From: "Tyner, Donna" <dtyner@entergy.com>
To: "Bo Pham" <BMP@nrc.gov>
Date: 11/28/2007 4:48:37 PM
Subject: IPEC RAI Reply - Reactor Vessel Surveillance Program and the Reactor Neutron Embrittlement Time-Limited Aging Analyses

Bo,

Attached is an advance electronic copy of our RAI Reply on Reactor Vessel Surveillance Program and the Reactor Neutron Embrittlement Time-Limited Aging Analyses. Hard copy will follow in mail.

Thanks,

Donna

Hearing Identifier: IndianPointUnits2and3NonPublic
Email Number: 3

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From: "Tyner, Donna" <dtyner@entergy.com>

Created By: dtyner@entergy.com

Recipients
"Bo Pham" <BMP@nrc.gov>

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Thanks,
Donna

Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249
Tel 914 734 6700

Fred Dacimo
Site Vice President

November 28, 2007

Re: Indian Point Units 2 & 3
Docket Nos. 50-247 & 50-286

NL-07-140

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

SUBJECT: **Reply to Request for Additional Information**
Regarding License Renewal Application

Reference: NRC letter dated October 29, 2007; "Requests for Additional Information for the Review of the Indian Point Nuclear Generating Unit Nos. 2 and 3, License Renewal Application"

Dear Sir or Madam:

Entergy Nuclear Operations, Inc is providing, in Attachment I, the additional information requested in the referenced letter pertaining to NRC review of the License Renewal Application for Indian Point 2 and Indian Point 3. The additional information provided in this transmittal addresses staff questions regarding the Reactor Vessel Surveillance Program and the Reactor Neutron Embrittlement Time-Limited Aging Analyses.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. R. Walpole, Manager, Licensing at (914) 734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 11/28/07

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

cc: next page

cc: Mr. Bo M. Pham, NRC Environmental Project Manger
Ms. Kimberly Green, NRC Safety Project Manager
Mr. John P. Boska, NRC NRR Senior Project Manager
Mr. Samuel J. Collins, Regional Administrator, NRC Region I
Mr. Mark Cox, NRC Senior Resident Inspector, IP2
Mr. Paul Cataldo, NRC Senior Resident Inspector, IP3
Mr. Paul D. Tonko, President, NYSERDA
Mr. Paul Eddy, New York State Dept. of Public Service

ATTACHMENT I TO NL-07-140

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING

LICENSE RENEWAL APPLICATION

(Reactor Vessel Surveillance Program and the Reactor Neutron Embrittlement Time-Limited Aging Analyses)

ENTERGY NUCLEAR OPERATIONS, INC
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 and 3
DOCKETs 50-247 and 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
LICENSE RENEWAL APPLICATION (LRA)
REQUESTS FOR ADDITIONAL INFORMATION (RAI)

The U.S. Nuclear Regulatory Commission (NRC or staff) has reviewed the information related to the Reactor Vessel Surveillance Program and the Reactor Vessel Neutron Embrittlement Time-Limited Aging Analyses provided by the applicant in the Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3) LRA. The staff has identified that additional information is needed to complete the review as addressed below.

RAI B.1.1.32-1

The Reactor Vessel Surveillance Program is identified as consistent with the program described in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Section XI.M31, "Reactor Vessel Surveillance," with enhancements. The enhancements are: (1) to withdraw and test a standby capsule to cover the peak reactor vessel fluence that is expected through the end of the period of extended operation; and (2) to revise procedures to require that tested and untested specimens from all capsules pulled from the reactor vessel be maintained in storage. Identify the lead factors for each standby capsule, the materials available to be tested in each capsule, and the date for capsule withdrawal to ensure that the fluence of the surveillance capsule will be equal or greater than the peak reactor vessel fluence through the end of the period of extended operation.

Response for RAI B.1.1.32-1

There are three available surveillance capsules in each IPEC reactor vessel as follows.

IP2

Capsule U, Capsule W and Capsule X have lead factors of 1.2 and contain the following materials: Plates B2002-1, B2002-2 and B2002-3 as well as correlation monitor material. None of these capsules contain weld material specimens.

To ensure that the fluence of the surveillance capsule will be equal or greater than the peak reactor vessel fluence through the end of the period of extended operation, at least one capsule will remain in the reactor vessel until approximately 40 EFPY. This burnup should be attained approximately 8 years prior to the end of the period of extended operation or around 2025.

IP3

Capsule W, with a lead factor of 1.52, contains the following materials.

- Plate B2803-3, longitudinal specimens
- Plate B2803-3, transverse specimens
- Plate B2802-1, longitudinal specimens
- Weld material

Capsule U, with a lead factor of 1.52, contains the following materials.

- Plate B2803-3, longitudinal specimens
- Plate B2803-3, transverse specimens
- Plate B2802-1, longitudinal specimens
- Weld material

Capsule V, with a lead factor of 1.52, contains the following materials.

- ASTM reference material, longitudinal specimens
- Plate B2803-3, transverse specimens
- Weld material
- Weld heat affected zone material

To ensure that the fluence of the surveillance capsule will be equal or greater than the peak reactor vessel fluence through the end of the period of extended operation, a capsule must remain in the reactor vessel until approximately 32 EFPY. This burnup should be attained approximately 16 years prior to the end of the period of extended operation or around 2019.

Per commitment 22, IP2 and IP3 will revise the specimen capsule withdrawal schedules to withdraw and test a standby capsule to cover the peak reactor vessel fluence expected through the end of the period of extended operation.

The withdrawal schedules will be submitted as required by 10 CFR Part 50, Appendix H, Section III.B.3.

RAI 4.2.1-1

The Charpy Upper-Shelf Energy (USE) and Pressurized Thermal Shock analyses utilize the neutron fluence at 48 effective full power years (EFPY) to represent the neutron fluence for the reactor vessels at the end of the period of extended operation.

A) What were the EFPY achieved for each unit prior to the last refueling outage? What capacity factors and neutron flux were assumed for each unit from the last refueling outage to the end of the period of extended operation to result in 48 EFPY at the end of the period of extended operation? Explain why these capacity factors and neutron flux values are applicable for determining the neutron fluence for the reactor vessels at the end of the period of extended operation.

B) How will future capacity factors, neutron flux and neutron fluence values be monitored to ensure 48 EFPY values bound the actual conditions of the reactor vessels at the end of the period of extended operation?

Response for RAI 4.2.1-1

A) At the end of Cycle 17 (4/19/06), Indian Point 2 had operated for 21.802 EFPY. At the end of Cycle 14 (3/6/07), Indian Point 3 had operated for 19.311 EFPY.

Neutron flux corresponding to the licensed reactor power rating was assumed from the end of the last refueling outage through the end of the period of extended operation. A two-year cycle (730 days), includes a 25-day refueling outage (705 operating days). IP2 would have to operate at a capacity factor of 0.99 during the periods between refueling outages to attain 48 EFPY at the end of the period of extended operation (September 13, 2033). IP3 would have to operate at a capacity factor of greater than 1.0 during the periods between refueling outages to attain 48 EFPY at the end of the period of extended operation (December 15, 2035).

B) Future neutron fluence values will be monitored to ensure 48 EFPY values bound the actual conditions of the reactor vessels in the same way current neutron fluence is monitored to ensure P-T curves remain valid. Plant service lifetime in EFPY is routinely reviewed by engineering and licensing personnel. Additionally, accumulated reactor vessel fluence is checked on a cyclic basis as part of the core reload change package. Because of shutdowns for refueling, plant operation cannot exceed 48 EFPY. If rated power level is increased at a future date, the associated engineering evaluations will ensure the resulting increase in flux is properly accounted for in determining the neutron fluence for the reactor vessels at the end of the period of extended operation.

RAI 4.2.2-1

Table 4.2-2 in the LRA indicates that the percentage drop in Charpy USE for plate B2803-3 is 21.3 percent at 48 EFPY. The percentage drop in Charpy USE for plate B2803-3 was determined using its surveillance data, in accordance with Position 2.2 of Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." Provide the analysis that was used to determine the percentage drop in Charpy USE for plate B2803-3, include all surveillance data (unirradiated and irradiated Charpy USE and surveillance capsule neutron fluence) and references for the surveillance data.

Response for RAI 4.2.2-1

The drop in USE was determined following the guidance of RG 1.99, Rev. 2, Position 2.2, which states "The decrease in upper-shelf energy may be obtained by plotting the reduced plant surveillance data on Figure 2 of this guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data. This line should be used in preference to the existing graph."

However, to maximize accuracy, IPEC used a spreadsheet and the equations for the RG 1.99 Figure 2 curves (available in NUREG/CR-5799) to effectively plot a parallel curve. Four sets of surveillance data were reviewed, and a correction factor for each surveillance point was determined. The lowest correction factor (giving the highest % drop in USE) was then used in the formula to predict the 48 EFPY % drop in USE.

NUREG/CR-5799, Equation 5: Percent drop in USE for base metal = (100 Cu + 14) * fluence ^ 0.2368

For surveillance data, Measured $Reduction_{sample}$ = (100 Cu + Correction Factor) * $fluence_{sample}$ ^ 0.2368

(These are curves parallel to the RG 1.99 curves and solved for the Correction Factors.)

The 48 EFPY predicted percent drop in USE, adjusted for surveillance data, now equals
(100 Cu + Correction Factor) * $fluence_{48\ EFPY}$ ^ 0.2368

The surveillance data from the reactor vessel integrity database (RVID2) as shown below was used to determine the correction factor.

Type	*Heat ID*	*Capsule*	*Lead Factor*	*Fluence, 10E19*	*%Cu*	*%Ni*	*USE, Unirr*	*Measured USE*	*% drop in USE offset line*
Plate	*A0512-2*	*T*	*3.74*	*0.31*	*0.24*	*0.52*	*67*	*58*	*1.24451*
Plate	*A0512-2*	*Y*	*3.74*	*0.72*	*0.24*	*0.52*	*67*	*57*	*1.20619*
Plate	*A0512-2*	*Z*	*3.46*	*1.04*	*0.24*	*0.52*	*105*	*82*	*1.33662*
Plate	*A0512-2*	*Z*	*3.46*	*1.04*	*0.24*	*0.52*	*67*	*56*	*1.21140*

RAI 4.2.2-2

According to Table 4.2-1 in the LRA, plates B2002-3 and B2003-1 in IP2 are projected to have a Charpy USE of 47.4 ft-lb and 49.8 ft-lb at 48 EFPY, respectively. This table also indicates that the material type was A302-B. Since the Charpy USE values are below 50 ft-lb, Section 4.2.2 indicates that the USE is acceptable in accordance with the Westinghouse 4-loop plant analysis described in WCAP-13587, Revision 1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation For Westinghouse Pressurized Water Reactors." WCAP-13587, Revision 1 was reviewed by the staff in a safety assessment in a letter dated April 21, 1994 to W. H. Rasin (Nuclear Management and Resources Council, NUMARC). This safety assessment indicates that licensees must confirm that the bounding plate used in the report has a lower J-R curve than any beltline material in the vessel. This safety assessment indicates that the A302-B material J-R curve methodology used in WCAP-13587, Revision 1 was unacceptable. However, the A302-B J-R curve methodology in a letter from D. B. Waters (Carolina Power and Light Company) dated December 21, 1993, includes appropriate adjustments for USE and temperature corrections and is acceptable to the staff.

A) Provide the J-R curve for the limiting IP2 plate, using corrections for material type used in the IP-2 reactor vessel, at 48 EFPY. Compare this curve to the J-R applied values in WCAP-13587, Revision 1 to demonstrate the analysis in the WCAP is bounding for the IP2 reactor vessel.

B) Table 3-1 in WCAP-13587, Revision 1 indicates the Westinghouse 4-loop plant J-R applied values are applicable for reactor vessels with a thickness of 8.5 inches and an inner radius of 86.5 inches and subject to Level A, B, C, and D conditions specified in Section 3.0 of the WCAP. Compare the wall thickness and inner radius of the IP2 reactor vessel at its beltline to the values used in the Westinghouse 4-loop plant analysis. Compare the Level A, B, C, and D conditions specified in Section 3.0 of the WCAP to the Level A, B, C and D conditions for IP2. Explain why the Westinghouse 4-loop plant analysis in WCAP-13587, Revision 1 is applicable to IP2.

Response for RAI 4.2.2-2

A) Studies done in the early 1990's indicated that the ability of SA 302 Grade B material to resist crack propagation may be inferior to that of SA 533, grade B. The SER mentioned in this question specifically excluded SA 302 Grade B from being bounded by the WOG evaluation.

Combustion Engineering Form N-1A Manufacturers Data Report for Nuclear Vessels lists the IP2 reactor vessel shell material as low alloy steel SA-302-B (Code Case 1339). An excerpt from Code Case 1339 (Special Ruling) states: "1) Plates made of materials conforming to the requirements of SA 302 Grade B except the composition may be modified to include from 0.4 to 1.0 per cent nickel and the thickness may exceed 4 inches."

The material listed in Table 4.2-1 in the LRA is SA 302, Grade B modified (with nickel added). In addition, this material is vacuum treated as required by CE Specification No. P3F12(a) and Code Case 1339.

The behavior of the IP2 and the IP3 vessel material is similar to that of SA 533 Grade B, which has improved material properties as compared to SA 302 Grade B. This conclusion is also consistent with the fact that the WCAP-13587, Revision 1 analysis included SA-302 Grade A, SA-302 grade B, SA-302 Grade B modified, SA-508 Class 2 and Class 3 and SA-533 Class 1 (see WCAP Table 2-3) and the SER only took exception to the material properties of SA-302 Grade B; it did not take exception to the material properties used for SA-302 Grade B modified.

WCAP-13587, Revision 1, Table 2-3 shows that the 4 loop plant EOL USE (ft-lb) bounding value is 44. The limiting plates for IP2 RPV are acceptable and bounded by the Westinghouse curves based on their USE values (47.4 and 49.8 ft-lb).

B) <u>Compare the wall thickness and inner radius of the IP2 reactor vessel at its beltline to the values used in the Westinghouse 4-loop plant analysis</u>

The IP2 reactor vessel is 173" inside diameter with an 8.625" nominal wall thickness. Therefore the 86.5" inner radius and the 8.5" minimal wall thickness dimensions used in WCAP-13587 bound the IP2 reactor vessel dimensions since these values result in conservative pressure stresses and have no significant impact on the through wall thermal stresses.

Compare the Level A, B, C, and D conditions specified in Section 3.0 of the WCAP to the Level A, B, C and D conditions for IP2

The Level A and B condition used in WCAP-13587 was a cool down rate of 100 degrees F per hour. This cool down rate is the same as the cool down rate for IP2.

The Level C conditions used in WCAP-13587 correspond to a small steam line break with the pressure and temperature time histories provided in Figure 3-1. These conditions were reviewed against the analyses provided in Chapter 14 of the IP2 FSAR and it was concluded that the IP2 FSAR conditions were bounded by the conditions analyzed in the WCAP.

The Level D conditions used in the WCAP analysis were associated with the large steam line break and are provided in Figure 3-2. These conditions were compared to the large steam line break conditions provided in Chapter 14 of the IP2 FSAR. This comparison concluded that the conditions provided in the FSAR are bounded by the conditions used in the WCAP analysis.

Based on the above, it is concluded that the evaluations provided in WCAP-13587 are applicable to the IP2 reactor vessel.

RAI 4.2.5-1

A) Table 4.2-3 in the LRA indicates that the ΔRT_{NDT} value caused by irradiation for the intermediate shell axial welds and the lower shell axial welds in IP2 were determined using surveillance data reported in WCAP-15629, Revision 1, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation." This WCAP has surveillance data from IP2, IP3, and H.B. Robinson, Unit 2. The IP2 fluences were calculated using approved methodologies (WCAP-15557-R0, "Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology," and WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves") that are based on RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001, (This RG requires the use of ENDF/BVI for determining neutron cross-sections which are included in the BUGLE-96 cross-section file). In addition, there is excellent agreement between calculated and corresponding measured values. The IP3 capsule analyses also used approved methods and cross sections, thus, they are acceptable. The H.B. Robinson calculations are reported in WCAP-14044, "Westinghouse Surveillance Capsule Neutron Fluence Re-evaluation," that was issued in 1994 (before the issuance of RG 1.190 and the availability of BUGLE-96 and ENDF/BVI). WCAP-15629, Revision 1 indicates that 15% was added to the values reported in WCAP-14044. Explain why 15% was added to the values reported in WCAP-14044. Provide neutron fluence values derived using a methodology that adheres to the guidance in RG 1.190. If the revised analysis results in a change in neutron fluence for the H.B. Robinson, Unit 2 surveillance capsules, provide the ΔRT_{NDT} value caused by irradiation and the RT_{PTS} value for the intermediate shell axial welds and the lower shell axial welds in IP2 and provide the surveillance data analysis required by 10 CFR 50.61(c)(2)(i).

B) Table 4.2-4 in the LRA indicates that the ΔRT_{NDT} caused by irradiation for the lower shell plate B2803-3 in IP3 was determined using surveillance data reported by the licensee's response to Generic Letter (GL) 92-01, "Reactor Vessel Structural Integrity." This surveillance data was reported in Attachment I to a September 4, 1998, letter from J. Knubel (New York Power Authority). As discussed in RAI 4.2.5-1A, the surveillance data from IP3 is also reported in WCAP-15629, Revision 1. The neutron fluence values for the IP3 surveillance capsule that are reported in WCAP-15629, Revision 1 and in the September 4, 1998, letter have different values. The applicant is requested to revise the PTS analyses using neutron fluence values for the surveillance capsules that are determined using the guidance in RG 1.190 and to provide the surveillance data analysis required by 10 CFR 50.61(c)(2)(I).

<u>Response for RAI 4.2.5-1</u>

A) The reference to WCAP-15629 in LRA Table 4.2-3 pertains to chemistry factors used to determine the 48 EFPY RT_{PTS}. Per Table 4 of WCAP-15629, the chemistry factor calculation for the surveillance weld material uses two H. B. Robinson (HBR) surveillance data points and five IP2/IP3 data points to determine axial weld chemistry factors.

The fluences shown for the HBR plant are based on the data taken before the RG 1.190 guidance was available. Westinghouse, in preparation of WCAP-15629 Rev 1, used the information at hand and added a 15% penalty for a conservative margin. Revised fluences for HBR consistent with the guidance of RG 1.190 have been calculated by Westinghouse. These revised fluences are to be incorporated into a new WCAP, which is being prepared to support an upcoming IP2 license amendment request for Technical Specification pressure-temperature curves. The draft calculations to date using the revised HBR data result in a lower chemistry factor for the IP2 surveillance weld material. Entergy anticipates submittal of the amendment request in the second quarter of 2008.

Since the revised HBR fluence results in a lower chemistry factor than initially reported, the RT_{PTS} value for 48 EFPY will be lower than the value of 261°F cited in the LRA.

B) The 48 EFPY neutron fluence values reported in LRA Table 4.2-4 were taken from a 2003 Westinghouse calculation supporting stretch power uprate. The neutron transport and dosimetry evaluation methods used to determine the fluence in the 2003 calculation followed the guidance of RG 1.190.

The calculation of IP3 RT_{PTS} values do not rely on WCAP-15269 and therefore the fluence values reported in WCAP-15269 do not impact LRA Table 4.2-4. The response to GL 92-01 referenced in LRA Table 4.2-4 provided only the chemistry factors used to determine the 48 EFPY RT_{PTS} values. These chemistry factors remain valid and are consistent with the values reported in the Reactor Vessel Integrity Database (RVID2).

RAI 4.2.5-2

10 CFR 50.61(b)(4) indicates that each pressurized water nuclear power reactor for which the analysis required by PTS rule indicates that if there is no reasonably practicable flux reduction program to prevent the RT_{PTS} value from exceeding the PTS screening criteria based on the neutron fluence at the expiration date of the operating license, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criteria is allowed. The analysis must be submitted at least three years before the RT_{PTS} value is projected to exceed the PTS screening criteria.

Section 4.2.5 in the LRA indicates that the RT_{PTS} value for plate B2803-3 in IP3 will exceed the PTS screening criterion. Identify the flux reduction program initiated by the applicant to prevent the RT_{PTS} value for plate B2803-3 in IP3 from exceeding the PTS screening criterion. Based on the information provided in response to RAI 4.2.5-1(B) and RAI 4.2.1-1, identify when the RT_{PTS} value for plate B2803-3 in IP3 is projected to exceed the PTS screening criterion.

Response for RAI 4.2.5-2

Plate B2803-3 will reach the screening criterion at approximately 37 EFPY. Using a plant capacity factor of 0.97 after 2007, IP3 will achieve 37 EFPY approximately 9 years after entering the period of extended operation.

With regard to flux reduction, IP3 implemented a low-low leakage loading plan in 1986 by placing fresh fuel in the interior of the core. Flux suppressors consisting of Pyrex glass were added to eight corner locations of the core in 1995. Since 1999, the suppressor material has been unclad hafnium.

These flux reduction methods have been successful. However, these methods alone will not prevent plate B2803-3 from reaching the screening criterion during the period of extended operation.

Commitment 32 states,

As required by 10 CFR 50.61(b)(4), IP3 will submit a plant-specific safety analysis for plate B2803-3 to the NRC three years prior to reaching the RT_{PTS} screening criterion. Alternatively, the site may choose to implement the revised PTS (10 CFR 50.61) rule when approved, which would permit use of Regulatory Guide 1.99, Revision 3.

Application of Regulatory Guide 1.99, Revision 3 to plate B2803-3 is expected to result in an acceptable RT_{PTS} value at 48 EFPY for IP3.