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John T. Herron
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September 11, 2002

Re: Indian Point Unit No. 2
Docket No. 50-247
NL-02-119

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

SUBJECT: **30-Day Response to NRC Bulletin 2002-02**

References:

1. NRC Bulletin 2002-02; "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs," dated August 9, 2002.
2. ENO letter to NRC, NL-02-101; "Reactor Pressure Vessel Head Penetration Nozzle Inspection Plan for 2002 Refueling Outage (2R15)," dated July 23, 2002.
3. EPRI Report 1006284, "PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48)," dated August 2001.

Dear Sir:

On August 9, 2002, the Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2002-02 (Reference 1), which requests licensees to summarize nondestructive examination (NDE) techniques that will be used to supplement current inspection programs or provide justification for continued reliance on visual examinations of the reactor pressure vessel (RPV) head and control rod drive mechanism (CRDM) penetration nozzles. Entergy Nuclear Operations, Inc (ENO) previously submitted (Reference 2) the inspection plans for the next refueling outage at Indian Point 2 (IP2), scheduled to being in October 2002. ENO has reviewed the information provided in Bulletin 2002-02 and concludes that the proposed inspection plan remains the appropriate course of action to demonstrate compliance with applicable regulations. The inspection plan consists of bare metal visual (BMV) examination and includes provisions for supplementing the inspection with ultrasonic NDE. The scope of coverage for the supplemental NDE portion of the inspection will be based on the outage duration. Additional information regarding the inspection plan is provided in Attachment I.

ENO has evaluated the expected status of IP2 with regard to accrued Effective Full Power Years (EFPY) and Effective Degradation Years (EDY) in accordance with Reference 3 with the following results:

Outage Date:	October 26, 2002
Accrued EFPY:	18.67 years
Calculated EDY:	7.998 years

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Although the calculated EDY is less than 8, ENO is implementing 100% BMV examination at the next refueling outage (2R15) which is consistent with the example recommendations in the Bulletin for the '8 to 12 year' EDY category. The example recommendation also identifies NDE techniques to be implemented beginning with the subsequent refueling outage (2R16 for IP2). However, ENO intends to implement supplemental NDE techniques beginning with the upcoming outage, 2R15, as stated in Reference 2 and summarized in Attachment I.

ENO is continuing to monitor industry experience regarding this issue and will use this information along with the results of the 2R15 inspection results to formulate inspection plans for future refueling outages. Attachment II provides additional information that addresses the six discussion points contained in the Bulletin regarding the need for supplementing visual examinations with NDE techniques.

No new commitments are being made in this letter. If you have any questions, please contact Mr. John McCann (914) 734-5074, Licensing Manager.

I declare under penalty of perjury that the foregoing is true and correct.

Very truly yours,

Executed on 9-11-02
(Date)

Mr. John Herron
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ATTACHMENT I TO NL-02-119

30-DAY RESPONSE TO NRC BULLETIN 2002-02
FOR INDIAN POINT UNIT 2

SUMMARY DISCUSSION OF SUPPLEMENTAL INSPECTIONS
TO BE IMPLEMENTED DURING REFUELING OUTAGE, 2R15

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

The following is a summary of the supplemental inspections that Entergy Nuclear Operations, Inc (ENO) plans to conduct during the next refueling outage at Indian Point 2 (IP2), scheduled to begin in October 2002. This information is consistent with the inspection plan previously submitted by ENO letter to NRC (NL-02-101) dated July 23, 2002.

A. EDY

The calculated EDY based on continued operation at full power until the next refueling outage scheduled to begin on October 26, 2002 is calculated to be 7.998 years.

B. METHOD

ENO intends to supplement the 100% bare metal visual (BMV) examination with ultrasonic testing of the vessel head penetrations (VHPs) and Control Rod Drive Mechanism (CRDM) nozzles. Note that 'supplemental inspection' refers to additional inspection of VHPs that are not already subject to more detailed inspection as a result of the bare metal visual examination. VHPs found to exhibit evidence of boric acid deposits consistent with primary coolant leakage from a through wall defect will be subject to additional inspections as previously described in NL-02-101.

C. SCOPE

The scope of the supplemental ultrasonic examination of VHPs will include a length of the associated Control Rod Drive Mechanism (CRDM) nozzle sufficient to span the J-groove weld as well as the upper and lower heat affected zones of the weld.

D. COVERAGE

The IP2 design has 97 VHPs / CRDMs and the bare metal visual examination will cover essentially 100% of the CRDM to Reactor Pressure Vessel head junctions. The coverage of the supplemental ultrasonic examination to be completed during 2R15 will be based on outage duration. The coverage completed will be reported by ENO in the inspection results summary due to be submitted within 30 days after plant restart. ENO is procuring and configuring the inspection equipment so that 100% coverage is feasible.

E. FREQUENCY

ENO is continuing to monitor industry experience regarding this issue and has not yet finalized specific plans regarding the appropriate frequency for inspections. Consistent with the example inspection table in Bulletin 2002-02, ENO currently believes that additional inspection in the subsequent refueling outage, 2R16, would not be required, depending on the results and actual completed coverage of the 2R15 inspection.

F. QUALIFICATION REQUIREMENTS

BMV examinations will be performed by VT-2 qualified personnel as required by ASME Section XI. In addition, ENO has contracted with Westinghouse to develop the inspection techniques and the associated tooling to complete ultrasonic inspections performed during the next refueling outage. Personnel, equipment and inspection techniques used during these ultrasonic inspections will be demonstrated prior to the refueling outage in EPRI samples specifically built for this purpose.

G. ACCEPTANCE CRITERIA

NDE results will be evaluated in accordance with the rules provided in Subsection IWB-3600 of the ASME Section XI Code. All flaws that exceed the requirements of IWB-3600 will be repaired in accordance with the requirements of ASME Section XI, and approved relief requests, prior to returning the plant to service.

ATTACHMENT II TO NL-02-119

30-DAY RESPONSE TO NRC BULLETIN 2002-02
FOR INDIAN POINT UNIT 2

ADDITIONAL INFORMATION REGARDING
BULLETIN 2002-02 DISCUSSION POINTS

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

Bulletin 2002-02 identified six concerns relative to the reliance on visual inspections as the primary inspection method. ENO provides the following responses as justification for continued reliance on visual examinations during the next refueling outage as the primary method to detect degradation in the RPV head. The visual examinations will be based on the MRP Inspection Plan (Refs 1, 2, and 3) that has been developed by PWR utilities to provide assurance that unacceptable wastage will not occur between refueling outages. The following responses address the reliability and effectiveness of visual examinations as they relate to the six concerns cited in Bulletin 2002-02.

Concern 1: Circumferential cracking of CRDM nozzles was identified by the presence of relatively small amounts of boric acid deposits. This finding increases the need for more effective visual and non-visual NDE inspection methods to detect the presence of degradation in CRDM nozzles before nozzle integrity is compromised.

Response: Since the initial discovery of circumferential cracks above the J-groove weld in 2001, visual inspection techniques and approaches employed have been dramatically improved and a heightened sense of awareness exists for the range in size and appearance of visual indications that must be further investigated. Non-visual techniques similarly have and continue to evolve to more effectively examine the penetration tube and associated welds for evidence of cracks. Nothing in the recent events at Davis-Besse has altered the fundamental inspection capability requirements previously established as necessary to identify the presence of primary water stress corrosion cracking (PWSCC) and subsequent associated wastage. The effectiveness of inspection techniques continues to be evaluated and improved.

EPRI MRP has published detailed guidance for performing visual examinations of RPV heads (Ref 3). RPV head bare metal visual inspections at Indian Point 2 will be performed and documented in accordance with written procedures and acceptance criteria that comply with the guidance of the MRP Inspection Plan. Evaluations and corrective actions will be in accordance with the plant corrective action program.

In order for outside diameter (OD) circumferential cracks above the J-groove weld to initiate and grow, a leak path must first be established to the CRDM annulus region from the inner wetted surface of the reactor pressure vessel (RPV) head. If primary water does not leak to the annulus, the environment does not exist to cause circumferential OD cracking. Axial cracks in the CRDM nozzles or cracks in J-groove welds must first initiate and grow through wall. Experience has shown that through wall axial cracks will result in observable leakage at the base of the penetration on the outer surface of the vessel, even with interference fits. Alloy 600 steam generator drain pipes at Shearon Harris (1988) and pressurizer instrument nozzles at Nogent 1 and Cattenom 2 (1989) were all roll expanded but still developed leaks during operation (Ref 4). Plant specific top head gap analyses have been performed for a large number of plants, with nozzle initial interference fits ranging from 0 to 0.0034". These analyses have confirmed the presence of a physical leak path in essentially all nozzles under normal operating pressure and temperature conditions (Ref 4).

The probability of detecting small control rod drive mechanism (CRDM) leaks by visual inspections alone is high. "Visual inspections of the reactor coolant system pressure boundary have been proven to be an effective method for identifying leakage from PWSCC cracks in Alloy 600 base metal and Alloy 82/182 weld metal. Specifically, visual inspections have detected leaks in RPV head CRDM nozzles, RPV head thermocouple nozzles, pressurizer heater sleeves, pressurizer instrument nozzles, hot leg instrument nozzles, steam generator drain lines, a RPV hot leg nozzle weld, a power operated relief valve (PORV) safe end, and a pressurizer manway diaphragm plate" (Ref 5). To date, no leaking (CRDM) nozzles have been discovered by non-visual NDE examinations except for the three nozzles at Davis-Besse where leakage would have been detected visually had there been good access for visual inspections and the head cleaned of pre-existing boric acid deposits from other sources (Ref 4).

Finally, as described under Concern 3 below, detailed probabilistic fracture mechanics (PFM) analyses have been performed to demonstrate the effectiveness of visual inspections in protecting the CRDM nozzles against failure due to circumferential cracking (Ref 6). Even though the above discussion illustrates that visual inspections performed in accordance with MRP recommendations have a high probability of detecting through-wall leakage, a very low probability of detection was assumed in the PFM analyses. The PFM analyses assume only a 60% probability that leakage will be detected if a CRDM nozzle is leaking at the time a visual inspection is performed. Furthermore, if a nozzle has been inspected previously, and leakage was missed, subsequent visual inspections are assumed to have only a 12% probability of detecting the leak. Even with these conservative assumptions regarding the probability of detection, the PFM analyses show that visual inspection every outage reduces the probability of a nozzle ejection to an acceptable level for plants with 18 or more EDY. Visual inspections of plants with fewer than 18 EDY in accordance with the MRP Inspection Plan will maintain the probability of nozzle ejection for these plants more than an order of magnitude lower than that for the greater-than-18 EDY plants.

In summary, the industry has responded to the need to detect small amounts of leakage by increased visual inspection sensitivity, increased inspection frequencies, and improved inspection capabilities. Small amounts of leakage can be detected visually and it has been shown that timely detection by visual examination will ensure the structural integrity of the RPV head penetrations with respect to circumferential cracking. These conclusions are especially applicable to plants with relatively low EDY values, such as IP2, as recognized by the graded inspection approach provided in Bulletin 2002-02.

Concern 2: Cracking of 82/182 weld metal has been identified in CRDM nozzle J-groove welds for the first time and can precede cracking of the base metal. This finding raises concerns because examination of weld metal material is more difficult than base metal.

Response: Cracks in the J-groove weld do not pose an increased risk regarding nozzle ejection as compared to penetration base metal cracks. J-groove weld cracks that initiate and grow through-wall will leak the same as cracks in the penetration base metal. Therefore, weld cracks pose a similar risk as cracks in the base material and are equally detectable by visual examination. Although higher crack growth rates have been observed in laboratory testing of weld metal, the industry model of time-to-leakage includes plants that have had weld metal cracking as well as base metal cracking.

Concern 3: Through-wall circumferential cracking from the outside diameter of the CRDM nozzle has been identified for the first time. This raises concerns about the potential for failure of CRDM nozzles and control rod ejection, causing a LOCA.

Response: Probabilistic fracture mechanics (PFM) analyses using a Monte-Carlo simulation algorithm were performed to estimate the probability of nozzle failure and control rod ejection due to through wall circumferential cracking (Ref 6). The PFM analyses conservatively assume that, once a leak path has extended to the annulus region, an OD circumferential crack develops instantaneously, with a length encompassing 30° of the nozzle circumference. Fracture mechanics crack growth calculations are then performed for this initially assumed crack, using material crack growth rate data from EPRI Report MRP-55 (Ref 7). The parameters used in the PFM model were benchmarked against the most severe cracking found to date in the industry (B&W Plants) and produced results that are in agreement with experience to date. The analyses were used to determine probability of nozzle failure versus EFPY for various head operating temperatures. Analyses were then performed to estimate the effect of visual and non-visual (NDE) inspections of the plants in the most critical inspection category, using the conservative assumption discussed above (see Concern #1 response) for probability of leakage detection by visual inspection. These analyses demonstrate that performing visual inspections significantly reduces the probability of nozzle ejection, and that performing such examinations on a regular basis effectively maintains the probability of nozzle ejection at an acceptably low level indefinitely.

In the extremely unlikely event that nozzle failure and rod ejection were to occur due to an undetected circumferential crack, an acceptable margin of safety to the public would still be maintained (Ref 8). The consequences of such an event are similar to that of a small-break LOCA, which is a design-basis event. The probability of core damage given a nozzle failure (assuming that failure leads to ejection of the nozzle from the head) has been estimated to be 1×10^{-3}. The PFM analyses demonstrate that periodic visual inspections are capable of maintaining the probability of nozzle failure due to circumferential cracking well below 1×10^{-3}. Therefore, the PFM analyses demonstrate that the resulting incremental change in core damage frequency due to CRDM nozzle cracking can be maintained at less than 1×10^{-6} ($10^{-3} \times 10^{-3} = 10^{-6}$) per plant year, through a program of periodic visual examinations performed in accordance with the MRP Inspection Plan. This result is consistent with NRC Regulatory Guide 1.174 that defines an acceptable change in core damage frequency (1×10^{-6} per plant year) for changes in plant design parameters, technical specifications, etc.

Concern 4: The environment in the CRDM housing/RPV head annulus will likely be more aggressive after any through-wall leakage because potentially highly concentrated borated primary water may become oxygenated. This raises concerns about the technical basis for current crack growth rate models.

Response: The MRP panel of international experts on SCC (including representatives from ANL/NRC Research), prior to the Davis-Besse incident, gave extensive consideration to the likely environment in the annulus between a leaking CRDM nozzle and the RPV head and revisited this issue subsequently (Ref 7). When revisited, the relevant arguments remain valid for leak rates that are less than 1 liter/h or 0.004 gpm, which plant experience has shown to be the usual case. The conclusions were

1. An oxygenated crevice environment is highly unlikely because:

   - Back diffusion of oxygen is too low compared to counterflow of escaping steam (two independent assessments based on molecular diffusion models were examined).
   - Oxygen consumption by the metal walls would further reduce its concentration.
   - Presence of hydrogen from leaking water and diffusion through the upper head results in a reducing environment.
   - Even if the concentration of hydrogen was depleted by local boiling, coupling between low alloy steel and Alloy 600 would keep the electrochemical potential low.

- Corrosion potential will be close to the Ni/NiO equilibrium, resulting in PWSCC susceptibility similar to normal primary water.

2. The most likely crevice environments are either hydrogenated steam or PWR primary water within normal specifications and both would result in similar, i.e. non-accelerated, susceptibility of the Alloy 600 penetration material to PWSCC.

3. If the boiling interface happens to be close to the topside of the J-weld, itself a low probability occurrence, concentration of PWR primary water solutes, lithium hydroxide and boric acid, can in principle occur. Of most concern here would be the accelerating effect of elevated pH on SCC, but calculations and experiments show that any changes are expected to be small, in part because of the buffering effects of precipitates. A factor of 2 on the crack growth rate (CGR) conservatively covers possible acceleration of PWSCC, even up to a high-temperature pH of around 9.

For larger leakage rates, which could lead to local cooling of the head, concentration of boric acid, and development of a sizeable wastage cavity adjacent to the penetration, the above arguments no longer directly apply. However, limited data (Berge et al., 1997) on SCC in concentrated boric acid solutions indicate that

- Alloy 600 is very resistant to transgranular SCC (material design basis).
- High levels of oxygen <u>and</u> chloride are necessary for intergranular cracking to occur at all.
- The effects are then worse at intermediate temperatures, suggesting that the mechanism is different from PWSCC.

The above considerations show that there is no basis for assuming that any post-leakage, crevice environment in the CRDM housing/RPV head annulus would be significantly more aggressive with regard to SCC of the Alloy 600 penetration material than normal PWR primary water, irrespective of the assumed leakage rate and/or annulus geometry. The current industry model (Ref 7), which includes a factor of 2 on CGR to cover residual uncertainty in the composition of the annulus environment, remains valid.

Concern 5: The presence of boron deposits or residue on the RPV head, due to leakage from mechanical joints, could mask pressure boundary leakage. This raises concerns that a through-wall crack may go undetected for years.

Response: The experience at Davis-Besse has clearly demonstrated that effective visual inspection for leakage from CRDM nozzle and weld PWSCC requires unobstructed inspection access and that the head surface be free of pre-existing boric acid deposits. Accumulations of debris and boric acid deposits from other sources can interfere with a determination as to the presence or absence of boric acid deposits extruding from the tube-to-head annulus. Therefore, to effectively perform a visual examination of the RPV head outer surface for penetration leakage, such deposits and debris accumulations must be carefully inspected, removed, and the area re-inspected. Evaluation may show that it is necessary to perform a non-visual examination to establish the source of the leakage.

Accordingly, each inspection at Indian Point 2 will be conducted with a questioning attitude and any boric acid deposit on the vessel head will be evaluated to determine its source in accordance with existing industry guidance, supplemented by additional more recent industry experience that may be available at the time of the inspection. These requirements are incorporated in the visual inspection guidance contained in the MRP Inspection Plan. Implementation of these requirements will preclude the cited condition of a through wall crack remaining undetected for years.

Concern 6: The causative conditions surrounding the degradation of the RPV head at Davis-Besse have not been definitively determined. The staff is unaware of any data applicable to the geometries of interest that support accurate predictions of corrosion mechanisms and rates.

Response: The causes of the Davis-Besse degradation are sufficiently well known to avoid significant wastage. The root cause evaluation performed by the utility (Ref 9) clearly identifies the root cause as PWSCC of CRDM nozzles followed by boric acid corrosion. The large extent of degradation has been attributed to failure of the utility to address evidence that had been accumulating over a five-year period of time (Figure 26 of Ref 9).

The industry has provided utilities with guidance for vessel top head visual inspections to ensure that conditions approaching those at Davis-Besse will not occur. Visual inspection guidelines have been provided (Ref 3), and a workshop was conducted to thoroughly review industry experience, regulatory requirements, leakage detection, and analytical work performed to understand the causes of high wastage rates (Ref 10).

Subsequent to significant wastage being discovered on the Davis-Besse RPV head, the industry has performed analytical work to determine how a small leak such as seen at several plants can progress to the significant amounts of wastage discovered at Davis-Besse. This work is referenced within the basis for the MRP Inspection Plan (Ref 11) and was previously presented to the NRC (Ref 12).

The analytical work shows that the corrosion rate is a strong function of the leakage rate. Finite element thermal analyses show that leak rates must reach approximately 0.1 gpm for there to be sufficient cooling of the RPV top head surface to support concentrated liquid boric acid that will produce high corrosion rates. The leak rate is in turn a strong function of the crack length. The effect of crack length above the J-groove weld on crack opening displacement and area has been confirmed by finite element modeling of nozzles including the effects of welding residual stresses and axial cracks. Leak rates have been calculated using crack opening displacements and areas determined by the finite element analyses and leak rate models based on PWSCC cracks in steam generator tubes.

Cracks that just reach the annulus through the base metal or weld metal will result in small leaks such as those that produced small volumes of boric acid deposits on several vessel heads at locations where the CRDM nozzles penetrate the RPV head outside surface. These leaks are typically on the order of 10^{-6} to 10^{-4} gpm. There is no report of any of these leaks resulting in significant corrosion. A leak rate of 10^{-3} gpm will result in the release of about 500 in^3 of boric acid deposits in an 18-month operating cycle, which will be detectable by visual inspections.

The time for a crack to grow from a length that will produce a leak rate of 10^{-3} gpm to a leak rate of 0.1 gpm has been estimated by deterministic analyses based on the MRP crack growth models to be 1.7 years for plants with 602°F head temperatures. Probabilistic analyses show that there is less than a 1×10^{-3} probability that corrosion will proceed to the point that the inside surface cladding of the head would be uncovered over a significant area before the wastage would be detected by supplemental visual inspections as required under the MRP Inspection Plan. During the transition from leak rates of 10^{-3} gpm to 0.1 gpm, loss of material will be by relatively slow processes (Ref 11).

The ability to detect leakage prior to the risk of structural failure is illustrated by Figure 26 of the Davis-Besse root cause analysis report. There was visual evidence of boric acid deposits on the vessel head for five years prior to the degradation being detected. Guidance provided in the MRP Inspection Plan would not permit these conditions to exist without determining the source of the leak, including nondestructive examinations if necessary.

Therefore, while the exact timing of the event progression at Davis-Besse cannot be definitively established, the probable durations can be predicted with sufficient certainty to conclude that a visual inspection regimen can ensure continued structural integrity of the RCS pressure boundary.

REFERENCES

1. EPRI Letter MRP 2002-086, "Transmittal of "PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan, Revision 1, August 6, 2002", from Leslie Hartz, MRP Senior Representative Committee Chairman, August 15, 2002.

2. EPRI Document MRP-75, Technical Report 1007337, "PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan, Revision 1", September 2002.

3. EPRI Technical Report 1006899, "Visual Examination for Leakage of PWR Reactor Head Penetrations on Top of the RPV Head: Revision 1", March 2002.

4. Appendix B of EPRI Document MRP-75 "Probability of Detecting Leaks in RPV Top Head Nozzles," September 2002.

5. EPRI TR-103696, "PWSCC of Alloy 600 Materials in PWR Primary System Penetrations", July 1994.

6. Appendix A of EPRI Document MRP-75 "Technical Basis for CRDM Top Head Penetration Inspection Plan," September 2002.

7. EPRI Document MRP-55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," July 2002.

8. Walton Jensen, NRC, Reactor Systems Branch, Division of Systems Safety and Analysis (DSSA), Sensitivity Study of PWR Reactor Vessel Breaks, memo to Gary Holahan, NRC, DSSA, May 10, 2002.

9. Davis-Besse Nuclear Power Station Report CR2002-0891, "Root Cause Analysis Report – Significant Degradation of the Reactor Pressure Vessel Head," April 2002.

10. EPRI Technical Report 1007336, "Proceedings of the EPRI Boric Acid Corrosion Workshop, July 25–26, 2002 (MRP-77)", September 2002.

11. Appendix C of EPRI Document MRP-75 "Supplemental Visual Inspections to Ensure RPV Closure Head Structural Integrity," September 2002.

12. Glenn White, Chuck Marks and Steve Hunt, Technical Assessment of Davis-Besse Degradation, Presentation to NRC Technical Staff, May 22, 2002.