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August 31, 2001
IPN-01-063

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

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Thirty-Day Response to NRC Bulletin 2001-01
"Circumferential Cracking of Reactor
Vessel Pressure Vessel Head Penetration Nozzles"

References:
1. NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles," dated August 3, 2001.
2. "PWR Material Reliability Program Response to NRC Bulletin 2001-01," (MRP-48), EPRI, Palo Alto, California, 2001, TP-1006284.

Dear Sir:

Attached is Entergy Nuclear Operations' Inc. (ENOI) response to NRC Bulletin 2001-01 (Reference 1) for the Indian Point 3 Nuclear Power Plant.

ENOI recognizes the potential safety significance associated with cracks in Alloy 600 reactor coolant system components and is committed to the timely and complete resolution of this issue. Because of the difficulties associated with removing taped and asbestos cemented insulation from the vessel head, we are not proposing to conduct "bare metal" inspections during the next refueling outage at this time. ENOI is proposing to improve its existing inspection "above the insulation" program and to closely monitor the results of other vessel head penetration (VHP) inspections that will be conducted in the eighteen months before the next refueling outage. If new evidence significantly increases the potential for VHP cracks at Indian Point 3, ENOI will conduct volumetric examinations as soon as reasonable.

A088

Final VHP inspection plans will be submitted to the NRC staff ninety-days before the start of the next refueling outage currently scheduled for April of 2003. The inspection plans in Attachment 1 are based on currently available information and may be revised to reflect new information.

To strengthen our technical abilities in this area, Entergy is planning to form an alliance with a major NSSS vendor to develop new state-of-the-art tooling and methods for volumetric examination at known susceptible cracking areas of CRDM penetrations. This alliance will include the development of both mitigation and repair methods.

ENOI is participating in the MRP-48 (Reference 2) integrated response to Bulletin 2001-01. The integrated response is described in MRP-48. MRP-48 provides background information on all pressurized water reactor (PWR) plants, ranking of the plants relative to Oconee 3 by the MRP, the basis for recommended inspections meeting applicable regulatory requirements, and references to previous MRP submittals containing supporting information.

If you have any questions, please contact Mr. John Donnelly at 914-736-8310.

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

Very truly yours,

Executed on August 31, 2001
(Date)

Mr. Michael Kansler
Vice President and
Chief Operating Officer

cc: Next page.

Attachments:

1. Thirty-Day Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles.

2. Summary of Commitments

cc: Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. P. Milano Senior Project Manager
Project Directorate I-1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop: O-8-E20
Washington, DC 20555-0001

Resident Inspector's Office
Indian Point 3 Nuclear Power Plant
U. S. Nuclear Regulatory Commission
P. O. Box 337
Buchanan, NY 10511

Thirty-Day Response to NRC Bulletin 2001-01 "Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles"

Entergy Nuclear Operations, Inc.
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

INTRODUCTION

This attachment is Entergy Nuclear Operations' Inc. (ENOI) response to the NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles" (Reference 1) for the Indian Point 3 Nuclear Power Plant. The format of this document mirrors the five sections of NRC Bulletin 2001-01.

The inspection plans are based on currently available information and may be revised to reflect new information.

REQUEST 1

1. All addressees are requested to provide the following information:

a. *the plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;*

b. *a description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles;*

c. *a description of the RPV head insulation type and configuration;*

d. *a description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;*

e. *a description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.*

RESPONSE 1a - PLANT SPECIFIC RANKING

Indian Point 3 has been ranked for potential primary water stress corrosion cracking (PWSCC) of the reactor pressure vessel (RPV) top head nozzles using the time-at-temperature model and plant-specific input data reported in MRP-48 (Reference 3). Table 2-1 of MRP-48

(Reference 3), indicates the number of EFPYs (effective full power years) of additional operation from March 1, 2001, to reach the same time-at-temperature that Oconee Nuclear Power Station Unit 3 (ONS3) had at the time that its leaking nozzles were discovered in February 2001.

Using the criteria stated in NRC Bulletin 2001-01, Indian Point 3 falls into the NRC category of plants "with greater than 5 EFPYs and less than 30 EFPY until reaching the Oconee 3 time at temperature."

RESPONSE 1b - DESCRIPTION OF VESSEL HEAD PENETRATIONS

Indian Point 3 has 79 total RPV (reactor pressure vessel) head nozzles. This includes 78 "J-Grooved" CRDM nozzles and 1 butt-welded head vent nozzle. The requested nozzle information is provided in Table 2-3 of MRP-48 (Reference 3).

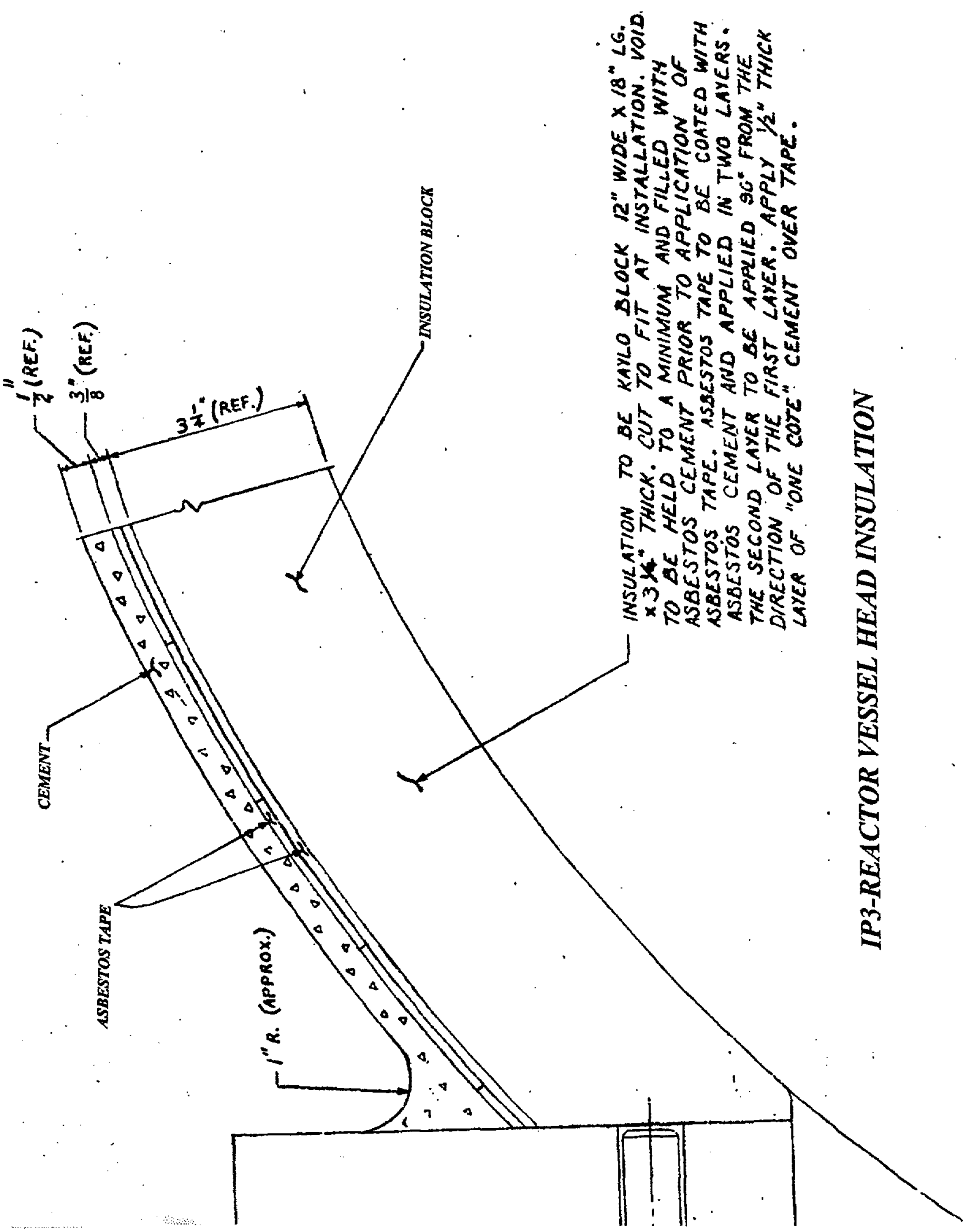
RESPONSE 1c - DESCRIPTION OF HEAD INSULATION

As reported in Table 2-1 of MRP-48 (Reference 3), Indian Point 3 has block contoured RPV head insulation. A note on a plant drawing (Figure 1, Reference 4) describes it as:

> *"Insulation to be Kaylo Block 12" wide x 18" lg. x 3-1/4" thick. Cut to fit at installation. Voids to be held to a minimum and filled with asbestos cement prior to application of asbestos tape. Asbestos tape to be coated with asbestos cement and applied in two layers. The second layer to be applied 90 degrees from the direction of the first layer. Apply ½" thick layer of 'One Cote' cement over tape."*

The actual application shows evidence of trowel-applied material extending upwards at the side of some of the penetrations, indicating use of cement-adhesive at these intersections. See Image T2-2338.

Figure 1
INDIAN POINT 3 REACTOR VESSEL HEAD INSULATION



RESPONSE 1d - INSPECTIONS OVER PAST FOUR YEARS

As reported in Table 2-1 of MRP-48, Indian Point 3 has performed RPV head and nozzle inspections within the past four years.

ENOI has established an Indian Point 3 CRDM nozzle inspection program based on the guidance in NRC Generic Letters 88-05 and 97-01. The program was enhanced for the spring 2001 refueling outage (RO11), to address pressure boundary boric acid leaks detected at some plants at the nozzle-to-head interface on the exterior surface.

A "best-effort" inspection was performed during RO11 with primary emphasis of detecting leakage of boric acid crystals at accessible nozzles to head interface on the exterior surface. Using a remote camera, over 60% of nozzles were inspected by a VT-2 equivalent examination from above the vessel head insulation. Inspection limitations included limited accessibility to the balance of nozzles. The inspection was performed using Entergy Procedure 3PT-R114, "RCS Boric Acid Leakage and Corrosion Inspection," Revision 6 (Reference 5). Due to the position of the reactor vessel head, the head had to be rotated to allow for the inspection from two different entry ports. The inspection was completed on May 10, 2001, and was videotaped.

Entergy compared the RO11 inspection with an inspection videotaped during the previous refueling outage - RO10. There appear to be no changes in the condition of the vessel head under the cooling shroud with the exception of the Conoseal No. 4 penetration tube and canopy leakage discovered prior to the recent 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 to the vessel head and then under the shroud to the exposed vessel surface. The results of the inspection show there are minor streaks of boron residue on this surface at the location of stud hole No. 38.

The inspection videotapes were reviewed by the a team of Entergy engineers including the system engineer, consulting metallurgist, inservice inspection engineer and quality control Level II, VT2 Inspector. Based on this review, the review team determined that there were no apparent areas of concern. The white deposits observed on the woven insulation are insulation repair material. This was verified by comparison of the texture and shape of the deposits with known boric acid deposits as shown on the video inspections performed. Boric acid appears crystalline with shiny facets while the insulation appears amorphous and dull. Some of the penetrations inspected showed rust staining and boric acid deposits running down part of their length but all showed no signs of degradation.

Attached are eight photographs of the inspection, captured from the videotape.

Table 1 - Description of VHP Video Captures

Image No.	Description
T1-2106	These penetrations show no evidence of leakage or corrosion products.
T1-2924	These penetrations show no evidence of leakage or corrosion products.
T2-1330	These penetrations show no evidence of leakage or corrosion products. The white deposit on the woven insulation and joint is insulation repair material.
T2-1811	These penetrations show no evidence of leakage or corrosion products. The white deposit on the penetration is insulation repair material.
T2-1820	These penetrations show no evidence of leakage or corrosion products.
T2-2015	These penetrations show no evidence of leakage or corrosion products. The white deposit on the woven insulation is insulation repair material.
T2-2338	These penetrations show no evidence of leakage or corrosion products. The white deposit on the insulation, joint and penetration is insulation repair material.
T2-2608	These penetrations show no evidence of leakage or corrosion products. The white deposit on the insulation, penetration and shroud is insulation repair material.

The penetration/vessel head joints have insulation around them and were inspected for evidence that degradation or a leak had occurred at the joint. The evidence would be dark rust brown and/or boric acid white discoloration. The inspection showed that the insulation was only lightly discolored where there was evidence of rust stains found to have run down the associated penetration from above.

The penetrations with boric acid deposits showed no signs of corrosion because the boric acid had not progressed to or collected in the penetration/vessel head joints.

Leakage from the Conoseal No. 4 penetration did not collect in the penetration/vessel head joint because, as the boric acid flowed down the side of the clamp, it fell on the vessel head and rolled down the slope away from the joint. The seal was repaired during the outage.

In summary, the penetration/vessel head joints, including the joint associated with the Conoseal No. 4 penetration, showed no evidence of leakage or corrosion products that could be attributed to boric acid collection from above or from a penetration/vessel joint failure.

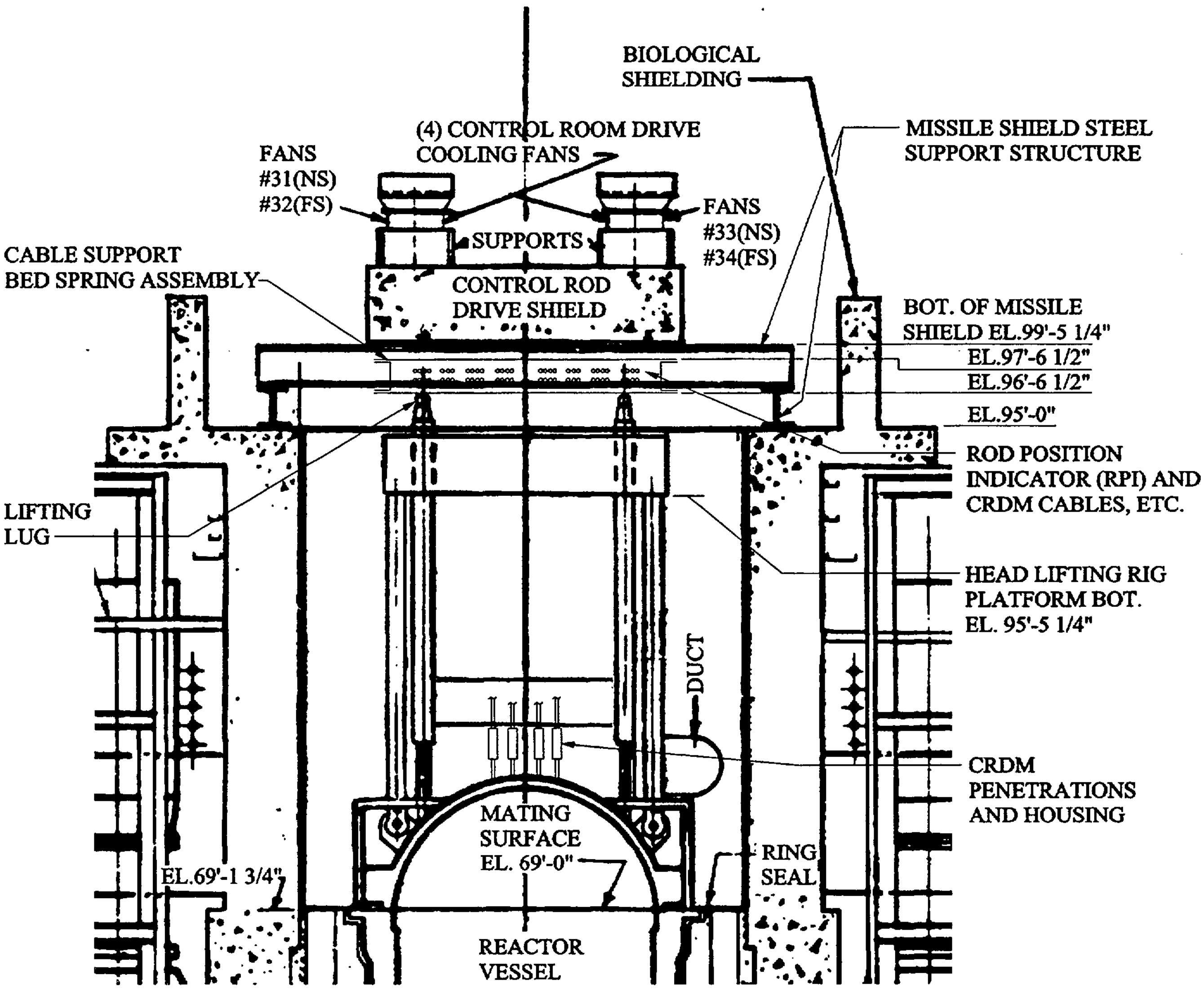
NRC inspectors reviewed the video recording of the remote visual inspection of the CRDM stub-tube-to-reactor-head with the Entergy inspection personnel who performed the test (Reference 16). It was also noted that a leaking Conoseal was discovered and repaired. During a May 23, 2001 meeting with the NRC, the inspector acknowledged that Indian Point 3 took a proactive approach and did a full VT2 exam during RO11 (Reference 17).

RESPONSE 1e - ABOVE REACTOR VESSEL HEAD AREA DESCRIPTION

General Description

Westinghouse provided Indian Point 3's nuclear steam supply system (reactor, ECCS, reactor coolant pumps, etc.). The control rod drive mechanisms (CRDM) attach to the top of the head at the CRDM nozzles (see Figure 2). The nozzles support the CRDM housings. The cables are supported (i.e., Rod Position Indicator, CRDM, etc.) by the bedspring assembly located just below the missile shield. A superstructure frame supported by a concrete structure supports the missile shield. Solenoid valves are located below the bedspring assembly. Except for cables, CRDM cooling fans and ducts are outside of the missile shield perimeter.

Figure 2
GENERAL VIEW OF INDIAN POINT 3
CRDMs AND MISSILE SHIELD
LOOKING WEST
(Not to scale)



Missile Shield

The Control Rod Drive (CRD) missile shield is a concrete and steel structure located directly above the CRDMs and reactor vessel. It measures 17' x 17' x 4' thick. The missile shield consists of four interconnected (stepped) reinforced concrete blocks. Nelson Studs were used to attach a 2-inch thick steel plate to the bottom of each block. Similarly, each concrete block /steel bottom plate was secured to two 24WF145 girders by 1 inch diameter bolts. The structural steel sub-framing includes 2-12WF40 and 2-12WF27 beams. The steel structure supporting the missile shield is anchored to the 95' - 0" elevation (Reference 7 and 8)

The reinforced concrete blocks were configured so as to utilize, to the extent possible, the resistance of the adjoining block to prevent lifting of the blocks (and subsequent drop onto the reactor vessel head) in the event of a postulated missile.

The CRD missile shield was installed to preclude damage to the containment liner and engineered safeguards systems and components from missiles originating from a postulated rupture of a RCC housing. The missile shield was designed as a Seismic Class I Structure (Reference 9 and 10). Calculations considered both OBE and DBE seismic loading.

The NRC, in their Indian Point 3 Safety Evaluation Report (Reference 11) states:

> *"A structure over the control rod drive mechanisms has been provided to block any such potential missiles. We have concluded that the measures taken to provide protection against internally generated missiles are acceptable."*

CRDM Housings and their Support/Restraint System

The CRDM housings are attached to the top of the CRDM nozzles at the top of the reactor vessel head. A seismic support frame restrains the upper portions of the housings. (References 15 and 20)

Electrical Cabling Arrangement and Other Components and Structures from Reactor Head to below Missile Shield

The incore thermocouple (TC) cables exit the RPV through five Conoseal assemblies at approximate elevations of 80' to 82' - 8". These cables are routed from the Conoseals through flexible conduit up to the bedspring located just below the missile shield. The bedspring consists of a structural steel frame with messenger wires suspended within the frame. The messenger wires are placed in three layers in a grid pattern. The Rod Position Indicator (RPI) and Control Rod Drive Mechanism (CRDM) cables are suspended from the messenger wires and are plugged into their respective RPI or CRDM connectors. The CRDM and RPI connectors are at an approximate elevation of 96' with the bedspring assembly at an approximate elevation of 96' - 6.5" to 97' - 6.5".

The bottom of the Missile Shield is at elevation 99' - 5.25".

Four reactor head vent solenoid valves (SOVs) are located at the top of the reactor head-lifting rig, just below the bedspring assembly. Cabling from the 4 SOVs is routed through conduit and junction boxes attached to the bedspring assembly. Cabling attached to the bedspring exits from the hinged side and is routed through floor sleeves or conduit and is routed through electrical raceways to their destinations.

Four CRDM cooling fans are located on the outside of the missile shield support frame. Power cables to these fans are routed exposed across the top of the missile shield blocks to plug connectors located adjacent to the reactor cavity above elevation 95' (Reference 13)

REQUEST 2 - PREVIOUS LEAKAGE

2. *If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide the following information:*

RESPONSE 2

Indian Point 3 has not previously experienced leakage from, or cracking in, VHP nozzles.

REQUEST 3 - SUSCEPTIBILITY RANKING

3. *If the susceptibility ranking for your plant is within 5 EFPY of ONS3, addressees are requested to provide the following information:*

RESPONSE 3

As detailed in MRP-48, the susceptibility ranking of Indian Point 3 is moderate - greater than 5 EFPY of ONS3.

REQUEST 4 - MODERATE SUSCEPTIBILITY INFORMATION REQUEST

4. *If the susceptibility ranking for your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information:*

    a. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

    b. *your basis for concluding that the inspections identified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

        (1) *If your future inspection plans do not include a qualified (acceptable) visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

        (2) *The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.*

RESPONSE 4a

1. ENOI will conduct additional "above the insulation" inspections of the RPV head and nozzles during the next (RO12) refueling outage. These inspections will be conducted using an enhanced version of the Indian Point 3 CRDM nozzle inspection program originally described in References 18 and 19. This program was originally based on the guidance in NRC Generic Letters 88-05 and 97-01. ENOI will enhance this program to include the following elements:

    - Research and evaluate improved camera delivery systems to improve access for VT2 visual. This should facilitate inspections of greater than 60% of the VHPs.

    - Increased camera resolution will improve the ability of inspectors to see signs of leakage.

    - If leakage is identified during these visual examinations

        - A volumetric examination of the suspect area will be conducted to further characterize the flaw

        - "Extent of condition" inspection sample size will be increased based on MRP and/or plant specific recommendations.

        - Repairs will be performed, as required, to meet acceptance criteria requirements.

2. A detailed inspection plan will be submitted to the NRC no less than 90 days before the start of the next (RO12) refueling. Refer to response 1(d) for additional information regarding the Indian Point 3 inspection program.

3. ENOI will continue to monitor the results of VHP inspections (visual and volumetric) conducted at similar commercial nuclear power plants. If the results of these examinations significantly increase the probability of PWSCC cracks in VHP at Indian Point 3, ENOI will consider expanding its inspection plans to include a volumetric sampling examination of the reactor vessel head.

4. ENOI will visually inspect (by VT2) any VHP that may be exposed to bare metal during an outage.

5. ENOI will assess the effectiveness of acoustic emission monitoring systems on the head during pressure testing.

Entergy is planning to form an alliance with a major NSSS vendor to develop new state-of-the-art tooling and methods to enable volumetric examination at known susceptible cracking areas of CRDM penetrations. This alliance will include the development of mitigation and repair methods.

JUSTIFICATION FOR PROPOSED INSPECTION PLAN

Asbestos Head Insulation

Vessel head insulation at Indian Point 3 cannot be readily removed for inspection access. The insulation conforms to the contour of the vessel head and is covered by layers of asbestos tape and coated with asbestos cement. Voids were filled with asbestos cement prior to application of asbestos tape. Plants with more typical insulation configurations, such as raised reflective metal insulation, have a gap above the vessel head that allows inspection underneath. Plants with insulation blankets sitting on the vessel head allow easier removal than the insulation installed at Indian Point 3.

Removing reactor vessel head insulation at Indian Point 3 would include the following:

- Removal of obstructions, such as sections of shroud, and other present interferences, to gain access.
- Destructive removal of insulation, by sections, with high personnel asbestos and radiation exposure.
- Management of asbestos issues, including contaminated airborne particles.
- Potential damage to vessel head due to destructive removal, by tooling, of sections of adhered insulation.
- Potential destruction of visual leakage evidence.
- Complete vessel head clean up to establish base line for future visual inspections.
- Disposal of contaminated hazardous material.
- Design, procurement and installation of a new insulation package for ease of future inspections.

Personnel Radiation Exposure - ALARA

Asbestos management techniques are generally slow and arduous. The level of effort to gain access to the bare metal of the vessel head would be dose intensive and would result in considerably more cumulative personnel radiation exposure than has been incurred by plants with more typical insulation configurations. The proposed inspection plan is consistent with NRC Bulletin 2001-01, which states that

> *"...Licensees should ensure that all activities related to the inspection of VHP nozzles and the repair of identified degradation are planned and implemented to keep personnel exposures as low as reasonably achievable (ALARA), consistent with the NRC ALARA policy."*

Indian Point 3's Susceptibility to Primary Water Stress Corrosion Cracking

Indian Point 3 is considered to have a moderate susceptibility to PWSCC based upon the MRP susceptibility evaluation ranking. MRP-48 (Reference 3) Table 2-1 indicates that it would take Indian Point 3 several additional EFPYs from March 1, 2001, to reach the same time at temperature as Oconee 3 at the time that leaking nozzles were discovered. Even with the application of a 10-year uncertainty margin, there is time plan and implement an efficient and cost-effective inspection.

Vessel Head Designer and Manufacturer

Westinghouse designed Indian Point 3 reactor vessel. It was built by Combustion Engineering. While shallow indications have been identified at some units to date, none of the inspected CRDM penetrations at domestic Westinghouse or Combustion Engineering operating units have detected any leaking penetrations.

Risk Aspects of Deferred Inspections until RO13

A qualitative review of the potential risk in core damage frequency (CDF) at Indian Point 3 associated with delaying the inspection until RO13 shows that the risk is very small. This conclusion is based on several factors. First, no circumferential cracks have been observed from the 830 nozzle penetrations inspected to-date for Combustion Engineering and Westinghouse plants. The leaks that were discovered at B&W plants had significant structural margin remaining. In addition, several other plants in the high susceptibility group had no evidence of leaking. The worst case crack found at a high susceptibility plant had a remaining ligament safety factor of about 6 to failure. Loss-of-coolant accidents are analyzed events and procedures are in place to mitigate their consequences.

A quantitative screening was also performed which showed the increase in CDF, as well as the increase in core damage probability (CDP) due to delaying the inspection until refueling outage RO13, to be very small and within the acceptance criteria contained in NRC Regulatory Guide 1.174. This conclusion is based on conservative estimates of the probabilities for small, medium and large-break LOCAs (and assuming CRDM ejection) caused by circumferential cracks and taking into account the corresponding conditional core damage probabilities. These estimates were based on inspection data available as of the date of this submittal.

RESPONSE 4b(1)

The technical basis for concluding that regulatory bases are met for Indian Point 3 is provided in MRP-48 (Reference 3).

Compliance with the regulatory documents referred to in NRC Bulletin 2001-01 or MRP-48 is detailed in the Indian Point 3 UFSAR, and other plant-specific licensing bases documents. The general design criteria (GDC), as outlined in Bulletin 2001-01, 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.

RESPONSE 4b(2)

See response 4a.

REQUEST 5 - REFUELING OUTAGE PLANS

5. *Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage:*

   a. *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

b. *if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.*

RESPONSE 5

Entergy Nuclear Operations, Inc. will submit the information requested within 30 days after plant restart from the next refueling outage, which is currently scheduled to begin in April 2003.

REFERENCES

1. NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Vessel Pressure Vessel Head Penetration Nozzles," dated August 3, 2001.

2. TP-1001491, Part 2, PWR Materials Reliability Project Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44), Interim Report, May 2001

3. PWR Material Reliability Program Response to NRC Bulletin 2001-01 (MRP-48), EPRI, Palo Alto, California, 2001. TP-1006284.

4. Drawing E-234-058-1, Closure Head Insulation Assembly & Details

5. Entergy Procedure, 3PT-R114, RCS Boric Acid Leakage and Corrosion Inspection, Revision 6 (RO11)

6. PEP-RAP-2001-048, CRDM/RPV VT-2 Inspection, May 11, 2001

7. Drawing 9321-F-13553-15, Containment Biological & Missile Shield Walls Above El. 95'-0'

8. Indian Point 3 Updated FSAR, Section 4.5.4, "Missile Protection"

9. IUP-9361, Indian Point 3-CRD Missile Shield Design Basis and CRDM Cooling Fan Support Brackets, by UE&C, April 7, 1993

10. IUP-7755, Indian Point 3-Seismic Analysis of Reactor Vessel Head Shield, by UE&C, April 21, 1987

11. NRC Safety Evaluation Report by the Directorate of Licensing, U.S. Atomic Energy Commission, in the matter on Consolidated Edison Company of New York, Inc. Indian Point 3, Docket No. 50-286, September 21, 1973

12. Combustion Engineering, Inc. Report No. 1122, Analytical Report for Indian Point Reactor Vessel Unit No. 3, June, 1969.

13. Drawing 9321-F-30773, Sheet 1, Conduit Layout Containment Building El. 95' - 0"; Drawings 9321-F-30993-5, 9321-F-31433-1 and 9321-F-32453-2, Cable Arrangement Reactor Head, Sheets 1, 2 and 3.

14. NEI letter, A. Marion to B. Sheron, dated August 21, 2001 regarding "Generic Information for Use by Licensees in Response to Generic Letter 2001-01, Project Number 689.

15. Westinghouse drawing 110E193, Indian Point Plant Unit No. 2/3, CRDM Seismic Support Frames Details, Rev. 3.

REFERENCES (cont'd)

16. NRC letter, C. Cowgill to R. Barrett regarding NRC Inspection Report 50-286/01-05 dated July 27, 2001.

17. Entergy memorandum, S. Rokerya to P. Kokolakis, (IP-LIC-01-033) dated May 23. 2001 regarding notes from the NRC ISI inspection exit meeting held on May 23, 2001.

18. NYPA letter, H. P. Salmon to USNRC, (IPN-97-055), dated April 29, 1997, regarding "30-day response to NRC Generic Letter 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations."

19. NYPA letter, J. Knubel to USNRC, (IPN-97-097) dated, July 21, 1997 regarding "120-day response to NRC Generic Letter 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations."

20. Westinghouse Nuclear Services Division letter, S. M Ira to P. Okas (ENOI), (INT-01-033) dated August 29, 2001 regarding identification of proper drawings for Indian Point 3 CRDM supports.

Image T1-2106
These penetrations show no evidence of leakage or corrosion products.

Image T1-2924
These penetrations show no evidence of leakage or corrosion products.

Image T2-1330
This penetration shows no evidence of leakage or corrosion products. The white deposit on the woven insulation and joint is insulation repair material.

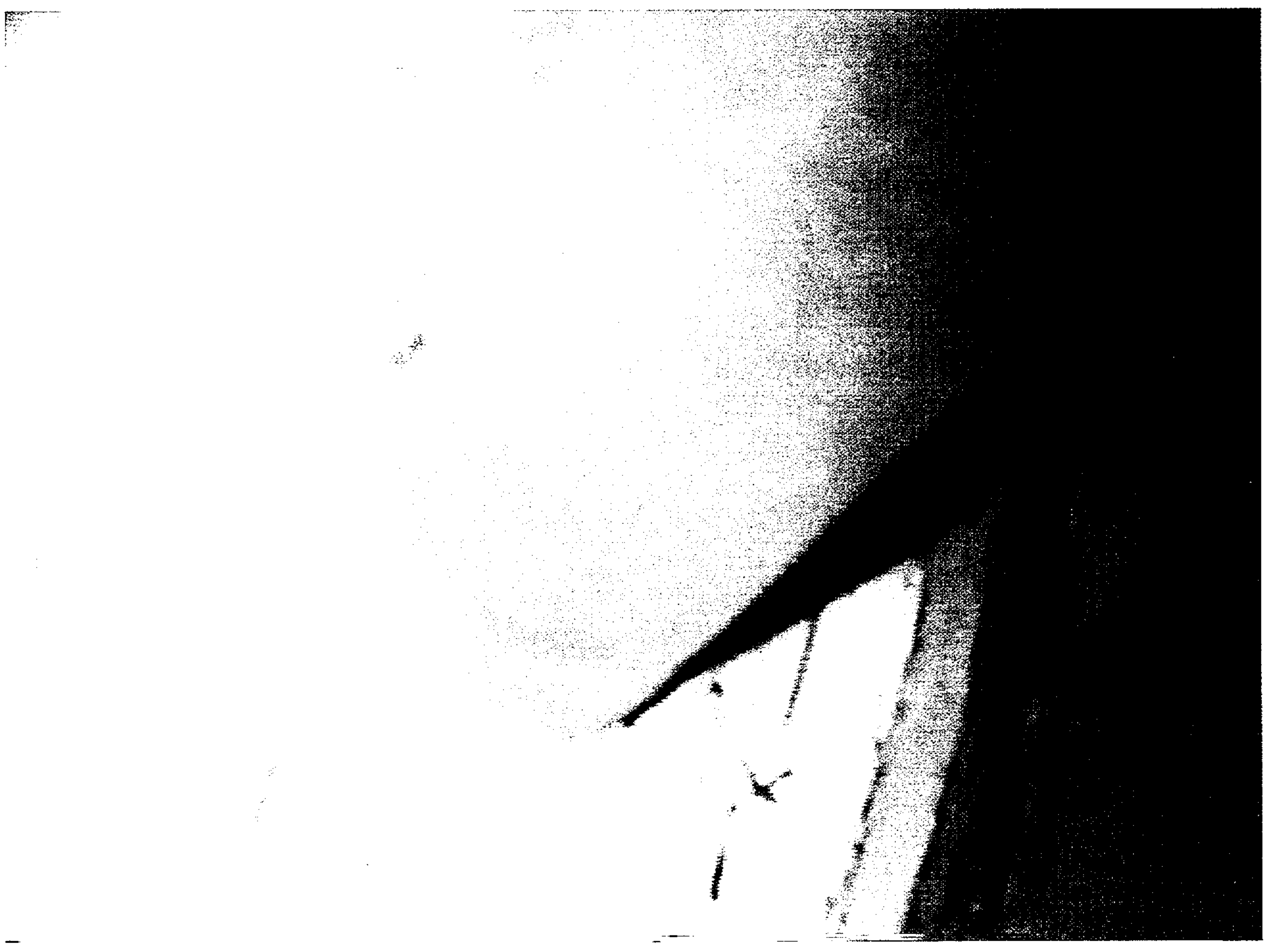


Image T2-1811
This penetration joint shows no evidence of leakage or corrosion products. The white deposit on the penetration insulation is insulation repair material.

Image T2-1820
This penetration joint shows no evidence of leakage or corrosion products.

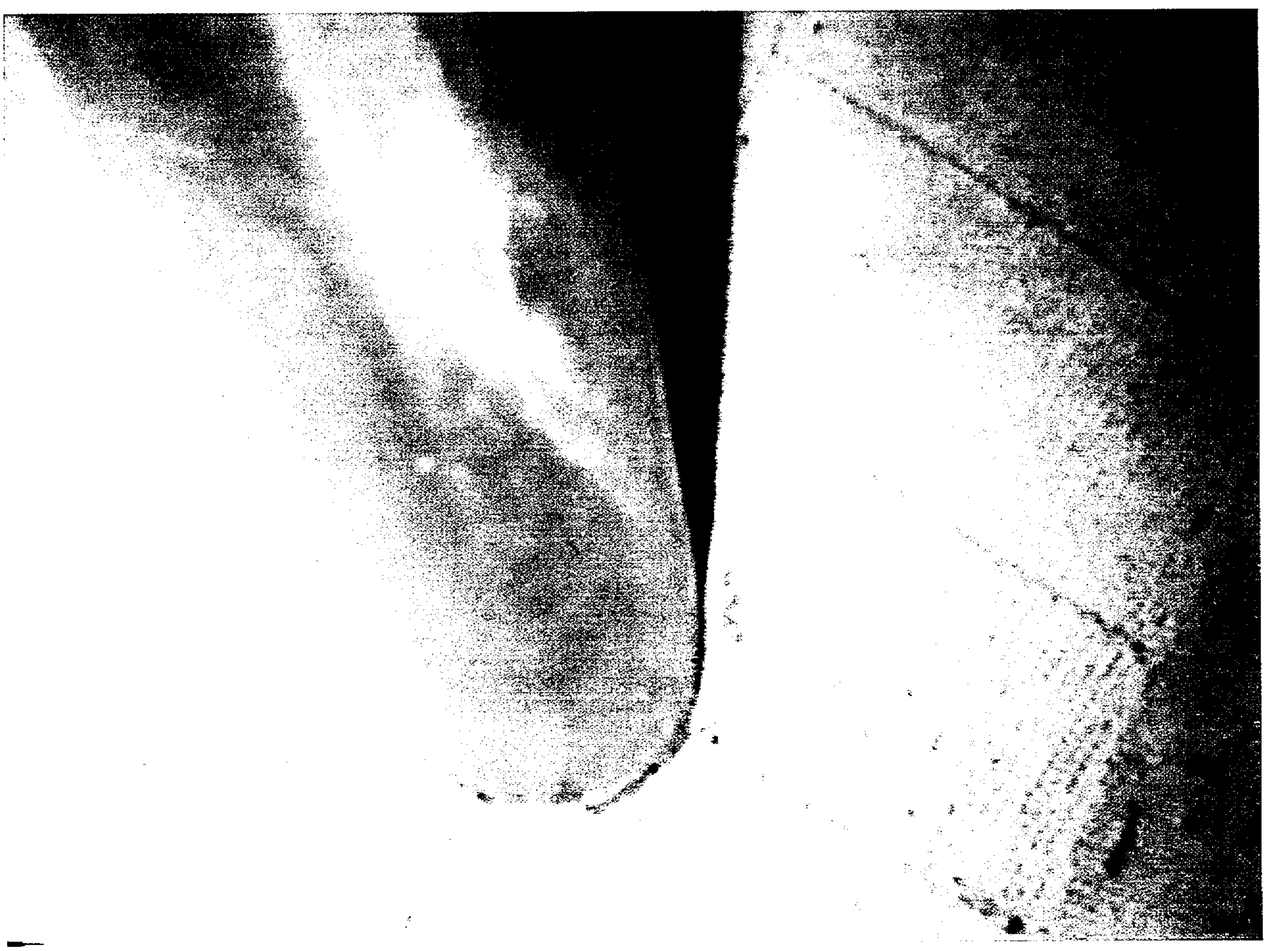


Image T2-2015

This penetration shows no evidence of leakage or corrosion products. The white deposits on the woven insulation are insulation repair material.

Image T2-2338

This penetration shows no evidence of leakage or corrosion products. The white deposits on the insulation, joint and penetration are insulation repair material.

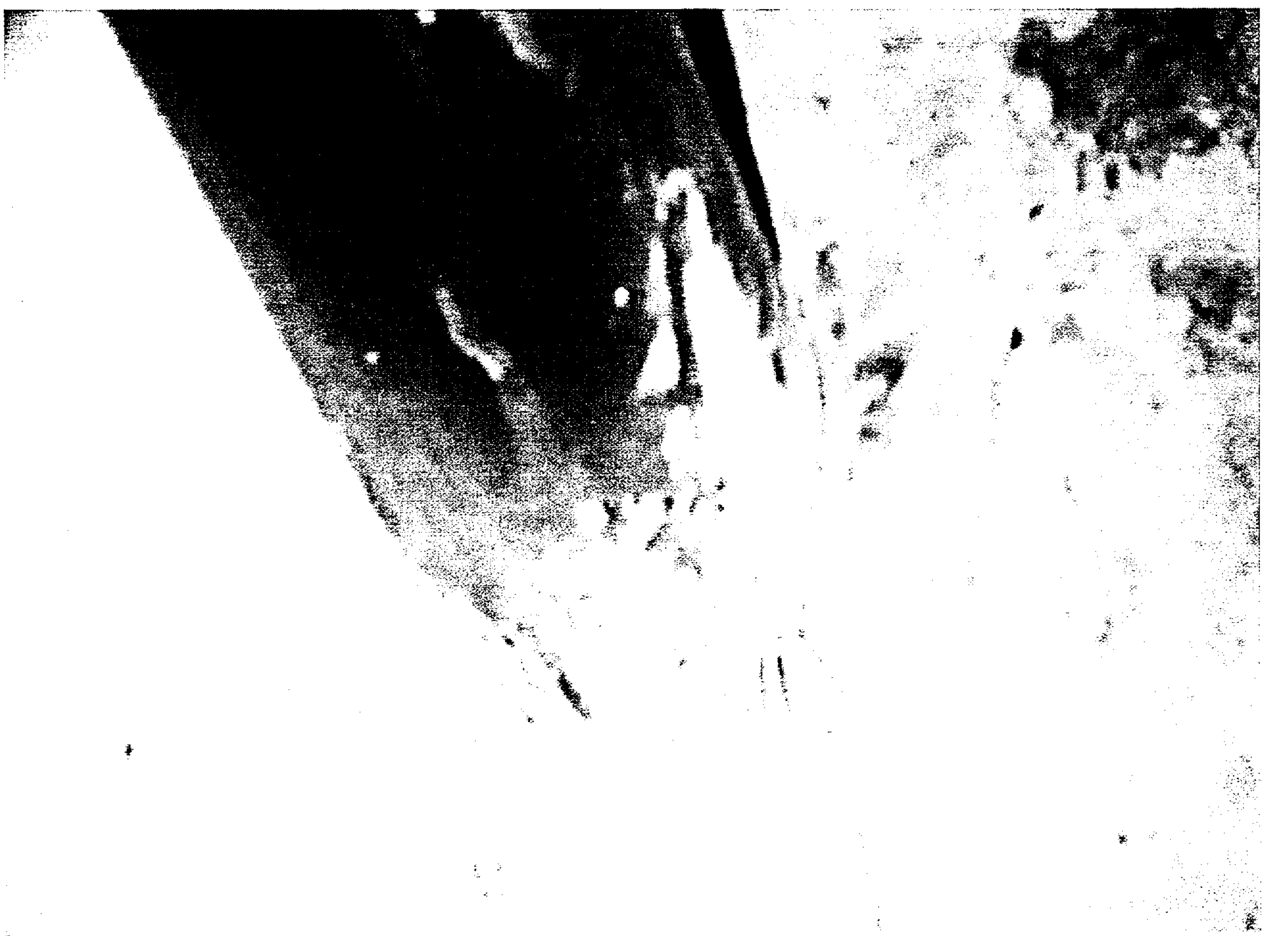
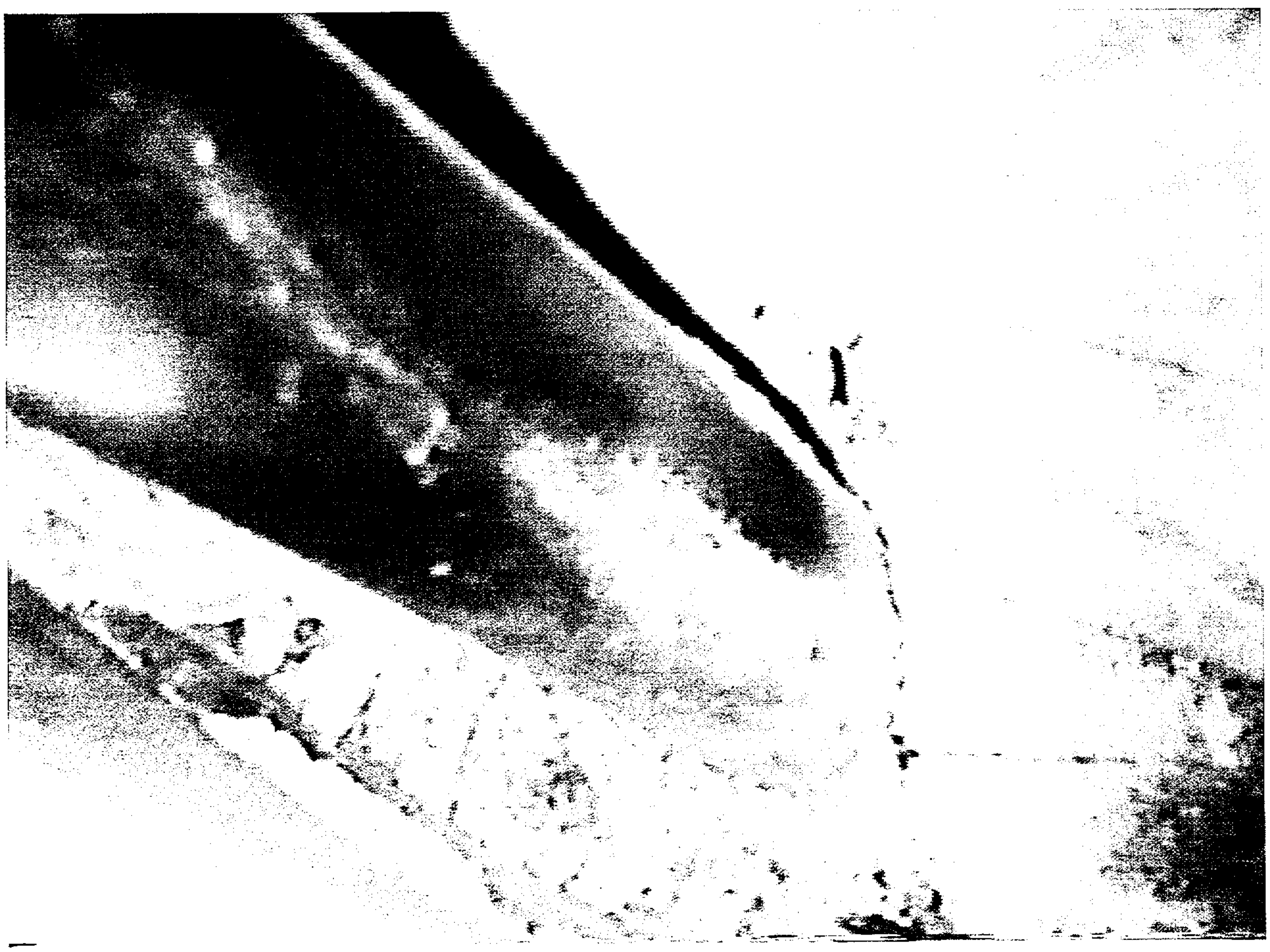


Image T2-2608

This penetration joint shows no evidence of leakage or corrosion products. The white deposits on the insulation, penetration and shroud are insulation repair material.



Summary of Commitments

Commitment ID	Description	Due Date
IPN-01-063-01	Provide the following information: a. a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected; b. if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.	Within 30 days after plant restart following the next refueling outage

IPN-01-063-02	ENOI will conduct additional "above the insulation" inspections of the RPV head and nozzles. These inspections will be conducted using an enhanced Indian Point 3 CRDM nozzle inspection program that was originally based on the guidance in NRC Generic Letters 88-05 and 97-01. ENOI will enhance this program to include the following elements:
	
	• Research and evaluate improved camera delivery systems to improve access for VT2 visual. This should facilitate inspections of greater than 60% of the VHPs.
	
	• Increased camera resolution will improve the ability of inspectors to see signs of leakage.
	
	• If leakage is identified during these visual examinations
	
	    • A volumetric examination of the suspect area will be conducted to further characterize the flaw
	
	    • "Extent of condition" inspection sample size will be increased based on MRP and/or plant specific recommendations.
		During the next (RO12) refueling outage.

Commitment ID	Description	Due Date
	• Repairs will be performed, as required, to meet acceptance criteria requirements.	
IPN-01-063-03	ENOI will continue to monitor the results of VHP inspections (visual and volumetric) conducted at similar commercial nuclear power plants. If the results of these examinations significantly increase the probability of PWSCC cracks in VHP at Indian Point 3, ENOI will consider expanding its inspection plans to include a volumetric sampling examination of the reactor vessel head.	Ongoing.
IPN-01-063-04	ENOI will visually inspect (by VT2) any VHP that may be exposed to bare metal during an outage.	As-required.
IPN-01-063-05	ENOI will assess the effectiveness of acoustic emission monitoring systems for on the head during pressure testing.	Ongoing.
IPN-01-063-06	Final VHP inspection plans will be submitted to the NRC staff.	Ninety-days before the start of the next refueling outage.