ATTACHMENT 2 TO NL-13-123

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

REGARDING P-T LIMITS AND LTOP REQUIREMENTS

VESSELS AND INTERNALS INTEGRITY BRANCH (EVIB)

(Non-proprietary version)

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

Response To Request For Additional Information

By letter dated February 6, 2013, (Agencywide Documents Access and Management System Accession No. ML13052A018) Entergy Nuclear Operations, Inc., (Entergy), submitted a license amendment request for Indian Point Nuclear Generating Station, Unit No. 2 (IP2) for the implementation of new reactor vessel heatup and cooldown curves (pressure-temperature limits[P-T Limits]) and low temperature over pressure (LTOP) requirements for staff review and approval.

In order to complete its review the U.S. Nuclear Regulatory Commission staff requires a response to the following questions:

RAI EVIB.1

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G requires that P-T limits be developed for the ferritic materials in the reactor vessel (RV) beltline (neutron fluence ≥ 1×10^{17} n/cm^2, E > 1 MeV). Confirm that all materials projected to receive the above noted fluence or greater at 48 Effective Full-Power Years have been analyzed as beltline materials for the purpose of generating the supplied P-T limits.

Response to RAI EVIB.1

The maximum 48 EFPY fast neutron (E > 1 MeV) exposure of the Indian Point Unit 2 reactor pressure vessel, including the projected effect of a core power uprate to 3216 MWt, was calculated in order to determine the beltline materials (neutron fluence $\geq 1 \times 10^{17}$ n/cm^2, E > 1 MeV). The maximum neutron fluence projected for the Indian Point Unit 2 reactor pressure vessel was calculated using an NRC approved methodology that follows the guidance and meets the requirements of Regulatory Guide 1.190 (Reference 4). The overall analytical methodology is described in References 2, 5, and 6. The NRC approvals are noted in References 2 and 6. Table 1 summarizes the maximum projected neutron fluence values for potential reactor vessel beltline materials at Indian Point Unit 2. It is noted that the maximum neutron fluence values for the intermediate shell, lower shell, intermediate and lower shell longitudinal welds, and intermediate to lower shell circumferential weld are consistent with those reported in WCAP-16752-NP (Reference 1).

From Table 1, it is noted that although the nozzle shell plates and associated nozzle shell longitudinal and nozzle shell to intermediate shell circumferential welds are projected to exceed the 1×10^{17} n/cm^2 threshold, the nozzles themselves as well as the nozzle to nozzle shell welds remain below 1×10^{17} n/cm^2 through 48 EFPY. Likewise, the lower shell to bottom head weld remains out of the beltline region through 48 EFPY. Therefore, no further consideration or analysis is necessary to address RAI EVIB.1 for these materials.

Table 1
Indian Point Unit 2 Calculated Neutron Fluence Projections on Reactor Vessel Extended Beltline Materials at 48 EFPY

Reactor Vessel Material	Fluence (n/cm^2, E > 1.0 MeV)
Inlet Nozzles – Lowest Extent	1.69E+16[(a)]
Outlet Nozzles – Lowest Extent	< 1.48E+16[(a)]
Nozzle Shell Plates	2.26E+17
Nozzle Shell Longitudinal Welds	1.95E+17
Nozzle Shell to Intermediate Shell Circumferential Weld	2.88E+17
Lower Shell to Lower Vessel Head Circumferential Weld	< 1.0E+17

Note for Table 1:

(b) Fluence values at the lowest extent of the nozzles.

As shown above in Table 1, the nozzle shell plates, nozzle shell longitudinal welds, and nozzle shell to intermediate shell circumferential weld receive projected fluence values greater than 1 x 10^{17} n/cm^2 at 48 EFPY. Note that these materials are collectively considered to be the extended beltline region materials; whereas, the traditional beltline materials typically include the intermediate shell, lower shell and associated longitudinal and circumferential welds, which were documented in WCAP-16752-NP (Reference 1). The reactor vessel extended beltline materials are analyzed in this section and compared to the traditional beltline materials already analyzed in Reference 1.

The ART values for the Indian Point Unit 2 extended beltline materials were calculated at 48 EFPY and were determined using the methodology contained in Regulatory Guide 1.99, Revision 2 (Reference 7). Best-estimate nickel (Ni) and copper (Cu) weight-percent (wt%) values along with initial RT_{NDT} values were obtained from Reference 8 for each of the extended beltline materials. The chemistry factor (CF) values were calculated using the Regulatory Guide 1.99, Revision 2 (Reference 7) methodology. The Position 1.1 CF values were calculated using the wt% copper and nickel values along with Tables 1 and 2 of Regulatory Guide 1.99, Revision 2.

Per Reference 8, the nozzle shell longitudinal welds and nozzle shell to intermediate shell circumferential weld were fabricated with weld heat W5214 and Linde 1092 flux type, which is the same weld heat and flux type as the beltline longitudinal weld seams, as documented in WCAP-16752-NP (Reference 1). The Position 2.1 CF values for the weld materials use surveillance capsule data from previously withdrawn capsules containing the W5214 weld material. Since the extended beltline weld materials were fabricated of the same weld heat and flux type as the beltline longitudinal welds, the Position 2.1 CF value for weld heat W5214 was taken from WCAP-16752-NP (Reference 1).

The ART values for the extended beltline materials were calculated using the Regulatory Guide 1.99, Revision 2 methodology and are documented in Tables 2 and 3 for 1/4T and 3/4T locations, respectively.

Table 2

ART Calculations at the 1/4T Location for the Indian Point Unit 2 Reactor Vessel Extended Beltline Materials at 48 EFPY

Reactor Vessel Material	Wt % Cu[a]	Wt % Ni[a]	CF[b] (°F)	1/4T Fluence[c] (n/cm^2, E > 1.0 MeV)	1/4T FF[c]	$RT_{NDT(U)}$[d] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[e] (°F)	Margin (°F)	ART (°F)
Nozzle Shell Plate B-2001-1	0.20	0.50	137.0	1.35E+17	0.1337	24	18.3	0	9.2	18.3	60.6
Nozzle Shell Plate B-2001-2	0.14	0.43	92.4	1.35E+17	0.1337	18	12.4	0	6.2	12.4	42.7
Nozzle Shell Plate B-2001-3	0.19	0.50	131.0	1.35E+17	0.1337	25	17.5	0	8.8	17.5	60.0
Nozzle Shell Longitudinal Welds (Heat W5214)	0.21	1.01	230.2	1.16E+17	0.1213	-56	27.9	17	14.0	44.0	15.9
Using Surveillance Data	0.21	1.01	251.8	1.16E+17	0.1213	-56	30.6	17	14.0	44.0	18.6
Nozzle Shell to Intermediate Shell Circumferential Weld (Heat W5214)	0.21	1.01	230.2	1.72E+17	0.1563	-56	36.0	17	18.0	49.5	29.5
Using Surveillance Data	0.21	1.01	251.8	1.72E+17	0.1563	-56	39.4	17	14.0	44.0	27.4

Notes for Table 2:

(a) Cu and Ni wt% values taken from Reference 8.

(b) Position 1.1 CF values were calculated using the Cu and Ni wt% values and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2. Position 2.1 CF value for weld heat W5214 was taken from WCAP-16752-NP (Reference 1) and is consistent with value reported in Reference 8.

(c) 1/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the Indian Point Unit 2 reactor vessel beltline thickness of 8.625 inches and surface fluences reported in Table 1.

(d) Initial RT_{NDT} values were taken from Reference 8. The initial RT_{NDT} values for the plates are based on measured data; whereas, the initial RT_{NDT} values for the welds are generic.

(e) Per WCAP-16752-NP (Reference 1), the surveillance weld data (heat W5214) was deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

Table 3
ART Calculations at the 3/4T Location for the Indian Point Unit 2 Reactor Vessel Extended Beltline Materials at 48 EFPY

Reactor Vessel Material	Wt % Cu[a]	Wt % Ni[a]	CF[b] (°F)	3/4T Fluence[c] (n/cm^2, E > 1.0 MeV)	3/4T FF[c]	$RT_{NDT(U)}$[d] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[e] (°F)	Margin (°F)	ART (°F)
Nozzle Shell Plate B-2001-1	0.20	0.50	137.0	4.78E+16	0.0649	24	8.9	0	4.4	8.9	41.8
Nozzle Shell Plate B-2001-2	0.14	0.43	92.4	4.78E+16	0.0649	18	6.0	0	3.0	6.0	30.0
Nozzle Shell Plate B-2001-3	0.19	0.50	131.0	4.78E+16	0.0649	25	8.5	0	4.2	8.5	42.0
Nozzle Shell Longitudinal Welds (Heat W5214)	0.21	1.01	230.2	4.13E+16	0.0581	-56	13.4	17	6.7	36.5	-6.1
Using Surveillance Data	0.21	1.01	251.8	4.13E+16	0.0581	-56	14.6	17	7.3	37.0	-4.4
Nozzle Shell to Intermediate Shell Circumferential Weld (Heat W5214)	0.21	1.01	230.2	6.10E+16	0.0775	-56	17.8	17	8.9	38.4	0.2
Using Surveillance Data	0.21	1.01	251.8	6.10E+16	0.0775	-56	19.5	17	9.8	39.2	2.7

Notes for Table 3:

(a) Cu and Ni wt% values taken from Reference 8.

(b) Position 1.1 CF values were calculated using the Cu and Ni wt% values and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2. Position 2.1 CF value for weld heat W5214 was taken from WCAP-16752-NP (Reference 1) and is consistent with value reported in Reference 8.

(c) 3/4T fluence and FF were calculated using Regulatory Guide 1.99, Revision 2, and the Indian Point Unit 2 reactor vessel beltline thickness of 8.625 inches and surface fluences reported in Table 1.

(d) Initial RT_{NDT} values were taken from Reference 8. The initial RT_{NDT} values for the plates are based on measured data; whereas, the initial RT_{NDT} values for the welds are generic.

(e) Per WCAP-16752-NP (Reference 1), the surveillance weld data (heat W5214) was deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1; the weld metal σ_Δ = 28°F for Position 1.1 and, with credible surveillance data, σ_Δ = 14°F for Position 2.1. However, σ_Δ need not exceed $0.5^*\Delta RT_{NDT}$.

Table 4 presents the limiting 1/4T and 3/4T ART values for Indian Point Unit 2 extended beltline materials at 48 EFPY along with the comparison of the limiting Indian Point Unit 2 48 EFPY ART values used in the development of the P-T limit curves (Reference 1) with the limiting 48 EFPY ART values of the extended beltline materials calculated in Tables 2 and 3. The limiting ART values used in the development of the 48 EFPY P-T limit curves (Reference 1) bound the 48 EFPY ART values of the extended beltline materials. Therefore, in response to RAI EVIB.1, the Indian Point Unit 2 48 EFPY P-T limit curves consider all reactor vessel beltline and extended beltline materials that are projected to receive a neutron fluence ≥ 1 x 10^{17} n/cm^2, E > 1 MeV.

Table 4

Comparison of the Indian Point Unit 2 Traditional and Extended Beltline Limiting ART Values at 48 EFPY

	1/4T Location		3/4T Location	
	Existing 48 EFPY Curves documented in WCAP-16752-NP (Traditional Beltline)	**Extended Beltline Evaluation (Table 2)**	**Existing 48 EFPY Curves documented in WCAP-16752-NP (Traditional Beltline)**	**Extended Beltline Evaluation (Table 3)**
Limiting ART (°F)	237	60.6	187	42.0
Limiting Material	Intermediate Shell Plate B-2002-3	Nozzle Shell Plate B-2001-1	Intermediate Shell Plate B-2002-3	Nozzle Shell Plate B-2001-3

RAI EVIB.2

It states in 10 CFR Part 50, Appendix G, Paragraph IV.A that:

> *"the pressure-retaining components of the reactor coolant pressure boundary [RCPB] that are made of ferritic materials must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code [ASME Code, Section III], supplemented by the additional requirements set forth in* [paragraph IV.A.2, "Pressure-Temperature (P-T) Limits and Minimum Temperature Requirements"]..."

Therefore, 10 CFR Part 50, Appendix G requires that P-T limits be developed for the ferritic materials in the reactor vessel (RV) beltline (neutron fluence $\geq 1 \times 10^{17}$ n/cm^2, E > 1 MeV), as well as ferritic materials not in the RV beltline (neutron fluence $< 1 \times 10^{17}$ n/cm^2, E > 1 MeV). Furthermore, 10 CFR Part 50, Appendix G requires that all RCPB components must meet the ASME Code, Section III requirements. The relevant ASME Code, Section III requirement that will affect the P-T limits is the lowest service temperature requirement for all RCPB components specified in Section III, NB-2332(b).

The P-T limit calculations for ferritic RCPB components that are not RV beltline shell materials may define P-T curves that are more limiting than those calculated for the RV beltline shell materials due to the following factors:

3. RV nozzles, penetrations, and other discontinuities have complex geometries that may exhibit significantly higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature (RT_{NDT}) for these components is not as high as that of RV beltline shell materials that have simpler geometries.

4. Ferritic RCPB components that are not part of the RV may have initial RT_{NDT} values, which may define a more restrictive lowest operating temperature in the P -T limits than those for the RV beltline shell materials.

Consequently, please describe how the P-T limit curves submitted for IP2, and the methodology used to develop these curves, considered all RV materials (beltline and non-beltline) and the lowest service temperature of all ferritic RCPB materials, consistent with the requirements of 10 CFR Part 50, Appendix G.

Response to RAI EVIB.2

Reactor Vessel Non-Beltline Components

WCAP-14040-A, Revision 4 does not consider the embrittlement of ferritic materials in the area adjacent to the beltline, specifically the stressed inlet and outlet nozzles. The inside corner regions of these nozzles are the most highly stressed ferritic component outside the beltline region of the reactor vessel; therefore, these components are analyzed in this section.

The ART values for the Indian Point Unit 2 nozzle corner regions were calculated at 48 EFPY for each reactor vessel inlet and outlet nozzle forging. These ART values were determined using the methodology contained in Regulatory Guide 1.99, Revision 2 (Reference 7) along with the inputs described below.

Nozzle Chemistry Data

Best-estimate nickel wt% values were obtained directly from the material-specific analyses documented in the respective Certified Material Test Report (CMTR) for each of the Indian Point Unit 2 reactor vessel inlet and outlet nozzles. The CMTRs did not contain copper wt% values because at the time that the Indian Point Unit 2 nozzles were manufactured, it was not required for SA-508, Class 2 low-alloy steel. Therefore, a best-estimate Cu wt% value of [[ ]]$^{\{E\}}$ from Section 4 of the NRC-approved BWRVIP report, BWRVIP-173-A (Reference 9), was utilized for the Indian Point Unit 2 inlet and outlet nozzles. A mean plus two standard deviations methodology was applied to the data in BWRVIP-173-A to determine conservative Cu wt% value of [[ ]]$^{\{E\}}$. The data in the BWRVIP report were tabulated from an industry-wide database of SA-508, Class 2 forging materials. Therefore, as stated above, the conservative best-estimate Cu wt% from the BWRVIP report of [[ ]]$^{\{E\}}$ was assigned to the Indian Point Unit 2 inlet and outlet nozzles.

The CF values used in this NRC RAI response were calculated using the Regulatory Guide 1.99, Revision 2 (Reference 7) methodology. The CF values were calculated using the wt% copper value of [[ ]]$^{\{E\}}$ from BWRVIP-173-A and the material-specific wt% nickel values along with Table 2 of Regulatory Guide 1.99, Revision 2.

Nozzle Initial RT_{NDT} Values

The initial RTNDT values were determined for each of the Indian Point Unit 2 reactor vessel inlet and outlet nozzle forging materials using the BWRVIP-173-A, Alternative Approach 2 methodology, contained in Appendix B of that report. For six of the eight nozzle materials, CVGraph Version 5.3 was utilized to plot the material-specific Charpy V-Notch impact energy data from the CMTRs to determine the transition temperatures at 35 ft-lb and 50 ft-lb as specified in the Alternative Approach 2 methodology. Two outlet nozzles (B-2012-2 and B-2012-4) had limited test data available only at 10°F, which was taken to be the 35 ft-lb transition temperature since 35 ft-lbs was achieved. For these two nozzles, the lowest Charpy V-notch data point (ft-lbs) was used to obtain the 50 ft-lbs transition temperature by adding 2°F/ft-lb, as described in the methodology. The 35 ft-lb and 50 ft-lb temperatures were then evaluated, per the Alternative Approach 2 methodology presented in BWRVIP-173-A, to determine the initial RTNDT values for the inlet and outlet nozzle materials for Indian Point Unit 2. It should be noted that the orientation of the Charpy V-Notch forging specimens was not clearly identified in the CMTRs; therefore, for conservatism, it was assumed that the forging specimens were oriented in the strong direction. The 50 ft-lb transition temperatures were increased by 30°F to provide conservative estimates for specimens oriented in the weak direction.

Nozzle Neutron Fluence Values

The Indian Point Unit 2 calculated neutron fluence projections at the reactor vessel clad/base metal interface at 48 EFPY for the nozzle materials were documented in the response to RAI EVIB.1. As previously documented, the inlet nozzles are projected to achieve a maximum fluence of 1.69 x 10^{16} n/cm2 (E > 1 MeV) at the lowest extent of the nozzles at 48 EFPY. Similarly, the outlet nozzles are projected to achieve a maximum fluence less than 1.48 x 10^{16} n/cm2 (E > 1 MeV) at the lowest extent of the nozzles at 48 EFPY. Note that the fluence values used in the ART calculations were calculated at the lowest extent of the nozzles (i.e., the nozzle to nozzle shell weld locations) and were chosen at an elevation lower than the actual elevation of the postulated flaw, which is at the inside corner of the nozzle, for conservatism. Additionally, the projected 48 EFPY fluence values at the lowest extent of the nozzles were increased by a factor of 50%. The 50% increase in the 48 EFPY fluence is meant to add conservatism to the calculation so that this evaluation may be demonstrated to be bounding in the future in the event that the nozzle fluence projections should increase (e.g. due to a power uprate).

Nozzle ART Values

The ART values for the nozzle corner regions were calculated using the Regulatory Guide 1.99, Revision 2 methodology and are documented in Table 5. The ART values were conservatively calculated at the clad/base metal interface, rather than at the standard vessel 1/4T location. These ART values were then used for the 1/4T flaw evaluation at the nozzle corner region.

Table 5

ART Calculations for the Indian Point Unit 2 Reactor Vessel Nozzle Materials Using Conservative Fluence Bounding 48 EFPY

Reactor Vessel Material	Wt % Cu[a]	Wt % Ni[a]	CF[b] (°F)	Surface Fluence[c] (n/cm^2, E > 1.0 MeV)	FF[c]	$RT_{NDT(U)}$[d] (°F)	ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[e] (°F)	Margin (°F)	ART (°F)
Inlet Nozzle B-2011-1	[[ ]]{E}	0.75	139.3	2.54E+16	0.0397	-88	5.5	0	2.8	5.5	-76.9
Inlet Nozzle B-2011-2	[[ ]]{E}	0.72	138.2	2.54E+16	0.0397	-78	5.5	0	2.7	5.5	-67.0
Inlet Nozzle B-2011-3	[[ ]]{E}	0.73	138.6	2.54E+16	0.0397	-88	5.5	0	2.8	5.5	-77.0
Inlet Nozzle B-2011-4	[[ ]]{E}	0.70	137.5	2.54E+16	0.0397	-52	5.5	0	2.7	5.5	-41.1
Outlet Nozzle B-2012-1	[[ ]]{E}	0.75	139.3	2.22E+16	0.0357	-11	5.0	0	2.5	5.0	-1.1
Outlet Nozzle B-2012-2	[[ ]]{E}	0.73	138.6	2.22E+16	0.0357	13	4.9	0	2.5	4.9	22.9
Outlet Nozzle B-2012-3	[[ ]]{E}	0.72	138.2	2.22E+16	0.0357	-20	4.9	0	2.5	4.9	-10.1
Outlet Nozzle B-2012-4	[[ ]]{E}	0.72	138.2	2.22E+16	0.0357	10	4.9	0	2.5	4.9	19.9

Notes for Table 5:

(a) Cu wt% values are the best-estimate values for SA-508, Class 2 low-alloy steel as documented in BWRVIP-173-A. The Ni wt% values are material-specific values as documented in each respective material CMTR.

(b) CF values were calculated using the Cu and Ni wt% values and Table 2 of Regulatory Guide 1.99, Revision 2.

(c) The 48 EFPY fluence values increased by a factor of 50%. These are considered conservative fluence values that bound the nozzle fluence values at the clad/base metal inner surface at 48 EFPY. See Table 1 for fluence values at the lowest extent of the nozzles for the inlet and outlet forgings at 48 EFPY. FF values were calculated using Regulatory Guide 1.99, Revision 2.

(d) $RT_{NDT(U)}$ values were determined using the Alternative Approach 2 methodology as described in Appendix B of BWRVIP-173-A.

(e) Per Regulatory Guide 1.99, Revision 2, the base metal nozzle forging materials σ_Δ = 17°F for Position 1.1 without surveillance data. However, σ_Δ need not exceed 0.5 * ΔRT_{NDT}.

A summary of the limiting inlet and outlet nozzle ART values at Indian Point Unit 2 is presented in Table 6.

Table 6
Summary of the Limiting ART Values for the Inlet and Outlet Nozzle Materials at Indian Point Unit 2

EFPY	Nozzle Material and ID Number	Limiting ART Value (°F)
> 48 (Conservative Projection)	Inlet Nozzle B-2011-4	-41.1
	Outlet Nozzle B-2012-2	22.9

Nozzle P-T Limits

A calculation of the Indian Point Unit 2 nozzle cooldown P-T limits was completed using the inlet and outlet nozzle ART values at 48 EFPY to account for nozzle embrittlement. The stress intensity factor correlations used for the nozzle corners are consistent with the ASME PVP2011-57015 (Reference 10) and ORNL study, ORNL/TM-2010/246 (Reference 11). 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 100°F/hour. The stresses used for the Indian Point Unit 2 nozzle corners are consistent with those used in the 4-loop Pressurized Water Reactor (PWR) nozzle analyses in ORNL/TM-2010/246 (Reference 11), which was based on a three-dimensional (3-D) finite element model (FEM).

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 flaw on 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 flaw on 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 Indian Point Unit 2 inlet and outlet nozzle P-T limit curves are shown in Figures 1 and 2, respectively, based on the stress intensity factor expression discussed above; also shown in these figures are the traditional beltline P-T limits from WCAP-16752-NP. The nozzle P-T limits are provided for a cooldown rate of -100°F/hr, along with a steady-state curve.

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 P-T limits provided in WCAP-16752-NP for 48 EFPY are still applicable for the beltline and non-beltline reactor vessel components.

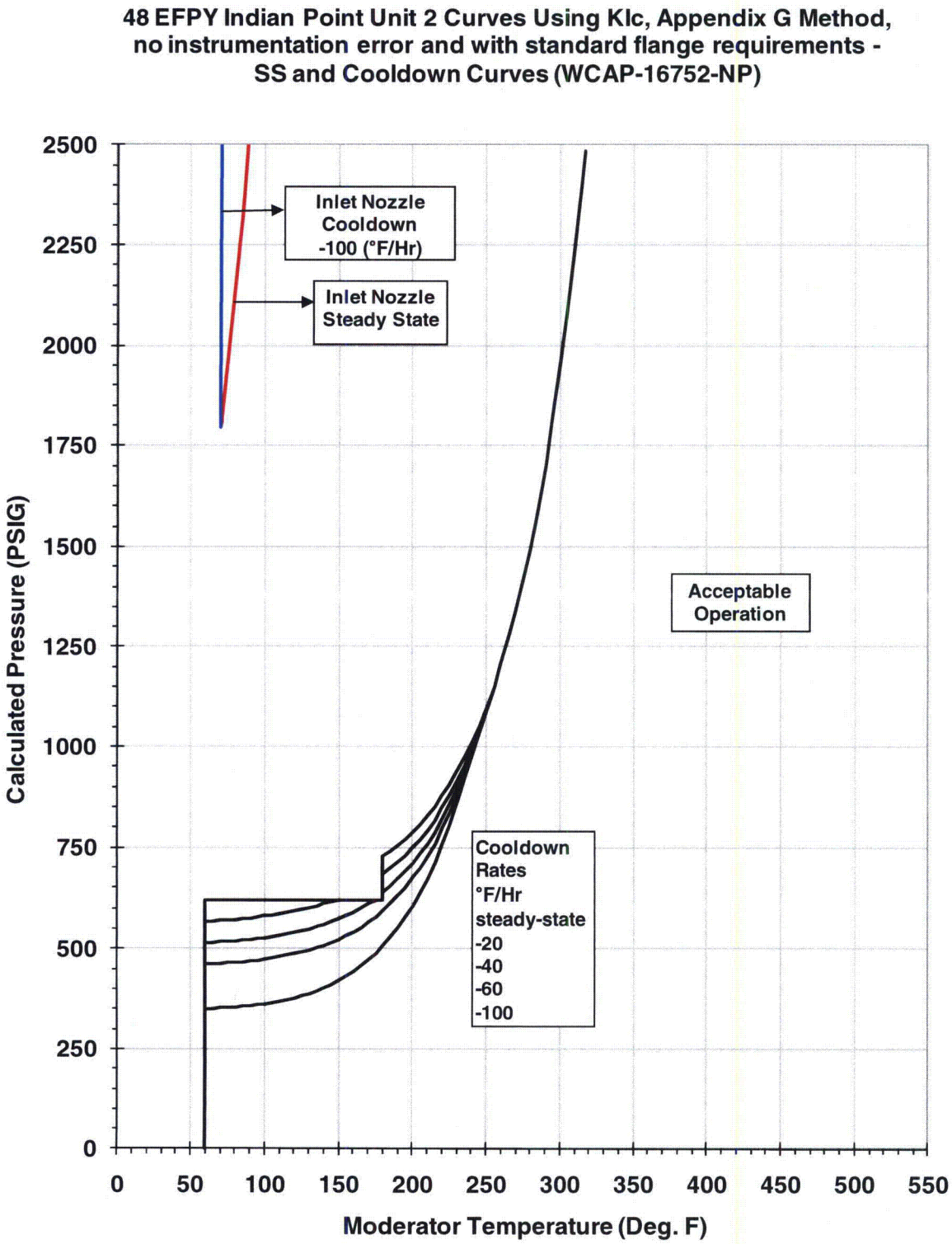

Figure 1: Comparison of IP2 WCAP-16752-NP P-T Limits to Inlet Nozzle Limits

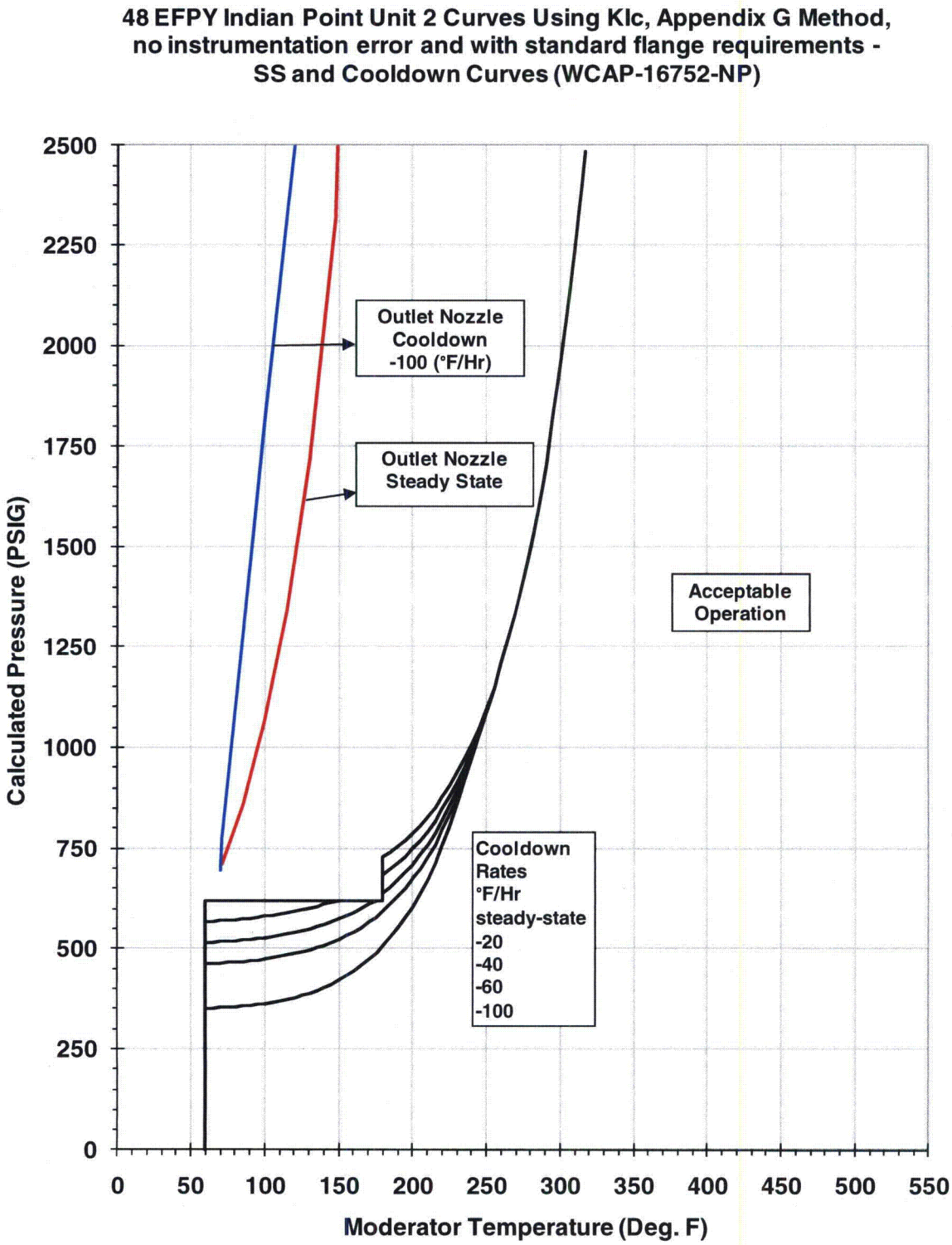

Figure 2: Comparison of IP2 WCAP-16752-NP P-T Limits to Outlet Nozzle Limits

Other Ferritic Components in the Reactor Coolant Pressure Boundary

The ferritic RCPB components that are not part of the RV consist of the Model 44F Replacement Steam Generator (RSG) and the original Model D Series 84 Pressurizer. It should be noted that the RSG and pressurizer have been analyzed for the non-ductile facture mechanics analysis as per ASME Section XI Appendix G. The Appendix G evaluation was performed for the Stretch Power Uprate (SPU) program at Indian Point Unit 2 in 2004 for the pressure retaining components of the reactor coolant pressure boundary as per Sections 3.6.2.2 and 3.6.2.3 in Reference 12.

The lowest service temperature (LST) requirement of NB-2332(b) is applicable to material for ferritic piping, pumps and valves with a nominal wall thickness greater than 2 ½ inches (Reference 13). Note that the Indian Point 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 Indian Point Unit 2 P-T limits.

The discussion in the subsequent pages will further demonstrate that the P-T limits for the ferritic components in the RCPB (i.e. RSG and pressurizer) are less limiting than the P-T limit curves developed for the Reactor Vessel beltline in WCAP-16752-NP (Reference 1).

Replacement Steam Generator

In order to demonstrate that the 48 EFPY P-T limit curves in WCAP-16752-NP for the Indian Point Unit 2 RV beltline region bounds the RSG ferritic components, two locations in the RSG were reviewed based on the ASME Section XI, Appendix G fracture mechanics analysis. The first location is the steam generator (SG) channel head to tube sheet region lower junction, and the second location is the primary nozzle knuckle region (nozzle corner). These two locations are highly stressed regions in the SG and contain discontinuities that should be considered for a non-ductile failure evaluation.

SG Tube Sheet to Channel Head Junction

To determine if the SG tube sheet to channel head junction location is more limiting for P-T limits than the RV beltline, allowable pressures based on ASME Section XI, Appendix G are calculated at several temperature values and compared with that of the RV beltline region for Indian Point Unit 2.

For the fracture mechanics analysis at the RSG tube sheet to the channel head junction, an inside surface axial flaw with an aspect ratio of 6:1 is considered, per ASME Section XI, Appendix G. The cooldown transient (ramp down of 100°F/hr) is also considered, as it will produce high tensile stresses on the inside surface for pressure and thermal transients. The limiting tensile stress components are chosen to determine the primary and secondary stress intensity factors, with the appropriate safety factors, and membrane and bending geometric factors consistent with the ASME Section XI, Appendix G evaluation.

The initial RT_{NDT} value is based on the design specification for the Indian Point Unit 2 RSG ferritic materials (base metals and welds); this value is conservatively set to 40°F for the tube sheet junction location. The initial RT_{NDT} from the actual CMTR (certified material test reports) is 10°F.

The P-T limit values for the SG tube sheet junction are plotted on Figure 3 and compared with the Indian Point Unit 2 RV beltline P-T limits for the cooldown transient. The data from Figure 3 demonstrates that the RV beltline P-T limit curves, from WCAP-16752-NP for Indian Point Unit 2, bound the tube sheet to channel head junction P-T limit curves.

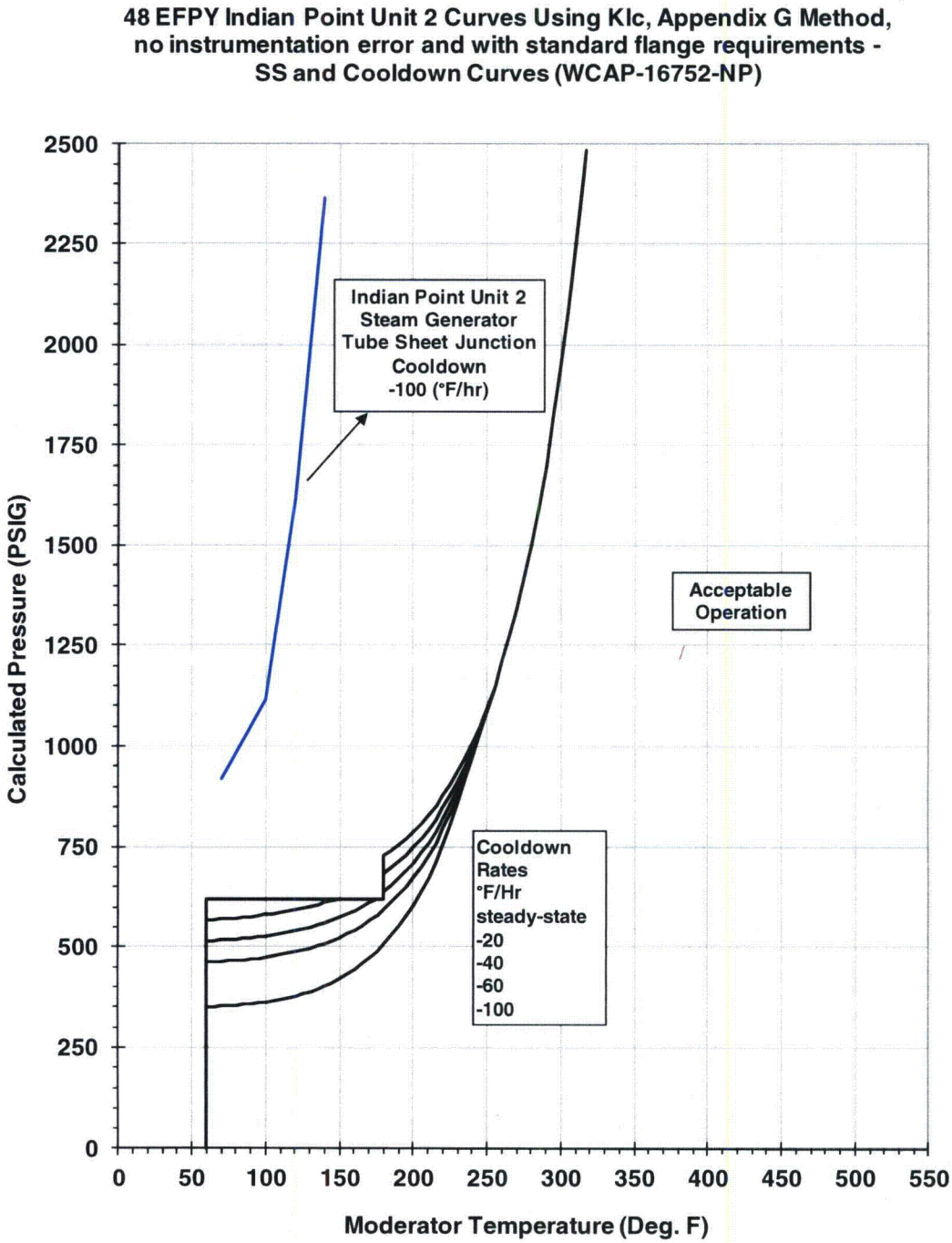

Figure 3: P-T Limit Curve Comparison of Tradition Beltline and the RSG Tube Sheet to Channel Head Junction for Indian Point Unit 2

Replacement Steam Generator Primary Nozzle Knuckle Region

The Indian Point Unit 2 RSG primary nozzle knuckle region is also considered to be another potential limiting location for a non-ductile failure evaluation for an ASME Section XI, Appendix G evaluation due to the discontinuity at the nozzle corner region. Therefore, the RSG nozzle corner region was reviewed to determine its impact on the P-T limit curves as compared to the reactor pressure vessel P-T limit curves.

In order to demonstrate that the P-T limits for the Indian Point RSG primary nozzle knuckle regions are less limiting than the reactor vessel beltline, a comparison of the stresses and the component material properties was performed. In the response to the RAI concerning the RV outlet nozzle corner region in the earlier section, it was demonstrated that the P-T limits for the RV outlet nozzle corner regions are less limiting than the reactor vessel beltline P-T limits. For the RSG nozzle corner herein, it will be demonstrated that the RSG nozzle corner or knuckle region P-T limits are less limiting than the reactor vessel outlet nozzle corner for a 1/4T inside surface corner flaw using the stresses due to the cooldown transient.

Fracture Toughness Material Property

Based on the plant-specific CMTR for the RSG primary nozzles, the RT_{NDT} is +10°F. This maximum RT_{NDT} is considered for evaluation as discussed below. Since the RSG nozzles experience negligible neutron irradiation, the maximum RT_{NDT} of 10°F for this material can be taken as the Adjusted Reference Temperature (ART). For the RV outlet nozzle corner P-T limit development, the limiting ART value for the RV outlet nozzle with the irradiation shift is 22.9°F, as discussed in Table 6 of this report. As a result, for the entire cooldown transient as well as at the lowest temperature of 70°F, the fracture toughness K_{Ic} for the reactor vessel outlet nozzle corner regions are less (more limiting) than the RSG nozzles, since the RT_{NDT} for the reactor vessel outlet nozzles are higher than the RSG nozzles.

Primary and Secondary Stresses

The Model 44F RSG has a larger inside knuckle radius than the reactor vessel outlet nozzle corners; therefore, the peak stresses for the Model 44F primary nozzle inside corner are less than the reactor vessel outlet nozzle corner.

The hoop pressure stresses due to a unit pressure stress of 1000 psi were compared to the RSG nozzle and the Indian Point Unit 2 reactor vessel outlet nozzle corners. Based on this comparison, the pressure stresses at the RSG nozzle corner region were less than the RV outlet nozzle corner regions from the inside surface up to 80% of the wall thickness. Therefore, it was concluded that the K_{IP} (stress intensity factor due to pressure) at the RSG primary nozzle knuckle region is less than that of the RV outlet nozzles.

The Indian Point Unit 2 RSG primary nozzle wall thickness at the knuckle region is approximately 10.2 inches while the RV outlet nozzle corner through-wall thickness is approximately 18.3 inches. Therefore, the cooldown thermal transient hoop stresses for the RSG primary nozzles are also less than the RV outlet nozzles since the wall thickness of the RSG nozzles are smaller than the RV outlet nozzles at the location of interest. Therefore it is concluded that the K_{It} values for a

postulated 1/4T flaw at the 10.2 inches RSG primary nozzles are also less than the RV outlet nozzle corner regions.

Pressure-Temperature Limit Discussion

Since both the primary and secondary stress intensity factors for the RSG nozzles are less than the RV outlet nozzle corner regions, the total applied K_I for the RSG nozzles is also less than the RV outlet nozzles. Furthermore, the K_{IC} allowable value for the RSG nozzle is greater than the RV outlet nozzles since the RT_{NDT} values for the RSG nozzles are less than the RV outlet nozzle. Therefore, it is concluded that the allowable pressures for the P-T limits for the RSG nozzle corner region are higher (less limiting) than the RV outlet nozzle corners. As a result, the RSG nozzle corners have P-T limit values that are less limiting than the RV beltline region for Indian Point Unit 2, since based on the earlier section of this letter report, the P-T limit curve for the RV outlet nozzle corners is bounded by the reactor vessel beltline region.

In addition to the discussion presented above for the RSG, a comprehensive ASME Section XI, Appendix G analysis has been performed for all the ferritic pressure-retaining components of the RCPB, including the RSG, as part of the SPU program in 2004 at Indian Point Unit 2. Indian Point Unit 2 received a safety evaluation report for the stretch power uprate from the NRC to amend their Technical Specification as referenced in "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment RE: 3.26 Percent Power Uprate (TAC No. MC1865)," NRC Adams Accession No: ML042960007 (Reference 12). The safety evaluation report concluded in Section 3.6.2.3 that,

> "...the licensee has adequately addressed fracture integrity evaluations for ferritic pressure-retaining components of the reactor coolant pressure boundary. Based on this assessment, the NRC staff concludes that the IP2 facility will continue to meet the requirements of 10 CFR Part 50 Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with GDC-14 and GDC-31 in this respect following the implementation of the proposed SPU."

Therefore, the ASME Section XI Appendix G evaluation performed for all the pressure-retaining components in the RCPB, including the RSG, as part of the SPU program can also be used to supplement the discussion provided in this report.

Pressurizer

For Westinghouse plants, the technical specification limits for the pressurizer heatup and cooldown limits are set to 100°F/hr and 200°F/hr, respectively. However, the pressurizer may experience sudden insurge or outsurge events during these heatup and cooldown conditions. These insurge and outsurge events can produce significant temperature transients in the pressurizer lower head resulting from the temperature difference between the hot leg and the pressurizer. Consequently, this results in the lower head ferritic regions of pressurizer to be limiting from a P-T limits point of view. The surge nozzle inside corner region experiences high thermal and pressure stresses along with large fatigue usage factors during the insurge and outsurge conditions. Therefore, the surge nozzle ferritic component of the pressurizer is used to determine if the surge nozzle is more limiting for the pressure temperature limits than the RV beltline during the heatup and cooldown conditions.

For the pressurizer surge nozzle, the pressure-temperature limits are determined based on the Appendix G evaluation criteria with the use of the nozzle corner stress intensity factor expression provided in the Oak Ridge National Laboratory study ORNL/TM-2010/246 (Reference 11). Various plant-specific heatup and cooldown transients were considered with different system ΔT, temperature difference, between the hot leg and the pressurizer, i.e. ΔT = 250°F and ΔT = 320°F.

Based on the plant-specific thermal and pressure stresses for the insurge and outsurge conditions, allowable pressure values are calculated for a postulated 1/4T inside surface corner flaw at the surge nozzle. An outside surface flaw is not considered since the pressure stresses are lower at the outside surface than at the inside surface of the nozzle. Furthermore, the thermal gradient stresses at the 1/4T flaw location for the heat-up condition are compressive as compared to the cooldown transient, which are tensile at the inside surface. Therefore, the limiting condition considered in the evaluation for the surge nozzle corner is the cooldown transient.

The fracture toughness (K_{IC}) is conservatively calculated based on the cooldown fluid temperature as opposed to the crack tip temperature at the 1/4T location through the wall thickness. The pressurizer components undergo a minimal amount of neutron embrittlement; therefore, it is acceptable to use the material initial RT_{NDT} for the calculation of the fracture toughness. Based on the Indian Point Unit 2 CMTR (certified material test report), an initial RT_{NDT} value of +40°F is used to determine K_{IC}.

Based on the methodology provided in ASME Section XI Appendix G, the allowable pressure for the surge nozzle corner region at a temperature of 70°F was determined to be approximately 2250 psig during the cooldown transient. The allowable pressure for the RV beltline based on Figure 1 is approximately 620 psig at a temperature of 70°F for the cooldown transient from WCAP-16752-NP (Reference 1). Therefore, the allowable pressure at the surge nozzle of the pressurizer demonstrates sufficient margin beyond the RV beltline P-T limits. As a result, it can be concluded that the P-T limits for the ferritic components within the pressurizer are less limiting than that for the reactor vessel beltline region as provided in WCAP-16752-NP.

In addition to the discussion presented above for the pressurizer, a comprehensive ASME Section XI, Appendix G analysis has been performed for all the ferritic pressure-retaining components of the RCPB, including the pressurizer, as part of the SPU program in 2004 at Indian Point Unit 2. Indian Point Unit 2 received a safety evaluation report for the stretch power uprate from the NRC to

amend their Technical Specification as referenced in "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment RE: 3.26 Percent Power Uprate (TAC No. MC1865)," NRC Adams Accession No: ML042960007 (Reference 12). The safety evaluation report concluded in Section 3.6.2.3 that,

> "...the licensee has adequately addressed fracture integrity evaluations for ferritic pressure-retaining components of the reactor coolant pressure boundary. Based on this assessment, the NRC staff concludes that the IP2 facility will continue to meet the requirements of 10 CFR Part 50 Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with GDC-14 and GDC-31 in this respect following the implementation of the proposed SPU."

Therefore, the ASME Section XI Appendix G evaluation performed for all the pressure-retaining components in the RCPB, including the pressurizer, as part of the SPU program can also be used to supplement the discussion provided in this report.

References

1. Westinghouse Report WCAP-16752-NP, Revision 1, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," March 2009. *[Revision 1 was issued to correct an editorial error. There was no impact to the technical evaluation contained in Revision 0.]*
2. 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.
3. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, December 19, 1995.
4. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.
5. Westinghouse Report WCAP-15557, Revision 0, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology," August 2000.
6. Westinghouse Report WCAP-16083-NP-A, Revision 0, "Benchmark Testing of the FERRET Code for Least Squares Evaluations of Light Water Reactor Dosimetry," May 2006.
7. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, May 1988.
8. Entergy Letter NL-08-143, "Additional Information Regarding License Renewal Application – Reactor Vessel Fluence Clarification," September 24, 2008. *(NRC ADAMS Accession Number ML082760402)*
9. *BWRVIP-173-A: BWR Vessel and Internals Project: Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials.* EPRI, Palo Alto, CA: 2011. 1022835.
10. ASME PVP2011-57015, "Additional Improvements to Appendix G of ASME Section XI Code for Nozzles," G. Stevens, H. Mehta, T. Griesbach, D. Sommerville, July 2011.
11. Oak Ridge National Laboratory Report, ORNL/TM-2010/246, "Stress and Fracture Mechanics Analyses of Boiling Water Reactor and Pressurized Water Reactor Pressure Vessel Nozzles – Revision 1," June 2012.
12. "Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment RE: 3.26 Percent Power Uprate (TAC no. MC1865)." *(NRC ADAMS Accession Number: ML042960007)*
13. ASME B&PV Code Section III, Division I, NB-2332, "Material for Piping Pumps, and Valves, Excluding Bolting Material," 2004 Edition.