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U. S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

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

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

Dear Sir/Madam:

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

The Nuclear Regulatory Commission (NRC) staff determined that additional information is needed to complete its LAR review. A draft of that information request was received by DEP via electronic mail message dated March 2, 2016, which provided four (4) requests for additional information (RAIs). An RAI clarification call was held on March 15, 2016 between NRC staff and DEP during which DEP requested a revised RAI response date for RAI No. 3 to accommodate outstanding vendor deliverables necessary for DEP's response. The DEP responses to RAIs 1, 2, and 4 were provided to NRC via letter dated March 31, 2016. The attachment to this letter provides Westinghouse report, MCOE-LTR-16-33, Rev. 0, which is intended to address RAI No. 3.

Please address any comments or questions regarding this matter to Mr. Tony Pilo, Acting Manager – Nuclear Regulatory Affairs at (843) 857-1409.

There are no new regulatory commitments made in this letter.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 14, 2016.

Sincerely,

FOR

R. Michael Glover
Site Vice President

RMG/jmw

Attachment: Westinghouse Report MCOE-LTR-16-33, Rev. 0

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

U. S. Nuclear Regulatory Commission
Enclosure to Serial: RNP-RA/16-0073
23 Pages (including this cover page)

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

REQUEST FOR ADDITIONAL INFORMATION
LICENSE AMENDMENT REQUEST
REVISION OF REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS

DUKE ENERGY PROGRESS, INC.
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2
DOCKET NO. 50-261

RAI-3

Title 10 of the Code of Federal Regulations (10 CFR), Part 50, Appendix G, requires that P-T limits be developed to bound all ferritic materials in the reactor pressure vessel (RPV). Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," dated October 14, 2014 (ADAMS Accession No. ML14149A165) clarifies that P-T limit calculations for ferritic RPV materials other than those materials with the highest reference temperature may define P-T curves that are more limiting because the consideration of stress levels from structural discontinuities (such as RPV inlet and outlet nozzles) may produce a lower allowable pressure. The staff noted that the licensee addressed the fluence levels of the RPV inlet and outlet nozzles for the 50 EFPY in WCAP-15827, Table 6, "Summary of the Vessel Surface, 1/4T and 3/4T Fluence Values used for the Generation of the 30, 35, 40, 45 and 50 EFPY Heatup/Cooldown Curves," and reported the adjusted reference temperatures (ART) for the RPV inlet and outlet nozzles in Table 16, "Calculation of the ART Values for the 1/4T Location @ 50 EFPY" and Table 17, "Calculation of the ART Values for the 3/4T Location @ 50 EFPY." However, WCAP-15827 does not have P-T limit calculations for the RPV inlet and outlet nozzles, and therefore, does not demonstrate how the P-T limit curves developed for 50 EFPY bound all ferritic pressure boundary components of the RPV.

Therefore, the staff requests the licensee to provide P-T limit calculations for the Robinson RPV inlet and outlet nozzles or otherwise demonstrate how the P-T limit curves developed for 50 EFPY in WCAP-15827 bound all ferritic pressure boundary components of the RPV. In the P-T limit calculations for the Robinson RPV inlet and outlet nozzles, the staff requests the following to be used: 1) the ART values of the Robinson RPV inlet nozzle, outlet nozzle, and "Nozzle Welds" in Tables 16 and 17 of WCAP-15827, and 2) consideration of the stress levels in the welds that attach the Robinson RPV inlet and outlet nozzles to the RPV. Lastly, the staff requests the licensee to confirm that here are no other ferritic pressure boundary components of the Robinson RPV that need to be considered for P-T limit evaluation for the period of extended operation.

DEP Response:

See following pages containing Westinghouse Report, MCOE-LTR-16-33, Rev. 0.

NP-Attachment
(Non-Proprietary)

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

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

Introduction

For Westinghouse nuclear steam supply systems such as H.B. Robinson Unit 2, the Topical Report WCAP-14040-NP-A, Revision 2 [Ref. 1] describes the methodology that is used to comply with the requirements of 10 CFR 50 Appendix G, "Fracture Toughness Requirements" [Ref. 2]. Since the reactor vessel materials surrounding the core region receive significant neutron fluence and undergo neutron embrittlement, the reactor vessel beltline region is considered to be the limiting reactor coolant system (RCS) component for P-T limits. The methodology in WCAP-14040-NP-A, Revision 2 only addresses the reactor vessel beltline region of the RCS as the most limiting for the P-T limits. The original Nuclear Regulatory Commission (NRC) Safety Evaluation (SE) for this topical report states, "We find the report to be acceptable for referencing in the administrative controls section of technical specifications for license amendment applications to the extent specified and under the limitations delineated in the report and the associated NRC safety evaluation, which is enclosed. The safety evaluation defines the basis for acceptance of the report." The SE further states, "The staff finds the WCAP-14040 methodology consistent with Appendix G to Section III of the ASME Code and SRP Section 5.3.2." and "T is the metal temperature and RT_{NDT} is the ART value of the limiting vessel material" thereby confirming that the reactor vessel is the limiting component evaluated in the development of the P-T limits. Table 1 of the NRC SE provides requirements regarding the fluence methodology, surveillance capsule program requirements, low-temperature overpressure protection system (LTOPS) requirements, adjusted reference temperature (ART) calculation, and 10 CFR 50 Appendix G temperature requirements, which have all been addressed for H.B. Robinson Unit 2 in WCAP-15827, Revision 0 [Ref. 3], consistent with the NRC SE.

Before implementation, the applicability of the 50 effective full-power years (EFPY) P-T limit curves for H.B. Robinson Unit 2 documented in WCAP-15827, Revision 0 [Ref. 3] must be confirmed. An updated fluence analysis, considering all currently completed fuel cycles and the most up-to-date prediction of future fuel cycles was completed and documented in WCAP-18100-NP, Revision 0 [Ref. 4]. The fluence values documented in WCAP-18100-NP, Revision 0 [Ref. 4] and all relevant sister-plant data are used herein to confirm that the 50 EFPY P-T limit curves documented in WCAP-15827, Revision 0 [Ref. 3] remain applicable through 50 EFPY for H.B. Robinson Unit 2.

Additionally, the H.B. Robinson reactor vessel inlet and outlet nozzles must now be considered during development of pressure-temperature (P-T) limit curves due to the recent issuance of the Regulatory Issue Summary (RIS) 2014-11 [Ref. 5]. Although one quarter-thickness (1/4T) and three quarter-thickness (3/4T) Adjusted Reference Temperature (ART) values were calculated in WCAP-15827, Revision 0 [Ref. 3], nozzle P-T limit curves based on nozzle-specific stresses have not been previously analyzed for H.B. Robinson Unit 2. The reactor vessel inlet and outlet nozzle corner region is more stressed than the cylindrical vessel regions due to the geometric discontinuity at the nozzle inner radius. As a result, the inlet and outlet nozzle corner regions, along with the vessel flange region, are considered to be the bounding stress concentration within the reactor vessel. Although the nozzle welds are subjected to primary (pressure) and secondary (cooldown transient) stresses, the nozzle welds are bounded by the larger nozzle corner region stresses because of the discontinuities at the nozzle inner radius. Also, the nozzle welds are included in the cylindrical shell P-T limits.

Note that the 10 CFR 50, Appendix G [Ref. 2] flange requirements have previously been incorporated into the 50 EFPY cylindrical shell P-T limits per WCAP-15827, Revision 0 [Ref. 3]. Therefore, the 50 EFPY cylindrical shell P-T limit curves previously generated for H.B. Robinson Unit 2 must now be checked against the nozzle P-T limit curves to ensure that the 50 EFPY cylindrical shell curves remain the most limiting.

50 EFPY Cylindrical Shell P-T Limit Curves Applicability Check

The H.B. Robinson Unit 2 calculated maximum neutron fluence projections at 50 EFPY for the reactor vessel materials were originally documented in Table 6 of WCAP-15827, Revision 0 [Ref. 3]. However, these values were recently updated in an Ex-Vessel Neutron Dosimetry (EVND) analysis, WCAP-18100-NP [Ref. 4]. Table 1 compares the beltline fluence projections from the original 50 EFPY cylindrical shell P-T limit curves report [Ref. 3] with those in the most recent EVND fluence analysis [Ref. 4]. Per RIS 2014-11 [Ref. 5] guidance, beltline materials were determined to be the reactor vessel materials that will be exposed to a neutron fluence greater than or equal to 1 x 10^{17} n/cm^2 (E > 1.0 MeV) at end of license (EOL, 50 EFPY).

Table 1
H.B. Robinson Unit 2 Calculated Neutron Fluence Projections on the Reactor Vessel Beltline Materials at 50 EFPY

Reactor Vessel Material	50 EFPY Fluence (n/cm^2, E > 1.0 MeV)	
	WCAP-15827	WCAP-18100-NP
Inlet Nozzle to Upper Shell Weld – Lowest Extent[(a)]	3.93 x 10^{17}	1.14 x 10^{17}
Outlet Nozzle to Upper Shell Weld – Lowest Extent[(a)]	2.53 x 10^{17}	1.35 x 10^{17}
Upper Shell Plates	2.50 x 10^{19}	2.45 x 10^{19}
Upper Shell Longitudinal Welds	2.50 x 10^{19}	1.80 x 10^{19}
Upper Shell to Intermediate Shell Circumferential Weld	2.50 x 10^{19}	2.45 x 10^{19}
Intermediate Shell Plates	6.01 x 10^{19}	5.69 x 10^{19}
Intermediate Shell Longitudinal Welds	4.46 x 10^{19}	4.18 x 10^{19}
Intermediate to Lower Shell Circumferential Weld	2.05 x 10^{19}	2.03 x 10^{19}
Lower Shell Plates	2.05 x 10^{19}	2.03 x 10^{19}
Lower Shell Longitudinal Welds	2.05 x 10^{19}	2.03 x 10^{19}

Note for Table 1:

(a) The fluence for the inlet and outlet nozzle to upper shell welds was also conservatively used as the fluence for the inlet and outlet nozzle materials. The actual nozzle forging fluence values, at the location of a postulated 1/4T flaw along the nozzle corner region, are expected to be lower since they are further away from the active core.

Per Table 1, the fluence values reported in WCAP-15827, Revision 0 [Ref. 3] are conservative when compared to those from the revised analysis, WCAP-18100-NP, Revision 0 [Ref. 4]. Thus, the ART values calculated in development of the P-T limit curves in WCAP-15827, Revision 0 [Ref. 3] are conservative for all materials analyzed, except potentially the upper to intermediate shell circumferential weld material (Heat # W5214). The ART value for this material must be re-calculated, because additional sister plant data now exists for this weld heat that was not available for incorporation in WCAP-15827, Revision 0 [Ref. 3]. Thus, the upper to intermediate shell circumferential weld material is the only cylindrical shell material re-analyzed as a part of this P-T limits applicability evaluation.

The H.B. Robinson Unit 2 upper to intermediate shell circumferential weld material was fabricated using weld Heat # W5214. In addition to being contained in the H.B. Robinson Unit 2 surveillance program, weld Heat # W5214 is contained in the Palisades supplemental surveillance program, as well as in the Indian Point Units 2 and 3 surveillance programs. Thus, the H.B. Robinson Unit 2, Palisades, and Indian Point Units 2 and 3 data will be used in the calculation of the Position 2.1 chemistry factor (CF) value for the H.B. Robinson Unit 2 weld Heat # W5214. Table 2 summarizes the applicable surveillance capsule data pertaining to weld Heat # W5214, and Table 3 shows the calculation of the Position 2.1 CF. The combined surveillance data is considered not fully credible per SIA Report 0901132.401, Revision 0 [Ref. 6]. 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 Reference 6. This calculation methodology is consistent with the NRC-approved conclusions of ML112870050 [Ref. 7] and ML113480303 [Ref. 8].

H.B. Robinson Unit 2 upper to intermediate shell circumferential weld seam (Heat # W5214) 1/4T and 3/4T ART values were calculated per Regulatory Guide 1.99, Revision 2 [Ref. 9]. These ART calculations are summarized in Tables 4 and 5.

Table 2
Surveillance Capsule Data for Weld Heat # W5214[a]

Weld Metal Heat # W5214	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured ΔRT_{NDT} (°F)	$T_{capsule}$ (°F)	Temperature Adjustment[b] (°F)	Wt. % Cu	Wt. % Ni	Position 1.1 CF[c] (°F)
H.B. Robinson Unit 2 Data	V	0.530[d]	208.8	547	0.0	0.34	0.66	217.7
	T	3.87[d]	289.1	547	0.0			
	X	4.49[d]	265.6	547	0.0			
Palisades Data	SA-60-1	1.50	259	535.0	-12.0	0.307	1.045	266.5
	SA-240-1	2.38	280.1	535.7	-11.3			
Indian Point Unit 2 Data	Y	0.455	193.9	529.1	-17.9	0.20	1.03	226.3
	V	0.492	197.5	524	-23.0			
Indian Point Unit 3 Data	T	0.263	149.8	539.4	-7.6	0.16	1.12	206.2
	Y	0.692	171.1	539.5	-7.5			
	X	0.874	192.5	539.7	-7.3			
	Z	1.04	228.3	538.9	-8.1			

Notes for Table 2:

(a) All data contained here taken from SIA Report 0901132.401, Revision 0 [Ref. 6], unless otherwise noted.

(b) Temperature adjustment = 1.0*($T_{capsule}$ - T_{plant}), where T_{plant} = 547°F for H.B. Robinson Unit 2 (applied to the weld ΔRT_{NDT} data for each of the Palisades, Indian Point Unit 2, and Indian Point Unit 3 capsules in the Position 2.1 chemistry factor calculation).

(c) CF = Chemistry factor. Calculated using Table 1 of Regulatory Guide 1.99, Revision 2 [Ref. 9]. Note that per WCAP-15827, Revision 0 [Ref. 3] the Position 1.1 Chemistry Factor (CF) of the H.B. Robinson Unit 2 upper to intermediate shell circumferential weld seam is 230.2°F.

(d) The fluence values assigned to the H.B. Robinson Unit 2 capsules are consistent with the values documented in WCAP-15827, Revision 0 [Ref. 3] and WCAP-18100-NP, Revision [Ref. 4].

Table 3
Calculation of H.B. Robinson Unit 2 Position 2.1 Chemistry Factor Value for Weld Heat # W5214 Using Surveillance Capsule Data

Weld Metal Heat # W5214	Capsule	Capsule Fluence[a] ($x\ 10^{19}\ n/cm^2$, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
H.B. Robinson Unit 2 Data	V	0.530	0.823	221.33 (208.8)	182.07	0.677
	T	3.87	1.349	306.45 (289.1)	413.40	1.820
	X	4.49	1.381	281.54 (265.6)	388.71	1.906
Palisades Data	SA-60-1	1.50	1.112	212.42 (259)	236.27	1.237
	SA-240-1	2.38	1.234	231.17 (280.1)	285.23	1.522
Indian Point Unit 2 Data	Y	0.455	0.781	179.52 (193.9)	140.17	0.610
	V	0.492	0.802	177.99 (197.5)	142.78	0.643
Indian Point Unit 3 Data	T	0.263	0.637	159.26 (149.8)	101.41	0.405
	Y	0.692	0.897	183.23 (171.1)	164.31	0.804
	X	0.874	0.962	207.42 (192.5)	199.59	0.926
	Z	1.04	1.011	246.62 (228.3)	249.33	1.022
	SUM :				2503.25	11.573
	$CF_{Heat\ \#\ W5214} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (2503.25) \div (11.573) =$ **216.3°F**					

Notes for Table 3:

(a) Taken from Table 2 herein.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. The ΔRT_{NDT} values are adjusted first by the difference in operating temperature then using the ratio procedure to account for differences in the surveillance weld chemistry and the reactor vessel weld chemistry (pre-adjusted values are listed in parentheses and were taken from Table 2 of this report). The temperature adjustments are listed in Table 2 herein. Ratio applied to the H.B. Robinson Unit 2 surveillance data = $CF_{Vessel\ Weld} / CF_{Surv.\ Weld}$ = 230.2°F / 217.7°F = 1.06. Ratio applied to the Palisades surveillance data = $CF_{Vessel\ Weld} / CF_{Surv.\ Weld}$ = 230.2°F / 266.5°F = 0.86. Ratio applied to the Indian Point Unit 2 surveillance data = $CF_{Vessel\ Weld} / CF_{Surv.\ Weld}$ = 230.2°F / 226.3°F = 1.02. Ratio applied to the Indian Point Unit 3 surveillance data = $CF_{Vessel\ Weld} / CF_{Surv.\ Weld}$ = 230.2°F / 206.2°F = 1.12. The Position 1.1 CF for the H.B. Robinson Unit 2 reactor vessel weld of 230.2°F was taken from WCAP-15827, Revision 0 [Ref. 3].

Table 4

ART Calculations at the 1/4T Location for the H.B. Robinson Unit 2 Upper to Intermediate Shell Circumferential Weld Seam 10-273[a]

Reactor Vessel Material and ID Number	Heat Number	CF (°F)	1/4T Fluence[d] (n/cm^2, E > 1.0 MeV)	1/4T FF[d]	$RT_{NDT(U)}$[e] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[f] (°F)	Margin (°F)	ART (°F)
Upper to Intermediate Shell Circumferential Weld 10-273	W5214	230.2[b]	1.40×10^{19}	1.0936	-56	251.8	17	28	65.5	261
Using surveillance data	W5214	216.3[c]	1.40×10^{19}	1.0936	-56	236.6	17	28	65.5	246

Notes for Table 4:

(a) The Regulatory Guide 1.99, Revision 2 [Ref. 9] methodology was utilized in the calculation of the ART values.

(b) This Position 1.1 CF was taken from WCAP-15827, Revision 0 [Ref. 3]

(c) This Position 2.1 CF was taken from Table 3.

(d) 1/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2 [Ref. 9], the H.B. Robinson Unit 2 reactor vessel beltline thickness of 9.313 inches, and the surface fluence value reported in Table 1.

(e) Taken from Table 1 WCAP-15827, Revision 0 [Ref. 3].

(f) As documented previously, the surveillance data for weld Heat # W5214 was deemed not fully credible. Thus, 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 (consistent with NRC-approved conclusions of References 7 and 8).

Table 5

ART Calculations at the 3/4T Location for the H.B. Robinson Unit 2 Upper to Intermediate Shell Circumferential Weld Seam 10-273[a]

Reactor Vessel Material and ID Number	Heat Number	CF (°F)	3/4T Fluence[d] (n/cm^2, E > 1.0 MeV)	3/4T FF[d]	$RT_{NDT(U)}$[e] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[f] (°F)	Margin (°F)	ART (°F)
Upper to Intermediate Shell Circumferential Weld 10-273	W5214	230.2[b]	4.58×10^{18}	0.7828	-56	180.2	17	28	65.5	190
Using surveillance data	W5214	216.3[c]	4.58×10^{18}	0.7828	-56	169.3	17	28	65.5	179

Notes for Table 5:

(a) The Regulatory Guide 1.99, Revision 2 [Ref. 9] methodology was utilized in the calculation of the ART values.

(b) This Position 1.1 CF was taken from WCAP-15827, Revision 0 [Ref. 3]

(c) This Position 2.1 CF was taken from Table 3.

(d) 3/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2 [Ref. 9], the H.B. Robinson Unit 2 reactor vessel beltline thickness of 9.313 inches, and the surface fluence value reported in Table 1.

(e) Taken from Table 1 WCAP-15827, Revision 0 [Ref. 3].

(f) As documented previously, the surveillance data for weld Heat # W5214 was deemed not fully credible. Thus, 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 (consistent with NRC-approved conclusions of References 7 and 8).

In order to check the applicability of the 50 EFPY cylindrical shell curves documented in WCAP-15827, Revision 0 [Ref. 3], the limiting 1/4T and 3/4T ART values used to develop the curves must be compared to the revised upper to intermediate shell circumferential weld 1/4T and 3/4T ART values documented in Tables 4 and 5, respectively. Table 6 compares the revised ART values for the H.B. Robinson Unit 2 upper to intermediate circumferential weld at 50 EFPY with those used in the development of the P-T limit curves documented in WCAP-15827, Revision 0 [Ref. 3]. The values calculated in WCAP-15827, Revision 0 [Ref. 3] for the cylindrical shell remain bounding; thus, the 50 EFPY cylindrical shell P-T limit curves documented in WCAP-15827, Revision 0 [Ref. 3] remain applicable through 50 EFPY.

Table 6
H.B. Robinson Unit 2 Revised and Limiting Cylindrical Shell ART Values at 50 EFPY[(a)]

		ART Value (°F)	
		1/4T Location	**3/4T Location**
Upper to Intermediate Shell Circumferential Weld 10-273 (Heat # W5214)	Limiting[(a)]	263	191
	Re-calculated, herein[(b)]	246	179

Notes for Table 6:

(a) Values were taken from WCAP-15827, Revision 0 [Ref. 3] and represent circumferential flaw ART values. These values are conservative when compared to the revised ART values based on updated fluence and all applicable capsule results. Thus, the 50 EFPY cylindrical shell P-T limit curves documented in WCAP-15827, Revision 0 [Ref. 3] remain applicable through 50 EFPY.

(b) Values were taken from Tables 4 and 5 and represent circumferential flaw ART values. Since the data was not fully credible, a full margin term was used; however, the Position 2.1 ART value was assigned to this material, since the Position 2.1 value was lower (consistent with NRC-approved conclusions of References 7 and 8).

Nozzle Initial RT_{NDT} and ART Calculations

H.B. Robinson reactor vessel inlet and outlet nozzle material properties were used to determine initial RT_{NDT} and the ART values per Regulatory Guide 1.99, Revision 2 [Ref. 9]. The initial RT_{NDT} values for the reactor vessel inlet nozzle forging materials were determined using the Branch Technical Position (BTP) 5-3 Position 1.1(4) methodology [Ref. 10] since limited Charpy V-notch tests were performed at a single temperature for the inlet nozzles. The initial RT_{NDT} values for the reactor vessel outlet nozzle forging materials were determined using the BWRVIP-173-A, Appendix B, Alternative Approach 2 methodology [Ref. 11] since no drop-weight test data was available for the outlet nozzles. These updated initial RT_{NDT} values are summarized in Table 7. The copper (Cu) and nickel (Ni) chemistry information and CF values (determined per Reference 9) for each of the H.B. Robinson Unit 2 reactor vessel nozzle materials are also summarized in Table 7.

Table 7
Summary of the H.B. Robinson Unit 2 Reactor Vessel Nozzle Material Properties[a]

Reactor Vessel Material	Material Heat ID #	Wt. % Cu	Wt. % Ni	Position 1.1 CF[b] (°F)	$RT_{NDT(u)}$[c] (°F)
Inlet Nozzle W-10207-1	X15156/X53163	0.02	0.90	20	10
Inlet Nozzle W-10207-2	X15156/X53163	0.02	0.90	20	10
Inlet Nozzle W-10207-3	X15156/X53163	0.02	0.90	20	10
Outlet Nozzle B-3201-1	BT2305	[[____{E}]][d]	0.71	137.9	-7.8
Outlet Nozzle B-3201-2	BT2305	[[____{E}]][d]	0.71	137.9	1.6
Outlet Nozzle B-3201-3	BT2305	[[____{E}]][d]	0.71	137.9	7.2

Notes for Table 7:

(a) Nozzle information was determined per the H.B. Robinson Unit 2 Certified Material Test Reports (CMTRs). Nozzle chemistry values are the maximum of the reported values in the CMTRs, unless otherwise noted.

(b) The Position 1.1 CF values for the nozzle materials are calculated based on Table 2 of Regulatory Guide 1.99, Revision 2 [Ref. 9] using the Cu and Ni weight percent (wt. %) values summarized in this table.

(c) Inlet nozzle forging initial RT_{NDT} values were determined using the Branch Technical Position (BTP) 5-3 Position 1.1(4) methodology [Ref. 10]. Outlet nozzle forging initial RT_{NDT} values for the reactor vessel outlet nozzle forging materials were determined using the BWRVIP-173-A, Appendix B, Alternative Approach 2 methodology [Ref. 11].

(d) Since no Cu wt. % data exists for these materials, Cu wt. % values are the best-estimate values for SA-508, Class 2 low allow steels as documented in BWRVIP-173-A [Ref. 11]. This is Electric Power Research Institute (EPRI) proprietary information.

Table 8 shows the calculation of the maximum ART values for the H.B. Robinson Unit 2 nozzle materials. The fast neutron fluences (E > 1.0 MeV) for the inlet and outlet nozzle to upper shell welds (lowest extent) were calculated conservatively in Reference 4. The fluences for the inlet and outlet nozzle to upper shell welds (lowest extent) were taken from an elevation several centimeters closer to the core than the actual location of the welds. In addition, the methodology that was used to calculate the fast neutron fluences (E > 1.0 MeV) utilized the two-dimensional/one-dimensional fluence rate synthesis technique to obtain three-dimensional (3D) synthesized fluence rates. The fluence rate synthesis technique provides more conservative results (more than 50% in relative difference) near nozzle elevations, compared to full 3D radiation transport solutions. Since the calculated values of ΔRT_{NDT} are less than 25°F for each nozzle material, embrittlement effects may be neglected per the following conclusions of Section 4 of TLR-RES/DE/CIB-2013-01 [Ref. 12]:

> *Based on the results of the studies documented in this report, the RES staff recommends the following definition for the RPV* [Reactor Pressure Vessel] *beltline region:*
>
> 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} [equal to ΔRT_{NDT}] *estimated using an ETC* [Embrittlement Trend Correlation] *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.*

Table 8
Surface ART Calculations for the H.B. Robinson Unit 2 Nozzle Forging Materials[(a)]

Reactor Vessel Material and ID Number	Heat Number	CF[(b)] (°F)	Surface Fluence[(c)] (n/cm^2, E > 1.0 MeV)	FF[(c)]	$RT_{NDT(U)}$[(b)] (°F)	ΔRT_{NDT}[(d)] (°F)	σ_I (°F)	σ_Δ[(e)] (°F)	Margin (°F)	ART (°F)
Inlet Nozzle W-10207-1	X15156/X53163	20	1.14×10^{17}	0.1198	10	0 (2.4)	0	0	0.0	10
Inlet Nozzle W-10207-2	X15156/X53163	20	1.14×10^{17}	0.1198	10	0 (2.4)	0	0	0.0	10
Inlet Nozzle W-10207-3	X15156/X53163	20	1.14×10^{17}	0.1198	10	0 (2.4)	0	0	0.0	10
Outlet Nozzle B-3201-1	BT2305	137.9	1.35×10^{17}	0.1339	-7.8	0 (18.5)	0	0	0.0	-7.8
Outlet Nozzle B-3201-2	BT2305	137.9	1.35×10^{17}	0.1339	1.6	0 (18.5)	0	0	0.0	1.6
Outlet Nozzle B-3201-3	BT2305	137.9	1.35×10^{17}	0.1339	7.2	0 (18.5)	0	0	0.0	7.2

Notes for Table 8:

(a) The Regulatory Guide 1.99, Revision 2 [Ref. 9] methodology was utilized in the calculation of the ART values.

(b) Values taken from Table 7.

(c) Surface fluence was taken from Table 1. FF values were calculated using Regulatory Guide 1.99, Revision 2 [Ref. 9].

(d) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. 12]. Actual calculated ΔRT_{NDT} values are listed in parentheses for these materials for information purposes.

(e) Per Regulatory Guide 1.99, Revision 2 [Ref. 9], σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Note that the nozzle corner fracture mechanics calculations have utilized the limiting inlet and outlet nozzle ART values as input to the nozzle P-T limit curves. If the projected fluence at the H.B. Robinson Unit 2 nozzle materials should increase in the future (e.g., due to a power uprate), the embrittlement calculations can be checked to ensure that the ΔRT_{NDT} for these materials remains less than 25°F and can subsequently be set equal to 0°F per TLR-RES/DE/CIB-2013-01 [Ref. 12], thus allowing the conclusions of this letter to remain unchanged. Based on the nozzle material properties documented in Table 7, the inlet nozzle ΔRT_{NDT} values will remain below 25°F until they reach a fluence value of 2.54 x 10^{19} n/cm^2 (E > 1.0 MeV). The outlet nozzle ΔRT_{NDT} values will remain below 25°F until they reach a fluence value of 2.18 x 10^{17} n/cm^2 (E > 1.0 MeV). The reactor vessel inlet and outlet nozzle forging limiting ART values for H.B. Robinson Unit 2 are summarized in Table 9.

Table 9
Summary of Limiting ART Values for the Inlet and Outlet Nozzle Materials for H.B. Robinson Unit 2

EFPY	Nozzle Material and ID Number	Limiting ART Value (°F)
50	Inlet Nozzles W-10207-1, W-10207-2, and W-10207-3	10
	Outlet Nozzle B-3201-3	7.2

Nozzle P-T Limits

A calculation of the H.B. Robinson Unit 2 nozzle cooldown P-T limits was completed using the inlet and outlet nozzle ART values from Table 9. The stress intensity factor correlations used for the nozzle corners are consistent with the ASME PVP2011-57015 [Ref. 13] and Oak Ridge National Laboratory (ORNL) study, ORNL/TM-2010/246 [Ref. 14]. The methodology used included postulating an inside surface 1/4T nozzle corner flaw, along with calculating through-wall nozzle corner stresses for a cooldown rate of 100ºF/hour.

The through-wall stresses at the nozzle corner location were fitted based on a third-order polynomial of the form:

$$\sigma = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

where,

σ = through-wall stress distribution

x = through-wall distance from inside surface

A_0, A_1, A_2, A_3 = coefficients of polynomial fit for the third-order polynomial, used in the stress intensity factor expression discussed below

The stress intensity factors generated for a rounded nozzle corner for the pressure and thermal gradient were calculated based on the methodology provided in ORNL/TM-2010/246. The stress intensity factor expression for a rounded corner is:

$$K_I = \sqrt{\pi a}\left[0.706A_0 + 0.537\left(\frac{2a}{\pi}\right)A_1 + 0.448\left(\frac{a^2}{2}\right)A_2 + 0.393\left(\frac{4a^3}{3\pi}\right)A_3\right]$$

where,

K_I = stress intensity factor for a circular corner crack on a nozzle with a rounded inner radius corner

a = crack depth at the nozzle corner, for use with 1/4T (25% of the wall thickness)

The H.B. Robinson Unit 2 inlet and outlet nozzle P-T limit curves shown in Figures 1 and 2 are based on the stress intensity factor expression discussed above; also shown in these figures are the traditional beltline P-T limits from WCAP-15827, Revision 0 [Ref. 3]. The nozzle P-T limits and the traditional beltline P-T limits are applicable to 50 EFPY. The nozzle P-T limits are provided for a cooldown rate of 100°F/hr, as well as at steady-state.

It should be noted that an outside surface flaw in the nozzle was not considered because the pressure stress is significantly lower at the outside surface than the inside surface. A heatup nozzle P-T limit curve is not provided, since it would be less limiting than the cooldown nozzle P-T limit curve in Figures 1 and 2 for an inside surface flaw.

Based on the results shown in Figures 1 and 2, it is concluded that the nozzle P-T limits are bounded by the traditional beltline curves. Therefore, the 50 EFPY P-T limits contained in WCAP-15827, Revision 0 [Ref. 3] are applicable for the beltline and non-beltline reactor vessel components of H.B. Robinson Unit 2 through 50 EFPY based on the technical evaluations contained herein.

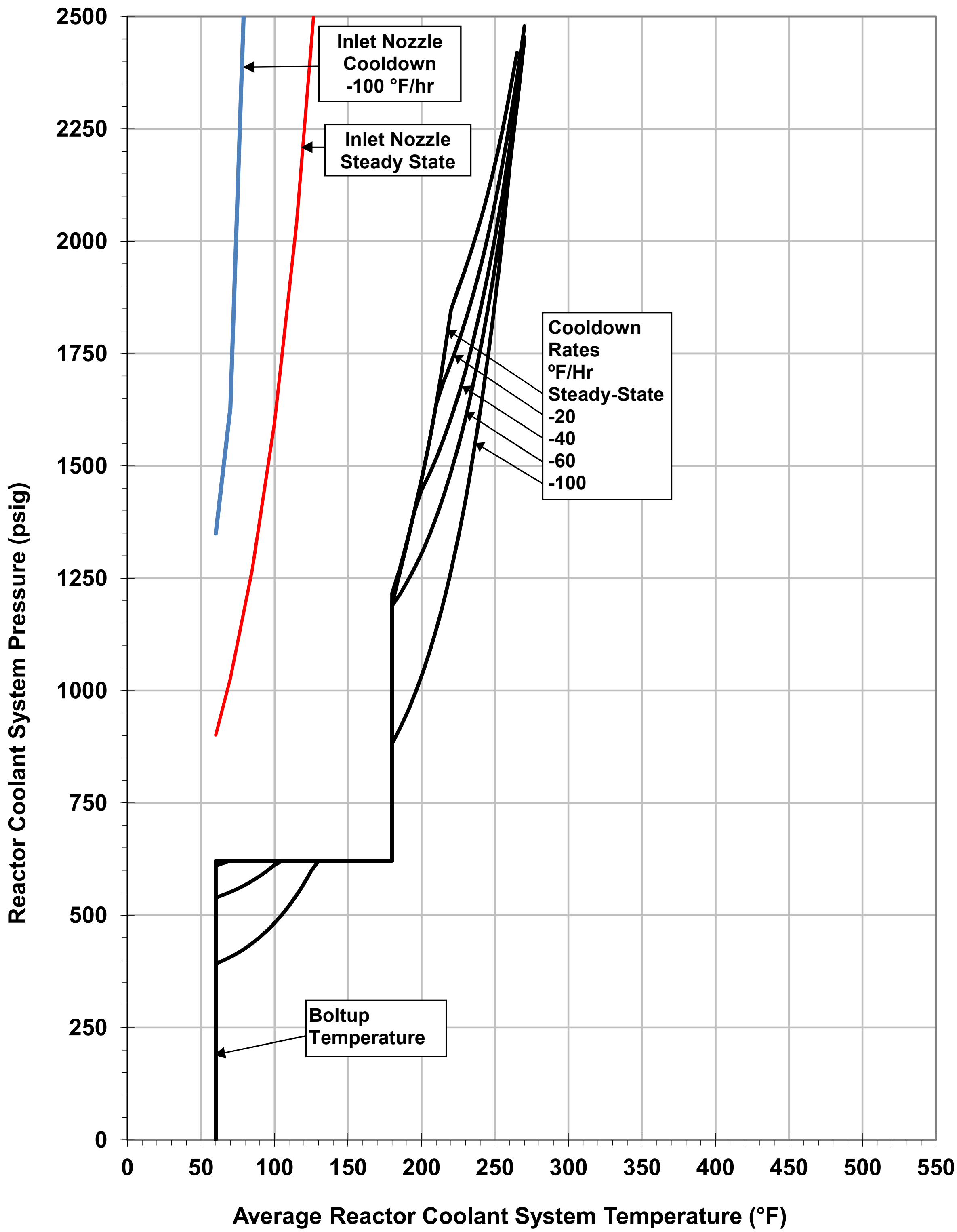

Figure 1: Comparison of H.B. Robinson Unit 2 50 EFPY Inlet Nozzle P-T Limits and Cylindrical Shell P-T Limits

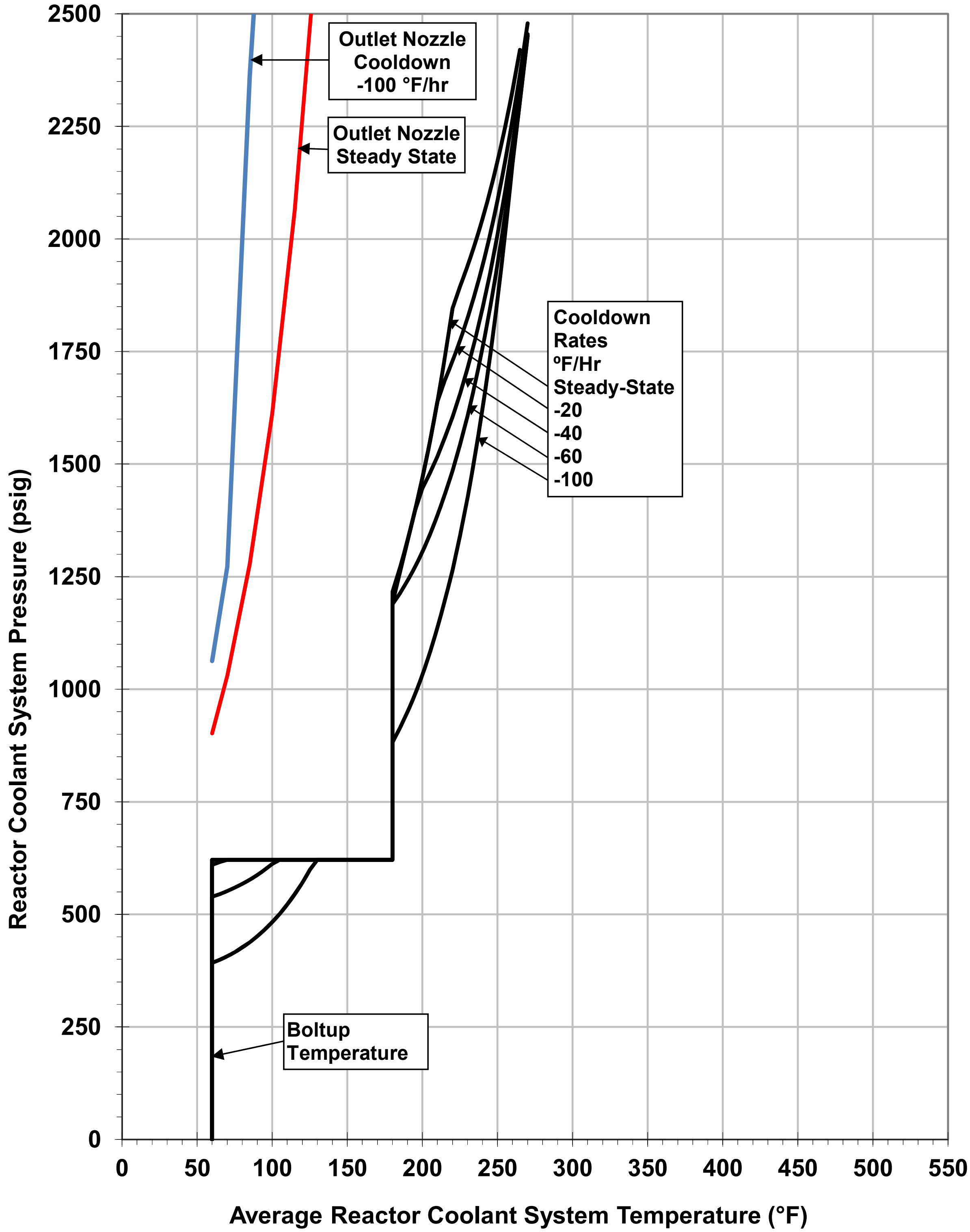

Figure 2: Comparison of H.B. Robinson Unit 2 50 EFPY Outlet Nozzle P-T Limits and Cylindrical Shell P-T Limits

Other Ferritic Components in the Reactor Coolant Pressure Boundary (RCPB)

10 CFR Part 50, Appendix G [Ref. 2], requires that all RCPB components meet the requirements of Section III of the ASME Code. The ferritic RCPB components that are not part of the reactor vessel consist of the pressurizer, replacement reactor vessel head, and replacement steam generators. The H.B. Robinson Unit 2 pressurizer was built in approximately 1967 and therefore was analysed to the ASME Code Section III 1965 Edition. The H.B. Robinson Unit 2 replacement reactor vessel head was analysed to the ASME Section III 1998 Edition with 2000 Addendum. The H.B. Robinson Unit 2 replacement steam generators were analysed to the ASME Code Section III 1965 Edition with Summer 1966 Addendum. These components met all applicable requirements at the time of construction; therefore, no further consideration is necessary for these components with regards to P-T limits.

The lowest service temperature (LST) requirement of NB-2332(b) of ASME Section III is applicable to material for ferritic piping, pumps, and valves with a nominal wall thickness greater than 2 ½ inches [Ref. 15]. Note that the H.B. Robinson Unit 2 reactor coolant system does not have ferritic materials in the Class 1 piping, pumps, or valves. Therefore, the LST requirements of NB-2332(b) are not applicable to the H.B. Robinson Unit 2 P-T limits.

Conclusions

Based on a comparison of the fluence values assigned to the H.B. Robinson Unit 2 materials in the P-T limits report containing the 50 EFPY cylindrical shell curves [Ref. 3] to the fluence values assigned to the same materials in the most recent EVND fluence analysis [Ref. 4] in Table 1, it was concluded that the 50 EFPY cylindrical shell curves documented in WCAP-15827, Revision 0 [Ref. 3] remain applicable for all materials analyzed, except potentially those materials using sister plant data. The only H.B. Robinson Unit 2 material that uses sister plant data is the upper to intermediate shell circumferential weld (Heat # W5214). The Position 2.1 CF was calculated for this material in Table 3, and this Position 2.1 CF was used to recalculate the 1/4T and 3/4T ART values for the upper to intermediate shell circumferential weld in Tables 4 and 5. As determined in Table 6, the limiting ART values used for the development of the 50 EFPY P-T limit curves in WCAP-15827, Revision 0 [Ref. 3] are bounding when compared to the upper to intermediate shell circumferential weld ART values calculated using updated fluence and all applicable capsule results. Thus, the H.B. Robinson Unit 2 cylindrical shell 50 EFPY P-T limit curves documented in WCAP-15827, Revision 0 [Ref. 3] remain applicable through 50 EFPY.

The chemistry, CF, and initial RT_{NDT} values of the H.B. Robinson Unit 2 inlet nozzle and outlet nozzle materials are presented in Table 7. Chemistry information and original test data were taken from the H.B. Robinson Unit 2 CMTRs. Inlet nozzle forging initial RT_{NDT} values were determined using the Branch Technical Position (BTP) 5-3 Position 1.1(4) methodology [Ref. 10], while outlet nozzle forging initial RT_{NDT} values for the reactor vessel outlet nozzle forging materials were determined using the BWRVIP-173-A, Appendix B, Alternative Approach 2 methodology [Ref. 11].

The calculated reactor vessel inlet and outlet nozzle forging ART values for H.B. Robinson Unit 2 at 50 EFPY are presented in Table 8. Since the 50 EFPY nozzle material ΔRT_{NDT} values are below 25°F, embrittlement effects are considered negligible per TLR-RES/DE/CIB-2013-01 [Ref. 12]. The limiting inlet and outlet nozzle ART values for H.B. Robinson Unit 2 at 50 EFPY are summarized in Table 9.

Although neutron embrittlement does not need to be considered for the H.B. Robinson Unit 2 reactor vessel inlet and outlet nozzles through 50 EFPY, the inside corner regions of the reactor vessel nozzles are considered to be highly stressed. P-T limit curves were developed for the reactor vessel nozzles at 50 EFPY using the ART values from Table 9. Figures 1 and 2 display the inlet and outlet nozzle P-T limit curves, and demonstrate that the 50 EFPY P-T limit curves documented in WCAP-15827, Revision 0 [Ref. 3] bound the reactor vessel nozzle P-T limit curves through 50 EFPY for H.B. Robinson Unit 2.

Additionally, other ferritic components in the RCPB were considered. The H.B. Robinson Unit 2 pressurizer, replacement reactor vessel head, and replacement steam generators met all applicable ASME Code Section III requirements at the time of construction; therefore, no further consideration is necessary for these components with regards to P-T limits. Furthermore, the LST requirements of NB-2332(b) are not applicable to the P-T limits since the H.B. Robinson Unit 2 reactor coolant systems do not have ferritic materials in the Class 1 piping, pumps, or valves.

In conclusion, the H.B. Robinson Unit 2 50 EFPY P-T limit curves documented in WCAP-15827, Revision 0 [Ref. 3] remain applicable through 50 EFPY and no further changes are necessary.

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