Progress Energy

AUG 1 6 2007

SERIAL: HNP-07-104
10 CFR 54

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

Subject: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400 / LICENSE NO. NPF-63

RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION - LICENSE RENEWAL APPLICATION SECTION 4.2 AND SUBSECTION B.2.17

References:
1. Letter from Cornelius J. Gannon to the U. S. Nuclear Regulatory Commission (Serial: HNP-06-136), "Application for Renewal of Operating License," dated November 14, 2006
2. Letter from Maurice Heath (NRC) to Robert J. Duncan II, "Requests for Additional Information for the Review of the Shearon Harris Nuclear Power Plant, Unit 1, License Renewal Application," dated July 20, 2007

Ladies and Gentlemen:

On November 14, 2006, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, requested the renewal of the operating license for the Shearon Harris Nuclear Power Plant, Unit No. 1, also known as the Harris Nuclear Plant (HNP), to extend the term of its operating license an additional 20 years beyond the current expiration date.

By letter dated July 20, 2007, the Nuclear Regulatory Commission (NRC) provided requests for additional information (RAIs) concerning the HNP License Renewal Application (LRA). The enclosure to this letter provides responses to the RAIs. The response to each of the RAIs indicates that a change to the LRA is required, and the response to RAI-B.2.17 involves a modification to existing License Renewal Commitment #13 described in Enclosure 1 of Reference 1. A transmittal to document these changes will be provided at a later date. This document contains no new Regulatory Commitments. Any actions discussed in this letter should be considered intended or planned actions that are included for information.

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P. O. Box 165
New Hill, NC 27562

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NRR

Please refer any questions regarding this submittal to Mr. Roger Stewart, Supervisor - License Renewal, at (843) 857-5375.

I declare, under penalty of perjury, that the foregoing is true and correct (Executed on AUG 16 2007).

Sincerely,

T. J. Natale
Manager - Support Services
Harris Nuclear Plant

TJN/mhf

Enclosure: Responses to Requests for Additional Information dated July 20, 2007

cc:

Mr. P. B. O'Bryan (NRC Senior Resident Inspector, HNP)
Ms. B. O. Hall (Section Chief, N.C. DENR)
Mr. M. L. Heath (NRC License Renewal Project Manager, HNP)
Dr. W. D. Travers (NRC Regional Administrator, Region II)
Ms. M. G. Vaaler (NRC Project Manager, HNP)

Responses to Requests for Additional Information dated July 20, 2007

Background

On November 14, 2006, Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., requested the renewal of the operating license for the Shearon Harris Nuclear Power Plant, Unit No. 1, also known as the Harris Nuclear Plant (HNP), to extend the term of its operating license an additional 20 years beyond the current expiration date.

By letter dated July 20, 2007, the Nuclear Regulatory Commission (NRC) provided a request for additional information (RAI) concerning the HNP License Renewal Application. This enclosure provides the responses to the NRC RAI. Note that NRC RAI numbers 4.2.1 and 4.2.2 are not used.

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NRC RAI-B.2.17

(A) The applicant states that the surveillance capsule that is to be withdrawn during the 16th refueling outage would have been exposed to a neutron fluence value that is equivalent to the peak reactor pressure vessel (RPV) fluence at 55 effective full power year (EFPY). Please confirm this statement. The staff requests that the applicant provide the following information related to this test:

(1) Lead factor of the surveillance capsule
(2) Identification number of the capsule, and
(3) Heat number of the surveillance material in the capsule

(B) Program element 6, item 2 of aging management program (AMP) B.2.17 states that the applicant intends to test one surveillance capsule after the 16th refueling outage. The staff requests that the applicant submit the following information that pertains to the test:

(1) The projected refueling outage of withdrawal
(2) Projected capsule neutron fluence value at the time of withdrawal
(3) Corresponding EFPY for the peak RPV fluence to equal the capsule fluence
(4) The identification number of the capsule, and
(5) Heat number of the surveillance material in the capsule

(C) The staff requests that the applicant confirm that the withdrawal schedule of the final two capsules for the extended period of operation is consistent with the requirements, specifically the limitations on lead factor, specified in paragraph 7.6.2 of the American Society of Testing Materials E 185 (ASTM E 185), "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

(D) Section 5.3.1.6 of the final safety analysis report states that the applicant intends to use two standby capsules with identifications Y and Z for future tests. However, the "operating experience" Section of AMP B.2.17 indicates that three capsules will remain in the RPV for future tests to manage neutron embrittlement during the extended period of operation. The staff requests that the applicant provide an explanation for this inconsistency.

(E) The staff requests that the applicant include the following statements in the commitment table of the license renewal application (LRA):

> (1) The applicant will notify the staff if there is any change in the withdrawal schedules of the surveillance capsules.
> (2) If a standby capsule is removed from the RPV without the intent to test it, the capsule will be stored in manner which maintains it in a condition which would permit its future use, including during the period of extended operation, if necessary.

RAI-B.2.17 Response

(A) Confirmation of statement in (A): The withdrawal of the next capsule will occur during Refueling Outage (RFO)-16, at which time the capsule fluence is projected to be equivalent to the 60-year (i.e., 55 EFPY) maximum vessel fluence of 6.8×10^{19} n/cm^2 in accordance with ASTM E 185-82.

- A(1) The lead factor of the surveillance capsule to be withdrawn during RFO-16 is 2.68.
- A(2) The identification of this capsule is Capsule W.
- A(3) The heat number of the surveillance base metal in Capsule W is B4197-2. The surveillance weld wire in Capsule W is 5P6771, Linde 124 flux, lot number 0342.

(B) Program Element 6, Item 2 of AMP B.2.17 states:

> ...HNP will evaluate neutron exposure for the remaining capsules, based on the analysis of the capsule withdrawn during RFO-16. The neutron exposure and withdrawal schedule for the capsules remaining after RFO-16 will be optimized to provide meaningful metallurgical data.

In addition to the response to RAI-B.2.17 questions B(1) through B(5), plans for the capsules remaining inside the reactor vessel are discussed in the response to RAI-B.2.17(C).

- B(1) The projected refueling outage for the withdrawal after RFO-16 (i.e., after the withdrawal of capsule W) will be determined after the analysis of Capsule W.

B(2) The projected neutron fluence for the next capsule to be withdrawn after RFO-16 will not exceed twice the 60-year maximum vessel fluence of $6.8x10^{19}$ n/cm^2 in accordance with ASTM E 185-82.

B(3) The corresponding EFPY for the peak reactor vessel fluence equal to the capsule fluence for the capsule to be withdrawn after RFO-16 will be determined after the analysis of Capsule W.

B(4) The identification number of the capsule to be withdrawn after RFO-16 is either Capsule Y or Z.

B(5) The heat number of the surveillance base metal in Capsules Y and Z is B4197-2. The surveillance weld wire in Capsules Y and Z is 5P6771, Linde 124 flux, lot number 0342.

(C) HNP is currently in Cycle 14. At the current time, Capsules W, Y, and Z remain inside the reactor vessel. Capsules W, Y, and Z are projected to exceed the 60-year (i.e., 55 EFPY) maximum vessel fluence prior to the end of 40 years. All remaining capsules currently have a lead factor of 2.68, which is used to determine the withdrawal schedule of Capsules W, Y, and Z in accordance with ASTM E 185-82. Capsule W is scheduled to be withdrawn during RFO-16, at which time the capsule fluence is projected to be equivalent to the 55 EFPY maximum vessel fluence of $6.8x10^{19}$ n/cm^2. Capsules Y and Z will remain in the reactor vessel after Capsule W is withdrawn.

Based on the above, the following changes to Section B.2.17 of the HNP LRA are required. These changes require a modification to HNP LR Commitment #13 from Enclosure 1 of the letter from Cornelius J. Gannon to the U. S. Nuclear Regulatory Commission (Serial: HNP-06-136), "Application for Renewal of Operating License," dated November 14, 2006.

Enhancement 1 to Element 6 in the HNP LRA will be modified as follows: Withdrawal of the next capsule (i.e., Capsule W) will occur during RFO-16, at which time the capsule fluence is projected to be equivalent to the 60-year maximum vessel fluence of $6.8x10^{19}$ n/cm^2 in accordance with ASTM E 185-82.

Enhancement 2 to Element 6 in the HNP LRA will be modified as follows: The analysis of Capsule W, to be withdrawn during RFO-16, will be used to evaluate neutron exposure for remaining Capsules Y and Z, as required by 10 CFR 50 Appendix H. The withdrawal schedule for one of the remaining capsules (i.e., Capsule Y or Z) will be adjusted, based on the analysis of Capsule W, so that the capsule fluence will not exceed twice the 60-year maximum vessel fluence in accordance with ASTM E 185-82. The neutron exposure and withdrawal schedule for the last capsule will be optimized to provide meaningful metallurgical data. If the last capsule is projected to significantly exceed a meaningful fluence value, it will either be relocated to a lower flux position or withdrawn for possible testing or re-insertion. The remaining Capsules Y and Z (and archived test specimens available for reconstitution) will be available for the monitoring of neutron exposure if additional license renewals are sought (i.e., 80 years of operation).

(D) HNP FSAR Section 5.3.1.6 states that Capsules U, V, and X have been withdrawn from the reactor vessel and tested, and that Capsules W, Y, and Z remain inside the vessel. FSAR Section 5.3.1.6 also states that Capsule W is scheduled for removal from the vessel, and that Capsules Y and Z are standby capsules. Therefore, FSAR Section 5.3.1.6 is stating that three capsules (i.e., Capsules W, Y, and Z) are currently in the reactor vessel. Section B.2.17 of the HNP LRA states that three capsules remain inside the vessel, exposed to additional neutron flux, providing a source for future data that will be used to manage neutron embrittlement aging effects for the period of extended operation. Therefore the statements in FSAR Section 5.3.1.6 and HNP LRA Section B.2.17 are consistent.

(E) Request for further commitments:

(E)(1) The HNP procedure entitled "Technical Specification Equipment List Program and Core Operating Limits Report," Attachment 3, states:

> Changes to the reactor materials surveillance schedule must receive NRC approval prior to implementation. (Reference: Section III.B.3 of 10 CFR 50, Appendix H).

Therefore, an additional commitment in the HNP LRA is not needed.

(E)(2) Both Commitment #13, Item 1, from Enclosure 1 of the letter from Cornelius J. Gannon to the U. S. Nuclear Regulatory Commission (Serial: HNP-06-136), "Application for Renewal of Operating License," dated November 14, 2006, and HNP LRA Section B.2.17, Enhancements, indicate that the Reactor Vessel Surveillance Program will be enhanced such that tested and untested specimens from all capsules pulled from the reactor vessel must be kept in storage to permit future reconstitution use and that the identity, traceability, and recovery of the capsule specimens shall be maintained throughout testing and storage. Therefore, an additional commitment in the HNP LRA is not needed.

An amendment to the LRA is required in response to RAI B.2.17 (C).

NRC RAI-4.2.3 (Editorial Correction)

In Tables 4.2-2 and 4.2-3 of the LRA, the chemical composition values of elements Copper and Nickel for the surveillance capsule test sample representing the intermediate shell plate (heat number-B4197-2) and the RPV's intermediate shell plate (heat number-B4197-2) are identical. However, the chemistry factors are different. The staff requests that the applicant add a footnote stating that the chemistry factor for the surveillance capsule test sample representing the intermediate shell plate is derived from the surveillance test data.

RAI-4.2.3 Response

A footnote will be added to Tables 4.2-2 and 4.2-3 in the LRA, stating that the chemistry factors for the surveillance test capsule representing the Intermediate Shell Plate (heat number B4197-2) and the Intermediate Shell-to-Lower Shell Circumferential Weld (100%) (heat number 5P6771) are derived from the surveillance data.

An amendment to the LRA is required in response to RAI-4.2.3.

NRC RAI-4.2.4

The staff requests that the applicant include the following items in Section 4.2.4 of the LRA:

> (A) The current pressure-temperature (P-T) limits are valid through 36 EFPY. The P-T limits for the extended period of operation will be managed by using approved fluence calculations when there are changes in power of core design in conjunction with surveillance capsule results.
>
> (B) Any change in P-T curves will be implemented by the license amendment process (i.e., modifications of technical specifications) and will meet the requirements of Title 10 of the *Code of Federal Regulations* Section 50.60 (10 CFR 50.60) and 10 CFR Part 50, Appendix G.

RAI-4.2.4 Response

> (A) The following will be added to the HNP LRA, Section 4.2.4, at the end of the "Analysis" Subsection:
>
> > The current P-T limits are valid through 36 EFPY. The P-T limits for the extended period of operation will be managed by using approved fluence calculations when there are changes in power or core design in conjunction with surveillance capsule results.
>
> (B) The following will be added to the HNP LRA, Section 4.2.4 at the end of the "Analysis" Subsection:
>
> > P-T limits have been imposed on operational parameters at HNP, thereby assuring that the reactor vessel is operated within required safety margins in accordance with the requirements of 10 CFR 50.60 and 10 CFR 50, Appendix G. HNP has implemented changes in the P-T curves throughout the current period of operation using the license amendment process, and expects to continue to use the license amendment process to implement future changes in P-T curves for the remainder of the current period of operation and for the extended period of operation.

An amendment to the LRA is required in response to RAI-4.2.4.

NRC RAI-4.2.5

Since the P-T limits for the extended period of operation are not yet developed, the applicant should make a statement in the LRA that they will submit the appropriate analysis for the low temperature overpressure (LTOP) setpoints that will be valid for the license renewal period. Any change in the LTOP setpoints will be implemented by the license amendment process (i.e., modifications of technical specifications) and will meet the requirements of 10 CFR 50.60 and 10 CFR Part 50, Appendix G.

RAI-4.2.5 Response

The following will be added to Section 4.2.5 of the HNP LRA:

> HNP will submit the appropriate analysis for LTOP setpoints that will be valid for the period of extended operation. LTOP setpoints have been imposed on operational parameters at HNP, thereby assuring that the reactor vessel is operated within required safety margins in accordance with the requirements of 10 CFR 50.60 and 10 CFR 50, Appendix G. HNP has implemented changes in the LTOP setpoints throughout the current period of operation using the license amendment process, and expects to continue to use the license amendment process to implement future changes in LTOP setpoints for the remainder of the current period of operation and for the extended period of operation.

An amendment to the LRA is required in response to RAI-4.2.5.

NRC RAI-4.2.6

During the audit at the Harris Nuclear Plant, the staff was informed by the applicant that one reactor vessel nozzle was projected to achieve a neutron fluence greater than 1×10^{17} n/cm^2 (E > 1 MeV) at the end of the extended period of operation. This nozzle material was not listed in Tables 4.2-1, 4.2-2 and 4.2-3 of the LRA. According to Table 1V A-2 of NUREG-1801, Revision 1, ferritic materials are subject to neutron embrittlement when they are exposed to neutron fluence greater than 1×10^{17} n/cm^2 (E > 1 MeV) at the end of the extended period of operation. Therefore, the staff requests that the applicant provide the following for this nozzle material and its associated welds:

> (1) The RT_{PTS} value of the nozzle material and its associated welds per the requirements of Title 10 of the Code of Federal Regulations (CFR) Section 50.61.
>
> (2) The adjusted reference temperature value of the nozzle material and its associated welds that will be used for developing pressure-temperature limits per the requirements of 10 CFR Part 50, Appendix G.

(3) The upper shelf energy value of the nozzle material and its associated welds per the requirements of 10 CFR Part 50, Appendix G.

RAI-4.2.6 Response

The reactor vessel materials outside the traditional beltline region that are exposed to a 55 EFPY fluence greater than 10^{17} n/cm^2 (E > 1.0 MeV) were evaluated to determine if these materials should be considered beltline materials for the period of extended operation. The beltline is defined in 10 CFR 50.61(a)(3) as the region of the reactor vessel that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection for the most limiting material.

The evaluation found five reactor vessel materials above the traditional beltline region with 55 EFPY fluence values greater than 10^{17} n/cm^2 that were not previously analyzed for irradiation damage. The materials were:

1) Upper to Intermediate Circumferential Weld AC (Heat 4P4784, Linde 124),
2) Upper Shell (conservatively C0123-1),
3) Inlet Nozzle Weld 15-A, 15-B, 15-C (Heat 3P4966, Linde 124),
4) Inlet Nozzle (conservatively 438B-5), and
5) Upper Shell Longitudinal Welds BE/BF (Heat 4P4784, Linde 124).

The reactor vessel materials below the traditional beltline region did not include any additional materials that required analysis for irradiation damage, in accordance with 10 CFR 50.61.

Table 4.2-4 summarizes the decrease in Charpy upper shelf energy (C_VUSE) for the five materials above the traditional beltline region with 55 EFPY fluence values greater than 10^{17} n/cm^2 (E > 1.0 MeV). These values were calculated per the requirements of 10 CFR 50, Appendix G. Based on the C_VUSE analysis, the material locations above the traditional beltline region are not limiting since they are projected to maintain Charpy upper shelf energies greater than that of the intermediate shell plate B4197-2, which is located inside the traditional beltline region. C_VUSE for the intermediate shell plate B4197-2, the limiting reactor vessel material, is 52.8 ft-lbs at ¼-thickness.

Table 4.2-5 summarizes the reactor vessel pressurized thermal shock (PTS) reference temperature (RT_{PTS}) for the five materials above the traditional beltline region with 55 EFPY fluence values greater than 10^{17} n/cm^2 (E > 1.0 MeV). These values were calculated per the requirements of 10 CFR 50.61. Based on the RT_{PTS} analysis, none of the material locations above the traditional beltline region are limiting, since they are projected to maintain RT_{PTS} values less than that of the intermediate shell plate, heat number B4197-2, which is located inside the traditional beltline region. RT_{PTS} for the intermediate shell plate B4197-2, the limiting reactor vessel material, is 199.9°F.

Table 4.2-6 summarizes the reactor vessel adjusted reference temperatures (ART) at the ¼-thickness and ¾-thickness wall locations for the five materials above the traditional beltline region with 55 EFPY fluence values greater than 10^{17} n/cm^2 (E > 1.0 MeV). These values were calculated per the requirements of 10 CFR 50, Appendix G. Based on the ART analysis, none of the material locations above the traditional beltline region are limiting, since they are projected to maintain ART values less than that of the intermediate shell plate, heat number B4197-2, which is located inside the traditional beltline region. ART for the intermediate shell plate B4197-2, the limiting reactor vessel material, is 195.3°F at ¼-thickness and 183.6°F at ¾-thickness.

An amendment to the LRA is required in response to RAI-4.2.6.

TABLE 4.2-4 UPPER SHELF ENERGY (C_VUSE) EVALUATION THROUGH YEAR 60 (55 EFPY) FOR MATERIALS ABOVE THE TRADITIONAL BELTLINE REGION WITH 55 EFPY FLUENCE VALUES GREATER THAN 10^{17} n/cm^2 (E > 1.0 MeV)

Material Description			Cu wt%	¼T Fluence[1] (x 10^{19}) n/cm^2	Initial C_VUSE ft-lbs	Predicted C_VUSE Per R.G. 1.99, Revision 2	
Reactor Vessel Beltline Region Location	Heat Number	Type				C_VUSE ft-lbs	% Decrease
Upper to Intermediate Circumferential Weld (AC)	4P4784	ASA/Linde 124	0.06	0.2073	95[4]	81.9[5]	13.8
Upper Shell	C0123-1[2]	SA-533 Gr. B	0.12	0.2073	84	71.8	14.5
Inlet Nozzle Weld (15-A, 15-B, 15-C)	3P4966	ASA/Linde 124	0.02	0.0113	63[4]	59.5[5]	5.5
Inlet Nozzle	438B-5[3]	SA-508 Cl. 2	0.35	0.0113	128	108.5	15.2
Upper Shell Longitudinal WeldS (BE/ BF)	4P4784	ASA/Linde 124	0.06	0.2073	95 [4]	81.9[5]	13.8

1. Calculated based on guidelines in RG 1.99, Revision 2. The 55 EFPY inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel. The ¼T location fluence value is determined by calculating the ¼T depth into the vessel and adding the minimum cladding thickness.
2. Upper Shell Plate C0123-1 exhibited a higher value for initial RT_{NDT} than the other Upper Shell Plate C0224-1. Therefore, Upper Shell Plate C0123-1 was chosen as the more conservative plate for the purpose of the embrittlement evaluation.
3. Inlet Nozzle 438B-5 exhibited a higher value for initial RT_{NDT} than the other Inlet Nozzles (438B-4 and 438B-6). Therefore, Inlet Nozzle 438B-5 was chosen as the most conservative inlet nozzle for the purpose of the embrittlement evaluation.
4. As defined by ASTM E 185-82, these Charpy data are from the transition region and not the upper shelf region of the Charpy curve, since the specimen fracture surfaces exhibit less than 95% shear. Therefore, these Charpy data do not represent upper shelf energy levels and are considered conservative.
5. Predicted value is conservative, since initial Charpy data are from the transition region and not the upper shelf region of the Charpy curve.

TABLE 4.2-5 PTS REFERENCE TEMPERATURE EVALUATION THROUGH YEAR 60 (55 EFPY) FOR MATERIALS ABOVE THE TRADITIONAL BELTLINE REGION WITH 55 EFPY FLUENCE VALUES GREATER THAN 10^{17} n/cm^2 (E > 1.0 MeV)

Material Description				Chemical Composition		Initial RT_{NDT}	Chemistry Factor	55 EFPY Fluence at the Inside Wetted Surface 10^{19} n/cm^2(1)	Fluence Factor	ΔRT_{PTS} °F	Margin °F	RT_{PTS} °F	Screening Criteria °F
Reactor Vessel Beltline Region Location	Material ID	Heat Number	Type	Cu wt%	NI wt%								
RT_{PTS} Calculation Per 10 CFR 50.61 Using Tables													
Upper to Intermediate Circumferential Weld	AC	4P4784	ASA/Linde 124	0.06	0.91	-20	82.0	0.3401	0.703	57.6	65.5	103.1	300
Upper Shell	C0123-1(2)	C0123-1(2)	SA-533 Gr. B	0.12	0.60	42	83.0	0.3401	0.703	58.4	34.0	134.4	270
Inlet Nozzle Weld	15-A, 15-B, 15-C	3P4966	ASA/Linde 124	0.02	0.92	-56	27.0	0.0203	0.173	4.7	34.3	-17.0	270
Inlet Nozzle	438B-5(3)	438B-5(3)	SA-508 Cl. 2	0.35	0.89	0	255.0	0.0203	0.173	44.1	34.0	78.1	270
Upper Shell Longitudinal Welds	BE/BF	4P4784	ASA/Linde 124	0.06	0.91	-20	82.0	0.3401	0.703	57.6	65.5	103.1	270

1. RT_{PTS} is normally calculated using the fluence at the clad/base metal interface in accordance with 10 CFR 50.61. However, HNP calculated RT_{PTS} using the 55 EFPY inside wetted surface fluence, which is higher than the 55 EFPY fluence at the clad/base metal interface.
2. Upper Shell Plate C0123-1 exhibited a higher value for initial RT_{NDT} than the other Upper Shell Plate C0224-1. Therefore, Upper Shell Plate C0123-1 was chosen as the more conservative plate for the purpose of the embrittlement evaluation.
3. Inlet Nozzle 438B-5 exhibited a higher value for initial RT_{NDT} than the other Inlet Nozzles (438B-4 and 438B-6). Therefore, Inlet Nozzle 438B-5 was chosen as the most conservative inlet nozzle for the purpose of the embrittlement evaluation.

TABLE 4.2-6 ADJUSTED REFERENCE TEMPERATURE EVALUATION THROUGH YEAR 60 (55 EFPY) FOR MATERIALS ABOVE THE TRADITIONAL BELTLINE REGION WITH 55 EFPY FLUENCE VALUES GREATER THAN 10^{17} n/cm^2 (E > 1.0 MeV)

Material Description				Chemical Composition		Initial RT_{NDT}	Chemistry Factor	55 EFPY Fluence 10^{19} n/cm^2			ΔRT_{NDT}, °F at 55 EFPY		Margin		ART, °F at 55 EFPY	
Reactor Vessel Beltline Region Location	Material ID	Heat Number	Type	Cu wt%	Ni wt%			Inside surface	¼T[1] Location	¾T[1] Location	¼T Location	¾T Location	¼T Location	¾T Location	¼T Location	¾T Location
RG 1.99, Revision 2, Position 1.1																
Upper to Intermediate Circumferential Weld	AC	4P4784	ASA/ Linde 124	0.06	0.91	-20	82.0	0.3401	0.2073	0.0818	47.4	31.0	58.3	46.0	85.7	57.0
Upper Shell	C0123-1[2]	C0123-1[2]	SA-533 Gr. B	0.12	0.60	42	83.0	0.3401	0.2073	0.0818	48.0	31.4	34.0	31.4	124.0	104.8
Inlet Nozzle Weld	15-A, 15-B, 15-C	3P4966	ASA/ Linde 124	0.02	0.92	-56	27.0	0.0203	0.0113	0.00372	3.2	1.5	34.2	34.0	-18.6	-20.5
Inlet Nozzle	438B-5[3]	438B-5[3]	SA-508 Cl. 2	0.35	0.89	0	255.0	0.0203	0.0113	0.00372	30.3	13.8	30.3	13.8	60.6	27.6
Upper Shell Longitudinal Welds	BE/BF	4P4784	ASA/ Linde 124	0.06	0.91	-20	82.0	0.3401	0.2073	0.0818	47.4	31.0	58.3	46.0	85.7	57.0

1. Calculated based on guidelines in RG 1.99, Revision 2. The 55 EFPY inside surface fluence is the calculated value at the "wetted" surface of the reactor vessel. The ¼T and ¾T location fluence values are determined by calculating the ¼T and ¾T depth into the vessel and adding the minimum cladding thickness.
2. Upper Shell Plate C0123-1 exhibited a higher value for initial RT_{NDT} than the other Upper Shell Plate C0224-1. Therefore, Upper Shell Plate C0123-1 was chosen as the more conservative plate for the purpose of the embrittlement evaluation.
3. Inlet Nozzle 438B-5 exhibited a higher value for initial RT_{NDT} than the other Inlet Nozzles (438B-4 and 438B-6). Therefore, Inlet Nozzle 438B-5 was chosen as the most conservative inlet nozzle for the purpose of the embrittlement evaluation.