			dpa		
			Upper Shell	Intermediate Shell	Lower Shell	Upper Shell	Intermediate Shell	Lower Shell
1	1.34	1.34	9.81E+17	2.52E+18	2.07E+18	1.62E-03	4.14E-03	3.42E-03
2	0.85	2.19	1.80E+18	4.08E+18	3.49E+18	2.96E-03	6.71E-03	5.76E-03
3	1.14	3.33	2.56E+18	5.84E+18	4.97E+18	4.22E-03	9.62E-03	8.18E-03
4	0.83	4.16	3.45E+18	7.55E+18	6.56E+18	5.68E-03	1.24E-02	1.08E-02
5	0.85	5.00	4.24E+18	9.18E+18	8.04E+18	6.98E-03	1.51E-02	1.32E-02
6	0.88	5.88	5.03E+18	1.08E+19	9.54E+18	8.29E-03	1.78E-02	1.57E-02
7	0.81	6.69	5.75E+18	1.23E+19	1.09E+19	9.47E-03	2.03E-02	1.80E-02
8	0.84	7.53	6.53E+18	1.38E+19	1.23E+19	1.08E-02	2.28E-02	2.02E-02
9	0.86	8.38	6.93E+18	1.46E+19	1.30E+19	1.14E-02	2.41E-02	2.14E-02
10	0.87	9.26	7.30E+18	1.56E+19	1.31E+19	1.20E-02	2.57E-02	2.16E-02
11	0.92	10.17	7.67E+18	1.66E+19	1.33E+19	1.26E-02	2.73E-02	2.19E-02
12	0.98	11.15	8.02E+18	1.76E+19	1.35E+19	1.32E-02	2.89E-02	2.22E-02
13	0.98	12.14	8.34E+18	1.84E+19	1.37E+19	1.37E-02	3.03E-02	2.25E-02
14	1.00	13.13	8.65E+18	1.93E+19	1.38E+19	1.43E-02	3.17E-02	2.28E-02
15	1.07	14.21	8.98E+18	2.02E+19	1.40E+19	1.48E-02	3.32E-02	2.31E-02
16	1.08	15.28	9.53E+18	2.17E+19	1.42E+19	1.57E-02	3.57E-02	2.35E-02
17	1.18	16.47	1.01E+19	2.32E+19	1.45E+19	1.67E-02	3.83E-02	2.38E-02
18	1.35	17.81	1.07E+19	2.48E+19	1.47E+19	1.76E-02	4.08E-02	2.43E-02
19	1.41	19.23	1.13E+19	2.63E+19	1.50E+19	1.86E-02	4.32E-02	2.47E-02
20	1.42	20.65	1.19E+19	2.76E+19	1.52E+19	1.95E-02	4.54E-02	2.51E-02
21	1.40	22.05	1.23E+19	2.88E+19	1.55E+19	2.03E-02	4.74E-02	2.56E-02
22	1.44	23.49	1.31E+19	3.06E+19	1.58E+19	2.16E-02	5.04E-02	2.60E-02
23	1.31	24.80	1.37E+19	3.20E+19	1.60E+19	2.26E-02	5.27E-02	2.64E-02
24	1.45	26.24	1.43E+19	3.34E+19	1.62E+19	2.36E-02	5.49E-02	2.68E-02
25	1.37	27.62	1.49E+19	3.46E+19	1.65E+19	2.45E-02	5.68E-02	2.72E-02
26	1.34	28.95	1.52E+19	3.54E+19	1.67E+19	2.51E-02	5.82E-02	2.75E-02
27	1.35	30.31	1.57E+19	3.64E+19	1.69E+19	2.59E-02	5.99E-02	2.79E-02
28	1.46	31.77	1.65E+19	3.83E+19	1.72E+19	2.72E-02	6.30E-02	2.83E-02
29	1.41	33.18	1.71E+19	3.97E+19	1.74E+19	2.82E-02	6.53E-02	2.88E-02
Future		35.00	1.79E+19	4.16E+19	1.77E+19	2.95E-02	6.83E-02	2.93E-02
Future		40.00	2.01E+19	4.67E+19	1.86E+19	3.31E-02	7.67E-02	3.07E-02
Future		45.00	2.23E+19	5.18E+19	1.95E+19	3.68E-02	8.51E-02	3.22E-02
Future		50.00	2.45E+19	5.69E+19	2.03E+19	4.04E-02	9.35E-02	3.37E-02
Future		55.00	2.67E+19	6.20E+19	2.12E+19	4.41E-02	1.02E-01	3.51E-02
Future		60.00	2.90E+19	6.71E+19	2.21E+19	4.77E-02	1.10E-01	3.66E-02
Future		65.00	3.12E+19	7.22E+19	2.30E+19	5.14E-02	1.19E-01	3.80E-02
Future		70.00	3.34E+19	7.74E+19	2.38E+19	5.50E-02	1.27E-01	3.95E-02
Future		75.00	3.56E+19	8.25E+19	2.47E+19	5.86E-02	1.36E-01	4.09E-02
Future		80.00	3.78E+19	8.76E+19	2.56E+19	6.23E-02	1.44E-01	4.24E-02

Table 2-12. Upper Shell Longitudinal Welds - Exposure

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (E > 1.0 MeV) (n/cm^2)			dpa		
			190°	70°	310°	190°	70°	310°
1	1.34	1.34	6.89E+17	3.39E+17	1.84E+17	1.14E-03	5.60E-04	3.01E-04
2	0.85	2.19	1.28E+18	6.40E+17	3.44E+17	2.13E-03	1.05E-03	5.62E-04
3	1.14	3.33	1.82E+18	9.28E+17	5.10E+17	3.02E-03	1.53E-03	8.32E-04
4	0.83	4.16	2.45E+18	1.23E+18	6.70E+17	4.06E-03	2.02E-03	1.09E-03
5	0.85	5.00	3.02E+18	1.50E+18	8.14E+17	5.00E-03	2.47E-03	1.33E-03
6	0.88	5.88	3.58E+18	1.78E+18	9.60E+17	5.94E-03	2.93E-03	1.57E-03
7	0.81	6.69	4.10E+18	2.05E+18	1.10E+18	6.80E-03	3.38E-03	1.80E-03
8	0.84	7.53	4.66E+18	2.33E+18	1.24E+18	7.72E-03	3.83E-03	2.03E-03
9	0.86	8.38	4.95E+18	2.54E+18	1.38E+18	8.21E-03	4.17E-03	2.25E-03
10	0.87	9.26	5.24E+18	2.75E+18	1.49E+18	8.68E-03	4.52E-03	2.43E-03
11	0.92	10.17	5.53E+18	2.99E+18	1.61E+18	9.18E-03	4.91E-03	2.62E-03
12	0.98	11.15	5.83E+18	3.24E+18	1.72E+18	9.67E-03	5.33E-03	2.82E-03
13	0.98	12.14	6.10E+18	3.49E+18	1.85E+18	1.01E-02	5.74E-03	3.02E-03
14	1.00	13.13	6.38E+18	3.75E+18	1.97E+18	1.06E-02	6.17E-03	3.22E-03
15	1.07	14.21	6.67E+18	4.04E+18	2.10E+18	1.11E-02	6.64E-03	3.44E-03
16	1.08	15.28	7.06E+18	4.23E+18	2.22E+18	1.17E-02	6.95E-03	3.63E-03
17	1.18	16.47	7.48E+18	4.45E+18	2.34E+18	1.24E-02	7.32E-03	3.83E-03
18	1.35	17.81	7.92E+18	4.69E+18	2.47E+18	1.32E-02	7.71E-03	4.04E-03
19	1.41	19.23	8.37E+18	4.95E+18	2.62E+18	1.39E-02	8.13E-03	4.29E-03
20	1.42	20.65	8.78E+18	5.20E+18	2.79E+18	1.46E-02	8.55E-03	4.57E-03
21	1.40	22.05	9.15E+18	5.43E+18	2.95E+18	1.52E-02	8.94E-03	4.82E-03
22	1.44	23.49	9.70E+18	5.66E+18	3.07E+18	1.61E-02	9.32E-03	5.02E-03
23	1.31	24.80	1.01E+19	5.87E+18	3.18E+18	1.68E-02	9.65E-03	5.21E-03
24	1.45	26.24	1.06E+19	6.09E+18	3.32E+18	1.76E-02	1.00E-02	5.44E-03
25	1.37	27.62	1.10E+19	6.30E+18	3.44E+18	1.82E-02	1.04E-02	5.63E-03
26	1.34	28.95	1.13E+19	6.48E+18	3.56E+18	1.87E-02	1.07E-02	5.83E-03
27	1.35	30.31	1.16E+19	6.69E+18	3.69E+18	1.93E-02	1.10E-02	6.04E-03
28	1.46	31.77	1.22E+19	6.94E+18	3.83E+18	2.02E-02	1.14E-02	6.27E-03
29	1.41	33.18	1.26E+19	7.16E+18	3.96E+18	2.09E-02	1.18E-02	6.48E-03
Future		35.00	1.32E+19	7.45E+18	4.13E+18	2.19E-02	1.23E-02	6.76E-03
Future		40.00	1.48E+19	8.25E+18	4.60E+18	2.45E-02	1.36E-02	7.53E-03
Future		45.00	1.64E+19	9.05E+18	5.07E+18	2.72E-02	1.49E-02	8.29E-03
Future		50.00	1.80E+19	9.85E+18	5.54E+18	2.98E-02	1.62E-02	9.06E-03
Future		55.00	1.96E+19	1.07E+19	6.01E+18	3.25E-02	1.76E-02	9.83E-03
Future		60.00	2.12E+19	1.15E+19	6.48E+18	3.51E-02	1.89E-02	1.06E-02
Future		65.00	2.28E+19	1.23E+19	6.95E+18	3.78E-02	2.02E-02	1.14E-02
Future		70.00	2.44E+19	1.31E+19	7.42E+18	4.05E-02	2.15E-02	1.21E-02
Future		75.00	2.60E+19	1.39E+19	7.89E+18	4.31E-02	2.29E-02	1.29E-02
Future		80.00	2.76E+19	1.47E+19	8.36E+18	4.58E-02	2.42E-02	1.37E-02

Table 2-13. Intermediate Shell Longitudinal Welds – Exposure

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (E > 1.0 MeV) (n/cm^2)			dpa		
			10°	250°	130°	10°	250°	130°
1	1.34	1.34	1.77E+18	8.71E+17	4.73E+17	2.93E-03	1.44E-03	7.72E-04
2	0.85	2.19	2.90E+18	1.44E+18	7.78E+17	4.81E-03	2.38E-03	1.27E-03
3	1.14	3.33	4.15E+18	2.11E+18	1.16E+18	6.88E-03	3.48E-03	1.90E-03
4	0.83	4.16	5.36E+18	2.68E+18	1.47E+18	8.89E-03	4.42E-03	2.40E-03
5	0.85	5.00	6.52E+18	3.25E+18	1.76E+18	1.08E-02	5.36E-03	2.88E-03
6	0.88	5.88	7.67E+18	3.82E+18	2.06E+18	1.27E-02	6.30E-03	3.37E-03
7	0.81	6.69	8.76E+18	4.38E+18	2.36E+18	1.45E-02	7.22E-03	3.85E-03
8	0.84	7.53	9.86E+18	4.93E+18	2.64E+18	1.64E-02	8.12E-03	4.31E-03
9	0.86	8.38	1.04E+19	5.35E+18	2.91E+18	1.73E-02	8.80E-03	4.75E-03
10	0.87	9.26	1.12E+19	5.93E+18	3.21E+18	1.86E-02	9.75E-03	5.24E-03
11	0.92	10.17	1.20E+19	6.57E+18	3.53E+18	1.99E-02	1.08E-02	5.76E-03
12	0.98	11.15	1.28E+19	7.29E+18	3.86E+18	2.12E-02	1.20E-02	6.31E-03
13	0.98	12.14	1.35E+19	7.96E+18	4.20E+18	2.24E-02	1.31E-02	6.86E-03
14	1.00	13.13	1.43E+19	8.69E+18	4.55E+18	2.37E-02	1.43E-02	7.43E-03
15	1.07	14.21	1.51E+19	9.48E+18	4.92E+18	2.50E-02	1.55E-02	8.03E-03
16	1.08	15.28	1.62E+19	1.00E+19	5.24E+18	2.69E-02	1.64E-02	8.56E-03
17	1.18	16.47	1.73E+19	1.06E+19	5.54E+18	2.87E-02	1.73E-02	9.06E-03
18	1.35	17.81	1.85E+19	1.12E+19	5.87E+18	3.06E-02	1.84E-02	9.59E-03
19	1.41	19.23	1.96E+19	1.18E+19	6.26E+18	3.24E-02	1.94E-02	1.02E-02
20	1.42	20.65	2.06E+19	1.24E+19	6.66E+18	3.40E-02	2.04E-02	1.09E-02
21	1.40	22.05	2.15E+19	1.30E+19	7.04E+18	3.56E-02	2.13E-02	1.15E-02
22	1.44	23.49	2.28E+19	1.35E+19	7.32E+18	3.77E-02	2.22E-02	1.20E-02
23	1.31	24.80	2.38E+19	1.40E+19	7.59E+18	3.94E-02	2.30E-02	1.24E-02
24	1.45	26.24	2.47E+19	1.45E+19	7.90E+18	4.10E-02	2.38E-02	1.29E-02
25	1.37	27.62	2.56E+19	1.50E+19	8.18E+18	4.24E-02	2.46E-02	1.33E-02
26	1.34	28.95	2.63E+19	1.54E+19	8.48E+18	4.35E-02	2.53E-02	1.38E-02
27	1.35	30.31	2.70E+19	1.58E+19	8.76E+18	4.48E-02	2.60E-02	1.43E-02
28	1.46	31.77	2.83E+19	1.64E+19	9.10E+18	4.69E-02	2.70E-02	1.48E-02
29	1.41	33.18	2.94E+19	1.69E+19	9.41E+18	4.86E-02	2.78E-02	1.53E-02
Future		35.00	3.07E+19	1.76E+19	9.82E+18	5.09E-02	2.89E-02	1.60E-02
Future		40.00	3.44E+19	1.95E+19	1.09E+19	5.70E-02	3.19E-02	1.78E-02
Future		45.00	3.81E+19	2.13E+19	1.20E+19	6.31E-02	3.50E-02	1.96E-02
Future		50.00	4.18E+19	2.31E+19	1.31E+19	6.92E-02	3.80E-02	2.14E-02
Future		55.00	4.55E+19	2.50E+19	1.42E+19	7.53E-02	4.10E-02	2.32E-02
Future		60.00	4.92E+19	2.68E+19	1.54E+19	8.14E-02	4.41E-02	2.50E-02
Future		65.00	5.29E+19	2.87E+19	1.65E+19	8.75E-02	4.71E-02	2.68E-02
Future		70.00	5.66E+19	3.05E+19	1.76E+19	9.36E-02	5.01E-02	2.86E-02
Future		75.00	6.02E+19	3.23E+19	1.87E+19	9.97E-02	5.32E-02	3.04E-02
Future		80.00	6.39E+19	3.42E+19	1.98E+19	1.06E-01	5.62E-02	3.22E-02

Table 2-14. Lower Shell Longitudinal Welds - Exposure

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (E > 1.0 MeV) (n/cm^2)		dpa	
			0°	120° & 240°	0°	120° & 240°
1	1.34	1.34	2.07E+18	5.39E+17	3.42E-03	8.81E-04
2	0.85	2.19	3.49E+18	9.32E+17	5.76E-03	1.53E-03
3	1.14	3.33	4.97E+18	1.37E+18	8.18E-03	2.24E-03
4	0.83	4.16	6.56E+18	1.77E+18	1.08E-02	2.89E-03
5	0.85	5.00	8.04E+18	2.15E+18	1.32E-02	3.51E-03
6	0.88	5.88	9.54E+18	2.55E+18	1.57E-02	4.16E-03
7	0.81	6.69	1.09E+19	2.94E+18	1.80E-02	4.80E-03
8	0.84	7.53	1.23E+19	3.30E+18	2.02E-02	5.39E-03
9	0.86	8.38	1.30E+19	3.63E+18	2.14E-02	5.92E-03
10	0.87	9.26	1.31E+19	4.02E+18	2.16E-02	6.57E-03
11	0.92	10.17	1.33E+19	4.47E+18	2.19E-02	7.30E-03
12	0.98	11.15	1.35E+19	4.97E+18	2.22E-02	8.11E-03
13	0.98	12.14	1.37E+19	5.46E+18	2.25E-02	8.90E-03
14	1.00	13.13	1.38E+19	5.98E+18	2.28E-02	9.75E-03
15	1.07	14.21	1.40E+19	6.55E+18	2.31E-02	1.07E-02
16	1.08	15.28	1.42E+19	6.88E+18	2.35E-02	1.12E-02
17	1.18	16.47	1.45E+19	7.24E+18	2.38E-02	1.18E-02
18	1.35	17.81	1.47E+19	7.63E+18	2.43E-02	1.24E-02
19	1.41	19.23	1.50E+19	8.05E+18	2.47E-02	1.31E-02
20	1.42	20.65	1.52E+19	8.47E+18	2.51E-02	1.38E-02
21	1.40	22.05	1.55E+19	8.88E+18	2.56E-02	1.45E-02
22	1.44	23.49	1.58E+19	9.20E+18	2.60E-02	1.50E-02
23	1.31	24.80	1.60E+19	9.51E+18	2.64E-02	1.55E-02
24	1.45	26.24	1.62E+19	9.85E+18	2.68E-02	1.60E-02
25	1.37	27.62	1.65E+19	1.02E+19	2.72E-02	1.65E-02
26	1.34	28.95	1.67E+19	1.05E+19	2.75E-02	1.71E-02
27	1.35	30.31	1.69E+19	1.08E+19	2.79E-02	1.76E-02
28	1.46	31.77	1.72E+19	1.12E+19	2.83E-02	1.82E-02
29	1.41	33.18	1.74E+19	1.15E+19	2.88E-02	1.88E-02
Future		35.00	1.77E+19	1.20E+19	2.93E-02	1.95E-02
Future		40.00	1.86E+19	1.32E+19	3.07E-02	2.15E-02
Future		45.00	1.95E+19	1.45E+19	3.22E-02	2.35E-02
Future		50.00	2.03E+19	1.57E+19	3.37E-02	2.56E-02
Future		55.00	2.12E+19	1.70E+19	3.51E-02	2.76E-02
Future		60.00	2.21E+19	1.82E+19	3.66E-02	2.96E-02
Future		65.00	2.30E+19	1.94E+19	3.80E-02	3.16E-02
Future		70.00	2.38E+19	2.07E+19	3.95E-02	3.36E-02
Future		75.00	2.47E+19	2.19E+19	4.09E-02	3.57E-02
Future		80.00	2.56E+19	2.32E+19	4.24E-02	3.77E-02

Table 2-15. Circumferential Welds - Exposure

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (E > 1.0 MeV) (n/cm^2)			dpa		
			Upper Shell to Inter. Shell	Inter. Shell to Lower Shell	Lower Shell to Lower Vessel Head	Upper Shell to Inter. Shell	Inter. Shell to Lower Shell	Lower Shell to Lower Vessel Head
1	1.34	1.34	9.81E+17	2.07E+18	4.05E+14	1.62E-03	3.42E-03	3.52E-06
2	0.85	2.19	1.80E+18	3.49E+18	7.05E+14	2.96E-03	5.76E-03	6.14E-06
3	1.14	3.33	2.56E+18	4.97E+18	9.99E+14	4.22E-03	8.18E-03	8.70E-06
4	0.83	4.16	3.45E+18	6.56E+18	1.34E+15	5.68E-03	1.08E-02	1.16E-05
5	0.85	5.00	4.24E+18	8.04E+18	1.65E+15	6.98E-03	1.32E-02	1.43E-05
6	0.88	5.88	5.03E+18	9.54E+18	1.97E+15	8.29E-03	1.57E-02	1.71E-05
7	0.81	6.69	5.75E+18	1.09E+19	2.27E+15	9.47E-03	1.80E-02	1.97E-05
8	0.84	7.53	6.53E+18	1.23E+19	2.54E+15	1.08E-02	2.02E-02	2.21E-05
9	0.86	8.38	6.93E+18	1.30E+19	2.69E+15	1.14E-02	2.14E-02	2.34E-05
10	0.87	9.26	7.30E+18	1.31E+19	2.73E+15	1.20E-02	2.16E-02	2.38E-05
11	0.92	10.17	7.67E+18	1.33E+19	2.78E+15	1.26E-02	2.19E-02	2.43E-05
12	0.98	11.15	8.02E+18	1.35E+19	2.83E+15	1.32E-02	2.22E-02	2.48E-05
13	0.98	12.14	8.34E+18	1.37E+19	2.88E+15	1.37E-02	2.25E-02	2.52E-05
14	1.00	13.13	8.65E+18	1.38E+19	2.93E+15	1.43E-02	2.28E-02	2.57E-05
15	1.07	14.21	8.98E+18	1.40E+19	2.99E+15	1.48E-02	2.31E-02	2.62E-05
16	1.08	15.28	9.53E+18	1.42E+19	3.04E+15	1.57E-02	2.35E-02	2.68E-05
17	1.18	16.47	1.01E+19	1.45E+19	3.11E+15	1.67E-02	2.38E-02	2.73E-05
18	1.35	17.81	1.07E+19	1.47E+19	3.17E+15	1.76E-02	2.43E-02	2.80E-05
19	1.41	19.23	1.13E+19	1.50E+19	3.25E+15	1.86E-02	2.47E-02	2.86E-05
20	1.42	20.65	1.19E+19	1.52E+19	3.32E+15	1.95E-02	2.51E-02	2.93E-05
21	1.40	22.05	1.23E+19	1.55E+19	3.38E+15	2.03E-02	2.56E-02	2.99E-05
22	1.44	23.49	1.31E+19	1.58E+19	3.46E+15	2.16E-02	2.60E-02	3.05E-05
23	1.31	24.80	1.37E+19	1.60E+19	3.52E+15	2.26E-02	2.64E-02	3.11E-05
24	1.45	26.24	1.43E+19	1.62E+19	3.59E+15	2.36E-02	2.68E-02	3.17E-05
25	1.37	27.62	1.49E+19	1.65E+19	3.65E+15	2.45E-02	2.72E-02	3.23E-05
26	1.34	28.95	1.52E+19	1.67E+19	3.70E+15	2.51E-02	2.75E-02	3.27E-05
27	1.35	30.31	1.57E+19	1.69E+19	3.76E+15	2.59E-02	2.79E-02	3.32E-05
28	1.46	31.77	1.65E+19	1.72E+19	3.84E+15	2.72E-02	2.83E-02	3.40E-05
29	1.41	33.18	1.71E+19	1.74E+19	3.90E+15	2.82E-02	2.88E-02	3.46E-05
Future		35.00	1.79E+19	1.77E+19	3.99E+15	2.95E-02	2.93E-02	3.54E-05
Future		40.00	2.01E+19	1.86E+19	4.23E+15	3.31E-02	3.07E-02	3.75E-05
Future		45.00	2.23E+19	1.95E+19	4.47E+15	3.68E-02	3.22E-02	3.97E-05
Future		50.00	2.45E+19	2.03E+19	4.70E+15	4.04E-02	3.37E-02	4.18E-05
Future		55.00	2.67E+19	2.12E+19	4.94E+15	4.41E-02	3.51E-02	4.40E-05
Future		60.00	2.90E+19	2.21E+19	5.18E+15	4.77E-02	3.66E-02	4.61E-05
Future		65.00	3.12E+19	2.30E+19	5.42E+15	5.14E-02	3.80E-02	4.83E-05
Future		70.00	3.34E+19	2.38E+19	5.66E+15	5.50E-02	3.95E-02	5.05E-05
Future		75.00	3.56E+19	2.47E+19	5.90E+15	5.86E-02	4.09E-02	5.26E-05
Future		80.00	3.78E+19	2.56E+19	6.14E+15	6.23E-02	4.24E-02	5.48E-05

3 NEUTRON DOSIMETRY EVALUATIONS

The current H. B. Robinson Unit 2 measurement database consists of four evaluated sensor sets from in-vessel capsules that were withdrawn and analyzed as an integral part of the Reactor Vessel Materials Surveillance Program (Reference 4), and five ex-vessel sensor sets irradiated as a part of the Ex-Vessel Neutron Dosimetry Program (References 8 and 9). The Materials Surveillance Program was initiated in the first operating fuel cycle at H. B. Robinson Unit 2 while the Ex-Vessel Neutron Dosimetry Program was initiated prior to Cycle 10. Thus, the dosimetry data obtained from these two complementary programs have provided continuous monitoring of the neutron exposure of the pressure vessel since plant startup. As noted in Section 2.4 of Regulatory Guide 1.190:

"… the dosimetry irradiated with metallurgical specimens is only available at infrequent intervals. …Recent pressure vessel benchmark experiments have demonstrated that the ex-vessel dosimetry can provide useful exposure information within the pressure vessel wall. When placed at appropriate circumferential locations, this dosimetry is a good monitor of the effectiveness of low-leakage core strategies."

With this philosophy in mind, the EVND Program installed at H. B. Robinson Unit 2 was designed to supplement the measurement data obtained from the in-vessel surveillance capsules, thus, providing a large comprehensive plant-specific database. A description of the EVND Program is provided in Appendix A of this report.

3.1 METHOD OF ANALYSIS

Evaluations of neutron sensor sets contained in the in-vessel and ex-vessel dosimetry capsules withdrawn to date from the H. B. Robinson Unit 2 reactor were completed using current state of the art least-squares methodology.

Least-squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a best-estimate neutron energy spectrum with associated uncertainties. Best-estimates for key exposure parameters such as fast neutron fluence rate (E > 1.0 MeV) and dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to reactor dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties.

For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ ($\delta_{\sigma_{ig}}$ and δ_{ϕ_g}, respectively). The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the H. B. Robinson Unit 2 dosimetry, the FERRET code (References 10 and 11) was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters along with associated uncertainties.

The application of the least-squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location
2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set
3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor set

For the current application, the calculated neutron spectrum at each measurement location was obtained from the results of plant-specific neutron transport calculations described in Section 2 of this report. The spectrum at each sensor set location was input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements. The sensor reaction rates were derived from the measured specific activities of each sensor set and the operating history of the respective fuel cycles. The dosimetry reaction cross sections were obtained from the SNLRML dosimetry cross-section library (Reference 12).

In addition to the magnitude of the calculated neutron spectra, the measured sensor set reaction rates, and the dosimeter set reaction cross sections, the least-squares procedure requires uncertainty estimates for each of these input parameters. The following provides a summary of the uncertainties associated with the least-squares evaluation of the H. B. Robinson Unit 2 dosimetry.

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, the irradiation history corrections, and the corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM consensus standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least-squares evaluation:

Reaction	Uncertainty
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	5%
$^{46}Ti\ (n,p)\ ^{46}Sc$	5%
$^{54}Fe\ (n,p)\ ^{54}Mn$	5%
$^{58}Ni\ (n,p)\ ^{58}Co$	5%
$^{238}U\ (n,f)\ ^{137}Cs$	10%
$^{237}Np\ (n,f)\ ^{137}Cs$	10%
$^{59}Co\ (n,\gamma)\ ^{60}Co$	5%

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

As noted above, the reaction rate cross sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross sections and uncertainties are provided in a fine multigroup structure for use in least-squares adjustment applications. These cross sections have been tested with respect to their accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources. Detailed discussions of the contents of the SNLRML library along with the evaluation process for each of the sensors are provided in Reference 12.

For sensors included in the H. B. Robinson Unit 2 dosimetry sets, the following uncertainties in the fission spectrum-averaged cross sections are provided in the SNLRML documentation package:

Reaction	Uncertainty
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	4.08 – 4.16%
$^{46}Ti\ (n,p)\ ^{46}Sc$	4.50 – 4.87%
$^{54}Fe\ (n,p)\ ^{54}Mn$	3.05 – 3.11%
$^{58}Ni\ (n,p)\ ^{58}Co$	4.49 – 4.56%
^{238}U (n,f) FP	0.54 – 0.64%
^{237}Np (n,f) FP	10.32 – 10.97%
$^{59}Co\ (n,\gamma)\ ^{60}Co$	0.79 – 3.59%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrum Uncertainties

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks, and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

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

where R_n specifies an overall fractional normalization uncertainty, and the fractional uncertainties $R_{g'}$ and R_g specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

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

where:

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

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of $\delta_{gg'}$ is 1.0 when $g = g'$ and 0.0 otherwise.

The set of parameters defining the input covariance matrix for the H. B. Robinson Unit 2 calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	25%
(E < 0.68 eV)	50%
Short-Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

These uncertainty assignments are consistent with an industry consensus uncertainty of 15-20% (1σ) for the fast neutron portion of the spectrum and provide for a reasonable increase in the uncertainty for neutrons in the intermediate and thermal energy ranges.

3.2 DOSIMETRY EVALUATIONS

During the course of the first 29 operating fuel cycles at H. B. Robinson Unit 2, four in-vessel capsules (Reference 7) and five ex-vessel neutron dosimetry sets (References 8 and 9) were irradiated and subsequently withdrawn for analysis. The locations and time of irradiation of each of these multiple foil sensor sets analyzed in this report are provided in Table 3-1. Additional details of the EVND are provided in Appendix A of this report.

Table 3-1. Location and Time of Irradiation for Sensor Sets Analyzed

Capsule ID	Sensor Location	Azimuthal Location	Axial Elevation[1]	Cycles of Irradiation
S	In-Vessel	10°	Core Midplane	1
V	In-Vessel	20°	Core Midplane	1-3
T	In-Vessel	0°	Core Midplane	1-8
X	In-Vessel	40°, 0°	Core Midplane	1-20
BC	Ex-Vessel	3°	+8.92 ft	10
BD	Ex-Vessel	3°	-0.83 ft	10
BE	Ex-Vessel	9°	+8.92 ft	10
BF	Ex-Vessel	9°	-0.83 ft	10
BG	Ex-Vessel	27°	+8.92 ft	10
BH	Ex-Vessel	27°	-0.83 ft	10
A	Ex-Vessel	3°	+5.85 ft	11
B	Ex-Vessel	3°	-3.95 ft	11
C	Ex-Vessel	9°	+5.85 ft	11
D	Ex-Vessel	9°	-3.95 ft	11
E	Ex-Vessel	27°	+7.85 ft	11
F	Ex-Vessel	27°	-1.95 ft	11
G	Ex-Vessel	0°	+5.53 ft	13
H	Ex-Vessel	0°	-4.31 ft	13
I	Ex-Vessel	15°	+5.53 ft	13
J	Ex-Vessel	15°	-4.31 ft	13
K	Ex-Vessel	30°	+5.53 ft	13
L	Ex-Vessel	30°	-4.31 ft	13
M	Ex-Vessel	0°	+5.53 ft	14,15
Q	Ex-Vessel	30°	+5.53 ft	14,15
R	Ex-Vessel	30°	-4.31 ft	14,15
O	Ex-Vessel	45°	+5.53 ft	14,15
P	Ex-Vessel	45°	-4.31 ft	14,15
U	Ex-Vessel	30°	+5.53 ft	16-29
V	Ex-Vessel	30°	-4.31 ft	16-29

Note:
1. Relative to core midplane at z = 0 ft

In this section, comparisons of the measurement results from each of the sensor set irradiations with corresponding analytical predictions at the measurement locations are presented. These comparisons are provided on two levels. On the first level, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. This level of comparison is not impacted by the least-squares evaluations of the sensor sets. On the second level, calculated values of neutron exposure rates in terms of fast neutron fluence rate (E > 1.0 MeV) and dpa/s are compared with the best-estimate exposure rates obtained from the least-squares evaluation.

In Table 3-2, comparisons of measured-to-calculation (M/C) reaction rate ratios are listed for the threshold sensors contained in in-vessel Capsules S, V, T, and X. From Table 3-2, it is noted that for the individual threshold sensors, the average M/C ratio ranges from 1.12 to 1.24 with an overall average of 1.17 and an associated standard deviation of 11.5%. In this case, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

Table 3-2. Comparison of Measured (M) to Calculated (C) Reaction Rates Sensitive in the Fast Energy Range for In-Vessel Neutron Dosimetry

Capsule	M/C					
	^{63}Cu(n,α)	^{46}Ti(n,p)	^{54}Fe(n,p)	^{58}Ni(n,p)	^{238}U(n,f)	^{237}Np(n,f)
S	1.23	-	1.05	0.89	-	-
V	1.30	-	1.39	1.31	-	-
T	1.26	-	1.19	1.12	1.32	1.17
X	1.17	-	1.00	1.15	1.01	1.12
Average	1.24	-	1.16	1.12	1.17	1.15
% Standard Deviation	4.4	-	15.1	15.5	18.8	3.1
Average	1.17					
% Standard Deviation	11.5					

Comparisons of the M/C ratios for the ex-vessel threshold sensor sets withdrawn from the off-midplane measurement locations in the reactor cavity are provided in Tables 3-3 and 3-4. Table 3-5 provides the combined EVND database. Measured and calculated ^{54}Fe (n,p) ^{54}Mn and ^{58}Ni (n,p) ^{58}Co reaction rates of the gradient chains are plotted in Figures 3-1 and 3-2, respectively, for the 300° (30° from core cardinal axis) location.

Below are notes on the processing of the EVND data presented in this report:

- Solid-state track recorder data used in the earlier EVND is generally excluded from the evaluations due to uncertainty in the initial fissile material content of the recorders.
- Fission monitors are generally excluded from the evaluations for the historical EVND (that is, Cycles 10 through 15) due to questions about data traceability and initial impurity.
- Niobium results are excluded for all evaluations due to potentially high initial Tantalum content.

Table 3-3. Comparison of Measured and Calculated Reaction Rates for Fast Neutron Reactions of Ex-Vessel Capsules – Above Core Midplane and Opposite the Active Fuel

Capsule	M/C					
	^{63}Cu(n,α)	^{46}Ti(n,p)	^{54}Fe(n,p)	^{58}Ni(n,p)	^{238}U(n,f)	^{237}Np(n,f)
A – Cycle 11	1.22	-	0.95	-	-	-
C – Cycle 11	1.26	-	1.07	-	-	-
G – Cycle 13	0.87	0.98	0.83	0.97	-	-
I – Cycle 13	0.75	0.97	0.71	0.81	-	-
K – Cycle 13	0.70	0.79	0.67	0.73	-	-
M – Cycles 14/15	0.81	0.93	0.77	0.90	-	-
O – Cycles 14/15	0.69	0.81	0.67	0.76	-	-
Q – Cycles 14/15	0.72	0.78	0.66	0.74	-	-
U – Cycles 16-29	0.75	0.88	0.80	0.91	1.03	1.19
Average	0.86	0.88	0.79	0.83	-	-
% Standard Deviation	25.6	9.7	17.7	11.4	-	-
Average	0.86					
% Standard Deviation	18.6					

Table 3-4. Comparison of Measured and Calculated Reaction Rates for Fast Neutron Reactions of Ex-Vessel Capsules – Below Core Midplane and Opposite the Active Fuel

Capsule	M/C					
	^{63}Cu(n,α)	^{46}Ti(n,p)	^{54}Fe(n,p)	^{58}Ni(n,p)	^{238}U(n,f)	^{237}Np(n,f)
BD – Cycle 10	-	-	1.31	1.25	-	-
BF – Cycle 10	-	-	1.40	1.28	-	-
BH – Cycle 10	-	-	0.96	0.85	-	-
B – Cycle 11	1.04	-	1.13	-	-	-
D – Cycle 11	1.09	-	1.09	-	-	-
F – Cycle 11	1.38	-	1.44	-	-	-
H – Cycles 12/13	1.13	1.26	1.10	1.08	-	-
J – Cycles 12/13	1.30	1.39	1.28	1.27	-	-
L – Cycles 12/13	1.22	1.36	1.27	1.27	-	-
P – Cycles 14/15	1.25	1.35	1.20	1.23	-	-
R – Cycles 14/15	1.25	1.33	1.22	1.24	-	-
V – Cycles 16-29	1.09	1.16	1.05	1.14	1.26	1.38
Average	1.19	1.31	1.20	1.18		
% Standard Deviation	9.5	6.5	12.0	11.9		
Average	1.22					
% Standard Deviation	10.6					

Table 3-5. Comparison of Measured and Calculated Reaction Rates for Fast Neutron Reactions of Ex-Vessel Capsules Opposite the Active Fuel

Capsule	M/C					
	^{63}Cu(n,α)	^{46}Ti(n,p)	^{54}Fe(n,p)	^{58}Ni(n,p)	^{238}U(n,f)	^{237}Np(n,f)
A – Cycle 11	1.22	-	0.95	-	-	-
C – Cycle 11	1.26	-	1.07	-	-	-
G – Cycles 13	0.87	0.98	0.83	0.97	-	-
I – Cycles 13	0.75	0.97	0.71	0.81	-	-
K – Cycles 13	0.70	0.79	0.67	0.73	-	-
M – Cycles 14/15	0.81	0.93	0.77	0.90	-	-
O – Cycles 14/15	0.69	0.81	0.67	0.76	-	-
Q – Cycles 14/15	0.72	0.78	0.66	0.74	-	-
U – Cycles 16-29	0.75	0.88	0.80	0.91	1.03	1.19
BD – Cycle 10	-	-	1.31	1.25	-	-
BF – Cycle 10	-	-	1.40	1.28	-	-
BH – Cycle 10	-	-	0.96	0.85	-	-
B – Cycle 11	1.04	-	1.13	-	-	-
D – Cycle 11	1.09	-	1.09	-	-	-
F – Cycle 11	1.38	-	1.44	-	-	-
H – Cycles 13	1.13	1.26	1.10	1.08	-	-
J – Cycles 13	1.30	1.39	1.28	1.27	-	-
L – Cycles 13	1.22	1.36	1.27	1.27	-	-
P – Cycles 14/15	1.25	1.35	1.20	1.23	-	-
R – Cycles 14/15	1.25	1.33	1.22	1.24	-	-
V – Cycles 16-29	1.09	1.16	1.05	1.14	1.26	1.38
Average	1.03	1.08	1.03	1.03	1.15	1.29
% Standard Deviation	23.4	22.1	24.4	20.8	14.2	10.5
Average	1.05					
% Standard Deviation	22.2					

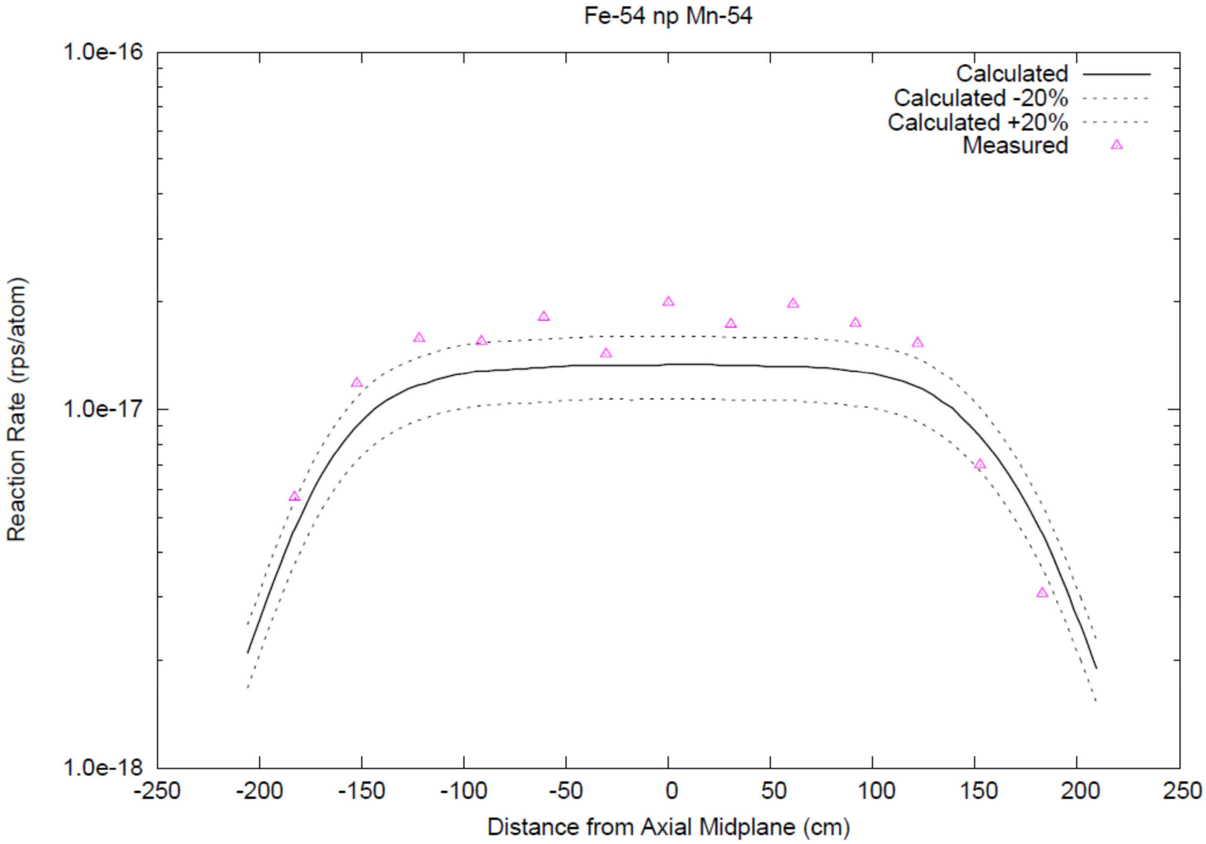

Figure 3-1. Measured and Calculated EVND Gradient Chain Irradiated in Cycles 16 through 29; ^{54}Fe (n,p) ^{54}Mn Reaction Rates at 300°

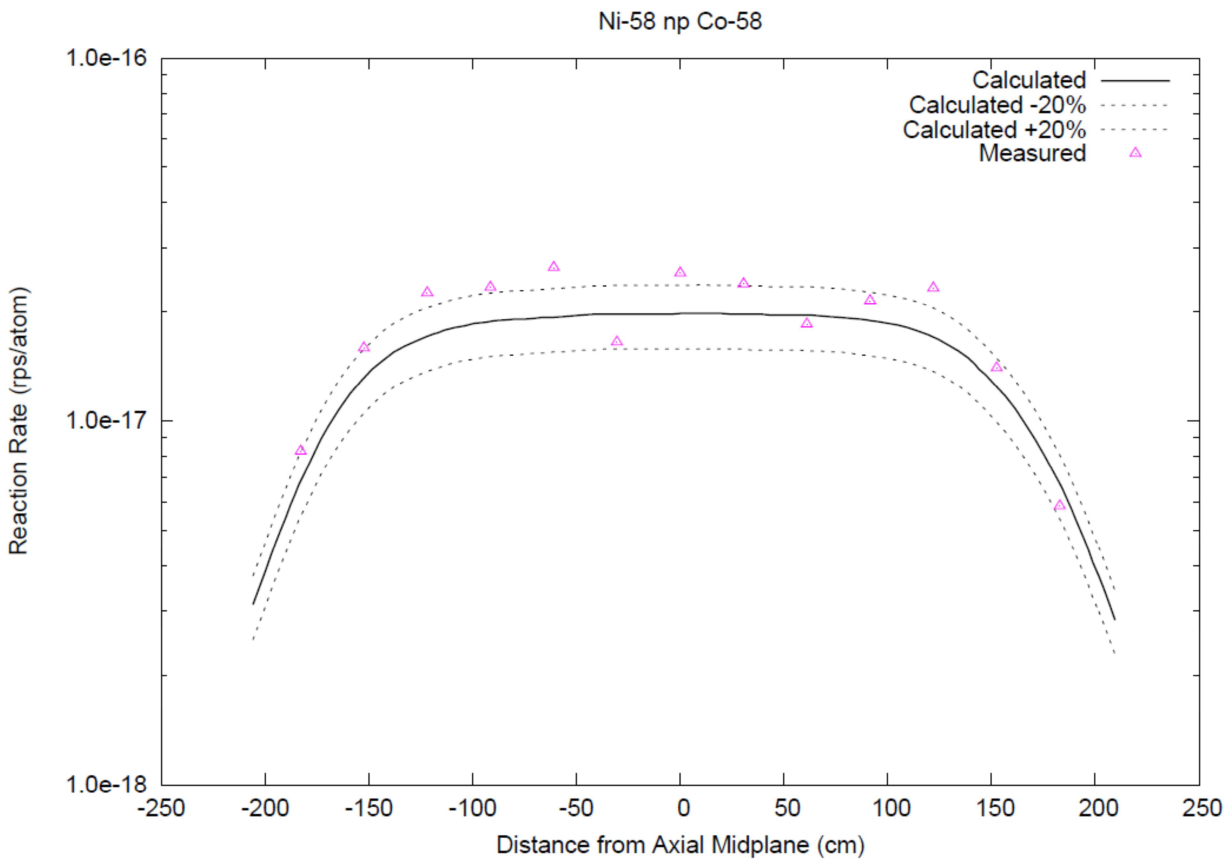

Figure 3-2. Measured and Calculated EVND Gradient Chain Irradiated in Cycles 16 through 29; ^{58}Ni (n,p) ^{58}Co Reaction Rates at 300°

In Table 3-6, best-estimate-to-calculated (BE/C) ratios for fast neutron fluence rate (E > 1.0 MeV) and iron displacements per atom per second resulting from the least-squares evaluation of each dosimetry set are provided for in-vessel irradiations. For the in-vessel capsules, the average BE/C ratio is 1.15 with an associated uncertainty of 11.1% for fast neutron fluence rate (E > 1.0 MeV) and 1.14 with an associated uncertainty of 9.8% for iron displacements per atom per second.

Table 3-6. Best-Estimate-to-Calculated Exposure Rates for In-Vessel Capsules

Capsule	BE/C	
	Neutron Fluence Rate (E > 1.0 MeV)	Iron Displacements per Atom per Second (dpa/s)
S	1.03	1.04
V	1.31	1.29
T	1.18	1.16
X	1.06	1.06
Average	1.15	1.14
% Standard Deviation	11.1	9.8

For the midplane in-vessel capsules, the linearly averaged M/C ratio of 1.17 are in good agreement with the resultant least-squares BE/C ratio of 1.14 for fast neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate.

Regulatory Guide 1.190 Section 1.4.2 states that:

"*Differences between measurements and calculations should be consistent with the combined uncertainty estimates for the measurements and calculations. (Note that the uncertainties in both the calculations and measurements will contribute to the observed measurement-to-calculation differences.) The calculated reaction rates (using the methods described in Regulatory Positions 1.1 through 1.3) typically agree with the measurements to within about 20% for in-vessel surveillance capsules and 30% for cavity dosimetry. Deviations greater than these values must be investigated and, when the cause of the deviation is determined to be an error in the calculation, the calculations must be modified.*"

The comparisons in this analysis demonstrate that the calculated results provided in Section 2 of this report are validated within 20% uncertainty for in-vessel surveillance capsules and 30% uncertainty for cavity dosimetry.

4 UNCERTAINTY IN FLUENCE CALCULATIONS

The uncertainty associated with the calculated neutron exposure of the H. B. Robinson Unit 2 reactor pressure vessel is based on the recommended approach provided in Regulatory Guide 1.190. In particular, the qualification of the methodology used in the H. B. Robinson Unit 2 reactor pressure vessel neutron exposure evaluations was carried out in the following four stages:

1. Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator (Reference 13) at the Oak Ridge National Laboratory (ORNL).
2. Comparison of calculations with surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment (Reference 14).
3. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant-specific transport calculations used in the neutron exposure assessments.
4. Comparisons of the calculations with dosimetry results from measurement programs carried out at the H. B. Robinson Unit 2 reactor.

The first phase of the methods qualification addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross sections. This phase, however, did not test the accuracy of commercial core neutron source calculations, nor did it address uncertainties in operational and geometric variables that impact power reactor calculations. The second phase of the qualification addressed uncertainties that are primarily methods-related and would tend to apply generically to all fast neutron exposure evaluations. The third phase of the qualification identified the potential uncertainties introduced into the overall evaluation due to calculational methods approximations as well as to a lack of knowledge relative to various plant-specific parameters. The overall calculational uncertainty was established from the results of these three phases of the methods qualification.

The comparison of the calculated results with the available plant-specific dosimetry results was used solely to demonstrate the adequacy of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used to bias the final transport calculation results in any way.

The following summarizes the uncertainties associated with the H. B. Robinson Unit 2 neutron exposure calculations:

PCA Benchmark Comparisons	3%
H. B. Robinson Cycle 9 Benchmark Comparisons	3%
Analytical Sensitivity Studies	11%
Internals Dimensions	3%
Vessel Inner Radius	5%
Water Temperature	4%
Peripheral Assembly Source Strength	5%
Axial Power Distribution	5%
Peripheral Assembly Burnup	2%
Spatial Distribution of the Source	4%
Other Factors	5%

The category designated “Other Factors” is intended to attribute an additional uncertainty to other geometrical or operational variables that individually have an insignificant impact on the overall uncertainty, but collectively should be accounted for in the assessment. In the overall uncertainty evaluation, these uncertainty components are considered to be at the 1σ level.

A more detailed discussion of the PCA and H. B. Robinson benchmark comparisons as well as the analytical sensitivity studies is provided in Reference 11.

When the uncertainty components tabulated above are combined in quadrature, the resultant uncertainty in the H. B. Robinson Unit 2 calculated exposure levels listed in Section 2 of this report was determined to be 13% at the 1σ level.

The comparisons of the transport calculation results with data from the H. B. Robinson Unit 2 in-vessel and ex-vessel measurement programs support the fourth phase of the methods qualification and demonstrate that the requirements of Regulatory Guide 1.190 are met.

5 REFERENCES

1. United States of America Code of Federal Regulations Title 10 Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Materials Surveillance Requirements," January 2008.

2. Nuclear Regulatory Commission Regulatory Issue Summary 2014-11, “Information on Licensing Applications For Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” October 14, 2014.

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4. Westinghouse Report, WCAP-7373, Rev 0, "Carolina Power and Light Co. H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program," January 1970.

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7. Westinghouse Report, WCAP-15805, Rev. 0, "Analysis of Capsule X from the Carolina Power & Light Company H.B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program," March 2002.

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10. RSICC Peripheral Shielding Routine PSR-145, "FERRET, Least-Squares Solution to Nuclear Data and Reactor Physics Problems," Radiation Safety Information Computational Center (RSICC), Oak Ridge National Laboratory, January 1980.

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13. NUREG/CR-6454 (ORNL/TM-13205), “Pool Critical Assembly Pressure Vessel Facility Benchmark,” Oak Ridge National Laboratory, July 1997.

14. NUREG/CR-6453 (ORNL/TM-13204), “H. B. Robinson-2 Pressure Vessel Benchmark,” Oak Ridge National Laboratory, February 1998.

APPENDIX A DESCRIPTION OF EX-VESSEL NEUTRON DOSIMETRY

The Westinghouse reactor cavity measurement program started in the beginning of Cycle 10 with temporary ex-vessel neutron dosimetry installed before the Cycle 10 and Cycle 11 operations, at three azimuthal locations at an axial elevation above the active fuel and three corresponding azimuthal locations at an axial elevation approximately one foot below the core midplane. At the beginning of Cycle 12, the permanent ex-vessel neutron dosimetry program was established by placing multiple foil sensor sets such that spectra evaluations could be made at three azimuthal locations at an axial elevation opposite the lower circumferential weld and three corresponding azimuthal locations adjacent to the upper circumferential weld. The intent was to determine changes in spectra caused by varying amounts of water located between the core and the pressure vessel as well as to directly monitor the exposure of the beltline welds. Due to the irregular shape of the reactor core, water thickness varies significantly as a function of azimuthal angle. At each of the three azimuthal locations selected for spectra measurements and at one additional azimuthal location corresponding to the minimum in the neutron fluence rate distribution, gradient chains extended over a 14 foot height centered on the core midplane. During the irradiations preceding Cycle 12, temporary support fixtures were employed that resulted in azimuthal and axial measurement locations that differed slightly from those utilized in the permanent installation. The specific locations used in these prior irradiations were determined by access limitations resulting from the design of the temporary fixtures and will not be repeated as part of the long term measurement program.

During Cycles 10 and 11, the strings were located at azimuthal positions of 3, 9, 27, and 45 degrees relative to the core cardinal axis; while, for Cycles 12 and beyond, the permanent installation provided azimuthal locations of 0, 15, 30 and 45 degrees. During all cycles, the sensor strings were hung in the annular gap between the pressure vessel insulation and the primary biological shield at a nominal radius of 93.7 inches relative to the core centerline.

The sensor sets used to characterize the neutron spectra within the reactor cavity were retained in 3.87 inch × 1.00 inch × 0.50 inch rectangular aluminum 6061 capsules. Each capsule included three compartments to hold the neutron sensors. The top compartment (position 1) was intended to accommodate bare radiometric monitors, whereas, the two remaining compartments (positions 2 and 3) were meant to house cadmium shielded packages. The separation between positions 1 and 2 was such that cadmium shields inserted into position 2 did not introduce perturbations in the thermal fluence rate in position 1. Aluminum 6061 was selected for the dosimeter capsules in order to minimize neutron fluence rate perturbations at the sensor set locations as well as to limit the radiation levels associated with post-irradiation shipping and handling of the capsules.

The latest removed dosimetry (Cycles 16-29) was as follows:

- 270° transport loop section – approximately 25′ feet long
- 300° transport loop section – approximately 25′ feet long
- 300° gradient chain – includes Capsules U and V and is approximately 25′ long

Among the removed dosimetry, only the 300° data was utilized. The overall length of the 300° dosimetry is 169-5/8 inches. The 300° capsules positions are given in Table A-1. The capsule contents are given in Table A-2.

Table A-1. EVND Capsule Positions on Gradient Chains – Irradiation Cycles 16 through 29

Azimuthal Location (°)	Capsule ID	Capsule Location	Position (ft)(1)
300	U	Approximately at upper circumferential weld	5.53
300	V	Approximately at lower circumferential weld	-4.31

Note:

1. Capsule positions are relative to the core midplane located at z = 0 ft.

Table A-2. Dosimeter Capsule Contents – Irradiation Cycles 16 through 29

Capsule ID	Bare or Cadmium Shielded	Radiometric Monitor Foil ID							
		Fe	Ni	Ti	Cu	Nb	Co	U-238	Np-237
U-1	Bare	E	-	-	-	-	CE	-	-
U-2	Cd	F	C	C	C	C	CF	-	-
U-3	Cd	-	-	-	-	-	-	10	3
V-1	Bare	G	-	-	-	-	CG	-	-
V-2	Cd	H	D	D	D	D	CH	-	-
V-3	Cd	-	-	-	-	-	-	11	5

APPENDIX B DOSIMETRY RESULTS FOR IN-VESSEL CAPSULES

The results of the dosimetry evaluations performed for the H. B. Robinson Unit 2 in-vessel capsules S, V, T, and X, are provided in this appendix. The data tabulations for each capsule evaluation include the following information:

1. The measured, calculated, and best-estimate reaction rates (rps/atom) for each sensor.
2. The measurement-to-calculation (M/C), best-estimate-to-measurement (BE/M), and best-estimate-to-calculation (BE/C) reaction rate ratios for each sensor.
3. The calculated and best-estimate values of fast neutron fluence rate (E > 1.0 MeV and E > 0.1 MeV) and iron atom displacement rate with associated uncertainties.
4. The best-estimate-to-calculation ratio (BE/C) for both fast neutron fluence rate (E > 1.0 MeV and E > 0.1 MeV) and iron displacements per atom per second.

The M/C and BE/M ratios for the individual sensors establish a comparison between measurement and calculation before and after the least-squares evaluation. The improvement in these reaction rate ratios for the best-estimate case is an indication of the improvement in the neutron spectrum and corresponding reduction in uncertainty brought about by the application of the least-squares procedure. The comparisons of calculated and best-estimate values of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate with associated uncertainties also provided an indication of the improved results obtained with the least-squares procedure.

Table B-1. Least-Squares Evaluation of Dosimetry in Surveillance Capsule S (10° Azimuth, Core Midplane); Cycle 1 Irradiation

Reaction	**Reaction Rate**			**M/C**	**BE/M**	**BE/C**
	Measured (M)	**Calculated (C)**	**Best-Estimate (BE)**			
Cu-63(n,α)Co-60	7.76E-17	6.30E-17	7.45E-17	1.23	0.96	1.18
Fe-54(n,p)Mn-54	7.95E-15	7.57E-15	8.18E-15	1.05	1.03	1.08
Ni-58(n,p)Co-58	9.36E-15	1.05E-14	1.13E-14	0.89	1.21	1.08
Co-59(n,g)Co-60	7.86E-12	5.93E-12	7.81E-12	1.33	0.99	1.32
Co-59(Cd)(n,g)Co-60	3.06E-12	2.75E-12	3.07E-12	1.11	1.00	1.12
Average of Fast Energy Threshold Reactions				1.06	1.07	1.11
% Standard Deviation				16.1	12.1	5.2

Integral Quantity	**Calculated (C)**	**% Unc.**	**Best-Estimate (BE)**	**% Unc.**	**BE/C**
Fluence rate E > 1.0 MeV (n/cm^2-s)	1.14E+11	13	1.18E+11	8	1.03
Fluence rate E > 0.1 MeV (n/cm^2-s)	3.92E+11	-	4.05E+11	12	1.03
dpa/s	1.92E-10	13	2.00E-10	9	1.04

Table B-2. Least-Squares Evaluation of Dosimetry in Surveillance Capsule V (20° Azimuth, Core Midplane); Cycles 1 through 3 Irradiation

Reaction	**Reaction Rate**			**M/C**	**BE/M**	**BE/C**
	Measured (M)	**Calculated (C)**	**Best-Estimate (BE)**			
Cu-63(n,α)Co-60	5.03E-17	3.87E-17	5.06E-17	1.30	1.01	1.31
Fe-54(n,p)Mn-54	5.53E-15	3.97E-15	5.34E-15	1.39	0.97	1.35
Ni-58(n,p)Co-58	7.06E-15	5.40E-15	7.15E-15	1.31	1.01	1.32
Co-59(n,g)Co-60	3.05E-12	2.17E-12	3.03E-12	1.40	0.99	1.39
Co-59(Cd)(n,g)Co-60	1.10E-12	9.89E-13	1.11E-12	1.11	1.00	1.12
Average of Fast Energy Threshold Reactions				1.33	1.00	1.33
% Standard Deviation				3.7	2.3	1.6

Integral Quantity	**Calculated (C)**	**% Unc.**	**Best-Estimate (BE)**	**% Unc.**	**BE/C**
Fluence rate E > 1.0 MeV (n/cm^2-s)	5.06E+10	13	6.64E+10	7	1.31
Fluence rate E > 0.1 MeV (n/cm^2-s)	1.56E+11	-	1.96E+11	11	1.26
dpa/s	8.18E-11	13	1.05E-10	8	1.29

Table B-3. Least-Squares Evaluation of Dosimetry in Surveillance Capsule T (0° Azimuth, Core Midplane); Cycles 1 through 8 Irradiation

Reaction	Reaction Rate Measured (M)	Reaction Rate Calculated (C)	Reaction Rate Best-Estimate (BE)	M/C	BE/M	BE/C
Cu-63(n,α)Co-60	1.11E-16	8.82E-17	1.08E-16	1.26	0.97	1.23
Fe-54(n,p)Mn-54	1.29E-14	1.09E-14	1.30E-14	1.19	1.00	1.19
Ni-58(n,p)Co-58	1.69E-14	1.51E-14	1.76E-14	1.12	1.04	1.17
U-238(Cd)(n,f)Cs-137	7.39E-14	5.60E-14	6.62E-14	1.32	0.90	1.18
Np-237(Cd)(n,f)Cs-137	5.19E-13	4.45E-13	5.19E-13	1.17	1.00	1.17
Co-59(n,g)Co-60	8.25E-12	8.24E-12	8.29E-12	1.00	1.00	1.01
Co-59(Cd)(n,g)Co-60	4.22E-12	3.84E-12	4.21E-12	1.10	1.00	1.10
Average of Fast Energy Threshold Reactions				1.21	0.98	1.19
% Standard Deviation				6.5	5.3	2.1

Integral Quantity	Calculated (C)	% Unc.	Best-Estimate (BE)	% Unc.	BE/C
Fluence rate E > 1.0 MeV (n/cm^2-s)	1.63E+11	13	1.92E+11	6	1.18
Fluence rate E > 0.1 MeV (n/cm^2-s)	5.54E+11	-	6.37E+11	9	1.15
dpa/s	2.73E-10	13	3.17E-10	7	1.16

Table B-4. Least-Squares Evaluation of Dosimetry in Surveillance Capsule X (40°,0° Azimuth, Core Midplane); Cycles 1 through 20 Irradiation

Reaction	Reaction Rate			M/C	BE/M	BE/C
	Measured (M)	Calculated (C)	Best-Estimate (BE)			
Cu-63(n,α)Co-60	6.53E-17	5.59E-17	6.35E-17	1.17	0.97	1.14
Fe-54(n,p)Mn-54	6.50E-15	6.47E-15	6.90E-15	1.00	1.06	1.07
Ni-58(n,p)Co-58	1.02E-14	8.92E-15	9.82E-15	1.15	0.96	1.10
U-238(Cd)(n,f)Cs-137	3.26E-14	3.23E-14	3.44E-14	1.01	1.05	1.07
Np-237(Cd)(n,f)Cs-137	2.82E-13	2.51E-13	2.73E-13	1.12	0.97	1.09
Co-59(n,g)Co-60	4.82E-12	4.52E-12	4.81E-12	1.07	1.00	1.06
Co-59(Cd)(n,g)Co-60	2.11E-12	2.10E-12	2.12E-12	1.01	1.00	1.01
Average of Fast Energy Threshold Reactions				1.09	1.00	1.09
% Standard Deviation				7.3	4.9	2.6

Integral Quantity	Calculated (C)	% Unc.	Best-Estimate (BE)	% Unc.	BE/C
Fluence rate E > 1.0 MeV (n/cm^2-s)	9.30E+10	13	9.85E+10	6	1.06
Fluence rate E > 0.1 MeV (n/cm^2-s)	3.09E+11	-	3.27E+11	10	1.06
dpa/s	1.54E-10	13	1.64E-10	7	1.06

APPENDIX C DOSIMETRY COMPARISONS FOR EX-VESSEL CAPSULES IRRADIATED DURING CYCLES 16-29

The results of the dosimetry evaluations performed for the H. B. Robinson Unit 2 EVND capsules withdrawn at end-of-Cycle 29 from locations near the top and bottom of the active core region are provided in this appendix. The data tabulations for the capsules irradiated in Cycles 16-29 include the following information:

1. The measured, calculated, and best-estimate reaction rates (rps/atom) for each sensor.
2. The M/C, BE/M, and BE/C reaction rate ratios for each sensor.
3. The calculated and best-estimate values of fast neutron fluence rate (E > 1.0 MeV and E > 0.1 MeV) and iron atom displacement rate with associated uncertainties.
4. The BE/C ratio for both fast neutron fluence rate (E > 1.0 MeV and E > 0.1 MeV) and iron displacements per atom per second.

The M/C and BE/M ratios for the individual sensors establish a comparison between measurement and calculation before and after the least-squares evaluation. The improvement in these reaction rate ratios for the best-estimate case is an indication of the improvement in the neutron spectrum and corresponding reduction in uncertainty brought about by the application of the least-squares procedure. The comparisons of calculated and best-estimate values of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate with associated uncertainties also provided an indication of the improved results obtained with the least-squares procedure.

Table C-1. Least-Squares Evaluation of Dosimetry in EVND Capsule U (30-Degree, 168.59 cm above Core Midplane, Cycles 16-29 Irradiation)

Reaction	Reaction Rate [rps/atom]			M/C	BE/M	BE/C
	Measured (M)	Calculated (C)	Best-Estimate (BE)			
Cu-63(n,α)Co-60	6.11E-20	8.17E-20	6.39E-20	0.75	1.05	0.78
Ti-46(n,p)Sc-46	9.95E-19	1.13E-18	9.47E-19	0.88	0.95	0.84
Fe-54(n,p)Mn-54	5.45E-18	6.84E-18	5.81E-18	0.80	1.07	0.85
Ni-58(n,p)Mn-54	9.22E-18	1.01E-17	9.03E-18	0.91	0.98	0.89
Co-59(n,γ)Co-60	2.13E-14	5.09E-14	2.27E-14	0.42	1.07	0.45
Co-59(Cd)(n,γ)Co-60	1.64E-14	2.19E-14	1.58E-14	0.75	0.96	0.72
U-238(n,f)Cs-137	4.55E-17	4.41E-17	4.17E-17	1.03	0.92	0.95
Np-237(n,f)Cs-137	1.22E-15	1.02E-15	1.12E-15	1.19	0.92	1.09
Average of Fast Energy Threshold Reactions				0.93	0.98	0.90
% Standard Deviation				17.4	6.6	12.1
Integral Quantity	**Calculated**	**% Unc.**	**Best-Estimate (BE)**	**% Unc.**	**BE/C**	
Fluence Rate E > 1.0 MeV (n/cm^2-s)	1.69E+08	13	1.68E+08	6	0.99	
Fluence Rate E > 0.1 MeV (n/cm^2-s)	2.75E+09	-	2.83E+09	11	1.03	
dpa/s	8.68E-13	13	8.76E-13	10	1.01	

Table C-2. Least-Squares Evaluation of Dosimetry in EVND Capsule V (30-Degree, 131.45 cm below Core Midplane, Cycles 16-29 Irradiation)

Reaction	Reaction Rate [rps/atom]			M/C	BE/M	BE/C
	Measured (M)	Calculated (C)	Best-Estimate (BE)			
Cu-63(n,α)Co-60	1.52E-19	1.40E-19	1.53E-19	1.09	1.01	1.09
Ti-46(n,p)Sc-46	2.26E-18	1.94E-18	2.18E-18	1.16	0.96	1.12
Fe-54(n,p)Mn-54	1.20E-17	1.14E-17	1.26E-17	1.05	1.06	1.11
Ni-58(n,p)Mn-54	1.90E-17	1.67E-17	1.89E-17	1.14	1.00	1.13
Co-59(n,γ)Co-60	4.48E-14	5.85E-14	4.54E-14	0.77	1.01	0.78
Co-59(Cd)(n,γ)Co-60	2.39E-14	2.57E-14	2.39E-14	0.93	1.00	0.93
U-238(n,f)Cs-137	8.83E-17	7.03E-17	8.18E-17	1.26	0.93	1.16
Np-237(n,f)Cs-137	2.04E-15	1.48E-15	1.90E-15	1.38	0.93	1.28
Average of Fast Energy Threshold Reactions				1.18	0.98	1.15
% Standard Deviation				10.3	5.2	6.0
Integral Quantity	**Calculated**	**% Unc.**	**Best-Estimate (BE)**	**% Unc.**	**BE/C**	
Fluence Rate E > 1.0 MeV (n/cm^2-s)	2.63E+08	13	3.13E+08	6	1.19	
Fluence Rate E > 0.1 MeV (n/cm^2-s)	3.80E+09	-	4.58E+09	11	1.21	
dpa/s	1.21E-12	13	1.45E-12	10	1.19	

APPENDIX D DOSIMETRY DATA FOR THE EX-VESSEL GRADIENT CHAIN IRRADIATED DURING CYCLES 16-29

The results of the dosimetry evaluations performed for the H.B. Robinson Unit 2 ex-vessel gradient chains are provided in Table E-1 of this appendix. The data tabulations for each chain evaluation include the following information:

- Measured and Calculated reaction rates for each of the three measured reactions

Table D-1. Comparison of Measured and Calculated Reaction Rates from the H.B. Robinson Unit 2 Ex-Vessel Neutron Dosimetry Set Gradient Chain Samples

Distance from Midplane (feet)	M/C 300° Chain					
	Fe-54 (n,p) Mn-54		Ni-58 (n,p) Co-58		Co-59 (n,γ) Co-60	
	Measured Reaction Rate (rps/atom)	Calculated Reaction Rate (rps/atom)	Measured Reaction Rate (rps/atom)	Calculated Reaction Rate (rps/atom)	Measured Reaction Rate (rps/atom)	Calculated Reaction Rate (rps/atom)
6	3.07E-18	4.66E-18	5.84E-18	6.93E-18	1.64E-14	3.62E-14
5	7.00E-18	7.93E-18	1.40E-17	1.17E-17	3.25E-14	5.04E-14
4	1.53E-17	1.16E-17	2.33E-17	1.71E-17	5.14E-14	6.77E-14
3	1.74E-17	1.28E-17	2.15E-17	1.89E-17	7.00E-14	7.78E-14
2	1.97E-17	1.32E-17	1.85E-17	1.96E-17	7.77E-14	8.30E-14
1	1.73E-17	1.33E-17	2.39E-17	1.97E-17	6.87E-14	8.46E-14
0	1.99E-17	1.34E-17	2.56E-17	1.98E-17	5.65E-14	8.42E-14
-1	1.43E-17	1.33E-17	1.65E-17	1.97E-17	5.49E-14	8.13E-14
-2	1.81E-17	1.31E-17	2.65E-17	1.94E-17	5.67E-14	7.59E-14
-3	1.55E-17	1.28E-17	2.34E-17	1.88E-17	5.11E-14	6.60E-14
-4	1.58E-17	1.18E-17	2.26E-17	1.73E-17	4.51E-14	5.63E-14
-5	1.18E-17	9.31E-18	1.59E-17	1.36E-17	3.45E-14	4.34E-14
-6	5.69E-18	4.76E-18	8.26E-18	7.02E-18	1.78E-14	2.85E-14