Entergy

Entergy Nuclear Northeast
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Patric Conroy
Licensing Manager
Indian Point Energy Center

June 8, 2005

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-05-072

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

SUBJECT: Request For Relief To Extend The Third 10-Year Inservice Inspection Interval For The Reactor Vessel Weld Examination

References: 1. R. Gramm of the NRC to G. Bischoff of the WOG, "Summary of teleconference with the Westinghouse Owners Group regarding potential one cycle relief of reactor pressure vessel shell weld inspections at pressurized water reactors related to WCAP-16168-NP, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Intervals", January 27, 2005.

Dear Sir:

Entergy Nuclear Operations, Inc. (Entergy) is submitting Relief Request No. 73 (RR-73) (Enclosure 1 and associated Attachments 1 and 2) for Indian Point Unit No. 2 (IP2) pertaining to an extension of the inservice inspection (ISI) interval for the subject Reactor Vessel (RV) Weld examinations by one refueling cycle beyond the 10-year interval.

The requirements for a technical basis to extend the 10-year RV ISI interval by one refueling cycle are contained in a letter, from R. Gramm of the NRC to G. Bischoff of the Westinghouse Owners Group, dated January 27, 2005 (Reference 1). This letter provides the basis for the one refueling cycle extension of the 10-year inspection interval for the subject examinations.

Entergy has concluded that the proposed alternative provides an acceptable level of quality and safety. The relief is requested under the provisions of 10CFR 50.55a(a)(3)(i).

Entergy requests approval of the relief request by February 2006, to support IP2 Refueling Outage (RFO) – 2R17.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. Patric W. Conroy, Licensing Manager at 914-734-6668.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. Patric W. Conroy, Licensing Manager at 914-734-6668

Very truly yours,

Patric W. Conroy
Licensing Manager
Indian Point Energy Center

Enclosure 1: IP2 Relief Request Number RR-73

cc: Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission

Resident Inspector's Office
Indian Point Unit 2
U.S. Nuclear Regulatory Commission

Mr. John P. Boska, Sr. Project Manager, Section 2
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Mr. Peter R. Smith, President
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ENCLOSURE 1 TO NL-05-072

REQUEST FOR RELIEF TO EXTEND THE THIRD 10-YEAR INSERVICE INSPECTION INTERVAL FOR THE REACTOR VESSEL WELD EXAMINATION
RELIEF REQUEST RR-73

(7 Pages)

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

INDIAN POINT ENERGY CENTER UNIT NO. 2
THIRD 10-YEAR INSERVICE INSPECTION INTERVAL

IPEC UNIT 2 RELIEF REQUEST NO. RR-73

1.0 ASME Code Component(s) Affected

The affected component is the IP2 (IP2) reactor vessel (21RV), specifically the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code Section XI Examination Categories and Item Numbers covering examinations of the reactor vessel (RV). These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME BPV Code, Section XI.

Examination Category	Item No.	Description	Component IDs
B-A	B1.11	Circumferential Shell Welds	RPVC2, RPVC3, RPVC4
B-A	B1.12	Longitudinal Shell Welds	RPVL1 thru RPVL8
B-A	B1.22	Meridional Shell Welds	RPVM1 thru RPVM6
B-A	B1.30	Shell-to-Flange Weld	RPVC1
B-D	B3.90	Nozzle-to-Vessel Welds	RPVN1 thru RPVN8
B-D	B3.100	Nozzle Inner Radius Areas	RPVN1(IR) thru RPVN8(IR)

(Throughout this request the above examination categories are referred to as "the subject examinations" and the ASME BPV Code, Section XI, is referred to as "the Code.")

2.0 Applicable Code Edition and Addenda

ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Code 1989 Edition, No Addenda.

3.0 Applicable Code Requirement

IWB-2412, Inspection Program B, requires volumetric examination of essentially 100% of reactor pressure vessel pressure retaining welds identified in Table IWB-2500-1 once each ten-year interval. In accordance with IWA-2430(d) and IWA-2430(e), the IP2 third inspection interval is currently scheduled to conclude on or before December 31, 2006.

4.0 Reason for Request

An alternative is requested from the requirement of IWA-2412, Inspection Program B, that volumetric examination of essentially 100% of reactor pressure vessel pressure retaining welds, Examination Categories B-A and B-D welds, be performed once each ten-year interval. Extension of the inspection interval for the subject examinations by one refueling cycle beyond the currently scheduled 10-year interval is requested.

The intent of the requested one refueling cycle extension is to allow for deferment of the subject examinations to allow time for NRC review of industry efforts (References 1 and 2) to extend the in-service inspection (ISI) interval for the subject examinations from 10 to 20 years. These efforts use ASME Section XI Code Case N-691 (Reference 3) as a basis for using risk-informed insights to show that extending the inspection interval from 10 to 20 years results in a change in reactor vessel failure frequency that satisfies the requirements of Regulatory Guide 1.174 (Reference 4). Following NRC approval of these efforts, Entergy intends to submit a separate request to extend the current 10-year interval for IP2 to 20 years. This 20-year inspection interval will result in a reduction in man-rem exposure and examination costs.

5.0 Proposed Alternative and Basis for Use

The third inspection interval for IP2 started on July 1, 1994 and will end on or before December 31, 2006. This inspection interval includes the IWA-2430(d) allowed one year extension and an IWA-2430(e) allowed 647 day extension. The subject examinations are currently scheduled to be performed in the Spring of 2006 during refueling outage 2R17. The proposed inspection date is one refueling outage beyond the Code allowed inspection interval and enables the subject examinations to be performed during refueling outage 2R18 in the Spring of 2008. In accordance with 10 CFR 50.55a(a)(3)(i), this interval extension is requested on the basis that the current inspection interval can be extended while providing an acceptable level of quality and safety.

The requirements for a technical basis to extend the 10-year RV ISI interval by one refueling cycle are contained in a letter, from R. Gramm of the NRC to G. Bischoff of the Westinghouse Owners Group, dated January 27, 2005 (Reference 5). This letter provides the basis for the one refueling cycle extension of the 10-year inspection interval for the subject examinations.

The technical justification for the extension of the inspection interval for the subject examinations was developed based on the guidance provided in Reference 5. The technical justification consists of five areas.

These are:

5.1 Plant specific reactor vessel inservice inspection history
5.2 Fleet-wide reactor vessel inservice inspection history
5.3 Degradation mechanisms in the reactor vessel
5.4 Material condition of the reactor vessel relative to embrittlement
5.5 Operational experience relative to RV structural integrity challenging events

5.1 Indian Point Unit 2 Reactor Vessel Inservice Inspection History

IP2 is in its third inservice inspection interval for the reactor pressure vessel. Therefore, the pre-service inspection and two inservice inspections have been performed on the Category B-A and B-D welds.

Inspections of the Category B-A and B-D welds have been performed in accordance with Regulatory Guide 1.150 (Reference 6), have achieved acceptable coverage, and with the exception of one indication (see the discussion below), no reportable indications have been found. Based on the examination method and coverage obtained, it is reasonable to conclude that the examinations were of sufficient quality to detect any significant flaws which could challenge reactor vessel integrity. The last inspection history for the welds to which the subject examinations apply is contained in Attachment 1.

Reactor Vessel Indication at 345 Degrees Vessel Azimuth

During the 1st 10-Year Reactor Vessel examination in 1984, an indication was reported in the reactor vessel lower shell course located in Circumferential Weld RPVC3, approximately 240 inches below the vessel flange at vessel azimuth 345 degrees. Based on the detected dimensions, the indication could not be found to be acceptable per ASME Section XI Table IWB-3510-1. While the NRC in their October 16, 1984 safety evaluation concurred that the size of the indication was acceptable for plant operation, they required an augmented inspection program for the reactor vessel, which was incorporated into the IP2 Technical Specifications. This indication was re-inspected in 1987, where the indication characterization (size, orientation, reflector type) was established by the use of more advanced sizing techniques. This indication was conservatively bounded by a length of 3.9 inches, a through-wall dimension of 0.37 inches and a ligament distance (metal between the flaw bottom edge and the outside diameter surface) of 0.23 inches. Based on these dimensions, the indication was found to be acceptable per ASME Section XI Table IWB-3510-1. In the safety evaluation dated July 12, 1988, the NRC concluded that the required augmented inspection could be discontinued (Reference 7). Furthermore, it was concluded that the indication was most likely a subsurface welding inclusion that has existed unchanged since the vessel was fabricated.

During the 2nd 10-year reactor vessel examination in 1995, the indication located at approximately 240 inches below the vessel flange at vessel azimuth 345 degrees was of particular interest. While there were some variations in the examination results due to differences in the examination system and examination methodology from the prior exams, the indication characteristics as noted in the 1984, 1987, and 1995 examinations remained the same. The indication is embedded and its dimensions were compared to the acceptance criteria of ASME Section XI Table IWB-3510-1 and found to be acceptable.

5.2 Fleet wide Reactor Vessel Inservice Inspection History

As part of the technical basis for ASME Code Case N-691, "Application of Risk-Informed Insights to Increase the Inspection Interval for Pressurized Water Reactor Vessels," (Reference 3) a survey of reactor vessel ISI history for 14 pressurized water reactors was performed. These 14 plants represented 301 total years of service and included reactor vessels fabricated by various vendors. The plants surveyed reported that no reportable findings had been discovered during examinations of their reactor vessel Category B-A and B-D welds of their reactor vessels.

It is widely recognized in the fracture mechanics community that fatigue crack growth of embedded flaws is substantially smaller than that of surface breaking flaws. Surface breaking flaws in the reactor vessel cladding are typically a result of lack of fusion defects between bands of cladding. In studies performed by Pacific Northwest National Laboratory for the NRC Pressurized Thermal Shock (PTS) Risk Reevaluation (Reference 8), it was determined that in plants with multi-pass cladding, for a flaw to exist through the cladding, two flaws would have to be aligned on top of one another. The probability of this occurring is very low (<0.0001). The

IP2 reactor vessel is constructed with multi-pass cladding and therefore has a low probability of containing through-cladding surface-breaking flaws.

All PWR plants have performed their 1st 10-Yr ISI for the subject examinations and many have completed their 2nd, or 3rd 10-Yr ISI. No surface-breaking or unacceptable near-surface flaws have been found in any of these inspections performed per the requirements of Regulatory Guide 1.150 (Reference 6) or ASME Section XI Appendix VIII.

5.3 Degradation Mechanisms in the Reactor Vessel

The welds for which the subject examinations are conducted are similar metal low alloy steel welds. The only currently known degradation mechanism for this type of weld is fatigue due to thermal and mechanical cycling from operational transients. Studies have shown that while flaw growth of simulated flaws in a reactor vessel would be small, the operational transient, which has the greatest contribution to flaw growth, is the cooldown transient. The cooldown transient is a low frequency transient and is not expected to occur more than a few instances during the requested inspection extension period. Therefore, any flaw growth during the requested deferral period will be inherently small.

With regard to the indication in RPVC-3 discussed in Section 5.1, flaw growth is very unlikely. This indication is embedded close to the surface of the outside diameter. As discussed in Section 5.2 above, fatigue crack growth of embedded flaws is significantly less than that of surface flaws. However, of greater significance for the subject indication is its location of 0.23 inches from the outside diameter of the reactor vessel. Given that cooldown transients do not cause tensile stresses on the outside diameter of the reactor vessel, growth of the indication due to cooldown transient is improbable. Furthermore, flaws on or near the outside diameter of the reactor vessel present a minimal challenge to structural integrity in PTS scenarios in contrast to inside diameter flaws. It is possible for an outside diameter flaw to experience flaw growth during a heatup scenario. However, due to system limitations, and the Technical Specifications that limit heatup rates to a maximum of 100°F/hour, the potential for growth of the indication during a heatup is small. Therefore, given the location of this indication, it does not present a significant challenge to structural integrity and it is highly unlikely that this indication would experience enough growth during the requested interval extension for it to exceed the Table IWB-3510-1 acceptance standards.

The fatigue usage factors for the welds in the subject examinations are much less than the ASME Code design limit of 1.0 after 40 years of operation. These usage factors are calculated using a very conservative design duty cycle. It is very unlikely that more than a few of these events (e.g. heatup or cooldown) would actually occur during the extension period of this proposed alternative.

It is important to note that this request does not apply to any dissimilar metal welds, including Alloy 600 base-metal or Alloy 82/182 weld material where primary water stress corrosion cracking (PWSCC) is a concern.

5.4 Material Condition of the Reactor Vessel Relative to Embrittlement

The reactor vessel beltline is the limiting area in terms of embrittlement for the subject examinations. The composition of each material in the reactor vessel beltline, along with the fluence and embrittlement data, can be found in the NRC Reactor Vessel Integrity Database (RVID) (Reference 9). Prior to the Stretch Power Uprate (SPU) license amendment request, this information was re-evaluated for IP2 considering the fluences associated with the SPU in WCAP-15629, Revision 1 (Reference 10) and was approved by the NRC staff (Reference 11). In support of the SPU license amendment request, additional analyses were performed to incorporate the actual thermal and power history data associated with the operating cycles after the effort supporting WCAP-15629, Revision 1 was completed. The NRC staff reviewed this information and concluded in Reference 11 that the effect of the additional operating data was negligible. This information is provided for IP2 in Attachment 2.

10CFR50.61 currently provides Pressurized Thermal Shock (PTS) screening criteria of RTPTS equal to 270°F for plates and axial welds and RTPTS equal to 300°F for circumferential welds. For IP2, the intermediate shell plate (B-2002-3) is the limiting material and its RTPTS value at 32 EFPY is well below the current PTS screening criteria (See Attachment 2). Furthermore, it is recognized by the NRC and industry that a large amount of conservatism exists in the current PTS screening criteria. In the NRC PTS Risk Re-Evaluation, results have shown that it may be possible to remove an amount of conservatism equivalent to reducing a plant's RTPTS value by at least 70°F. While the exact amount of conservatism that will be removed has not been determined, it is clear that IP2 will be below the current PTS screening criteria during the extension period and further below the potential revised PTS screening criteria.

5.5 Operational Experience Relative to RV Structural Integrity Challenging Events

It is widely recognized that the greatest possible challenge to reactor pressure vessel integrity for a pressurized water reactor (PWR) is pressurized thermal shock (PTS). A PTS event can be generally described as a rapid cooling of the reactor vessel followed by a late repressurization. Plants have taken steps such as implementing emergency operating procedures and operator training to lower the likelihood of a PTS event occurring. Due to the implementation of such measures, the number of occurrences of PTS events fleetwide is very small. When considered over the combined fleetwide PWR operating history, the frequency of PTS events is very small. When considering the frequency of PTS events and the length of the requested extension, the probability of a PTS event occurring during the requested extension is also very low. Combining the low probability of a PTS event with the low probability of an unacceptable flaw existing in the reactor vessel (given the previously discussed inspection history), the probability of reactor vessel failure due to PTS is also very small.

Entergy has implemented emergency operating procedures and operator training to prevent the occurrence of PTS events. IP2 Emergency Procedure, FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition," provides actions to avoid, or limit thermal shock or pressurized thermal shock to the reactor pressure vessel, or overpressure conditions at low temperature. The IP2 Control Room Operators systematically get trained on the Training Simulator on EOP Emergency Procedure, FR-P.1 at least once every two years.

Entergy has not performed an analysis in accordance with Regulatory Guide 1.154 (Reference 12). IP2 minimizes the amount of neutron fluence accumulated at the RV beltline using a low leakage core, to keep the RV below the PTS screening criterion, obviating the need to perform this analysis.

Summary

The current requirements for inspection of reactor vessel pressure-containing welds is the 1989 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The industry has expended significant cost and man-rem exposure that have shown no service-induced flaws in the reactor vessel (RV) for ASME Section XI Category B-A and B-D RV welds. ASME Section XI Code Case N-691 and industry efforts have shown that risk-insights can be used to extend the reactor vessel inservice inspection interval from 10 to 20 years. This extension satisfies the change in risk requirements of Regulatory Guide 1.174 and, in accordance with 10CFR 50.55a(a)(3)(i) maintains an acceptable level of quality and safety. Based on these efforts having shown that the risk of vessel failure with a 10-year inspection interval extension is low and achieves an acceptable level of quality and safety, it is reasonable to conclude that a one refueling cycle extension will also achieve an acceptable level of quality and safety. Furthermore, items 5.1 through 5.5 of Section 5 provide a qualitative basis that the risk associated with extending the inspection interval by one refueling cycle is small. Therefore, Entergy considers the proposed alternative for the subject examinations at IP2 to provide an acceptable level of quality and safety in accordance with 10CFR 50.55a(a)(3)(i).

6.0 Duration of Proposed Alternative

This request is applicable to Entergy's ISI Program for the Third Ten-Year Interval for IP2.

7.0 References

1. WCAP-16168-NP, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," October 2003.

2. NRC to WOG, "WOG Request for the Staff Review of Topical Report WCAP-16168-NP Risk-Informed Extension of Reactor Vessel In-Service Inspection Intervals," August 18, 2004.

3. ASME Boiler and Pressure Vessel Code, Code Case N-691, "Application of Risk-Informed Insights to Increase the Inspection Interval for Pressurized Water Reactor Vessels," Section XI, Division 1, November 2003.

4. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated November 2002.

5. R. Gramm of the NRC to G. Bischoff of the WOG, "Summary of teleconference with the Westinghouse Owners Group regarding potential one cycle relief of reactor pressure vessel shell weld inspections at pressurized water reactors related to WCAP-16168-NP,

"Risk-Informed Extension of Reactor Vessel In-Service Inspection Intervals"," January 27, 2005.

6. Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Pre-Service and Inservice Examinations," dated February 1983.

7. "NRC Safety Evaluation by the Office of Nuclear Reactor Regulation, Evaluation of the Flaw Indication Detected in the Reactor Pressure Vessel, Consolidated Edison Company, Indian Point Unit 2, Docket No, 50-247," July 12, 1988. (TAC No 64457)

8. NRC Memorandum, Thadani to Collins, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule (10CFR50.61)," December 31, 2002.

9. Nuclear Regulatory Commission Reactor Vessel Integrity Database, Version 2.0.1, July 6, 2000.

10. WCAP-15629, Revision 1, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," December 2001.

11. "NRC Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 241 to Facility Operating License No. DPR-26, Entergy Nuclear Operations Inc., Indian Point Nuclear Generating Unit No. 2, Docket No. 50-247," October 27, 2004. (TAC NO. MC1865)

12. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," dated January 1987.

Attachment 1 TO NL-05-072

Indian Point Unit 2 Last RV 10-Year Inspection Report Summary

(2 Pages)

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

ATTACHMENT 1

Indian Point Unit 2 Last RV 10-Year Inspection Report Summary

Item No.	ASME CAT.	ASME ITEM	Weld ID	Description	Date Last Inspected (Note 1)	Percent Coverage Obtained	Number of Reportable Indications	Number of Indications Currently Being Monitored	Growth of Indications Currently Being Monitored (in)
1	B-A	B1.11	RPVC2	RV CIRC SHELL WELD	March, 1995	100.0	0	0	0
2	B-A	B1.11	RPVC3	RV CIRC SHELL WELD	March, 1995	100.0	0 (Note 2)	0	0
3	B-A	B1.11	RPVC4	RV CIRC SHELL WELD	March, 1995	91.0	0	0	0
4	B-A	B1.12	RPVL1	RV Longitudinal Weld @007	March, 1995	90.3	0	0	0
5	B-A	B1.12	RPVL2	RV Longitudinal Weld @127	March, 1995	91.8	0	0	0
6	B-A	B1.12	RPVL3	RV Longitudinal Weld @247	March, 1995	91.6	0	0	0
7	B-A	B1.12	RPVL4	RV Longitudinal Weld @060	March, 1995	100.0	0	0	0
8	B-A	B1.12	RPVL5	RV Longitudinal Weld @180	March, 1995	100.0	0	0	0
9	B-A	B1.12	RPVL6	RV Longitudinal Weld @300	March, 1995	100.0	0	0	0
10	B-A	B1.12	RPVL7	RV Longitudinal Weld @165	March, 1995	95.0	0	0	0
11	B-A	B1.12	RPVL8	RV Longitudinal Weld @345	March, 1995	95.0	0	0	0
13	B-A	B1.22	RPVM1	RV Lower Head MW @150	March, 1995	56.0	0	0	0
14	B-A	B1.22	RPVM2	RV Lower Head MW @090	March, 1995	56.0	0	0	0
15	B-A	B1.22	RPVM3	RV Lower Head MW @030	March, 1995	56.0	0	0	0
16	B-A	B1.22	RPVM4	RV Lower Head MW @330	March, 1995	56.0	0	0	0
17	B-A	B1.22	RPVM5	RV Lower Head MW @270	March, 1995	56.0	0	0	0
18	B-A	B1.22	RPVM6	RV Lower Head MW @210	March, 1995	56.0	0	0	0
19	B-A	B1.30	RPVC1	RV Shell to Flange Circ Weld	March, 1995	100.0	0	0	0
20	B-D	B3.90	RPVN1	Nozzle-to-Vessel Weld RO@022	March, 1995	100.0	0	0	0
21	B-D	B3.90	RPVN2	Nozzle-to-Vessel Weld RI@067	March, 1995	100.0	0	0	0
22	B-D	B3.90	RPVN3	Nozzle-to-Vessel Weld RI@113	March, 1995	100.0	0	0	0
23	B-D	B3.90	RPVN4	Nozzle-to-Vessel Weld RO@158	March, 1995	100.0	0	0	0
24	B-D	B3.90	RPVN5	Nozzle-to-Vessel Weld RO@202	March, 1995	100.0	0	0	0
25	B-D	B3.90	RPVN6	Nozzle-to-Vessel Weld RI@247	March, 1995	100.0	0	0	0
26	B-D	B3.90	RPVN7	Nozzle-to-Vessel Weld RI@293	March, 1995	100.0	0	0	0

ATTACHMENT 1
Indian Point Unit 2 Last RV 10-Year Inspection Report Summary

Item No.	ASME CAT.	ASME ITEM	Weld ID	Description	Date Last Inspected (Note 1)	Percent Coverage Obtained	Number of Reportable Indications	Number of Indications Currently Being Monitored	Growth of Indications Currently Being Monitored (in)
27	B-D	B3.90	RPVN8	Nozzle-to-Vessel Weld RO@338	March, 1995	100.0	0	0	0
28	B-D	B3.100	RPVN1(IR)	RV Noz Inside Rad RO@022	March, 1995	100.0	0	0	0
29	B-D	B3.100	RPVN2(IR)	RV Noz Inside Rad RI@067	March, 1995	100.0	0	0	0
30	B-D	B3.100	RPVN3(IR)	RV Noz Inside Rad RI@113	March, 1995	100.0	0	0	0
31	B-D	B3.100	RPVN4(IR)	RV Noz Inside Rad RO@158	March, 1995	100.0	0	0	0
32	B-D	B3.100	RPVN5(IR)	RV Noz Inside Rad RO@202	March, 1995	100.0	0	0	0
33	B-D	B3.100	RPVN6(IR)	RV Noz Inside Rad RI@247	March, 1995	100.0	0	0	0
34	B-D	B3.100	RPVN7(IR)	RV Noz Inside Rad RI@293	March, 1995	100.0	0	0	0
35	B-D	B3.100	RPVN8(IR)	RV Noz Inside Rad RO@338	March, 1995	100.0	0	0	0

Notes: (1) Due to improvements in inspection technology, the most recent inspection is considered to be of the greatest quality of the inspections performed. In some instances, indications were found during inspections and then, in later inspections with improved equipment, were determined to be reflections rather than indications. Therefore, the inspection data provided in this table is for the most recent inservice inspection.

(2) During the 1st 10-Yr Reactor Vessel examination in 1984, an indication was reported in the vessel shell located in RPVC3, approximately 240 inches below the vessel flange at vessel azimuth 345 degrees. As sized in 1984 this indication was unacceptable per ASME Section XI Table IWB-3510-1. This indication has since been re-inspected in 1987 and 1995 using more advanced inspection techniques and has been found acceptable per Table IWB-3510-1. (NRC SER TAC NO. 64457)

Attachment 2 TO NL-05-072

Indian Point Unit 2 Reactor Vessel Materials Information for EOL (32 EFPY)

(1 Page)

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

ATTACHMENT 2
Indian Point Unit 2 Reactor Vessel Materials Information for EOL (32 EFPY)

Major Material Region Description				Cu (wt%)	Ni (wt%)	RT_{NDT} (Unirradiated)	Chemistry Factor (Note 1)	Margin (°F)	RT_{PTS} (°F) (Note 3)	USE (ft-lb) (Note 2)
Item No.	Type	ID	Shell Location							
1	Plate	B-2002-1	Intermediate	0.19	0.65	34 °F	144°F (Pos 1.1)	34 °F	222 °F	56
	Plate	B-2002-1	Intermediate	0.19	0.65	34 °F	114 °F (Pos 2.1)	17 °F	173 °F	56
2	Plate	B-2002-2	Intermediate	0.17	0.46	21 °F	115.1 °F (Pos 1.1)	34 °F	178 °F	58
	Plate	B-2002-2	Intermediate	0.17	0.46	21 °F	118.2 °F (Pos 2.1)	34 °F	182 °F	58
3	Plate	B-2002-3	Intermediate	0.25	0.60	21 °F	176 °F (Pos 1.1)	34 °F	244 °F	50.3
	Plate	B-2002-3	Intermediate	0.25	0.60	21 °F	181.9 °F (Pos 2.1)	17 °F	233 °F	50.3
4	Plate	B-2003-1	Lower	0.20	0.66	20 °F	152 °F (Pos 1.1)	34 °F	217 °F	52
5	Plate	B-2003-2	Lower	0.19	0.60	-20 °F	142 °F (Pos 1.1)	34 °F	166 °F	61
6	Axial Weld	(Heat # W5214)	Intermediate & Lower	0.21	1.01	-56 °F	230.2 °F(Pos 1.1)	65.5 °F	231 °F	69
	Axial Weld	(Heat # W5214)	Intermediate & Lower	0.21	1.01	-56 °F	254.7 °F (Pos 2.1)	44 °F	233 °F	69
7	Circumferential Weld	(Heat # 34B009)	Intermediate & Lower	0.19	1.01	-56 °F	220.9 °F (Pos 1.1)	65.5 °F	246 °F	56

Notes: 1. "Pos 1.1" and "Pos 2.1" refer to positions in Reg Guide 1.99 Rev. 2. Whenever data was available, RT-PTS was calculated using both positions.

2. All upper shelf energies are above 50 ft-lb at EOL.

3. All RT-PTS values are significantly below the Reg Guide 1.99 Rev. 2 screening criteria values of 270 °F (plates and axial welds) and 300 °F (circumferential welds) at EOL.

Cu Copper

Ni Nickel

RT_{NDT} Reference Temp., Nil-Ductility Transition

wt% Weight by percentage

RT_{PTS} Reference Temp., Pressurized Thermal Shock

EOL End Of Life