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Serial: RNP-RA/17-0082
February 7, 2018

10 CFR 50.90

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

SUBJECT: LICENSE AMENDMENT REQUEST PROPOSING TO REVISE TECHNICAL SPECIFICATION 3.4.3, "RCS PRESSURE AND TEMPERATURE (P/T) LIMITS"

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy) hereby submits a license amendment request (LAR) for H.B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP). The proposed change will revise HBRSEP Technical Specification (TS) 3.4.3, "RCS Pressure and Temperature (P/T) Limits," to reflect that Figures 3.4.3-1 and 3.4.3-2 (P/T limit curves) are applicable up to 46.3 effective full power years (EFPY) instead of 50 EFPY with the removal of part length shield assemblies (PLSAs) and migration to 24-month fuel cycles. At the beginning of HBRSEP Cycle 32, PLSAs will no longer be used and 24-month cycles will go into effect.

The Enclosure provides a description and assessment of the proposed change. Attachment 1 provides the existing HBRSEP TS pages marked to show the proposed change. Attachment 2 provides a non-proprietary version of the Westinghouse report, WCAP-18215, "H.B. Robinson Unit 2 End-of-License Extension Reactor Vessel Integrity Evaluations and Feasibility Study."

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed change involves no significant hazards consideration. The bases for these determinations are included in the Enclosure.

Duke Energy requests approval of the proposed amendment within one year of the date of this submittal. Once approved, Duke Energy will implement the license amendment within 120 days.

There are no new regulatory commitments contained in this letter.

In accordance with 10 CFR 50.91, Duke Energy is notifying the State of South Carolina of this license amendment request by transmitting a copy of this letter and enclosure to the designated State Official.

If there are any questions or if additional information is needed, please contact Mr. Kevin Ellis, Manager - Regulatory Affairs at 843-951-1329 or Kevin.Ellis@duke-energy.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 7, 2018.

Sincerely,

Ernest J. Kapopoulos, Jr.
Site Vice President

EJK/jlv

Enclosure: Description and Assessment of the Proposed Change

Attachments:

1. Technical Specification Page Markups
2. Westinghouse Report WCAP-18215, "H.B. Robinson Unit 2 End-of-License Extension Reactor Vessel Integrity Evaluations and Feasibility Study" (Non-Proprietary Class 3)

cc (with Enclosure/Attachments):

C. Haney, USNRC Region II – Regional Administrator
J. Rotton, USNRC Senior Resident Inspector – RNP
D. Galvin, NRR Project Manager – RNP
A. Gantt, Chief, Bureau of Radiological Health (SC)
A. Wilson, Attorney General (SC)
S. E. Jenkins, Manager, Radioactive and Infectious Waste Management (SC)

bcc (with Enclosure/Attachments):

Chris Nolan
Art Zaremba
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Ernie Kapopoulos
Jim Kammer
Chuck Sherman
Kevin Ellis
Heidi Walters (For RNP Licensing/Nuclear Records Files)

ENCLOSURE

Description and Assessment of the Proposed Change

Subject: License Amendment Request Proposing to Revise Technical Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits"

ATTACHMENTS:

1. Technical Specification Page Markups
2. Westinghouse report, WCAP-18215, "H.B. Robinson Unit 2 End-of-License Extension Reactor Vessel Integrity Evaluations and Feasibility Study" (Non-Proprietary Class 3)

1. SUMMARY DESCRIPTION

Duke Energy Progress, LLC (Duke Energy) hereby submits a license amendment request (LAR) for H.B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP). The proposed change will revise HBRSEP Technical Specification (TS) 3.4.3, “RCS Pressure and Temperature (P/T) Limits,” to reflect that Figures 3.4.3-1 and 3.4.3-2 (P/T limit curves) are applicable up to 46.3 effective full power years (EFPY) instead of 50 EFPY with the removal of part length shield assemblies (PLSAs) and migration to 24-month fuel cycles. At the beginning of HBRSEP Cycle 32, PLSAs will no longer be used and 24-month cycles will go into effect.

2. DETAILED DESCRIPTION

2.1 System Design and Operation

All components of the HBRSEP Reactor Coolant System (RCS) are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients and reactor trips. HBRSEP is required to limit the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation.

The HBRSEP TSs contain P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing and data for the maximum rate of change of reactor coolant temperature. Each P/T limit curve defines an acceptable region for normal operation. The typical use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

HBRSEP Updated Final Safety Analysis Report Section 5.3.2 (Reference 1) provides additional details regarding the methodology used to develop the P/T limit curves that are contained in TSs and are the subject of this amendment request.

2.2 Current Technical Specifications Requirements

HBRSEP Limiting Condition for Operation (LCO) 3.4.3 requires that RCS pressure, RCS temperature and RCS heatup and cooldown rates be maintained within the limits specified in Figures 3.4.3-1 and 3.4.3-2 (P/T curves) that are associated with this proposed change. The two elements of LCO 3.4.3 are:

- The limit curves for heatup, cooldown and ISLH testing; and
- Limits on the rate of change of temperature.

LCO 3.4.3 limits apply to all components of the RCS, except the pressurizer and define allowable operating regions and permit a large number of operating cycles while providing a wide margin to non-ductile failure. Violating LCO 3.4.3 limits would result in placing the reactor vessel outside of the bounds of the stress analyses and could increase stresses in other reactor coolant pressure boundary (RCPB) components.

2.3 Reason for the Proposed Change

Westinghouse recently issued technical report WCAP-18215, "H.B. Robinson Unit 2 End-of-Life License Extension Reactor Vessel Integrity Evaluations and Feasibility Study" (see Attachment 2) (Reference 1) to Duke Energy that includes an evaluation of the HBRSEP reactor pressure vessel (RPV) with respect to reactor vessel integrity (RVI) under the scenario that includes a migration to 24-month fuel cycles and no longer using PLSAs. At the beginning of HBRSEP Cycle 32, PLSAs will no longer be used and 24-month cycles will go into effect. Within the evaluation provided in Attachment 2, Adjusted Reference Temperature (ART) values were calculated at 50 EFPY, which is deemed end-of-license extension for HBRSEP. The 50 EFPY ART values were used to perform an applicability check on the existing P/T limit curves for HBRSEP that are in TS 3.4.3. Consideration of the fluence values provided under the scenario that includes 24-month fuel cycles and no use of PLSAs indicates that a reduction of the applicability term for the existing HBRSEP 50 EFPY P/T limit curves is required. It was determined that the HBRSEP P/T limit curves will require revision after 46.3 EFPY.

Therefore, for the existing P/T limit curves in TS 3.4.3 (i.e., Figures 3.4.3-1 and 3.4.3-2) to remain valid, the applicability term must be reduced form 50 EFPY to 46.3 EFPY.

2.4 Description of the Proposed Change

TS 3.4.3, Figure 3.4.3-1, "Reactor Coolant System Heatup Limits Applicable Up to 50 EFPY," is revised as follows:

- The words "Limiting ART Values at 50 EFPY" are revised to state "Limiting ART Values at 46.3 EFPY".

- The words "Curves applicable for heatup rates up to 60°F/Hr for service period up to 50 EFPY" are revised to state "Curves applicable for heatup rates up to 60°F/Hr for service period up to 46.3 EFPY".

- The words "Criticality Limit based on Inservice Hydrostatic test temperature (221°F) for the service period up to 50 EFPY" are revised to state "Criticality Limit based on Inservice Hydrostatic test temperature (221°F) for the service period up to 46.3 EFPY".

- The title of Figure 3.4.3-1 ("Reactor Coolant System Heatup Limits Applicable Up to 50 EFPY") is revised to state "Reactor Coolant System Heatup Limits Applicable Up to 46.3 EFPY".

TS 3.4.3, Figure 3.4.3-2, "Reactor Coolant System Cooldown Limitations Applicable Up to 50 EFPY," is revised as follows:

- The words "Limiting ART Values at 50 EFPY" are revised to state "Limiting ART Values at 46.3 EFPY".

- The words "Curves applicable for cooldown rates up to 100°F/Hr for the service period up to 50 EFPY." are revised to state "Curves applicable for cooldown rates up to 100°F/Hr for the service period up to 46.3 EFPY."

- The title of Figure 3.4.3-2 ("Reactor Coolant System Cooldown Limitations Applicable Up to 50 EFPY") is revised to state "Reactor Coolant System Cooldown Limitations Applicable Up to 46.3 EFPY".

3. TECHNICAL EVALUATION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting material in the beltline region of the RPV. The most limiting RT_{NDT} of the material in the core (beltline) region of the RPV was determined by using the unirradiated RPV material fracture toughness properties and estimating the irradiation-induced shift (Δ RT_{NDT}).

RT_{NDT} increases as the material is exposed to fast-neutron irradiation; therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, Δ RT_{NDT} due to the radiation exposure associated with that time period was added to the original unirradiated Δ RT_{NDT}. Using the ART values, P/T limit curves were determined in accordance with the requirements of 10 CFR 50, Appendix G, as augmented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code.

P/T limit curves for normal heatup and cooldown of the primary RCS for HBRSEP were originally developed in Westinghouse report WCAP-15827 for 50 EPFY (Reference 2). Report WCAP-15827 was previously provided to the NRC staff in a HBRSEP submittal dated November 2, 2015 (Reference 3). However, these curves were developed using only a subset of the currently available surveillance data, outdated fluence values and some material properties differing from those that are included within Attachment 2 of this submittal.

To confirm the existing applicability of the 50 EFPY P/T limit curves developed in WCAP-15827 for HBRSEP, the limiting reactor vessel material ART values from the beltline (with consideration of the updated fluence values, one revised chemistry value factor and recalculated initial Δ RT_{NDT} values for the nozzle forging materials) must be shown to be less than or equal to the limiting beltline material ART values used in development of the existing P/T limit curves contained in WCAP-15827 at 50 EFPY. For the scenario that uses 24-month fuel cycles and the removal of PLSAs (i.e., the configuration of HBRSEP beginning in Cycle 32), the Regulatory Guide 1.99, Revision 2 methodology was used along with the fluence values documented in Section 2 of Attachment 2 (Reference 1) of this submittal to calculate ART values for the HBRSEP reactor vessel materials at 50 EFPY. The ART calculations are summarized in several tables in Section 6 of Attachment 2. The summary of the limiting ART values used in the applicability evaluation for the scenario that applies 24-month fuel cycles and the removal of PLSAs is provided in Table 6.1-5 of Attachment 2 (Reference 1).

Comparison of the limiting ART values calculated as part of the end-of-license extension RPV integrity evaluation in Attachment 2, taking into consideration the fluence values provided for the scenario with 24-month fuel cycles and removal of PLSAs, to those used in calculation of the existing P/T limit curves, indicates that a reduction to the applicability term of the existing HBRSEP 50 EFPY P/T limit curves documented in WCAP-15827 is required. For the scenario evaluated in Attachment 2 with 24-month fuel cycles and removal of PLSAs, the HBRSEP P/T limit curves would require revision after 46.3 EFPY. Since HBRSEP is migrating to 24-month fuel cycles and will be removing PLSAs, the existing P/T limit curves in TSs will remain exactly the same, except for the term to which they apply (proposed to be 46.3 EFPY instead of 50 EFPY as demonstrated in Section 2.4 above).

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The following regulatory requirements and guidance documents are applicable to the proposed change.

10 CFR 50.36

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, “Technical specifications,” establishes the requirements related to the content of the TSs. Pursuant to 10 CFR 50.36(c) TSs will include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) LCOs, (3) SRs, (4) design features; and (5) administrative controls.

HBRSEP LCO 3.4.3 limits the pressure and temperature changes during RCS heatup and cooldown (i.e., to the right and below the P/T curves in Figures 3.4.3-1 and 3.4.3-2), to prevent non-ductile RPV failure. The proposed change reduces the applicability of the P/T limits to 46.3 EFPY, which does impact the function of the LCO to limit RCS operation to within approved P/T limits. Based on the determination that the existing Figures 3.4.3-1 and 3.4.3-2 are acceptable up to 46.3 EFPY (see Section 3 above and Attachment 2), Duke Energy concludes that LCO 3.4.3 will continue to meet 10 CFR 50.36(c)(2)(i) with the proposed change by providing for the requisite functional capability or performance level required for safe operation.

HBRSEP Surveillance Requirement (SR) 3.4.3.1 verifies that RCS operation is within the limits of Figures 3.4.3-1 and 3.4.3-2 when RCS pressure and temperature conditions are undergoing planned changes. The proposed change reduces the applicability of the P/T limits to 46.3 EFPY, which does not impact the function of SR 3.4.3.1 to verify that the RCS is operated within approved P/T limits. Therefore, Duke Energy concludes Figures 3.4.3-1 and 3.4.3-2 (with a reduction in applicability of the P/T limits) in SR 3.4.3.1 continues to meet 10 CFR 50.36(c)(3) by providing sufficient test, calibration or inspection requirements to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits and that the LCO will be met.

HBRSEP TS 3.4.12, “Low Temperature Overpressure Protection (LTOP) System,” addresses the condition when the accumulator is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in TS 3.4.3, Figures 3.4.3-1 and 3.4.3-2. This includes the Applicability Note in LCO 3.4.12, Condition C and Required Action C.1, and Condition D and Required Action D.2. LCO 3.4.12 provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. The Notes permit accumulator surveillances to be performed when the accumulator pressure does not exceed the P/T limits. Action statements C and D address an improperly isolated accumulator. The proposed change in this LAR reduces the applicability of the P/T limit curves to 46.3 EFPY, which does not impact the Conditions and Required Actions in LCO 3.4.12. Therefore, Duke Energy concludes the use of the proposed Figures 3.4.3-1 and 3.4.3-2 in TS 3.4.12 with an applicability term of 46.3 EFPY continues to meet 10 CFR 50.36(c)(2)(i) by providing for the requisite functional capability or performance level required for safe operation.

10 CFR 50.60

Section 50.60 of 10 CFR, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation,” imposes fracture toughness and material surveillance program requirements, which are set forth in 10 CFR 50, Appendices G, “Fracture Toughness Requirements,” and H, “Reactor Vessel Material Surveillance Program Requirements.” The proposed change only reduces the applicability term of the P/T limit curves to 46.3 EFPY and does not propose to revise any other aspects of the limit curves. The existing curves will meet the 10 CFR 50.60 requirements up to 46.3 EFPY. Therefore, the proposed change to reduce the applicability term to 46.3 EFPY is consistent with the requirements of 10 CFR 50.60.

10 CFR 50, Appendix G

Appendix G to 10 CFR 50 requires that the P/T limits for the facility’s reactor pressure vessel (RPV) be at least as conservative as those obtained by following the linear elastic fracture mechanics methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The P/T limits in TS 3.4.3 are unchanged. Only the applicability term is proposed to be reduced to 46.3 EFPY instead of 50 EFPY. Therefore, Duke Energy concludes for the proposed change that the HBRSEP RPV will continue to meet RPV integrity regulatory requirements through 46.3 EFPY.

10 CFR 50, Appendix H

Appendix H to 10 CFR 50 establishes requirements for each facility related to its RPV material surveillance. These regulatory requirements will continue to be met for the proposed change with the surveillance capsule removal schedule prescribed in Section 7 (Table 7-2) of Attachment 2 to this submittal.

Regulatory Guide (RG) 1.99, Revision 2

RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” contains guidance on methodologies the NRC considers acceptable for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. This RG was used for the calculation of ART values at the 1/4T and 3/4T locations that were previously calculated in Reference 2 and are shown on the existing HBRSEP P/T limit curves. The proposed change does not alter the existing ART values in Figures 3.4.3-1 and 3.4.3-2. The proposed change only reduces the applicability term to 46.3 EFPY. Therefore, the proposed change has no effect on the application of RG 1.99, Revision 2.

RG 1.190

RG. 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated March 2001, describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence. The analysis methodologies used to calculate the HBRSEP RPV fluences for the scenario associated with the proposed change that includes 24-month fuel cycles and removal of PLSAs satisfy the guidance set forth in RG 1.190.

UFSAR Section 3.1

The General Design Criteria (GDC) in existence at the time HBRSEP was licensed for operation (July 1970) were contained in Proposed Appendix A to 10 CFR 50, “General Design Criteria for Nuclear Power Plants,” published in the Federal Register on July 11, 1967.

HBRSEP UFSAR Section 3.1.2.9, “Reactor Coolant Pressure Boundary” (GDC 9), states that:

> The reactor coolant pressure boundary (RCPB) shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrollable leakage throughout its design lifetime.

The proposed change to revise TS 3.4.3 to reflect that Figures 3.4.3-1 and 3.4.3-2 (P/T limit curves) are applicable up to 46.3 EFPY with the removal of PLSAs and migration to 24-month fuel cycles does not affect HBRSEP’s compliance with the intent of UFSAR Section 3.1.2.9 (i.e., July 1967 GDC 9).

HBRSEP UFSAR Section 3.1.2.34, “RCPB Rapid Propagation Failure Prevention” (GDC 34), states that:

> The RCPB shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given:
>
> a) To the provisions for control over service temperature and irradiation effects which may require operational restrictions
>
> b) To the design and construction of the reactor pressure vessel (RPV) in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range, and for absorption of energy by plastic deformation
>
> c) To the design and construction of RCPB piping and equipment in accordance with applicable codes.

The proposed change to revise TS 3.4.3 to reflect that Figures 3.4.3-1 and 3.4.3-2 (P/T limit curves) are applicable up to 46.3 EFPY with the removal of PLSAs and migration to 24-month fuel cycles does not affect HBRSEP’s compliance with the intent of UFSAR Section 3.1.2.34 (i.e., July 1967 GDC 34).

The proposed change does not affect plant compliance with any of the above regulations or guidance and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

4.2 No Significant Hazards Consideration Determination

Duke Energy requests approval of a change to the H.B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP) Technical Specifications (TS). The proposed change will revise HBRSEP TS 3.4.3, “RCS Pressure and Temperature (P/T) Limits,” to reflect that Figures 3.4.3-1 and 3.4.3-2 (P/T limit curves) are applicable up to 46.3 effective full power years (EFPY) instead of 50 EFPY with the removal of part length shield assemblies (PLSAs) and migration to 24-month fuel

cycles. At the beginning of HBRSEP Cycle 32, PLSAs will no longer be used and 24-month cycles will go into effect.

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of Amendment,” as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change revises TS 3.4.3 to reflect that Figures 3.4.3-1 and 3.4.3-2 (P/T limit curves) are applicable up to 46.3 EFPY instead of 50 EFPY with the removal of PLSAs and migration to 24-month fuel cycles. The proposed change does not involve physical changes to the plant or alter the reactor coolant system (RCS) pressure boundary (i.e., there are no changes in operating pressure, materials or seismic loading). The P/T limit curves and Adjusted Reference Temperature (ART) values will remain as-is. Only the term to which the limit curves applies is effected by the proposed change. The P/T limit curves in TS 3.4.3 with an applicability term of 46.3 EFPY provide continued assurance that the fracture toughness of the reactor pressure vessel (RPV) is consistent with analysis assumptions and NRC regulations. The methodology used to develop the existing P/T limit curves provides assurance that the probability of a rapidly propagating failure will be minimized. The P/T limit curves, with the applicability term reduced to a proposed 46.3 EFPY, will continue to prohibit operation in regions where it is possible for brittle fracture of reactor vessel materials to occur, thereby assuring that the integrity of the RCS pressure boundary is maintained.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change revises TS 3.4.3 to reflect that Figures 3.4.3-1 and 3.4.3-2 (P/T limit curves) are applicable up to 46.3 EFPY instead of 50 EFPY with the removal of PLSAs and migration to 24-month fuel cycles. The proposed change does not affect the design or assumed accident performance of any structure, system or component, or introduce any new modes of system operation or failure modes. Compliance with the proposed P/T curves (same as the existing P/T curves with the applicability term reduced to 46.3 EFPY) will provide sufficient protection against brittle fracture of reactor vessel materials to assure that the RCS pressure boundary performs as previously evaluated.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change revises TS 3.4.3 to reflect that Figures 3.4.3-1 and 3.4.3-2 (P/T limit curves) are applicable up to 46.3 EFPY instead of 50 EFPY with the removal of PLSAs and migration to 24-month fuel cycles. HBRSEP adheres to applicable NRC regulations (i.e., 10 CFR 50, Appendices G and H) and NRC-approved methodologies (i.e., Regulatory Guides 1.99 and 1.190) with respect to the P/T limit curves in TS 3.4.3 in order to provide an adequate margin of safety to the conditions at which brittle fracture may occur. The P/T limit curves, with the applicability term reduced to 46.3 EFPY, continue to provide assurance that the established P/T limits are not exceeded.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

4.3 Conclusions

In conclusion, based on the considerations discussed above: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission’s regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. Westinghouse Report WCAP-18215, Revision 0, *H.B. Robinson Unit 2 End-of-License Extension Reactor Vessel Integrity Evaluations and Feasibility Study*, March 2017.

2. Westinghouse Report WCAP-15827, Revision 0, *H.B. Robinson Unit 2 Heatup and Cooldown Limit Curves for Normal Operation*, March 2003.

3. Duke Energy letter, *Request for Technical Specification Change to Reactor Coolant System Pressure and Temperature Limits*, November 2, 2015 (ADAMS Accession No. ML15307A069).

Attachment 1

Technical Specification Page Markups

MATERIALS PROPERTIES BASE
Controlling Material: Upper Shell Plate W10201-1 & Girth Weld 10-273
Limiting ART Values at ~~50~~ 46.3 EFPY: 1/4T, 172°F & 263°F
3/4T, 153°F & 191°F

Curves applicable for heatup rates up to 60°F/Hr for service period up to ~~50~~ 46.3 EFPY
Heatup Curves include +20°F and -80 psig Allowance for Instrumentation error.

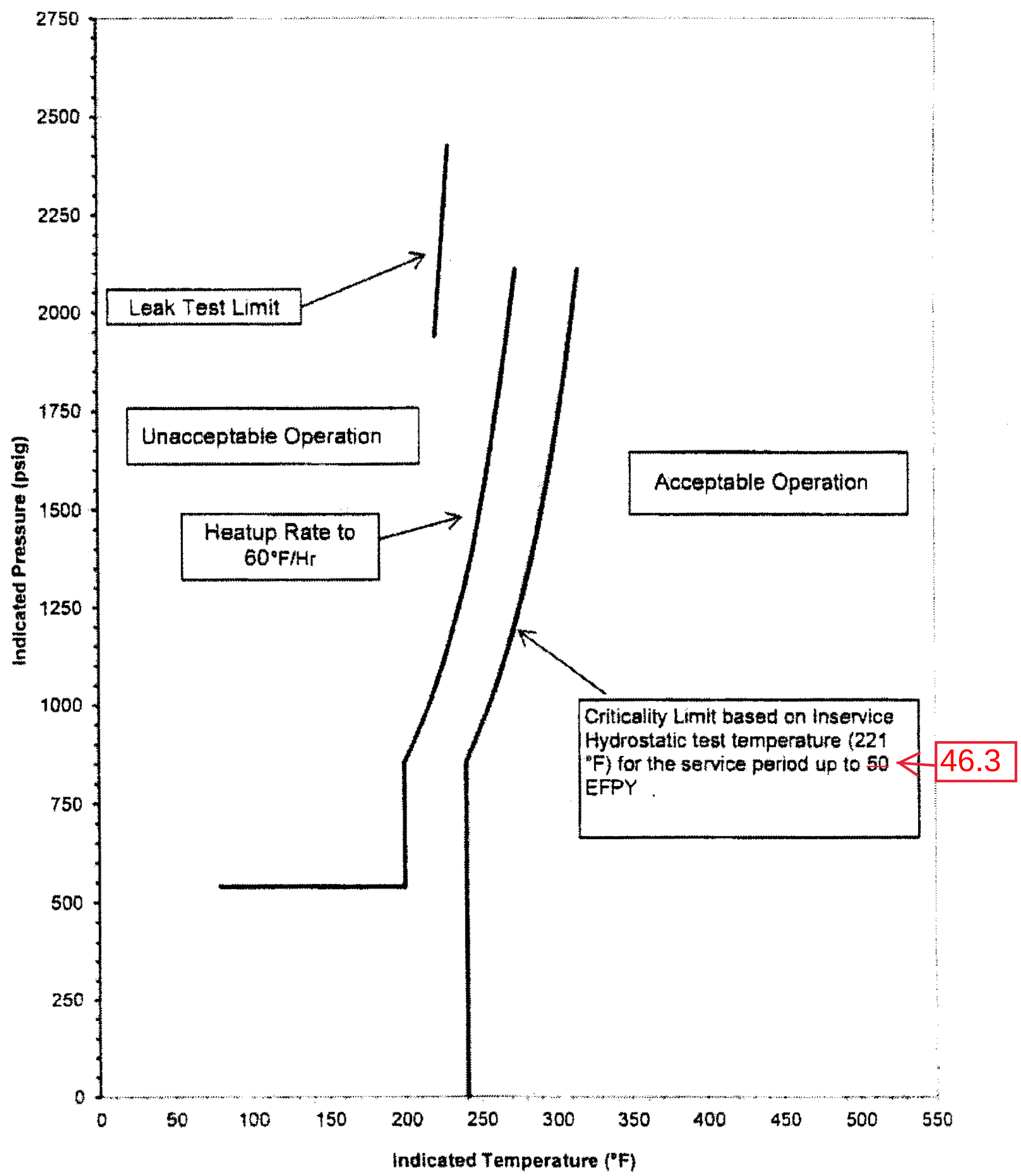

Figure 3.4.3-1
Reactor Coolant System Heatup Limits
Applicable Up to ~~50~~ 46.3 EFPY

MATERIALS PROPERTIES BASE
Controlling Material. Upper Shell Plate W10201-1 & Girth Weld 10-273
Limiting ART Values at ~~50~~ 46.3 EFPY: 1/4T, 172°F & 263°F
3/4T, 153°F & 191°F

Curves applicable for cooldown rates up to 100°F/Hr
for the service period up to ~~50~~ 46.3 EFPY.
Curves include +20°F and -80 PSIG Allowance for
Instrumentation error.

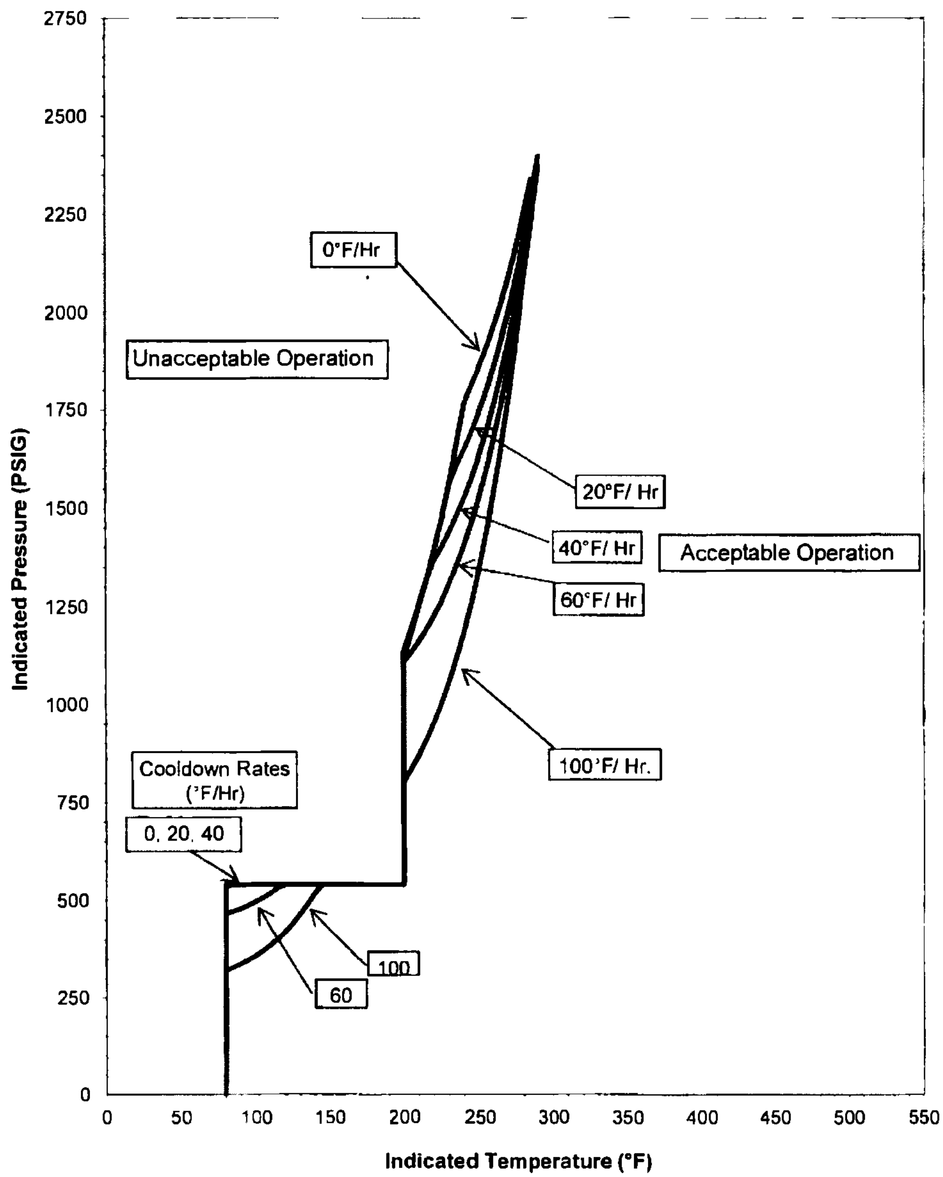

Figure 3.4.3-2
Reactor Coolant System Cooldown Limitations
Applicable Up to ~~50~~ 46.3 EFPY

Attachment 2

Westinghouse Report WCAP-18215, "H.B. Robinson Unit 2 End-of-License Extension Reactor Vessel Integrity Evaluations and Feasibility Study" (Non-Proprietary Class 3)

WCAP-18215-NP
Revision 0

March 2017

H.B. Robinson Unit 2 End-of-License Extension Reactor Vessel Integrity Evaluations and Feasibility Study

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-18215-NP
Revision 0

H.B. Robinson Unit 2 End-of-License Extension Reactor Vessel Integrity Evaluations and Feasibility Study

Michael R. Ickes*
Materials Center of Excellence

Sean T. Kinnas*
Nuclear Operations & Radiation Analysis

March 2017

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	Jianwei Chen* Nuclear Operations & Radiation Analysis		Laurent P. Houssay*, Manager Nuclear Operations & Radiation Analysis

*Electronically approved records are authenticated in the electronic document management system.

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RECORD OF REVISION

Revision 0: Original Issue

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EXECUTIVE SUMMARY

This report presents the evaluation of the H.B. Robinson Unit 2 reactor pressure vessel (RPV) with respect to Reactor Vessel Integrity (RVI) under two scenarios. Scenario # 1 represents current anticipated operation, which includes 18-month cycles and the use of part length shield assemblies (PLSAs). Scenario # 2 represents a case study, which includes 24-month cycles and no use of PLSAs both starting at the beginning of Cycle 32. H.B. Robinson Unit 2 has been approved for license extension to a total of 60 calendar years of operation. The purpose of this report is to document RVI evaluations applicable through 60 years of operation for both Scenario #1 and Scenario #2. The evaluations in this report projected for 60 years of operation are applicable through 50 effective full power years (EFPY), which is deemed end-of-license extension (EOLE).

A summary of results for the H.B. Robinson Unit 2 RVI evaluation under either scenario is provided below. Based on the results presented herein, it is concluded that the H.B. Robinson Unit 2 RPV will continue to meet RPV integrity regulatory requirements through the extended period of operation under Scenario #1. Scenario #2 will require P-T limit curves to be re-evaluated before 50 EFPY.

Neutron Fluence

The RPV beltline neutron fluence values applicable through a 20-year license renewal period were calculated for the H.B. Robinson Unit 2 RPV materials under both operational scenarios. The analysis methodologies used to calculate the H.B. Robinson Unit 2 RPV fluences satisfy the guidance set forth in Regulatory Guide 1.190. See Section 2 for more details.

Fracture Toughness and Material Properties

Fracture toughness properties of the H.B. Robinson Unit 2 RPV beltline materials are summarized herein. See Section 3 for more details.

EOLE Pressurized Thermal Shock

The RT_{PTS} values of all of the beltline materials in the H.B. Robinson Unit 2 RPV are below the RT_{PTS} screening criteria of 270°F for base metal and/or longitudinal welds, and 300°F for circumferentially oriented welds (per 10 CFR 50.61), through EOLE (50 EFPY) under either operational scenario. However, Scenario #1 maintains additional margin to the screening criteria. See Section 4 for more details.

EOLE Emergency Response Guideline Limits

The Emergency Response Guideline (ERG) limits were developed in order to establish guidance for operator action in the event of an emergency situation, such as a PTS event. Under both operational scenarios, H.B. Robinson Unit 2 remains in ERG Category IIIb through EOLE. See Section 4 for more details.

EOLE Upper-Shelf Energy

Under both scenarios, multiple materials drop below the 50 ft-lb USE screening criterion per 10 CFR 50, Appendix G. However, the upper-shelf energy (USE) values of all of the beltline materials in the H.B. Robinson Unit 2 RPV are projected to remain above the required USE value of 42 ft-lbs per WCAP-13587 Revision 1 for a three-loop Westinghouse designed plant under either operational scenario. Scenario #1 maintains additional margin to the USE requirement. See Section 5 for more details.

EOLE Adjusted Reference Temperatures

Adjusted Reference Temperatures (ART) values are calculated at 50 EFPY for all H.B. Robinson Unit 2 beltline materials. The 50 EFPY ART values are used to perform an applicability check on the existing pressure-temperature (P-T) limit curves for H.B. Robinson Unit 2 documented in WCAP-15827. With consideration of the fluence values provided for Scenario #1 it is concluded that this scenario requires no reduction of the applicability term of the existing H.B. Robinson Unit 2 50 EFPY P-T limit curves.

Consideration of the Scenario #2 fluence values indicates that a reduction of the applicability term of the existing H.B. Robinson Unit 2 50 EFPY P-T limit curves is required. Under Scenario #2, the H.B. Robinson Unit 2 P-T limit curves will require revision after 46.3 EFPY.

Surveillance Capsule Withdrawal Schedule

H.B. Robinson Unit 2 has pulled and tested four surveillance capsules. In order to satisfy the surveillance capsule provisions of ASTM E185-82, as required by 10 CFR 50, Appendix H, through EOL (32 EFPY), one additional surveillance capsule must be withdrawn. With consideration of 60 years of operation (50 EFPY), five total capsules must be withdrawn to satisfy ASTM E185-82 and to meet the recommendations of the Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 2. Recommendations for additional capsule withdrawals under both operational scenarios are provided in Section 7. See Section 7 for more details.

1 INTRODUCTION

The purpose of this report is to evaluate the beltline materials in the H.B. Robinson Unit 2 RPV with respect to RVI under two potential operational scenarios. These materials are evaluated to determine their RT_{PTS}, USE, and ART values at EOLE, which corresponds to 50 EFPY. The applicability period of the current H.B. Robinson Unit 2 P-T limit curves is also analyzed to ensure that the applicability period of the curves is not impacted as a result of fluence, CF, or initial RT_{NDT} updates.

The two operating scenarios considered are Scenario #1, where H.B. Robinson Unit 2 continues on 18-month fuel cycles with the current fuel design, and Scenario #2, where H.B. Robinson Unit 2 changes to 24-month fuel cycles and removes the PLSAs from the fuel design. For the purposes of this feasibility study, the change from Scenario #1 to Scenario #2 is assumed to occur after the completion of fuel Cycle 31 and implemented for fuel Cycle 32.

Historically, only those materials directly adjacent to the active core, commonly referred to as the traditional beltline, have been evaluated with respect to RVI. However, as plants, such as H.B. Robinson Unit 2, obtain License Renewal, additional materials are now being included in RVI analyses. The Generic Aging Lessons Learned (GALL) Report (NUREG-1801, Revision 2 [Reference 1]) states that any materials exceeding 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) must be monitored to evaluate the changes in fracture toughness. Any materials not previously evaluated for RVI that have predicted fluence levels greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) are now commonly referred to as the extended beltline. However, per Section 3, the traditional H.B. Robinson Unit 2 beltline region encompasses all of the necessary beltline materials.

Section 2 of this report discusses the methodologies used to evaluate the neutron fluence and presents the neutron fluence values. Section 3 identifies the materials in the H.B. Robinson Unit 2 reactor vessel beltline region and provides fracture toughness and material properties. RT_{PTS} and USE values are determined in Sections 4 – 5, respectively. The ART values and the P-T limit curve applicability period are determined in Section 6. Section 7 provides an updated surveillance capsule withdrawal schedule.

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

A discrete ordinates (S_N) transport analysis was performed for the H.B. Robinson Unit 2 reactor to determine the neutron radiation environment within the reactor pressure vessel. The projected exposure of the H. B. Robinson Unit 2 RPV and surveillance capsule locations was developed based on a series of fuel-cycle-specific neutron transport calculations. The methodology and calculations are substantially similar to those described in WCAP-18100-NP (Reference 2). However, beginning with Cycle 32 and continuing through Cycle 35, the calculations reflected the removal of the PLSAs and the transition to 24-month fuel cycles. Projections beyond the end of Cycle 35 to 50 EFPY also reflect PLSA removal and 24-month fuel cycles. In this section, the neutron transport methodology is discussed, and the calculated results applicable to the RPV and surveillance capsule locations are presented.

All of the calculations described in this section were based on nuclear cross-section data derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). Furthermore, the neutron transport evaluation methodologies follow the guidance of Regulatory Guide 1.190 (Reference 3). Additionally, the methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-14040-A, Revision 4 (Reference 4).

2.2 DISCRETE ORDINATES ANALYSIS

The methodology used to perform the cycle-specific forward transport calculations for fuel cycles 1 through 29 are described in WCAP-18100-NP (Reference 2). Fast neutron exposure values for Cycles 30 and 31 were projected using the average fluence values of cycles 27 through 29. In performing the fast neutron exposure evaluations for the H.B. Robinson Unit 2 reactor vessel for fuel cycles 32 and beyond, a series of fuel-cycle-specific forward transport calculations were performed using the following three-dimensional fluence rate synthesis technique:

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

where $\varphi(r,\theta,z)$ is the synthesized three-dimensional neutron fluence rate distribution, $\varphi(r,\theta)$ is the transport solution in r,θ geometry, $\varphi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\varphi(r)$ 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. This synthesis procedure was carried out for each operating cycle at H.B. Robinson Unit 2.

For this analysis, 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 all 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 conservative 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.

The locations of the H.B. Robinson Unit 2 vessel welds and plates are provided in Table 2-1. The axial position of each material is indexed to z = 0.0 cm, which corresponds to the midplane of the active fuel stack. Pressure vessel material maximum exposures in terms of fast neutron fluence (E > 1.0 MeV) and displaced iron atoms (dpa) for Scenario # 2 are given in Tables 2-2 through 2-8. Similar values for Scenario #1 are available in Section 2.4 of WCAP-18100-NP (Reference 2). Calculated values through Cycle 29 were taken from Section 2.4 of WCAP-18100-NP (Reference 2). Values herein for Cycles 30 and 31 were projected based on the average values from Cycles 27 through 29. Then, beginning with Cycle 32 and continuing through Cycle 35, calculations were performed reflecting the removal of the part length shield assemblies (PLSAs) and the transition to 24-month fuel cycles. Projections beyond the end of Cycle 35 to 50 EFPY also reflect PLSA removal and 24-month fuel cycles.

Additionally, 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).

Results of the discrete ordinates analyses pertinent to each of these in-vessel surveillance capsule locations are provided in Table 2-9 for Scenario #2. Similar values for Scenario #1 are available in Section 2.3 of WCAP-18100-NP (Reference 2). In Table 2-9, the calculated fuel-cycle-specific fast neutron fluence rate (E > 1.0 MeV) are provided at the geometric center of the five capsule radial locations described above at an elevation corresponding to the core midplane. Calculated values through Cycle 29 were taken from Table 2-6 of WCAP-18100-NP (Reference 2). Values herein for Cycles 30 and 31 were projected based on the average values from Cycles 27 through 29. Then, beginning with Cycle 32 and continuing through Cycle 35, calculations were performed reflecting the removal of the part length shield assemblies (PLSAs) and the transition to 24-month fuel cycles. Projections beyond the end of Cycle 35 to 50 EFPY also reflect PLSA removal and 24-month fuel cycles.

2.3 CALCULATIONAL UNCERTAINTIES

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

1. Comparison of calculations with benchmark measurements from the Pool Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL).

2. Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment.

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 plant-specific calculations with all available dosimetry results from the H.B. Robinson Unit 2 surveillance program.

The first phase of the methods qualification (PCA comparisons) 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 or geometric variables that impact power reactor calculations.

The second phase of the qualification (H.B. Robinson comparisons) addressed uncertainties in these additional areas that are primarily methods related and would tend to apply generically to all fast neutron exposure evaluations.

The third phase of the qualification (analytical sensitivity study) 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 input parameters. The overall calculational uncertainty applicable to the H.B. Robinson Unit 2 analysis was established from results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with H.B. Robinson Unit 2 measurements) was used solely to demonstrate the validity 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 in any way to modify the calculated surveillance capsule or pressure vessel neutron exposures.

Table 2-10 summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in WCAP-16083-NP, Revision 1 (Reference 7). The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random and no systematic bias was applied to the analytical results. The plant-specific measurement comparisons given in WCAP-18100-NP (Reference 2) support these uncertainty assessments for H.B. Robinson Unit 2.

Table 2-1 Pressure Vessel Material Locations

Material	Axial Location[(a)] (cm)	Azimuthal Location (Degrees)
Outlet Nozzle to Upper Shell Welds – Lowest Extent		
Nozzle 1	246.8	10
Nozzle 2	246.8	130
Nozzle 3	246.8	250
Inlet Nozzle to Upper Shell Welds – Lowest Extent		
Nozzle 1	248.7	80
Nozzle 2	248.7	200
Nozzle 3	248.7	320
Upper Shell		
Plate 1	170.0 to 466.5	70 to 190
Plate 2	170.0 to 466.5	190 to 310
Plate 3	170.0 to 466.5	310 to 70
Upper Shell Longitudinal (Long.) Welds		
Weld 1	170.0 to 466.5	70
Weld 2	170.0 to 466.5	190
Weld 3	170.0 to 466.5	310
Upper Shell to Intermediate (Int.) Shell Circumferential (Circ.) Weld – Centerline	170.0	0 to 360
Intermediate Shell		
Plate 1	-129.7 to 170.0	10 to 130
Plate 2	-129.7 to 170.0	130 to 250
Plate 3	-129.7 to 170.0	250 to 10
Intermediate Shell Longitudinal Welds		
Weld 1	-129.7 to 170.0	10
Weld 2	-129.7 to 170.0	130
Weld 3	-129.7 to 170.0	250
Intermediate Shell to Lower Shell Circumferential Weld – Centerline	-129.7	0 to 360
Lower Shell		
Plate 1	-308.1 to -129.7	0 to 120
Plate 2	-308.1 to -129.7	120 to 240
Plate 3	-308.1 to -129.7	240 to 360
Lower Shell Longitudinal Welds		
Weld 1	-308.1 to -129.7	0
Weld 2	-308.1 to -129.7	120
Weld 3	-308.1 to -129.7	240
Lower Shell to Lower Vessel Head Circumferential Weld – Centerline	-308.1	0 to 360

Note:

(a) Axial elevations are indexed to $z = 0.0$ cm at the core midplane

Table 2-2 Outlet Nozzle to Upper Shell Welds – Lowest Extent - Exposure (Scenario #2)

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
30	1.60	34.78	9.94E+16	5.52E+16	3.06E+16	2.07E-04	1.14E-04	6.30E-05
31	1.45	36.23	1.03E+17	5.69E+16	3.16E+16	2.14E-04	1.17E-04	6.51E-05
32	1.90	38.13	1.10E+17	6.05E+16	3.37E+16	2.30E-04	1.27E-04	7.01E-05
33	1.89	40.02	1.18E+17	6.43E+16	3.60E+16	2.47E-04	1.37E-04	7.57E-05
34	1.91	41.93	1.25E+17	6.81E+16	3.84E+16	2.64E-04	1.48E-04	8.14E-05
35	1.89	43.82	1.32E+17	7.19E+16	4.07E+16	2.80E-04	1.58E-04	8.69E-05
--	--	46.00	1.41E+17	7.63E+16	4.33E+16	3.00E-04	1.70E-04	9.34E-05
--	--	48.00	1.49E+17	8.03E+16	4.58E+16	3.18E-04	1.81E-04	9.93E-05
--	--	50.00	1.57E+17	8.43E+16	4.82E+16	3.35E-04	1.92E-04	1.05E-04

Table 2-3 Inlet Nozzle to Upper Shell Welds – Lowest Extent - Exposure (Scenario #2)

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
30	1.60	34.78	8.38E+16	4.65E+16	2.59E+16	1.78E-04	9.81E-05	5.41E-05
31	1.45	36.23	8.67E+16	4.80E+16	2.67E+16	1.84E-04	1.01E-04	5.58E-05
32	1.90	38.13	9.25E+16	5.10E+16	2.85E+16	1.99E-04	1.10E-04	6.06E-05
33	1.89	40.02	9.91E+16	5.42E+16	3.04E+16	2.15E-04	1.20E-04	6.58E-05
34	1.91	41.93	1.05E+17	5.74E+16	3.24E+16	2.31E-04	1.30E-04	7.12E-05
35	1.89	43.82	1.11E+17	6.06E+16	3.43E+16	2.46E-04	1.39E-04	7.65E-05
--	--	46.00	1.19E+17	6.43E+16	3.66E+16	2.65E-04	1.51E-04	8.26E-05
--	--	48.00	1.25E+17	6.77E+16	3.86E+16	2.81E-04	1.61E-04	8.81E-05
--	--	50.00	1.32E+17	7.10E+16	4.07E+16	2.98E-04	1.71E-04	9.37E-05

Table 2-4 Upper Shell, Intermediate Shell, and Lower Shell Plates - Exposure (Scenario #2)

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
30	1.60	34.78	1.78E+19	4.14E+19	1.77E+19	2.94E-02	6.80E-02	2.93E-02
31	1.45	36.23	1.85E+19	4.29E+19	1.79E+19	3.05E-02	7.05E-02	2.97E-02
32	1.90	38.13	1.96E+19	4.50E+19	1.99E+19	3.24E-02	7.42E-02	3.32E-02
33	1.89	40.02	2.09E+19	4.73E+19	2.22E+19	3.46E-02	7.82E-02	3.71E-02
34	1.91	41.93	2.21E+19	4.95E+19	2.44E+19	3.67E-02	8.21E-02	4.09E-02
35	1.89	43.82	2.32E+19	5.16E+19	2.63E+19	3.87E-02	8.57E-02	4.42E-02
--	--	46.00	2.46E+19	5.42E+19	2.88E+19	4.11E-02	9.02E-02	4.86E-02
--	--	48.00	2.59E+19	5.65E+19	3.11E+19	4.33E-02	9.43E-02	5.25E-02
--	--	50.00	2.72E+19	5.89E+19	3.33E+19	4.55E-02	9.83E-02	5.64E-02

Table 2-5 Upper Shell Longitudinal Welds - Exposure (Scenario #2)

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
30	1.60	34.78	1.31E+19	7.42E+18	4.11E+18	2.18E-02	1.22E-02	6.73E-03
31	1.45	36.23	1.36E+19	7.66E+18	4.25E+18	2.25E-02	1.26E-02	6.95E-03
32	1.90	38.13	1.44E+19	8.06E+18	4.49E+18	2.39E-02	1.33E-02	7.36E-03
33	1.89	40.02	1.53E+19	8.50E+18	4.76E+18	2.55E-02	1.40E-02	7.81E-03
34	1.91	41.93	1.61E+19	8.95E+18	5.03E+18	2.70E-02	1.48E-02	8.28E-03
35	1.89	43.82	1.70E+19	9.39E+18	5.30E+18	2.85E-02	1.56E-02	8.73E-03
--	--	46.00	1.80E+19	9.90E+18	5.61E+18	3.02E-02	1.64E-02	9.26E-03
--	--	48.00	1.89E+19	1.04E+19	5.89E+18	3.18E-02	1.72E-02	9.74E-03
--	--	50.00	1.98E+19	1.08E+19	6.18E+18	3.34E-02	1.80E-02	1.02E-02

Table 2-6 Intermediate Shell Longitudinal Welds - Exposure (Scenario #2)

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
30	1.60	34.78	3.05E+19	1.75E+19	9.76E+18	5.05E-02	2.88E-02	1.59E-02
31	1.45	36.23	3.16E+19	1.80E+19	1.01E+19	5.23E-02	2.96E-02	1.64E-02
32	1.90	38.13	3.31E+19	1.88E+19	1.05E+19	5.49E-02	3.10E-02	1.71E-02
33	1.89	40.02	3.48E+19	1.96E+19	1.10E+19	5.79E-02	3.24E-02	1.80E-02
34	1.91	41.93	3.64E+19	2.04E+19	1.15E+19	6.07E-02	3.38E-02	1.88E-02
35	1.89	43.82	3.79E+19	2.12E+19	1.20E+19	6.33E-02	3.52E-02	1.97E-02
--	--	46.00	3.98E+19	2.21E+19	1.26E+19	6.66E-02	3.68E-02	2.06E-02
--	--	48.00	4.15E+19	2.30E+19	1.31E+19	6.95E-02	3.82E-02	2.15E-02
--	--	50.00	4.31E+19	2.39E+19	1.36E+19	7.25E-02	3.97E-02	2.24E-02

Table 2-7 Lower Shell Longitudinal Welds - Exposure (Scenario #2)

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
30	1.60	34.78	1.77E+19	1.19E+19	2.93E-02	1.94E-02
31	1.45	36.23	1.79E+19	1.22E+19	2.97E-02	2.00E-02
32	1.90	38.13	1.99E+19	1.28E+19	3.32E-02	2.10E-02
33	1.89	40.02	2.22E+19	1.34E+19	3.71E-02	2.20E-02
34	1.91	41.93	2.44E+19	1.40E+19	4.09E-02	2.30E-02
35	1.89	43.82	2.63E+19	1.46E+19	4.42E-02	2.41E-02
--	--	46.00	2.88E+19	1.53E+19	4.86E-02	2.53E-02
--	--	48.00	3.11E+19	1.60E+19	5.25E-02	2.64E-02
--	--	50.00	3.33E+19	1.66E+19	5.64E-02	2.74E-02

Table 2-8 Circumferential Welds - Exposure (Scenario #2)

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
30	1.60	34.78	1.78E+19	1.77E+19	3.98E+15	2.94E-02	2.93E-02	3.53E-05
31	1.45	36.23	1.85E+19	1.79E+19	4.05E+15	3.05E-02	2.97E-02	3.60E-05
32	1.90	38.13	1.96E+19	1.99E+19	4.48E+15	3.24E-02	3.32E-02	3.99E-05
33	1.89	40.02	2.09E+19	2.22E+19	4.97E+15	3.46E-02	3.71E-02	4.44E-05
34	1.91	41.93	2.21E+19	2.44E+19	5.43E+15	3.67E-02	4.09E-02	4.86E-05
35	1.89	43.82	2.32E+19	2.63E+19	5.85E+15	3.87E-02	4.42E-02	5.25E-05
--	--	46.00	2.46E+19	2.88E+19	6.39E+15	4.11E-02	4.86E-02	5.74E-05
--	--	48.00	2.59E+19	3.11E+19	6.87E+15	4.33E-02	5.25E-02	6.18E-05
--	--	50.00	2.72E+19	3.33E+19	7.35E+15	4.55E-02	5.64E-02	6.62E-05

Table 2-9 Calculated Fast Neutron Fluence (E > 1.0 MeV) at the Geometric Center of the Surveillance Capsule Locations at the Core Midplane (Scenario #2)

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
30	1.60	34.78	1.16E+20	8.39E+19	4.45E+19	3.47E+19	2.38E+19
31	1.45	36.23	1.20E+20	8.69E+19	4.58E+19	3.57E+19	2.46E+19
32	1.90	38.13	1.26E+20	9.10E+19	4.78E+19	3.72E+19	2.57E+19
33	1.89	40.02	1.33E+20	9.57E+19	4.99E+19	3.88E+19	2.70E+19
34	1.91	41.93	1.39E+20	1.00E+20	5.20E+19	4.05E+19	2.83E+19
35	1.89	43.82	1.45E+20	1.04E+20	5.41E+19	4.22E+19	2.96E+19
--	--	46	1.52E+20	1.10E+20	5.65E+19	4.41E+19	3.10E+19
--	--	48	1.59E+20	1.14E+20	5.87E+19	4.59E+19	3.24E+19
--	--	50	1.66E+20	1.19E+20	6.09E+19	4.76E+19	3.37E+19

Table 2-10 Calculational Uncertainties

Description	Capsule and Vessel IR
PCA Comparisons	3%
H.B. Robinson Comparisons	3%
Analytical Sensitivity Studies	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%
Net Calculational Uncertainty	13%

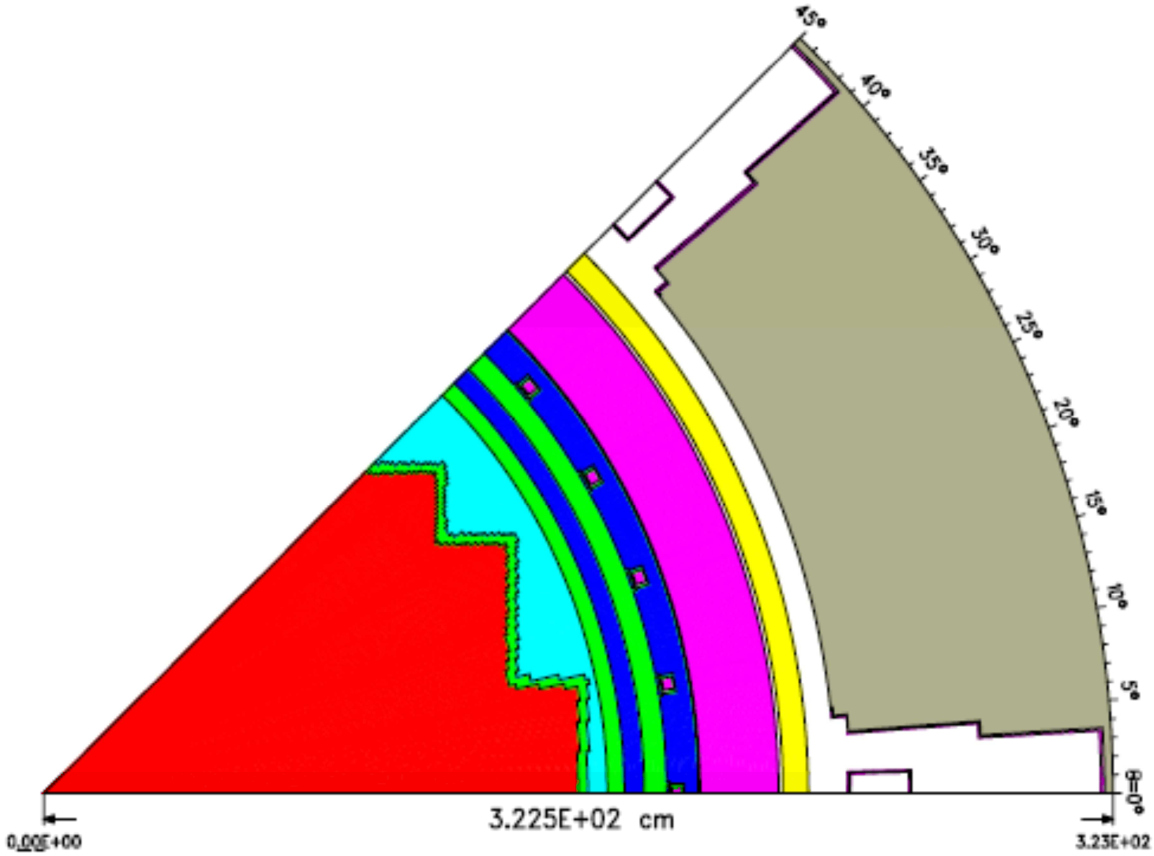

Figure 2-1 H.B. Robinson Unit 2 r,θ Reactor Geometry Plan View at the Core Midplane with Surveillance Capsules

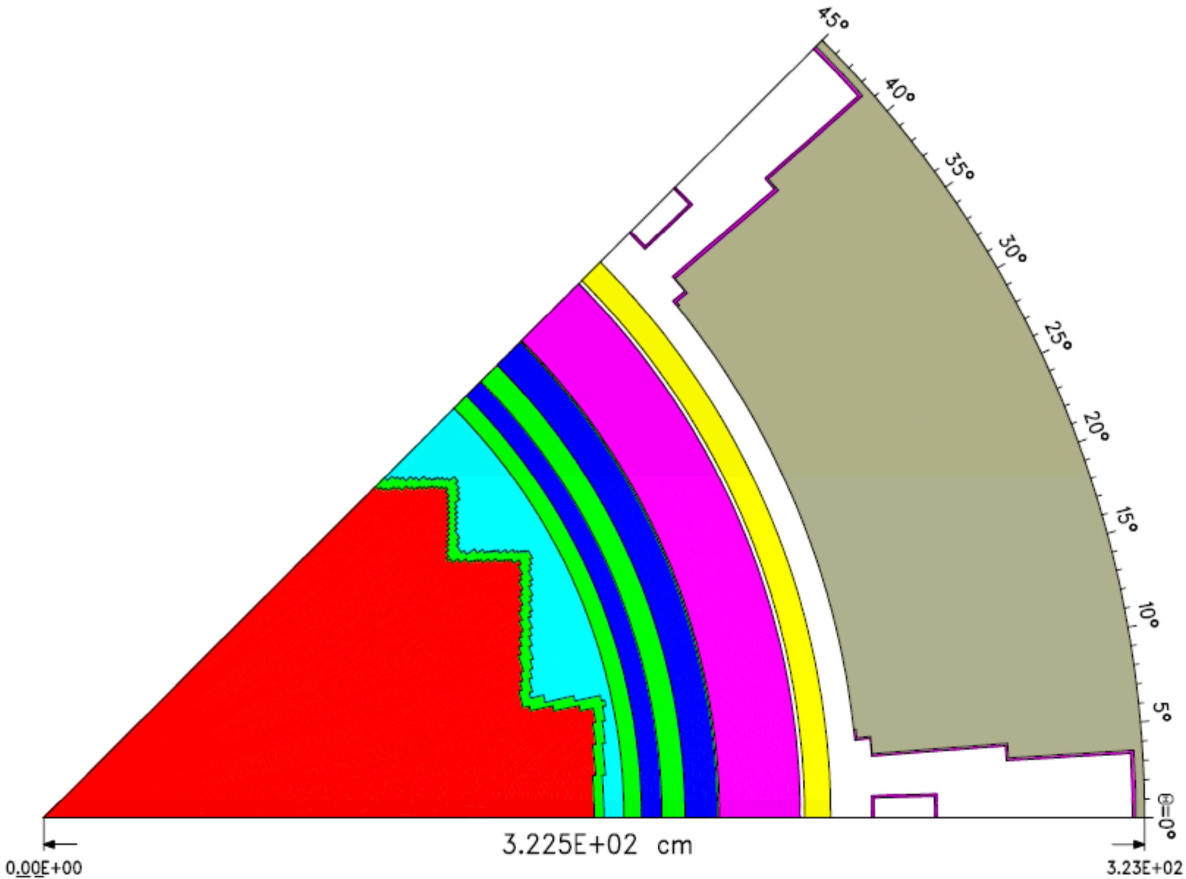

Figure 2-2 H.B. Robinson Unit 2 r,θ Reactor Geometry Plan View at the Core Midplane without Surveillance Capsules

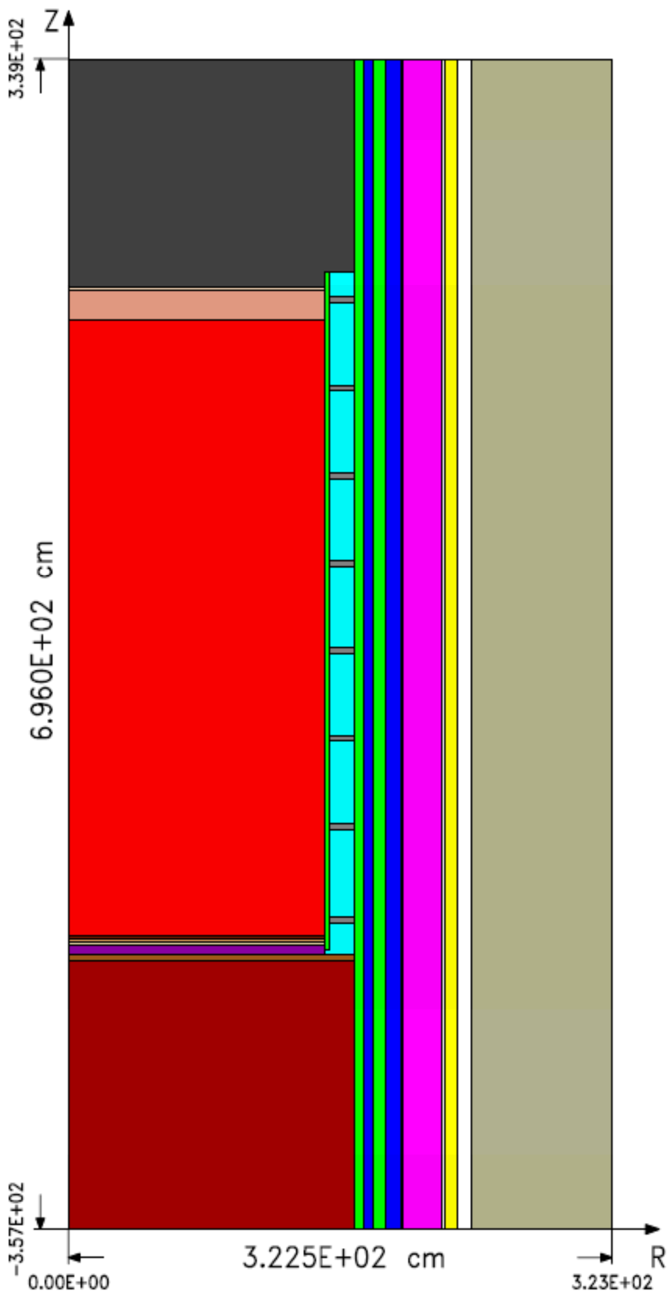

Figure 2-3 H.B. Robinson Unit 2 r,z Reactor Geometry Section View

3 MATERIAL PROPERTY INPUT

As described in NRC Regulatory Issue Summary (RIS) 2014-11 (Reference 8), any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in reactor vessel integrity evaluations. The materials that exceed this fluence threshold are commonly referred to as beltline materials and are evaluated to ensure that the applicable acceptance criteria are met. As seen from Tables 2-9 through 2-15 of WCAP-18100-NP (Reference 2) for Scenario #1 and Tables 2-2 through 2-9 of this report for Scenario #2, under either Scenario #1 or Scenario #2, the beltline materials include the inlet nozzles, the outlet nozzles, the nozzle to upper shell welds, the upper shell plates, the upper shell axial welds, the upper to intermediate shell circumferential weld, the intermediate shell plates, the intermediate shell axial welds, the intermediate to lower shell circumferential weld, the lower shell plates, and the lower shell axial welds. The H.B. Robinson Unit 2 reactor vessel does not have an extended beltline region, as all of the necessary beltline materials are encompassed in the traditional H.B. Robinson Unit 2 beltline region. Note that for reactor vessel welds, the terms “girth” and “circumferential” are used interchangeably; herein, these welds shall be referred to as circumferential welds. Similarly, for reactor vessel welds, the terms “axial” and “longitudinal” are used interchangeably; herein, these welds shall be referred to as axial welds.

Table 3-1 contains a summary of the chemistry (Cu and Ni weight percent [wt. %]) values, initial RT_{NDT}, and initial USE for each of the H.B. Robinson Unit 2 RPV beltline materials. Table 3-2 contains chemistry factor (CF) values for each of the H.B. Robinson RPV beltline materials. These values serve as input to the reactor vessel integrity evaluations contained herein.

Table 3-1 Summary of the Best-Estimate Chemistry, Initial RT_{NDT}, and Initial Upper-Shelf Energy Values for the H.B. Robinson Unit 2 Reactor Vessel Beltline Materials

Reactor Vessel Material and Identification Number	Chemical Composition[(a)]		Fracture Toughness Property	
	Wt. % Cu	Wt. % Ni	Initial RT_{NDT}[(a)] (°F)	Initial Upper-Shelf Energy[(b)] (ft-lb)
Reactor Vessel Beltline Materials				
Upper Shell Plate W10201-1	0.13	0.11	69	>62
Upper Shell Plate W10201-2	0.15	0.25	30	80
Upper Shell Plate W10201-3	0.11	0.08	36	62
Intermediate Shell Plate W10201-4	0.12	0.09	20	62
Intermediate Shell Plate W10201-5	0.10	0.12	20	64
Intermediate Shell Plate W10201-6	0.09	0.09	45	74
Lower Shell Plate W9807-3	0.12	0.10	50	78
Lower Shell Plate W9807-5	0.15	0.10	33	74
Lower Shell Plate W9807-9	0.14	0.15	9	77
Upper Shell Axial Welds 1-273 A, B, & C (Heat #86054B)	0.22	0.05	-56	105
Intermediate Shell Axial Welds 2-273 A, B, & C (Heat #86054B)	0.22	0.05	-56	105
Lower Shell Axial Welds 3-273 A, B, & C (Heat #86054B)	0.22	0.05	-56	105
Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214)	0.21	1.01	-56	112
Intermediate Shell to Lower Shell Circumferential Weld 11-273 (Heat # 34B009)	0.19	0.98	-77	106
Nozzle Welds[(c)]	Note (c)	Note (c)	-56	65
Inlet Nozzle W-10207-1	0.02	0.90	10	70
Inlet Nozzle W-10207-2	0.02	0.90	10	70
Inlet Nozzle W-10207-3	0.02	0.90	10	70
Outlet Nozzle B-3201-1	[[__ {E}]][(e)]	0.71	-7.8	70
Outlet Nozzle B-3201-2	[[__ {E}]][(e)]	0.71	1.6	70
Outlet Nozzle B-3201-3	[[__ {E}]][(e)]	0.71	7.2	70
Reactor Vessel Surveillance Materials[(d)]				
Intermediate Shell Plate W10201-4	0.12	0.09	- - -	- - -
Intermediate Shell Plate W10201-5	0.10	0.12	- - -	- - -
Intermediate Shell Plate W10201-6	0.09	0.09	- - -	- - -
H.B. Robinson Unit 2 – Surveillance Weld Material (Heat # W5214)	0.34	0.66	- - -	- - -

Notes on the following page.

Notes:

(a) Chemistry and initial RT_{NDT} values for the beltline (except nozzle forging) materials were taken from Table 1 of WCAP-15828 (Reference 9), unless otherwise noted. These values for the nozzle forging materials were taken from Table 7 of MCOE-LTR-16-33 (Reference 10). All initial RT_{NDT} values are measured values except for weld Heat # 86054B and # W5214.

(b) The initial upper-shelf energy values for the beltline materials were taken from Tables A1 and A2 of WCAP-15828 (Reference 9).

(c) Consistent with WCAP-15828 (Reference 9), since no chemistry values are available for this material, the chemistry factor for this material is set equal to the highest chemistry factor of the welds contained in the H.B. Robinson Unit 2 vessel. Consistent with Table A2 of WCAP-15828 (Reference 9), a Cu wt. % value of 0.22 is utilized for the nozzle welds USE decrease calculation.

(d) Since the surveillance plates were cut from prolongations of the beltline material, the reactor vessel surveillance plate chemistry values were taken to be equal to the chemistry values of the beltline plates. The reactor vessel surveillance weld chemistry values were taken from Table 1 of WCAP-15828 (Reference 9).

(e) Use of this number specifically to support H.B. Robinson Unit 2 is permitted since Duke is an EPRI MRP member. This number may not be used for generic work or other plant-specific evaluations if the utility is not an MRP member without EPRI consent.

Table 3-2 Summary of the H.B. Robinson Unit 2 Positions 1.1 and 2.1 Chemistry Factors

Reactor Vessel Material and Identification Number	Chemistry Factor (°F)	
	Position 1.1[(a)]	Position 2.1
Reactor Vessel Beltline Materials		
Upper Shell Plate W10201-1	62.9	- - -
Upper Shell Plate W10201-2	84.8	- - -
Upper Shell Plate W10201-3	51.8	- - -
Intermediate Shell Plate W10201-4	57.1	67.1[(b)]
Intermediate Shell Plate W10201-5	51.2	38.8[(b)]
Intermediate Shell Plate W10201-6	44.2	45.9[(b)]
Lower Shell Plate W9807-3	58.0	- - -
Lower Shell Plate W9807-5	70.5	- - -
Lower Shell Plate W9807-9	70.5	- - -
Upper Shell Axial Welds 1-273 A, B, & C (Heat #86054B)	100.8	- - -
Intermediate Shell Axial Welds 2-273 A, B, & C (Heat #86054B)	100.8	- - -
Lower Shell Axial Welds 3-273 A, B, & C (Heat #86054B)	100.8	- - -
Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214)	230.2	216.3[(c)]
Intermediate Shell to Lower Shell Circumferential Weld 11-273 (Heat # 34B009)	217.1	- - -
Nozzle Welds	230.2	- - -
Inlet Nozzle W-10207-1	20	- - -
Inlet Nozzle W-10207-2	20	- - -
Inlet Nozzle W-10207-3	20	- - -
Outlet Nozzle B-3201-1	137.9	- - -
Outlet Nozzle B-3201-2	137.9	- - -
Outlet Nozzle B-3201-3	137.9	- - -
Reactor Vessel Surveillance Materials		
Intermediate Shell Plate W10201-4	57.1	- - -
Intermediate Shell Plate W10201-5	51.2	- - -
Intermediate Shell Plate W10201-6	44.2	- - -
H.B. Robinson Unit 2 – Surveillance Weld Material (Heat # W5214)	210.7	- - -

Notes:

(a) Position 1.1 chemistry factor values for the beltline (except nozzle forging) materials were taken from Table 5 of WCAP-15828 (Reference 9). Position 1.1 chemistry factors were calculated using the copper and nickel weight percent values presented in Table 3-1 of this report and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 (Reference 14).

(b) Position 2.1 chemistry factor values for the intermediate shell plates were taken from Table 5 of WCAP-15828. Per Appendix D of WCAP-15805 (Reference 11), surveillance data for Intermediate Shell Plates W10201-4 and W10201-6 was deemed non-credible, while the surveillance data for Intermediate Shell Plate W10201-6 was deemed credible

(c) The Position 2.1 chemistry factor for Upper to Intermediate Shell Circumferential Weld 10-273 (Heat # 5214) was taken from Westinghouse letter MCOE-LTR-16-33 (Reference 10). The combined surveillance data utilized in calculation of this value is considered not fully credible per SIA Report 0901132.401 (Reference 12). Thus, the Position 2.1 CF can be used for determining ART values, but a full margin term must be included in the calculation as established in SIA Report 0901132.401. This calculation methodology is consistent with the NRC-approved conclusions of Westinghouse letter MCOE-LTR-16-33 (Reference 10).

4 PRESSURIZED THERMAL SHOCK AND EMERGENCY RESPONSE GUIDELINE LIMITS EVALUATION

4.1 PRESSURIZED THERMAL SHOCK

Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a loss-of-coolant accident (LOCA) or steam line break. Such transients may challenge the integrity of the reactor pressure vessel (RPV) under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high repressurization, significant degradation of vessel material toughness caused by radiation embrittlement, and the presence of a critical-size defect anywhere within the vessel wall.

In 1985, the U.S. NRC issued a formal ruling on PTS (10 CFR 50.61 [Reference 13]) that established screening criteria on Pressurized Water Reactor (PWR) vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of license, termed RT_{PTS}. RT_{PTS} screening values were set by the U.S. NRC for beltline axial welds, forgings or plates, and for beltline circumferential weld seams for plant operation to the end of plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end of license. The U.S. NRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values consistent with the methods given in Regulatory Guide 1.99, Revision 2 (Reference 14).

These accepted methods were used with the clad/base metal interface fluence values of Section 2. of WCAP-18100-NP (Reference 2) for Scenario #1 and Section 2 of this report for Scenario #2 to calculate the following RT_{PTS} values for the H.B. Robinson Unit 2 RPV materials at 50 EFPY (EOLE). The EOLE RT_{PTS} calculations are summarized for Scenario #1 and Scenario #2 in Table 4.1-1 and Table 4.1-2, respectively.

The following two conclusions from Section 4 of TLR-RES/DE/CIB-2013-01 (Reference 15) were utilized, as appropriate:

1. *The beltline is defined as the region of the RPV adjacent to the reactor core that is projected to receive a neutron fluence level of $1x10^{17}$ n/cm^2 (E > 1.0 MeV) or higher at the end of the licensed operating period.*
2. *Embrittlement effects may be neglected for any region of the RPV if either of the following conditions are met: (1) neutron fluence is less than $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL, or (2) the mean value of ΔT_{30} estimated using an ETC acceptable to the staff is less than 25°F at EOL. The estimate of ΔT_{30} at EOL shall be made using best-estimate chemistry values.*

Therefore, embrittlement of reactor vessel materials with ΔT_{30} (which is equivalent to ΔRT_{NDT}) values less than 25°F need not be considered in the subsequent RT_{PTS} calculations documented in Tables 4.1-1 and 4.1-2.

Table 4.1-1 RT_{PTS} Calculations for the H.B. Robinson Unit 2 Reactor Vessel Materials at 50 EFPY under Scenario #1[a]

Reactor Vessel Material and Identification Number	CF[b] (°F)	Surface Fluence[c] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF	$RT_{NDT(U)}$[d] (°F)	ΔRT_{NDT}[e] (°F)	σ_U[d] (°F)	σ_Δ[e] (°F)	M (°F)	RT_{PTS} (°F)
Upper Shell Plate W10201-1	62.9	2.45	1.2411	69	78.1	0	17.0	34.0	181
Upper Shell Plate W10201-2	84.8	2.45	1.2411	30	105.2	0	17.0	34.0	169
Upper Shell Plate W10201-3	51.8	2.45	1.2411	36	64.3	0	17.0	34.0	134
Intermediate Shell Plate W10201-4	57.1	5.69	1.4270	20	81.5	0	17.0	34.0	135
→ Using non-credible surveillance data	67.1	5.69	1.4270	20	95.7	0	17.0	34.0	150
Intermediate Shell Plate W10201-5	51.2	5.69	1.4270	20	73.1	0	17.0	34.0	127
→ Using credible surveillance data	38.8	5.69	1.4270	20	55.4	0	8.5	17.0	92
Intermediate Shell Plate W10201-6	44.2	5.69	1.4270	45	63.1	0	17.0	34.0	142
→ Using non-credible surveillance data	45.9	5.69	1.4270	45	65.5	0	17.0	34.0	144
Lower Shell Plate W9807-3	58.0	2.03	1.1930	50	69.2	0	17.0	34.0	153
Lower Shell Plate W9807-5	70.5	2.03	1.1930	33	84.1	0	17.0	34.0	151
Lower Shell Plate W9807-9	70.5	2.03	1.1930	9	84.1	0	17.0	34.0	127
Upper Shell Axial Welds 1-273 A, B, & C (Heat # 86054B)	100.8	1.80	1.1613	-56	117.1	17	28.0	65.5	127
Intermediate Shell Axial Welds 2-273 A, B, & C (Heat # 86054B)	100.8	4.18	1.3657	-56	137.7	17	28.0	65.5	147
Lower Shell Axial Welds 3-273 A, B, & C (Heat # 86054B)	100.8	2.03	1.1930	-56	120.3	17	28.0	65.5	130
Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214)	230.2	2.45	1.2411	-56	285.7	17	28.0	65.5	295
→ Using not fully credible surveillance data	216.3	2.45	1.2411	-56	268.5	17	28.0	65.5	278
Intermediate Shell to Lower Shell Circumferential Weld 11-273 (Heat # 34B009)	217.1	2.03	1.1930	-77	259.0	0	28.0	56.0	238
Nozzle Welds	230.2	0.0135	0.1339	-56	30.8	17	15.4	45.9	21
Inlet Nozzle W-10207-1	20.0	0.0114	0.1198	10	0 (2.4)	0	0.0	0.0	10

Table 4.1-1 RT_{PTS} Calculations for the H.B. Robinson Unit 2 Reactor Vessel Materials at 50 EFPY under Scenario #1[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	Surface Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF	$RT_{NDT(U)}$[(d)] (°F)	ΔRT_{NDT}[(e)] (°F)	σ_U[(d)] (°F)	σ_Δ[(e)] (°F)	M (°F)	RT_{PTS} (°F)
Inlet Nozzle W-10207-2	20.0	0.0114	0.1198	10	0 (2.4)	0	0.0	0.0	10
Inlet Nozzle W-10207-3	20.0	0.0114	0.1198	10	0 (2.4)	0	0.0	0.0	10
Outlet Nozzle B-3201-1	137.9	0.0135	0.1339	-7.8	0 (18.5)	0	0.0	0.0	-7.8
Outlet Nozzle B-3201-2	137.9	0.0135	0.1339	1.6	0 (18.5)	0	0.0	0.0	1.6
Outlet Nozzle B-3201-3	137.9	0.0135	0.1339	7.2	0 (18.5)	0	0.0	0.0	7.2

Notes:

(a) The 10 CFR 50.61 (Reference 13) methodology was utilized in the calculation of the RT_{PTS} values.

(b) CF values taken from Table 3-2.

(c) Fluence values taken from WCAP-18100-NP (Reference 2).

(d) Initial RT_{NDT} ($RT_{NDT(U)}$) values taken from Table 3-1. $RT_{NDT(U)}$ values are measured values for all base metal materials; thus, σ_U = 0°F. $RT_{NDT(U)}$ values are generic for all welds except the Intermediate to Lower Shell Circumferential Weld. Thus, σ_U = 17°F for all welds except the Intermediate to Lower Shell Circumferential Weld. For the Intermediate to Lower Shell Circumferential Weld, σ_U = 0°F, since $RT_{NDT(U)}$ is a measured value for this material.

(e) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 (Reference 15). Actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(f) As documented in WCAP-15805 (Reference 11), the surveillance capsule data for Intermediate Shell Plate W10201-4 and Intermediate Shell Plate W10201-6 were deemed non-credible, while the surveillance capsule data for Intermediate Shell Plate W10201-5 were deemed credible. Per the guidance of 10 CFR 50.61 (Reference 13), the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data. The base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. The weld metal σ_Δ = 28°F for Position 1.1. The surveillance capsule data for Upper Shell to Intermediate Shell Circumferential Weld 10-273 were deemed not fully credible per Reference 12; a full margin term (σ_Δ = 28°F) must be used for both Position 1.1 and 2.1. However, the lower of the two values may be taken as the RT_{PTS} value. σ_Δ need not exceed 0.5*ΔRT_{NDT} per Reference 13.

Table 4.1-2 RT_{PTS} Calculations for the H.B. Robinson Unit 2 Reactor Vessel Materials at 50 EFPY under Scenario #2[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	Surface Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF	$RT_{NDT(U)}$[(d)] (°F)	ΔRT_{NDT}[(e)] (°F)	σ_U[(d)] (°F)	σ_Δ[(e)] (°F)	M (°F)	RT_{PTS} (°F)
Upper Shell Plate W10201-1	62.9	2.72	1.2671	69	79.7	0	17.0	34.0	183
Upper Shell Plate W10201-2	84.8	2.72	1.2671	30	107.4	0	17.0	34.0	171
Upper Shell Plate W10201-3	51.8	2.72	1.2671	36	65.6	0	17.0	34.0	136
Intermediate Shell Plate W10201-4	57.1	5.89	1.4333	20	81.8	0	17.0	34.0	136
→ Using non-credible surveillance data	67.1	5.89	1.4333	20	96.2	0	17.0	34.0	150
Intermediate Shell Plate W10201-5	51.2	5.89	1.4333	20	73.4	0	17.0	34.0	127
→ Using credible surveillance data	38.8	5.89	1.4333	20	55.6	0	8.5	17.0	93
Intermediate Shell Plate W10201-6	44.2	5.89	1.4333	45	63.4	0	17.0	34.0	142
→ Using non-credible surveillance data	45.9	5.89	1.4333	45	65.8	0	17.0	34.0	145
Lower Shell Plate W9807-3	58.0	3.33	1.3152	50	76.3	0	17.0	34.0	160
Lower Shell Plate W9807-5	70.5	3.33	1.3152	33	92.7	0	17.0	34.0	160
Lower Shell Plate W9807-9	70.5	3.33	1.3152	9	92.7	0	17.0	34.0	136
Upper Shell Axial Welds 1-273 A, B, & C (Heat # 86054B)	100.8	1.98	1.1865	-56	119.6	17	28.0	65.5	129
Intermediate Shell Axial Welds 2-273 A, B, & C (Heat # 86054B)	100.8	4.31	1.3721	-56	138.3	17	28.0	65.5	148
Lower Shell Axial Welds 3-273 A, B, & C (Heat # 86054B)	100.8	3.33	1.3152	-56	132.6	17	28.0	65.5	142
Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214)	230.2	2.72	1.2671	-56	291.7	17	28.0	65.5	301
→ Using not fully credible surveillance data	216.3	2.72	1.2671	-56	274.1	17	28.0	65.5	284
Intermediate Shell to Lower Shell Circumferential Weld 11-273 (Heat # 34B009)	217.1	3.33	1.3152	-77	285.5	0	28.0	56.0	265
Nozzle Welds	230.2	0.0157	0.1477	-56	34.0	17	17.0	48.1	26

Table 4.1-2 RT_{PTS} Calculations for the H.B. Robinson Unit 2 Reactor Vessel Materials at 50 EFPY under Scenario #2[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	Surface Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surface FF	$RT_{NDT(U)}$[(d)] (°F)	ΔRT_{NDT}[(e)] (°F)	σ_U[(d)] (°F)	σ_Δ[(e)] (°F)	M (°F)	RT_{PTS} (°F)
Inlet Nozzle W-10207-1	20.0	0.0132	0.1320	10	0 (2.6)	0	0.0	0.0	10
Inlet Nozzle W-10207-2	20.0	0.0132	0.1320	10	0 (2.6)	0	0.0	0.0	10
Inlet Nozzle W-10207-3	20.0	0.0132	0.1320	10	0 (2.6)	0	0.0	0.0	10
Outlet Nozzle B-3201-1	137.9	0.0157	0.1477	-7.8	0 (20.4)	0	0.0	0.0	-7.8
Outlet Nozzle B-3201-2	137.9	0.0157	0.1477	1.6	0 (20.4)	0	0.0	0.0	1.6
Outlet Nozzle B-3201-3	137.9	0.0157	0.1477	7.2	0 (20.4)	0	0.0	0.0	7.2

Notes:

(a) The 10 CFR 50.61 (Reference 13) methodology was utilized in the calculation of the RT_{PTS} values.

(b) CF values taken from Table 3-2.

(c) Fluence values taken from Section 2 of this WCAP.

(d) Initial RT_{NDT} ($RT_{NDT(U)}$) values taken from Table 3-1. $RT_{NDT(U)}$ values are measured values for all base metal materials; thus, σ_U = 0°F. $RT_{NDT(U)}$ values are generic for all welds except the Intermediate to Lower Shell Circumferential Weld. Thus, σ_U = 17°F for all welds except the Intermediate to Lower Shell Circumferential Weld. For the Intermediate to Lower Shell Circumferential Weld, σ_U = 0°F, since $RT_{NDT(U)}$ is a measured value for this material.

(e) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 (Reference 15). Actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(f) As documented in WCAP-15805 (Reference 11), the surveillance capsule data for Intermediate Shell Plate W10201-4 and Intermediate Shell Plate W10201-6 were deemed non-credible, while the surveillance capsule data for Intermediate Shell Plate W10201-5 were deemed credible. Per the guidance of 10 CFR 50.61 (Reference 13), the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data. The base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. The weld metal σ_Δ = 28°F for Position 1.1. The surveillance capsule data for Upper Shell to Intermediate Shell Circumferential Weld 10-273 were deemed not fully credible per Reference 12; a full margin term (σ_Δ = 28°F) must be used for both Position 1.1 and 2.1. However, the lower of the two values may be taken as the RT_{PTS} value. σ_Δ need not exceed 0.5*ΔRT_{NDT} per Reference 13.

PTS Conclusion

The H.B. Robinson Unit 2 limiting RT_{PTS} value for base metal or longitudinal weld materials at 50 EFPY is 181°F under Scenario #1 (see Table 4.1-1) and 183°F under Scenario #2 (see Table 4.1-2), both values correspond to Upper Shell Plate W10201-1 (using Position 1.1). The H.B. Robinson Unit 2 limiting RT_{PTS} value for circumferentially oriented welds at 50 EFPY is 278°F under Scenario #1 (see Table 4.1-1) and 284°F under Scenario #2 (see Table 4.1-2), both values correspond to Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214, using Position 2.1 and not fully credible surveillance data).

Therefore, all of the beltline materials in the H.B. Robinson Unit 2 reactor vessel are below the RT_{PTS} screening criteria of 270°F for base metal and/or longitudinal welds, and 300°F for circumferentially oriented welds through EOLE (50 EFPY) under Scenario #1 and Scenario #2. However, additional margin to the PTS screening criteria exists under Scenario #1 when compared to Scenario #2.

The Alternate PTS Rule (10 CFR 50.61a [Reference 16]) was published in the Federal Register by the NRC in 2010. This alternate rule is less restrictive than the PTS Rule (10 CFR 50.61) and is intended to be used for situations in which the 10 CFR 50.61 criteria cannot be met. H.B. Robinson Unit 2 currently meets the criteria for the PTS Rule through EOLE and therefore does not need to utilize the Alternate PTS Rule at this time.

4.2 EMERGENCY RESPONSE GUIDELINE LIMITS

The Emergency Response Guideline (ERG) limits were developed to establish guidance for operator action in the event of an emergency situation, such as a PTS event (Reference 17). Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest value of RT_{NDT} for which the generic category ERG limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Therefore, if the limiting vessel material has an RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG P-T limits must be developed.

The ERG category is determined by the magnitude of the limiting RT_{NDT} value, which is calculated the same way as the RT_{PTS} values are calculated in Section 4.1 of this report. The material with the highest RT_{NDT} defines the limiting material, which for H.B. Robinson Unit 2 is Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214). Table 4.2-1 identifies ERG category limits and the limiting material RT_{NDT} values at 50 EFPY under both scenarios for H.B. Robinson Unit 2.

Table 4.2-1 Evaluation of H.B. Robinson Unit 2 ERG Limit Category under Scenario #1 and Scenario #2

ERG Pressure-Temperature Limits (Reference 17)	
Applicable RT_{NDT} Value[(a)]	**ERG P-T Limit Category**
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category III b
Limiting RT_{NDT} Value[(b)]	
Scenario	**RT_{NDT} Value @ 50 EFPY**
Scenario #1	278°F
Scenario #2	284°F

Notes:

(a) Longitudinally oriented flaws are applicable only up to 250°F; circumferentially oriented flaws are applicable up to 300°F.

(b) Values taken from Tables 4.1-1 and 4.1-2 and correspond to the Upper Shell to Intermediate Shell Circumferential Weld 10-273 Heat # W5214 (Position 2.1).

Per the ERG limit guidance document (Reference 17), some vessels do not change categories for operation through the end of license. However, when a vessel does change ERG categories between the beginning and end of operation, a plant-specific assessment must be performed to determine at what operating time the category changes. Thus, the ERG classification need not be changed until the operating cycle during which the maximum vessel value of actual or estimated real-time RT_{NDT} exceeds the limit on its current ERG category.

Per Table 4.2-1, the limiting material for H.B. Robinson Unit 2 (Upper Shell to Intermediate Shell Circumferential Weld 10-273 Heat # W5214 [Position 2.1]) is a circumferential flaw material and has an RT_{NDT} greater than 250°F but less than 300°F through 50 EFPY. Therefore, H.B. Robinson Unit 2 remains in ERG Category III b through EOLE (50 EFPY) under either Scenario #1 or Scenario #2.

Conclusion of ERG P-T Limit Categorization

As summarized above, under either operational scenario H.B. Robinson Unit 2 is currently in ERG Category III b and will remain in ERG Category III b through EOLE (50 EFPY).

5 UPPER-SHELF ENERGY EVALUATION

Charpy upper-shelf energy (USE) is associated with the determination of acceptable RPV toughness during the license renewal period when the vessel is exposed to additional irradiation.

The requirements on USE are included in 10 CFR 50, Appendix G (Reference 18). 10 CFR 50, Appendix G requires utilities to submit an analysis at least three years prior to the time that the USE of any RPV material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing. The most common analysis is called an equivalent margins analysis (EMA). A generic EMA was completed and documented in WCAP-13587, Revision 1 (Reference 19). As a three-loop Westinghouse-designed reactor, the H.B. Robinson Unit 2 reactor vessel materials must maintain USE values above 42 ft-lbs per Table 3-5 of WCAP-13857, Revision 1 (Reference 19).

There are two methods that can be used to predict the decrease in USE with irradiation, depending on the availability of credible surveillance capsule data as defined in Regulatory Guide 1.99, Revision 2 (Reference 14). For vessel beltline materials that are not in the surveillance program or have non-credible data, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in Regulatory Guide 1.99, Revision 2.

When two or more credible surveillance sets become available from the reactor, they may be used to determine the Charpy USE of the surveillance material. The surveillance data are then used in conjunction with the Regulatory Guide to predict the change in USE (Position 2.2) of the RPV material due to irradiation.

The 50 EFPY (EOLE) Position 1.2 USE values of the vessel materials can be predicted using the corresponding 1/4T fluence projection, the copper content of the materials, and Figure 2 in Regulatory Guide 1.99, Revision 2.

The predicted Position 2.2 USE values are determined for the reactor vessel materials that are contained in the surveillance program by using the reduced plant surveillance data along with the corresponding 1/4T fluence projection. The reduced plant surveillance data was obtained using data from Table B-1 of WCAP-15805 (Reference 11) for H.B. Robinson Unit 2. The surveillance data was plotted in Regulatory Guide 1.99, Revision 2, Figure 2 (see Figures 5-1 and 5-2 of this report) using the surveillance capsule fluence values documented in Table 2-8 of WCAP-18100-NP (Reference 2). This data was fitted by drawing a line parallel to the existing lines as the upper bound of all the surveillance data. These reduced lines were used instead of the existing lines to determine the Position 2.2 EOLE USE values.

The projected USE values were calculated to determine if the H.B. Robinson Unit 2 beltline materials remain above the 42 ft-lb criterion established by WCAP-13587, Revision 1 (Reference 19) at 50 EFPY (EOLE). These calculations are summarized in Table 5-1 for Scenario #1 and Table 5-2 for Scenario #2.

Table 5-1 Predicted USE Values at 50 EFPY (EOLE) for H.B. Robinson Unit 2 under Scenario #1

Reactor Vessel Material and Identification Number	Wt % Cu(a)	1/4T Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE(a) (ft-lb)	Projected USE Decrease(c) (%)	Projected EOLE USE (ft-lb)
Position 1.2					
Upper Shell Plate W10201-1	0.13	1.40	>62(d)	24.0	>62(d)
Upper Shell Plate W10201-2	0.15	1.40	80	26.0	59.2
Upper Shell Plate W10201-3	0.11	1.40	62	22.0	48.4
Intermediate Shell Plate W10201-4	0.12	3.25	62	28.0	44.6
Intermediate Shell Plate W10201-5	0.10	3.25	64	25.5	47.7
Intermediate Shell Plate W10201-6	0.09	3.25	74	25.5(e)	55.1
Lower Shell Plate W9807-3	0.12	1.16	78	22.0	60.8
Lower Shell Plate W9807-5	0.15	1.16	74	25.0	55.5
Lower Shell Plate W9807-9	0.14	1.16	77	24.0	58.5
Upper Shell Axial Welds 1-273 A, B, & C	0.22	1.03	105	36.5	66.7
Intermediate Shell Axial Welds 2-273 A, B, & C	0.22	2.39	105	44.5	58.3
Lower Shell Axial Welds 3-273 A, B, & C	0.22	1.16	105	37.5	65.6
Upper Shell to Intermediate Shell Circumferential Weld 10-273	0.21	1.40	112	38.0	69.4
Intermediate Shell to Lower Shell Circumferential Weld 11-273	0.19	1.16	106	34.5	69.4
Nozzle Welds	0.22	0.00772(e)	65	14.5	55.6
Inlet Nozzle W-10207-1	0.02	0.00652(e)	70	7.5	64.8
Inlet Nozzle W-10207-2	0.02	0.00652(e)	70	7.5	64.8
Inlet Nozzle W-10207-3	0.02	0.00652(e)	70	7.5	64.8
Outlet Nozzle B-3201-1	[[__{E}]]	0.00772(e)	70	11.0	62.3
Outlet Nozzle B-3201-2	[[__{E}]]	0.00772(e)	70	11.0	62.3
Outlet Nozzle B-3201-3	[[__{E}]]	0.00772(e)	70	11.0	62.3
Position 2.2					
Intermediate Shell Plate W10201-4	0.12	3.25	62	17.0	51.5
Intermediate Shell Plate W10201-5	0.10	3.25	64	17.5	52.8
Intermediate Shell Plate W10201-6	0.09	3.25	74	8.0	68.1
Upper Shell to Intermediate Shell Circumferential Weld 10-273	0.21	1.40	112	49.0	57.1

Notes:

(a) Cu and initial USE values taken from Table 3-1.

(b) 1/4T fluence values calculated using Regulatory Guide 1.99, Revision 2 (Reference 14) methodology, surface fluence values from WCAP-18100-NP (Reference 2), and the H.B. Robinson Unit 2 beltline thickness of 9.313 inches.

(c) Projected USE decreases were calculated in accordance with Regulatory Guide 1.99, Revision 2 (Reference 14), Positions 1.2 and 2.2. When the Cu wt. % value was below the value of the lowest line given on the Regulatory Guide 1.99, Revision 2 (Reference 14) figure, the lowest line on the figure was conservatively used.

(d) Per WCAP-15828 (Reference 9), since the fluence value of this material is less than the fluence value of Capsule T (3.87 x 10^{19} n/cm^2 [E >1.0 MeV]), the USE value remains above 62 ft-lbs.

(e) Figure 2 of Regulatory Guide 1.99, Revision 2 (Reference 14) is limited to fluence values of 2.0 x 10^{17} n/cm^2, (E > 1.0 MeV) and greater. Therefore, the fluence of this material is conservatively assumed to be 2.0 x 10^{17} n/cm^2, (E > 1.0 MeV).

Table 5-2 Predicted USE Values at 50 EFPY (EOLE) for H.B. Robinson Unit 2 under Scenario #2

Reactor Vessel Material and Identification Number	Wt % Cu[(a)]	1/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE[(a)] (ft-lb)	Projected USE Decrease[(c)] (%)	Projected EOLE USE (ft-lb)
Position 2.1					
Upper Shell Plate W10201-1	0.13	1.56	>62[(d)]	24.5	>62[(d)]
Upper Shell Plate W10201-2	0.15	1.56	80	27.0	58.4
Upper Shell Plate W10201-3	0.11	1.56	62	22.5	48.1
Intermediate Shell Plate W10201-4	0.12	3.37	62	28.0	44.6
Intermediate Shell Plate W10201-5	0.10	3.37	64	25.5	47.7
Intermediate Shell Plate W10201-6	0.09	3.37	74	25.5[(e)]	55.1
Lower Shell Plate W9807-3	0.12	1.90	78	24.5	58.9
Lower Shell Plate W9807-5	0.15	1.90	74	28.0	53.3
Lower Shell Plate W9807-9	0.14	1.90	77	27.0	56.2
Upper Shell Axial Welds 1-273 A, B, & C	0.22	1.13	105	37.5	65.6
Intermediate Shell Axial Welds 2-273 A, B, & C	0.22	2.46	105	45.0	57.8
Lower Shell Axial Welds 3-273 A, B, & C	0.22	1.90	105	42.0	60.9
Upper Shell to Intermediate Shell Circumferential Weld 10-273	0.21	1.56	112	39.0	68.3
Intermediate Shell to Lower Shell Circumferential Weld 11-273	0.19	1.90	106	38.5	65.2
Nozzle Welds	0.22	0.00898[(e)]	65	14.5	55.6
Inlet Nozzle W-10207-1	0.02	0.00755[(e)]	70	7.5	64.8
Inlet Nozzle W-10207-2	0.02	0.00755[(e)]	70	7.5	64.8
Inlet Nozzle W-10207-3	0.02	0.00755[(e)]	70	7.5	64.8
Outlet Nozzle B-3201-1	[[__{E}]]	0.00898[(e)]	70	11.0	62.3
Outlet Nozzle B-3201-2	[[__{E}]]	0.00898[(e)]	70	11.0	62.3
Outlet Nozzle B-3201-3	[[__{E}]]	0.00898[(e)]	70	11.0	62.3
Position 2.2					
Intermediate Shell Plate W10201-4	0.12	3.37	62	17.5	51.2
Intermediate Shell Plate W10201-5	0.10	3.37	64	18.0	52.5
Intermediate Shell Plate W10201-6	0.09	3.37	74	8.0	68.1
Upper Shell to Intermediate Shell Circumferential Weld 10-273	0.21	1.56	112	50.0	56.0

Notes:

(a) Cu and initial USE values taken from Table 3-1.

(b) 1/4T fluence values calculated using Regulatory Guide 1.99, Revision 2 (Reference 14) methodology, surface fluence values taken from Section 2 of this WCAP, and the H.B. Robinson Unit 2 beltline thickness of 9.313 inches.

(c) Projected USE decreases were calculated in accordance with Regulatory Guide 1.99, Revision 2 (Reference 14), Positions 1.2 and 2.2. When the Cu wt. % value was below the value of the lowest line given on the Regulatory Guide 1.99, Revision 2 (Reference 14) figure, the lowest line on the figure was conservatively used.

(d) Per WCAP-15828 (Reference 9), since the fluence value of this material is less than the fluence value of Capsule T (3.87 x 10^{19} n/cm^2 [E >1.0 MeV]), the USE value remains above 62 ft-lbs.

(e) Figure 2 of Regulatory Guide 1.99, Revision 2 (Reference 14) is limited to fluence values of 2.0 x 10^{17} n/cm^2, (E > 1.0 MeV) and greater. Therefore, the fluence of this material is conservatively assumed to be 2.0 x 10^{17} n/cm^2, (E > 1.0 MeV).

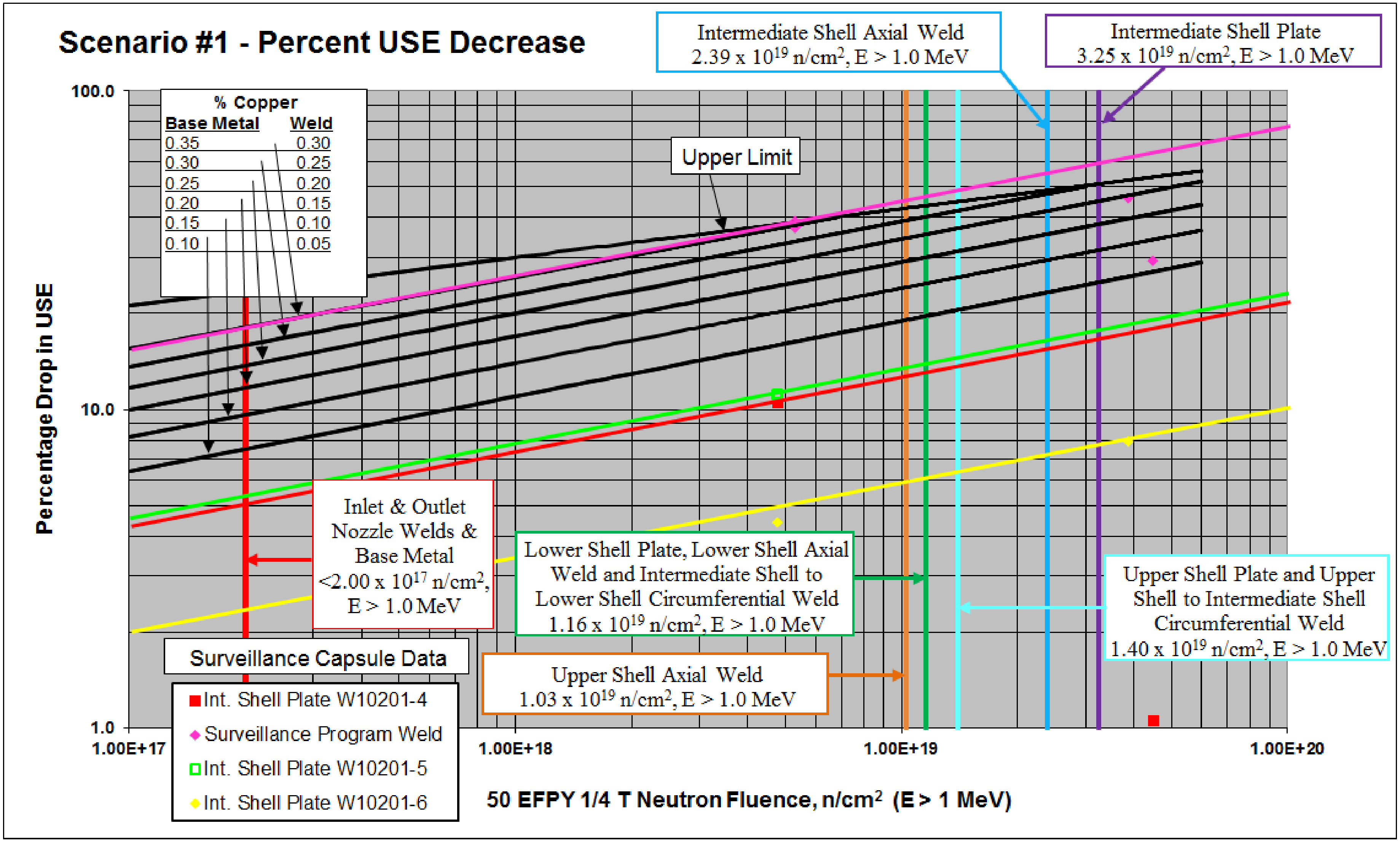

Figure 5-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence under Scenario #1

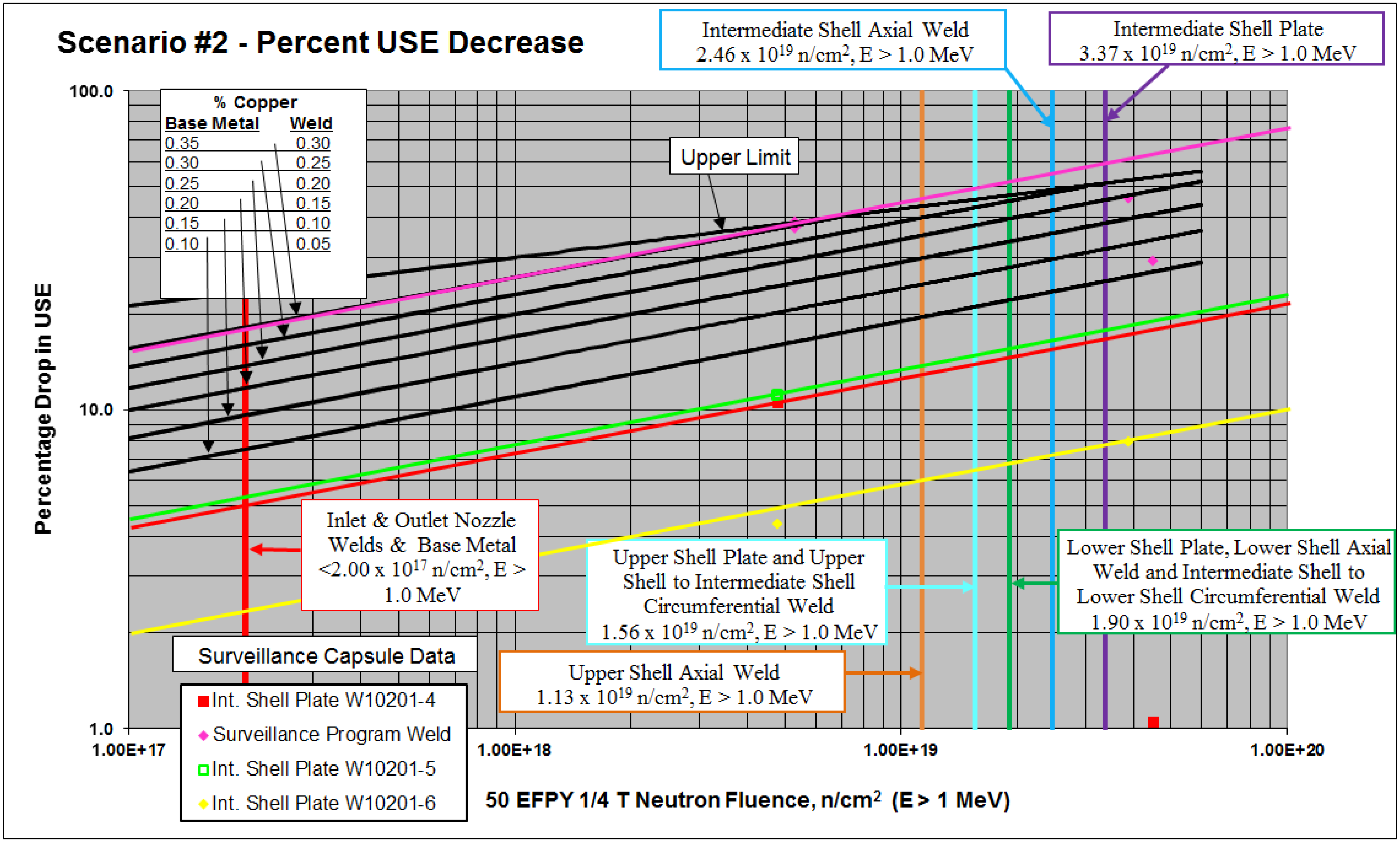

Figure 5-2 Regulatory Guide 1.99, Revision 2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence under Scenario #2

USE Conclusion

For H.B. Robinson Unit 2, the limiting USE value at 50 EFPY is 48.4 ft-lb under Scenario #1 (see Table 5-1) and 48.1 ft-lb under Scenario #2 (see Table 5-2); these values correspond to the Upper Shell Plate W10201-3. Therefore, all of the beltline materials in the H.B. Robinson Unit 2 reactor vessel are projected to remain above the required USE value of 42 ft-lb (per WCAP-13587, Revision 1) through EOLE (50 EFPY) under either operational scenario. However, additional margin to the required USE value exists under Scenario #1 when compared to Scenario #2.

As a result of the low USE values for some RV materials, H.B. Robinson Unit 2 takes advantage of the generic Westinghouse EMA (Reference 19) which concluded that a minimum USE value of 42 ft-lbs is allowable for a Westinghouse-designed three-loop plant, instead of 50 ft-lbs, as required by 10 CFR Part 50, Appendix G (Reference 18).

Finally, note that Position 2.2 USE values are utilized in lieu of Position 1.2 values when available, even though some surveillance data is deemed non-credible for ΔRT_{NDT} calculations. This usage is allowable per Regulatory Guide 1.99, Revision 2 (Reference 14), since the upper shelf can be clearly determined for each of these materials.

6 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting material in the beltline region of the RPV. The most limiting RT_{NDT} of the material in the core (beltline) region of the RPV is determined by using the unirradiated RPV material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}).

RT_{NDT} increases as the material is exposed to fast-neutron irradiation; therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. Using the adjusted reference temperature (ART) values, pressure-temperature (P-T) limit curves are determined in accordance with the requirements of 10 CFR Part 50, Appendix G (Reference 18), as augmented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code (Reference 20).

P-T limit curves for normal heatup and cooldown of the primary reactor coolant system for H.B. Robinson Unit 2 were originally developed in WCAP-15827 (Reference 21) for 50 EFPY. However, these curves were developed using only a subset of the currently available surveillance data, outdated fluence values, and some material properties differing from those summarized herein. Additionally, extended beltline materials must now be considered per NRC RIS 2014-11 (Reference 8); however, the traditional H.B. Robinson Unit 2 beltline region encompasses all of the necessary beltline materials.

To confirm the existing applicability of the 50 EFPY P-T limit curves developed in WCAP-15827 (Reference 21) for H.B. Robinson Unit 2, the limiting reactor vessel material ART values from the beltline (with consideration of the updated fluence values, one revised Position 2.1 chemistry factor value, and recalculated initial RT_{NDT} values for the nozzle forging materials) must be shown to be less than or equal to the limiting beltline material ART values used in development of the existing P-T limit curves contained in Reference 21 at 50 EFPY. The Regulatory Guide 1.99, Revision 2 (Reference 14) methodology was used along with the fluence values documented in Section 2 of WCAP-18100-NP (Reference 2) for Scenario #1 and those documented in Section 2 of this report for Scenario #2 to calculate ART values for the H.B. Robinson Unit 2 reactor vessel materials at 50 EFPY. The ART calculations are summarized in Tables 6.1-1 through 6.1-4 for H.B. Robinson Unit 2. Note that per Table 6.1-5 the 50 EFPY ART values for Scenario #1 are bounded by those used to develop the H.B. Robinson Unit 2 P-T limit curves WCAP-15827 (Reference 21). However, the Scenario #2 50 EFPY ART values will require a reduction in the applicability term of the current P-T limit curves for H.B. Robinson Unit 2.

Existing P-T Limit Curves Applicability Conclusions

Comparison of the limiting ART values calculated as part of this EOLE RPV integrity evaluation to those used in calculation of the existing P-T limit curves are contained in Table 6.1-5 for H.B. Robinson Unit 2. With consideration of the fluence values provided for Scenario #1, it is concluded that this scenario requires no reduction to the applicability term of the existing H.B. Robinson Unit 2 50 EFPY P-T limit curves documented in WCAP-15827 (Reference 21).

Consideration of the Scenario #2 fluence values indicates that a reduction to the applicability term of the existing H.B. Robinson Unit 2 50 EFPY P-T limit curves documented in WCAP-15827 (Reference 21) is required. Under Scenario #2, the H.B. Robinson Unit 2 P-T limit curves will require revision after 46.3 EFPY. Note that a similar applicability check and a nozzle P-T limit curve evaluation (Reference 10) has been previously submitted and approved for Scenario #1. These conclusions regarding the nozzles have are also confirmed for Scenario #2.

6.1 ADJUSTED REFERENCE TEMPERATURES AND PRESSURE-TEMPERATURE LIMIT CURVES APPLICABILITY

Tables 6.1-1 through 6.1-4 summarize the 1/4T and 3/4T ART calculations for H.B. Robinson Unit 2 at 50 EFPY under Scenario #1 and Scenario #2, respectively. The limiting 50 EFPY circumferential flaw ART values for H.B. Robinson Unit 2 correspond to Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214, Position 2.1 - using surveillance data). The limiting 50 EFPY axial flaw ART values for H.B. Robinson Unit 2 correspond to Upper Shell Plate W10201-1 (Position 1.1).

Table 6.1-5 compares the updated limiting ART values at 50 EFPY to the limiting ART values used in development of the existing 50 EFPY P-T limit curves documented in WCAP-15827 (Reference 21). The limiting circumferential flaw ART values used to develop the existing 50 EFPY P-T limit curves are documented in Table 18 of Reference 21 and correspond to Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214, Position 2.1). The limiting 50 EFPY axial flaw ART values for H.B. Robinson Unit 2 correspond to Upper Shell Plate W10201-1 (Position 1.1).

As shown in Table 6.1-5, the revised limiting ART values at 50 EFPY are less than the limiting ART values used to develop the existing 50 EFPY P-T limit curves for Scenario #1. For Scenario #2, the revised limiting axial flaw material ART values at 50 EFPY are greater than the limiting axial flaw material ART values used to develop the existing 50 EFPY P-T limit curves.

P-T Limits Applicability Conclusion

For H.B. Robinson Unit 2, it is concluded that the existing 50 EFPY P-T limit curves in WCAP-15827 (Reference 21) do not require a reduction of applicability and remain valid through 50 EFPY under Scenario #1. The existing 50 EFPY P-T limit curves in WCAP-15827 (Reference 21) do require a reduction of applicability and remain valid through 46.3 EFPY under Scenario #2.

Table 6.1-1 Adjusted Reference Temperature Evaluation for the H.B. Robinson Unit 2 Reactor Vessel Beltline Materials through 50 EFPY at the 1/4T Location under Scenario #1[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	1/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4 T FF	$RT_{NDT(U)}$[(d)] (°F)	Predicted[(e)] ΔRT_{NDT} (°F)	σ_I[(d)] (°F)	σ_Δ[(f)] (°F)	M (°F)	ART (°F)
Upper Shell Plate W10201-1	62.9	1.40	1.0936	69	68.8	0	17	34.0	172
Upper Shell Plate W10201-2	84.8	1.40	1.0936	30	92.7	0	17	34.0	157
Upper Shell Plate W10201-3	51.8	1.40	1.0936	36	56.7	0	17	34.0	127
Intermediate Shell Plate W10201-4	57.1	3.25	1.3099	20	74.8	0	17	34.0	129
→ Using non-credible surveillance data	67.1	3.25	1.3099	20	87.9	0	17	34.0	142
Intermediate Shell Plate W10201-5	51.2	3.25	1.3099	20	67.1	0	17	34.0	121
→ Using credible surveillance data	38.8	3.25	1.3099	20	50.8	0	8.5	17.0	88
Intermediate Shell Plate W10201-6	44.2	3.25	1.3099	45	57.9	0	17	34.0	137
→ Using non-credible surveillance data	45.9	3.25	1.3099	45	60.1	0	17	34.0	139
Lower Shell Plate W9807-3	58.0	1.16	1.0417	50	60.4	0	17	34.0	144
Lower Shell Plate W9807-5	70.5	1.16	1.0417	33	73.4	0	17	34.0	140
Lower Shell Plate W9807-9	70.5	1.16	1.0417	9	73.4	0	17	34.0	116
Upper Shell Axial Welds 1-273 A, B, & C (Heat # 86054B)	100.8	1.03	1.0081	-56	101.6	17	28	65.5	111
Intermediate Shell Axial Welds 2-273 A, B, & C (Heat # 86054B)	100.8	2.39	1.2350	-56	124.5	17	28	65.5	134
Lower Shell Axial Welds 3-273 A, B, & C (Heat # 86054B)	100.8	1.16	1.0417	-56	105.0	17	28	65.5	115
Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214)	230.2	1.40	1.0936	-56	251.8	17	28	65.5	261
→ Using not fully credible surveillance data	216.3	1.40	1.0936	-56	236.6	17	28	65.5	246
Intermediate Shell to Lower Shell Circumferential Weld 11-273 (Heat # 34B009)	217.1	1.16	1.0417	-77	226.1	0	28	56.0	205
Nozzle Welds	230.2	0.00772	0.0917	-56	0 (21.1)	17	0	34.0	-22
Inlet Nozzle W-10207-1	20	0.00652	0.0813	10	0 (1.6)	0	0	0.0	10

Table 6.1-1 Adjusted Reference Temperature Evaluation for the H.B. Robinson Unit 2 Reactor Vessel Beltline Materials through 50 EFPY at the 1/4T Location under Scenario #1[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	1/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4 T FF	$RT_{NDT(U)}$[(d)] (°F)	Predicted[(e)] ΔRT_{NDT} (°F)	σ_I[(d)] (°F)	σ_Δ[(f)] (°F)	M (°F)	ART (°F)
Inlet Nozzle W-10207-2	20	0.00652	0.0813	10	0 (1.6)	0	0	0.0	10
Inlet Nozzle W-10207-3	20	0.00652	0.0813	10	0 (1.6)	0	0	0.0	10
Outlet Nozzle B-3201-1	137.9	0.00772	0.0917	-7.8	0 (12.6)	0	0	0.0	-7.8
Outlet Nozzle B-3201-2	137.9	0.00772	0.0917	1.6	0 (12.6)	0	0	0.0	1.6
Outlet Nozzle B-3201-3	137.9	0.00772	0.0917	7.2	0 (12.6)	0	0	0.0	7.2

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values. See Reference 14 for details.

(b) CF values taken from Table 3-2.

(c) 1/4T fluence values calculated using Regulatory Guide 1.99, Revision 2 (Reference 14) methodology, surface fluence values from WCAP-18100-NP (Reference 2), and the H.B. Robinson Unit 2 beltline thickness of 9.313 inches.

(d) Initial RT_{NDT} ($RT_{NDT(U)}$) values taken from Table 3-1. $RT_{NDT(U)}$ values are measured values for all base metal materials; thus, σ_I = 0°F. $RT_{NDT(U)}$ values are generic for all welds except the Intermediate to Lower Shell Circumferential Weld. Thus, σ_I = 17°F for all welds except the Intermediate to Lower Shell Circumferential Weld. For the Intermediate to Lower Shell Circumferential Weld, σ_I = 0°F, since $RT_{NDT(U)}$ is a measured value for this material.

(e) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 (Reference 15). Actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(f) As documented in WCAP-15805 (Reference 11), the surveillance capsule data for Intermediate Shell Plate W10201-4 and Intermediate Shell Plate W10201-6 were deemed non-credible, while the surveillance capsule data for Intermediate Shell Plate W10201-5 were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 14), the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data. The base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. The weld metal σ_Δ = 28°F for Position 1.1. The surveillance capsule data for Upper Shell to Intermediate Shell Circumferential Weld 10-273 were deemed not fully credible per Reference 12; a full margin term (σ_Δ = 28°F) must be used for both Position 1.1 and 2.1. However, the lower of the two values may be taken as the ART value. σ_Δ need not exceed 0.5*ΔRT_{NDT} per Reference 14.

Table 6.1-2 Adjusted Reference Temperature Evaluation for the H.B. Robinson Unit 2 Reactor Vessel Beltline Materials through 50 EFPY at the 3/4T Location under Scenario #1[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	3/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4 T FF	$RT_{NDT(U)}$[(d)] (°F)	Predicted[(e)] ΔRT_{NDT} (°F)	σ_I[(d)] (°F)	σ_Δ[(f)] (°F)	M (°F)	ART (°F)
Upper Shell Plate W10201-1	62.9	0.458	0.7828	69	49.2	0	17	34.0	152
Upper Shell Plate W10201-2	84.8	0.458	0.7828	30	66.4	0	17	34.0	130
Upper Shell Plate W10201-3	51.8	0.458	0.7828	36	40.5	0	17	34.0	111
Intermediate Shell Plate W10201-4	57.1	1.06	1.0174	20	58.1	0	17	34.0	112
→ Using non-credible surveillance data	67.1	1.06	1.0174	20	68.3	0	17	34.0	122
Intermediate Shell Plate W10201-5	51.2	1.06	1.0174	20	52.1	0	17	34.0	106
→ Using credible surveillance data	38.8	1.06	1.0174	20	39.5	0	8.5	17.0	76
Intermediate Shell Plate W10201-6	44.2	1.06	1.0174	45	45.0	0	17	34.0	124
→ Using non-credible surveillance data	45.9	1.06	1.0174	45	46.7	0	17	34.0	126
Lower Shell Plate W9807-3	58.0	0.380	0.7321	50	42.5	0	17	34.0	126
Lower Shell Plate W9807-5	70.5	0.380	0.7321	33	51.6	0	17	34.0	119
Lower Shell Plate W9807-9	70.5	0.380	0.7321	9	51.6	0	17	34.0	95
Upper Shell Axial Welds 1-273 A, B, & C (Heat # 86054B)	100.8	0.337	0.7003	-56	70.6	17	28	65.5	80
Intermediate Shell Axial Welds 2-273 A, B, & C (Heat # 86054B)	100.8	0.782	0.9310	-56	93.8	17	28	65.5	103
Lower Shell Axial Welds 3-273 A, B, & C (Heat # 86054B)	100.8	0.380	0.7321	-56	73.8	17	28	65.5	83
Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214)	230.2	0.458	0.7828	-56	180.2	17	28	65.5	190
→ Using not fully credible surveillance data	216.3	0.458	0.7828	-56	169.3	17	28	65.5	179
Intermediate Shell to Lower Shell Circumferential Weld 11-273 (Heat # 34B009)	217.1	0.380	0.7321	-77	158.9	0	28	56.0	138
Nozzle Welds	230.2	0.00253	0.0396	-56	0.0 (9.1)	17	0	34.0	-22
Inlet Nozzle W-10207-1	20	0.00213	0.0346	10	0.0 (0.7)	0	0	0.0	10

Table 6.1-2 Adjusted Reference Temperature Evaluation for the H.B. Robinson Unit 2 Reactor Vessel Beltline Materials through 50 EFPY at the 3/4T Location under Scenario #1[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	3/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4 T FF	$RT_{NDT(U)}$[(d)] (°F)	Predicted[(e)] ΔRT_{NDT} (°F)	σ_I[(d)] (°F)	σ_Δ[(f)] (°F)	M (°F)	ART (°F)
Inlet Nozzle W-10207-2	20	0.00213	0.0346	10	0.0 (0.7)	0	0	0.0	10
Inlet Nozzle W-10207-3	20	0.00213	0.0346	10	0.0 (0.7)	0	0	0.0	10
Outlet Nozzle B-3201-1	137.9	0.00253	0.0396	-7.8	0.0 (5.5)	0	0	0.0	-7.8
Outlet Nozzle B-3201-2	137.9	0.00253	0.0396	1.6	0.0 (5.5)	0	0	0.0	1.6
Outlet Nozzle B-3201-3	137.9	0.00253	0.0396	7.2	0.0 (5.5)	0	0	0.0	7.2

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values. See Reference 14 for details.

(b) CF values taken from Table 3-2.

(c) 3/4T fluence values calculated using Regulatory Guide 1.99, Revision 2 (Reference 14) methodology, surface fluence values from WCAP-18100-NP (Reference 2), and the H.B. Robinson Unit 2 beltline thickness of 9.313 inches.

(d) Initial RT_{NDT} ($RT_{NDT(U)}$) values taken from Table 3-1. $RT_{NDT(U)}$ values are measured values for all base metal materials; thus, σ_I = 0°F. $RT_{NDT(U)}$ values are generic for all welds except the Intermediate to Lower Shell Circumferential Weld. Thus, σ_I = 17°F for all welds except the Intermediate to Lower Shell Circumferential Weld. For the Intermediate to Lower Shell Circumferential Weld, σ_I = 0°F, since $RT_{NDT(U)}$ is a measured value for this material.

(e) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 (Reference 15). Actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(f) As documented in WCAP-15805 (Reference 11), the surveillance capsule data for Intermediate Shell Plate W10201-4 and Intermediate Shell Plate W10201-6 were deemed non-credible, while the surveillance capsule data for Intermediate Shell Plate W10201-5 were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 14), the base metal σ_Δ = 17°F for Position 1.1. The weld metal σ_Δ = 28°F for Position 1.1 and 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. The surveillance capsule data for Upper Shell to Intermediate Shell Circumferential Weld 10-273 were deemed not fully credible per Reference 12; a full margin term (σ_Δ = 28°F) must be used for both Position 1.1 and 2.1. However, the lower of the two values may be taken as the ART value. σ_Δ need not exceed 0.5*ΔRT_{NDT} per Reference 14.

Table 6.1-3 Adjusted Reference Temperature Evaluation for the H.B. Robinson Unit 2 Reactor Vessel Beltline Materials through 50 EFPY at the 1/4T Location under Scenario #2[a]

Reactor Vessel Material and Identification Number	CF[b] (°F)	1/4T Fluence[c] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4 T FF	$RT_{NDT(U)}$[d] (°F)	Predicted[e] ΔRT_{NDT} (°F)	σ_I[d] (°F)	σ_Δ[f] (°F)	M (°F)	ART (°F)
Upper Shell Plate W10201-1	62.9	1.56	1.1221	69	70.6	0	17	34.0	174
Upper Shell Plate W10201-2	84.8	1.56	1.1221	30	95.2	0	17	34.0	159
Upper Shell Plate W10201-3	51.8	1.56	1.1221	36	58.1	0	17	34.0	128
Intermediate Shell Plate W10201-4	57.1	3.37	1.3178	20	75.2	0	17	34.0	129
→ Using non-credible surveillance data	67.1	3.37	1.3178	20	88.4	0	17	34.0	142
Intermediate Shell Plate W10201-5	51.2	3.37	1.3178	20	67.5	0	17	34.0	121
→ Using credible surveillance data	38.8	3.37	1.3178	20	51.1	0	8.5	17.0	88
Intermediate Shell Plate W10201-6	44.2	3.37	1.3178	45	58.2	0	17	34.0	137
→ Using non-credible surveillance data	45.9	3.37	1.3178	45	60.5	0	17	34.0	139
Lower Shell Plate W9807-3	58.0	1.90	1.1763	50	68.2	0	17	34.0	152
Lower Shell Plate W9807-5	70.5	1.90	1.1763	33	82.9	0	17	34.0	150
Lower Shell Plate W9807-9	70.5	1.90	1.1763	9	82.9	0	17	34.0	126
Upper Shell Axial Welds 1-273 A, B, & C (Heat # 86054B)	100.8	1.13	1.0347	-56	104.3	17	28	65.5	114
Intermediate Shell Axial Welds 2-273 A, B, & C (Heat # 86054B)	100.8	2.46	1.2427	-56	125.3	17	28	65.5	135
Lower Shell Axial Welds 3-273 A, B, & C (Heat # 86054B)	100.8	1.90	1.1763	-56	118.6	17	28	65.5	128
Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214)	230.2	1.56	1.1221	-56	258.3	17	28	65.5	268
→ Using not fully credible surveillance data	216.3	1.56	1.1221	-56	242.7	17	28	65.5	252
Intermediate Shell to Lower Shell Circumferential Weld 11-273 (Heat # 34B009)	217.1	1.90	1.1763	-77	255.4	0	28	56.0	234
Nozzle Welds	230.2	0.00898	0.1019	-56	0 (23.4)	17	0	34.0	-22
Inlet Nozzle W-10207-1	20	0.00755	0.0903	10	0 (1.8)	0	0	0.0	10

Table 6.1-3 Adjusted Reference Temperature Evaluation for the H.B. Robinson Unit 2 Reactor Vessel Beltline Materials through 50 EFPY at the 1/4T Location under Scenario #2[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	1/4T Fluence[(c)] (x 10^{19} n/cm², E > 1.0 MeV)	1/4 T FF	$RT_{NDT(U)}$[(d)] (°F)	Predicted[(e)] ΔRT_{NDT} (°F)	σ_I[(d)] (°F)	σ_Δ[(f)] (°F)	M (°F)	ART (°F)
Inlet Nozzle W-10207-2	20	0.00755	0.0903	10	0 (1.8)	0	0	0.0	10
Inlet Nozzle W-10207-3	20	0.00755	0.0903	10	0 (1.8)	0	0	0.0	10
Outlet Nozzle B-3201-1	137.9	0.00898	0.1019	-7.8	0 (14.0)	0	0	0.0	-7.8
Outlet Nozzle B-3201-2	137.9	0.00898	0.1019	1.6	0 (14.0)	0	0	0.0	1.6
Outlet Nozzle B-3201-3	137.9	0.00898	0.1019	7.2	0 (14.0)	0	0	0.0	7.2

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values. See Reference 14 for details.

(b) CF values taken from Table 3-2.

(c) 1/4T fluence values calculated using Regulatory Guide 1.99, Revision 2 (Reference 14) methodology, surface fluence values taken from Section 2 of this WCAP, and the H.B. Robinson Unit 2 beltline thickness of 9.313 inches.

(d) Initial RT_{NDT} ($RT_{NDT(U)}$) values taken from Table 3-1. $RT_{NDT(U)}$ values are measured values for all base metal materials; thus, σ_I = 0°F. $RT_{NDT(U)}$ values are generic for all welds except the Intermediate to Lower Shell Circumferential Weld. Thus, σ_I = 17°F for all welds except the Intermediate to Lower Shell Circumferential Weld. For the Intermediate to Lower Shell Circumferential Weld, σ_I = 0°F, since $RT_{NDT(U)}$ is a measured value for this material.

(e) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 (Reference 15). Actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(f) As documented in WCAP-15805 (Reference 11), the surveillance capsule data for Intermediate Shell Plate W10201-4 and Intermediate Shell Plate W10201-6 were deemed non-credible, while the surveillance capsule data for Intermediate Shell Plate W10201-5 were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 14), the base metal σ_Δ = 17°F for Position 1.1. The weld metal σ_Δ = 28°F for Position 1.1 and 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. The surveillance capsule data for Upper Shell to Intermediate Shell Circumferential Weld 10-273 were deemed not fully credible per Reference 12; a full margin term (σ_Δ = 28°F) must be used for both Position 1.1 and 2.1. However, the lower of the two values may be taken as the ART value. σ_Δ need not exceed 0.5*ΔRT_{NDT} per Reference 14.

Table 6.1-4 Adjusted Reference Temperature Evaluation for the H.B. Robinson Unit 2 Reactor Vessel Beltline Materials through 50 EFPY at the 3/4T Location under Scenario #2[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	3/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4 T FF	$RT_{NDT(U)}$[(d)] (°F)	Predicted[(e)] ΔRT_{NDT} (°F)	σ_I[(d)] (°F)	σ_Δ[(f)] (°F)	M (°F)	ART (°F)
Upper Shell Plate W10201-1	62.9	0.509	0.8114	69	51.0	0	17	34.0	154
Upper Shell Plate W10201-2	84.8	0.509	0.8114	30	68.8	0	17	34.0	133
Upper Shell Plate W10201-3	51.8	0.509	0.8114	36	42.0	0	17	34.0	112
Intermediate Shell Plate W10201-4	57.1	1.10	1.0271	20	58.6	0	17	34.0	113
→ Using non-credible surveillance data	67.1	1.10	1.0271	20	68.9	0	17	34.0	123
Intermediate Shell Plate W10201-5	51.2	1.10	1.0271	20	52.6	0	17	34.0	107
→ Using credible surveillance data	38.8	1.10	1.0271	20	39.9	0	8.5	17.0	77
Intermediate Shell Plate W10201-6	44.2	1.10	1.0271	45	45.4	0	17	34.0	124
→ Using non-credible surveillance data	45.9	1.10	1.0271	45	47.1	0	17	34.0	126
Lower Shell Plate W9807-3	58.0	0.623	0.8674	50	50.3	0	17	34.0	134
Lower Shell Plate W9807-5	70.5	0.623	0.8674	33	61.2	0	17	34.0	128
Lower Shell Plate W9807-9	70.5	0.623	0.8674	9	61.2	0	17	34.0	104
Upper Shell Axial Welds 1-273 A, B, & C (Heat # 86054B)	100.8	0.370	0.7255	-56	73.1	17	28	65.5	83
Intermediate Shell Axial Welds 2-273 A, B, & C (Heat # 86054B)	100.8	0.806	0.9396	-56	94.7	17	28	65.5	104
Lower Shell Axial Welds 3-273 A, B, & C (Heat # 86054B)	100.8	0.623	0.8674	-56	87.4	17	28	65.5	97
Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214)	230.2	0.509	0.8114	-56	186.8	17	28	65.5	196
→ Using not fully credible surveillance data	216.3	0.509	0.8114	-56	175.5	17	28	65.5	185
Intermediate Shell to Lower Shell Circumferential Weld 11-273 (Heat # 34B009)	217.1	0.623	0.8674	-77	188.3	0	28	56.0	167
Nozzle Welds	230.2	0.00294	0.0447	-56	0 (10.3)	17	0	34.0	-22
Inlet Nozzle W-10207-1	20	0.00247	0.0389	10	0 (0.8)	0	0	0.0	10

Table 6.1-4 Adjusted Reference Temperature Evaluation for the H.B. Robinson Unit 2 Reactor Vessel Beltline Materials through 50 EFPY at the 3/4T Location under Scenario #2[(a)]

Reactor Vessel Material and Identification Number	CF[(b)] (°F)	3/4T Fluence[(c)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4 T FF	$RT_{NDT(U)}$[(d)] (°F)	Predicted[(e)] ΔRT_{NDT} (°F)	σ_I[(d)] (°F)	σ_Δ[(f)] (°F)	M (°F)	ART (°F)
Inlet Nozzle W-10207-2	20	0.00247	0.0389	10	0 (0.8)	0	0	0.0	10
Inlet Nozzle W-10207-3	20	0.00247	0.0389	10	0 (0.8)	0	0	0.0	10
Outlet Nozzle B-3201-1	137.9	0.00294	0.0447	-7.8	0 (6.2)	0	0	0.0	-7.8
Outlet Nozzle B-3201-2	137.9	0.00294	0.0447	1.6	0 (6.2)	0	0	0.0	1.6
Outlet Nozzle B-3201-3	137.9	0.00294	0.0447	7.2	0 (6.2)	0	0	0.0	7.2

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values. See Reference 14 for details.

(b) CF values taken from Table 3-2.

(c) 3/4T fluence values calculated using Regulatory Guide 1.99, Revision 2 (Reference 14) methodology, surface fluence values taken from Section 2 of this WCAP, and the H.B. Robinson Unit 2 beltline thickness of 9.313 inches.

(d) Initial RT_{NDT} ($RT_{NDT(U)}$) values taken from Table 3-1. $RT_{NDT(U)}$ values are measured values for all base metal materials; thus, σ_I = 0°F. $RT_{NDT(U)}$ values are generic for all welds except the Intermediate to Lower Shell Circumferential Weld. Thus, σ_I = 17°F for all welds except the Intermediate to Lower Shell Circumferential Weld. For the Intermediate to Lower Shell Circumferential Weld, σ_I = 0°F, since $RT_{NDT(U)}$ is a measured value for this material.

(e) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 (Reference 15). Actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(f) As documented in WCAP-15805 (Reference 11), the surveillance capsule data for Intermediate Shell Plate W10201-4 and Intermediate Shell Plate W10201-6 were deemed non-credible, while the surveillance capsule data for Intermediate Shell Plate W10201-5 were deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 (Reference 14), the base metal σ_Δ = 17°F for Position 1.1. The weld metal σ_Δ = 28°F for Position 1.1 and 2.1 with non-credible surveillance data, and σ_Δ = 14°F for Position 2.1 with credible surveillance data. The surveillance capsule data for Upper Shell to Intermediate Shell Circumferential Weld 10-273 were deemed not fully credible per Reference 12; a full margin term (σ_Δ = 28°F) must be used for both Position 1.1 and 2.1. However, the lower of the two values may be taken as the ART value. σ_Δ need not exceed 0.5*ΔRT_{NDT} per Reference 14.

Table 6.1-5 Summary of the H.B. Robinson Unit 2 Limiting ART Values used in the Applicability Evaluation of the Existing 50 EFPY Reactor Vessel Heatup and Cooldown Curves under Scenario #1 and Scenario #2

Limiting Material	1/4T Limiting ART (°F)			3/4T Limiting ART (°F)		
	Existing Curves[(a)]	Scenario #1[(b)]	Scenario #2[(d)]	Existing Curves[(a)]	Scenario #1[(c)]	Scenario #2[(e)]
Upper Shell to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214, Position 2.1) (Circumferential Flaw)	263	246	252	191	179	185
Upper Shell Plate W10201-1 (Position 1.1) (Axial Flaw)	172	172	174[(f)]	153	152	154[(g)]

Notes:

(a) Values taken from Table 18 of WCAP-15827, Revision 0 (Reference 21).
(b) Values taken from Table 6.1-1.
(c) Values taken from Table 6.1-2.
(d) Values taken from Table 6.1-3.
(e) Values taken from Table 6.1-4.
(f) This value will exceed that used to develop the existing curves at 46.3 EFPY.
(g) This value will exceed that used to develop the existing curves at 47.6 EFPY.

7 SURVEILLANCE CAPSULE REMOVAL SCHEDULE

The following surveillance capsule removal schedules (Table 7-1 and Table 7-2 for Scenario #1 and Scenario #2, respectively) meet the recommendations of ASTM E185-82 (Reference 22) as required by 10 CFR 50, Appendix H (Reference 18). Note that it is recommended for future capsule(s) to be removed from the H.B. Robinson Unit 2 reactor vessel.

Table 7-1 H.B. Robinson Unit 2 Surveillance Capsule Withdrawal Schedule – Scenario #1[(a)]

Capsule	Location	Lead Factor[(a)]	Withdrawal EFPY[(b)]	Fluence (n/cm^2, E > 1.0 MeV)
S	280°	1.90	EOC 1 (1.34)	4.79×10^{18}
V	290°	0.91	EOC 3 (3.33)	5.30×10^{18}
T	270°	2.80	EOC 8 (7.53)	3.87×10^{19}
X	270° (50°)	1.63	EOC 20 (20.65)	4.49×10^{19}
U	280° (30°)	1.95	39.7 – 58.7[(c)]	7.61×10^{19} – 1.14×10^{20} [(c)]
Y	290° (150°)	0.89	Standby	Note (d)
W	270° (40°)	2.72	Standby	Note (d)
Z	40°	0.46	EOC 1 (1.34)	Note (e)

Notes:

(a) For Capsules S, V, T, X, and Z the lead factors shown are the lead factor of the capsule over the entire irradiation period. For Capsules U, Y, and W the lead factors shown are the ratio of the fluence rate of the capsule at its current location to the vessel maximum fluence rate.

(b) EFPY from plant startup.

(c) In order to satisfy the 5th capsule requirement for both a 60-year and 80-year life, Capsule U should be withdrawn after 39.7 EFPY, which corresponds to a fluence value of approximately 7.61×10^{19} n/cm^2 (projected maximum EOSLR vessel fluence value). However, Capsule U must be withdrawn before 58.7 EFPY, which corresponds to a fluence value of approximately 1.14×10^{20} n/cm^2 (two times the projected maximum EOLE vessel fluence value). Capsule U should be withdrawn the cycle following 39.7 EFPY. This is consistent with the recommendations of MRP-326 (Reference 23).

(d) Capsules Y and W should remain in the reactor. If additional metallurgical data is needed for H.B. Robinson Unit 2, relocation, withdrawal, and/or testing of these capsules can be considered.

(e) Capsule Z was inadvertently removed from the reactor vessel per Reference 11.

Table 7-2 H.B. Robinson Unit 2 Surveillance Capsule Withdrawal Schedule – Scenario #2[(a)]

Capsule	Location	Lead Factor[(b)]	Withdrawal EFPY[(b)]	Fluence (n/cm^2, E > 1.0 MeV)
S	280°	1.90	EOC 1 (1.34)	4.79 x 10^{18}
V	290°	0.91	EOC 3 (3.33)	5.30 x 10^{18}
T	270°	2.80	EOC 8 (7.53)	3.87 x 10^{19}
X	270° (50°)	1.63	EOC 20 (20.65)	4.49 x 10^{19}
U	280° (30°)	1.91	41.3 – 57.1[(c)]	8.09 x 10^{19} – 1.18 x 10^{20} [(c)]
Y	290° (150°)	0.94	Standby	Note (d)
W	270° (40°)	2.98	Standby	Note (d)
Z	40°	0.46	EOC 1 (1.34)	Note (e)

Notes:

(a) For Capsules S, V, T, X, and Z the lead factors shown are the lead factor of the capsule over the entire irradiation period. For Capsules U, Y, and W the lead factors shown are the ratio of the fluence rate of the capsule at its current location to the vessel maximum fluence rate.

(b) EFPY from plant startup.

(c) In order to satisfy the 5th capsule requirement for both a 60-year and 80-year life, Capsule U should be withdrawn after 41.3 EFPY, which corresponds to a fluence value of approximately 8.09 x 10^{19} n/cm^2 (projected maximum EOSLR vessel fluence value). However, Capsule U must be withdrawn before 57.1 EFPY, which corresponds to a fluence value of approximately 1.18 x 10^{20} n/cm^2 (two times the projected maximum EOLE vessel fluence value). Capsule U should be withdrawn the cycle following 41.3 EFPY. This is consistent with the recommendations of MRP-326 (Reference 23).

(d) Capsules Y and W should remain in the reactor. If additional metallurgical data is needed for H.B. Robinson Unit 2, relocation, withdrawal, and/or testing of these capsules can be considered.

(e) Capsule Z was inadvertently removed from the reactor vessel per Reference 11.

8 REFERENCES

1. U.S. Nuclear Regulatory Commission, *Generic Aging Lessons Learned (GALL) Report*, NUREG-1801, Revision 2, December 2010. *[Agencywide Document Access and Management System (ADAMS) Accession Number ML103490041]*

2. Westinghouse Report WCAP-18100-NP, Revision 0, *Ex-Vessel Neutron Dosimetry Program for H. B. Robinson Unit 2 Cycles 16 through 29*, March 2016.

3. U.S. Nuclear Regulatory Commission Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001.

4. Westinghouse Report WCAP-14040-A, Revision 4, *Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves*, May 2004.

5. RSICC Computer Code Collection CCC-650, *DOORS 3.2: One, Two- and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System*, April 1998.

6. RSICC Data Library Collection DLC-185, *BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications*, March 1996.

7. Westinghouse Report WCAP-16083-NP, Revision 1, *Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry*, April 2013.

8. U.S. Nuclear Regulatory Commission Regulatory Issue Summary (RIS) 2014-11, *Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components*, dated October 14, 2014. *[ADAMS Accession Number ML14149A165]*

9. Westinghouse Report WCAP-15828, Revision 0, *Evaluation of Pressurized Thermal Shock for H.B. Robinson Unit 2*, March 2003.

10. Westinghouse Letter MCOE-LTR-16-33, Revision 0, *H.B. Robinson Unit 2 Cylindrical Shell Pressure-Temperature Limit Curve Applicability Check and Reactor Vessel Nozzle Pressure-Temperature Limit Curve Development for Extended Plant Operation*, August 2016.

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

12. Structural Integrity Associates Report 0901132.401, Revision 0, *Evaluation of Surveillance Data for Weld Heat No. W5214 for Application to Palisades PTS Analysis*, April 2010. *[ADAMS Accession Number ML110060693]*

13. Code of Federal Regulations 10 CFR 50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events*, Federal Register, Volume 60, No. 243, dated December 19, 1995, effective January 18, 1996.

14. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials*, May 1988.

15. U.S. Nuclear Regulatory Commission Technical Letter Report TLR-RES/DE/CIB-2013-01, *Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels*, Office of Nuclear Regulatory Research [RES], dated November 14, 2014. *[ADAMS Accession Number ML14318A177]*

16. Code of Federal Regulations, 10 CFR 50.61a, *Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events*, Federal Register, Volume 75, No. 1, dated January 4, 2010, with corrections dated February 3, 2010 (No. 22), March 8, 2010 (No. 44), and November 26, 2010 (No. 227).

17. Westinghouse Owners Group Document HF04BG, *Background Information for Westinghouse Owners Group Emergency Response Guidelines, Critical Safety Function Status Tree, F-0.4 Integrity, HP/LP-Rev. 2*, April 2005.

18. Code of Federal Regulations 10 CFR 50, Appendix G, *Fracture Toughness Requirements,* and Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, Federal Register, Volume 60, No. 243, dated December 19, 1995.

19. Westinghouse Report WCAP-13587, Revision 1, *Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors*, September 1993.

20. Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, *Fracture Toughness Criteria for Protection Against Failure.*

21. Westinghouse Report WCAP-15827, Revision 0, *H. B. Robinson Unit 2 Heatup and Cooldown Curves for Normal Operation*, March 2003.

22. ASTM E185-82, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels*, American Society for Testing and Materials, 1982.

23. *Materials Reliability Program: Coordinated PWR Reactor Vessel Surveillance Program (CRVSP) Guidelines (MRP-326).* EPRI, Palo Alto, CA: 2011. 1022871.