Entergy

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Fred Dacimo
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January 18, 2005

Re: Indian Point Unit 3
Docket No. 50-286
NL-05-010

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Stop O-P1-17
Washington, DC 20555-0001

Subject: **Reply to RAI regarding Indian Point Unit 3 Relaxation Requests for Inspection of Reactor Pressure Vessel Head per NRC First Revised Order, EA-03-009 (TAC No. MB3195)**

References:

1. NRC letter dated July 29, 2004, "Request for Additional Information Regarding Requests for Relaxation from Revised Order on Reactor Vessel Nozzles, Indian Point Generating Unit Nos. 2 and 3 (TAC Nos. MC3194 and MB3195)".
2. Entergy letter to NRC (NL-04-060), "Relaxation Requests for Inspection of Reactor Pressure Vessel Head", dated May 19, 2004.
3. Entergy letter to NRC (NL-04-104), "Reply to RAI regarding Relaxation Requests for Inspection of Reactor Pressure Vessel Head per NRC First Revised Order, EA-03-009", dated August 23, 2004.
4. NRC letter dated October 15, 2004, "Relaxation of First Revised Order on Reactor Vessel Nozzles, Indian Point Nuclear Generating Unit No. 2 (TAC No. MC3194)".
5. Issuance of First Revised NRC Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors, dated February 20, 2004.

Dear Sir;

Entergy Nuclear Operations, Inc. (Entergy) is providing a response to the NRC request for additional information (RAI) in Reference 1 regarding proposed relaxation requests (Reference 2) for Indian Point 3 (IP3).

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This letter addresses the additional information requested regarding Attachments 1 and 3 of Reference 1. The response to questions on Attachment 2 of Reference 1 was previously provided by Reference 3 (for both Indian Point Units 2 and 3, IP2 and IP3). The relaxation request for IP2 was approved on October 15, 2004 (Reference 4). The NRC indicated in the October 15th letter that the relaxation requests (3 total) for IP3 will be evaluated in a separate letter.

Entergy requests approval of the three (3) relaxation requests for IP3, as initially submitted in Reference 2, and as supplemented in Reference 3 and this letter, by February 18, 2005 to support the IP3 Spring Outage 3R13.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. Anders Eng at 914-272-3523.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

Attachment 1 (Reply To NRC Request For Additional Information Regarding Relaxation Requests For Inspection Of IP3 Reactor Pressure Vessel Head)

cc: next page

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ATTACHMENT 1 TO NL-05-010

REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING RELAXATION REQUESTS FOR INSPECTION

OF IP3 REACTOR PRESSURE VESSEL HEAD

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

Entergy's Reply to Request for Additional Information from NRC Letter dated July 29, 2004 (Accession Number ML042110166)

A. Questions regarding Entergy Letter NL-04-060, Attachment 1 (dated May 19, 2004):

Question 1:

Provide a summary of the methodology and results for the stress analysis performed for the 5 RPV penetration nozzles identified in the relaxation request. The results should include the specific operating stress levels for the uphill and downhill sides of the nozzles and the angle of each nozzle relative to the upper surface of the RPV head.

Response:

Extensive analyses were done for the Indian Point 2 (IP2) RPV top head. Finite element analyses were performed for the IP2 top head configuration by Dominion Engineering/Structural Integrity Associates. These calculations addressed weld residual stresses and applied stresses for the limiting nozzles at IP2, under several repair scenarios. Entergy/Structural Integrity Associates has compared the geometries and operating conditions of Indian Point 3 (IP3) to those analyzed for IP2, and have determined that the analyzed results for IP2 are applicable and bounding for IP3. This conclusion is based on the fact that head geometries for IP2 are similar to IP3's and in some cases even more limiting than the corresponding IP3 geometries for the top head penetrations. In particular, the head geometries for the two units are essentially identical in major dimensions, (i.e., head thickness, diameter, and CRDM tube diameter and wall thickness). The materials for both reactor heads are the same. The weld joint geometries are slightly more severe for IP2, in that the J-groove weld leg for IP2 is typically slightly larger and therefore has more volume than the IP3's at the limiting nozzle locations. Since weld residual stress is driven to a great extent by the volume of weld metal through the cooling process, residual stresses for the IP3 welds are expected to be comparable to, if not less than, those at IP2. This conclusion is valid for the original as-welded J-groove welds.

The stress analysis performed for IP2 to identify the limiting configuration nozzle locations could be used to identify the IP3 limiting configuration nozzle locations (i.e., nozzles #74, 75, 76, 77, 78). These nozzles are considered as limiting configuration locations because previous experience with CRDM penetration welding residual stress analyses for IP2 has established two general trends: (1) predicted welding residual stresses in the nozzle tend to increase for nozzles that are further from the center of the head, and (2) predicted welding residual stresses in the nozzle tend to increase with increasing nozzle yield strength. In order to conservatively bound both of these effects, the most peripheral nozzle geometry in the IP2/IP3 heads (48.8° angle of incidence between the nozzle centerline and the head inside surface) was modeled and a nozzle yield strength of 63.0 ksi (highest reported yield strength for any IP2 and IP3 nozzles) was used. The angle of incidence between the nozzle centerline and the head inside surface is essentially the same for both the IP2 and IP3 outer-most nozzles. For IP3, nozzles #74, 75, 76, 77, and 78 are the outer-most nozzles and are expected to have the highest stress levels.

The finite element analysis includes the simulation of the weld deposition for the J-groove weld buttering and the subsequent stress relief of the head and buttering, in addition to the J-groove welding. The results of this analysis provided the operating condition stress levels and distribution effects of welding residual stresses, hydrostatic testing, and steady state operating pressure and temperature. Hoop stress results of the performed analysis of original welding plus operating loads for the 5 RPV penetration nozzles (#74, 75, 76, 77, and 78) are summarized as follows:

Nozzle No's:	Operating Stress (PSI):	Location:
74, 75, 76, 77, and 78	9,786	(*) Downhill side
74, 75, 76, 77, and 78	-1,170	(**) Uphill side

Notes:

(*) Downhill side at 0.96" below the lowest point at the toe of the J-groove weld.
(**) Uphill side at 5.476" below the lowest point at the toe of the J-groove weld.

Question 2:

Provide a summary of the crack growth calculation describing the methodology used, including the input assumptions and results. Discuss whether the crack growth rates assessed were based on the equations in MRP-55 and whether the crack growth evaluation was based on the as-built weld geometry.

Response:

A crack growth evaluation was performed using the methods of MRP-55 for crack growth caused by primary water stress corrosion cracking at a head temperature of 592 degrees F. The nozzle is an open-ended tube so that the operating pressure and temperature are essentially the same at both the inside and outside surface of the tube. For purposes of this analysis, a conservatively high hoop stress of 30 ksi was assumed, as compared to the calculated stresses of less than 20 ksi as noted above (Reference 5, First Revised Order EA-03-009, page 10). An initial hypothetical flaw is assumed to exist with a worst-case orientation (axial) and flaw growth occurs in a single direction toward the J-groove weld. Because the location of the hypothetical flaw is in the threaded region, at least 0.96 inches from the J-groove weld, the weld residual stresses are also negligible. The evaluation confirmed that the hypothetical flaw would not grow to the J-groove weld over at least 4 years of operation. The weld geometry dimensions are taken from the IP3 plant specific drawing (IP3V-0439-1680) entitled: "Indian Point Unit 3, RVH Penetration Inspection, Interface Dimensions". Refer to the attached Figure 1 for the location of these dimensions on the penetration nozzles.

Question 3a:

For the five nozzles, are there funnels threaded and pinned to the bottom of the nozzles? Are there guide/thermal sleeves installed inside the five penetration nozzles? Discuss the hardship that would be encountered to remove the funnels (dose, safety risks, etc.).

Response:

These 5 nozzles (nos. 74, 75, 76, 77, and 78) are equipped with retaining collar/guide funnels which are welded to the bottom of the nozzles on the OD side. Removal of the guide funnels would require use of the Electrical Discharge Machining (EDM) process. It would take approximately 24 hours to set up the equipment, and approximately 22 hours to cut out all 5 thermal sleeve/guide funnels and re-install. The replacement guide funnels would have to be fabricated to match the as-built configuration of the nozzle since these are not the standard size. After the inspections, the replacement guide funnels would be installed by welding. It is estimated that the total radiation exposure for the removal and reinstallation activities would be about 6 rem.

Since IP3 intends to perform the non-visual NDE examinations with a combination UT and ECT probe from the inside of the nozzle tube (a single probe assembly that contained a pair of transducers for the UT examination and an eddy current coil for the ECT examination), removal of these OD guide funnels will not increase the NDE examination coverage or accuracy. Furthermore, removal of these guide funnels, as identified above, is a labor intensive activity and the additional stay-times by personnel in the radiation field required to perform the removal and reinstallation tasks represent a hardship without a compensating increase in the effectiveness of the non-visual NDE examination.

Question 3b:

The First Revised Order allows either ultrasonic testing (UT) examination or a surface examination (i.e. liquid penetrant or eddy current) to be performed. Discuss why a surface examination is not being considered for the threaded area of the five penetration nozzles identified. A surface examination would provide meaningful results on the threaded regions. Discuss any hardships that would be imposed by performing the surface examination (dose, safety risks etc.).

Response:

The inspection methods and results performed during Refueling Outage 3R12 in April 2003 were submitted in Entergy's letter to NRC, NL-03-098, dated June 12, 2003. The non-visual NDE portion of the examination for the referenced five penetration nozzles was performed using a single probe assembly that contained a pair of transducers for the UT examination and an eddy current coil for the ECT surface examination. The axial coverage with this probe assembly extended down to approximately 0.75 inches from the bottom of the nozzle (for the UT exam) and down to approximately 0.25 inches from the bottom of the nozzle (for the ECT surface exam). Meaningful UT data below approximately 0.75 inches was limited by signal dispersion in the threaded region. Meaningful ECT data below approximately 0.25 inches was also limited because the eddy current coil tends to lose contact with the examination surface as it reaches the lead-in chamfer region (see attached Figure 1). As discussed in the response to Question 3a above, the hardships in removing the guide funnels welded to the bottom of the nozzles precludes examination of the OD wetted surface areas of the nozzle tubes.

Based on these NDE inspections and coupled with the remote bare metal visual (BMV) examination of the entire top surface of the vessel head performed in 3R12 in 2003, ENO concluded that there were no signs of reactor pressure vessel head degradation or primary water stress corrosion cracking of the Alloy 600 penetration nozzles.

Question 4:

The licensee stated that IP3 will remain in the moderate category during Refueling Outage (RFO) 3R13, scheduled for Spring 2005. In accordance with the information provided in Reactor Vessel Closure Head Penetration Safety Assessment (MRP-110), the effective degradation years (EDYs) calculated for RFO 3R13 will be over 12 years, which will put IP3 into the high susceptibility category. For those plants in a high susceptibility category, the inspection requirements for RPV head and head penetration nozzle inspections shall be performed using the techniques of paragraphs IV.C.(5)(a) and (b) of the First Revised Order.

Provide a comprehensive discussion of the site-specific calculations that support the statement that IP3 will remain in the moderate susceptibility category including differences between the site-specific calculations and those upon which the MRP-110 results are based.

Response:

In response to NRC Bulletin 2001-01, the industry elected to assign the average bulk fluid temperature in the upper head regions of the plants as the "head" temperature. These "head" temperatures were based on plant design parameters with the results scaled to attempt to adjust the data to reflect plant vessel inlet temperature data. Temperatures were recalculated for IP3, utilizing the <u>actual</u> plant specific operating conditions on a cycle-by-cycle and a within-a-cycle bases. The re-calculation took into account the actual plant power levels, vessel inlet temperatures, flow rates, and core power distributions within each cycle as opposed to the general design conditions.

The re-calculation was performed using the "THRIVE"™ computer code. The methodology within the THRIVE code for calculating the average bulk fluid temperature of the reactor vessel upper head region has been benchmarked against scale model and in-plant testing.

The basic methodology used to determine the revised IP3 EDY is the same methodology used to provide the results reflected in MRP-110, Table 4.2 and is consistent with the requirements as prescribed in the First Revised NRC Order EA-03-009. The only difference is that the revised EDY calculations provide a more detailed determination of the head temperatures using actual plant operations data.

Since the re-calculated average bulk fluid temperature of the upper head region is based on the actual IP3 operating conditions and history, it is a more accurate representation of the upper head region average bulk fluid temperature time history.

The revised IP3-specific calculations indicate that the IP3 reactor vessel head will be at 11.17 effective degradation years (EDY's) by March 11, 2005 (the scheduled start date of 3R13). Therefore, IP3 presently remains in the moderate susceptibility category per the criteria established in Section IV.B of the First Revised NRC Order EA-03-009.

B. Question regarding Entergy Letter NL-04-060, Attachment 2 (dated May 19, 2004):

Question 1:

The licensee stated that, although the reflective metal insulation support ring is removable, the other components of the insulation package supported by the support ring and the control rod drive mechanism cooling shroud would have to be removed first to achieve a 100 percent bare metal visual coverage of the RPV head.

Provide a detailed discussion that identifies the difficulties in removing the insulation, including the other components that would need to be removed. Provide sketches to show all the components that need to be removed in order to achieve a 100 percent bare metal coverage. Discuss any unique challenges posed by the removal of these components.

Response:

The response to this question was previously provided in Entergy letter NL-04-104, dated August 23, 2004 (Reference 3). The response provided was applicable to both IP2 and IP3. The relaxation request for IP2 was approved on October 15, 2004. The relaxation request for IP3, as submitted in Attachment 2 of the May 19, 2004 letter (Reference 2) and as supplemented in Reference 3, is identical to the approved IP2 submittal.

C. Questions regarding Entergy Letter NL-04-060, Attachment 3 (dated May 19, 2004):

Question 1:

Provide a similar discussion regarding susceptibility as requested in question 4 for Attachment 1.

Response:

In response to NRC Bulletin 2001-01, the industry elected to assign the average bulk fluid temperature in the upper head regions of the plants as the "head" temperature. These "head" temperatures were based on plant design parameters with the results scaled to attempt to adjust the data to reflect plant vessel inlet temperature data. Temperatures were recalculated for IP3, utilizing the actual plant specific operating conditions on a cycle-by-cycle and a within-a-cycle bases. The re-calculation took into account the actual plant power levels, vessel inlet temperatures, flow rates, and core power distributions within each cycle as opposed to the general design conditions.

The re-calculation was performed using the "THRIVE"™ computer code. The methodology within the THRIVE code for calculating the average bulk fluid temperature of the reactor vessel upper head region has been benchmarked against scale model and in-plant testing.

The basic methodology used to determine the revised IP3 EDY is the same methodology used to provide the results reflected in MRP-110, Table 4.2 and is consistent with the requirements as prescribed in the First Revised NRC Order EA-03-009. The only difference is that the revised EDY calculations provide a more detailed determination of the head temperatures using actual plant operations data.

Since the re-calculated average bulk fluid temperature of the upper head region is based on the actual IP3 operating conditions and history, it is a more accurate representation of the upper head region average bulk fluid temperature time history.

The revised IP3-specific calculations indicate that the IP3 reactor vessel head will at 11.17 effective degradation years (EDY's) by March 11, 2005 (the scheduled start date of 3R13). Therefore, IP3 presently remains in the moderate susceptibility category per the criteria established in section IV.B of the First Revised NRC Order EA-03-009.

Question 2:

Provide the head temperature for IP3 and the details as to how and where the head temperature is calculated/and or measured. Provide a discussion of the differences between the methodology used to determine the IP3 EDY and the methodology used to provide the results reflected in MRP-110. Provide the basis regarding why the licensee's calculations are a more accurate (or conservative) representation of RPV head temperature and EDY than the results presented in MRP-110.

Response:

Same response and discussions as provided in response to Question C.1 above.

References:

1. NRC letter dated July 29, 2004, "Request for Additional Information Regarding Requests for Relaxation from Revised Order on Reactor Vessel Nozzles, Indian Point Generating Unit Nos. 2 and 3 (TAC Nos. MC3194 and MB3195)".

2. Entergy letter to NRC (NL-04-060), "Relaxation Requests for Inspection of Reactor Pressure Vessel Head", dated May 19, 2004.

3. Entergy letter to NRC (NL-04-104), "Reply to RAI regarding Relaxation Requests for Inspection of Reactor Pressure Vessel Head per NRC First Revised Order, EA-03-009", dated August 23, 2004.

4. NRC letter dated October 15, 2004, "Relaxation of First Revised Order on Reactor Vessel Nozzles, Indian Point Nuclear Generating Unit No. 2 (TAC No. MC3194)"

5. Issuance of First Revised NRC Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors, dated February 20, 2004

Attachment:

Figure 1. IP3 Penetration Nozzles Nos. 74 through 78 (TYP.) – 1 page

Attachment I to NL-05-010
Docket No. 50-286
Figure 1

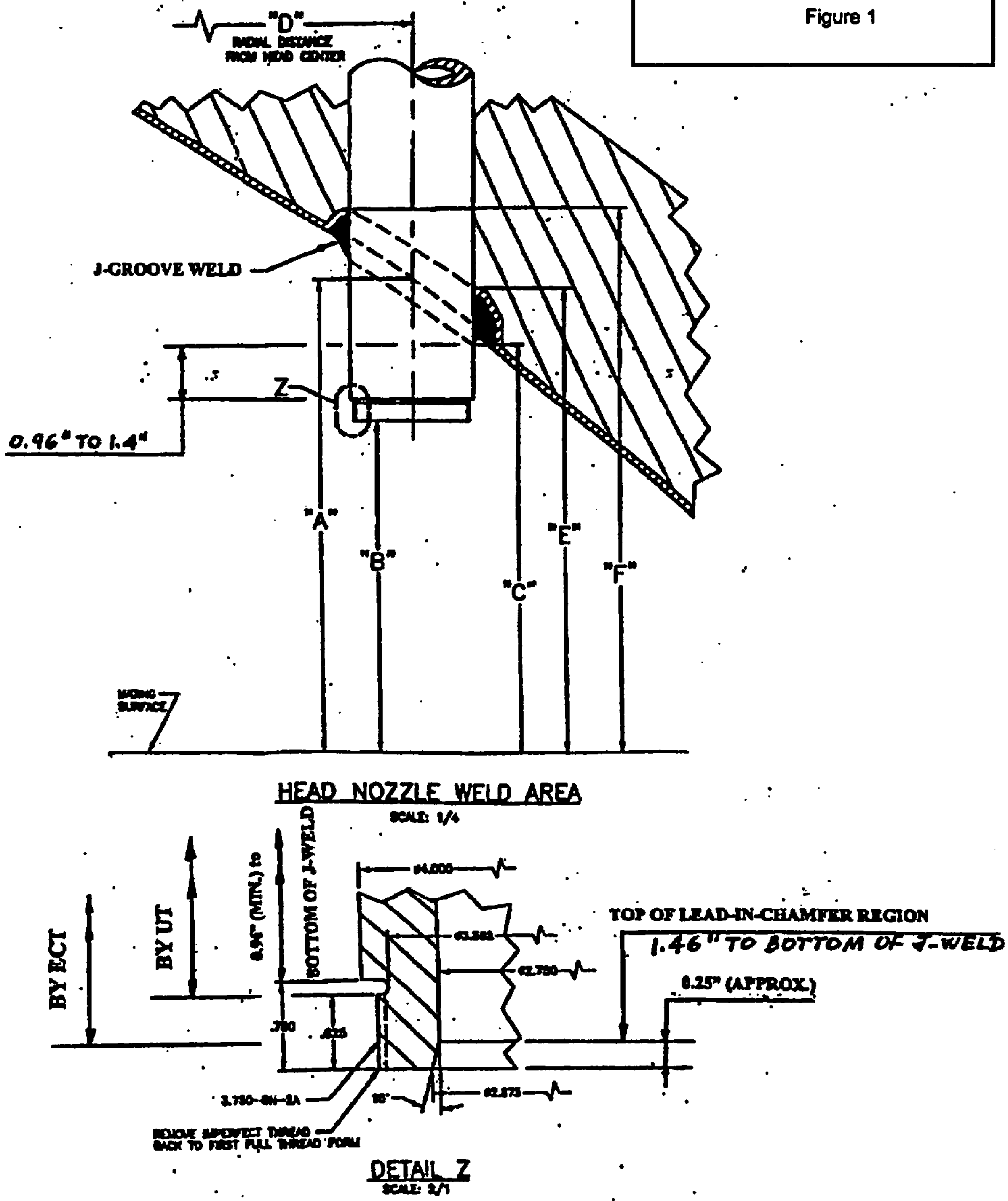

FIGURE 1
IP3 - PENETRATION NOZZLES
NOS. 74 THROUGH 78 (TYP.)