May 12, 2005

Mr. Fred R. Dacimo
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000286/2005002

Dear Mr. Dacimo:

On March 31, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Unit 3 (IP3). The enclosed integrated inspection report documents the inspection findings, which were discussed on April 21, 2005, with Mr. Chris Schwarz and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of the inspection, two findings of very low safety significance (Green) were identified. These findings were determined to be violations of NRC requirements. However, because of their very low safety significance, and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at Indian Point 3.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the

NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Brian J. McDermott, Chief
Projects Branch 2
Division of Reactor Projects

Docket No. 50-286
License No. DPR-64

Enclosure: Inspection Report No. 05000286/2005002
w/Attachment: Supplemental Information

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-286
License No.	DPR-64
Report No.	05000286/2005002
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Unit 3
Location:	295 Broadway, Suite 3 Buchanan, NY 10511-0308
Dates:	January 1, 2005 - March 31, 2005
Inspectors:	T. Hipschman, Senior Resident Inspector P. Habighorst, Senior Resident Inspector R. Berryman, Resident Inspector M. Cox, Resident Inspector J. Schoppy, Senior Reactor Inspector R. Nimitz, Senior Health Physicist J. Noggle, Senior Health Physicist C. Long, Reactor Inspector A. Passarelli, Reactor Inspector M. Snell, Reactor Engineer B. Wittick, Reactor Inspector
Approved by:	Brian J. McDermott, Chief Projects Branch 2 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000286/2005002; 01/01/2005 - 03/31/2005, Indian Point Nuclear Generating Unit 3; Problem Identification and Resolution, Event Followup, and Cross-Cutting Areas.

The report covers a 3-month period of inspection by resident inspectors and regional inspectors. Two Green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

Green A Green self-revealing non-cited violation of Technical Specification 3.7.11 was identified involving Entergy's failure to maintain the proper configuration of a damper actuator in the safety-related control room ventilation system. On January 26, 2005, during tracer gas testing, Entergy discovered that control room ventilation system damper B was operating in the reverse direction due to it's actuator and position indicator both being installed backwards. Entergy's investigation determined that the actuator and position indication were installed backwards during maintenance in 2001. As a result of the damper's actuator being reversed, the control room ventilation system would not have protected operators from toxic gases.

This finding is more than minor because Entergy failed to meet Technical Specification 3.7.11, "Control Room Ventilation System," which states that two control room ventilation system (CRVS) trains shall be operable. Contrary to this requirement, due to the improper installation of damper B, the CRVS was considered inoperable since May 5, 2001. Entergy's failure to properly maintain the proper configuration of the CRVS was determined to have very low safety significance (Green) based on a Phase 3 analysis in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Although the damper misalignment represented a degradation of the barrier function of the control room against smoke and/or toxic gas intrusion, compensatory measures are pre-planned. In addition, the control room dose limits per 10 CFR 50, Appendix A, General Design Criteria (GDC) 19 would not have been exceeded during a design basis event. Because this failure to maintain the CRVS was entered into the licensee's corrective action program (reference CR-IP3-2005-00315), this violation is being treated as an NCV consistent with Section VI.A. of the NRC Enforcement Policy. (Section 4OA3)

Cornerstone: Public Radiation Safety

Green A Green self-revealing non-cited violation of 10 CFR 20.2001 was identified associated with the transfer of waste, by Entergy's Indian Point Energy Center, for disposal, that did not meet Barnwell Low-Level Waste Disposal facility license requirements as required by 10 CFR 30.41. Specifically, a shipment (0205-12578) of low-level radioactive waste, from the Indian Point Energy Center, was identified on February 11, 2005, at the Barnwell Low-level Waste Disposal Facility, to have loose radioactive waste material inside the shipping cask (and outside of the waste disposal container) contrary to the disposal facility's site operating license (License No. 097, Amendment 47, Condition 61).

This finding is considered to be more than minor because Entergy failed to meet a waste disposal facility license requirement that was reasonably within its ability to foresee, correct, and prevent. This radioactive material control transportation finding was evaluated against criteria specified in NRC Manual Chapter 0609, Appendix D, and determined to be of very low safety significance (Green) because: 1) no external radiation or contamination limits were exceeded; 2) no package breach was involved; 3) no failure to make a notification was involved; and 4) although a low-level burial ground non-conformance was involved, burial ground access was not denied and no 10 CFR 61.55 waste classification issue was involved. In addition, although the finding did involve a certificate of compliance issue; the finding was a minor contents deficiency with low risk significance relative to causing a radioactive release to the public or public or occupational exposure. The small quantity of waste material was contained within the NRC approved shipping cask. Entergy temporarily suspended this type of shipment from the Indian Point Energy Center and placed the issue in the corrective action program. (Section 4OA2).

B. Licensee-Identified Violations.

None.

REPORT DETAILS

Summary of Plant Status

Indian Point 3 (IP3) operated at or near full power until March 10 and 11, 2005, when the unit reduced power for pre-outage testing. On March 12, 2005, the unit shut down for a planned refueling outage (3R13) and remained shutdown through the remainder of the inspection period.

1. **REACTOR SAFETY**

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope (71111.01 - 2 samples of actual adverse weather)

The inspectors reviewed Entergy procedure OAP-048, "Seasonal Weather Preparation," and 3-COL-RW-2, "Service Water System" to verify that Entergy completed these procedures in accordance with procedural requirements. During the week of January 3, the inspectors performed a risk-informed sample to independently verify that Entergy's actions to assure freeze protection of plant equipment were completed due to the very low ambient temperatures, snow, and icy conditions during that period. The inspectors performed walkdowns of accessible areas of the Unit 3 auxiliary feedwater building, emergency diesel generators (EDGs), and service water (SW) intake structure to assess the adequacy of freeze protection measures. The inspectors also looked for any vulnerable components not previously identified by Entergy. Specifically, from February 1 to February 9, the inspectors reviewed the effects of freezing weather on the condensate storage tank breather valves (CR-IP3-2005-00366).

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope (71111.04Q - 4 samples)

Partial System Walkdowns. The inspectors performed system walkdowns during periods of system train unavailability in order to verify that the alignment of the available train was proper to support the availability of safety functions, and to assure that Entergy had identified and properly addressed equipment discrepancies that could potentially impair the functional capability of the available train. The specific information reviewed to verify correct system alignment is referenced in the Supplemental Information attachment at the end of this report. The following system walkdowns were performed:

- On January 5-6, 2005, the inspectors performed a partial system walkdown of the SW pumps, traveling screens, and strainer pit while the 34 SW pump was unavailable for planned maintenance. The inspectors reviewed system drawings and checkoff lists to verify proper alignment of risk-significant SW valves at the SW intake and in the EDG rooms. In addition, the inspectors walked down the 480V vital switchgear and independently verified several SW work clearance tags. The inspectors also walked down risk-significant SW components following the maintenance to ensure that the SW system was returned to an operable condition.
- On January 11, 2005, the inspector performed a partial system walkdown of the 31 and 32 instrument air system during and after the maintenance on the 33 instrument air compressor.
- On February 22 and March 29, 2005, the inspector performed a partial system walkdown of the spent fuel pool cooling and back-up spent fuel pool systems to determine their readiness and performance during high heat load conditions during the full core offload during the 3R13 refueling outage.
- On March 22, 2005, the inspector performed a partial system walkdown of the service water supply to the 31, 32 and 33 EDGs during piping repairs that required the installation of a temporary service water discharge.

1R05 Fire Protection (71111.05Q - 12 samples)

a. Inspection Scope

The inspectors toured areas that were identified as important to plant safety and risk significant. The inspectors consulted Section 4.0, “Fire,” and the top risk significant fire zones in Table 4.4.4.2, “Core Damage Frequency for Fire Zones,” within the Indian Point 3 Individual Plant Examination of External Events (IPEEE). The objective of this inspection was to determine if Entergy had adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, and had adequately established compensatory measures for degraded fire protection equipment. The inspectors evaluated conditions related to: 1) control of transient combustibles and ignition sources; 2) the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and 3) the fire barriers used to prevent fire damage or fire propagation. Reference material used by the inspectors to determine the acceptability of the observed conditions in the fire zones are referenced in the Supplemental Information section of this report. The areas reviewed were:

- Fire Zones: 60A, 73A on January 3
- Fire Zones: 7A, 74A on January 3
- Fire Zone 15 on January 7
- Fire Zones: 37A, 40A on January 14
- Fire Pre-Plan 371 on January 25
- Fire Pre-Plan 373 on January 25
- Fire Zone 14 on January 28
- Fire Zone 20 on February 1

- Fire Zone 1 on February 11
- Fire Zone 4 on February 17
- Fire Zone 35A on February 25
- Fire Zone 5 on March 16

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope (71111.06 - 1 internal sample)

The inspector reviewed Entergy’s internal flood analysis, flood mitigation procedures and design features to verify whether they were consistent with IP3's design requirements. The inspector walked down selected internal plant areas that contained equipment important to safety. The inspector evaluated the condition and adequacy of mitigation equipment to assess whether flood protection design features were adequate and operable.

The inspector reviewed and toured the 73-ft, and 55-ft elevations of the PAB. The plant areas selected contained risk significant equipment based on the IP3 Individual Plant Examination for External Events (IPE), Appendix C, “Internal Flood Analysis.” Internal flooding initiated from a residual heat removal system service water line break from the 73-ft or 55-ft elevation of the PAB was an internal flooding scenario specifically evaluated by the IP3 IPE. The inspectors verified the accuracy of the descriptive text in the IPE and compared it with actual plant conditions in the PAB.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope (71111.08 - 6 Samples)

The inspector assessed the effectiveness of the licensee’s program for monitoring degradation of the reactor coolant system boundary. The inspection focused on the boric acid corrosion control and nondestructive examination activities on Class 1 & 2 piping as well as containment system boundaries. The steam generators did not undergo eddy current testing during 3R13. The operational assessment for the current operating cycle was reviewed to assure acceptability for not performing this testing.

For the nondestructive examination activities, the inspector conducted interviews with the ultrasonic (UT), surface and visual (VT) examination personnel and engineering personnel to assess the planning and preparation for the activities. The inspector reviewed training and qualification records to verify the licensee's personnel qualification process adequately prepared the assigned staff to perform the examination. The examination procedure was reviewed to determine whether it provided adequate guidance and examination criteria to implement the examination plan. For a manual UT on the main steam line, the inspector witnessed the calibration of the ultrasonic equipment and the demonstration of the procedure on a mock-up with built-in flaws similar to that expected in the materials to be examined to verify the UT equipment as calibrated would be able to find and accurately characterize flaws on the examined welds. The inspector observed documentation of UT results in the field for this exam.

Planning personnel involved with Class I piping exams were interviewed. For the UT indications found in the pressurizer auxiliary spray line, the inspector reviewed the weld repair and piping replacement design and made sure that the criteria used to expand examination scope was in accordance with the risk-informed program guidelines.

The inspector also reviewed the results of the manual UT's performed on the reactor head meridinal welds. Records of an indication left in service in a section of the reactor head-to-flange weld were reviewed to ascertain whether the flaw had changed in magnitude since it was first examined.

The inspector reviewed the results of a liquid penetrant exam on a pressurizer 6" line pipe support. The record was reviewed to ensure that the surface indication observed in the field was of acceptable size to leave in service, in accordance with code requirements.

The inspector assessed the ability of the licensee's inspection activities to identify boric acid corrosion and leaks. The licensee's boric acid inspection procedure was reviewed to determine if it provided adequate scope and guidance on examination criteria and corrective action required when boric acid deposits are found. An inspector conducted a boric acid walkdown of containment to verify that the licensee effectively inspected for active boric acid leaks. The inspector reviewed the licensee's boric acid walkdown report for indications of active boric acid leaks or boric acid corrosion of carbon steel components and the associated condition reports and corrective actions assigned.

The inspector observed licensee staff perform a visual exam of containment components including electrical penetrations. The procedure was reviewed as well as the resulting work orders documenting conditions of degradation to assure that the minimum amount of required visual exams were performed to meet the ASME code standards.

b. <u>Findings</u>

No findings of significance were identified.

1R11 Operator Requalification Inspection

a. Inspection Scope (71111.11Q - 1 sample)

The inspectors observed simulator training for licensed operators on Operations Team 3-A on February 24, 2005. The inspectors reviewed an “as found” simulator scenario to determine if the scenario contained: 1) clear event descriptions with realistic initial conditions; 2) clear start and end points; 3) clear descriptions of visible plant symptoms for the crew to recognize; and 4) clear expectations of operator actions in response to abnormal conditions.

During the simulator exercise, the inspectors evaluated the team’s performance for: 1) clarity and formality of communications; 2) correct use and implementation of emergency operating procedures (EOPs) and off-normal operating procedures (ONOPs); 3) operators’ ability to properly interpret and verify alarms; and 4) operators’ ability to take timely actions in a safe direction based on transient conditions. In addition, the inspectors evaluated the control room supervisor’s ability to exercise effective oversight and control of the crew’s actions during the exercise. The inspectors verified that the feedback from the instructors was thorough that they identified specific areas for improvement, and that they reinforced management expectations regarding crew competencies in the areas of procedure use, communications, and peer checking. The inspectors also evaluated Entergy’s post-scenario critique.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope (71111.12Q - 3 samples)

The inspectors evaluated Entergy’s work practices and follow-up corrective actions for selected systems, structures, and components (SSC) issues to assess the effectiveness of maintenance activities. The inspectors reviewed the performance history of those SSCs and assessed extent of condition determinations performed by Entergy personnel for those issues with potential common cause or generic implications to evaluate the adequacy of corrective actions. The inspectors reviewed problem identification and resolution actions for these issues identified by Entergy personnel to evaluate whether they had appropriately monitored, evaluated, and dispositioned the issues in accordance with Entergy's procedures and the requirements of 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance.” In addition, the inspectors reviewed selected SSC classification, performance criteria and goals, and Entergy’s corrective actions that were taken or planned, to verify whether the actions were reasonable and appropriate. The inspectors specifically reviewed the following samples in addition to the scope of this inspection:

- The inspector reviewed maintenance activities to correct deficiencies with the mechanical seals on the 33 safety injection pump. This included observation of the replacement of the inboard pump mechanical seal on January 28, 2005. The inboard mechanical seal was replaced in August 2004 and then again in January. The inspector discussed these maintenance activities with operations, engineering and maintenance personnel. Additionally, the inspectors reviewed maintenance history, post work and surveillance test data.

- The inspector reviewed maintenance activities to correct deficiencies with the structures at Indian Point 3. The licensee declared the structures at Indian Point 3 to be in (a) (1) status in accordance with 10 CFR 50.65 (a) (1) due to incomplete current inspection data of plant structures. This inspection activity included numerous field observations by the inspectors of safety-related and risk-significant structures to verify that any deficiencies were either being corrected or were dispositioned by the licensee's corrective actions program. The inspector discussed these maintenance activities with engineering personnel. Additionally, the inspectors reviewed licensee structural inspection reports and calculations.

- **The inspector reviewed maintenance activities related to the station batteries including open elective maintenance activities, condition reports, and work orders to ensure deficiencies were properly classified and corrected, maintenance rule failures were identified and properly scoped, and unavailability and unreliability was monitored. The inspector discussed corrective actions with operations, engineering, and maintenance personnel. Additionally, the inspectors reviewed maintenance, post work and surveillance test data.**

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control

a. Inspection Scope (71111.13 - 5 samples)

The inspector observed selected portions of emergent and planned maintenance work activities to assess Entergy's risk management in accordance with 10 CFR 50.65(a)(4). The inspector verified that Entergy took the necessary steps to plan and control emergent work activities, to minimize the probability of initiating events, and to maintain the functional capability of mitigating systems. The inspector observed and/or discussed risk management with maintenance and operations personnel. The specific information reviewed is referenced in the Supplemental Information attachment at the end of this report. The following three emergent and two planned activities were observed:

- WO IP3-05-10839: Planned replacement of feed flow steam flow mismatch bistable 3FC-438C.
- WO IP3-05-10565: Emergent repairs to plant vent radiation monitor R-27.
- WO IP3-05-16641: Emergent ultrasonic testing and venting of the RHR/SI systems.
- WO IP3-04-12504: Planned repairs to 345 KV Breaker. 1
- WO IP3-05-16478: Emergent repairs to fan cooler unit 32.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions and Events

a. Inspection Scope (71111.14 - 2 samples)

For the non-routine events described below, the inspectors reviewed operator logs, plant computer data, and strip charts to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures.

- On January 27, the inspectors observed control room tracer gas testing used to verify analysis inputs associated with Entergy's license change request for use of an alternate source term.
- On February 26, the inspector observed recovery to full power following an unexpected negative reactivity addition during operations to place the mixed-bed ion exchanger in service. On March 1, the inspector observed initiation of the mixed-bed demineralizer.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15 - 5 samples)

The inspectors selected a sample of Entergy's operability evaluations for review on the basis of potential risk significance. The operability evaluations selected as samples are associated with the CRs listed below. The inspectors assessed the accuracy of the evaluations, the use and control of compensatory measures, if needed, and compliance with the TSs. The inspectors' review included a verification that the operability evaluations were made as specified by procedure ENN-OP-104, "Operability Determinations." The technical adequacy of the evaluations was reviewed. References used during these reviews included the TS, the Technical Requirements Manual, the Final Safety Analysis Report (FSAR), and associated design basis documents. The specific information reviewed is referenced in the Supplemental Information attachment at the end of this report.

- CR-IP3-2005-00263 33 Auxiliary Boiler Feedwater Pump recirculation line low temperatures.
- CR-IP3-2005-00036 Errors in Proto-Flo flow calculations for various safety-related systems.
- CR-IP3-2005-00366 Condensate storage tank level indication errors.
- CR-IP3-2005-00649 Refueling water storage tank level indication errors.
- CR-IP3-2005-00315 32 Fan Cooler Unit degradation.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope (71111.17 - 1 sample)

The inspectors reviewed the engineering evaluation for the installation of a vent valve downstream of SI-MOV-888 A/B (ER-05-3-025) to verify that the design bases, licensing bases, and performance capability of risk significant SSCs have not been degraded through modifications. During performance testing of PT-Q127, “Periodic SI Venting,” a gas void was discovered downstream of the SI-MOV-888 A/B valves. Although the gas void was not of sufficient size to impact the operability of the residual heat removal (RHR) or safety injection (SI) systems, excessive gas buildup could result in pump gas binding and/or a pressure transient. Installation of the permanent modification provided a positive means to vent gas downstream of the valves.. The inspectors reviewed the calculations, test data and thermal analyses involved with this analysis. The inspectors also conducted walk-downs to compare installed equipment to the equipment that was analyzed.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope (71111.19 - 5 samples)

The inspectors reviewed PMT procedures and associated testing activities to assess whether: 1) the effect of testing in the plant had been adequately addressed by control room personnel; 2) testing was adequate for the maintenance performed; 3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing documents; 4) test instrumentation had current calibrations, range, and accuracy for the application; and, 5) test equipment was removed following testing.

The selected testing activities involved components that were risk significant as identified in the IP3 Individual Plant Examination. The regulatory references for the inspection included TS 6.8.1.a and 10 CFR 50, Appendix B, Criteria XIV, "Inspection, Test, and Operating Status." The specific information reviewed is referenced in the Supplemental Information attachment at the end of this report. The following testing activities were evaluated:

- Tracer gas testing of the central control room following reconfiguration and sealing, WO IP3-04-13924.
- Nuclear Instrument 31 following circuit card replacement, WO IP3-02-24552.
- Control rod drive mechanism meggaring following splicing, WO IP3-05-14074.
- Safety injection header safety relief valve testing SI-855, WO IP3-02-15919.
- Inservice leak testing of the 31 EDG starting air receiver after cleaning, WO IP3-02-22880.

n. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20 - 1 sample)

a. Inspection Scope

The inspectors evaluated, observed and verified a number of activities associated with 3R13. The refueling outage occurred between March 12 through April 6, 2005.

Outage Risk Control Plan

The inspectors reviewed Entergy's refueling outage risk assessment activities to ensure that appropriate consideration was given to minimize the unavailability or mitigate/compensate for reduced reactivity control, core cooling, power availability, containment integrity, spent fuel cooling, and inventory control attributes. The inspectors observed that Entergy conducted a qualitative evaluation of the daily risk associated with planned outages of both safety and non-safety related systems which contribute to these six attributes. In addition, Entergy assigned an overall risk characterization based upon the collective risk of all those systems out-of-service. The inspectors reviewed Entergy's daily outage risk assessments to assess Entergy made some changes to the outage schedule and "Defense in Depth Contingency Plans" for those outage configurations which could not be otherwise modified to minimize the overall risk.

Monitoring of Plant Shutdown and Cooldown Activities

The inspectors observed control room and plant activities during the plant shutdown on March 12, 2005 The inspectors verified the operators took timely and appropriate actions per emergency operating procedures E-0 and ES-0.1 when the reactor was manually tripped as part of the normal shutdown sequence.

The inspectors observed the operators conducting the shutdown using procedures 3-POP 3.1, “Plant Shutdown,” and 3-POP 3.3, “Plant Cooldown,” and controlled plant parameters within the requirements of Technical Specifications. The inspectors verified that operators used the appropriate pressure and temperature instrumentation during cooldown. The reactor was cooled down below 350°F using the auxiliary feedwater (AFW) system until RHR was placed in service. The plant entered cold shutdown with the reactor coolant system (RCS) less than 200°F on March 12.

The inspectors observed the operators response to changing plant conditions. Specifically, inspectors verified operators’ block of the SI signal during cooldown, and verified appropriate soluble boron concentration to preserve shutdown margins in the core.

Control of Outage Activities

The inspectors performed walkdowns of various areas and systems during 3R13. Areas specifically evaluated during the outage were:

- Containment to perform a boric acid walkdown of the RCS
- EDG building
- EDG fuel oil transfer system
- Normal and back-up spent fuel pool (SFP) cooling system
- RHR system
- Low temperature overpressure protection system and controls
- Primary auxiliary building (PAB)
- AFW building
- Turbine building

During 3R13 the inspectors periodically verified adequate shutdown margin in accordance with technical specifications. The inspectors independently verified the adequacy of system tagout isolation and configuration controls. Specific items verified included:

- Low temperature overpressure protection injection source isolation
- Instrument Bus Circuit 17. Following application of the tagout for the 31 Main Transformer, the control room lost control of the high pressure steam dumps and they went shut. Cooldown was maintained on the atmospheric relief valves, and the high pressure steam dumps were restored approximately 6 minutes later (CR IP3-2005-00989).
- Isolation of component cooling water and charging to the reactor coolant pumps during pump backseat evolution
- Isolation of non-essential service water to the EDGs

- Isolation of component cooling water and charging to the reactor coolant pumps during 33 RCP CCW line weld repairs.

The inspectors periodically verified configuration management controls, including maintenance of defense-in-depth commensurate with the outage safety plan (OSP) for key safety functions and compliance with the applicable TSs when taking equipment out of service. Specific items verified included:

- Protected component cooling water and RHR pumps and heat exchangers
- EDGs while electrical power risk was elevated during offsite transmission work
- Periodic review of 3-PT-W019, “Electrical Verification - Offsite Power Sources and AC Distribution”

The inspectors periodically verified through control room and system walkdowns operation of the shutdown cooling system and found their observations to be consistent with system operating procedures and TS requirements. The inspectors periodically verified proper operation of the SFP cooling system.

Reduced Inventory and Mid-Loop Conditions

On March 17, 2005 inspectors observed reduced inventory operations, and on March 31, 2005 inspectors observed mid-loop operations. The inspectors verified plant configurations were consistent with commitments from NRC Generic Letter 88-17. The inspectors focused on unexpected conditions or emergent activities that could impact operators’ ability to maintain required reactor vessel level.

Refueling Activities

On March 19 and 20, 2005; and March 26 - 28, 2005, the inspectors observed refueling activities on the containment manipulator crane, containment fuel transfer system, the SFP, and the control room. The inspectors observed that foreign material exclusion was being maintained in the vicinity of the SFP and the reactor cavity. The inspectors verified that fuel loading was performed in a manner documented by the refueling manual as per design.

Plant Heatup and Startup Activities

The inspectors observed a number of plant restart activities within the control room, and conducted walkdowns of the containment, PAB, and the auxiliary feedwater (AFW) pump building. The specific activities, in part, included containment cleanliness, RCS leakage calculations, containment integrity, plant heat-up, and selected safety system alignment verifications.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22 - 8 samples)

The inspectors observed portions of the surveillance tests listed below and reviewed the test procedures to assess whether: 1) the test preconditioned any of the components; 2) the effect of the testing was adequately addressed in the control room; 3) the scheduling and conduct of the tests were consistent with plant conditions; 4) the acceptance criteria demonstrated system operability consistent with design requirements and the licensing basis; 5) the test equipment range and accuracy were adequate for the application, and the test equipment was properly calibrated; 6) the test was performed in the proper sequence in accordance with the test procedure; and, 7) the affected system was properly restored to the correct configuration following the test. The specific information reviewed is referenced in the Supplemental Information attachment at the end of this report.

- 3PT-M096C, “Containment Spray and Spray Addition System Monthly Alignment Verification,” Rev. 3, on January 9, 2005.
- 3PT-W001, “Emergency Diesel Generator Support System Inspection,” Rev. 37, on January 10, 2005.
- 3PT-Q99A, “Containment Pressure Functional Test - Channel IV,” Rev. 1, performed on January 5, 2005.
- 3PT-Q120C, “33 ABFP (Motor Driven) Surveillance and IST,” Rev. 7 performed on January 12, 2005.
- 3PT-Q100C, “Steam Flow/Feed Flow Mismatch Functional Test,” Rev. 3, on February 17, 2005.
- 3PT-R007B, “32 Auxiliary Boiler Feedwater Pumps Full Flow Test,” Rev. 12 performed on March 11, 2005.
- **3PT-R156B, “Station Battery 32 Load-Profile Service Test,” Rev. 8, performed on March 21, 2005**.
- 3PT-R160A, “31 EDG Capacity Test,” Rev 8, performed on March 29, 2005.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23 - 2 samples)

The inspector reviewed documentation on Temporary Alteration No: TA-04-3-090 "Disable the pressurizer spray line loop low temp alarm for TC451-X." The pressurizer spray line loop low temperature alarm normally receives inputs from temperature elements upstream of both spray valves PCV-455A and PCV-455B. Spray valve PCV-455A was isolated due to a valve packing leak at the time of the inspector's review. This resulted in a no flow condition through this spray line and as a result the temperature upstream of PCV-455A was below the alarm setpoint. The modification involved removing the wire to the alarm relay to prevent the low temperature upstream of PCV-455A from causing a locked-in "Pressurizer Spray Line Loop Low Temp." annunciator in the Central Control Room (CCR).

The inspector also reviewed documentation on Temporary Alteration No: TA-03-3-111 "Reconfigure Control Room Ventilation System During Tracer Gas Testing." The inspector reviewed the special system configuration which was allowed by TS LCO 3.7.11 for evaluation of system parameters in support of Entergy's license amendment request for the use of an alternate source term.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope (71114.06 - 1 sample)

The inspectors observed an emergency preparedness (EP) drill conducted on January 12, 2005. The inspectors used NRC Inspection Procedure 71114.06, "Drill Evaluation" as guidance and criteria for evaluation of the drill. The drill consisted of a steam generator tube leak followed by a loss of all on-site and offsite power. The inspectors observed the drill and conducted reviews from the participating facilities onsite, including the IP2 Plant Simulator, the Technical Support Center (TSC), and the Emergency Operations Facility (EOF). The inspectors focused the reviews on the identification of weaknesses and deficiencies in the classification and notification timeliness during the drill. The inspectors were briefed on Entergy's critique results and compared the NRC identified weaknesses and deficiencies to those identified by Entergy to ensure that problem areas were properly identified.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope (71121.01 - 11 samples)

During March 21-25, 2005, the inspector conducted the following activities during the IP3 Spring refueling outage to verify that the licensee was properly implementing physical, engineering, and administrative controls for access to high radiation areas, and other radiologically controlled areas, and that workers were adhering to these controls when working in these areas. Implementation of the access control program was reviewed against the criteria contained in 10 CFR 20, site technical specifications, and the licensee's procedures.

(1) The following exposure significant work areas were evaluated to determine if radiological controls (e.g., surveys, postings, and barricades) were acceptable.

- 204B charging valve bypass modification
- Safety injection valve 849C repair
- No. 32 reactor coolant pump seal replacement
- Scaffold building
- Radiation protection outage support

(2) Radiation work permits (RWPs) associated with the above work activities were reviewed with respect to high radiation area controls including electronic dosimeter alarm set points.

(3) With respect to the work activities listed in (1) above, walk downs of these work areas were conducted with a radiation survey instrument to determine whether radiation work permit (RWP), procedure, and engineering controls were in place, and whether licensee surveys and postings were complete and accurate, and that air samplers were properly located.

(4) Work activities listed in (1) above were reviewed against the radiological control requirements as specified in the applicable RWPs and ALARA reviews, as well as verbal instructions provided by RP technicians during radiological briefings to workers.

(5) With respect to the work activities listed in (1) above, the conduct of necessary system breach surveys and evolving radiological hazards associated with work activities were observed to evaluate the radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls.

(6) During observations of outage work activities listed in (1) above, radiation worker performance was evaluated with respect to radiological work requirements and radiological briefing instructions.

(7) During observations of outage work activities listed in (1) above, radiation protection technician performance was evaluated with respect to radiation protection procedure and work activity radiological surveillance requirements.

(8) Observation of outage scaffold work activity as a high radiation area work activity with significant dose rate gradients was reviewed with respect to exposure monitoring regulatory requirements.

(9) Based on the condition reports reviewed (See Section 4OA2), no repetitive deficiencies were identified for further followup.

(10) Condition reports reviewed (see Section 4OA2) were evaluated with respect to traceable trends in radiation worker performance.

(11) Condition reports reviewed (see Section 4OA2) were evaluated with respect to traceable trends in radiation protection technician performance.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

1. ALARA Outage Work Activities

a. Inspection Scope (71121.02 - 3 samples)

From March 21-25, 2005, the inspector conducted the following activities to verify that the licensee was properly maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and the licensee's procedures.

(1) Scheduled outage work activities were selected during the inspection period that were estimated to result in the highest collective exposures. These included:

- 3R13 outage valve work: 14.513 person-rem estimate
- Reactor coolant pump work: 6 person-rem estimate
- Scaffold building and inspections: 4.9 person-rem estimate
- Radiation protection support: 4.9 person-rem estimate

(2) Based on the work activities listed in (1) above, the conduct of these work activities was observed with respect to the licensee's use of engineering controls to achieve dose reductions.

(3) Based on the work activities listed in (1) above, the conduct of radiation worker and radiation protection technician performance was observed to evaluate if workers demonstrated ALARA in the performance of their work activities in these high dose areas.

b. Findings

No findings of significance were identified.

2. ALARA Review

a. Inspection Scope (71121.02 - 6 samples)

During January 31 through February 4, 2005, the inspector conducted the following activities to verify that the licensee was properly maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and the licensee's procedures.

(1) The plant collective exposure history trend and current 3-year rolling average collective exposure data was reviewed. Based on 2001-2003 exposure data, Indian Point Unit 2 performance of 94 person-rem, ranks in the third quartile, and Indian Point Unit 3 performance of 74 person-rem, ranks in the second quartile of U.S. pressurized water reactors.

(2) The following highest exposure work activities for the Unit 2 Fall 2004 refueling outage were selected for review.

- replace resistance temperature detectors
- outage scaffold support
- reactor disassembly / reassembly
- radiation protection support
- outage valve work
- temporary shielding
- fuel moves and associated work
- operations outage support

(3) The ALARA reviews for the outage work activities listed in (2) above were evaluated with respect to initial exposure estimates and any subsequent credits due to emergent work or increased dose rates, and then compared to the actual exposure results obtained. Any causes for exposure overruns were identified and quantified where appropriate.

(4) With respect to the ALARA reviews that were evaluated in (3) above, the methods for adjusting exposure estimates were reviewed relative to changes in work scope or increased dose rates in order to preserve the original work activity exposure performance measurement of the work activities.

(5) The site specific trend in source term was reviewed and found to be stable; approximately 70 mrem/hr average intermediate loop piping for Unit 2 and 32.5 mrem/hr for Unit 3. This compares favorably with the industry average of 100 mrem/hr.

(6) ALARA work planning and exposure estimates were reviewed for the upcoming Unit 3 Spring 2005 refueling outage. The refueling outage goal of 71 person-rem and its basis was reviewed along with the initial ALARA plans for the following work activities: reactor disassembly/reassembly, valve work, scaffold support, fuel moves and associated work, snubber inspections, pressurizer work, fuel transfer system repairs, reactor coolant pump work, and in-service inspection.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

1. Daily Review

a. Inspection Scope (71152)

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” and in order to help identify repetitive failures or specific human performance issues for follow-up, the inspectors screened all items entered into Entergy’s corrective action program. This review was accomplished by reviewing hard copies of each CR.

b. Findings

No findings of significance were identified.

2. PI&R Annual Sample - Control Rod Drive Mechanism (CRDM) Cable Splicing

a. Inspection Scope (71152 - 1 sample)

The inspectors entered the containment building during 3R13, to observe the condition of the CRDM power cables, and observed test results to verify proper control rod operation. The test consisted of meggaring the CRDMs and rod testing during start-up operations. The inspectors reviewed the scope of the testing to ensure it would encompass all repaired components and also reviewed the test data for completeness and accuracy.

During start-up on August 16, 2003 control rods G5 and E9 did not move when shutdown bank “A” was withdrawn during an attempted reactor start-up. Upon investigation, I&C determined that the power cables to the CRDMs for these rods had developed short-circuits in cable splices that were installed as part of a cable replacement modification in 1997. Entergy replaced eight cable splices (F8, G5, E9, B6, N9, C13, F14, and K10) that had shorted conductors, and opened a representative sample of eight other splices that were apparently not shorted to determine the extent of condition. Entergy attributed the splice failures to poor workmanship during the 1997 modification, and to the high temperature conditions around the reactor head (~130F) after the four CRDM fans and the five FCUs tripped following the loss of offsite power. This condition was initially documented in NRC Integrated Inspection Report No. 05000286/2003-008.

The inspectors evaluated Entergy’s corrective actions to ensure that they were appropriately focused to correct the identified problems. The procedures were reviewed to verify that appropriate changes had been made to properly implement the prescribed corrective actions. The inspectors also evaluated the changes for technical adequacy.

b. Findings

No findings of significance were identified.

3. PI&R Annual Sample - Barnwell Shipment

a. Inspection Scope (71152 - 1 sample)

The inspector reviewed the circumstances surrounding the shipment by Entergy, Indian Point 2, of low-level radioactive waste on February 7, 2005, to the Barnwell South Carolina waste disposal facility, that was found to be in nonconformance with the requirements of the State of South Carolina’s license (License No. 97, Amendment No. 47) issued to the Barnwell Waste Disposal facility.

The review was against requirements contained in 10 CFR 20 and applicable waste transfer requirements.

b. Findings

Introduction. A self-revealing non-cited violation of 10 CFR 20.2001 was identified associated with transfer of low-level radioactive waste, by Entergy Indian Point Energy Center for disposal, that did not meet Barnwell Low-Level Waste Disposal facility license requirements as required by 10 CFR 30.41. Specifically, during shipment unloading on February 11, 2005, at Barnwell, loose radioactive waste material (approximately 2 tablespoons), was identified within the annular space between the waste container and transport cask. Loose waste is prohibited by the facility license (License No. 097, Amendment 47, Condition 61) issued to Barnwell by the State of South Carolina.

Description. On February 11, 2005, personnel from the Barnwell waste disposal facility conducted an inspection of a shipment of radioactive waste (0205-12578) from the Entergy Indian Point Energy Center. Shipment 0205-12578 was a polyethylene liner filled with depleted filter media, placed inside an NRC-approved Type B shipping package (CNS 8-120B -2 cask). During off-loading and removal of the polyethylene liner (waste disposal container) from the cask at Barnwell, loose radioactive waste materials (approximately 2 tablespoons) were observed on the bottom of the shipping cask as the radioactive waste disposal package was removed from the cask. The waste material was collected, surveyed, and found to exhibit low radiation levels. Entergy Indian Point was subsequently notified by the Barnwell Low-Level Waste Disposal Facility that shipment 0205-12578, had radioactive waste materials outside the waste disposal container, contrary to the waste disposal facility's site operating license (License No. 097, Amendment 47, Condition 61). Entergy had packaged the shipment and was unaware that waste material was in the annular space between the shipping container (cask) and the waste disposal container (polyethylene liner).

Analysis. Failure to transfer waste to a licensed waste disposal facility, in accordance with the provisions of its disposal license, is a performance deficiency because, a requirement (disposal license condition) was not met by Entergy Indian Point which was reasonably within its ability to foresee and correct, and which should have been prevented.

The finding is not subject to traditional enforcement in that the finding did not have any actual safety consequence, did not have the potential for impacting the NRC's ability to perform its regulatory function, and there were no willful aspects.

The finding was greater than minor, in that it is associated with the program and process attribute (radioactive material control/transportation) of the Public Radiation Safety cornerstone and did affect the cornerstone. Specifically, Entergy Indian Point did not meet the general packaging conditions of the recipient's (Barnwell Disposal Facility) operating license and properly package waste for shipment to the waste disposal facility. Further, the NRC Certificate of Compliance (No. 9168, Revision 14) for this shipping cask specifically requires byproduct material, other than irradiated reactor components, to be contained within secondary containers. Failure to follow packaging conditions would not ensure adequate protection of public health and safety. The finding was evaluated against criteria specified in NRC Manual Chapter 0609, Appendix D, and determined to be of very low safety significance (Green), because: 1) no radiation or contamination limits were exceeded; 2) no package breach was involved; 3) no failure to make a notification was involved; and 4) although a low-level burial ground non-conformance was involved, burial ground access was not denied and no 10 CFR 61.55 waste classification issue was involved. In addition, although the finding did involve an NRC Certificate of Compliance issue, the finding was a minor contents deficiency with low risk significance relative to causing a radioactive release to the public or public or occupational exposure. The small quantity of loose waste was contained within the NRC -approved shipping cask. Entergy suspended similar shipments when notified and placed the issue in its corrective action program (CR-IP-2-2005-00613).

Enforcement. 10 CFR 20.01 and 10 CFR 30.41 require that the licensee may only transfer licensed materials to a person authorized to receive such material under terms of a specific license issued by an Agreement State. Condition 61 of License 097 (Amendment 48) issued for the operation of the Barnwell Waste Management Facility by the State of South Carolina (an Agreement State), prohibits loose radioactive waste residuals within shipping casks. Contrary to this requirement, loose radioactive waste material was found within the annulus space between the waste disposal container (polyethylene liner) and the shipping cask for Indian Point Energy Center shipment No. 0205-12578 on February 11, 2005. This is a violation of 10 CFR 20.2001.

Because this finding was of very low safety significance (Green), and Entergy Indian Point entered this finding into its corrective action program, this violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000286/2005002-01: Failure to transfer waste to a licensed waste disposal facility, in accordance with the provisions of its disposal license)**

4. PI&R Review for IP 71121

a. Inspection Scope (71121)

The inspector reviewed 13 corrective action condition reports that were initiated between February 2005 and March 20, 2005 and were associated with the radiation protection program. The inspector verified that problems identified by these condition reports were properly characterized in the licensee’s event reporting system, and that applicable causes and corrective actions were identified commensurate with the safety significance of the radiological occurrences.

b. Findings

No findings of significance were identified.

5. PI&R Review for IP 71111.08

a. Inspection Scope (71111.08)

The inspector reviewed a sample of issue reports that identified problems associated with inservice inspection. The inspector verified that the problems were accurately recorded in the issue reports and that the corrective action taken was appropriate. The condition reports reviewed are listed in Attachment 3 under “List of Documents Reviewed.”

b. Findings

No findings of significance were identified.

4OA3 Event Followup

7. (Closed) LER 05000286/2005-001-00, Plant in a Condition Prohibited by Technical Specifications due to Error Making Control Room Ventilation System Inoperable

Introduction. A Green self-revealing non-cited violation of Technical Specification 3.7.11 was identified involving Entergy's failure to maintain the proper configuration of a damper actuator in the safety-related control room ventilation system. On January 26, 2005 during tracer gas testing, Entergy discovered that control room ventilation system damper B was operating in the reverse direction due to it's actuator and position indicator both being installed backwards. Entergy's investigation determined that the actuator and position indications were installed backwards during maintenance in 2001. As a result of the damper's actuator being reversed, the control room ventilation system would not have protected operators from toxic gases.

Description. On January 26, 2005 during tracer gas testing, Entergy engineers and technicians discovered that control room ventilation system damper B was operating in the reverse direction due to its actuator and position indicator both being installed backwards. The result of this was that whenever the damper was selected to be shut, it was actually open and vice versa. The position indicator was also reversed which prevented operators from discovering the reverse operation of damper B.

Control room ventilation system damper B is normally selected to the shut position. In normal operation, operators would not notice a difference in control room ventilation system performance if damper B was not fully shut since the flow path from the outside environment through damper B is isolated downstream by the normally shut dampers F1, F2 and C. These dampers were installed properly.

Damper B is commanded open when the control room ventilation system is switched from normal operation to the "10% Incident Mode". In the "10% Incident Mode", damper B opens and allows approximately ten percent of the total control room ventilation flow to be taken from outside air and filtered through a charcoal bed before being circulated through the control room. The purpose of "10% Incident Mode" is to allow an amount of filtered air in to make sure that the control room is always maintained at a pressure greater than ambient during events that may result in a release of fission products to ensure the dose to the operators does not exceed the 10 CFR 50, General Design Criterion 19 limits.

Damper B is commanded shut when the control room ventilation system is switched to the "100% Recirculation Mode". Damper B is the only barrier to the outside environment when in the "100% Recirculation Mode" since dampers C and F1 or F2 will be opened in this mode. "100% Recirculation Mode" is selected by the operators in the event of toxic gases being detected in the control room in events such as a fire outside of the control room.

Entergy maintenance technicians disassembled damper B and found that the damper was approximately one inch from the fully shut position while selected to be open. As a result, enough flow passed damper B while the system was selected to "10% Incident Mode" to adequately pressurize the control room to ensure that operator dose would not exceed 10 CFR 50, Appendix A, General Design Criteria (GDC) 19 limits after a design basis accident. However, the damper's actuator being reversed would have prevented operators using the control room ventilation system to protect operators from toxic gases.

<u>Analysis</u>. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the Phase 1 screen required a Phase 3 analysis because the damper misalignment represented a degradation of the barrier function of the control room against smoke and/or toxic gas intrusion. The Region I Senior Reactor Analyst (SRA) conducted a qualitative Phase 3 assessment of this condition using the Indian Point 3 Individual Plant Examination for External Events (IPEEE) and available design documentation. Based upon this assessment, the analyst concluded the following:

The as-found condition of the control room ventilation damper did not have an adverse impact on the licensee's response to a fire event. This ventilation damper does not serve as an automatic or manual fire-rated barrier. Accordingly, this condition, as it relates to any specific fire scenario, would not have resulted in any incremental change to the overall core damage probability, including those fire events resulting in the evacuation of the control room.

The SRA postulated that operator response to external events involving control room toxic gas intrusion could potentially be adversely impacted by this condition. Specifically, any credit for control room ventilation being isolated and subsequently aligned to a recirculation mode to permit the control room staff to continue to respond to operational and off-normal events from the control room, may be compromised by this condition.

The analyst reviewed IPEEE Section 5.5, "Hazardous Chemical, Transportation and Nearby Facility Incidents," to assess the known toxic gas vulnerabilities of the unit and their respective estimated frequencies of occurrence. Based upon this review, the predicted frequencies of all known potential releases of toxic chemicals and asphyxiates (reference IPEEE Table 5.5.2.2) did not exceed the > 1E-6/year frequency criterion which would warrant further analysis per Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," dated June 1974. Accordingly, any calculated incremental increase in core damage probability as a result of the spectrum of analyzed initiating events coincident with a toxic gas release would be insignificant.

The SRA observed that the licensee’s evaluation of the more likely toxic gas intrusion events did credit detection by the toxic gas monitors and operator response to configure the control room ventilation to minimize gas intrusion to the control room envelope. The SRA and inspectors also confirmed that station operating procedures require the control room staff to don self-contained breathing apparatus anytime a toxic gas monitor alarm is received and verified. This action would further mitigate any potential increased risk associated with the degraded as-found damper and control room ventilation system.

The SRA concluded that this finding was of very low risk significance (Green).

This finding is associated with the cross-cutting area of human performance (personnel), in that the failure to properly install control room ventilation damper B impacted the ability of the control room ventilation system to protect the operators from smoke or toxic gas. (see Section 4OA4).

Enforcement. Technical Specification 3.7.11, “Control Room Ventilation System,” states that two control room ventilation system (CRVS) trains shall be operable. Contrary to this requirement, due to the improper installation of damper B, the CRVS was considerable inoperable since May 5, 2001. Entergy did not properly implement LCO 3.7.11.B which requires that one CRVS train shall be restored to operable status within 72 hours, or take additional required actions to shutdown per LCO 3.7.11.C. Because this failure to maintain the CRVS was entered into the licensee’s corrective action process (reference CR-IP3-2005-00315), this violation is being treated as an NCV consistent with Section VI.A. of the NRC Enforcement Policy.
(NCV 05000286/2005002-02, Control Room Ventilation System Inoperable due to Human Performance Maintenance Error).

4OA4 Cross Cutting Aspects of Findings

Section 40A3 describes a finding in which maintenance technicians failed to properly install CRVS damper B due to incomplete work instructions. This error impacted the operability of a barrier function. Consequently, both trains of technical specification required equipment were rendered inadvertently inoperable. This finding was determined to be associated with the cross-cutting area of human performance (personnel).

4OA5 Other Activities

8. TI 2515/150, Revision 3 - Reactor Pressure Vessel Head And Vessel Head Penetration Nozzles (NRC Order EA-03-009)

a. Inspection Scope

The inspection assessed the effectiveness of the licensee’s reactor pressure vessel (RPV) and head vessel penetration (VHP) nozzle inspection in detecting small amounts of boric acid, primary water stress corrosion cracking (PWSCC) in VHP nozzles, and boric acid flow through the interference zone of the fit of the VHP nozzles. The inspection consisted of interviews with ultrasonic, eddy current and visual examination personnel, data analysts, and engineering personnel. The data analysts' training and qualification records were reviewed to verify the licensee's personnel qualification process adequately prepared the assigned staff to perform the examination and analyze accumulated non-destructive examination data. Also, the inspector reviewed the examination procedures to determine the adequacy of the guidance and examination criteria to implement the examination plan.

The inspector reviewed accuracy of the licensee's RPV head's susceptibility calculation, which showed that IP3 was in the "Moderate Susceptibility" range. In accordance with NRC order EA-03-009, the equation requires the licensee to use a head temperature corresponding to 100% power for each of the operating cycles. The head temperatures used for the IP3 susceptibility calculations were furnished to the licensee by Westinghouse through the use of proprietary computer codes. A detailed methodology of the Westinghouse calculation was provided to the inspectors and determined to be accurate.

For the visual examination of the reactor head bare metal surface, the inspector observed that the camera operator used the appropriate test chart characters for a qualified VT-2 exam. The inspector observed the analyst who was reviewing the recorded tapes to verify that the approved procedures were being followed and appropriate examination criteria was available to the analyst and was being used to disposition any degraded conditions or evidence of boron. The inspector verified that appropriate corrective action was taken for indications identified during the examination process, including thorough documentation and effective cleaning of the head and penetrations. The inspector also verified that the licensee made all efforts to access and inspect the required surface area surrounding the penetrations on top of the reactor head.

For the ultrasonic/eddy current examination of the RPV penetrations, the inspector accessed records used to program the automatic probe to achieve full coverage of the J-groove weld area. Portions of the examination were observed to verify that the approved procedures were being followed. The ultrasonic and eddy current records for a CRDM nozzle were accessed to determine the validity the analyst's assessment of the integrity of the pressure boundary. The inspector reviewed the results for each penetration examined to verify that no cracks or significant indications associated with primary water stress corrosion cracking were present in the reactor upper head.

b. Findings

No findings of significance were identified.

The specific reporting requirements of TI 2515/150, Revision 3 are documented in Attachment 2.

2. TI 2515/152 - Reactor Pressure Vessel (RPV) Lower Head Penetration (LHP) Nozzles (NRC BULLETIN 2003-02)

a. Inspection Scope

The inspectors reviewed the licensee's response to NRC Bulletin 2003-02 which described the RPV lower head penetration inspection program. The inspectors reviewed the LHP nozzle examination procedure to determine whether it provided adequate guidance and examination criteria to implement the licensee's examination plan. The inspectors reviewed examination personnel training and qualification records to ensure that personnel were adequately prepared to perform the assigned examination activities.

The inspectors observed a portion of the LHP inspection activities and also reviewed photographs and examination reports to determine whether the inspection procedure was effectively implemented. The inspectors observed the review of several penetration nozzles to evaluate the effectiveness of the visual (VT) examination to verify that the penetration intersection location could be fully accessed to perform a 360-degree examination. The inspectors observed the actions taken by the licensee when the insulation package under the lower head impacted the viewing of some penetrations. The inspectors also reviewed corrective actions taken to clean the penetrations and document the baseline condition of the head, in expectation of future inspections. The inspectors questioned the licensee on possible options for improving the lower head insulation design.

b. Findings

No findings of significance were identified during this inspection.
The specific reporting requirements of TI 2515/152 are documented in Attachment 3.

4OA6 Meetings, including Exit

On April 21, 2005, the inspectors presented the inspection results to Mr. C. Schwarz and other Entergy staff members, who acknowledged the inspection results presented. Entergy did not identify any material as proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

R. Allen, NDE Analyst
N. Azevedo, Supervisor, Engineering
T. Beasley, Systems Engineer
J. Boccio, I&C Superintendent
V. Cambigianis, Systems Engineering Primary Systems Supervisor
T. Carson, Manager, Maintenance
J. Comiotes, Director, Nuclear Safety Assurance
P. Conroy, Manager, Licensing
F. Dacimo, Site Vice President
G. Dahl, Licensing
G. Dean, Assistant Operations Manager - Training
R. DeCensi, Technical Support Manager
R. Dolansky, ISI Engineer
P. Donahue, Senior Environmental Specialist
R. Drake, Supervisor, Engineering
A. Eng, Licensing, White Plains Office
C. English, Assistant Program Manager
K. Finucan, Quality Assurance
T. Foley, System Engineer
G. Garcia, Westinghouse Level III analyst
J. Goebel, Project Manager
C. Ingrassia, Systems Engineer
F. Inzirillo, Emergency Planning Manager
T. Jones, Licensing Supervisor
R. Kadin, Supervisor for WesDyne
D. Lancaster, Jamko Project Manager
D. Leach, Director, Site Engineering
M. Miele, Project Manager
T. McCaffrey, Manager, Systems Engineering
B. McGuire, Contractor/Investigator, VPA Corporation
E. O'Donnell, IP3 Assistant Operations Manager
J. O'Driscoll, Systems Engineer
P. Okas, Engineering Programs
T. Orlando, Manager, Programs and Components
J. Parrotia, QA Manager
R. Penny, Program Manager
F. Phillips, Emergency Planner
M. Rose, NDE Analyst
P. Rubin, Manager, Site Planning and Outage Services
C. Schwarz, General Manager, Plant Operations

J. Ventosa, Site Operations Manager
A. Vitale, Operations Manager, IP3
T. Walsh, WesDyne NDE Lead
S. Wilke, Fire Protection Engineer
C. Wend, Radiation Protection Manager
D. Shah, Systems Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000286/2005002-01	NCV	Transfer of Low-level Radioactive Waste, by Entergy Indian Point Energy Center for Disposal, That Did Not Meet Barnwell Low-level Waste Disposal Facility License
05000286/2005002-02	NCV	Control Room Ventilation System Inoperable due to Human Performance Maintenance Error

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures
3-SOP-IA-001, “Instrument Air System Operation,” Rev. 20
3-COL-IA-1, “Instrument Air System,” Rev. 27
3-OSP-EL-001, Emergency Diesel Generator Operation With Temporary Service Water Return Lines, Rev. 0
3-COL-RW-2, Rev. 39, “Service Water System”

Drawings
9321-F-20333, SH 1, “Flow Diagram Service Water System”
9321-F-27223, “Flow Diagram Service Water System Nuclear Steam Supply Plant”

Clearances
3-SWS-SWN-55

Work Orders
IP3-04-05974

Section 1R08: Inservice Inspection

Procedures
3-PT-R114, RCS Boric Acid Leakage and Corrosion Inspection
ENN-EP-S-001, IWE General Visual Containment Inspection
IP-3-RPT-VC-03071, Revision 4, Containment Inservice Inspection First Ten Year Class MC and CC Program

Drawings
ISI-IWE-001, Containment Metal Liner Roll-Out Drawing

Work Orders
WO # IP3-05-15721, ASME Section XI IWE Inspection, mechanical and electrical penetrations
WO # IP3-05-15720, ASME Section XI IWE Inspection, weld channel paint
WO # IP3-05 15719, ASME Section XI IWE Inspection, moisture barrier

Condition Reports
CR-IP3-2005-01002, RCS boric acid leakage inspection
CR-IP-3-2005-01345, weld between pipe and support, pressurizer to valve line
CR-IP3-2005-01549, 2" aux spray line repair indications
CR-IP3-2005-01587, containment liner

Miscellaneous
Tech Evaluation 05-000883, degraded studs and nuts
ENN-NDE-9.41 Rev. 0, Surface Examination Data, 6" Line 344
Memo, item 278 ISI inspection indication for RC-H-344-1
Calibration Data Sheet, Socket Welds
Augmented Risk Informed IP3 Component Surveillance Schedule
Memo, RPV Interior Inspection
Entergy Nuclear Northeast R-13 Schedule as of 0424003 LT-AD: ISI Inspection Program
Operational Assessment of Indian Point 3 Steam Generator Tubing for Cycle 13 and 14
Evaluation of Steam Generator Secondary Side Visual Inspections for Foreign Objects and Loose Parts at Indian Point 3
Integrated Leak Rate Test, Attachment 10, Containment Building Visual Inspection
ASME Code Article IWE-1000
Reactor Vessel Closure Head INT-1-3100-2, 3, 4, 5, 6 & 7, Thickness Data Sheet
RV Closure Head Calibration Data Sheet

Section 1R12: Maintenance Effectiveness

Calculations
IP3-CALC-SWS-02022
DRN-04-00917

Work Orders
IP3-05-00406

Condition Reports

CR-IP3-2005-00345	CR-IP3-2005-00362	CR-IP3-2004-01578	CR-IP2-2005-00212
CR-IP3-2003-05488	CR-IP3-2004-01578	CR-IP3-2004-01578	CR-IP3-2003-01600
CR-IP3-2003-02602	CR-IP3-2003-06146	CR-IP3-2004-01931	CR-IP3-2004-00315

Miscellaneous

IP-RPT-00090, Maintenance Rule Structural Monitoring Report, Rev. 0
IP3-RPT-STR-01932, Maintenance Rule Basis Document for System C09 IP3 Structures System, Rev. 0
"Maintenance Rule Basis Document for 125V DC Power System," Rev. 0
Unit 3 - DC Power - Fourth Quarter 2004 System Health Report
ER-04-3-022É, 32 Station Battery Replacement

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Drawings

IP3V-171-0357, Instrument Block Diagram Integrating Reactor Protection and Control System, Rev. 3
5651D72, Logic Diagrams Steam Generator Trip Signals, Rev. 7

Procedures

3PT-Q100C, Steam Flow/Feedwater Flow Mismatch Functional Test, Rev. 3
OAP-035, Technical Specifications and Technical Requirements Manual - License Adherence and Use, Rev. 1

Work Orders

IP3-05-10839 IP3-05-10565

Condition Reports

IP3-2005-00209 IP3-2005-00124 IP3-2005-00227

Section 1R15: Operability Evaluations

Procedures

3-SOP-SI-003, Recirculation and/or Purification of the Refueling Water Storage Tank, Rev. 16

Calculations

IP3-CALC-SI-03333, Engineering Evaluation of Postulated RWST Inventory Loss in Support of ACT 99-44077, Rev. 0

Condition Reports

CR-IP3-2005-00036	CR-IP3-2005-00263	CR-IP3-2005-00366	CR-IP3-2005-00510
CR-IP3-2005-00529	CR-IP3-2005-00560	CR-IP3-2005-00572	CR-IP3-2005-00649
CR-IP3-2005-00700			

Miscellaneous

NSE 99-3-035	RWST Purification Without Continuous Manning While Above Cold Shutdown, Rev. 1

Section 1R20: Refueling and Outage Activities

Procedures

3-POP-3.7	Plant Cooldown - Hot to Cold Shutdown, Rev. 44
3-POP-3.1	Plant Shutdown from 45% Power, Rev. 38
3-ARP-008	MTG Supervisory Alarm, Rev. 41

Condition Reports

CR-IP3-2005-00989 CR-IP3-2005-00992 CR-IP3-2005-00994

Section 1R22: Surveillance Testing

Procedures

3PT-R156B, "Station Battery #32 Load-Profile Service Test," Rev. 8
PFM-82, "BCT-2000 Battery Test Computer Calibration," Rev. 5

Work Orders
IP3-02-15923

Section 1R23: Temporary Plant Modifications

Procedures

3-ARP-003	Pressurizer Spray Line Loop Low Temperature, Rev. 38

Drawings

113E302	Miscellaneous Relay Racks Rack No. 1 (G2) Front, Rev. 15
9321-LD-72453	Pressurizer Spray Temperature Loop T-451 Diagram, Rev. 0

Work Orders

WO IP3-04-19893 WO IP3-04-19894

Section 4OA2: Problem Identification and Resolution

Condition Reports

CR-IP3-2004-3114	CR-IP2-2004-5906	CR-IP2-2004-4322	CR-IP2-2004-6271
CR-IP2-2004-5164	CR-IP2-2004-6219	CR-IP2-2004-5179	CR-IP2-2004-6632
CR-IP2-2005-00613	CR-IP2-2005-00737	CR-IP2-2005-01106	CR-IP3-2005-00400
CR-IP3-2005-00462	CR-IP3-2005-00463	CR-IP3-2005-00531	CR-IP3-2005-00781
CR-IP3-2005-00896	CR-IP3-2005-00934	CR-IP3-2005-00977	CR-IP3-2005-01209
CR-IP3-2005-01334	CR-IP3-2005-01416		

<u>Section 4OA5: Other Activities</u>

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<u>Calculations</u>
IP3 Calculation No. IP3-CALC-RV-03720 (rev. 1) "Estimation of Effective Degradation Years (EDYs) for IP3 Reactor Vessel Head”

<u>Procedures</u>
Procedure for Inspection of Reactor Vessel Head Penetrations with Gap Scanner
WDI-ET-003, IntraSpect Eddy Current Imaging Procedure for Inspection of Reactor Vessel Head Penetrations
WDI-ET-005, RPV Head CRDM Penetrations EC Examination for Wastage Detection Procedure
Wesdyne Standard Internal Review Sheet, WCAL-002, Pulser/Receiver Linearity Procedure

<u>Work Orders</u>
WO# IP3-04-12523, Perform EC and UT NDE
WO# IP3-04-12277, Bare Metal Visual (BMV) Examination of Top Head Penetrations Per NRC Order EA-03-009, Rev.1, Dated 2-20-2004, and its associated Examination Acceptance Criteria

<u>Condition Reports</u>
CR-IP3-2005-01409, initial visual exam of RPV upper head
CR-IP3-2005-01419, canopy seal area
CR-IP3-2005-01483, penetration number 69
CR-IP3-2005-01487, indications of new boric acid
CR-IP3-2005-01554, shroud to vessel flange
CR-IP3-2005-01516, FME identified
CR-IP3-2005-01360, reactor head inspections failed to meet performance expectations on nightshift
CR-IP3-2005-01611, during scan of penetration 22 data anomaly

<u>NRC Documents</u>
NRC Order EA-03-009 "Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors" ML040220391
NRC Order EA-03-009 Relaxation Request Regarding Inspection of Reactor Pressure Vessel Head Nozzles
Request for Additional Information Regarding Request for Relaxation from Order Establishing Interim Inspection Requirements of Reactor Vessel Heads, Indian Point Nuclear Generating Unit No. 3
NRC First Revised Order EA-03-009 Relaxation Requests for Inspection of Reactor Pressure Vessel Heads
Relaxation of First Revised Order on Reactor Vessel Nozzles, Indian Point Nuclear Generating Unit No.3

Miscellaneous

Upper Reactor Vessel Head Above the Insulation Inspection report
JTS-2005-3-CRDM, Reactor Vessel CRDM Visual Inspection for Indian Point Unit 3
ENN-NDE-10.02, VT-2 Examination
Jamko IP3 THOR Navigation Routes
Wesdyne International NDE Qualification and Certification Summary Records
Automated Ultrasonic Examination Calibration Data Sheet
Indian Point 3R13 Reactor Vessel Head Weld Coverage, Appendix A Examination Coverage Memo, Cleaning and UT/ET Inspection of Penetrations #21, 22, 58, 69, 73 and 78
MRS-SSP-1450, Appendix K Reference Table 3R12 RVHI Scan Information
IP3 3R13-Reactor Pressure Vessel Integrity Examination of RPV Head Condition Reports - Engineering Evaluation (Response to Corrective Action CR-IP3-2005-01487 CA #5 & #7)
Inspection Plans for Vessel Head - 3R13 Pre-Job Briefing
Various Reactor Head Visual Inspection Project Turnovers
IP3 Reactor Head UT/ET Status 3R13
IPEC Radiological Work Permit, Rx Head External Inspection
Indian Point Unit No. 3 Reactor Vessel Head Inspection Results; IP3, Spring 2003 Refueling Outage
Primavera System’s Outage Schedule on reactor head activities
Safety Evaluation for IP3 First Revised Order EA-03-009, Relaxation Requests for Inspection of Reactor Pressure Vessel Head and Penetration Nozzles
Evaluation of Work Performed on the Results of the 3R13 Reactor Head Penetration UT/ET Inspection
WDI-PJF-1303009-PSR-001, Reactor Vessel Head Penetration Inspection, IP3 Reactor Vessel Head Inspection Status Map - 3R13
Westinghouse Ultrasonic Report Sheet, Penetration No. 22
Westinghouse Eddy Current Report Sheet, Penetration No. 22
Entergy VT-2 Visual Examination Report, Reactor Vessel Upper Head BMV and CRDM Penetrations
WDI-UT-013, IntraSpect UT Analysis Guidelines
Westinghouse Standard Internal Review Sheet, WDI-ET-008, IntraSpect Eddy Current Imaging
WDI-ET-004, IntraSpect ET Analysis Guidelines
MRS-SSP-1450, Reactor Vessel Head Penetration Inspection Tool Operation for IP3

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Work Orders

Work Order IP3-04-12276 "Bottom Mounted Instrumentation per NRC Bulletin 2003-02" and its associated “Examination Acceptance Criteria”

Condition Reports

CR-IP3-2005-01214, indications of staining
CR-IP3-2005-1208, weld rods located on insulation

<u>Miscellaneous</u>
Jamko Field Service Instruction JTS-2005-3-BMI "Reactor Vessel BMI Visual Inspection for Indian Point Unit 3"
"Lower Reactor Head Bottom Mounted Instrumentation (BMI) Inspection" report
90-Day Response to NRC Bulletin 2003-02 Regarding Leakage From Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity
Various IPEC 3R13 Shift Outage Manager Turnovers
05VT043, VT-2 Examination Report for Reactor Vessel Lower Head BMI Penetrations
IPEC Radiological work permit for "3R13 Inspect Reactor Bottom"
Various BMI Project Turnovers

LIST OF ACRONYMS

ABFP	auxiliary boiler feedwater pump
ANS	alert and notification system
ALARA	As Low As is Reasonably Achievable
CAP	corrective action program
CCR	central control room
CFR	Code of Federal Regulations
COL	check-off list
CR	condition report
CVCS	chemical and volume control system
EDG	emergency diesel generator
EOF	Emergency Operations Facility
EOP	emergency operating procedure
EP	emergency preparedness
FSAR	final safety analysis report
IMC	inspection manual chapter
IP3	Indian Point Nuclear Generating Unit 3
IPEC	Indian Point Energy Center
IPEEE	Individual Plant Examination of External Events
KV	kilo volts
LER	Licensee Event Report
LGR	local government radio
NCV	non-cited violation
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
ONOP	off-normal operating procedure
PI	performance indicator
PMT	post maintenance test
PWT	post-work test
RCP	reactor coolant pump
QA	quality assurance
RCA	radiologically controlled area
RCS	reactor coolant system
REMP	radiological environmental monitoring program

RHR	residual heat removal
RMS	radiation monitoring system
RPV	reactor pressure vessel
RTD	resistance temperature detector
RWST	refueling water storage tank
SCBA	self-contained breathing apparatus
SDP	significance determination process
SI	safety injection
SOP	system operating procedure
SW	service water
TLD	thermoluminescent dosimeter
TS	technical specification
TSC	technical support center
WO	work order

ATTACHMENT 2

TI 2515/150, Revision 3 - REACTOR PRESSURE VESSEL HEAD AND VESSEL HEAD PENETRATION NOZZLES (NRC Order EA-03-009)

Reporting Requirements

a.1. The visual examination was performed by qualified and knowledgeable personnel with certification to the American Society of Mechanical Engineers (ASME), Section XI, Level III for visual examiners. In addition, the Level III examiners had received training in this type of inspection. The training included a review of industry experiences, lessons learned, inspection results and procedure requirements.

Ultrasonic and eddy current test personnel performing calibration or data analysis functions were qualified to a minimum of Level II. In addition, data analysis personnel had training in the analysis system, reactor head penetration (RHP) examination techniques and RHP analysis methods.

a.2. The surface, volumetric and visual examinations were performed using adequate procedures. The visual inspection procedure and robotic equipment had been previously demonstrated on a mock-up head. The procedures specified the extent of the inspection required, provided documentation requirements and provided clear inspection standards and acceptance criteria on which personnel were trained.

a.3. The examinations were adequate to identify, resolve, and disposition deficiencies.

a.4. The examinations performed were capable of identifying the primary water stress corrosion phenomena described in Order EA-03-009.

b. The reactor vessel head had some staining and streaking on the surface of the carbon steel, and limited areas where the original coating had peeled off and exhibited a thin layer of iron oxidation underneath. There was also a small amount of debris and granular matter. After the vessel head was cleaned several times to remove boric acid deposits from sources above the head, the final visual exams were completed to verify that no significant degradation to the reactor head surface had occurred. A robotic crawler with a camera attached, as well as a camera probe were used to inspect 360 degrees around each nozzle penetration. The camera probe was utilized primarily in the head areas closest to the mirror-insulation. This inspection was not affected by viewing obstructions in the nozzle annulus areas. A small amount of foreign objects were located during the head inspection and were subsequently removed.

c. Small boron deposits could be identified and characterized by the visual technique used. The robotic camera and the camera probe had high magnifications and a VT-2 visual card was used to verify correct focus.

d. One of the conoseals used on a thermocouple-containing penetration had leaked during the previous operating cycle. The conoseals were modified on each of the five penetrations containing thermocouple attachments during this outage, including the one that leaked, which was fully repaired.

e. There were no significant items that impeded an effective visual examination of the outside surface of the head. The mirror insulation that was installed over the head was temporarily jacked up to allow the robotic crawler camera to move freely over the surface of the head.

For the ultrasonic examination of some of the CRDM nozzles, the threading located at the bottom of the penetrations prevented the receipt of usable volumetric information for the area within 1" below the j-groove weld. Eddy current data was acquired for the required extent for the penetrations lacking complete UT data.

f. The head temperatures used for the Indian Point Unit 3 susceptibility calculations were furnished to the licensee by Westinghouse through the use of proprietary computer codes. The computer codes compiled a list of best estimates of reactor vessel upper head bulk mean fluid temperatures. The codes demonstrated that through appropriate conservatism and assumptions, the values of temperatures were within 10 degrees below average RCS hot leg temperatures for each of the operating cycles.

g. No indications were found during the UT examination that required the use of the flaw evaluation guidelines in Appendix D.

h. The procedure used for visual examination (ENN-NDE-10.02) specifies that if an area of general corrosion of a component resulting from leakage is to be traced to the source of origin. The BMV examination acceptance criteria associated with the BMV work order (PO-IP3-04-12277) specifies that if a deposit is identified, the examiner shall look for possible sources of the leak according to plant procedures, whether it is above or adjacent to the location of the deposit.

I. The licensee performed a visual examination above the upper head insulation and found evidence of leakage. As a result, a detailed inspection of 1) canopy seal weld regions of all penetrations including the mechanical clamps installed on the spare and core exit thermocouple penetrations 2) the conoseal locations 3) the reactor vessel head surface and 4) the top of the insulation. These inspection activities identified seven penetrations of interest which were further investigated by visual and NDE methods. All seven penetrations were volumetrically inspected from the inside surface using UT and ET techniques. After the head surface was cleaned, it was re-examined to determine whether any degradation of the carbon steel surface had occurred.

ATTACHMENT 3
TI 2515/152, REACTOR PRESSURE VESSEL LOWER HEAD PENETRATION NOZZLES (NRC BULLETIN 2003-02)

Reporting Requirements

a.1. The examination was performed by qualified and knowledgeable personnel with certification to the American Society of Mechanical Engineers (ASME), Section XI, Level II and Level III for visual examiners. In addition, Level II and Level III examiners had received a minimum period of training in this type of inspection. The training included a review of the penetration drawings, inspection techniques and use of visual aids, effects of surface conditions on detecting and evaluating indications, industry experience, lessons learned, inspection results and procedure requirements.

a.2. The examination was performed in accordance with demonstrated procedures, including ENN-NDE-10.02, Visual Examination.

a.3. The examination was adequate to identify, resolve, and disposition deficiencies. Optical light and resolution checks were performed as required providing the required character resolution.

a.4. The examination performed was capable of identifying active pressure boundary leakage and/or lower head corrosion as described in the bulletin.

b. No active boric acid deposits or leaks were identified at the interface between the vessel and the lower reactor head penetrations. The licensee used EPRI Report #1006296, Visual Examination for Leakage of PWR Reactor Head Penetrations, as a basis for determining that no active boric acid leaks were present.

c. The licensee used a combination of direct visual inspection and remote visual inspection by a camera probe that attached to the BMI tubes and rotated 360 degrees around the circumference of all nozzles. The center plate of insulation was lowered to attain direct visual access.

d. The licensee was able to achieve a complete 360 degrees circumferential visual on each penetration.

e. The reactor vessel lower head had contour insulation surrounding the bottom. The insulation parts had cut-outs of about 3 inch diameter around each penetration to allow viewing. Some of the access cut-outs were off center and the insulation panels had to be adjusted to allow for a full view of the penetrations. There were streaks of boric acid residue on the outer surface of the insulation and on parts of the lower head surface, which appeared to be stains that ran past the BMI penetrations underneath the insulation. It did not appear however that there were any boric acid deposits emanating from the penetrations annulus areas.

f. No material deficiencies that required repair were identified. Some of the original paint coating applied over the surface of the carbon steel had peeled and flaked off, but an inspection of the accessible areas where this happened showed that the boric acid residue had not resulted in any wastage of the head base material beyond minor surface corrosion.

g. As described above, the licensee removed small cover plates from each penetration nozzle which provided access to view the bottom-mounted instrumentation annulus area. Some of the access cut-outs were off center and the insulation panels had to be adjusted to allow for a full view of the penetrations. In order to make sure that boric acid had not collected under the insulation from the sources above, the center insulation plate was lowered, to perform a visual exam of a small area at the center bottom of the head.

h. No active boric acid leak deposits were noted on the lower vessel head. Some stains, attributed to prior reactor canopy and conoseal leaks from the upper head, were observed at multiple locations. These indications were evaluated by the licensee.

I. The licensee did not take any samples for chemical analysis. The criteria used to determine that the streaks on the lower head were not emulating from the lower penetration annulus areas were the locations of the streaks (running downslope) and the lining up of streaks below with sources above the lower head.

j. The penetrations were steam cleaned and examined to reconfirm the lack of pressure boundary leakage from the lower head.

k. A history of leakage going back to 1975, including canopy seal leaks in 1990, conoseal leaks in 2001, various valve packing leaks, and instrument well cover leaks, etc. is described in the licensee’s lower head inspection report. These historical leaks, as well as the leakage detected from the conoseal that was leaking in the most recent operating cycle, were characterized as small and intermittent, which corresponds to the type of streaking found on the lower head. The licensee acquired video records from this visual inspection to provide a baseline set of information before the next inspection.