February 2, 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 2 - NRC INTEGRATED INSPECTION REPORT NO. 05000247/2004012

Dear Mr. Dacimo:

On December 31, 2004 the US Nuclear Regulatory Commission (NRC) completed an inspection at the Indian Point Nuclear Generating Unit 2 (IP2). The enclosed integrated inspection report documents the inspection findings, which were discussed on January 13, 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. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of the inspection, nine findings of very low safety significance (Green) were identified. Eight of 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 (NCV) 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 2.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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. 50-247
License No. DPR-26

Enclosure: Inspection Report No. 05000247/2004012 w/Attachments

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

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	05000247/2004012
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Unit 2
Location:	295 Broadway, Suite 3 Buchanan, NY 10511-0308
Dates:	October 1, 2004 - December 31, 2004
Inspectors:	P. Habighorst, Senior Resident Inspector, IP2 M. Cox, Resident Inspector, IP2 R. Berryman, Resident Inspector, IP3 T. Hipschman, Senior Resident Inspector, IP3 C. Long, Reactor Inspector, DRP M. Snell, Reactor Inspector, DRP J. Schoppy, Senior Project Engineer (Acting), DRP J. Noggle, Senior Health Physicist, DRS T. O'Hara, Reactor Inspector, DRS A. Pascarelli, Reactor Inspector, DRS T. Burns, Senior Reactor Inspector, DRS T. Sicola, Reactor Inspector, DRS W. Held, General Engineer, NRR G. Meyer, Senior Project Engineer, DRS L. Cheung, Senior Reactor Inspector, DRS, (1/26-1/30/2004) G. Bowman, Reactor Inspector, DRS, (1/26 - 1/30/2004)
Approved by:	Brian J. McDermott, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000247/2004012; 10/01/2004 - 12/31/2004, Indian Point Nuclear Generating Unit 2; Equipment Alignment; Maintenance Risk Assessments and Emergent Work Control; Personnel Performance During Non-routine Plant Evolutions and Events; Refueling and Outage Activities; Surveillance Testing; Problem Identification and Resolution; Other Activities (Temporary Instruction (TI) 2515/152).

The report covers a 3-month period of inspection by resident inspectors, and eleven regional inspectors. Eight Green Non-cited Violations (NCVs), one Green finding, and two licensee identified violations 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). 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: Initiating Events

- Green. The inspector identified a self-revealing Green finding involving poor causal analysis associated with the main generator stator water cooling (SWC) system. The ineffective causal analysis was associated with the settings of the generator protection trip pressure switch (63-P79). The finding resulted in an automatic reactor trip due to a low inlet pressure condition on the main generator SWC system.

  The finding is more than minor since it impacts the Initiating Event cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions, and is associated with the equipment performance attribute. Specifically, the finding affects the likelihood of a reactor trip and challenges the critical safety function of auxiliary feedwater (AFW) initiation. The finding is of very low risk significance (Green) since it does not contribute to both the likelihood of a reactor trip and the likelihood of mitigation equipment functions being unavailable. The finding is associated with the cross-cutting area of problem identification and resolution (PI&R) based on the ineffective causal analysis for previously identified deficiencies affecting the SWC system. (Section 1R14.1)

- Green. The inspector identified a self-revealing Green non-cited violation of 10 CFR 50 Appendix B, Criterion V "Instructions, Procedures and Drawings." Maintenance personnel did not verify that the length of tubing between the RACK 20 bulkhead connection and the existing 21 Reactor Coolant Loop Flow (FT-416) Hi side impulse tubing was sufficient for a proper Swagelok connection pursuant to procedure IP-SMM-MA-108.

  The finding is more than minor since it impacts the Initiating Event cornerstone objective of limiting the likelihood of those events that upset plant stability and challenges critical safety functions, and is associated with the procedural quality attribute. Specifically, the finding affects the likelihood of a reactor coolant system (RCS) leak that upsets plant stability and challenges critical safety functions. This finding is of very low risk significance (Green) since worst case degradation would not result in exceeding the technical specification (TS) limit for identified leakage (10 gpm) and it does not affect the mitigation system's safety functions. (Section 1R14.2)

- Green. The inspector identified a self-revealing Green non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings" associated with a reactor vessel water level control issue during the drain down for the reactor vessel head re-installation on November 11, 2004. Specifically, an inappropriate level reduction rate existed by procedure, such that when communications to field operational personnel were temporarily lost and manual valve manipulations to reduce the rate were delayed, a two foot lower reactor vessel water level resulted.

  This finding is more than minor, because it potentially affects the Initiating Events cornerstone objective of limiting the likelihood of events that challenge critical safety functions during shutdown, and is associated with the procedural quality attribute. This finding is considered to be of very low safety significance (Green), because residual heat removal (RHR) shutdown cooling remained operable and gravity re-flood of the reactor without operator action would have limited the consequences of any potential loss of shutdown cooling. (Section 1R20)

Cornerstone: Mitigating Systems

- Green. The inspector identified a Green non-cited violation of TS 5.4.1 associated with Entergy's failure to properly implement procedure 2-COL 10.0, "Locked Safeguards Valves." Residual heat removal recirculation valve AC-1863 was left in the shut position during the restart from IP2 refueling outage No. 16 (2RF16). The valve was not locked open in accordance with 2-COL 10.0 prior to entering Mode 4 due to the sequence of procedures performed at the end of the refueling outage.

  The finding is more than minor because it is associated with the Mitigating Systems cornerstone attribute of configuration control and adversely affects the capability of systems that respond to initiating events to prevent undesirable consequences. The finding involves the unavailability of a design feature described in the Final Safety Analysis Report (FSAR) that would ensure the capability to continue high-head recirculation after a loss of coolant accident (LOCA) in the event of certain system failures. This finding is of very low safety significance (Green), because the normal flow paths for establishing flow to the safety injection (SI) pump suctions during high-head recirculation remained available for the duration of the period that valve AC-1863 was shut. This finding is associated with the cross-cutting area of human performance, in that, operators did not adequately assess a change in the sequence of procedures performed during the refueling outage. (Section 1R04)

- Green. The inspector identified a self-revealing Green non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings." A maintenance procedure for trip checks associated with the 345KV electrical feeder was inadequate since it did not provide appropriate directions for the test set-up. As a result, technicians unintentionally reset the main generator lock-out relays by using test stabs which defeated the station blackout (SBO) relays associated with the emergency diesel generators (EDGs) starting logic.

The finding is more than minor since it affects the procedure quality attribute of the Mitigating Systems cornerstone and impacts the cornerstone objective of ensuring availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. The finding is of very low safety significance (Green) due to low exposure time, credit for manual actions in the abnormal operating procedures (AOPs) to restore power to the safety-related 480 volt buses and start the required loads to stabilize plant conditions, and the availability of other mitigating equipment (ie. steam driven AFW pump and gas turbines 1 and 2). (Section 1R13.1)

- Green. The inspector identified a self-revealing Green, non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The finding involved improper maintenance on a 480 volt cross-tie breaker (52/3AT6A). Maintenance personnel did not install the main line contactors for breaker (52/3AT6A) consistent with maintenance procedure BRK-P-003-A, "Westinghouse Model DB-75 Breaker - Preventative Maintenance."

  The finding is more than minor since it affects the Mitigating Systems cornerstone objective of ensuring the availability of the RHR system and to prevent undesirable consequences such as core damage due to lack of core cooling during plant shutdown. The performance finding affects the Mitigating Systems cornerstone attribute of procedural quality (breaker preventative maintenance (PM) procedure). This finding is considered to be of very low safety significance since it did not degrade Entergy's ability to terminate a leak path or add reactor coolant inventory when needed, or degrade Entergy's ability to recover RHR once it is was lost. This finding is associated with the cross-cutting area of human performance, in that maintenance personnel did not implement a 480 volt breaker PM procedure correctly. (Section 1R13.2)

- Green. The inspectors identified a Green non-cited violation of 10 CFR 50 Appendix B, Criterion VI, "Document Control." Inadequate document control resulted in multiple surveillance procedures not meeting the criteria of the Improved Technical Specifications (ITS) surveillance requirements (SRs) or the applicable ITS basis document.

  The finding is more than minor since, if left uncorrected, it would become a more significant safety concern potentially impacting multiple SRs of safety-related equipment and equipment important to safety. The performance finding affects the Mitigating Systems cornerstone attribute of procedural quality. This finding is considered to be of very low risk significance (Green) since it had not resulted in a loss of safety function or in any inoperable equipment. (Section 1R22)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, "Corrective Action," for Entergy's failure to properly address a condition adverse to quality involving leakage from a canopy seal weld in

November 2002. The ineffective corrective actions for this conoseal leak led to boron accumulation on the reactor vessel head (RVH).

The finding is considered to be more than minor since, if left uncorrected, it could have led to a more significant problem. Specifically, the boric acid, if re-wetted, could have led to accelerated corrosion of the RVH. The finding is of very low significance since the RVH integrity was not affected by this problem. The finding is associated with the cross-cutting area of PI&R related to the ineffective corrective actions for the conoseal leak. (Section 4OA2.1)

- Green. The inspectors identified a non-cited violation of 10 CFR 50 Appendix B, Criterion IX, "Control of Special Processes," for Entergy's failure to provide adequate inspection criteria and guidance to evaluators prior to the inspection of the reactor vessel lower head penetration nozzles. In particular, Entergy personnel performed visual inspections of the reactor vessel bottom mounted instrumentation annulus area without adequate procedural guidance to define potential problems or indications.

  This finding is considered to be more than minor since inspection program deficiencies could allow a degraded component to remain inservice undetected. Specifically, the failure to develop adequate inspection guidance could result in a failure to detect a degraded lower RVH penetration boundary. The finding is of very low significance since the lower RVH integrity was not affected. (Section 4OA5.1)

B. Licensee-Identified Violations.

The inspectors reviewed two violations of very low safety significance, which were identified by Entergy. Corrective actions taken or planned by Entergy have been entered into Entergy's corrective action program (CAP). These violations and corrective actions are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Indian Point 2 (IP2) began the inspection period at 97 percent power due to an inoperable main turbine No. 2 stop valve. On October 4, 2004, the unit was restored to full power when Entergy reconfigured the turbine valves to close the associated No. 2 control valve. On October 11, 2004, the unit began a plant coast-down until the start of 2RF16. On October 22, Unit 2 performed a manual reactor trip from 25 percent power and began 2RF16. The reactor was made critical on November 21 and power ascension continued until November 26 when the unit experienced an automatic reactor trip due to SWC flow problems. The unit returned to full power on November 29. On December 3, operators took the main turbine off-line to affect repairs to a main steam line pressure impulse line steam leak. On December 5, the unit returned to full power and remained at this level throughout the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity [REACTOR - R]

1R01 Adverse Weather Protection (71111.01 - 3 samples)

a. Inspection Scope

The inspectors reviewed Entergy procedure OAP-048, Rev. 0, “Seasonal Weather Preparation,” and the associated station operating procedures and check-off lists (COLs) involving cold weather preparations (one sample of adverse weather preps). The review was to verify that these procedures and checklists were completed. The inspectors verified that the actions taken by Entergy to assure freeze protection of plant equipment were completed consistent with prevailing weather conditions for the months of October, November, and December 2004. The inspectors performed walkdowns of accessible areas of the Unit 2 power plant operating and auxiliary support structures including the gas turbine 1 building, emergency diesel generators (EDGs), and intake structure to assess the adequacy of freeze protection measures (one sample of actual adverse weather). The inspectors also looked for any vulnerable components or conditions not previously identified by Entergy.

The inspectors reviewed past condition reports (CRs) for any weather-related adverse trends or repeat problems to ensure Entergy had adequately addressed them through the corrective action program (CAP). The inspectors reviewed Entergy’s cold weather preparation progress and any uncorrected deficiencies (one sample of adverse weather preps). The inspectors also reviewed outstanding work orders (WOs) for selected systems to evaluate any impacts on the freeze protection and cold weather preparations.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q - 4 samples and 71111.04S - 1 sample)

a. Inspection Scope

Partial System Walkdowns

The inspectors performed four partial system walkdowns during periods of system train unavailability in order to verify that the alignment of the available train was proper to support its required safety functions, and to assure that Entergy had identified and properly addressed equipment discrepancies that could potentially impair the capability of the available train. Referenced documents are listed in the Supplemental Information attachment at the end of this report. The following system walkdowns were performed:

- On November 3, 2004, the inspectors performed a walkdown of the back-up spent fuel pool (SFP) cooling system. The inspectors discussed system operations with contractor operators, verified anti-siphon holes established on both suction and discharge lines from the SFP, verified flowrates within procedural limitations, and reviewed system operating logs.

- On November 15, 2004, the inspector performed a partial system walkdown of the 22 and 23 SI trains while the 21 safety injection (SI) pump was out of service for maintenance during 2RF16. The inspector reviewed system drawings and COLs to verify proper alignment of suction valves 887A, 887B, 1810, and 848B, and discharge valves 850A, 851A, 850B, and 851B. The inspector also observed the physical condition of the equipment during the verification.

- On November 16, 2004, the inspector performed a partial system walkdown of the 21 DC distribution system, including the 21 static inverter, 21 batteries, 21 battery charger, 21 power panel, and the 21, 21A, and 21B distribution panels while the plant was in 2RF16. The inspector reviewed system drawings and COLs to verify proper alignment and observed the physical condition of the equipment during the verification.

- On November 17, 2004, the inspector performed a partial system walkdown of locked safeguards valves and breakers required to transition the plant from mode 5 to mode 4 after 2RF16. Entergy's COL was used to verify the alignment of valves and breakers and the condition of the associated equipment.

Full Equipment Alignment

The inspectors performed an extensive walkdown of the residual heat removal (RHR) and recirculation portion of the SI systems. The inspectors walked down the systems using COL 4.2.1, "Residual Heat Removal System," Rev. 21 and 2-COL 10.1.1, "Safety Injection System," Rev. 24. The inspectors verified that components were in the proper position per the COL and verified that any position discrepancies were properly documented. The inspectors also verified that the field configuration was consistent with the current revision of the COL. Additionally, the inspectors evaluated the physical condition of the equipment during the walkdown.

b. Findings

Introduction. The inspector identified a Green non-cited violation of TS 5.4.1 associated with Entergy's failure to properly implement procedure 2-COL 10.0, “Locked Safeguards Valves,” Rev. 38. Residual heat removal recirculation valve AC-1863 was left in the shut position during the restart from 2RF16. The valve was not locked open in accordance with 2-COL 10.0 prior to entering Mode 4 due to the sequence of procedures performed at the end of the refueling outage.

Description. On November 24, 2004, the inspectors discovered valve AC-1863 was in the shut position. The inspectors brought this to the attention of the system’s engineer and the plant operators in the control room. Valve AC-1863 had been shut in accordance with 2-SOP-1.2, “Draining Reactor Coolant System,” Rev. 40. Vessel draining was completed and valve AC-1863 should have been repositioned to the locked open position after November 11, 2004.

The FSAR describes a contingency flow path from the discharge of the RHR pumps to the suction of the 22 SI pump using an alternate flow path which bypasses the RHR heat exchangers. This alternate flow path is available in the event of a loss of recirculation flow due to a blockage of the primary side of the RHR heat exchanger or a break in the SI pump’s suction header. The normal flow paths for establishing flow to the SI pump suctions during high-head recirculation remained available for the duration that valve AC-1863 was shut. Valve AC-1863 was improperly positioned for 13 days.

Analysis. Entergy’s failure to implement a procedure appropriate for emergency core cooling system (ECCS) operation is a performance deficiency associated with the Mitigating Systems cornerstone, and is contrary to NRC regulations. Traditional enforcement does not apply because an event did not occur that resulted in an actual safety consequence, the failure to have an adequate procedure did not impact the NRC’s regulatory function, and was not the result of a willful violation of NRC requirements or Entergy procedures. The finding is greater than minor because it is associated with the Mitigating Systems cornerstone attribute of configuration control and adversely affected the capability of systems that respond to initiating events to prevent undesirable consequences. The finding involved the unavailability of a design feature described in the FSAR that would ensure the availability of high-head recirculation after a loss of coolant accident (LOCA) in the event of certain system failures, and the evaluation used the screening criteria on Phase I SDP worksheet for the Mitigating Systems cornerstone. The finding was determined to be of very low safety significance (Green because the normal flow pats for establishing flow to the SI pump suctions during high-head recirculation remained available for the duration of the period that valve AC-1863 was shut.

This finding is associated with the cross-cutting area of human performance, in that, operators did not adequately assess a change in the sequence of procedures performed during the refueling outage. This error impacted the availability of mitigating systems (see Section 4OA4).

Enforcement. Technical Specification 5.4.1, “Procedures,” requires in part that written procedures be established, implemented and maintained per Regulatory Guide (RG) 1.33 except as provided in the quality assurance program or referenced in the updated FSAR. Appendix A to RG 1.33 states that procedures for operation of ECCS should be covered by written procedures. Contrary to the above, Entergy did not properly implement procedures for operation of the ECCS. Because the failure to implement appropriate procedures was entered into Entergy’s CAP (reference CR-IP2-2004-06448), this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. During an extent-of-condition review on November 24, 2004, Entergy also discovered test stop valve upstream of AC-885A valve A-74A was closed but not sealed and capped. This valve is required to be closed, sealed and capped in accordance with 2-COL 10.6.2, “Containment Integrity,” Rev. 25.
(NCV 05000247/2004012-01: Failure to implement adequate procedures for ECCS operation)

1R05 Fire Protection (71111.05Q - 8 samples)

a. Inspection Scope

The inspector toured areas that were identified as important to plant safety and risk significant. The inspector consulted Section 4.0, “Internal Fires Analysis,” and the top risk significant fire zones in Table 4.6-2, “Summary of Core Damage Frequency Contributions from Fire Zones,” within the Indian Point 2 Individual Plant Examination for 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 inspector 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 inspector to determine the acceptability of the observed conditions in the fire zones are referenced in the Supplemental Information attachment at the end of this report. The areas reviewed were:

- Zone 11
- Zone 12
- Zone 14
- Zone 23
- Zone 32A
- Zone 15
- Zone 62A
- Zone 65A

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08 - 3 samples)

a. Inspection Scope

The inspectors observed selected samples of non-destructive examination (NDE) activities in process. Also, the inspectors reviewed selected additional samples of completed NDE and repair/replacement activities. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage. The observations and documentation reviews were performed to verify the activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspectors reviewed a sample of Entergy's inspection reports and CRs initiated as a result of problems identified during in-service inspection (ISI) examinations. Also, the inspectors evaluated the effectiveness in the resolution of problems identified during selected ISI activities. The inspectors reviewed Entergy's boric acid corrosion control program walkdowns. The results of this review are discussed in Section R20 of this report.

The inspectors observed the performance of two in-process NDE activities and reviewed documentation and examination reports for an additional four NDE activities. The inspectors reviewed two samples of welding activities on a pressure boundary and, reviewed the package for a repair performed in accordance with the ASME Code during the previous operating cycle.

The inspectors observed manual ultrasonic testing (UT) and visual examination (VT) activities to verify the effectiveness of the examiner, process, and equipment to identify degradation of risk significant structures, systems, and components (SSCs) and to evaluate the activities for compliance with the requirements of ASME Section XI of the Boiler and Pressure Vessel Code.

The inspectors observed the UT performed on the reactor vessel-to-flange weld and two reactor vessel meridional welds and the VT of the reactor pressure vessel (RPV) lower head instrumentation nozzles.

The inspectors reviewed the UT examination reports of the reactor vessel closure head lifting rig and several reactor vessel closure holdown studs, two main steam line circumferential welds, visual testing of feedwater pipe supports, reactor coolant pump 23 support bracket welds, and magnetic particle testing of one feedwater line weld.

The inspectors reviewed four samples of NDE evaluations which had been initially rejected and subsequently accepted after evaluation. The inspectors also reviewed the radiographs and the examiners' interpretation of indications on two chemical volume control system valve replacement welds.

The inspectors reviewed report "Indian Point Unit 2 Condition Monitoring and Operational Assessment RFO-15 SG-SGDA-02-45, September 2003 Westinghouse Electric Company." This report documented that the steam generator (SG) degradation, measured in 2RF15, was low enough to defer SG inspection activities during 2RF16.

The inspectors reviewed the composition of the pressurizer nozzle material and verified that TI 2515/160, "Pressurizer Penetration Nozzles and Steam Space Piping Connections in US Pressurized Water Reactors," was not applicable to IP2.

b. Findings

No findings of significance were identified.

1R11 Operator Requalification Inspection (71111.11Q - 1 sample)

b. Inspection Scope

On December 21, 2004, during continuing training for Emergency Response Organization (ERO) facility leads, the inspectors evaluated classroom training and exercises related to identification and classification of plant events using the site emergency plan. During the practical exercises the inspectors evaluated the various ERO managers' performance for correct use and implementation of Emergency Action Levels (EALs). 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.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 samples)

a. Inspection Scope

The inspectors reviewed the maintenance activities listed below, and recent performance issues with systems and components to assess the effectiveness of Entergy's Maintenance Rule (MR) program. Using 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," the inspectors verified that Entergy was implementing their MR program in accordance with NRC regulations and guidelines, properly classifying equipment failures, and using the appropriate performance criteria for MR systems in 10 CFR 50.65 (a)(2) status.

The inspectors also reviewed WOs, and associated post-maintenance test activities to assess whether: 1) the effect of maintenance work in the plant had been adequately addressed by control room personnel; 2) work planning was adequate for the maintenance performed; 3) the acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing documents; and, 4) the equipment was effectively returned to service. Referenced documents are listed in the Supplemental Information attachment at the end of this report. The below-listed maintenance activities were observed and evaluated.

21 Emergency Diesel Generator Camshaft Damage/Replacement

The inspectors reviewed activities associated with the replacement of a damaged camshaft on 21 EDG. The damage was identified during an 8-year preventive maintenance inspection on the EDG. The inspectors observed portions of the maintenance and reviewed WO IP2-04-30835 to evaluate Entergy's work practices associated with the camshaft replacement. The inspectors reviewed work history associated with the camshaft, fuel pump, crosshead and roller assembly to determine if previous work may have impacted the camshaft. The inspectors also evaluated Entergy's extent of condition assessment.

Residual Heat Removal System

The inspector performed a review of maintenance issues associated with the (RHR system since October 2003. The inspector evaluated the MR basis document to determine system boundaries and verified that the system was being properly tracked in accordance with the requirements of 10 CFR 50.65, "Requirements of Monitoring the Effectiveness of Maintenance." The inspector reviewed the quarterly system health inspection report for the 2nd quarter of 2004 and evaluated the system performance monitoring criteria for scope and accuracy. The inspector reviewed CRs for the system and evaluated their proper classification for the MR and compliance with ENN-DC-171, Rev. 0, "Maintenance Rule Monitoring."

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 6 samples)

a. Inspection Scope

The inspectors observed selected portions of emergent maintenance work activities to assess Entergy's risk management in accordance with 10 CFR 50.65(a)(4). The inspectors 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 inspectors observed and/or discussed risk management with maintenance and operations personnel. The following planned activities were observed:

- WO IP2-04-34637, Power reduction due to 1st stage pressure steam leak
- WO IP2-02-58391, Offsite calibration of W-95 resulting in loss of blackout signal to EDGs

The following five emergent activities were observed:

- WO IP2-04-23687, Failure to open on demand SI-MOV-885B
- WOs IP2-04-33504/33510, Reactor coolant system RTD leakage and repairs
- WO IP2-02-33119, Loss of bus 6A during 2RF16
- WO IP2-04-15373, Service water leaks on 1-2-3 header to EDG's

b. Findings

.1 Introduction. The inspector identified a self-revealing Green non-cited violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings.” A maintenance procedure for trip checks associated with the 345KV electrical feeder was inadequate since it did not provide appropriate directions for the test set-up. As a result, technicians unintentionally reset the main generator lock-out relays by using test stabs which defeated the station blackout (SBO) relays associated with the emergency diesel generators (EDGs) starting logic.

Description. On November 11, 2004, technicians were scheduled to perform trip checks on the 345KV off going feeder (W95). The technicians received permission to commence the work from the Control Room Supervisor (CRS). In preparation for the trip checks the technicians opened the test stabs associated with the main turbine generator primary and backup lockout relays. The technicians performed this task based on system drawings. No WO instructions addressed this step of the task. After opening the stabs the technicians requested permission from the CRS to reset the lockout relays. The CRS realized that this would defeat the logic for the SBO relays in the EDG starting logic. The CRS denied this request since TSs required that these relays be operable for either Bus 5A or Bus 6A while in the current plant conditions. After reviewing the drawing the technicians determined that they had effectively reset the relays by opening the test stabs. The CRS directed that the stabs be shut. The EDG’s were declared inoperable due to the loss of the SBO function for 30 minutes.

Analysis. The inspectors determined that this was a performance deficiency since the procedural steps were not sufficient to ensure that the maintenance activity would not result in the loss of the SBO function. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC’s regulatory function and the finding was not the result of any willful violation of NRC requirements or Entergy’s procedures. The finding was determined to be greater than minor since it affected the procedure quality attribute of the Mitigating Systems cornerstone and impacted the cornerstone objective of ensuring availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. This finding was determined to be self-revealing based on the definitions in IMC 0612 since the issue was not identified through a licensee program or process that was specifically intended to identify the problem.

In the event of a SBO, the logic circuitry is designed to automatically start and shut the EDG output breakers. Required loads would then sequence on the bus. The inspectors determined that on a loss of off-site power with the SBO relays defeated, the EDGs would still be provided a start signal due to operation of the undervoltage relays; however, they would not automatically be tied to the buses and required loads would not sequence on. Recovery from this condition would require operator action to shut the EDG output breakers and start the required loads. Procedure AOP-480V-1, “Loss of Normal Power to Any 480 Volt Bus,” Rev. 2 provides direction for the operators to perform these actions.

Because the identified performance deficiency potentially affected shutdown risk, the inspectors conducted a Phase 1 screening based on the criteria established in IMC

0609 Appendix G “Shutdown Operations Significance Determination Process.” The inspectors evaluated the plant conditions (cold shutdown, RCS closed and steam generators available for decay heat removal, loops filled and inventory in pressurizer, time to boil less than two hours) in accordance with Checklist 2 of Appendix G, Attachment 1. The performance deficiency was associated with the power availability guidelines of Checklist 2. This checklist requires a Phase 2 analysis for findings that degrade Entergy’s ability to cope with a loss of off-site power.

Based on the Phase 2 analysis, the inspectors determined that the failure to have adequate procedures for the W95 trip checks was of very low safety significance due to low exposure time, that manual actions in the AOPs provide direction for the operators to restore power to the safety-related 480V buses and start the required loads to stabilize plant conditions, and the availability of other mitigating equipment such as the steam driven AFW pump and gas turbines 1 and 2.

Enforcement. 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures and Drawings,” states in part that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and be accomplished in accordance with these procedures. Contrary to this, Entergy relied on skill of the craft during preparations to perform electrical feeder trip checks. This resulted in defeating the SBO relay logic and all EDGs being declared inoperable. Because this failure to provide adequate procedures for the performance of this maintenance task has been entered into Entergy’s CAP (CR IP2-2004-6266) this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy.
(NCV 05000247/2004012-02, Inadequate maintenance procedure resulting in all EDGs being declared inoperable due to defeating SBO logic)

.2 Introduction. The inspector identified a self-revealing Green, non-cited violation of 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” The finding involved improper maintenance on a 480 volt cross-tie breaker (52/3AT6A). Maintenance personnel did not install the main line contactors for breaker (52/3AT6A) consistent with maintenance procedure BRK-P-003-A, “Westinghouse Model DB-75 Breaker - Preventative Maintenance,” Rev. 1.

The consequence of the performance finding was a loss of power to 480 volt safeguards bus 6A and an automatic start signal for all three EDGs. The loss of 480 volt power resulted in a loss of RHR cooling for approximately 6 minutes and a loss of SFP cooling for approximately 39 minutes. At the time of the event, the unit was in a refueling mode with fuel in the core and reactor cavity level greater than 23 feet above the vessel flange.

Description. On March 19, 2003, maintenance personnel completed a preventative maintenance (PM) on breaker 3AT6A as documented in WO IP2-02-33119. On April 18, 2003, non-licensed operators installed this tie breaker into the 480 volt buses.

The onsite power distribution system contains four 480 volt safeguards power buses 5A, 6A, 2A, and 3A. The four electrical buses can be supplied by either of two offsite circuits or three EDGs. Breaker 3AT6A is a cross-tie breaker between buses 3A and 6A that during power operations is normally racked open to maintain electrical power separation. System operating procedure (SOP) 27.1.5 and AOPs (2-AOP-138Kv and 2-AOP-480-1) allow closure of this tie breaker when the plant is in a cold shutdown condition to facilitate maintenance or to allow one EDG to be tied to two safeguards buses in response to an EDG failure.

On November 9, 2004, at 11:39 p.m. an actuation of the emergency AC electrical power system occurred as described in FSAR section 8.2.3.4. While in a refueling outage, a planned evolution was in progress to tie 480 volt bus 3A to 6A using a tie breaker. When the normal supply breaker was opened as part of the maintenance, the tie breaker's lack of primary disconnects resulted in a loss of power to bus 6A. Safeguards bus 6A de-energized, and the 21 and 22 EDG automatically started on an undervoltage signal and supplied bus sections 5A, 2A, and 3A. The 23 EDG was tagged out for maintenance. At 12:58 a.m. on November 10, Unit 2 was returned to its normal 480 volt lineup. Entergy informed the NRC via a 10 CFR 50.72 report on November 10, 2004, at 4:05 a.m.

<u>Analysis</u>. In March2003, Entergy maintenance personnel did not install the primary disconnects for 480 tie breaker 3AT6A in accordance with PM procedure BRK-003-A which is a performance deficiency associated with the Mitigating Systems cornerstone, and is contrary to NRC regulations. Traditional enforcement does not apply because an event did not occur that resulted in an actual safety consequence, the failure to have an adequate procedure did not impact the NRC's regulatory function, and it was not the result of a willful violation of NRC requirements or Entergy procedures. The finding is greater than minor since it affects the Mitigating Systems cornerstone to ensure availability of the RHR system and to prevent undesirable consequences such as core damage due to lack of core cooling during plant shutdown. The performance finding affects the Mitigating Systems cornerstone attribute of procedural quality (breaker PM procedure).

The inspector evaluated the risk significance of this issue using IMC 0609 Appendix G. IMC 0609 Appendix G is applicable during a refueling outage when Entergy has initiated RHR cooling. At the time of the actuation, the unit was in refueling with the refueling cavity greater than 23 feet above the reactor vessel flange. The inspector used IMC 0609 Appendix G, Phase 1, Checklist 4, PWR Refueling Operation: RCS level greater than 23 feet with time to boil greater than 2 hours (actual time to boil greater than 8 hours). The performance finding involved less than the minimum required RHR loops (for 6 minutes) and less than the minimum DC available sources (for 38 minutes). No loss of thermal margin occurred due to the loss of RHR for 6 minutes since it was well below the 20 percent margin on time to boil. The performance finding did not require a Phase 2 or Phase 3 analysis since the finding did not increase the likelihood of a loss of RCS inventory, did not degrade Entergy's ability to terminate a leak path or add RCS inventory when needed, or degrade Entergy's ability to recover RHR once it was lost. The finding was determined to be of very low safety significance (Green).

This finding is associated with the cross-cutting area of human performance, in that maintenance personnel did not implement a 480 volt breaker PM procedure correctly. This performance issue impacted the availability of RHR safety function during 2RF16. (See Section 4OA4).

Enforcement. 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures and Drawings,” states, in part, that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and be accomplished in accordance with these procedures. Contrary to this, on March 19, 2003, Entergy did not accomplish procedure BRK-P-003-A step 8.7.2 to reinstall the primary disconnect contacts for 480 volt tie breaker 3AT6A. This violation existed until April 18, 2003, when the breaker was installed in the electrical bus and on November 9, 2004, the consequence of this error resulted in a loss of 480 volt bus 6A. Because this failure to implement adequate maintenance for a 480 volt safeguards breaker was entered into Entergy’s CAP (CR IP2-2004-5927 and IP2-2004-6033), this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000247/2004012-03, Inadequate PM procedure implementation resulting in a loss of safeguards bus 6A)**

1R14 Personnel Performance During Non-routine Plant Evolutions and Events (71111.14 - 4 samples)

a. Inspection Scope

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 October 4, 2004, Entergy closed the No. 2 control valve on the main turbine due to failure of the No. 2 stop valve. During normal operations all four control valves are full or partially open depending on reactor power. During a plant trip the control valves and their associated stop valves all shut, securing steam flow to the turbine. With No. 2 stop valve failed it was determined that the No. 2 control valve should be shut to prevent a single failure from causing a turbine overspeed condition. A temporary procedure, 2-TOP-003, was written to perform this evolution since this condition is not covered in the normal operating procedures. The inspectors reviewed the procedure, observed the special evolution brief provided to the control room operations staff and observed the operator actions during performance of this evolution. The inspectors also evaluated plant response to ensure it conformed to documented expectations and that plant parameters remained within procedural limitations.

- On November 20, 2004, a Swagelok fitting on the transmitter for 21 reactor coolang system (RCS) flow failed resulting in a 1.9 gpm RCS leak inside containment.

- On November 26, 2004, the inspector observed the control room and plant operator activities during an automatic turbine trip/reactor trip. The inspector arrived in the control room within approximately one minute of the automatic turbine trip/reactor trip and observed that immediate and supplemental actions were completed in accordance with procedures E-0, “Reactor Trip or Safety Injection,” Rev. 45 and ES-0.1, “Reactor Trip Response,” Rev. 42. The inspector also observed various activities related to troubleshooting an apparent low flow condition in the main generator’s stator water cooling (SWC) system.

- On December 3, 2004, a plant down power occurred to repair a steam leak on the 1st stage pressure transmitter (PT-412B). The inspectors observed operators during the plant down power, confirmed weld repairs to the pressure transmitter, and evaluated the outage scope of maintenance activities.

b. Findings

.1 Introduction. The inspectors identified a self-revealing Green finding involving poor causal analysis associated with the main generator SWC system. The specific causal analysis was associated with the trip setting and pressure switch performance associated with generator protection trip/alarm pressure switch (63-P79). The finding resulted in an automatic reactor trip due to a low inlet pressure condition on the main generator SWC system.

Description. An automatic main generator/turbine/reactor trip occurred on November 26 at 1:22 p.m. due to an erroneous system low inlet pressure trip on the main generator SWC system. The purpose of the SWC system is to provide cooling for the main generator rectifier banks.

During 2RF16, Entergy completed a number of corrective maintenance issues associated with the SWC system. The activities involved cleaning of strainers and filters, overhauling of temperature control valve (Y-07), replacing head gaskets for both coolers, and overhauling of flow control valve (Y-63).

On November 19, 2004, maintenance on the SWC system was completed and operators attempted to place the system in service. Numerous system leaks and an unexpected main generator protection circuit energized on low system pressure. Entergy initiated CR IP2-2004-6339 to document system problems during restoration. The causal analysis for the unexpected generator protection trip was not fully evaluated. Specifically, pressure switch (63-P79) reset was set too high (above normal system pressure) and could not be reset until mechanically agitated.

At approximately 8:00 a.m., on November 26, 2004, Entergy noticed that system flow increased unexpectedly and operators noticed that the lower limit mechanical stop on flow control valve (Y-63) appeared to have loosened.

On August 26 and September 1, 2004, two system transients occurred on the stator cooling water system. On August 26, 2004, operators shifted SWC pumps and an unexpected low pressure alarm occurred. On September 1, 2004, the system cooled down following a reactor trip and an unexpected generator protection trip occurred. Causal analysis for these occurrences were ineffective.

Analysis. The performance issue was ineffective causal analysis associated with the SWC system during the period August 26 through November 26. The poor causal analysis was associated with the settings of the generator protection trip pressure switch (63-P79) as it related to system hydraulic performance. Traditional enforcement does not apply since no actual safety consequences existed, it did not impact NRC's ability to perform its regulatory function, and there were no willful aspects of the violation. The finding is more than minor since it impacted the Initiating Event cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions, and was associated with the equipment performance attribute. Specifically, the finding affected the likelihood of a reactor trip and challenged the critical safety function of AFW initiation. In accordance with Phase 1 of the SDP, the performance finding does not contribute to both the likelihood of a reactor trip and the likelihood of mitigation equipment functions being unavailable, thus the finding is screened to very low risk significance (Green).

This finding is associated with the cross-cutting area of PI&R based on the ineffective causal analysis for deficiencies in the SWC system between August 26 and November 26. The ineffective causal analysis was associated with the settings of the generator protection trip/alarm pressure switch (63-P79) as its relationship to system hydraulic performance. The finding impacted the Initiating Events cornerstone which resulted in an automatic generator/turbine/reactor trip (see Section 4OA4.6).

Enforcement. No violation of regulatory requirements occurred. The inspector determined that the inadequate corrective actions were not associated with safety-related equipment and, therefore, did not fall under the requirements of 10 CFR 50 Appendix B. **(FIN 05000247/2004012-04: Inadequate corrective actions associated with a SWC pressure switch)**

This event was documented in plant CR IP2-2004-06467. Corrective actions included development of single point failure analysis on a number of balance of plant equipment, improvements in trending of system issues, and development of an integrated post work test procedure to perform SWC system startup after each refueling outage.

.2 Introduction. The inspectors identified a self-revealing Green non-cited violation of 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures and Drawings.” Maintenance personnel did not verify that the length of tubing between the RACK 20 bulkhead connection and existing 21 Reactor Coolant Loop Flow (FT-416) Hi side impulse tubing was sufficient for a proper Swagelok connection pursuant to procedure IP-SMM-MA-108. Reactor coolant system leakage was estimated at approximately 1.9 gpm and lasted for 39 minutes until isolated by operators.

Description. On November 19, 2004, during plant heatup (RCS pressure at 2,040 psig, reactor sub-critical), a failure of a Swagelok fitting between a ½ inch RCS flow instrument tubing upstream of flow transmitter (FT)-416 and root stop valve 513CX1 resulted in tubing coming apart from the fitting. Operators entered into the applicable AOP (AOP-LEAK-1, “Excessive Reactor Coolant System Leakage” Rev. 4) and isolated the leak inside containment in approximately 39 minutes.

During 2RF16, Entergy implemented plant modification ER-04-2-033 that replaced twelve obsolete RCS loop flow transmitters and the associated instrument rack and power supplies. On November 4, 2004, contractors completed all tubing re-connections based on new transmitter rack installation.

The modification WO provided only one step and one sign off for contractors to tighten all Swagelok fittings (75 total) and for quality control inspectors to perform gap inspections per Step 6.6 of procedure IP-SMM-MA-108, “Swagelok Tubing Connections.” Entergy’s investigation found that correct assembly of the fitting did not occur due to interferences and that the length of tubing between RACK 20 and the FT-416 impulse tubing was insufficient.

Analysis. The performance issue is that quality procedures were inadequately implemented during modification ER-04-2-033, specifically associated with the Swagelock fitting associated with 21 reactor coolant loop flow transmitter. Traditional enforcement does not apply since no actual safety consequences existed, it did not impact NRC’s ability to perform its regulatory function, and there were no willful aspects of the violation. The finding is more than minor since it impacted the Initiating Event cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions, and was associated with the procedural quality attribute. Specifically, the finding affected the likelihood of a RCS leak that upset plant stability and challenged critical safety functions. In accordance with Phase 1 of the SDP, assuming worst case degradation, the finding would not result in exceeding the TS limit for identified leakage (10 gpm) and did not affect other mitigation systems resulting in a total loss of their safety function. The finding is screened to very low risk significance (Green).

Enforcement. 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” in part, requires that activities affecting quality shall be prescribed by documenting instructions of a type appropriate to the circumstances and shall be accomplished in accordance within these instructions. Contrary to the above, contractor maintenance personnel improperly installed a length of tubing between the instrument rack and the RCS loop flow Swagelok fitting. The installation work package did not provide adequate instruction to use a pre-swaging tool per Section 6.4 of IP-SMM-MA-108 for Swagelok installations in difficult locations. **(NCV 05000247/2004012-05: Improper installation of RCS loop flow tubing resulting in RCS leakage)**

Contractor maintenance personnel completed the connections on November 4, 2004. The connection failed during RCS pressurization on November 19. The actual consequence was an isolable RCS leak of approximately 1.9 gpm for approximately 39 minutes.

This event was documented in plant CR IP2-2004-06301. Corrective actions included replacement of the tubing from the rack to the transmitter, verification of other Swagelok fittings associated with modification ER-04-2-033, and extent of condition reviews on seven other safety-related systems.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

The inspectors selected operability evaluations that Entergy had generated that warranted review on the basis of potential risk significance. The selected samples are addressed in 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 and compared to the TSs, Technical Requirements Manual (TRM), the FSAR, and associated design basis documents.

- CR IP2-2004-5022, component cooling water heat exchanger acceptance criteria for tube plugging
- CR IP2-2004-6150, service water leak to 21 EDG
- CR IP2-2004-5343, 22 recirculation motor data analysis
- CR IP2-2004-5378, failure to achieve designed power factor on 22 EDG during surveillance testing
- CR IP2 2004-5497, -5505, -5559, -5752, -5844, -5916, -5930, -5956, -5957 containment cable separation issues

b. Findings

No findings of significance were identified. Two licensee-identified findings are documented further in report detail 4OA7.

1R17 Permanent Plant Modifications (71111.17B - 7 samples)

a. Inspection Scope

The inspectors reviewed seven risk-significant plant modification packages selected from among the design changes that were completed within the past two years. The review was to verify that: (1) the design bases, licensing bases, and performance capability of risk significant SSCs had not been degraded through modifications; and, (2) modifications performed during increased risk configurations did not place the plant in an unsafe condition.

The selected plant modifications were distributed among Initiating Event, Mitigating System, and Barrier Integrity cornerstones. For these selected modifications, the inspectors reviewed the design inputs, assumptions, and design calculations to determine the design adequacy. The inspectors also reviewed field change notices that were issued during the installation to confirm that the problems associated with the installation were adequately resolved. In addition, the inspectors also reviewed the post-modification testing, functional testing, and instrument and relay calibration records to determine readiness for operations. Finally, the inspectors reviewed the affected procedures, drawings, design basis documents, and FSAR sections to verify that the affected documents were appropriately updated.

For the accessible components associated with the modifications, the inspectors also walked down the systems to detect possible abnormal installation conditions.

The following seven modifications were reviewed:

- CGP-92-07512-C Central Control Room Anchorage Modification;
- FPX-97-12766-F Secondary Boiler Blowdown Purification System Piping Seismic Upgrade;

•	FCX-99-12029-C	Central Control Room HVAC Carbon Filter Bypass Leakage;
•	MMS-87-60819-00	Removal of Pipe Whip Restraint PWR-124;
•	DCP-200107381-E	Functional Removal of the BGIS/BGIR Circuit in the Emergency Diesel Generator Control Circuits;
•	DCP-02-2-005	Station Auxiliary Transformer Load Tap Changer - Safety Injection Modification;
•	DCP-02-2-006	Emergency Diesel Generator 22 Wiring and Terminations Upgrade.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 7 samples)

a. Inspection Scope

The inspector reviewed post-work test (PWT) 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 WO 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 IP2 Individual Plant Examination (IPE). The regulatory references for the inspection included TSs and 10 CFR 50, Appendix B, Criterion XIV, "Inspection, Test, and Operating Status." The following testing activities were evaluated:

- WO IP2-04-25578, feedwater regulating valve (FCV-437) following maintenance
- WO IP2-04-23687, verification of no leak-by on safety injection valve SI-858A
- PWT IP2-04-17593, verification of 21 charging pump separation modification
- WO IP2-04-15444, in-service leak test of RCS RTDs from ER-04-2-005
- WO IP2-04-20732, 21 stator cooling pump rotating element replacement between November 29 - December 3
- WO IP2-04-34771, 24 atmospheric dump PCV-1137 stroke test between November 29 and December 2
- WO IP2-04-30654, reconstitution of fuel assembly S-48

b. 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 2RF16. The refueling outage occurred between October 22 through November 22, 2004.

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 October 22, 2004. 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 scrammed at 12:00 a.m. on October 22 as part of the normal shutdown sequence.

The inspectors observed the operators conducting the shutdown using procedures POP 3.1, "Plant Shutdown," and POP 3.3, "Plant Cooldown," and controlled plant parameters within the requirements of TS 3.4.3-2. The inspectors verified that operators used the appropriate pressure and temperature instrumentation during cooldown. The reactor was cooled down below 350 F using the AFW system until RHR was placed in service. The plant entered cold shutdown with the RCS less than 200 F at 9:42 a.m. on October 23.

The inspectors observed the operators response to changing equipment conditions with the use of alarm response procedures and AOPs, when appropriate. 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 2RF16. Areas specifically evaluated during the outage were:

- Containment to perform a boric acid walkdown of the RCS

- EDG building
- EDG fuel oil transfer system
- Augmented SFP cooling system
- RHR system
- Low temperature overpressure protection system and controls
- Primary auxiliary building (PAB)
- AFW building
- Turbine building

During 2RF16 the inspectors periodically verified adequate shutdown margin in accordance with TS 3.1.1.1 and Entergy graphs RCS-4 and RCS-6. The inspectors independently verified the adequacy of system tagout isolation and configuration controls. Specific items verified included:

- Low temperature overpressure protection injection source isolation (2R16-01062-01065)
- 21 SI pump (2-SIS-SI Pump 21 Rev 0-0)
- 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
- 21 backup pressurizer heaters

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 2-PT-W019, "Electrical Verification - Offsite Power Sources and AC Distribution"

The inspectors periodically verified through control room instrument panel and in-field walkdowns operation of the shutdown cooling system and compared observations consistent with system operating procedure (SOP) 4.2-1 and TS requirements. The inspectors periodically verified proper operation of the SFP cooling system.

Reduced Inventory and Mid-Loop Conditions

On October 26, 2004 and November 12, 2004, 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.

On November 11, 2004, the reactor cavity was drained in preparation for reinstallation of the reactor vessel head, and communications problems inadvertently resulted in a vessel level 2 feet below the intended level. The inspectors reviewed the event details by means of logs and parameter traces, and preliminary event evaluations, including operating personnel interviews.

Refueling Activities

On October 29, November 8, and November 9 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. On November 4, 2004, the inspectors observed non-destructive examinations of fuel assemblies S-48, S-32, and S-45.

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

Introduction. The inspector identified a self-revealing Green non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings" associated with a reactor vessel water level control issue during the drain down for the reactor vessel head re-installation on November 11, 2004.

Description. On November 11, 2004, Entergy drained the 11 feet of water above the reactor vessel flange plus 1.2 feet additional to enable reinstallation of the reactor vessel head. The procedurally specified rate of draining was to be 2500 gpm until 2 feet above the flange, then 1200 gpm until at the flange, and then 120 gpm until at the intended end point 1.2 feet below the flange; the water was being pumped by the RHR system to the refueling water storage tank (RWST) in parallel with providing reactor cooling. Note that the reactor cavity above the vessel flange has a much greater water surface area than that of the reactor vessel, such that at the same drain rate, the level drops seven times faster below the flange compared to above the flange.

Concurrently with the planned transition from the 1200 gpm rate to the 120 gpm rate, radio communications were temporarily lost to the nuclear plant operator (NPO) in the field at the time to instruct him to throttle a valve to provide the lower drain rate. In the few minutes needed to reach the NPO by alternate means and to instruct another NPO to isolate the drain path via another means, the level dropped from 69 feet (flange) past the intended end point of 67.8 feet to 65.7 feet. At the existing 1200 gpm drain rate, the level was dropping about a foot every minute.

While the drain path was being stopped (both the throttle valve and a motor-operated isolation valve were closed), control room operators maximized the charging flow, and they recovered the level to 66 feet within three minutes and to the intended level of 67.8 feet within 30 minutes.

A subsequent Entergy evaluation found that the hand-held radio used by the NPO had experienced a battery failure, i.e., a capacity reduction to less than a shift due to age-related degradation.

Analysis. The inspectors concluded that the drain down rate was unnecessarily rapid in the vicinity of the reactor vessel flange, such that there was little margin for error, and that this represented a performance issue. At IP2 stopping the draining needed to be performed in the field (no controls in the control room), and the IP2 procedure (2-SOP-1.2, "Draining Reactor Coolant System") specified that the drain rate from 2 feet above the flange to the flange be approximately 1200 gpm or less and that the transition to approximately 120 gpm occur at the flange. For comparison purposes, IP3 had controls to close the isolation valve in the control room (thus not totally dependent on field actions), and the IP3 procedure (3-SOP-RP-020, "Draining the RCS/Refueling Cavity") specified the drain rate at 2 feet above the flange be less than 500 gpm and no reduction was needed at the flange. Thus, on three criteria - (1) ability to stop the draining from the control room, (2) drain rate approaching the flange, and (3) the need to reduce the drain rate at the flange, the IP3 approach was more conservative than the IP2 approach.

This finding is more than minor, because it potentially affected the Initiating Events cornerstone objective of limiting the likelihood of events that challenge critical safety functions during shutdown, and was associated with the procedural quality attribute. The consequences of not reducing the drain rate at the flange was that reactor vessel level was dropping rapidly, approximately one foot per minute, and within minutes could reach the hot leg nozzles four feet below the flange, potentially causing the RHR cooling suction path to be lost. During the November 11 event, the hot leg nozzles were not reached nor was RHR suction lost.

The Senior Reactor Analyst (SRA) evaluated this performance deficiency using IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." Based upon the inadvertent loss of RCS inventory in excess of two feet, this performance issue was screened using the Phase 1 Checklist 3, "PWR Cold Shutdown and Refueling Operation, RCS Open and Refueling Cavity Level< 23 Feet, Time to Boiling < 2 Hours." The SRA determined that the finding resulted in a potential increase in likelihood of RCS inventory loss, prompting a Phase 2 evaluation.

Phase 2 Worksheet 6, "SDP for a PWR Plant - Loss of Inventory in POS 2 (RCS Vented)," was used to assess this particular finding and the following assumptions were made: 1) the initiating event likelihood credit was zero, based upon the relatively short time period before the RHR pump suction would have been lost, given the rapid (1200 gpm) level drop; 2) time to boil was approximately 37 minutes, based upon Entergy's calculation; and, 3) based upon the gravity reflood of the vessel and cavity without operator action, the Worksheet 6 "RCS injection before CD" top event and associated core damage sequences were determined to be always successful (see inspector's review in following paragraph). Based upon this Phase 2 evaluation, this performance deficiency was of very low risk significance (Green).

The inspectors reviewed the basis for the self-limiting aspect (i.e., RWST gravity reflood of reactor) of the event risk analysis, including Calculation IP-CaLC-04-01731. If the level had dropped below the hot leg suction for the RHR pump, the pump would have cavitated and tripped. Then, due to the RWST being at a higher elevation, the RWST flow would have reversed to flow toward the reactor vessel. In effect, the two discharge paths (cooling to the reactor and RWST) with the pump would have become a discharge path (vessel) and a supply path (RWST) without the pump. The calculation was thorough, modeled the piping in detail, and showed that sufficient motive force would have existed to produce flow above 1000 gpm. The inspectors also reviewed a May 1989 surveillance test at IP3 which demonstrated gravity flow from the IP3 RWST to the IP3 reactor vessel of well above 1000 gpm; the IP2 and IP3 configurations are similar.

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall be prescribed in documented procedures and shall be accomplished in accordance with these procedures. Contrary to this requirement, on November 11, 2004, the IP2 reactor cavity was drained using procedure 2-SOP-1.2, "Draining the Reactor Coolant System," but was not accomplished in accordance with Step 4.4.5, which specified that "when the RCS level is at a level no less than 67 foot elevation, stop the drain down," in that RCS level was drained to the 65.7 foot elevation. However, because of the very low safety significance and because the issue was entered into Entergy's CAP via CR IP2-2004-05991, this finding is being treated as a non-cited violation, consistent with Section VI.A of the Enforcement Policy, issued May 1, 2002 (65FR25368). **(NCV 05000247/2004012-06; Failure to follow RCS drain down procedure due to inappropriate approach)**

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors reviewed surveillance test procedures and observed testing activities to assess whether: 1) the test preconditioned the component tested; 2) the effect of the testing was adequately addressed in the control room; 3) the acceptance criteria demonstrated operational readiness consistent with design calculations and licensing documents; 4) the test equipment range and accuracy were adequate and the equipment was properly calibrated; 5) the test was performed per the procedure; 6) test equipment was removed following testing; and 7) test discrepancies were appropriately evaluated. The surveillance tests observed were based upon risk significant components as identified in the IP2 IPE. The regulatory requirements that provided the acceptance criteria for this review were 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," Criterion XIV, "Inspection, Test, and Operating Status," Criterion XI, "Test Control," and TS 6.8.1.a. The following test activities were reviewed:

- PT-R06, "Main Steam Safety Valve Setpoint Determination," rev. 23 performed on October 22, 2004
- PT-R013, "Safety Injection System," rev. 25 performed on October 24 - 25, 2004
- 2-PT-R014, "Automatic Safety Injection System Electrical Load and Blackout Test," rev. 20 performed on October 25 - 26, 2004
- PT-R007A, "Motor Driven Auxiliary Boiler Feed Pump Full Flow," rev. 16 performed on October 18, 2004
- PT-Y28C, "23 Emergency Diesel Generator Mechanical Overspeed Test," performed on October 5, 2004

- PT-R 75, “Reactor Coolant System Pressure Test,“ rev. 10 on November 20, 2004

b. Findings

Introduction. The inspectors identified a Green non-cited violation of 10 CFR 50 Appendix B, Criterion VI, “Document Control.” Inadequate document control resulted in multiple surveillance procedures not meeting the criteria of the improved technical specifications (ITS), surveillance requirements (SRs), or the applicable ITS basis document.

Description. While performing routine baseline inspections the inspectors found multiple examples of surveillance tests in which the specified ITS SRs would not be met had the test been performed as written or the surveillance test did not conform to the limitations set forth in the applicable basis document.

2-PT-R014, “Automatic Safety Injection System Electrical Load and Blackout Test,” did not include requirements for a greater than five minute run or the appropriate frequency requirements for the EDGs as stated in SR 3.8.1.12. The acceptance criteria still referenced the requirements of the previous Custom Technical Specifications (CTS). This was identified by the inspectors and corrected by Entergy prior to the surveillance being performed.

The inspectors identified that 2-PT-R013, “Safety Injection System,” did not meet all of the requirements for EDG trips and bypasses as stated in SR 3.8.1.9. Entergy created three additional surveillance tests to ensure the appropriate requirements were met.

While reviewing PT-2Y 8C, “23 EDG Mechanical Overspeed Trip,” the inspectors found that the document stated this procedure met the requirements of SR 3.8.1.9. Entergy was preparing to perform this test during power operations. The inspectors identified that this specific SR should not normally be performed while at power and informed Entergy of this requirement. Entergy determined that this test did not actually meet the requirements of SR 3.8.1.9 and therefore could be performed online; however, the operators had not previously questioned this precaution in the procedure.

On December 12, 2003, Entergy implemented ITS in place of the previous CTS. Prior to this change Entergy implemented a process to review the SRs associated with ITS to ensure that the current surveillance procedures would meet all the requirements. Where the requirements were not met the procedures were changed to ensure compliance with ITS.

Analysis. The inspectors determined that this was a performance deficiency since surveillance procedures did not adequately implement TS SRs. Traditional enforcement does not apply since there was no actual safety consequence or potential for impacting the NRC’s regulatory function and the finding was not the result of any willful violation of NRC requirements or Entergy’s procedures. The inspectors determined that the finding was greater than minor since if left uncorrected it would become a more significant safety concern potentially impacting multiple SRs of safety-related equipment and equipment important to safety. This creates the potential for equipment to be inoperable due to not meeting the acceptance criteria of the ITS SRs without Entergy having knowledge of the condition since the criteria or requirements may not be reflected in their surveillance procedure. This performance finding affects the Mitigating Systems cornerstone attribute of procedural quality. The inspectors conducted a Phase 1 SDP screening and determined that the failure to maintain proper document controls was of very low safety significance since it had not resulted in a loss of safety function or in any inoperable equipment.

Enforcement. 10 CFR 50 Appendix B, Criterion VI, "Document Control," states in part that measures shall be established to ensure that documents, including changes, are reviewed for adequacy. Contrary to this, Entergy's surveillance review process as part of the conversion to ITS was not sufficient to ensure that the ITS SRs would be met by the procedures in their surveillance program as indicated by multiple deficiencies noted in several surveillance procedures sampled by the inspectors. Because this failure to maintain proper document control is of very low safety significance and has been entered into Entergy's CAP (CR IP2-2004-5326) this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: **(NCV 05000247/2004012-07: Multiple deficiencies in surveillance procedures associated with ITS conversion**)

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

Based on observations from a plant status walkdown in the AFW building on October 20, the inspectors reviewed documentation for a temporary leak repair on feedwater piping for the 24 SG. The purpose of this temporary repair was to stop a minor steam leak on a threaded cap on the end of the vent line downsteam of BFD-93-3 (high point vent isolation valve for the 24 SG feed regulating valve, low flow bypass line). Work Order IP2-03-06228 was initiated to evaluate and repair the steam leak and WO IP2-03-29437 tracks the permanent repair of the affected valve and cap. The inspectors reviewed the associated engineering evaluations and in-plant configuration to ensure that Entergy implemented this modification in accordance with plant procedures and in compliance with 10 CFR 50.59. The specific information reviewed is referenced in the Supplemental Information attachment at the end of this report.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

On December 15, 2004, the inspectors observed a full siren activation and the feedback system from the Indian Point Emergency Operating Facility and the Westchester County Emergency Operating Center. The inspectors reviewed siren reliability and other results from the test to verify the accuracy of announced results, as well as Entergy's follow-up of the full siren test to ensure that problem areas were properly identified.

Additionally, the inspectors reviewed siren availability following the failure of 88 of 156 sirens on November 4, 2004. The inspectors reviewed CR IP3-2004-0397 to determine that Entergy personnel took appropriate corrective actions.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 14 samples)

a. Inspection Scope

During November 1-5, 2004, the inspector conducted the following activities during 2RF16 to verify that Entergy was properly implementing physical, engineering, and administrative controls for access to high radiation areas (HRAs), and other radiological controlled areas (RCAs), 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 TSs, and Entergy's procedures.

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

- resistance temperature detector (RTD) replacements on the RCS
- scaffold erection inside Unit 2 containment
- outage valve work - 21 charging pump check valve 4000 and 4001 replacement
- in-service inspection - reactor head weld and penetration inspection
- radiation protection (RP) support of outage work activities in Unit 2 containment and PAB

(2) Radiation work permits (RWPs) associated with the above work activities were reviewed with respect to HRA 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 RWP, procedure, and engineering controls were in place, and whether Entergy's 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 RP 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, RP technician performance was evaluated with respect to RP procedure and work activity radiological SRs.

(8) There were no internal exposure dose assessments for review that were greater than 50 mrem committed effective dose equivalent (CEDE) during 2004 at either Unit 2 or Unit 3.

(9) There were no Occupational Exposure performance indicator (PI) occurrences for review during 2004.

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

(11) An interview with the RP manager was conducted to verify that procedural controls would be enacted before commencing certain plant operations that have the potential to become very high radiation areas, such as thimble withdrawal into the reactor cavity sump.

(12) Based on the CRs reviewed (see Section 4OA4.3), no repetitive deficiencies were identified for further followup.

(13) Condition reports reviewed (see Section 4OA4.3) were evaluated with respect to traceable trends in radiation worker performance.

(14) Condition reports reviewed (see Section 4OA4.3) were evaluated with respect to traceable trends in RP technician performance.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 3 samples)

a. Inspection Scope

During October 18-20 and November 1-5, 2004, the inspector conducted the following activities to verify that Entergy was properly maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). Implementation of the ALARA program for the site (IP2 and IP3) was reviewed against the criteria contained in 10 CFR 20.1101(b) and Entergy'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:

- Replace RTDs: 26.9 person-rem estimate
- Reactor disassembly/reassembly: 25 person-rem estimate
- Refurbish valves: 15 person-rem estimate
- Reactor head insulation modification: 6.5 person-rem estimate
- In-service inspection: 5.7 person-rem estimate

(2) Based on the work activities listed in (1) above, the conduct of these work activities was observed with respect to Entergy'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 RP 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.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151 - 3 samples)

Inspection Scope

The inspector reviewed Entergy's PI data for the High Pressure Injection System indicator to verify whether the data was accurate and complete. The inspector compared the PI data reported by Entergy to information gathered from the control room logs, CRs, and WOs for the 3rd, 4th quarters of 2003 and the 1st 2nd, 3rd quarters of 2004. In addition, the inspectors compared the PI data against the guidance in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2.

The inspector reviewed implementation of Entergy's Occupational Exposure Control Effectiveness Performance Indicator (PI) Program. Specifically, the inspector reviewed CRs, and RCA dosimeter exit logs for the past four calendar quarters. These records were reviewed for occurrences involving locked HRAs, very high radiation areas, and unplanned exposures against the criteria specified in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 2, to verify that all occurrences that met the NEI criteria were identified and reported.

The inspector reviewed a listing of relevant effluent release reports for the past four calendar quarters, for issues related to the Public Radiation Safety PI, which measures radiological effluent release occurrences per site that exceed 1.5 mrem/qtr whole body or 5.0 mrem/qtr organ dose for liquid effluents; 5mrads/qtr gamma air dose, 10 mrad/qtr beta air dose, and 7.5 mrads/qtr for organ dose for gaseous effluents.

The inspector reviewed the following documents to ensure Entergy met all requirements of the PI:

- monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases;
- quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases; and
- dose assessment procedures.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71111.08 and 71152)

.1 In-Service Inspection Related Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the corrective action reports listed in the Supplemental Information attachment at the end of this report, which involved ISI related problems, to ensure that Entergy properly addressed these issues.

b. Findings

Introduction. The inspectors identified a Green non-cited violation of 10 CFR 50 Appendix B, Criterion XVI, “Corrective Action,” for Entergy’s failure to properly address a condition adverse to quality involving leakage from a canopy seal weld onto the reactor vessel head (RVH) in November 2002.

Description. In October 2004, during 2RF16, Entergy discovered an amount of boron on the reactor vessel upper head near penetration 86. Entergy determined that the boron came from a control rod drive mechanical joint canopy seal weld (conoseal) leak that had been identified in November 2002 during the in-service pressure test and prior to the reactor start-up from 2RF15. The leak was documented and evaluated in CR-IP2-2002-10986. The inspectors interviewed Entergy personnel and reviewed the evaluation for CR-IP2-2002-10986 and determined that Entergy had not performed an inspection to ensure that the conoseal leakage had not reached the RVH. As a result, the boron remained on the RVH from November 2002 to November 2004. During 2RF16, Entergy removed this boron and inspected the RVH and did not identify any damage.

Analysis. The performance deficiency was ineffective corrective actions for a conoseal leak which led to boron accumulation on the RVH. The finding was considered to be more than minor because it could have led to a more significant problem. Specifically, the boric acid, if re-wetted, could have led to accelerated corrosion of the RVH. During the operating cycle, additional conoseal leaks developed which demonstrated the potential to wet existing boric acid on the RVH. This finding was evaluated using the Phase I worksheet of the NRC's Significance Determination Process and determined to be of very low significance (Green) since the RVH integrity was not affected by the performance deficiency.

This finding was associated with the cross-cutting area of PI&R, in that, Entergy did not take adequate corrective actions for the conoseal leak that they had initially identified in November 2002 (see Section 4OA2.6).

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, and nonconformances are promptly identified and corrected." Contrary to the above, Entergy did not implement appropriate corrective actions for conoseal leakage above the RVH for leakage that existed from November 2002 until the time of the inspection in November 2004. Because this performance deficiency was determined to be of very low significance since there were no actual safety consequences, and has been entered into Entergy's CAP (CR-IP2-2004-06050), this violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000247/2004012-08, Failure to properly address a condition adverse to quality involving leakage from a canopy seal weld onto the RVH in November 2002)**

.2 Daily Review (71152)

a. Inspection Scope

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 CAP. This review was accomplished by reviewing hard copies of each CR.

b. Findings

No findings of significance were identified.

.3 PI&R Radiation Protection Sample - Occupational Radiation Safety (71121)

a. Inspection Scope

The inspector reviewed 15 corrective action CRs that were initiated between May 2004 and November 4, 2004, and were associated with the RP program. The inspector verified that problems identified by these CRs were properly characterized in Entergy'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.

.4 PI&R Sample - Plant Modifications and 10 CFR 50.59 Evaluations

a. Inspection Scope

The inspectors reviewed CRs associated with 10 CFR 50.59 issues and plant modification issues to ensure that Entergy was identifying, evaluating, and correcting problems associated with these areas and that the planned or completed corrective actions for the issues were appropriate. The inspectors also reviewed five self-assessments related to 10 CFR 50.59 safety evaluation and plant modification activities at IP2.

The listing of the CRs and self assessments reviewed is provided in the Supplemental Information attachment at the end of this report.

b. Findings

No findings of significance were identified.

.5 PI&R Semi-annual Trend Review (71152 - 1 sample)

a. Inspection Scope

The inspectors reviewed Entergy's CAP database over the last two calendar quarters of 2004 in order to assess the total number and significance of CRs written in various subject areas such as equipment and processes. The results were evaluated on a per quarter basis to identify any notable trends. The assessment specifically consisted of CR reviews in the following areas:

C Level "A" CRs: which required a full root cause analysis and review by the Corrective Actions Review Board (CARB) prior to closeout; and Level "B" CRs: which required an apparent cause evaluation and an optional CARB review.

- The number and significance of CRs associated with plant equipment previously identified as having reliability issues.

- A review of the corrective action database to assess trends in the number of CRs written in the previous two quarters that were related to subject areas that reflect the quality of maintenance, work controls, operations, procedures, etc.

- A review of the Indian Point Energy Center Quarterly Integrated Self-Assessment/Trend Reports for 2Q04 and 3Q04 written by the IPEC Quality Assurance Department, which contained Entergy's assessments of CR trends during those quarters.

b. Findings

No findings of significance were identified.

.6 Cross-Cutting Aspects of Findings (PI&R)

Section 1R14.1 describes a finding associated with the cross-cutting area of PI&R based on ineffective causal analysis for deficiencies in the SWC system between August 26 and November 26. The ineffective causal analysis was associated with the settings of the generator protection trip/alarm pressure switch (63-P79) as its relationship to system hydraulic performance. The finding impacted Initiating Events cornerstone which resulted in an automatic generator/turbine/reactor trip.

Section 4OA4.1 describes a finding associated with the cross-cutting area of PI&R, in that, Entergy did not take adequate corrective actions for a conoseal leak that they had initially identified in November 2002. The finding leakage had the potential to adversely impact the Barrier Integrity cornerstone as the boric acid, if re-wetted, could have led to accelerated corrosion of the RVH.

4OA4 Cross-Cutting Aspects of Findings Involving Human Performance

Section 1R04 describes a finding associated with the cross-cutting area of human performance, in that, operators did not adequately assess a change in the sequence of procedures performed during the refueling outage. Due to the resulting valve misalignment, a backup method for establishing post-accident high head recirculation using the RHR pumps and their associated sump was not available. This error impacted the availability of mitigating systems.

Section 1R13.2 describes a finding associated with the cross-cutting area of human performance, in that maintenance personnel did not implement a 480 volt breaker PM procedure correctly. This performance issue impacted the availability of RHR safety function during 2RF16.

4OA5 Other Activities

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

a. Inspection Scope

The inspectors reviewed Entergy'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 Entergy'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 selected 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 VT to verify that the penetration intersection location could be fully accessed to perform a 360-degree examination. The inspectors noted that the original inspection procedure was modified to require manual (vs. automatic) video taping of some LHP nozzles due to interference between the reactor vessel lower head insulation and the LHP nozzles. The inspectors also reviewed the radiological exposure impact associated with the revised LHP inspection method (approximately 4.8 man-rem was expended for this job versus an estimated 1.2 man-rem). The inspectors reviewed the disposition and actions for observations related to oxidation, debris and dried boron that impacted the viewing of some penetrations on the lower head and also Entergy's efforts to clean the lower head to establish a baseline for future inspections. The inspectors reviewed the following CRs to ensure that Entergy's actions were appropriate:

- CR-IP2-2002-10417, "Dried boron on the reactor vessel lower head;"
- CR-IP2-2004-05290, "Boric acid residue throughout the reactor vessel lower head penetration area;"
- CR-IP-2004-05292 and CR-IP-2004-05351, "Partial reexamination required for twenty-eight reactor vessel lower head penetrations due to incomplete inspection coverage;" and,
- CR-IP2-2004-05329, "Evaluation required for residue and debris identified during the reactor vessel lower head penetration inspection."

b. Findings

The specific reporting requirements of TI 2515/152 are documented in Attachment 2.

Introduction. The inspectors identified a Green non-cited violation of 10 CFR 50 Appendix B, Criterion IX, "Control of Special Processes," for Entergy's failure to provide adequate inspection criteria and guidance to evaluators prior to the inspection of the reactor vessel lower head penetration nozzles.

Description. The NRC identified that, during the period between October 26 and 27, 2004, Entergy personnel were performing visual inspections of the reactor vessel bottom mounted instrumentation annulus area without adequate procedural guidance to define potential problems or indications. The visual inspections were performed to satisfy Entergy commitments made relative to NRC Bulletin 2003-02. Following inspector identification of this performance deficiency, Entergy revised the inspection procedure to provide detailed guidance for the identification and evaluation of potential indications.

During this period, the inspectors witnessed the evaluation of video taped inspection data and noted the absence of inspection criteria and evaluation guidance in inspection procedure, JTS-2004-BMI, Revision 0, 8/23/04, "Reactor Vessel BMI Visual Inspection for Indian Point 2." The inspectors observed that several penetrations had obvious indications of dried boron present in the lower penetration nozzle annulus area and noted that these indications were not identified by Entergy data evaluators for further evaluation. The inspectors interviewed the two evaluators and noted that they expressed different views regarding what observations to report and what to accept. The inspectors discussed this issue with Entergy management. Subsequently, Entergy initiated CR-IP2-2004-05279 and resolved the problem. The inspectors noted that eventually all indications were dis-positioned as acceptable because Entergy determined that they were not the result of pressure boundary leakage.

Analysis. This performance deficiency, involving an inadequate procedure for inspecting the lower reactor head for pressure boundary leakage, is more than minor because, if left uncorrected, it could have led to a more significant problem. Specifically, the failure to develop adequate inspection guidance could result in a failure to detect a degraded lower RVH penetration boundary. The inspectors evaluated this finding using the Phase I worksheet (RCS Barrier Integrity) of the NRC's Significance Determination Process and determined that this finding was of very low significance (Green) since the RCS pressure boundary was not actually degraded.

Enforcement. 10 CFR 50, Appendix B, Section IX, "Control of Special Processes," states, in part, that "Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements." Contrary to the above, prior to October 27, 2004, Entergy did not develop adequate procedural guidance for the identification and reporting of potential indications of pressure boundary leakage prior to conducting the reactor vessel lower head visual inspections. Because this failure to develop an adequate inspection procedure was of very low significance and has been entered into Entergy's CAP (CR-IP2-2004-05279), this violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000247/2004012-09, Failure to provide adequate inspection criteria and guidance to evaluators prior to the inspection of the reactor vessel lower head penetration nozzles)**

.2 TI 2515/150 - Reactor Pressure Vessel Head and Head Penetration Nozzles (NRC Order EA-03-009)

a. Inspection Scope

The inspectors reviewed Entergy's response to NRC Order EA-03-009 which described the Reactor Pressure Vessel Head and Head Penetration Nozzle inspections. The inspectors reviewed Entergy's Reactor Pressure Vessel Head and Head Penetration Nozzle inspection procedure to determine whether it provided adequate guidance and examination criteria to implement Entergy's examination plan. The inspectors reviewed examination personnel training and qualification records to verify that personnel were properly qualified to perform the examination activities.

The inspectors observed Entergy's inspection activities to verify proper performance of the procedure. Entergy had to revise the original inspection procedure due to interference between the mirror insulation on the RVH and the remote camera. This interference resulted in manually video taping part of the inspection. Some head penetration nozzle intersections were difficult to directly observe due to the initial magnification of the viewing camera and the presence of debris, insulation remnants, and dried boron on the RVH. All of the dried boron was removed and Entergy provided low magnification photographs which showed the penetration intersection area.

The inspectors reviewed video records of the inspection and observed the data evaluation by the Entergy Level III personnel. The inspectors selected several penetration nozzles to evaluate the effectiveness of the VT to verify that the penetration intersection location could be fully accessed to reliably perform a 360-degree examination of the intersection region. The inspectors verified, by observation and review of video tapes, that the RPV head was adequately covered during this inspection.

b. Findings

No findings of significance were identified. The specific reporting requirements of TI 2515/150 are documented in Attachment 3.

.3 Reactor Containment Sump Blockage (NRC Bulletin 2003-01)

Background

On June 3, 2003, the NRC issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," to all pressurized-water reactor licensees requesting that they provide a response within 60 days. On August 7, 2003, Entergy responded and stated that they had implemented the following interim compensatory measures (Option 2 as outlined in Bulletin 2003-01): (1) operator and staff training on indications of and responses to sump clogging; (2) procedural modifications, if appropriate, that would delay the switchover to containment sump recirculation; (3) ensuring that alternative water sources are available to refill the RWST or to otherwise provide inventory to inject into the reactor core and spray the containment atmosphere; (4) more aggressive containment cleaning and increased foreign material controls; (5) ensuring containment drainage paths are unblocked; and (6) ensuring sump screens are free of adverse gaps and breaches.

a. Inspection Scope (TI 2515/153)

During the weeks of October 5 and 12, the inspectors reviewed Entergy's activities in response to NRC Bulletin 2003-01 to assess whether Entergy effectively implemented reasonable compensatory measures. The inspectors independently verified that Entergy had implemented the interim compensatory measures or had planned and scheduled these activities consistent with their response. The inspectors reviewed data gathered during containment walkdowns in 1995 that were performed to quantify debris sources within the containment. The inspectors also reviewed plans for the October 2004 outage to reevaluate this data due to modifications performed since 1995.

The inspectors reviewed operator training records, procedures, documentation of containment inspections and foreign material control activities, and containment sump related corrective action reports. The inspectors also discussed Entergy's Bulletin response with the NRR Project Manager. During the inspection period, the resident inspectors interviewed four operators, representing two operating shifts, to assess their awareness of reactor containment sump blockage issues and expected operator mitigating actions.

b. Findings

No findings of significance were identified with respect to Entergy's response to NRC Bulletin 2003-01. A number of specific observations are listed below.

The inspectors were not able to find a specific lesson plan that presents the mechanisms and potential consequences of sump clogging. The NRC has issued a request for additional information regarding Entergy's response to NRC Bulletin 2003-01 regarding operator training.

Entergy's 60-day response to NRC Bulletin 2003-001 indicates Entergy would participate in a Westinghouse Owners Group (WOG) program to assess potential changes to the generic Emergency Response Guidelines. The NRC has issued a request for additional information regarding Entergy's response to NRC Bulletin 2003-001 regarding procedural modifications.

The inspectors reviewed Entergy procedure OAP-007, "Containment Entry and Egress" and noted that it contains guidance to ensure that no unrestrained items are left in containment that could block a drainage path. Entergy has concluded that additional compensatory measures regarding containment drainage paths are not necessary.

Entergy procedures OAP-007, "Containment Entry and Egress" and SAO-213, "Containment Entry, Egress and Inspection," contain guidance to verify the presence and size of any gaps that could create a bypass around the sump screens. During a containment walkdown on April 13, 2004, the inspectors noted several issues not previously identified by Entergy. The inspectors identified loose sump deck plate penetration cover plates and missing deck plate anchor bolts. Upon further review, the inspectors questioned the gap between the alignment collars and the pipes penetrating the sump. During a subsequent sump inspection, Entergy concluded that the annular gap between the alignment collars and the pipes all exceeded the design of 1/8". Entergy initiated CRs to address these deficiencies (CR-IP2-2004-01781, 2004-01820, 2004-01948, and 2004-01951). The inspectors verified that corrective actions based on gaps identified in the recirculation sump had been completed. Entergy procedure SAO-213, "Containment Entry, Egress and Inspection," Revision 4, Attachment V, requires personnel to "verify recirculation sump grating and floor in place and pipe collars in place" and to "verify ALL debris removed." Entergy last implemented Attachment V during their containment closeout in August 2003. The inspectors considered this a missed opportunity as Entergy should have identified these deficiencies prior to reactor startup in August 2003. The performance issues and the associated findings were documented in integrated inspection report 05000247/2004006.

4OA6 Meetings, including Exit

.1 Exit Meeting Summary

On January 13, 2005, the inspectors presented the inspection results to Mr. C. Schwarz and other Entergy staff members, who acknowledged the inspection results presented. The inspectors asked Entergy what materials examined during the inspection should be considered proprietary. No proprietary information is presented in this report.

.2 Management Site Visits

On December 15, 2004, Sam Collins, Regional Administrator, visited the Indian Point Energy Center, toured IP2 and IP3 plant areas, and met with senior members of Entergy Nuclear Northeast, Inc.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by Entergy and are violations of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- 10 CFR 50 CFR 50 Appendix B, Criteria III, “Design Controls,” requires, in part, that measures be established to assure that applicable regulatory requirements and design basis are correctly translated into specifications, drawings, procedures and instructions. This criteria also requires that design control measures shall provide for verifying or checking the adequacy of design and that the applicable design basis for SSCs be translated into procedures. Entergy identified a number of cable trays inside containment that have missing or damaged dividers and a divider for tray T-99G was not shown on plant drawing 206872. Entergy documented these deficiencies in their CAP as CR IP2-2004-5497, IP2-2004-5750 and CR IP2-2004-5752. This finding is only of very low safety significance (Green) because past operability of the condition was preserved through verification of free air space between trains of cables.

- 10 CFR 50 CFR 50 Appendix B, Criteria III, “Design Controls,” requires, in part, that measures be established to assure that applicable regulatory requirements and design basis are correctly translated into specifications, drawings, procedures and instructions. This criteria also requires that design control measures shall provide for verifying or checking the adequacy of design and that the applicable design basis for SSCs be translated into procedures. Entergy identified a number of examples whereas cables passed between two different safeguards routing channels. Entergy documented these deficiencies in their CAP as CR IP2- 2004-5505, IP2-2004-5559, IP2-2004-5502, IP2-2004-5565, IP2-2004-5916, and CR IP2-2004-5957. This finding is only of very low safety significance (Green) because of no impact on the worst case postulated electrical fault on unaffected cables and credit for dual electrical isolation to limit the fault current.

ATTACHMENTS: As Stated

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Key Points of Contact

J. Bencivenga	Sr. Engineer, Civil Design Engineering
C. Bergeren	In-Service Testing Engineer
J. Boccio	I&C Superintendent
T. Burns	NEM/Respiratory Protection Supervisor
J. Comiotes	Director, Nuclear Safety Assurance
P. Conroy	Manager, Licensing
F. Dacimo	Site Vice President
G. Dahl	Technical Specialist, Licensing
R. Daley	Systems Engineer
S. D'Auria	Design Engineer
G. Dean	Assistant Operations Manager - Training
R. DeCensi	Technical Support Manager
J. DeRoy	General Manager, Engineering
R. Deschamps	Radiation Protection Coordinator
A. DeVito	Senior Technical Instructor, Nuclear Training
R. Dolansky	In-Service Testing Engineer
P. Donahue	Senior Environmental Specialist
R. Drake	Supervisor, Mechanical Design Engineering
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F. Gumble	Reactor Engineer
J. Herrera	Systems Engineer
G. Hocking	Instruments and Dosimetry Supervisor
M. Imai	Systems Engineer
F. Inzirillo	Emergency Planning Manager
A. Irani	Supervisor, Nuclear Engineering Analysis
T. Jones	Nuclear Safety/Licensing Specialist, Licensing
M. Kahn	Electrical Engineer
R. LaVera	ALARA Supervisor
D. Mayer	Unit 1 Project Manager
B. Meek	System Engineer
R. Milici	Senior Engineer, Electrical Design Engineering
M. Miller	Supervisor, Operations Standards
V. Myers	Systems Engineering Primary Systems Supervisor
K. Naku	Unit 2 Instrumentation and Controls Assistant Superintendent
E. O'Donnell	IP3 Assistant Operations Manager
N. Papaiya	Quality Specialist, Quality Assurance
J. Perrotta	Quality Assurance Manager
J. Peters	Unit 2 Plant Chemist
S. Petrosi	Manager, Design Engineering
F. Phillips	Emergency Preparedness Staff
J. Pineda	Procedure Writer, Operations Standards
P. Platt	Unit 3 Instrumentation and Controls Supervisor
J. Raffaele	Supervisor, Electrical Design Engineering
P. Rubin	Manager, Site Planning and Outage Services

C. Schwarz	General Manager, Plant Operations
I. Sinert	Systems Engineer
A. Stewart	Nuclear Safety/Licensing Specialist, Licensing
R. Sutton	Systems Engineer
A. Unsal	Acting Supervisor, Civil-Structural Design Engineering
J. Ventosa	Site Operations Manager
A. Vitale	Operations Manager, IP3
R. Walpole	Labor Relations Response Coordinator
C. Wend	Radiation Protection Manager
S. Wilke	Fire Protection Engineer
J. Zarella	Programs and Components Engineer
W. Zolotas	Radiation Protection Technician

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000247/2004012-01	NCV	Failure to implement adequate procedures for ECCS operation (Section 1R04)
05000247/2004212-02	NCV	Inadequate maintenance procedure resulting in all EDGs being declared inoperable due to defeating SBO logic (Section 1R13.1)
05000247/2004012-03	NCV	Inadequate PM procedure implementation resulting in a loss of safeguards bus 6A (Section 1R13.2)
05000247/2004012-04	FIN	Inadequate corrective actions associated with SWC pressure switch (Section 1R14.1)
05000247/2004012-05	NCV	Improper installation of RCS loop flow tubing resulting in RCS leakage (Section 1R14.2)
05000247/2004012-06	NCV	Failure to follow RCS drain down procedure due to inappropriate approach (Section 1R20)
05000247/2004012-07	NCV	Multiple deficiencies in surveillance procedures associated with ITS conversion (Section 1R22)
05000247/2004012-08	NCV	Failure to properly address a condition adverse to quality involving leakage from a canopy seal weld onto the RVH in November 2002 (Section 4OA2.1)
05000247/2004012-09	NCV	Failure to provide adequate inspection criteria and guidance to evaluators prior to the inspection of the reactor vessel lower head penetration nozzles (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures

2-COL 10.6.2 Containment Integrity, Rev. 25
2-SOP-1.2, Rev. 40, “Draining Reactor Coolant System”
COL 4.2.1, Rev. 21, “Residual Heat Removal System"
COL 27.1.6, Rev. 20, “Instrument Buses, DC Distribution and PA Inverter”
2 COL 10.0, Rev. 38, “Locked Safeguards Valves”
2 COL 10.1.1, Rev. 24, “Safety Injection System”
OP-0291-1, Rev. 0, “Indian Point 2 SFP Temporary Cooling System Operating Procedure”

Drawings

9321-F-2735 Flow Diagram - Safety Injection System
IP2_SOD_005, Rev. 2, “ECCS System”, 4/29/02

Condition Reports

CR-IP2-2001-02909
CR-IP2-2004-06448
CR-IP3-2000-00121

Section 1R05: Fire Protection

Procedures

SMM-DC-901, “IPEC Fire Protection Program Plan,” Rev. 1
ENN-DC-161, “Transient Combustible Program,” Rev. 1
Transient Combustible Evaluation No. 04-004, dated 10/21/04
Fire Protection Implementation Plan, Pre-Fire Plans
Pre-Fire Plans, PFP-252, Rev. 0, May 2004, “Unit 2 Control Building - Cable Spreading Room, Elev. 33'-0"”
Pre-Fire Plans, PFP-259, Rev. 0, May 2004, “Unit 2 Aux. Feedwater Building, Elev. 18'-0"”
Pre-Fire Plans, PFP-251, Rev. 0, May 2004, “Unit 2 Control Building - 480V Switchgear Room, Elev. 15'-0"”
Pre-Fire Plans, PFP-213, Rev. 0, May 2004, “Unit 2 Electrical Tunnel, Elev. 33'-0" to 68'-0"”

Condition Reports

CR-IP2-2004-05043

Section 1R08: Inservice Inspection Activities

See documents listed under “Section 4OA5.1, 4OA5.2, and 1R08" below.

Section 1R12: Maintenance Effectiveness

QA Surveillance Report QS-2004-IP-17, MOV Controllers at IP2
1st Quarter Residual Heat Removal System Health Report
Maintenance Rule Basis Document, Revision 7

Condition Reports (CR-IP2-)

2004-2912	2004-3932	2004-5784	2004-6125
2004-3389	2004-5269	2004-5820	2004-6448

Section 1R14: Operator Performance During Non-Routine Evolutions

Procedures

E-0	Reactor Trip or Safety Injection, Rev. 45
ES-0.1	Reactor Trip Response, Rev. 42

Section 1R17 (also applies to Section 1R2 of Inspection Report 05000247/2004002)

10 CFR 50.59 Screened-out Evaluations

03-1125-TR-00-RS	Temporary Repair of Leaking Pipe at BFD-93-3
DCP-02-02552-M	Main Steam to Auxiliary Steam Pressure Reducing Station
03-0281-TM-00-RS	Block Alarm Function of –44 Flux Deviation Only
02-0517-MM-00-AD	Electrical Isolation of "FQ" Integrator in loop of Plant Exhaust Monitoring System at Units 1 & 2
02-0277-DE-00-AD	Battery 22 Replacement, Rev. 0
02-0218-MD-00-RS	Core Exit Thermocouple Nozzle Assemblies
02-0437-MD-01-RS	Reactor Vessel Top Dome Insulation Replacement
02-481-MM-00-RS	Narrow Range RTD Padding Resistor Removal and Wiring Replacement
02-415-MM-00-RS	Condensate Flow Control System Upgrade
02-0405-MM-00-RS	Coating Exposed Structural Steel in FSS/CCR Fire Barrier and Refinishing of Pipe Chase
02-0342-MM-00-RS	Station Auxiliary Transformer Load Tap Changer - Safety Injection Modification
01-0897-MM-00-RS	Functional Removal of the BGIS/BGIR Circuit in the Emergency Diesel Generator Control Circuits

Self-Assessments

2003-00081	IPEC Snapshot Self-Assessment Report of the Modification Closeout Process
2003-00285	Focused Self-Assessment Report of the IPEC 10 CFR 50.59 Program
02-AR-15-EN	Implementation Engineering - Design Change Process, April 30, 2002
2003-00356	IP2 Permanent Modifications-
Indian Point Unit 1&2,	Nuclear Quality Assurance Independent Oversight Programs; Engineering - Outage Modifications

Condition Reports (CR-IP2-)

1999-09492	2003-00091	2003-05461	2003-00846
2000-05366	2003-01850	2003-00010	2003-05294
2000-09088	2003-02024	2003-00910	2003-02714
2000-05226	2003-01162	2003-00671	2003-04250
2003-04960	2003-05306	2003-01026	2003-05092

Procedures

SAO-405	Engineering Change Process, Revision 17
ENN-DC-141	Design Inputs, Revision 0
	10 CFR 50.59 Resource Manual, Revision 0
ENN-LI-101	10 CFR 50.59 Review Process (Superceded SAO-460), Revision 4
ENN-DC-103	Design Process, Revision 1
ENN-DC-117	Post Modification Testing and Special Testing Instruction, Revision 2
ENN-DC-102	Operating Plant Changes and Modification, Revision 1
ENN-DC-134	Design Verification, Revision 0
2-AOP-LEAK-1	Sudden Increase in Reactor Coolant System Leakage, Revision 1
AOI 1.7	Excessive Reactor Coolant System Leakage, Revision 16
AOI 27.1.9.2	Providing Appendix R Power from Unit 3, Revision 0
2-POP-3.1	Plant Shutdown, Mode 1 to Mode 3, Revision 43
TOI-271	RCS Depressurization for Reseating of Leaking Pressurizer Safety Relief Valves, Revision 1
AOP-AIR	Air System Malfunctions, Revision 1
GRAPH RCS-2	Pressurizer Level vs. T_{AVG}, Revision 4;
GRAPH RPC-4	RCS T-REF vs. Reactor Power, Revision 4
2-ROP-1.1	Plant Restoration, Mode 5 to Mode 3, Revision 67
2-SOP-1.7	Reactor Coolant System Leakage Surveillance, Revision 37

Drawings

A250907-21	Electrical Distribution and Transmission System
1998MB6707-AA	Pipe Support for Integrated Liquid Waste Continuous Blowdown, Revision 1
DMD 206759-AA	Modification of Pipe Support for Integrated Liquid Waste Continuous Blowdown, Revision 0
DMD 206761-AA	Modification of Pipe Support for Integrated Liquid Waste Continuous Blowdown, Revision 0
DMD 238672-AD	Main Steam to Auxiliary Steam Pressure Reducing Station D/C for Rad Monitor "R45" Modification, Revision 0

Section 1R19: Post-Maintenance Testing

Work Orders

WO # IP2-04-20732
WO # IP2-04-34771

Section 1R20: Refueling and Outage Activities

Procedures

2-SOP-1.2, “Draining Reactor Coolant System”
3-SOP-RP-020, “Draining the RCS/Refueling Cavity”
ENG-375, Rev. 0, “Gravity Flow Test from the RWST to the RCS”
Westinghouse Refueling Manual (FP-IPP-R16)
System operating procedure (SOP) 17.31, “Refueling Operation Surveillance Verification.”
2-REF-003-GEN, Reactor Core Refueling, Revision 0
2-REF-003-GEN, Section 2.3, Reactor Vessel - Debris Inspection, Revision 0
2-REF-003-GEN, Section 3.1, Fuel Movement Requirements - Core Reload
OAD-38, Outage Risk Management, Revision 9
2-AOP-RHR-1, Loss of RHR, revision 3
2-AOP-SF-1, Loss of Spent Fuel Pit Cooling, revision 1
OSP 1.1, Filling and Venting Reactor Coolant System, revision 2
SOP 17.30, Manipulator Crane Operations, Revision 0
OSP 24.1.2, Support Procedure - Service Water Header Operation with the RCS less than 350 F, Revision 1
2-POP-3.1, Plant Shutdown, Mode 1 to Mode 3, Revision 46
2-POP-3.3, Plant Cooldown, Modes 3 to 5,
2-SOP-1.2.1, Draining Reactor Coolant System No Fuel In the Reactor and Vessel Head Less than Fully Tensioned, Revision 13
SOP 4.2.2, Operation with Reduced Reactor Coolant System Inventory, Revision 17
2-SOP-17.31, Refueling Operation Surveillance, Revision 24
2-TOP-001, Unit 2 Reactor Cavity Cleanup, revision 0
2-POP 1.2, Reactor Startup, Mode 3 to Mode 2, Revision 44
2-POP-1.1, Plant Restoration Mode 5 to Mode 3, Revision 71
2-SOP-1.2, Draining Reactor Coolant System, Revision 40
0-SYS-014-GEN, “Scaffolding Construction And Control,” Rev. 2
NRC Information Notice 96-58: “RCP Seal Replacement With Pump on Backseat,” dated 10/30/96
2-OSO-1.3, “Support Procedure - Reactor Coolant Pump Startup and Shutdown,” Rev. 1
2-SOP-27.3.1.1, “21 Emergency Diesel Generator Manual Operation,” Rev. 12
2-ARP-003, “Diesel Generator,” Rev. 0
2-POP-3.3, “Plant Cooldown, Mode 3 to Mode 5,” Rev. 64
2-SOP-4.2.1, “Residual Heat Removal System Operation,” Rev. 58
Dwg. No. A208168-53, “Chemical & Volume Control System”
Dwg. No. 9321-F-2738-113, “Reactor Coolant System”
2-AOP-FH-1, “Fuel Damage or Loss of SFP/Refueling Cavity Level,” Rev. 1
OAP-030, “Infrequently Performed Tests and Evolutions, “ Rev. 0
2-ARP-SGF, “Spent Fuel Pit Level 6",” Rev. 31
2-PT-W019, “Electrical Verification - Offsite Power Sources and AC Distribution,” Rev. 1

Calculations

IP-CALC-04-01731

Condition Reports (CR-IP2-)

2004-05179	2004-05501	2004-05978	2004-06419
2004-05245	2004-05502	2004-05991	2004-06421
2004-05256	2004-05677	2004-06260	2004-06512
2004-05269	2004-05954	2004-06414	2004-06541
2004-05334			

Section 1R23: Temporary Plant Modifications

Engineering Evaluations

ER IP2-03-29397, “Leak Repair for BFD-93-3, “ Rev. 0
MMS-C-010-N Rev. 8, “Temporary repair of leaking 3/4" Pipe at BFD-93-3 (03-1125-TR-00-RS),” dated 11/6/03 [Entergy 50.59 Screen Control Form]
Engineering Order No. 24969C, “Team Inc. Calc. No. 237-11306, ER IP2-03-29397,” dated 11/6/03 [Vendor Document Distribution and Impact Review]

Section 20S1: Access Control to Radiologically Significant Areas

IP2-2004-2178	IP2-2004-3688	IP2-2004-5179	IP3-2004-2933
IP2-2004-2456	IP2-2004-3823	IP3-2004-1712	IP3-2004-3114
IP2-2004-3343	IP2-2004-4601	IP3-2004-1763	IP3-2004-3117
IP2-2004-3676	IP2-2004-4902	IP3-2004-2602	

Section 4OA5.1, 4OA5.2, and 1R08 (TI 2515/152, TI 2515/150, and ISI)

Action Request/Condition Report

IP2-2002-07204	IP2-2002-10277	IP2-2004-03909	IP2-2004-05697*
IP2-2002-07220	IP2-2002-10669	IP2-2004-03932	IP2-2004-05751*
IP2-2002-07972	IP2-2003-01466	IP2-2004-04524	IP2-2004-05756*
IP2-2002-08241	IP2-2003-02159	IP2-2003-04703	IP2-2004-05783
IP2-2002-08273	IP2-2003-02161	IP2-2004-05226	IP2-2004-05697
IP2-2002-08276	IP2-2003-04288	IP2-2004-05263*	IP2-2004-05709*
IP2-2002-08794	IP2-2003-04703	IP2-2004-05658	IP2-2004-05169
IP2-2002-08889	IP2-2003-06598	IP2-2002-10790	IP2-2004-05263
IP2-2002-09101	IP2-2003-06690	IP2-2002-10986	IP2-2004-05803*
IP2-2002-09571	IP2-2003-07127	IP2-2002-11305	IP2-2004-05892*
IP2-2002-09573	IP2-2003-07519	IP2-2003-00327	IP2-2004-06050*
IP2-2002-09606	IP2-2004-01251	IP2-2003-01202	IP2-2004-06119*
IP2-2002-10021	IP2-2004-03083	IP2-2004-05664*	
IP2-2002-10184	IP2-2004-03902*	IP2-2004-05674*	

* Indicates this was generated as a result of this inspection.

Procedures

ENN-DC-319, Revision 1; Boric Acid Corrosion Control Program
ENN-NDE-10.02, Revision 0, 12/2/03; VT-2 Examination
JTS-2004-BMI, Revision 0, 8/23/04; Reactor Vessel BMI Visual Inspection for Indian Point 2
2-SOP-1.7, Revision 39, 9/22/04; Reactor Coolant System Leakage Surveillance
IP-SMM-MA-121, Revision 0, 9/1/044; Fluid Leak Management Program
ENN-NDE-9.23, Revision 0, 8/12/03; Ultrasonic Examination of Austenitic Pipings Welds (ASME Section XI)
ENN-NDE-10.01, Revision 0, 12/2/03; VT-1 Examination
ENN-NDE-10.03, Revision 0, 12/3/03; VT-3 Examination
ENN-NDE-9.31, Revision 0; ??/??/??; Magnetic Particle Examination (MT) for ASME Section XI
2-PT-D001, Revision 5, 9/29/04, Control Room Operations Surveillance Requirements
JTS-2004-CRDM, Revision 0, 9/9/04, Reactor Vessel CRDM Visual Inspection for Indian Point 2
JTS-2004-CRDM, Revision 1, 11/3/04, Reactor Vessel CRDM Visual Inspection for Indian Point 2
EN-LI-102, Revision 1; 11/11/04; Corrective Action Process
2-PTR-156, Revision 0; 3/25/04; RCS Boric Acid Leakage and Corrosion Inspection
EN-WM-100, Revision 0, 6/1/04; Work Request (WR) Generation, Screening, and Classification
ER-IP2-03-14929; 10/8/04; RV Bottom Head Insulation Modification
IP-SMM-WM-100, Revision 2; 6/3/04; Work Control Process
SOP-RCS-005, Revision 18, Reactor Coolant Leakage Evaluation
ER IP2-04-31511, 11/6/04, Installation of Canopy Seal Clamp Assembly on Reactor Vessel Head Penetration Nozzles

NDE Examination Reports

RV Head Lifting Rig UT Examination, 10/29/01
- Sling Leg Connecting Pins (upper and lower)
- Head Lifting Lug (3) Connecting Pins
- Platform Legs Clevis Pins and welds
- Head Lifting Lug - 3 locations
- Platform Leg Upper Clevis Block - 3 locations
- Upper Clevis Weld to Top Cover Plate - 3 Locations
- Platform Leg Brace & Weldments - 3 locations

00-VT014; CVC valve bolts, 5/3/00; Rejected then accepted
00-VT-031:RHR Valve Internals; 5/3/00; Rejected then accepted
04-VT007: Pipe Hanger Threaded Rod Engagement; 10/27/04; Rejected then accepted
04-VT006: Pipe Hanger Threaded Rod Engagement; 10/27/04; Rejected then accepted
04-VT111; RCP 23 Circumferential Weld, 11/11/04
04-VT110; RCP 23 Support Bracket Welds, 11/11/04
04-VT033; FW Line Pipe Support Visual, 11/8/04
Radiographic Inspection Report, IP2-CVCS-215-029, 11/10/04
Radiographic Inspection Report, IP2-CVCS-217-021, 11/10/04
04-UT091; MS Circ Weld, C-F-2/C5.51,11/12/04
04-UT086; MS Circ Weld, C-F-2/C5.51,11/12/04
04-UT049; RV Threads In Flange, 11/9/04
04-UT024; RV Threads In Flange, 11/9/04

04-UT025; RV Threads In Flange, 11/9/04
04-UT026; RV Threads In Flange, 11/9/04
04-UT027; RV Threads In Flange, 11/9/04
04-UT028; RV Threads In Flange, 11/9/04
04-UT029; RV Threads In Flange, 11/9/04
04-UT030; RV Threads In Flange, 11/9/04
04-UT031; RV Threads In Flange, 11/9/04
04-UT032; RV Threads In Flange, 11/9/04
04-UT033; RV Threads In Flange, 11/9/04
04-UT034; RV Threads In Flange, 11/9/04
04-UT035; RV Threads In Flange, 11/9/04
04-UT036; RV Threads In Flange, 11/9/04
04-UT037; RV Threads In Flange, 11/9/04
04-UT038; RV Threads In Flange, 11/9/04
04-UT039; RV Threads In Flange, 11/9/04
04-UT040; RV Threads In Flange, 11/9/04
04-UT041; RV Threads In Flange, 11/9/04
04-UT042; RV Threads In Flange, 11/9/04
04-UT043; RV Threads In Flange, 11/9/04
04-UT044; RV Threads In Flange, 11/9/04
04-UT045; RV Threads In Flange, 11/9/04
04-UT046; RV Threads In Flange, 11/9/04
04-UT047; RV Threads In Flange, 11/9/04
04-UT050; RV Threads In Flange, 11/9/04
04-UT084; RV Upper Head Meridional Weld 5-6 @ 300, RVHM6, 11/10/04
04-UT097; RV Upper Head Meridional Weld 23-24 @ 060, RVHM2, 11/10/04
Cal. Data Sheet #IP2R16-001, RVHC2, Head To Flange Weld, From #43 stud to #1 stud, and stud #36 to stud #43
04-CA005; Cal. Report, 11/11/04
04-CA004; UT Cal. Report, RV Flange Threads, 11/9/04
Data Sheet #IP2R16-003, 11/11/04, Steam Generator Tubesheet Weld SGC 21R-6
Data Sheet #IP2R16-002, 11/11/04, Steam Generator Tubesheet Weld SGC21R-7
04-MT004; FW Line MT Data Sheet, 11/9/04

Repair-Replacement Work Order

W.O. IP2-03-23245, Revision 1, 11/3/04; Bare Metal Visual (BMV) Examination of Top Head Exterior Surface, and CRDM Penetration
Work Order #IP2-03-23245, Bare Metal Visual Examination

Drawings/Isometrics

Drawing 35R05016, IP2 Carbon Steel Inspection Map

Calculations

Calculation FMX-00323, Revision 1, 9/15/03; "Indian Point Unit 2 Condition Monitoring and Operational Assessment RFO-15 SG-SGDA-02-45 September 2003 Westinghouse Electric Company"
Entergy Calculation FCX-00538-00, Revision 0; "Estimation of Effective Degradation Years (EDY's) for IP2 Reactor Vessel Head"
Calculation Number: IP3 - CALC - RV - 04038

Miscellaneous Documents

83-0051, Revision 1; 11/1/02; Brooks: Reactor Vessel Head Remote Visual Inspection
PWR Reactor Pressure Vessel Upper Head Penetrations Inspection Plan; MRP-75, Rev. 1
PO-IP2-03-23245, Visual Examination for Leakage of RPV Head and CRDM Penetrations Examination Acceptance Criteria
IP2 - RPV Head Integrity: Visual Examination, 2R16 Pre-job Briefing
Entergy letter to NRC dated April 2, 2002, "Submittal of 15-day Response to NRC Bulletin 2002-01"
Entergy letter to NRC dated May 15, 2002, "Submittal of 60-Day Response to NRC Bulletin 2002-01"
Entergy letter to NRC dated September 11, 2002, "30-Day Response to NRC Bulletin 2002-02, RPV and Vessel Head Penetration Nozzle Inspection Programs"
Entergy letter to NRC dated May 19, 2004, "NRC First Revised Order EA-03-009 Relaxation Requests for Inspection of Reactor Pressure Vessel Heads"
Jamko Technical Solutions, Inc. letter to Entergy dated October 21, 2004, "Rivit & Thor Qualification"
IP2 Contractor Personnel Qualification Document, IP-SMM SEC-106, Rev. 1, Out Processing Checkout Sheet
Entergy Nuclear Northeast VT 1-3 Level III Certification, ASNT, Inc. NDT Level III Certification for Reed, S.
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Entergy letter to NRC dated July 23, 2002, NRC Bulletin 2001-01 - Reactor Pressure Vessel Head Penetration Nozzle Inspection Plan for 2002 Refueling Outage (2R15)
Entergy letter to NRC dated November 18, 2003, 90-Day Response to NRC Bulletin 2003-02 Regarding Leakage From Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity
Entergy letter to NRC dated July 26, 2004, Indian Point Nuclear Generating Units No. 2 and No.3, Docket No. 50-247, and 50-286, Response to NRC Bulletin 2004-01 Regarding Inspection of Alloy 82/182/600 Materials Used In Pressurizer Penetrations and Steam Space Piping Connections
Entergy letter to NRC dated August 23, 2004, Reply to RAI regarding Relaxation Requests for Inspection of Reactor Pressure Vessel Head per NRC First Revised Order, EA-03-009
NRC Letter, Dated October 15, 2004, Relaxation Of First Revised Order On Reactor Vessel Nozzles, Indian Point Nuclear Generating Unit No. 2 (TAC NO. MC3194)

Section 4OA5.3 Reactor Containment Sump Blockage (NRC Bulletin 2003-01)

Procedures

E-1, Loss of Reactor or Secondary Coolant
ECA1.1, Loss of Emergency Coolant Recirculation
SOP-SI-002, Refilling the Refueling Water Storage Tank

Miscellaneous

LRQ-EOP-07, EOP ECA-1.1 and ECA-1.2
Simulator Training Guide EOP034E1, RCS/PZR Review and EOP's E-1/ES-1.3/ES-1.4

LIST OF ACRONYMS

AFW	auxiliary feedwater
ALARA	As Low As is Reasonably Achievable
AOP	abnormal operating procedure
ASME	American Society of Mechanical Engineers
BGIS/BGIR	BGIS Switch and BGIR Relay
CAP	corrective action program
CARB	Corrective Actions Review Board
CCR	central control room
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulation
COL	check-off list
CR	condition report
CRS	Control Room Supervisor
CTS	Custom Technical Specifications
EAL	emergency action level
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOP	Emergency Operating Procedure
ERO	Emergency Response Organization
FSAR	Final Safety Analysis Report
GPM	gallons per minute
HRA	high radiation area
HVAC	heating, ventilating and air conditioning
IMC	Inspection Manual Chapter
IP2	Indian Point 2
IP3	Indian Point 3
IPEC	Indian Point Energy Center
IPE	Individual Plant Evaluation
IPEEE	Individual Plant Evaluation for External Events
ISI	in-service inspection
ITS	Improved Technical Specifications
KV	kilo volts
LHP	lower head penetration

LOCA	loss of coolant accident
MR	maintenance rule
NCV	Non-cited Violation
NDE	non-destructive examination
NEI	Nuclear Energy Institute
NPO	nuclear plant operator
NRC	Nuclear Regulatory Commission
OSP	outage safety plan
PAB	primary auxiliary building
PFP	Pre-Fire Plan
PI	performance indicator
PI&R	problem identification and resolution
PM	preventative maintenance
PWT	post-work test
RCA	radiological controlled area
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RP	radiation protection
RPV	reactor pressure vessel
RTD	resistance temperature detector
RVH	reactor vessel head
RWP	radiation work permit
RWST	refueling water storage tank
SBO	station blackout
SDP	significance determination process
SFP	spent fuel pool
SG	steam generator
SI	safety injection
SOP	system operating procedure
SR	surveillance requirement
SRA	Senior Reactor Analyst
SSC	structure, system, and component
SWC	stator water cooling
TI	temporary instruction
TRM	technical requirements manual
TS	Technical Specification
UT	ultrasonic testing
VT	visual examination
WO	work order
WOG	Westinghouse Owners Group
2RF15	IP2 refueling outage No. 15
2RF16	IP2 refueling outage No. 16

ATTACHMENT 2

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 initially performed using an inadequate procedure as described in the attached inspection report. Entergy subsequently revised the procedure and provided clear standards and acceptance criteria for the inspection observations.

a.3. The examination was adequate to identify, resolve, and disposition deficiencies.

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

b. The reactor vessel lower head contained dirt, debris, insulation, dried boron residue, and oxidation and which potentially impacted the ability to observe early indication of leakage. No active boric acid deposits or leaks were identified at the interface between the vessel and the lower reactor head penetrations.

c. The inspection was conducted by direct visual inspection of video camera data by examination personnel. The inspection effort achieved examination for 360 degrees around the circumference of all nozzles.

d. If present, small, active boric acid deposits representing reactor coolant leakage, as described in Bulletin 2003-02, could be identified and characterized.

e. No material deficiencies were identified. No indications were identified at the time this inspection was performed.

f. This was the first inspection performed at Indian Point Unit 2 subsequent to issuance of Bulletin 2003-02, and only limited access to the lower head was provided. Entergy removed small cover plates from each penetration nozzle which provided access to view the bottom mounted instrumentation annulus area, but did not provide access to view the general condition of the lower reactor vessel surface.

g. No active boric acid leak deposits were noted on the lower vessel head. Some stains, attributed to prior reactor cavity seal leakage, were observed at multiple locations. These indications were evaluated by Entergy as not pressure boundary leakage.

h. Several locations were sampled for dried boron deposits and chemically analyzed. All deposits were discovered to be greater than two years old and evaluated as being from prior reactor cavity seal leakage.

i. An attempt was made to clean selected penetrations intersecting the lower vessel head. The cleaning was not effective and residue remained on multiple penetrations on the lower head.

j. Entergy noted the above conditions and recorded these indications on the RVH data sheets. Additionally, Entergy retained the video records from this visual inspection. The condition of the lower vessel head away from the immediate area (approximately 1-2 inch diameter around each penetration) was not inspected due to the presence of insulation panels.

ATTACHMENT 3

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

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 using adequate procedures. The procedure specified the extent of the inspection required, provided detailed documentation requirements and provided clear inspection standards and acceptance criteria on which personnel were trained.

a.3. The examination was adequate to identify, resolve, and disposition deficiencies.

a.4. The examination performed was capable of identifying pressure boundary leakage as described in the bulletin.

b. The RVH and head penetration had some debris, insulation residue and oxidation which made it difficult to view the penetration intersection area when using the magnified viewing camera. Entergy provided photographic evidence that these materials did not appear significant when viewed using normal magnification. No active boric acid deposits were identified at the interface between the vessel and the penetrations.

c. The inspection was conducted by direct visual inspection by examination personnel and by the use of a video camera. The inspection used either a remote controlled video camera crawler or a manually controlled video camera. The inspection effort achieved examination for 360 degrees around the circumference of all RVH penetrations in their entirety.

d. No material deficiencies were identified which required repair.

e. These inspections were initially impeded by tight clearance between the upper head and the insulation package. Entergy subsequently modified their inspection technique to gain adequate access to all penetrations.

f. The temperatures used in the susceptibility ranking calculation were plant specific measured values.

g. During this outage only a VT of the RVH and head penetrations was conducted.

h. Plant procedures did exist to detect leakage from pressure-retaining components above the RVH. Boris acid deposits were discovered on the RVH and removed. No damage was observed to the RVH.

i. During 2RF16, Entergy did perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components above the RPV head including additional inspections and chemical analysis of discovered boric acid deposits.