April 2, 2002
NL-02-050
IPN-02-023

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

SUBJECT: Indian Point 2 and 3 Nuclear Power Plants
Docket Nos. 50-247 and 50-286
License Nos. DPR-26 and DPR-64
"Submittal of 15 Day Response to NRC Bulletin 2002-01"

Reference:

1. NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002

2. Entergy Letter NL-01-106, dated September 4, 2001, "Thirty-Day Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"

3. Entergy letter IPN-01-063, dated August 31, 2001, "Thirty-Day Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"

4. Entergy letter IPN-01-079/NL-01-133, dated November 13, 2001, Revised Vessel Head Penetration Inspection Plans, NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"

Dear Sir:

Attached is Entergy Nuclear Operations' Inc. (ENO) response to Bulletin 2002-01 (Reference 1) for Indian Point Units 2 and 3, the Indian Point Entergy Center (IPEC). Attachments 1 and 2 contain the responses for Indian Point Units 2 and 3, respectively.

ENO recognizes the safety significance of the events discussed in the Bulletin and is committed to a timely and complete resolution of the issue. At this time, ENO believes there is reasonable assurance that regulatory requirements are currently being met and will continue to be met. ENO will continue to monitor industry experience regarding this Bulletin for applicability to Indian Point Units 2 and 3.

No new commitments are being made in this letter. If you have any questions, please contact Mr. John McCann (914) 734- 5074 or Mr. John Donnelly at (914) 736-8310, at Units 2 and 3, respectively.

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

Very truly yours,

[signed 4/2/02 by J. Herron]

Executed on ____________________________
(Date)

Mr. John Herron
Senior Vice President
Indian Point Units 2 and 3
Entergy Nuclear Operations, Inc.

Attachment: As stated

cc: Mr. Hubert J. Miller
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Indian Point 2

15-Day Response to Bulletin 2002-01

<u>Required Information</u>

1. Within 15 days of the date of this bulletin, all PWR addressees are required to provide the following:

   A. A summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant,

Response: Indian Point 2 performs a VT-2 visual examination of the reactor coolant system including the reactor vessel and its attachments each refueling outage, as required by the ASME Section XI Code. A review of the most recent pressure test inspection results shows that ASME Section XI inspection criteria were met.

In addition to the visual examinations performed during the pressure tests, enhanced visual/video examinations have been performed in the past, to evaluate and disposition canopy seal weld defects, which resulted in canopy seal weld leaks. The latest documented video examination of a portion of the top of the head insulation was performed during a 1998 outage under the direction of the Indian Point 2, Level III examiner who was trained to report non-conforming conditions such as boron deposits resulting from primary coolant leakage. This inspection was performed after implementation of the modification to the partial length Control Rod Drive Mechanism (CRDM) nozzles, which were cut and capped to eliminate the potential for leaks. During this modification one canopy seal weld was found leaking and was subsequently repaired. A review of the video tape for the final inspection (i.e. after the modification and weld repairs had been implemented) indicated that the reactor coolant system had been pressurized for approximately one week and no leaks were detected during this inspection.

In addition to the above documented inspections, disassembly of the reactor pressure vessel head during refueling outages provides additional opportunities to detect abnormal conditions such as those typical of primary coolant leaks which result in visible boron deposits. It is also anticipated that during plant operations, a change in the Reactor Coolant System (RCS) leakage would be noted by leakage monitoring, and/or the containment radiation detectors, and actions would be taken to determine the source of the leakage in containment.

   B. An evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse, Response: While the insulation installed on the IP2 reactor pressure vessel head prevents "bare metal" inspections, it does offer a protective barrier against boric acid due to leaks occurring above the insulation.

The insulation system applied to the Indian Point 2 vessel head is 3 1/4" "Kaylo Block" filled with asbestos cement prior to application of two layers of asbestos tape 90^0 from each other. A final coating of 1/2" thick "One Cote" asbestos cement was applied over the tape. This type of insulation provides a barrier which significantly reduces the probability that leakage from above the head will reach the carbon steel. Boric acid leakage from above the RPV head insulation (for example, canopy seal weld area) would have a tendency to fall on the insulation system (which has a hard, cement like surface) and thus not affect the RPV head. The RPV head is painted with a heat resistant silicone aluminum coating (Wisconsin Plasite #888 coating), which has been tested to be boric acid resistant. Severe leaks could soak the insulation, if the insulation system was damaged, but the heat resistant silicone aluminum coating would provide additional protection.

C. A description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions,

Response: A review of the Indian Point 2 operating history indicates that there have been three events of Conoseal leakage and three events of canopy seal weld leakage. The Conoseal leakage events occurred in 1988, 1996 and 1997. The three events of canopy seal weld leakage occurred in 1986, 1988 and in 1997. As a result of the 1986 leakage event, Indian Point 2 performed a root cause evaluation and performed weld repairs of the four leaking canopy seal welds prior to plant restart.

Indian Point 2 implemented inspections and testing during the1986, 1987, and 1988 outages to verify that the prior leakage had not adversely impacted the structural integrity of the head. This included the following actions:

(1) A detailed video examination of the head insulation surface to ensure that leaks were identified and corrected prior to plant restart (1988).

(2) Removal of sections of insulation (in 1986, 1987, and 1988) adjacent to leaking canopy seal welds and a bare metal inspection of the reactor vessel head base metal to ensure that the leakage experienced during the previous fuel cycle had not adversely impacted the structural integrity of the head. Although limited in scope, these bare metal inspections demonstrated the protective nature of the

insulation.

(3) Chemical resistance tests were performed in 1986 to evaluate the resistance of the silicone aluminum coating to a boric acid solution. After subjecting the test coupons to simulated head temperature conditions, including heatups and cooldowns, the coupons were microscopically examined to assess the condition of the coating. These tests showed no evidence of any deterioration of the coating. There was also no evidence of rusting, peeling, blistering or loss of adhesion. Subsequent bare metal visual inspections confirmed that the aluminum rich silicone coating (heat resistant Wisconsin Plasite #888 coating) applied during vessel fabrication, remained unaffected by the leakage experienced during the previous fuel cycle.

The 1997 canopy seal leak was also weld-repaired followed by a pressure test to confirm the effectiveness of the process. There has been no evidence of leakage through any of these locations since the 1997 refueling outage.

During a 1998 extended maintenance outage modifications to the part length control rod drive mechanism housings were made in response to concerns identified at Prairie Island Unit 2. No leaks have been identified since the modifications.

D. Your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria, and

Response: ENO has previously provided this information in Reference 4. The effective examination includes essentially 100% of the CRDM penetrations, which will be inspected during the next scheduled refueling outage. ENO is currently working with Westinghouse to develop and qualify a volumetric inspection technique capable of detecting flaws in the Alloy 600 CRDM nozzles. ENO is developing a blade probe for the inspection, which would allow the inspection to be completed in the area between the thermal sleeve and the CRDM penetration (tube material). This inspection will be implemented during the next refueling outage, which is currently scheduled to begin in October of 2002 for IP2. Flaws detected during the examinations will be evaluated against the requirements of the ASME Section XI Boiler and Pressure Vessel Code. Any flaws found to exceed the limits of ASME Section XI will be repaired consistent with the requirements of 10CFR50.55a. The flaw acceptance criteria are currently under development. Since the qualification of the inspection technique is currently on going, the details are not yet available. However, they will be provided to the NRC staff no later than 90 days prior to the start of the refueling outage. This would allow any industry “learned lessons” from the Winter/Spring 2002 refueling outages of other plants with a similar design to be included in the inspection program.

ENO has not finalized its plans to ensure that degradation similar to the Davis Besse degradation is detected and corrected prior to restart from the next refueling outage. However, ENO is evaluating several options, which could be implemented during the next refueling outage if the root cause evaluation resulting from the Davis Besse degradation indicates that additional inspections are necessary to ensure that the structural integrity of the head is maintained during future operation. The following options are under consideration:

(1) Removal of the insulation and performing a bare metal inspection of the head,

(2) Augmented Alloy 600 inspection techniques to verify sound base metal behind the CRDM nozzles,

(3) Straight beam UT from the cladded surface to verify the head thickness.

Any additional inspections deemed necessary to address the Davis-Besse type of degradation will be included in a report to be submitted to the NRC Staff no later than 90 days prior to the start of the next refueling outage.

E. Your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in response to Item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met.

Response: The recently discovered Davis-Besse head corrosion was likely a result of either (1) reactor coolant leakage through a crack in the Alloy 600 CRDM nozzle, (2) a result of above the head leakage which dripped on to the head outside surface or (3) a combination of both. The probability that either one of these driving mechanisms is present at IP2, is considered to be low for the following reasons:

(1) The presence of through wall cracks in the CRDM nozzles at Indian Point 2 is considered to be unlikely. This conclusion is based on the fact that Indian Point 2 was ranked as one of the lowest plants in the moderate susceptibility category in accordance with the MRP ranking criteria, which was used to respond to Bulletin 2001-01. In fact, IP2 has only accumulated approximately 40% of the time at temperature when compared to Davis-Besse as of March 1, 2001. Since the accumulated EFPY to date is directly proportional to the susceptibility of the CRDM nozzles to PWSCC (i.e. rather than number of EFPY to reach the Oconee

3 condition), Indian Point 2 is considered to be the least susceptible plant in the moderate susceptibility category. Based on this, Indian Point 2 is not expected to have any through wall cracks in the CRDM nozzles prior to our next scheduled inspection.

(2) Although Indian Point 2 has experienced leakage from Conoseal and from canopy seal welds, it is unlikely that any of this leakage has had any impact on the structural integrity of the reactor vessel head. This conclusion is based on the fact that the head is protected by a high temperature aluminum silicone based coating which has been demonstrated to be resistant to boric acid attack and is also protected by the head insulation. The coating and the insulation in combination with the high head temperatures during operation (i.e. conditions conducive to evaporation of any above the head leakage) are expected to provide an effective barrier against above the head leakage from contacting the head base metal. This conclusion is also supported by the results of the bare metal inspections performed during the 1986, 1987, and 1988 outages.

Regulatory requirements are currently being met and will continue to be met. Compliance with the regulatory documents referred to in NRC Bulletin 2002-01 or MRP-48 is detailed in the Indian Point 2 UFSAR, and other plant-specific licensing bases documents. The NRC Bulletin 2002-01 section on applicable regulatory requirements cited: General Design Criteria 14, 31, and 32 of Appendix A, 10 CFR 50; Criterion V, IX and XVI of Appendix B to 10 CFR 50; 10 CFR 50.55(a); plant Technical Specifications; and, Generic Letter 88-05.

The general design criteria (GDC), as outlined in Bulletin 2002-01, came into effect after the Indian Point 2 facility operating license was issued. The draft GDC that Indian Point 2 was licensed to was addressed in the FSAR at that time.

The requirements established for documented instructions, procedures, or drawings for activities affecting quality, for special processes, and for corrective action in Criterion V, IX, and XVI of Appendix B to 10 CFR 50 are satisfied. Criterion V and IX are forward looking criterion applicable to new inspections that must be performed and compliance is required in accordance with our Quality Assurance Manual requirements. Criterion XIV requires measures to assure that conditions adverse to quality are promptly identified and corrected. The plant has a corrective action program that requires industry feedback to be evaluated for applicability and corrective action.

10 CFR 50.55(a) - The Inservice Inspection Program performed in accordance with the ASME code and our program provides the basis for concluding that through wall leakage of the reactor coolant pressure boundary does not exist.

Technical Specification 3.1.F.2.c(1)(a) requires that the plant have a zero reactor coolant pressure boundary leakage. At this time no elevated leakage has been detected inside containment. Inservice inspection will determine whether any leakage below the threshold of detection has occurred.

The plant has a program for implementation of the requirements of Generic Letter 88-05.

The presence of through wall cracks in the CRDM nozzles at Indian Point 2 is considered unlikely, as it is considered to have a low susceptibility to Primary Water Stress Corrosion Cracking (PWSCC) based upon the industry evaluation ranking, as of March 1, 2001.

Based on the above, it is concluded that the probability that the Indian Point 2 reactor vessel head has experienced the same degradation as that detected in the Davis-Besse head is low. Therefore, ENO concludes that there is reasonable assurance that the regulatory requirements listed above are currently being met and will continue to be met.

References

1. NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2001

2. Entergy letter NL-01-106, dated September 4, 2001, "Thirty-Day Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"

3. Entergy letter IPN-01-063, dated August 31, 2001, "Thirty-Day Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"

4. Entergy letter IPN-01-079/NL-01-133, dated November 13, 2001, Revised Vessel Head Penetration Inspection Plans, NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"

Indian Point 3

15-Day Response to Bulletin 2002-01

<u>Required Information</u>

1. "Within 15 days of the date of this bulletin, all PWR addressees are required to provide the following:

   A. A summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant,"

Response: Indian Point 3 performs a VT-2 visual examination of the reactor coolant system including the reactor vessel and its attachments each refueling outage, as required by the ASME Section XI Code. A review of the most recent pressure test inspection results shows that ASME Section XI inspection criteria were met.

In addition, the Reactor Pressure Vessel (RPV) head area is inspected every refueling outage in accordance with NRC Generic Letter 88-05. This is completed in accordance with Procedure 3PT-R114, "RCS Boric Acid Leakage and Corrosion Inspection". This inspection includes inspecting the Control Rod Drive Mechanism (CRDMs) and a general visual to determine if there are boric acid deposits, which would indicate a Reactor Coolant System (RCS) leakage path. The program that has been implemented is considered to be adequate to detect RCS leakage paths under the shroud. When leaks are detected, affected areas are inspected to ensure that no Reactor Vessel head degradation has occurred.

In addition to the above documented inspections, disassembly of the reactor pressure vessel head during refueling outages provides additional opportunities to detect abnormal conditions such as those typical of primary coolant leaks which result in visible boron deposits. It is also anticipated that during plant operations, a change in the RCS leakage would be noted by leakage monitoring, and/or the containment radiation detectors, and actions would be taken to determine the source of the leakage in containment.

   B. An evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse,"

Response: While the insulation installed on the IP3 reactor pressure vessel head prevents "bare metal" inspections, it does offer a protective barrier against boric acid due to leaks occurring above the insulation.

The insulation system applied to the Indian Point 3 vessel head (similar to that used at IP2) is 3 1/4" "Kaylo Block" filled with asbestos cement prior to application of two layers of asbestos tape 90^0 from each other. A final coating of 1/2" thick "One Cote" asbestos cement was applied over the tape. The RPV head is painted with a heat resistant silicone aluminum coating. This type of insulation provides a barrier, which significantly reduces the probability that leakage from above the head will reach the carbon steel. Boric acid leakage from above the RPV head insulation (for example canopy seal weld area) would have a tendency to fall on the insulation system (which has a hard, cement like surface) and thus not affect the RPV head. Severe leaks could soak the insulation, if the insulation system was damaged, however, a heat resistant silicone aluminum paint, which meets MIL-P-14276B requirements and is similar to the IP2 coating (which has been tested to be boric acid resistant) would provide additional protection. If a significant leak from above the head manifests visible boron residues on top of the insulation, it would be identified by inspections. This area was inspected and videotaped during the inspection. No significant boric acid residue was noted.

C. "A description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions,"

Response: During the startup from a scheduled outage in March 1990, a canopy seal weld leak was detected during the performance of 3PT-R114 surveillance, which was developed for compliance with GL-88-05. This leak was from overhead and dripped onto the insulation. A Canopy Seal Clamp Assembly was installed at penetration #28 at the Canopy seal weld area. A section of the Reactor Vessel head insulation was removed that covered approx. 14 CRDM penetrations in order to perform the repair. No degradation of the RPV base material was noted after the insulation was removed.

A Canopy Seal Clamp Assembly was also installed at penetration #26 at the Canopy seal weld area as a precaution.

The spared capped and core exit thermocouple (CET) penetrations for IP3 have the Canopy Seal Clamp assembly installed. This prevents leakage from the canopy seal weld area. The CET penetration numbers are 74, 75, 76, 77 and 78. The spare penetrations are 2, 3, 4, 5, 15, 17, 19, 21, 26, 27, 28 and 29. During refueling outage 11 (spring of 2001), preventive maintenance was performed on the Canopy Seal assemblies for the CET and spare penetrations.

This included visual examinations for leakage and bolt torque checks.

During refueling outages 10 (Fall 1999) and 11 (Spring 2001) videotapes were taken to document the Reactor Vessel head inspections.

Refueling Outage 10

The exposed surface of the Reactor Vessel head outside the CRDM shroud assembly was inspected per 3PT-R114 during vessel disassembly by the Refueling crew. At the beginning of R10, the inspection noted boron deposits/streaks around vessel studs #5 through #18, which was documented in the test procedure and in the IP3 corrective action program. The corrective action implemented was an enhanced, video inspection under the vessel CRDM shroud using a camera mounted to an extended pole in lieu of the normal general visual inspection via the access ports. This inspection was videotaped for more comprehensive evaluation and to allow comparison with future inspections. Although evidence of the historical leaks remained (for example, boron streaking on the CRDMs that was determined to be coming from above the RPV head and insulation), no indication of any new leakage or degradation of insulation was found. It was postulated that the residue found on the exposed surface of the vessel head flange was due to humidity and entrainment of historical residue in the head ventilation system, which condensed outside the CRDM shroud during plant cooldown. The deposits on the exposed surface were cleaned and no degradation was noted.

Refueling Outage 11

Inspections were performed and boron deposits were noted around #4 Conoseal. This was documented in IP3's corrective action program. This leakage was initially noted on-line (during a leak inspection walkdown inside the containment by plant personnel. During RO11, the IP3 ISI engineer, Corporate Metallurgist and IP3 system engineer evaluated the areas that showed evidence of boron residue. No degradation of the Reactor vessel was noted. Conoseal #4 was disassembled, repaired and replaced during this refueling outage. The conoseal gaskets are replaced each refueling as part of the reactor disassembly / reassembly process.

Using a remote camera, a general area inspection of the RPV head (top of insulation and CRDMs) and approximately 60% of the nozzles (at the insulation interface) were inspected by a VT-2 equivalent examination from above the vessel head insulation. The RO11 inspection was compared with an inspection videotaped during the previous refuel outage - RO10. There were no apparent changes in the condition of the vessel head under the cooling shroud with the exception of the Conoseal No. 4 conoseal clamp leakage discovered prior to the RO11 outage. Boron had precipitated from this leak and collected on the alloy steel canopy clamp. Also, there is evidence that some traces did traverse down the tube and were entrained in the CRDM ventilation depositing on the exposed vessel head outside the cooling shroud. The inspection revealed minor streaks of boron residue on surface at the location of this stud hole No. 38, which were cleaned prior to return to service with no degradation noted.

In summary, there was no evidence of leakage from penetration/vessel head joints at inspected locations. An engineering evaluation was performed in Refueling outage 11 of all boric acid residues to ensure that there was no affect on RPV head integrity.

D. "Your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria, and"

Response: ENO has previously provided this information in Reference 4. The effective examination includes essentially 100% of the CRDM penetrations, which will be inspected during the next scheduled refueling outage. ENO is currently working with Westinghouse to develop and qualify a volumetric inspection technique capable of detecting flaws in the Alloy 600 CRDM nozzles. ENO is developing a blade probe for the inspection, which would allow the inspection to be completed in the area between the thermal sleeve and the CRDM penetration (tube material). This inspection will be implemented during the next refueling outage, which is currently scheduled to begin in April of 2003 for IP3. Flaws detected during the examinations will be evaluated against the requirements of the ASME Section XI Boiler and Pressure Vessel Code. Any flaws found to exceed the limits of ASME Section XI will be repaired consistent with the requirements of 10CFR50.55a. The flaw acceptance criteria are currently under development. Since the qualification of the inspection technique is currently on going, the details are not yet available. However, they will be provided to the NRC staff no later than 90 days prior to the start of the refueling outage. This would allow any industry "learned lessons" from the Winter/Spring 2002 refueling outages of other plants with a similar design to be included in the inspection program.

ENO has not finalized its plans to ensure that degradation similar to the Davis-Besse degradation is detected and corrected prior to restart from the next refueling outage. However, ENO is evaluating several options, which could be implemented during the next refueling outage if the root cause evaluation resulting from the Davis-Besse degradation indicates that additional inspections are necessary to ensure that the structural integrity of the head is maintained during future operation. The following options are under consideration:

(1) Removal of the insulation and performing a bare metal inspection of the head,

(2) Augmented Alloy 600 inspection techniques to verify sound base metal behind the CRDM nozzles,

(3) Straight beam UT from the cladded surface to verify the head thickness.

Any additional inspections deemed necessary to address the Davis-Besse type of degradation will be included in the report to be submitted to the NRC Staff no later than 90 days prior to the start of the next refueling outage.

E. "Your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in response to Item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met.

Response: The recently discovered Davis-Besse head corrosion was likely a result of either (1) reactor coolant leakage through a crack in the Alloy 600 CRDM nozzle, (2) a result of above the head leakage which dripped on to the head outside surface, or (3) a combination of both. The probability that either one of these driving mechanisms is present at IP3, is considered to be low for the following reasons:

The presence of through wall cracks in the CRDM nozzles at Indian Point 3 is considered to be unlikely. This conclusion is based on the fact that Indian Point 3 has only accumulated approximately 60% of the time at temperature when compared to Davis-Besse as of March 1, 2001. Under the MRP ranking used to respond to Bulletin 2001-01, Indian Point 3 was ranked as a moderately susceptible plant while Davis-Besse was ranked as a highly susceptible plant. Based on this, Indian Point 3 is not expected to have any through wall cracks in the CRDM nozzles prior to our next scheduled inspection.

(1) Although Indian Point 3 has experienced some leakage from Conoseal and from canopy seal welds, it is unlikely that any of this leakage has had any impact on

the structural integrity of the reactor vessel head. This conclusion is based on the fact that the head is protected by a high temperature aluminum silicone based coating and is also protected by the head insulation which was especially installed to eliminate voids which could provide a path for the leaking fluid to contact the vessel head. The coating and the insulation in combination with the high head temperatures during operation (i.e. conditions conducive to evaporation of any above the head leakage) are expected to provide an effective barrier against above the head leakage from contacting the head base metal.

Regulatory requirements are currently being met and will continue to be met. Compliance with the regulatory documents referred to in NRC Bulletin 2002-01 or MRP-48 is detailed in the Indian Point 3 UFSAR, and other plant-specific licensing bases documents. The NRC Bulletin 2002-01 section on applicable regulatory requirements cited: General Design Criteria 14, 31, and 32 of Appendix A, 10 CFR 50; Criterion V, IX and XVI of Appendix B to 10 CFR 50; 10 CFR 50.55(a); plant Technical Specifications; and, Generic Letter 88-05.

The general design criteria (GDC), as outlined in Bulletin 2002-01, came into effect after the Indian Point 3 facility operating license was issued. The draft GDC that Indian Point 3 was licensed to was addressed in the FSAR at that time.

The requirements established for documented instructions, procedures, or drawings for activities affecting quality, for special processes, and for corrective action in Criterion V, IX, and XVI of Appendix B to 10 CFR 50 are satisfied. Criterion V and IX are forward looking criterion applicable to new inspections that must be performed and compliance is required in accordance with our Quality Assurance Manual requirements. Criterion XIV requires measures to assure that conditions adverse to quality are promptly identified and corrected. The plant has a corrective action program that requires industry feedback to be evaluated for applicability and corrective action.

10 CFR 50.55(a) - The Inservice Inspection Program performed in accordance with the ASME code and our program provides the basis for concluding that through wall leakage of the reactor coolant pressure boundary does not exist.

Technical Specification 3.4.13 requires that the plant have a zero reactor coolant pressure boundary leakage. At this time no elevated leakage has been detected inside containment. Inservice inspection will determine whether any leakage below the threshold of detection has occurred.

The plant has a program for implementation of the requirements of Generic Letter 88-05.

The presence of through wall cracks in the CRDM nozzles at Indian Point 3 is considered unlikely, as it is considered to have a moderate susceptibility to Primary Water Stress Corrosion Cracking (PWSCC) based upon the industry evaluation ranking, as of March 1, 2001.

Based on the above, it is concluded that the probability that the Indian Point 3 reactor vessel head has experienced the same degradation as that detected in the Davis-Besse head is low. Therefore, ENO concludes that there is reasonable assurance that the regulatory requirements listed above are currently being met and will continue to be met.

References

1. NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2001

2. Entergy letter NL-01-106, dated September 4, 2001, "Thirty-Day Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"

3. Entergy letter IPN-01-063, dated August 31, 2001, "Thirty-Day Response to NRC Bulletin 2001-01, Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"

4. Entergy IPN-01-079/NL-01-133, dated November 13, 2001, Revised Vessel Head Penetration Inspection Plans, NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"