August 11, 2006

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. 05000247/2006003

Dear Mr. Dacimo:

On June 30, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Unit 2. The enclosed integrated inspection report documents the inspection results, which were discussed on June 28, 2006, with you and members of your staff.

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

Based on the results of this inspection, 11 findings of very low safety significance (Green) were identified. Nine of these findings were also determined to be violations of NRC requirements. However, because of their very low safety significance, and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV 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 Senior Resident Inspector at Indian Point Unit 2.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the

NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Eugene W. Cobey, Chief
Reactor Projects Branch 2
Division of Reactor Projects

Docket No. 50-247
License No. DPR-26

Enclosure: Inspection Report No. 05000247/2006003
w/Attachments: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	05000247/2006003
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Unit 2
Location:	295 Broadway, Suite 3 Buchanan, NY 10511-0308
Dates:	April 1, 2006 through June 30, 2006
Inspectors:	M. Cox, Senior Resident Inspector, Indian Point 2 G. Bowman, Resident Inspector, Indian Point 2 T. Hipschman, Senior Resident Inspector, Indian Point 3 B. Wittick, Resident Inspector, Indian Point 3 F. Arner, Senior Reactor Inspector, Region I S. Barr, Senior Emergency Preparedness Specialist, Region I G. Dentel, Senior Resident Inspector, Seabrook D. Jackson, Senior Project Engineer, Region I R. Codell, Senior Hydraulic Engineer J. Kottan, Senior Health Physics Inspector, Region I T. Nicholson, Senior Technical Advisor on Radionuclide Transport J. Noggle, Senior Health Physics Inspector, Region I L. Scholl, Senior Reactor Inspector, Region I M. Snell, Reactor Inspector, Region I D. Dempsey, Resident Inspector, FitzPatrick T. Burns, Reactor Inspector, Region I J. Josey, Reactor Inspector, Region I J. Lilliendahl, Reactor Inspector, Region I T. Setzer, Project Engineer, Region I
Other:	J. Williams, Groundwater Specialist, U.S. Geological Survery
Approved by:	Eugene W. Cobey, Chief Reactor Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000247/2006003; 04/01/2006 - 6/30/2006; Indian Point Nuclear Generating Unit 2; Equipment Alignment, Fire Protection, Maintenance Risk Assessment and Emergent Work Control, Operability Evaluations, Post-Maintenance Testing, Refueling and Outage Activities, Surveillance Testing, Access Control to Radiologically Significant Areas, Event Followup.

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

A. NRC Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. The inspectors identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Entergy's procedures failed to ensure that the 22 residual heat removal (RHR) pump suction pressure gauge was placed in service prior to starting the system in the shutdown cooling mode of operation. This gauge, which is used to identify degrading RHR pump performance when in shutdown cooling, was left isolated after the plant was depressurized. Entergy placed the pressure gauge in service and entered the issue into the corrective action program.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. The inspectors evaluated the significance of this finding using IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs [Pressurized Water Reactors] and BWRs [Boiling Water Reactors] and determined that this finding was of very low safety significance because the finding did not degrade the equipment, instrumentation, training or procedures needed for any shutdown safety function. The inspectors also determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that the procedure for placing the RHR system in the shutdown cooling mode of operation was complete and accurate. (Section 1R04)

Green. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because plant procedures for reactor coolant system venting following depressurization were inadequate. This resulted in the formation of an 850 gallon void in the reactor vessel head while the plant was shutdown and depressurized. Entergy entered this issue into the corrective action program for evaluation.

The inspectors determined that this finding, which was associated with the Initiating Events cornerstone, was more than minor because if it was left uncorrected, it would have become a more significant safety concern. The inspectors evaluated the significance of this finding using IMC 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs,” Checklist 3, and determined that a Phase 2 analysis was needed. The Region I Senior Reactor Analyst performed the Phase 2 analysis using IMC 0609, Appendix G, Attachment 2, “Phase 2 Significance Determination Process Template for PWR During Shutdown,” and determined that the finding was of very low safety significance based upon the availability of mitigating systems and the low initiating event (loss of inventory) likelihood. The inspectors also determined that this finding had a cross-cutting aspect in the area of human performance because Entergy’s shutdown procedures were not complete and accurate, in that, they failed to ensure the reactor vessel head was adequately vented. (Section 1R20)

Green. A Green self-revealing finding was identified because Entergy’s procedure for placing the standby main lube oil cooler in service was inadequate. A deficiency in the procedure resulted in a loss of main feedwater, an automatic start of the motor-driven auxiliary feedwater pumps, and a steam generator level transient. This issue was entered into the corrective action program, and the procedural deficiencies were resolved.

The inspectors determined that this finding was associated with the Initiating Events cornerstone; and, it was more than minor because it was similar to IMC 0612, Appendix E, “Examples of Minor Issues,” Example 4.b, since the inadequacies in Entergy’s procedure caused a plant transient. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” and determined that the finding was of very low safety significance because it did not contribute to the likelihood of both a reactor trip and the likelihood that mitigation equipment or functions would be unavailable. The inspectors also determined that the finding had a cross-cutting aspect in the area of human performance because Entergy’s procedures were not complete and accurate, in that, they failed to ensure the standby main lube cooler was properly filled and vented prior to being placed in service. (Section 4OA3)

Cornerstone: Mitigating Systems

Green. The inspectors identified a Green NCV of license condition 2.K. because Entergy failed to identify a condition adverse to fire protection related to a degraded fire door between the 21 and 22 RHR pump cells. A similar condition with the same door had been previously identified by the NRC in January 2006. Entergy took actions to correct the degraded fire door and entered the issue into the corrective action program.

The inspectors determined that this finding was more than minor because it was associated with the Protection Against External Factors attribute of the Mitigating

Systems cornerstone; and, it affected the cornerstone objective of ensuring the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using IMC 0609 Appendix F, "Fire Protection Significance Determination Process," and determined that the finding was of very low safety significance because the fire door, which was moderately degraded, provided a minimum of 20 minutes of fire endurance protection; and, the ignition sources and combustible materials in the RHR pump cells were situated in a manner that the degraded fire door would not have been subject to direct flame impingement. The inspectors also determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because operators who routinely traverse through the degraded fire door during performance of their rounds had not identified the degraded condition of the door. (Section 1R05)

Green. A Green self-revealing finding was identified because Entergy failed to take adequate corrective actions for a degraded service water pipe in the primary auxiliary building. Degradation of this pipe was identified in 2003, but was not adequately evaluated or repaired. Consequently, in April of 2006, the continued corrosion of this pipe led to a through-wall leak and, if not corrected, would have challenged the operability of the RHR pumps. Entergy implemented compensatory measures to protect the RHR pumps, repaired the degraded pipe, and entered the issue into the corrective action program.

The inspectors determined that this finding, which was associated with the Mitigating Systems cornerstone, was more than minor because if it was left uncorrected it would have become a more significant safety concern. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding was of very low safety significance because it represented a qualification deficiency that was confirmed not to result in the loss of operability per Part 9900 Technical Guidance, "Operability Determination Process for Operability and Functional Assessment." The inspectors also determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not implement timely and effective corrective actions for degraded service water piping in the primary auxiliary building. (Section 1R13)

Green. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because plant procedures were not followed during the installation of compensatory measures to restore operability of the RHR pumps following the identification of service water piping degradation in the primary auxiliary building. The inspectors also identified multiple deficiencies with the installation and implementation of the compensatory measures. In response, Entergy corrected the deficiencies associated with the compensatory measures and entered the issue into the corrective action program.

The inspectors determined that this finding, which was associated with the Mitigating Systems cornerstone, was more than minor because it was similar to IMC 0612, Appendix E, "Examples of Minor Issues," Example 3.a, in that, the deficiencies identified with Entergy's compensatory measures required significant rework to ensure RHR pump operability. The inspectors evaluated the significance of this finding using IMC 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs,” Checklist 2, and determined that the finding was of very low significance because the finding did not degrade the equipment, instrumentation, training, or procedures needed for any shutdown safety function. The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not follow plant procedures when implementing a temporary alteration required for the operability of safety-related equipment. (Section 1R15)

Green. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” because Entergy’s post-maintenance test on the 21 emergency diesel generator (EDG) following a governor replacement in November 2004 was not adequate to ensure it could perform its intended design function. Subsequent testing showed the EDG could not attain its rated load of 2300 kilowatts. Entergy corrected the deficiency with the 21 EDG, performed a post-maintenance test including a run at 2300 kilowatts, and entered the issue into the corrective action program.

The inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, “Significant Determination of Reactor Inspection Findings for At-Power Situations,” and determined that this finding was of very low safety significance because it was not a qualification deficiency; it did not represent a loss of safety function for a train or system as defined in the plant specific risk-informed inspection notebook; and it was not risk significant due to external event initiators. (Section 1R19)

Green. The inspectors identified a Green NCV of 10 CFR Part 50.65(a)(4) because Entergy did not assess the risk associated with maintenance on the discharge containment isolation valve from the 21 containment spray pump, SI-869A. This maintenance resulted in the unavailability of the 21 containment spray train for a period of approximately 90 minutes. Entergy entered this issue into the corrective action program, conducted an extent of condition review, and completed a causal analysis.

The inspectors determined that this finding, which was associated with the Mitigating Systems cornerstone, was more than minor because it was similar to Example 7.e in IMC 0612, Appendix E, “Examples of Minor Issues,” in that, the licensee’s risk assessment failed to consider maintenance activities on components that prevent containment failure. The inspectors evaluated the significance of this finding using IMC 0609, Appendix K, “Maintenance Risk Assessment and Risk Management Significance Determination Process,” Flowchart 1, and determined that the finding was of very low safety significance because the calculated risk deficit was not greater than 1×10^{-6}. The inspectors also determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not appropriately incorporate risk insights into planning work activities on SI-869A in accordance with 10 CFR Part 50.65(a)(4) and the Site Management Manual IP-SMM-WM-101, "Online Risk Assessment." (Section 1R22)

Green. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because plant surveillance procedure 2-PT-R84B, "22 EDG 8 Hour Load Run," was not adequate to ensure testing at the appropriate power factor limit prescribed by Technical Specifications Surveillance Requirement 3.8.1.10. Entergy entered this issue into the corrective action program and completed an evaluation to assess the operability of all three EDGs.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, " Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was of very low safety significance because it was not a qualification deficiency; it did not result in the loss of a system or train safety function; and it did not screen as potentially risk-significant due to external events. The inspectors also determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that procedure 2-PT-R84B, "22 EDG 8 Hour Load Run," was complete and accurate. (Section 1R22)

Cornerstone: Occupational Radiation Safety

Green. A Green self-revealing NCV of Technical Specification 5.4.1 was identified because Entergy failed to follow procedural requirements during the core support barrel installation activity. As a result, dose rates were significantly higher than expected during the work activity, and a worker received an unplanned and unintended radiation exposure. Entergy entered this issue into the corrective action program and completed a root cause analysis.

The inspectors determined that this finding was more than minor because it was associated with the Program and Process attribute of the Occupational Radiation Safety cornerstone; and, it affected the cornerstone objective of ensuring adequate protection of workers from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The inspectors evaluated the significance of this finding using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," and determined that the finding was of very low safety significance because it did not involve: (1) as low as reasonable achievable planning or work controls; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. The inspectors also determined that the finding had a cross-cutting aspect in the area of human performance because Entergy personnel failed to comply with plant procedures that were required and specified to support reinstallation of the core support barrel. (Section 2OS1)

Green A Green self-revealing NCV of 10 CFR Part 20.1501, "General," was identified because Entergy failed to take adequate radiation surveys during the installation of the core support barrel. As a result, Entergy did not recognize that actual radiological conditions were significantly different than expected, which contributed to unplanned and unintended exposure of a worker. Entergy entered this issue into the corrective action program and completed a root cause analysis.

The inspectors determined that this finding was more than minor because it was associated with the Program and Process attribute of the Occupational Radiation Safety cornerstone; and, it affected the cornerstone objective of ensuring adequate protection of workers from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. The inspector evaluated the significance of this finding using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," and determined that this finding was of very low safety significance because it did not involve: (1) as low as reasonable achievable planning or work controls; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. (Section 2OS1)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Indian Point Unit 2 began the inspection period at 100 percent power and remained at or near 100 percent power until operators commenced a planned reactor shutdown and plant cooldown on April 19, 2006, to begin the 17th refueling outage (2R17). The plant was returned to 100 percent power on May 25, 2006, and operated at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample of system-related weather preparations)

a. Inspection Scope

The inspectors reviewed Entergy's administrative controls and implementation of a maintenance program to ensure adequate protection of the auxiliary boiler feedwater pump building during warm weather conditions. This system was selected because its risk-significant functions could be affected by adverse weather. The inspectors also reviewed work orders, condition reports, and risk assessments associated with preparation for warm weather conditions. The inspectors evaluated Entergy's actions against the requirements of Technical Specification 5.4.1 and Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix B. The documents reviewed are listed in the Attachment. The review of this system represents one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q - 3 samples / 71111.04S - 1 sample)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors performed three partial system walkdowns to verify the operability of redundant or diverse trains and components during periods of system train unavailability or following periods of maintenance. The inspectors referenced the system procedures, the Updated Final Safety Analysis Report (UFSAR), and system drawings in order to verify that the alignment of the available train was proper to support its required safety functions. The inspectors also reviewed applicable condition reports and work orders to ensure that Entergy had identified and properly addressed equipment discrepancies that could potentially impair the capability of the available train, as required by

10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action.” The documents reviewed are listed in the Attachment. The inspectors performed a partial walkdown on the following systems which represents three samples:

- Emergency diesel generator (EDG) service water supply following maintenance;
- Auxiliary boiler feedwater system following full-flow testing; and
- Temporary spent fuel pool cooling system following installation.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Inspection

a. Inspection Scope

The inspectors performed a complete system alignment inspection on the residual heat removal (RHR) system during the Unit 2 refueling outage to determine whether the system was aligned and capable of providing decay heat removal in accordance with design basis requirements. The inspectors also completed a review of the RHR system alignment following the refueling outage to confirm it was capable of providing reactor vessel inventory makeup, as required by Technical Specifications. The inspectors reviewed operating procedures, surveillance test results, piping and instrumentation drawings, equipment lineup check-off lists, and the Updated Final Safety Analysis Report (UFSAR) to determine if the system was aligned to perform its safety functions. The inspectors reviewed a sample of condition reports and work orders written for deficiencies associated with the RHR system to ensure that they had been evaluated and resolved consistent with the requirements of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action.” The documents reviewed are listed in the Attachment. The walkdown of the RHR system represents one semi-annual sample.

b. Findings

Introduction: The inspectors identified a Green non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because Entergy’s procedures failed to ensure that the 22 RHR pump suction pressure gauge was placed in service prior to starting the system in the shutdown cooling mode of operation.

Description: During a walkdown of the control room following plant shutdown for refueling outage 2R17, the inspectors identified that the suction pressure gauge for the 22 RHR pump was indicating abnormally high. At the time, the plant was depressurized and both RHR pumps were running in parallel to remove decay heat. Entergy determined that the suction pressure gauge for the 22 RHR pump had been left isolated following plant depressurization. The suction pressure gauges for the RHR pumps are low range pressure gauges that are normally isolated until the plant is depressurized to prevent over-ranging. The inspectors reviewed procedures 2-SOP-1.2, “Draining

Reactor Coolant System," and 2-SOP-4.2.1, "Residual Heat Removal Operation." The inspectors determined that Attachment 10 to 2-SOP-1.2, which is used to line up the reactor coolant system for depressurized operation, contains steps to unisolate both RHR suction pressure gauges once the reactor coolant system has been depressurized. However, these steps are optional and are completed at the discretion of plant operators based on running pump alignment. At the time shutdown cooling was initiated, only the 21 RHR pump was running, so operators made the decision to leave the 22 RHR suction gauge isolated. Subsequently, the 22 RHR pump was started, but at the time, operators did not place its suction pressure gauge in service. The inspectors reviewed plant operating procedures and determined that there was no other procedural direction provided to operators to unisolate this gauge when the 22 RHR pump was subsequently started for shutdown cooling.

The inspectors reviewed the licensee's response to Generic Letter 88-17, "Loss of Decay Heat Removal," which discusses recommended actions to reduce the likelihood of loss of decay heat removal capability while shutdown and in a reduced inventory condition. Generic Letter 88-17 recommended, in part, the use of enhanced monitoring for RHR pump degradation, such as the use of RHR pump motor current or pump noise and vibration monitoring. The licensee's response stated that motor current, noise, and vibration monitoring would not be used, because "suction pressure is far more sensitive than pump current, vibration, or pump noise." The RHR pump suction pressure gauges are relied on by operators to identify degrading RHR pump performance when in shutdown cooling. Failure to place these gauges in service would make identification of an imminent loss of shutdown cooling more difficult.

<u>Analysis</u>: The inspectors determined that Entergy's failure to develop an adequate procedure for placing the RHR pump suction pressure gauges in service on initiation of shutdown cooling was a performance deficiency which was reasonably within Entergy's ability to foresee and prevent. 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 procedures.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Specifically, the inadequate procedure for placing the RHR system in the shutdown cooling mode of operation increased the likelihood of the loss of RHR initiating event. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs [Pressurized Water Reactors] and BWRs [Boiling Water Reactors]." The inspectors determined that Checklist 2 was applicable because the unit was in cold shutdown with the reactor coolant system closed and filled with inventory in the pressurizer and time to core boil less than 2 hours. The inspectors determined that this finding was of very low safety significance because the finding did not degrade the equipment, instrumentation, training or procedures needed for any shutdown safety function.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that the procedure for placing the RHR system in the shutdown cooling mode of operation was complete and accurate.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances. Contrary to the above, on April 19, 2006, Entergy did not provide a procedure that was appropriate to the circumstances for placing the RHR system in the shutdown cooling mode of operation. Specifically, the procedure did not ensure that the RHR pump suction gauges were in service when the shutdown cooling mode of operation was initiated. Because this issue is of very low safety significance and is entered into Entergy's corrective action program (CR-IP2-2006-01984), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000247/2006003-01, Inadequate Procedure for Placing RHR Pump Suction Pressure Gauges in Service.

In response to the inspectors' observation, Entergy unisolated the pressure gauge, entered the issue into the corrective action program (CAP), and implemented actions to upgrade their procedures.

1R05 Fire Protection (71111.05Q - 6 samples)

a. Inspection Scope

The inspectors toured areas that were identified as important to plant safety or contained risk-significant components. The inspectors consulted the Indian Point Unit 2 Individual Plant Examination for External Events, 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." The objective of this inspection was to determine if Entergy had adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, and had adequately established compensatory measures for degraded fire protection equipment. The inspectors evaluated the observed conditions against requirements of the Technical Requirements Manual and Entergy's fire protection plan. The inspectors evaluated conditions related to: (1) control of transient combustibles and ignition sources; (2) the material condition, operational status, and operational lineup of fire protection systems, equipment, and features; (3) the fire barriers used to prevent fire damage or fire propagation; and (4) compensatory measures for out-of-service, degraded, or inoperable fire protection equipment. The documents reviewed are listed in the Attachment.

The following six fire areas were reviewed and represent six inspection samples:

- Fire zones 70A, 71A, 72A, 75A, 76A, 77A, 78A;
- Fire zones 80A, 81A, 82A, 83A, 85A, 87A;
- Fire zone 86A;
- Fire zones 3 and 4;
- Fire zone 10; and
- Fire zone 59A.

b. Findings

Introduction: The inspectors identified a Green NCV of license condition 2.K. because Entergy failed to identify a condition adverse to fire protection related to a degraded fire door between the 21 and 22 RHR pump cells. A similar condition with the same door had been previously identified by the NRC in January 2006.

Description: On May 11, 2006, the inspectors performed a fire protection walkdown of the 21 and 22 RHR pump cells. The inspectors found that the three-hour rated swing-type fire door which separates these two fire zones would not latch properly, and that there was a gap along the lower section of the door. This door is installed between the RHR pump cells to prevent flames and hot gases from passing from one pump cell to the other. Degradation of this door could allow the propagation of a fire to impact both safety-related pumps. The inspectors had previously identified a gap between the door and its frame in January 2006, and issued NCV 05000247/2006002-01, "Degraded Residual Heat Removal Pump Cell Fire Door," because Entergy operators had failed to identify this condition during routine tours through the pump cells. In January, Entergy corrected the deficiencies with the door itself, but actions to increase operator awareness of deficient conditions were not effective in ensuring that future degradation would be promptly identified and corrected. As a result, the May 2006 condition was not identified by operators.

The inspectors informed plant operators of the issue. They declared the fire door inoperable and took appropriate compensatory measures in accordance with SAO-703, "Fire Protection Impairment Criteria and Surveillance." Indian Point maintenance staff repaired the fire door, and the issue was entered into the CAP.

Analysis: The inspectors determined that Entergy's failure to identify the degraded condition of the fire door, as required by license condition 2.K, was a performance deficiency. This issue was reasonably within Entergy's ability to foresee and correct, given that operators routinely pass through the door on their tours, and a similar issue had been identified by the NRC in January 2006. 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 inspectors determined that this finding was more than minor because it was associated with the Protection Against External Factors attribute of the Mitigating

Systems cornerstone and it affected the cornerstone objective of ensuring the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the degraded fire door was not capable of preventing the propagation of a fire between the RHR pump cells as required by Entergy's fire protection program. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process." The issue was determined to be of very low safety significance because the fire door, which was moderately degraded, provided a minimum of 20 minutes of fire endurance protection; and, the ignition sources and combustible materials in the RHR pump cells were situated in a manner that the degraded fire door would not have been subject to direct flame impingement.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because operators who routinely traverse through the degraded fire door during performance of their rounds had not identified the degraded condition of the door.

Enforcement: License condition 2.K. requires that Entergy implement and maintain in effect all provisions of the NRC-approved fire protection program, as approved in part by the NRC Safety Evaluation Report dated January 31, 1979. The January 31, 1979, Safety Evaluation Report requires administrative controls comparable to those described in NRC Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976." Branch Technical Position 9.5-1 requires that measures be established to assure that conditions adverse to fire protection, such as deficiencies, deviations, defective components, and non-conformities are promptly identified, reported, and corrected. Contrary to the above, on May 11, 2006, Entergy failed to promptly identify a degraded fire door between the 21 and 22 RHR pump cells. Because this violation is of very low safety significance and is entered into the corrective action program, it is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000247/2006003-02, Failure to Identify Degraded Residual Heat Removal Pump Cell Fire Door.

Once identified by the inspectors, Entergy declared the fire door inoperable, initiated appropriate compensatory measures, and entered the issue into the CAP (CR-IP2-2006-02934).

1R07 Heat Sink Performance (71111.07B - 3 samples)

a. Inspection Scope

Based on a plant specific risk assessment, past inspection results, and resident inspector input, the inspector selected three inspection samples consisting of:

- 21 component cooling water heat exchanger;
- 22 component cooling water heat exchanger; and
- 22 emergency diesel generator jacket water and lube oil heat exchangers.

The inspector reviewed documents to ensure that potential common cause heat sink performance problems that had the potential to increase risk were identified and corrected by Entergy. The inspector also reviewed records to ensure that potential macro fouling (silt, debris, etc.) issues and biofouling issues were closely examined by Entergy. To ensure adequate implementation of NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” the inspector reviewed Entergy’s inspection, cleaning, and eddy current testing methods and frequency with the responsible system engineers. The inspector compared surveillance test and inspection data, including as-found conditions and eddy current summary sheets, to the established acceptance criteria to verify that the results were acceptable and that system heat exchanger operation was consistent with design. The inspector reviewed heat exchanger design basis values and assumptions, plugging limit calculations, and vendor information to verify that they were incorporated into the heat exchanger inspection and maintenance procedures.

The inspector walked down the intake area, portions of the service water system, including the service water pump pit, Zurn strainer pit, component cooling water heat exchangers, and emergency diesel generator heat exchangers to assess the material condition and operational functioning of these systems and components. The inspector reviewed a sample of condition reports related to the component cooling water and emergency diesel generator heat exchangers, and the service water system to ensure that Entergy was appropriately identifying, characterizing, and correcting problems related to these systems and components. The documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (71111.08 - 4 samples)

a. Inspection Scope

The purpose of this inspection was to assess the effectiveness of Entergy’s inservice inspection program for monitoring degradation of the reactor coolant system boundary, risk-significant piping system boundaries, and the vapor containment boundary. The inspector assessed the inservice inspection activities using the criteria specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” and applicable NRC regulatory requirements.

The inspector observed portions of selected non-destructive examination activities, reviewed documents, and included samples of non-destructive examination activities associated with the repair and replacement of safety-related pressure boundary components. The sample selection was based on the inspection procedure objectives and risk significance. Specifically, the inspector focused on components and systems where degradation would result in a significant increase in risk of core damage. These reviews were conducted to verify the activities were performed in accordance with ASME Code requirements. The inspector reviewed a sample of examination reports and condition reports initiated during inservice inspection examinations to evaluate Entergy's effectiveness in the identification and resolution of problems.

The inspector reviewed the procedures used to perform visual examinations for indications of boric acid leaks from pressure retaining components including components above the reactor pressure vessel (RPV) head. Also, the inspector reviewed a sample of condition reports initiated as a result of the inspections performed in accordance with Entergy's boric acid control program. The inspector selected condition reports that identified evidence of leakage locations which could cause degradation of safety-significant components. The inspector reviewed nine condition reports which identified and characterized coolant leakage as active or inactive. The inspector reviewed operability evaluations to verify that the corrective actions performed were consistent with the requirements of the ASME Code and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action."

The inspector reviewed portions of the steam generator management plan, degradation assessment, and the final operational assessment to evaluate the steam generator inspection and management program. The inspector reviewed plant specific steam generator information, tube inspection criteria, integrity assessments, degradation modes, and tube plugging criteria. Entergy conducted eddy current testing of tubes in all steam generators to identify and quantify tube degradation mechanisms and to confirm tube integrity following the completion of two fuel cycles of operation. The inspector observed a sample of tubes examined from each generator to verify Entergy's examination of the entire length. The inspector interviewed data management and data acquisition personnel and resolution analysts. Also, the inspector reviewed examination data for selected tubes from each of the Unit 2 steam generators. These were selected by the inspector for review since they were noted to be representative of tubes which would most likely approach the tube plugging limit within the next three cycles. The inspector reviewed the characterization and disposition of the indications to assess the implementation of the steam generator inspection program.

The inspector observed portions of four in-process non-destructive examination activities and performed a documentation review of additional examinations. The activities included volumetric and surface examinations as follows:

- Ultrasonic test, volumetric, B feedwater system, 18 inch, 3/4 inch wall, carbon steel, weld 5-17;
- Magnetic particle test, surface, reactor pressure vessel, 42 inch length carbon steel meridional welds at azimuth 000, 120 and 240;
- Liquid penetrant test, surface, reactor coolant system, butt welded alloy 600 control rod drive mechanism penetration nozzles 13, 14 and 15; and
- Eddy current test, volumetric, reactor coolant system, steam generator tubes (one tube from steam generator 21 (R39C35), and two tubes from steam generator 24 (R38C57 and R38C64)).

The inspector reviewed documentation of two samples (work orders IP2-04-10333 and IP2-04-32627) of a repair/replacement activity which involved welding on an ASME pressure boundary and required the development and implementation of an ASME Section XI repair/replacement plan. The activity was performed during the previous operating cycle and involved the repair and replacement of portions of piping in the service water system. The inspector reviewed the ASME Section XI plan, piping replacement material, weld procedure specifications and qualifications, weld filler metals, specified non-destructive tests, acceptance criteria, and post-maintenance testing.

The inspector reviewed a sample of ultrasonic thickness measurements of a portion of service water system line 405 taken during refueling outage 2R17 following the identification of a through-wall leak in a segment of pipe (CR-IP2-2006-01883). The inspector reviewed the ultrasonic test procedure, acceptance criteria, proposed repair plan using welding techniques, material selection, applicable modification drawings, seismic calculations, and non-destructive test and acceptance criteria for weld locations.

The inspector visually examined selected portions of the vapor containment liner to verify compliance with the requirements of ASME Section XI, Section IWE, “Requirements for Class MC and Metallic Liners of Class CC Components.” The documents reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On June 12, 2006, the inspectors observed a simulator session conducted for licensed operator requalification training as required by 10 CFR Part 55.59, “Requalification.” The simulator exercise was performed to evaluate operator response to a steam generator tube leak (SGTL) and subsequent steam generator tube rupture (SGTR). The simulator session was conducted in accordance with lesson plan SES-E-3-LOOP, “SGTL, Feedwater Regulating Valve Controller Failure, SGTR with Loss of Off-Site Power.” The inspectors reviewed the simulator scenario to determine if the scenario contained: (1) clear event descriptions with realistic initial conditions; (2) clear start and end points; (3) clear descriptions of visible plant symptoms for the crew to recognize; and (4) clear expectations of operator actions in response to abnormal conditions.

During the simulator exercise, the inspectors evaluated the team’s performance for: (1) clarity and formality of communications; (2) correct use and implementation of emergency operating procedures and abnormal operating procedures; (3) operators’ ability to properly interpret and verify alarms; (4) operators’ ability to classify events in a timely fashion; and (5) operators’ ability to take timely actions in a safe direction based on transient conditions. In addition, the inspectors evaluated the control room supervisor’s ability to exercise effective oversight and control of the crew’s actions during the exercise. The inspectors verified that the feedback from the instructors was thorough, that they identified specific areas for improvement, and that they reinforced management expectations regarding crew competencies in the areas of procedure use, communications, and peer checking. The inspectors also evaluated Entergy's post-scenario critique. The documents reviewed are listed in the Attachment. The simulator session inspection represents one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 1 sample)

a. Inspection Scope

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

- Pressurizer pilot-operated relief valves.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control (71111.13 - 5 samples)

a. Inspection Scope

The inspectors observed selected portions of emergent and planned maintenance work activities to assess Entergy's risk management against the requirements of 10 CFR Part 50.65(a)(4) and plant procedures. 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 discussed risk management with maintenance and operations personnel. The documents reviewed are listed in the Attachment. The following activities represent five inspection samples:

- WO-IP2-05-21757, “Full-Flow Testing of the 22 Auxiliary Boiler Feedwater Pump with Feedwater Flow Transmitter Testing;”
- WO-IP2-06-18343, “Service Water Line 405 Leak;”
- WO-IP2-05-23569, “Service Water Valve SWN-840 Leak;”
- WO-IP2-05-26246, “Motor-Operated Valve Mechanical Interlock Failures and Extent of Condition Review;” and
- WO-IP2-06-23628, “22 Auxiliary Boiler Feedwater Pump Pressure Regulating Valve Repairs Following Failure of the Valve to Stroke Open.”

b. Findings

Introduction: A Green self-revealing finding was identified because Entergy failed to take adequate corrective actions for a degraded service water pipe in the primary auxiliary building. Degradation of this pipe was identified in 2003, but was not adequately evaluated or repaired. Consequently, in April 2006, continued corrosion of this pipe led to a through-wall leak.

Description: On April 19, 2006, plant operators identified a one pint per hour leak from one of the two 24 inch service water pipes running from the component cooling water heat exchangers to the discharge canal. Entergy’s initial testing and analysis determined that the pipe, which is designed to be seismically qualified, had degraded to a degree that it could not be relied upon to remain operable under design seismic loads. Failure of this pipe in a seismic event would lead to flooding of the lower elevation of the primary auxiliary building and loss of the safety-related RHR pumps, which are required to remain operable in a seismic event. As a result, Entergy declared both RHR pumps inoperable and entered the appropriate Technical Specification Limiting Condition for Operation.

A review of previous work orders and condition reports found that the corrosion of these pipes had previously been identified in February 2003 and documented in condition report CR-IP2-2003-00941. At that time, work orders were written to perform visual inspections, cleaning, and painting of the pipes. Once the work orders were written, the condition report was closed. The visual inspections were completed under WO-IP2-03-14399, and based on the condition of the pipe, the Entergy inspector recommended that ultrasonic testing measurements be taken. However, no condition report or work order was generated to track completion of the ultrasonic tests. In addition, work order IP2-03-13693, which was generated to clean and paint the pipe, was taken to “complete” status and closed with no cleaning or painting actually being completed. According to Entergy procedure EN-LI-102, “Corrective Action Process,” if a work order associated with a condition report is to be closed with no actions taken, a review is required by plant management. In the case of WO-IP2-03-13693, no management review took place. Additionally, the results of the visual inspections and the recommendation to complete followup ultrasonic testing were not documented in a condition report or work order.

EN-LI-102 requires a condition report be generated for adverse conditions. In this case, because no condition report was written, further testing was not conducted and the material condition of the pipe was not evaluated. Failure to take these actions contributed to continued degradation of the piping, which ultimately resulted in the leak and the declaration of RHR pump inoperability.

Entergy entered this issue into the corrective action program as condition report IP2-2006-02156, repaired the degraded pipe, and initiated actions to strengthen control over work orders associated with condition reports. Entergy completed a more detailed structural analysis of the pipe and determined that even in its degraded condition, the pipe would not have catastrophically failed under seismic loads.

Analysis: The inspectors determined that Entergy's failure to take adequate corrective actions for the previously identified degraded service water pipe was a performance deficiency. This issue was reasonably within Entergy's ability to foresee and prevent, given that the degradation had been identified and documented in a condition report and the requirements were addressed in Entergy procedure EN-LI-102. 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 procedures.

The inspectors determined that this finding, which was associated with the Mitigating Systems cornerstone, was more than minor because if it was left uncorrected it would have become a more significant safety concern. Specifically, had the service water pipe not developed a through-wall leak, the severe degradation would not have been identified; and the pipe would have continued to degrade to the point where it was unable to withstand seismic loads. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was of very low safety significance because it represented a qualification deficiency that was confirmed not to result in loss of operability per Part 9900 Technical Guidance, "Operability Determination Process for Operability and Functional Assessment."

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not implement timely and effective corrective actions for degraded service water piping in the primary auxiliary building.

Enforcement: Because this finding is associated with degradation of a non safety-related pipe, no violation of regulatory requirements occurred: FIN 05000247/2006003-03, Inadequate Corrective Actions for Degradation of Service Water Piping.

Entergy entered this issue into the corrective action program as condition report IP2-2006-02156, repaired the degraded pipe, and initiated actions to strengthen control over work orders associated with condition reports.

1R14 Personnel Performance During Non-routine Plant Evolutions and Events (71111.14 - 2 samples)

a. Inspection Scope

For the two non-routine planned evolutions described below, the inspectors reviewed plant procedures, operator logs, plant computer data, and strip charts to evaluate operator performance in coping with non-routine events and to determine if operator response was in accordance with required procedures and training. The documents reviewed are listed in the Attachment.

- From April 12 through 18, 2006, the inspectors observed control room activities associated with plant coastdown to refueling outage 2R17. The inspectors reviewed plant operating procedures to verify that they were adequate, ensured that precautions and limitations were appropriately observed, and verified that operator control over plant parameters was acceptable.

- On April 18, 2006, the inspectors observed preparation activities associated with biennial testing of the main steam safety valves. The inspectors reviewed the surveillance requirements to evaluate the associated precautions and limitations, and to ensure that plant conditions supported the test. The inspectors attended the pre-job brief for control room personnel, ensured that risk was appropriately evaluated, and observed control room operator activities during performance of the test.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 6 samples)

a. Inspection Scope

The inspectors selected a sample of Entergy's operability evaluations for review on the basis of potential risk significance. The inspectors assessed the accuracy of the evaluations, the use and control of compensatory measures, if needed, and compliance with the Technical Specifications. The inspectors' review included a verification that the operability evaluations were completed as specified by procedure ENN-OP-104, "Operability Determinations," and that they were technically adequate. References used during these reviews included the Technical Specifications, the Technical Requirements Manual, the Updated Final Safety Analysis Report, and associated design basis documents.

The following operability evaluations, which represent six samples, were reviewed:

- CR-IP2-06-03497, “Unexpected Communication Between the Clean and Dirty Sides of the Vapor Containment Sump;”
- CR-IP2-06-01625, “Service Water Valve SWN-840 Leak;”
- CR-IP2-06-02133, “RHR System Operability With Service Water Line 405 Degradation;”
- CR-IP2-06-02762, “Control Rod Drive H-6 Drag Test Failure;”
- CR-IP2-06-01894, “22 EDG West Air Start Motor With Missing Seismic Mounting Bolts;” and
- CR-IP2-06-03530, “EDGs Following the Identification of Inadequate Surveillance Testing.”

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” because plant procedures were not followed during the installation of compensatory measures to restore operability of the RHR pumps following the identification of service water piping degradation in the primary auxiliary building.

Description: On April 25, 2006, Entergy identified severe corrosion on a seismically-qualified service water line in the primary auxiliary building. The line was declared inoperable since it could not be shown that it would maintain its structural integrity during a design basis earthquake. In addition, both RHR pumps were declared inoperable because a failure of this line would result in flooding of the RHR pump cells. Entergy installed compensatory measures to mitigate the consequences of a service water line break and declared the RHR pumps operable.

The inspectors reviewed the operability evaluation and the associated compensatory measures for the RHR pumps. The compensatory measures consisted of: (1) blocking open an exit door from the primary auxiliary building to allow water to flow out of the building; (2) installation of sandbag walls at the entrance to both RHR pump cells; (3) blocking the RHR pump cells’ floor drains with sandbags; (4) placing a portable sump pump in one of the pump cells; and (5) stationing an operator in the vicinity of the RHR pump cells to operate the sump pump and to notify the control room of any flooding. The inspectors found that the compensatory measures were not installed or controlled as a temporary alteration. Procedures EN-OP-104, “Operability Determinations,” and ENN-DC-136, “Temporary Alterations,” both require that physical compensatory actions be evaluated as temporary alterations. In addition, the inspectors found multiple deficiencies with the compensatory measures. Specifically, there were no written procedures for the operator stationed at the RHR pump cells; and, the inspectors identified one instance where the stationed operator was not familiar with his assigned duties. The fire door between the cells was not blocked open which would have inhibited water flow to the temporary sump pump from one of the cells; and, one of the sandbag walls did not meet the minimum required height. The inspectors reviewed the requirements of EN-OP-104 and ENN-DC-136 and determined that, had these procedures been followed, the required reviews and more rigorous implementation requirements should have identified these deficiencies before the RHR pumps were declared operable.

Entergy entered this issue into the corrective action program and took actions to correct the deficiencies associated with the compensatory measures. Specifically, a procedure was developed for the operator, the height of the wall was raised, and the fire door between the RHR pump cells was blocked open to allow for water flow. Because a required fire door was made inoperable, Entergy also took the actions in accordance with procedure SAO-703, "Fire Protection Impairment Criteria and Surveillance."

Analysis: The inspectors determined that Entergy's failure to follow procedures for implementation of compensatory measures required for the operability of safety-related equipment was a performance deficiency. This issue was reasonably within Entergy's ability to foresee and prevent, given that the requirements were well-defined in their procedures. 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 procedures.

The inspectors determined that this finding, which was associated with the Mitigating Systems cornerstone, was more than minor because it was similar to Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," Example 3.a, in that, the deficiencies identified with Entergy's compensatory measures required significant rework to ensure RHR pump operability. If the compensatory measures had been installed through the use of the appropriate procedural requirements, the required structured reviews would have provided an opportunity to identify these deficiencies prior to declaring the RHR pumps operable. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs." The inspectors determined that Checklist 2 was applicable because the unit was in cold shutdown with the reactor coolant system closed and filled with inventory in the pressurizer, and, time to core boil was less than two hours. The inspectors determined that this finding was of very low safety significance because Entergy's subsequent analysis concluded that the service water pipe remained capable of withstanding a seismic event; hence, the finding did not degrade the equipment, instrumentation, training or procedures needed for any shutdown safety function.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not follow plant procedures when implementing a temporary alteration required for the operability of safety-related equipment.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. Contrary to the above, on April 25, 2006, Entergy did not follow procedures EN-OP-104, "Operability Determinations," and ENN-DC-136, "Temporary Alterations," during the installation of compensatory measures to restore operability of the RHR pumps following the identification of service water piping degradation in the primary auxiliary building. Because this violation is of very low safety significance and has been entered into Entergy's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000247/2006003-04, Failure to Follow Plant Procedures for Implementation of Compensatory Measures.

Entergy entered this issue into the corrective action program (CR-IP2-06-02133, -02298, -02221) and took actions to correct the deficiencies associated with the compensatory measures.

1R17 Permanent Plant Modifications (71111.17A - 1 sample)

a. Inspection Scope

The inspectors reviewed a modification associated with upgrades to the vapor containment and recirculation sumps. This modification was implemented using ER-04-2-234, "IP2 [Indian Point Unit 2] Recirculation Sump and Vapor Containment Sump Strainer Upgrade," to address concerns associated with pressurized-water reactor containment sump clogging. The inspectors reviewed the modification package to ensure it was technically adequate and conducted walkdowns of the modification to verify it was completed in accordance with the design. The inspectors reviewed quality control records to verify the strainers were installed correctly, interviewed Entergy quality control personnel, and observed Entergy inspectors performing post-installation strainer inspections. The inspectors reviewed in-progress engineering changes to ensure they would not have an adverse effect on sump operability. The inspectors evaluated their observations against the requirements of 10 CFR Part 50.59, "Changes, Tests, and Experiment;" 10 CFR Part 50, Appendix B; and Technical Specifications. The documents reviewed are listed in the Attachment. The review of this modification represents one sample.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 4 samples)

a. Inspection Scope

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

The selected testing activities involved components that were risk-significant as identified in the Indian Point Unit 2 Individual Plant Examination. The following testing activities were evaluated:

- WO- IP2-06-00620 and WO-IP2-04-32065, reactor coolant pump thermal barrier component cooling water containment isolation valve following corrective maintenance for seat leakage and overtorque conditions;
- WO-IP2-06-17981, -17981, -19930, -18006, -19966, -17994, -17998, -18007, -20670, and -19967, various safety-related motor-operated valves following corrective maintenance to the mechanical interlocks in the motor control cabinets;
- WO-IP2-06-18693, -18694, -18770, and -18771, corrective actions for deficiencies identified during the vapor containment integrated leak rate test; and
- WO-IP2-4-30948, 22 EDG following completion of eight year preventive maintenance items.

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," because Entergy's post-maintenance test on the 21 EDG following a governor replacement in November 2004 was not adequate to ensure it could perform its intended design function.

Description: On May 24, 2006, Entergy generated condition report IP2-2006-03286, which stated that an extent of condition review for post-maintenance test concerns associated with work performed on the 22 EDG had determined that the 21 EDG had not been run at its full load rating of 2300 kilowatts (kW) following a governor replacement in November 2004. An operability evaluation was performed to assess the impact of the condition on the 21 EDG's ability to perform its design function. The evaluation concluded that procedure 2-PT-R84A, "21 EDG 8 Hour Load Run," had been completed satisfactorily and there was no requirement to operate the 21 EDG at 2300 kW. Based on this, Entergy determined that the 21 EDG could perform its design function, was considered operable, and no further actions were completed to address operability.

On June 1, 2006, the inspectors raised concerns about the adequacy of the biennial eight hour EDG load run surveillance test conducted on all three EDGs because this test did not bound the electrical loading the EDGs would be required to carry during a design basis accident. Based on this and subsequent concerns with the associated operability evaluation, condition report IP2-2006-03685 was written on June 15, 2006. The initial evaluation of operability performed for this concern stated that the 22 and 23 EDGs were considered operable since they had been tested at 2300 kW following their governor replacements; however, the 21 EDG needed to be tested to 2300 kW prior to being declared operable. Surveillance procedure 2-PT-M21A, "Emergency Diesel Generator 21 Load Test," was modified to include running the EDG at 2300 kW. This test was performed on June 16, 2006. During the test, the 21 EDG achieved a load of 2250 kW, but could not achieve its rated load of 2300 kW. Troubleshooting revealed that the fuel rack linkage had not been properly aligned during the governor replacement in 2004.

This misalignment prevented the cylinders from receiving maximum fuel with the fuel rack at its full fuel position.

The inspectors reviewed calculation FEX-00039-02, “Emergency Diesel Generator Loading Study,” and determined that the worst case accident loading on the 21 EDG would be during a loss of coolant accident coincident with a loss of off-site power and the failure of the 23 EDG. The peak loading of 2268 kW occurs for a short period during the switch over to low pressure recirculation where the 21 recirculation pump is started prior to securing the 21 safety injection pump. The inspectors determined that EDG loading for all other accident sequences was less than 2250 kW.

Entergy repaired the misalignment, successfully retested the EDG at 2300 kW, and entered this issue into the corrective action program.

Analysis: The inspectors determined that the inadequate post-maintenance test following governor replacement on the 21 EDG was a performance deficiency. Entergy failed to ensure that all the necessary testing to confirm that the 21 EDG would perform satisfactorily in service was identified and performed, which was reasonably within Entergy’s ability to foresee and prevent. 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 inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, because Entergy failed to perform an adequate post-maintenance test on the 21 EDG, the governor misalignment was not identified, and the EDG was unable to achieve its rated load of 2300 kWs. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The inspectors determined that this finding was of very low safety significance because it was not a qualification deficiency; it did not represent a loss of safety function for a train or system as defined in the plant specific risk-informed inspection notebook; and it was not risk significant due to external event initiators.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” requires, in part, that a test program shall assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, in November 2004, Entergy’s post-maintenance testing failed to identify and incorporate all testing required to demonstrate that the 21 EDG could satisfactorily perform its intended safety function during a design basis accident scenario. Because this violation is of very low safety significance and has been entered in Entergy’s corrective action program this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC

Enforcement Policy: NCV 05000247/2006003-05, Inadequate Post-Work Test on 21 Emergency Diesel Generator.

Entergy repaired the misalignment, successfully retested the EDG at 2300 kW, and entered this issue into the corrective action program (CR-IP2-06-03396).

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

a. Inspection Scope

The inspectors reviewed the schedule and risk assessment documents associated with the Indian Point Unit 2 refueling outage 2R17 to confirm that Entergy appropriately considered risk, industry operating experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. Prior to the refueling outage, the inspectors reviewed Entergy's outage risk assessment to identify risk-significant equipment configurations and to determine whether planned risk management actions were adequate. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored Entergy's controls over the outage activities.

The inspectors observed the Unit 2 shutdown and cooldown on April 19, 2006, to verify that cooldown rates met Technical Specifications (TS) requirements. Inspectors also evaluated conditions within containment for indications of unidentified leakage and damaged equipment. The inspectors verified that Entergy managed the outage risk commensurate with the outage plan. Inspectors periodically observed refueling activities from the refueling bridge in containment and the spent fuel pool (SFP) to verify refueling gates and seals were properly installed and to determine whether foreign material exclusion boundaries were established around the reactor cavity. Core offload and reload activities were periodically observed from the control room and refueling bridge to verify whether operators adequately controlled fuel movements in accordance with procedures.

The inspectors verified that tagged equipment was properly controlled and equipment configured to safely support maintenance work. Equipment work areas were periodically observed to determine whether foreign material exclusion boundaries were adequate. During control room tours, the inspectors verified that operators maintained adequate RCS level and temperature and that indications were within the expected range for the operating mode.

The inspectors determined whether offsite and onsite electrical power sources were maintained in accordance with Technical Specification requirements and consistent with the outage risk assessment. Periodic walkdowns of portions of the onsite electrical buses and the emergency diesel generators were conducted during risk-significant electrical configurations. The inspectors verified through routine plant status activities that the decay heat removal safety function was maintained with appropriate redundancy as required by Technical Specifications and consistent with Entergy's outage risk assessment. During core offload conditions, the inspectors periodically determined whether the fuel pool cooling system was performing in accordance with applicable system operating procedures and consistent with Entergy's risk assessment for the refueling outage. Equipment and procedures to mitigate a loss of spent fuel cooling were reviewed by the inspectors to ensure they were available and ready for use.

Reactor coolant system inventory controls and contingency plans were reviewed by the inspectors to determine whether they met Technical Specification requirements and provided for adequate inventory control. Inspectors reviewed procedures and observed portions of activities in the control room when the unit was in reduced inventory modes of operation, including mid-loop operations. Water level and core temperature measurement instrumentation was reviewed by the inspectors to ensure they were installed and operational. Calculations that provide time to core boil information were also reviewed for reactor coolant system reduced inventory conditions as well as the spent fuel pool during high heat loads.

Containment status and procedural controls were reviewed by the inspectors during fuel offload and reload activities to verify that Technical Specification requirements and procedure requirements were met for containment. Specifically, the inspectors verified that during fuel movement activities, personnel, materials, and equipment were staged to close containment penetrations as assumed in the licensing basis. The documents reviewed are listed in the Attachment. The combined efforts described above represent one inspection sample.

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because plant procedures for reactor coolant system venting following depressurization were inadequate. This resulted in the formation of an 850 gallon void in the reactor vessel head while the plant was shutdown and depressurized.

Description: On April 21, 2006, Entergy depressurized the reactor coolant system and drained the pressurizer to a level of 68 percent. These actions were performed in accordance with plant operating procedure 2-POP-3.3, "Plant Cooldown, Mode 3 to Mode 5," Attachment 10, "Depressurizing/Draining Solid Pressurizer Using PRT [Pressurizer Relief Tank] Gas Volume." The operators completed Section 5 of the procedure to establish pressurizer level at 68 percent, and then stopped while preparations were made for performing containment leak rate testing. On April 25, 2006, following the containment test, the reactor vessel head vent was inadvertently opened due to a tagout restoration error. When the vent was opened, the operators noted that pressurizer level dropped approximately six percent, and that gas issued from the reactor head vent.

When the reactor coolant system is depressurized, gasses come out of solution and tend to collect in the reactor vessel head area. A pressurized nitrogen blanket is maintained on the volume control tank, which is part of the chemical and volume control system (CVCS). The nitrogen dissolves in the water until an equilibrium concentration is reached. As this water is transferred to the reactor coolant system, which is at atmospheric pressure, the nitrogen will come out of solution and collect at high points in the system. Therefore, if plant conditions are maintained such that the pressurizer is not drained and the CVCS system is in service, nitrogen gas will continue to build up in the reactor coolant system unless properly vented. This can result in inaccurate indication of reactor coolant system water inventory and adversely impact the time to core boil if decay heat removal capability is lost. In addition, if the gas continues to displace water from the reactor vessel, it can challenge the operation of the RHR pumps, which are used to remove the decay heat.

The inspectors reviewed plant computer data from April 21 through 25, 2006. Based on the observed changes in pressurizer level and containment pressure, the inspectors determined that a nitrogen bubble had formed in the reactor vessel head region following reactor coolant system depressurization. The six percent drop in pressurizer level equated to approximately 850 gallons of primary coolant that had transferred back into the reactor vessel when the reactor vessel head vent was opened. This corresponded to a gas space of approximately one foot in the reactor head region. As the gas built up, it forced primary coolant out of the reactor vessel and into the pressurizer. During this time period, the reactor vessel level indicating system was out of service for maintenance, and pressurizer level was the only indication of reactor coolant system inventory.

The inspectors reviewed procedure 2-POP-3.3, "Plant Cooldown, Mode 3 to Mode 5," and noted that Section 6 of Attachment 10 addressed venting of the reactor vessel head. The operators had stopped the procedure for the containment leak rate testing following the completion of Section 5. The inspectors determined that this was in accordance with the operations procedure usage rules; however, they also noted that the procedure had no precautions or limitations which would prevent maintaining the plant in a condition where the reactor coolant system is depressurized without the vessel head being periodically vented. Venting of the reactor vessel head would only be performed based on operator knowledge of gas buildup. The inspectors determined that the operator's ability to identify the need to vent the reactor vessel head would be challenged, since the primary means of identifying gas accumulation would be by the reactor vessel level indication system, which was not required to be in service.

Entergy entered this issue into the corrective action program and is evaluating changes to plant operating procedures.

<u>Analysis</u>: The inspectors determined that the failure to provide adequate procedural guidance to prevent gas accumulation in the reactor vessel during depressurized plant operations was a performance deficiency. This issue was reasonably within Entergy's ability to foresee and prevent. 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 inspectors determined that this finding, which was associated with the Initiating Events cornerstone, was more than minor because if it was left uncorrected, it would have become a more significant safety concern. Specifically, operating procedures did not require venting of the reactor coolant system to prevent gas buildup when plant conditions were conducive to this phenomena. Had the plant been maintained in these plant conditions without venting, the gas void would have continued to grow to the point where it could have challenged operation of the RHR pumps. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs.” The inspectors determined that Checklist 3 was applicable because the unit was in cold shutdown with the reactor coolant system open; refueling cavity level was less than 23 feet; and time to core boil was less than two hours. Based upon Appendix G, Attachment 1, Phase 2 screening criteria of Checklist 3, the inspectors concluded that a Phase 2 analysis was needed to characterize the risk significance of this finding. The Region I Senior Reactor Analyst performed the Phase 2 analysis, and evaluated this finding using Inspection Manual Chapter 0609, Appendix G, Attachment 2, “Phase 2 Significance Determination Process Template for PWR During Shutdown.” Based upon the plant operating state (plant operating state 2, reactor coolant system in cold shutdown and vented via the pressurizer); and the early time window (high decay heat load and short time to core boil); the Senior Reactor Analyst characterized this issue as a precursor finding and used Worksheet 2, “Loss of Level Control Initiating Event,” to assess the risk significance of this finding. The finding was determined to be of very low safety significance based upon the availability of mitigation systems and the low initiating event (loss of inventory) likelihood.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy’s shutdown procedures were not complete and accurate, in that, they failed to ensure the reactor vessel head was adequately vented.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. Contrary to the above, from April 21 through 25, 2006, Entergy failed to provide appropriate procedural direction in 2-POP-3.3, “Plant Cooldown, Mode 3 to Mode 5,” to ensure that a significant volume of gas would not accumulate in the reactor vessel head. Because this violation is of very low safety significance and has been entered into Entergy’s corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000247/2006003-06, Inadequate Procedure for Venting the Reactor Vessel Head While Shutdown.

Entergy entered this issue into the corrective action program (CR-IP2-06-02233) and is evaluating changes to plant operating procedures.

1R22 Surveillance Testing (71111.22 - 8 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) the test discrepancies were appropriately evaluated. The surveillance tests observed were based on risk-significant components as identified in the Unit 2 Individual Plant Examination. The documents reviewed are listed in the Attachment. The surveillance tests reviewed, which represent eight samples, included:

- 2-PT-R16, “Recirculation Pumps,” Revision 18;
- 2-PT-R13, “Safety Injection Test,” Revision 21;
- 2-PT-R14, “Automatic Safety Injection System Electrical Load and Blackout Test,” Revision 21;
- 2-PT-10Y1, “Integrated Leak Rate Test,” Revision 1;
- 2-PT-R84B, “22 EDG 8 Hour Load Test,” Revision 8;
- 2-PT-M21A, “Emergency Diesel Generator 21 Load Test,” Revision 14;
- 2-PT-R7A, “Motor Driven Auxiliary Feed Pumps Full Flow,” Revision 19 and 2-PT-R22A, “Steam Driven Auxiliary Feedwater Pump Full Flow,” Revision 13; and
- 2-PT-R26A, “21 Spray Pump Discharge Valve 869A,” Revision 10.

b. Findings

.1 Introduction: The inspectors identified a Green NCV of 10 CFR Part 50.65(a)(4) because Entergy did not assess the risk associated with maintenance on the discharge containment isolation valve from the 21 containment spray pump, SI-869A. This maintenance resulted in the unavailability of the 21 containment spray train for a period of approximately 90 minutes.

Description: On April 11, 2006, the 21 containment spray train was removed from service for pre-outage leakage testing of valve SI-869A. The containment spray system consists of two independent trains, and is used to reduce containment pressure and to remove iodine following a loss of coolant accident or steam line break inside the containment. The inspectors reviewed Entergy’s control room logs and their on-line risk assessment, which is used to ensure that testing and maintenance does not place the plant in an unacceptable risk configuration. The inspectors identified that while Unit 2 operators had appropriately followed plant Technical Specifications for having one of the two trains inoperable, this maintenance had not been included in the daily risk estimate.

10 CFR Part 50.65(a)(4) requires that licensees assess the risk of maintenance activities prior to conducting work on SSCs that a risk-informed evaluation process has shown to

be significant to public health and safety. Because the containment spray system is safety-related, maintenance which makes one of the trains of this system unavailable requires a risk assessment. Entergy procedure IP-SMM-WM-101, “On-Line Risk Assessment,” requires a risk evaluation prior to removing such equipment from service. In this case, no risk assessment was completed for the work on SI-869A, as required by Entergy’s procedures and 10 CFR Part 50.65(a)(4). Because this maintenance was not evaluated for risk, redundant and backup equipment, such as the 22 containment spray train and the containment fan cooler units, were not protected during the work, as required by Entergy procedure IP-SMM-WM-100, “Work Management Process.”

Entergy entered this issue into the corrective action program and conducted a review of other scheduled pre-outage maintenance activities to ensure that they were being risk assessed. The inspectors reviewed testing that was conducted the following day on the discharge containment isolation valve from the 22 containment spray pump and reviewed risk assessments for a sample of other pre-outage work. The inspectors confirmed that the risk of performing these maintenance items was appropriately assessed.

Analysis: The inspectors determined that Entergy’s failure to assess the risk of maintenance activities on valve SI-869A, as required by 10 CFR Part 50.65(a)(4), was a performance deficiency. The inspectors determined that this issue was within Entergy’s ability to foresee and prevent, given that procedural guidance directs a risk assessment for this type of work. Traditional enforcement does not apply because there were no actual safety consequences or potential for impacting the NRC’s regulatory function, nor was the finding the result of any willful violation of NRC requirements or Entergy procedures.

The inspectors determined that this finding, which was associated with the Mitigating Systems cornerstone, was more than minor because it was similar to Example 7.e in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” in that, the licensee’s risk assessment failed to consider maintenance activities on components that prevent containment failure. Specifically, the risk of maintenance on the 21 containment spray train, which is used to limit containment pressure following a reactor accident, was not evaluated. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix K, “Maintenance Risk Assessment and Risk Management Significance Determination Process,” Flowchart 1, “Assessment of Risk Deficit.” The aggregate risk for the equipment removed from service, not including the 21 containment spray train, represented a core damage frequency of 1.24×10^{-5} per year. The aggregate risk including the 21 containment spray train represented a core damage frequency of 1.59×10^{-5} per year. Although the actual aggregate core damage frequency for the maintenance activities was higher than initially calculated, the overall risk remained very low. The inspectors determined the incremental core damage probability deficit (ICDPD) from the licensee’s core damage frequency; the actual duration of maintenance activities (approximately 1.5 hours); and calculated the ICDPD to be 5.99×10^{-10}. This was determined to be a Green finding having very low safety significance because the calculated risk deficit was not greater that 1×10^{-6}.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not appropriately incorporate risk insights into the planning of work activities on SI-869A in accordance with 10 CFR Part 50.65(a)(4) and the Site Management Manual IP-SMM-WM-101, "Online Risk Assessment."

Enforcement: 10 CFR Part 50.65(a)(4) requires that before performing maintenance activities on SSCs that a risk-informed evaluation process has shown to be significant to public health and safety, the licensee must assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to SSCs that a risk-informed evaluation process has shown to be significant to public health and safety. Contrary to the above, on April 11, 2006, for a period of approximately 90 minutes, Entergy did not assess the risk associated with planned maintenance activities on SI-869A which caused the safety-related 21 containment spray train to be unavailable. Because this violation is of very low safety significance and is entered into Entergy's corrective action program, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000247/2006003-07, Failure to Assess the Risk of Maintenance Activities on Valve SI-869A.

Entergy entered this issue into the corrective action program (CR-IP2-06-01834) and conducted a review of other scheduled pre-outage maintenance activities to ensure that they were being risk assessed.

.2 Introduction: The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because plant surveillance procedure 2-PT-R84B, "22 EDG 8 Hour Load Run," was not adequate to ensure testing at the appropriate power factor limit prescribed by Technical Specifications Surveillance Requirement 3.8.1.10.

Description: On May 4, 2006, the inspectors observed an eight hour load test on the 22 EDG, which was conducted in accordance with surveillance procedure 2-PT-R84B, "22 EDG 8 Hour Load Run." One of the purposes of this test was to meet the requirements of Technical Specifications Surveillance Requirement 3.8.1.10, which requires each emergency diesel generator to be run for eight hours at a power factor equal to or less than 0.85. A note in the Technical Specifications states that if grid conditions do not permit, the power factor limit is not required to be met, but should be maintained as close to the limit as practicable. This power factor requirement ensures that the generator and excitation system will be appropriately loaded during the test since, for a given real power, the current loading on both will increase as the power factor decreases.

The inspectors reviewed the procedure, observed portions of the test, and analyzed data obtained over the eight hour run. The inspectors found the required 0.85 or less power factor was not achieved for a significant portion of the test. The procedure requires that operators reach a specified real and reactive power band. If both bands are met, the EDG would be operating at the appropriate power factor. The procedure also provides limitations on stator current, generator terminal voltage, and field current. Reactive loading was limited during the low power portion of the test by generator terminal voltage limits and during the higher power portion by field current limits.

The inspectors noted that this test is normally performed with the electrical system aligned to receive off-site power from the 138 kilovolt (kV) system. In that configuration, the operators can adjust the station auxiliary transformer tap changer to vary incoming voltage and transfer reactive load to the EDGs, thus assisting in achieving the required power factor. However, the procedure had been temporarily changed to allow the test to be performed while aligned to the 13.8 kV off-site power system. In this configuration, the operators cannot vary the incoming voltage to assist in adding reactive load to the EDGs. The procedure change stated that if adequate reactive load could not be obtained to meet the power factor requirement, the system operator was to be contacted to adjust system voltage. The inspectors determined that this method would be of limited value, since adjustment of system voltage would impact other transmission system customers, and would therefore not be an effective means of adjusting reactive load on the EDG.

In addition, the inspectors found the field current limits provided in the procedure, which limited the ability to reach the power factor requirement during the high power portion of the test, were not set at the machine limits. The limits specified in the procedure were the expected field currents at a 0.8 power factor and 480 volts terminal voltage. The actual field current limit is based on excitation field heating. The V-curves show the field is capable of 131 amps of field current at maximum load. The limitation in the procedure was set at 114 amps. The inspectors determined that the field current during the peak loading period of a design basis accident would exceed 114 amps.

Entergy entered this issue into the corrective action program, evaluated the impact of the inadequate test on operability of the 22 EDG, and completed an extent of condition review for the other EDGs.

Analysis: The inspectors determined that this issue was a performance deficiency because Entergy failed to provide a procedure that was adequate to meet the surveillance requirements of the Technical Specifications for the EDG eight hour load test. This issue was reasonably within Entergy's ability to foresee and prevent since they determined the limits and initial conditions in the procedure to meet the Technical Specifications requirements. 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 inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone; and, it affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, because of deficiencies in procedure 2-PT-R84B, Entergy did not load the 22 EDG during its biennial endurance run consistent with Technical

Specification Surveillance Requirement 3.8.1.10. Failure to appropriately test the 22 EDG limited the licensee's ability to identify degraded generator performance. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that this finding was of very low safety significance because it was not a qualification deficiency; it did not result in the loss of a system or train safety function; and it did not screen as potentially risk-significant due to external events. Although this deficiency created a condition in which the 22 EDG was not tested in accordance with Technical Specifications requirements, additional evaluations by Entergy showed that the power factor during a design basis accident would be approximately 0.89 and showed no EDG degradation since the last run where the EDG was tested at load with the appropriate power factor requirements.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that procedure 2-PT-R84B, "22 EDG 8 Hour Load Run," was complete and accurate.

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. Contrary to the above, on May 4, 2006, Entergy's procedure for the EDG endurance test was not appropriate, in that, it did not ensure that the EDGs were run at a power factor of less than or equal to 0.85 as required by Technical Specification Surveillance Requirement 3.8.1.10, when grid conditions were not limiting. The inspectors determined that the power factor requirement was not met due to a combination of the initial conditions allowed by the test and the unnecessarily restrictive field current limits, rather than by grid conditions. Based on this conclusion, Entergy did not meet the conditions to invoke the note stating the power factor was not required due to grid conditions. Because this violation is of very low safety significance and has been entered in Entergy's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000247/2006003-08, Inadequate Surveillance Test Procedure for Emergency Diesel Generators.

Entergy entered this issue into the corrective action program (CR-IP2-06-03286), evaluated the impact of the inadequate test on operability of the 22 EDG, and completed an extent of condition review for the other EDGs.

1R23 Temporary Modifications (71111.23 - 1 sample)

a. Inspection Scope

The inspectors reviewed one temporary modification to ensure that the effects on plant operation were well understood and to ensure that no unintended adverse consequences would result from the modification. Specifically, the inspectors reviewed modification ER-02-2-234, "IP2 [Indian Point Unit 2] Recirculation Sump and VC [vapor containment] Sump Strainer Upgrade." The inspectors reviewed modification documentation for accuracy and completeness, assessed the basis for the modification, and reviewed associated procedures or changes to procedures used to control operation of the temporary modification. The inspectors completed a walkdown of the modification to ensure it was installed in accordance with design. The acceptability of this modification was evaluated against the requirements of 10 CFR Part 50.59, "Changes, Tests, and Experiments;" Technical Specifications; 10 CFR Part 50, Appendix B; and associated design basis documents. The documents reviewed are listed in the Attachment. The review of the temporary modification represents one inspection sample.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05 (OA) - 1 sample)

a. Inspection Scope

A region-based specialist conducted an inspection of Entergy's corrective actions related to the current Indian Point alert and notification system, and also of the progress made in the design and installation of the new siren system. The inspection was conducted on June 6 and 7, 2006, per the Reactor Oversight Process deviation authorized by the NRC Executive Director of Operations in a memorandum signed on October 31, 2005.

In order to assess the continued effectiveness of Entergy's corrective actions, the inspector discussed and reviewed the corrective actions implemented and all condition reports written against the current siren system since the inspector's March 2006 inspection. To assess the effectiveness of the corrective actions and the performance of Entergy's communication systems used in conjunction with the siren system, the inspector observed the performance of the monthly emergency planning communication test conducted on June 7, 2006. This test was conducted, in part, to validate the proper operation of the recently installed Radiological Emergency Communication System and the local four county Executive Hotline. The inspector monitored the test from the Indian Point emergency operations facility and observed the use of the two phone systems to establish contact with the local four county operation centers and warning points, and with the New York State Emergency Management Office (SEMO). On June 28, 2006, the inspectors observed the full siren sounding test from the emergency offsite facility and at siren W-46.

The inspector discussed, with the assigned project manager, the status of the new siren system to understand Entergy's progress toward meeting the milestone dates required by the NRC's Confirmatory Order dated January 31, 2006. The inspector also reviewed and discussed the initial Indian Point Energy Center prompt alert and notification system design report, which Entergy had submitted to SEMO for review. The inspector learned of planned changes to this submittal, such as the intent to add two new transmission towers for siren actuation to supplement the two existing towers. The inspector also visited a site in Putnam County to observe the construction and installation of one of the new siren towers in order to assess the progress of the installation and compliance with the schedule. The inspection represented one inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors observed an emergency preparedness drill conducted on June 14, 2006. The inspectors used NRC Inspection Procedure 71114.06, "Drill Evaluation," as guidance and criteria for evaluation of the drill. The inspectors observed the drill and critiques that were conducted from the participating facilities on-site, including the Indian Point Unit 3 plant simulator, and the emergency operations facility. The inspectors focused the reviews on the identification of weaknesses and deficiencies in classification; and notification timeliness, quality, and accountability of essential personnel during the drill. The inspectors observed Entergy's critique and compared the licensee's self-identified issues with the observations from the inspectors' review to ensure that performance issues were properly identified. The observation of the drill represented one sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 8 samples)

a. Inspection Scope

From May 1 through 11, 2006, the inspector conducted the following activities during refueling outage 2R17 and during a subsequent in-office review of an unintended exposure of a worker during core support barrel installation.

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

- Sump strainer modification;
- Reactor disassembly/reassembly;
- Steam generator primary inspection;
- Scaffolding;

- Valve work;
- Reactor coolant pump work; and
- Core support barrel replacement.

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

(3) With respect to the work activities listed in (1) above, a walkdown of these work areas was conducted with a radiation survey instrument to determine whether:

- RWPs, procedures, and engineering controls were in place;
- Entergy's surveys and postings were complete and accurate; and
- To verify that air samplers were properly located.

(4) The work activities listed in (1) above were reviewed against the radiological control requirements as specified in the applicable RWP and "as low as reasonably achievable" (ALARA) reviews, as well as verbal instructions provided by radiation protection technicians during radiological briefings to workers.

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

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

(7) Corrective action reports related to access controls were reviewed to determine if the follow-up activities were conducted in an effective and timely manner commensurate with safety and risk (see Section 4OA2).

(8) Licensee documents associated with a May 5, 2006 Performance Indicator event were reviewed to determine if there were any overexposures or substantial potential for overexposure associated with this incident.

The inspector verified that Entergy was properly implementing physical, engineering, and administrative controls for access to high radiation areas and other radiologically controlled areas (RCA), 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 Part 20, Indian Point Unit 2 Technical Specifications, and Entergy's procedures. The inspections constituted eight inspection samples.

b. Findings

.1 Introduction: A Green self-revealing NCV of Technical Specification 5.4.1 was identified because Entergy failed to follow procedural requirements during the core support barrel installation activity. This resulted in significantly higher dose rates than expected, and led to a worker's unplanned and unintended radiation exposure.

Description: On May 4, 2006, while lifting the core support barrel from its lower cavity support stand and moving it to the reactor vessel for installation, Entergy encountered unexpected high radiation dose rates. Specifically, dose rates to the maximally exposed worker reached 12.4 rad per hour, a level approximately ten times greater than expected. Consequently, the worker received an accumulated dose of 474 mrem, which was 174 mrem of unintended exposure above the 300 mrem limit specified in the radiation work permit (RWP) for the electronic dosimeter alarm set point.

Entergy refueling procedure 2-REF-002-GEN, "Core Support Barrel Removal and Installation," requires that water level in the refueling cavity meet minimum level requirements prior to moving the core support barrel. Investigation revealed that approximately 1.5 hours prior to the core support barrel movement activity, the refueling senior reactor operator determined that the refueling cavity water level was at the minimum required level of 94 feet 8 inches. However, it was later determined that the individual had incorrectly determined the water level by referencing the level to the position of an electrical junction box conduit mounted on the cavity wall, which he assumed was located at the minimum required water level. In actuality, the conduit reference was determined to be approximately 94 feet 5 inches, or three inches lower than required.

In addition, while Entergy personnel were aware of pre-existing leakage of water from the refueling cavity at the rate of between four and ten gallons per minute, the effect of the leakage on actual cavity water level was not considered. Subsequent investigation found that at the time the core support barrel was moved, the leak had reduced the refueling cavity water level to approximately six inches below the procedurally required minimum level.

Further, 2-REF-002-GEN requires that an indicator arrow be placed on the manipulator crane structure to indicate the maximum lift height of 103 feet 7 inches. This maximum lift height ensures the core support barrel is high enough to clear the reactor vessel flange, while sufficiently submerged to assure that adequate radiation shielding is maintained while the component is lifted and moved in the cavity. However, the indicator arrow had not been installed and was not available as a reference. Entergy determined that the lift director relied on his experience to raise the core support barrel to the appropriate height without any specific reference to control the maximum lift height. Based on the known water level and the dose rates encountered, it was determined that the core support barrel was raised several inches higher than allowed, which further reduced the water shielding and contributed to the unexpected high dose rates that were encountered during this activity.

Additionally, this work was being performed, in part, using as low as is reasonably achievable (ALARA) Plan 06-513, "2R17 10 Year Reactor Vessel ISI," which is a work control document that pertains to the core support barrel replacement activity. This procedure specifies that the core support barrel work be treated as an infrequently performed evolution, and requires implementation of procedure OAP-30, "Infrequently Performed Tests and Evolutions." Procedure OAP-30 is used to ensure adequate planning and preparation for infrequent or high-risk work, and provides for increased management involvement and oversight during the performance of the activity. This procedure also requires additional preparatory actions including evaluation of stop work criteria, enhanced coordination of the activity, verification that adequate safety margins are maintained, development of prerequisites for performing the activity, and contingency actions to address any unexpected conditions that may be encountered. In the case of the core support barrel move, Entergy failed to implement OAP-30. As a result, this infrequently performed evolution was not subject to the required level of management oversight, planning, preparation, and coordination that was expected and required for tasks of this type. This lack of attention and coordination contributed to the inadequate performance exhibited in the performance of this potentially radiologically hazardous work activity.

Once the higher dose rates were identified, Entergy moved the core support barrel to its final location and lowered it, which reduced radiation dose.

Analysis: The inspectors determined that Entergy's failure to follow the requirements of procedures 2-REF-002-GEN and OAP-30 was a performance deficiency. The inspectors determined that this issue was within Entergy's ability to foresee and prevent, given that procedural guidance was available to ensure adequate planning, preparation and implementation of this work activity. Traditional enforcement does not apply because 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 procedures.

The inspectors determined that this finding was more than minor because it was associated with the Program and Process attribute of the Occupational Radiation Safety cornerstone; and, it affected the cornerstone objective of ensuring adequate protection of workers from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, Entergy's failure to properly implement 2-REF-002-GEN and OAP-30 resulted in higher than expected dose rates and the unintended exposure to a worker. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process." The inspectors determined that this finding was of very low safety significance because it did not involve: (1) as low as reasonable achievable planning or work controls; (2) an overexposure; (3) a substantial potential for overexposure; (4) or an impaired ability to assess dose.

The inspectors determined that the finding had a cross-cutting aspect in the area of human performance because Entergy personnel failed to comply with plant procedures that were required and specified to support reinstallation of the core support barrel.

Enforcement: Technical Specification 5.4.1 requires that written procedures be established, implemented, and maintained for the activities specified in Regulatory Guide 1.33, Revision 2, Appendix A. Appendix A of Regulatory Guide 1.33 includes procedures for refueling and core alterations. Contrary to Technical Specification 5.4.1, on May 4, 2006, Entergy failed to adequately implement refueling procedure, 2-REF-002-GEN, “Core Support Barrel Removal and Installation,” and OAP-30, “Infrequently Performed Tests and Evolutions.” Once the higher dose rates were identified, Entergy moved the core support barrel to its final location and lowered it, which reduced radiation dose. Because this finding is of very low safety significance and has been entered into the corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000247/2006003-09, Failure to Implement Procedural Requirements Associated With Core Support Barrel Replacement.

This issue was entered into the corrective action program (CR-IP2-2006-02635) and a root cause analysis was completed.

.2 Introduction: A Green self-revealing NCV of 10 CFR Part 20.1501, “General,” was identified because Entergy failed to take adequate radiation surveys during the installation of the core support barrel. As a result, Entergy did not recognize that actual radiological conditions were significantly different than expected, which contributed to unplanned and unintended exposure of a worker.

Description: On May 4, 2006, Entergy relied on limited radiation detection equipment and devices to monitor dose rates as the core support barrel was being lifted and moved for reinstallation in the reactor vessel. Principally, reliance was placed on remote radiation monitoring devices that were intended only for general area coverage of the refueling floor, and were not positioned to be effective in evaluating the magnitude and extent of radiation associated with this specific work activity. Remote teledosimetry devices and personnel electronic dosimeters were employed to monitor personnel exposures. However, there was no effective use or application of field radiation survey equipment or other radiation detection equipment to directly survey and evaluate the magnitude and extent of the significantly high radiation dose rates that could be expected to result from this activity.

Further, the remote teledosimetry devices employed for area and personnel monitoring were not optimized to provide immediate radiation exposure and dose rate information. Specifically, the teledosimetry devices were set to provide radiation information at 40 second intervals, and consequently were not effective in providing actual real-time information concerning the magnitude and extent of the radiation dose rates that were actually being encountered during this three minute evolution. As a result, radiological control personnel were not able to immediately recognize that the radiological conditions at the onset of work and during the actual lift and movement of the component were significantly different than was expected.

By the time Entergy recognized the unexpected radiological conditions, and that some personnel may exceed authorized administrative exposure limits, there were few options available to mitigate the condition. The component move was temporarily stopped to quickly reassess the situation and consult with the radiation protection supervisor. Subsequently, a decision was made to complete the evolution so that the component could be restored to a shielded configuration as soon as possible, and the radiation dose rate reduced to prevent further unplanned exposure to personnel.

Analysis: The inspectors determined that Entergy's failure to adequately evaluate the radiological conditions associated with the core support barrel move was a performance deficiency. The inspectors determined that this issue was within Entergy's ability to foresee and prevent. Traditional enforcement does not apply because 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 procedures.

The inspectors determined that this finding was more than minor because it was associated with the Program and Process attribute of the Occupational Radiation Safety cornerstone; and, it affected the cornerstone objective of ensuring adequate protection of workers from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Specifically, because Entergy did not conduct an adequate radiation survey of actual conditions, they failed to detect the unexpectedly high dose rates, which resulted in an unplanned and unintended exposure to personnel. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process." The inspectors determined that this finding was of very low safety significance because it did not involve: (1) as low as reasonable achievable planning or work controls; (2) an overexposure; (3) a substantial potential for overexposure; (4) or an impaired ability to assess dose.

Enforcement: 10 CFR Part 20.1501 requires that licensees make radiation surveys that are necessary to ensure compliance with 10 CFR Part 20.1201 and are reasonable under the circumstance to evaluate the magnitude of radiation levels, the extent of radiation levels, and the potential radiological hazard. Contrary to 10 CFR Part 20.1501, on May 4, 2006, Entergy failed to conduct adequate radiological surveys to support reinstallation of the core support barrel, an activity having potential for high dose rates and radiological hazard. Because this finding is of very low safety significance and has been entered into the corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000247/2006003-10, Failure to Perform Adequate Surveys to Evaluate Radiation Levels During Core Support Barrel Replacement.

Once the higher dose rates were identified, Entergy moved the core support barrel to its final location and lowered it, which reduced radiation dose. This issue was entered into the corrective action program (CR-IP2-2006-02635) and a root cause analysis was completed.

2OS2 ALARA Planning and Controls (71121.02 - 5 samples)

a. Inspection Scope

From May 1 through 11, 2006, 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 was reviewed against the criteria contained in 10 CFR Part 20.1101(b) and Entergy's procedures. The following inspections constituted five inspection samples.

(1) The plant collective exposure history trend and current three-year rolling average collective exposure data was reviewed.

(2) The following highest exposure work activities for the Unit 2 Spring 2006 refueling outage were selected for review:

- Sump strainer modification;
- Reactor disassembly/reassembly;
- Steam generator primary inspection;
- Scaffolding;
- Valve work;
- Reactor coolant pump work; and
- Core support barrel replacement.

(3) The site procedure associated with maintaining occupational exposures ALARA, 0-RP-RWP-400, "RWP Preparation and ALARA Planning," was reviewed to evaluate the processes used to estimate and track work activity exposures.

(4) Job sites were observed to evaluate if surveys and ALARA controls were implemented as planned.

(5) Radiation worker and radiation protection technician performance was observed to ensure their performance demonstrated the ALARA principles.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS1 Gaseous and Liquid Effluents (71122.01 - 10 samples)

a. Inspection Scope

The inspector reviewed the following documents to evaluate the effectiveness of the licensee's radioactive gaseous and liquid effluent control programs. The requirements for radioactive effluent controls are specified in the Technical Specifications and the

Offsite Dose Calculation Manual (ODCM). The following inspections constituted ten inspection samples.

(1) The 2004 and 2005 Radiological Annual Effluent Release Reports were reviewed, including calculated public dose assessments. No anomalous results were noted in these reports. The 2005 Radiological Annual Effluent Release Report contained information on abnormal liquid releases of tritium and strontium-90 (sr-90) due to the presence of these radionuclides in groundwater. The inspector evaluated Entergy's analysis of this abnormal release pathway and verified that they had in place a program of sampling and dose assessment for this effluent pathway. Chapter 11 of the Updated Final Safety Analysis Report for Units 1, 2 and 3 were reviewed for those sections that described the gaseous and liquid radioactive waste systems. The latest quality assurance audit, QA-06-2005-IP-1, "IPEC Effluent and Environmental Programs," was also reviewed.

(2) The inspector observed the following plant equipment and work activities to evaluate the effectiveness of Entergy's radioactive gaseous and liquid effluent control programs:

- Walkdown of the radioactive gaseous and liquid effluent radiation monitoring (RMS) and effluent sampling systems to determine equipment operability and material condition;
- Observation of effluent sampling and laboratory analysis of those samples;
- Walkdown to determine the operability and material condition of air cleaning systems;
- Control room walkdown of effluent RMS displays to verify that the control room and local RMS readouts were in agreement, and to verify the effluent RMS alarm set points; and
- Entergy's understanding of the on-site groundwater contamination problem that was leading to the abnormal off-site liquid release of tritium and strontium-90 via this pathway. The inspector verified that Entergy had developed a technical basis for the on-site groundwater monitoring program, understood the groundwater flow patterns for the site, and had in place acceptable dose calculation methodology for this pathway.

(3) Selected gaseous and liquid radioactive waste release permits for 2005 and 2006 to date were reviewed with respect to procedural and ODCM requirements. The calculations were independently verified.

(4) Recent Unit 2 liquid effluent releases through effluent radiation monitor R-54 were reviewed. This monitor was inoperable, and the required program for compensatory sampling and analysis for radioactive effluent releases was verified.

(5) Changes to the ODCM (Revision 9 for Units 1 and 2, Revision 17 for Unit 3) were reviewed along with the justifications for each change.

(6) Monthly radioactive effluent dose projections were reviewed for each month of 2005 and 2006 to date with respect to Technical Specifications and ODCM methodology, and 10 CFR Part 50, Appendix I public dose requirements. The inspector verified calculations to ensure no regulatory requirements were exceeded.

(7) The inspector reviewed the most recent air cleaning system filter surveillance results required by Technical Specifications. This included visual inspection, pressure differential, in-leakage tests, laboratory charcoal efficiency test, and air flow capacity test, as required for the following:

- Unit 2 control room ventilation system;
- Unit 3 control room ventilation system; and
- Unit 3 containment fan cooler unit 32.

(8) The inspector reviewed the most recent calibration results for the gaseous and liquid effluent radiation monitors and associated flow rate measurement devices as required by the ODCM for the following:

- Unit 2 liquid radioactive waste (R54);
- Unit 2 plant vent (R43/44);
- Unit 3 liquid radioactive waste (R18);
- Unit 3 plant vent (R14); and
- Unit 3 plant vent, wide range (R27).

The inspector also reviewed the calibrations of the laboratory instrumentation (gamma ray spectrometers and liquid scintillation counting systems) and the sample preparation apparatus used for the analysis of effluent samples. Quality control data for this instrumentation was reviewed in order to verify that the instrumentation was being operated within acceptable performance parameters.

(9) Implementation of the laboratory quality assurance program for the analysis of effluent samples was reviewed, including the interlaboratory quality control program. The inspector also reviewed quality assurance audit QA-06-2005-IP-1, conducted from September 19, 2005, to December 5, 2005, of the Indian Point effluent and environmental programs.

(10) The inspector reviewed eight condition reports relative to the effluent control programs at Units 1 and 2 and Unit 3 from April 2005 to May 2006 (see Section 4OA2).

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP) (71122.03 - 11 samples)

a. Inspection Scope

(1) The inspector reviewed the current Annual Radiological Environmental Operating Report, and Entergy assessment results, to verify that the REMP was implemented as required by Technical Specifications and ODCM. The review included changes to the ODCM with respect to environmental monitoring commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, interlaboratory comparison program, and analysis of data. The inspector also reviewed the ODCM to identify environmental monitoring stations. In addition, the inspector reviewed:

- Entergy self-assessments and audits;
- Event reports;
- Inter-laboratory comparison program results;
- The updated final safety analysis report (UFSAR) for information regarding the environmental monitoring program and meteorological monitoring instrumentation; and
- The scope of the audit program to verify that it met the requirements of 10 CFR Part 20.1101.

(2) The inspector walked down six air particulate and iodine sampling stations, three ground water sampling locations, and seven thermoluminescent dosimeter (TLD) monitoring locations to determine that they were located as described in the ODCM and to determine the equipment material condition.

(3) The inspector observed the collection and preparation of a variety of environmental samples to include airborne particulate, iodine, and Hudson River aquatic vegetation samples. Other sample locations and sample aliquot compositing methods were demonstrated to include water inlet and discharge points, and shoreline sediment samples. The inspector verified that environmental sampling was representative of the release pathways as specified in the ODCM and that sampling techniques were in accordance with procedures.

(4) The inspector reviewed meteorological instruments to ensure they were operable, calibrated, and maintained in accordance with guidance contained in the UFSAR, NRC Safety Guide 23, and Entergy procedures. The inspector reviewed the meteorological data readout and recording instruments reflecting the control room readout and the tower to ensure they were operable and provided the same data values.

(5) The inspector reviewed each event documented in the Annual Radiological Environmental Monitoring Report, which involved a missed sample, inoperable sampler, lost TLD, or anomalous measurement for the cause and corrective actions. The inspector conducted a review of Entergy's assessment of any positive sample results.

(6) The inspector reviewed any significant changes made by Entergy to the ODCM as the result of changes to the land census or sampler station modifications since the last inspection. The inspector also reviewed technical justifications for any changed sampling locations and verified that Entergy performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment.

(7) The inspector reviewed the calibration and maintenance records for air samplers. The inspector reviewed:

- The results of Entergy's interlaboratory comparison program to verify the adequacy of environmental sample analyses performed by Entergy;
- Entergy's quality control evaluation of the interlaboratory comparison program and the corrective actions for any deficiencies;
- Entergy's determination of any bias to the data and the overall effect on the REMP; and
- Quality Assurance audit results of the program to determine whether Entergy met the Technical Specifications and ODCM requirements.

In addition, the inspector verified that the appropriate detection sensitivities with respect to Technical Specifications and ODCM are utilized for counting samples and reviewed the results of the quality control program including the interlaboratory comparison program to verify the adequacy of the program.

(8) The inspector observed the health physics control point egress point from the radiologically controlled area where Entergy monitors potentially contaminated material leaving the radiologically controlled area. The inspector inspected the methods used for control, survey, and release from these areas including observing the performance of personnel surveying and releasing material for unrestricted use.

(9) The inspector inspected radiation monitoring instrumentation to ensure it was appropriate for the radiation types present and was calibrated with appropriate radiation sources. The inspector reviewed Entergy's equipment to ensure the radiation detection sensitivities were consistent with the NRC guidance contained in Circular 81-07, "Control of Radioactively Contained Material," Information Notice 85-92, "Surveys of Wasted Before Disposal from Nuclear Reactor Facilities," and HPPOS-221, "Lower Limit of Detection (LLD) for Potentially Contaminated Oil."

(10) The inspector reviewed Entergy's audits and self-assessments related to the radiological environmental monitoring program since the last inspection to determine if identified problems were entered into the corrective action program as appropriate. Selected corrective action reports were reviewed since the last inspection to determine if identified problems accurately characterized the causes and corrective actions were assigned to each commensurate with their safety significance. Any repetitive deficiencies were also assessed to ensure that

Entergy's self-assessment activities were identifying and addressing these deficiencies.

(11) The inspectors verified several commitments made by Entergy as described in Entergy letter to the NRC dated April 10, 2006. Quarterly tritium and strontium-90 analyses of site perimeter monitoring well samples (MW-38, 48, 51, and 40) had been performed during the second quarter of 2006 and the sampling requirements were incorporated in Radiation Protection Standing Order RPSO-2006-03, "Monitoring Well/REMP Water Sampling."

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 2 samples)

a. Inspection Scope

The inspectors reviewed Entergy's data submitted to the NRC for the performance indicators (PI) listed below, and performed an independent verification that the source data was consistent with plant records. The inspectors reviewed Entergy's collecting and reporting process for PI data as described in procedure EN-LI-114, "Performance Indicator Process." The purpose of these reviews was to determine whether the methods for reporting PI data were consistent with the guidance contained in Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guidelines," Revision 2. The inspection included a review of the indicator definitions, data reporting elements, calculation methods, definition of terms, and clarifying notes for the performance indicators. Plant records and data were sampled and compared to the reported data. This inspection activity represents the completion of two samples.

Initiating Events Cornerstone

- Scrams With Loss of Normal Heat Removal (January 2004 - December 2005)

Barrier Integrity Cornerstone

- Reactor Coolant System Activity (January 2004 - December 2005)

The inspectors reviewed operator log entries, daily morning reports (including the daily condition report descriptions), the monthly operating reports, and Performance Indicator data sheets to determine whether Entergy adequately reported the above performance indicators during the previous eight quarters. In addition, the inspectors also interviewed licensee personnel responsible, as necessary for the Performance Indicator data collection, evaluation, and distribution.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Daily Review

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 corrective action program. This review was accomplished by reviewing hard copies of each condition report.

b. Findings

No findings of significance were identified.

.2 Semi-annual Trend Review (71152 - 1 sample)

a. Inspection Scope

The inspectors performed a semi-annual review to identify trends that might indicate the existence of a more significant safety issue. The inspectors included in this review repetitive or closely related issues that may have been documented by Entergy outside of the normal corrective action program (CAP), such as trend reports, performance indicators, major equipment problem lists, maintenance rule assessments and maintenance and CAP backlogs.

The inspectors reviewed Entergy's CAP database during the first and second quarters of 2006 to assess the total number and significance of condition reports written in various subject areas such as equipment or processes and to discern any notable trends in these areas. The inspectors reviewed Entergy's quarterly assessment/trend reports for both CAP and quality assurance for the fourth quarter of 2005 and the first quarter of 2006 to ensure they were appropriately evaluating and trending identified conditions. **The documents reviewed during the inspection are listed in the Attachment.**

b. Findings

No findings of significance were identified.

.3 PI&R Annual Sample - Selected Issue Follow-up Inspection - Corrective Actions Associated with NCV 05000247/2005004-01, Incorrect Setting of Relief Valve SI-855 Above System Design Pressure and Failure to Submit Required Changes to the Safety Analysis Report (71152 - 1 sample)

a. Inspection Scope

The inspectors conducted a review of Entergy's corrective actions associated with the setting of relief valve SI-855 above system design pressure and the failure to make required changes to the Updated Final Safety Analysis Report (UFSAR) as documented in condition report IP2-2005-03469.

In 1991, Indian Point Unit 2 raised the setpoint of relief valve SI-855 to 1670 pounds per square inch gauge (psig) to prevent repetitive lifting of the relief valve while operating the safety injection pumps. In 2005, Entergy engineering was reviewing a similar change to the Indian Point Unit 3 safety relief valve SI-855 setpoint due to periodic lifting and failure of relief valves. A review of planned changes to the Indian Point Unit 3 safety relief valve identified that contrary to the UFSAR, the setpoint for the Indian Point Unit 2 relief valves was set above system design pressure, no changes had been made to the UFSAR, and calculations supporting the 1991 safety evaluation report underlying the 1991 relief setpoint change had errors of a non-conservative nature. The NRC issued NCV 05000247/2005004-01 in October 2005 addressing these deficiencies.

The inspectors reviewed Entergy's operability and reportability evaluations, classification and prioritization of the resolution of the problem commensurate with its safety significance, the focus of corrective actions to correct the problem, and the completion of corrective actions in a timely manner commensurate with the safety significance of the issue. This inspection was accomplished by review of condition report IP2-2005-03469 and discussion with Entergy engineering and licensing personnel.

b. Findings and Observations

No findings of significance were identified. The inspectors determined that condition report IP2-2005-03469 was properly classified and prioritized based on Entergy procedure EN-LI-102, "Corrective Action Process." Corrective actions were completed in a timely manner commensurate with the safety significance of the issue, and Entergy properly addressed the associated reportability concerns. However, the inspectors determined that Entergy's operability evaluation was not sufficiently comprehensive, in that, calculations were not initially completed for maximum stresses based on the maximum expected temperatures provided in the UFSAR, and allowances for pressure transients described in American National Standards Institute piping code B31.1, "Power Piping Systems," were incorrectly used to increase design values. Additionally, the Entergy 1991 safety evaluation report, which contained several errors, was used verbatim as the 10 CFR Part 50.59 evaluation for the current acceptance of the change. Subsequent calculations revising the 1991 safety evaluation report demonstrated satisfactory system operability, and that the safety injection system was safe for continued operation with the increased setpoint. Entergy's causal evaluation of this issue identified historical weaknesses in modification processes and attention to detail in calculations. Errors in the most recent operability evaluation demonstrate that similar problems still exist. These issues were evaluated by the inspectors and determined to be of minor significance, because while they represented a lack of attention to detail and affected the results of the calculation in a non-conservative manner, they did not invalidate Entergy's operability conclusion.

.4 Identification and Resolution of Problems - Inservice Inspection (71111.08 - 1 sample)

a. Inspection Scope

The inspector reviewed a sample of condition reports which documented identified flaws and other nonconforming conditions discovered during refueling outage 2R17 and the previous outage. The inspector verified that the nonconforming conditions identified were reported, characterized, evaluated, and appropriately dispositioned in the corrective action program. The documents reviewed are listed in the Attachment. The review of the condition reports represents one inspection sample.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems - Occupational Radiation Safety (71121.01 and 71121.02 - 1 sample)

a. Inspection Scope

The inspector reviewed 25 corrective action condition reports that were initiated between July 2005 and May 2006, and were associated with the radiation protection program. The inspector verified that problems identified by these condition reports 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. The documents reviewed are listed in the Attachment. The review of the condition reports represents one inspection sample.

b. Findings

No significant findings were identified.

.6 Identification and Resolution of Problems - Public Radiation Safety (71122.01 - 1 sample)

a. Inspection Scope

The inspector reviewed the eight condition reports initiated between April 2005 and May 2006, relative to the radioactive gaseous and liquid effluent control programs. The inspector verified that problems identified by these condition reports 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 occurrences. The documents reviewed are listed in the Attachment. The review of the condition reports represents one inspection sample.

c. Findings

No significant findings were identified.

.7 Identification and Resolution of Problems - Public Radiation Safety (71122.03)

a. Inspection Scope

The inspector reviewed the following four corrective action condition reports that were initiated between January 2004 and May 2006, and were associated with the radiological environmental monitoring program:

- CR-IP3-2006-1784, "Control Room Met Tower Wind Speed Reads Inaccurately;"
- LO-JAFLO-2005-0134, "Radiation Protection Self Assessment Tracking Report;"
- CR-IP3-2004-3863, "Meteorological Data Review Showed 122 m to 10 m Delta-T Reading High for Existing Conditions;" and
- CR-IP3-2005-4690, "122 m Temperature Sensor Failed and is Reading Low."

The inspector verified that problems identified by these condition reports were properly characterized in the licensee's event reporting system, and that applicable causes and corrective actions were identified commensurate with the safety significance of the radiological occurrences.

b. Findings

No significant findings were identified.

4OA3 Event Followup (71153 - 4 samples)

.1 (Closed) Licensee Event Report (LER) 05000247/2005-001-01 Technical Specification Prohibited Condition Due to Exceeded the Allowed Completion Time for One Inoperable Train of ECCS Caused by an Inoperable Auxiliary Component Cooling Water Check Valve.

The inspectors reviewed Revision 1 of LER 05000247/2005-001. This LER was written to address an issue associated with inoperability of the 22 auxiliary component cooling water pump discharge check valve. Binding of this valve prevented it from fully closing, and resulted in the failure of the 21 auxiliary component cooling water pump to achieve the required flow during quarterly surveillance testing. The inspectors previously reviewed Revision 0 of this LER in Inspection Report 05000247/2005-002, and issued NCV 05000247/2005003-01, "Inadequate Post-Work Test Resulting in a Safety-Related System Exceeding the Allowed Outage Time," because Entergy failed to perform an adequate post-maintenance test following valve maintenance in November 2004. This

LER was revised to incorporate the post-maintenance test deficiency as a contributing cause, and to describe the associated corrective actions. In addition, the revision provided an updated risk assessment of the condition. No additional findings of significance or violations of NRC requirements were identified. This LER is closed.

.2 (Closed) LER 05000247/2005-002-00 Technical Specification Prohibited Condition Due to Exceeding the Allowed Completion Time for One Inoperable Train of ECCS Caused by an Inoperable Safety Injection Pump.

This LER was written to address the inoperability of the 23 safety injection pump due to gas void formation within the pump casing. Following identification of this issue, Entergy corrected the source of gas intrusion, implemented periodic venting and monitoring of safety injection system piping, and conducted training for plant staff. This issue was reviewed by the NRC and dispositioned in Inspection Reports 05000247/2005006 and 05000247/2005013. No additional findings of significance or violations of NRC requirements were identified. This LER is closed.

.3 (Closed) LER 05000247/2005-003-00 Automatic Start of Both Motor Driven Auxiliary Feedwater Pumps Due to Trip of 22 Main Feedwater Pump Caused by Low Lube Oil Pressure Due to Inadequate Procedure.

a. Inspection Scope

On December 22, 2005, a loss of main feedwater and an automatic start of both motor-driven auxiliary feedwater pumps occurred when plant operators attempted to place the standby main lube oil cooler in service. At the time, the plant was at approximately 67 percent power and a shutdown was in progress to repair a leaking feedwater regulating valve. Entergy subsequently determined that the cause of the loss of main feedwater was a momentary loss of oil pressure to the 22 main feedwater pump due to improper filling and venting of the standby cooler. The inspectors reviewed LER 05000247/2005-003-00, CR-IP2-05-05254, and Entergy's causal analysis of this issue. The inspectors also reviewed Entergy's corrective actions, including procedure changes made to prevent recurrence. A Green self-revealing finding, detailed below, was identified. This LER is closed.

b. Findings

Introduction: A Green self-revealing finding was identified because Entergy's procedure for placing the standby main lube oil cooler in service was inadequate. A deficiency in the procedure resulted in a loss of main feedwater, an automatic start of the motor-driven auxiliary feedwater pumps, and a steam generator level transient.

Description: On December 21, 2005, Entergy identified a packing leak on the feedwater regulating valve to the 24 steam generator (SG) . A plant shutdown was initiated to repack the valve. With the plant at approximately 67 percent power, Entergy operators made the decision to switch main lube oil coolers. There are two main lube oil coolers which are arranged in parallel. One cooler is normally in service to remove heat from the oil system, which supplies the main turbine and main feedwater pumps. The decision to switch coolers was based on a previously identified degraded condition associated with the service water supply to the in-service main lube oil cooler, and the drop in lube oil temperature as a result of the reduction in plant power.

When plant operators attempted to switch coolers, the control room received a momentary low oil pressure alarm on the 22 main feedwater pump and a subsequent pump trip. At the time, the 21 main feedwater pump was in recirculation mode and was not being used to feed the steam generators. The trip of the 22 main feedwater pump resulted in a loss of feedwater to the steam generators. Both motor-driven auxiliary boiler feedwater pumps automatically started, as designed. A subsequent investigation by Entergy identified that the standby lube oil cooler had not been properly filled and vented following maintenance. Due to a procedural deficiency, the cooler was also not filled and vented prior to being placed in service. Entergy procedure 2-SOP-21.4, “Main Boiler Feed Pump Lube Oil System,” required plant operators to check the cooler sight glass to ensure it was full. However, this was not adequate to ensure that no air was present in high points in the cooler.

Following the loss of main feedwater, steam generator water levels dropped from an approximate level of 50 percent to 30 percent in three minutes. If level in any one of the four steam generator drops to 9 percent, the reactor is automatically tripped. Plant operators were able to restore steam generator water levels prior to reaching the reactor trip setpoint, and secured the motor-driven auxiliary feedwater pumps.

<u>Analysis</u>: The inspectors determined that Entergy’s failure to develop an adequate procedure for filling and venting the main lube oil coolers prior to placing them in service was a performance deficiency. This issue was reasonably within Entergy’s ability to foresee and prevent. 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 inspectors determined that this finding was associated with the Initiating Events cornerstone; and, it was more than minor because it was similar to Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” Example 4.b, because the inadequacies in Entergy’s procedure caused a plant transient. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The inspectors determined that the finding was of very low safety significance because it did not contribute to the likelihood of both a reactor trip and the likelihood that mitigation equipment or functions would be unavailable. While the finding resulted in a loss of main feedwater, the low oil pressure condition was momentary, and both main feedwater pumps could have been promptly restored from the control room, if needed.

The inspectors determined that the finding had a cross-cutting aspect in the area of human performance because Entergy’s procedures were not complete and accurate, in that, they failed to ensure the standby main lube cooler was properly filled and vented prior to being placed in service.

Enforcement: Because this finding is related to a procedural deficiency associated with the nonsafety-related main lube oil system, no violation of regulatory requirements occurred: FIN 05000247/2006003-12, Inadequate Procedure for Placing Standby Main Lube Oil Cooler in Service.

Entergy entered this issue into the corrective action program (CR-IP2-05-05254) and revised 2-SOP-21.4 to require filling and venting of the standby cooler prior to placing it in service. Additional requirements to fill and vent the coolers following system startup were also added to the procedure.

.4 (Closed) LER 05000247/2005-004-00 Automatic Start of Both Motor Driven Auxiliary Feedwater Pumps Due to 22 Steam Generator High-High Level Signal Caused by Personnel Error.

On December 22, 2005, with the plant off-line to repair a packing leak on the 24 steam generator feedwater regulating valve, both motor-driven auxiliary feedwater pumps received automatic start signals. At the time, operators were using these pumps to manually control steam generator water level. Due to a personnel error, level in the 22 steam generator reached the high-high level setpoint, which resulted in signals to trip the main turbine and isolate the feedwater system. The subsequent trip of the main boiler feedwater pumps caused a start signal to each motor-driven auxiliary feedwater pump. At the time of the event, the motor-driven auxiliary feedwater pumps were being used to control steam generator water level and were unaffected, and the main boiler feedwater pumps were both isolated. Entergy operators restored level in the 22 steam generator to the normal band and entered the issue into the corrective action program as CR-IP2-05-05252. This LER was reviewed by the inspectors; no findings of significance were identified, and no violation of NRC requirements occurred. This LER is closed.

4OA5 Other Activities

.1 Reactor Pressure Vessel Head And Vessel Head Penetration Nozzles

a. Inspection Scope

The inspector reviewed Entergy's examination activities performed in response to NRC Order, EA-03-009, "Issuance of First Revised NRC Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," dated February 20, 2004. In addition, the inspector reviewed, "Relaxation Request of the First Revised Order on Reactor Vessel Nozzles - Indian Point Unit 2," dated February 27, 2006. The request proposed relaxation from the inspection coverage for the nondestructive examination, using ultrasonic testing techniques of the reactor pressure vessel head penetration nozzles that are limited by a threaded section that is less than the one inch lower boundary limit specified in Section IV, paragraph C.5(b) of the Order. The inspector noted that the NRC staff authorized the proposed relaxation and alternative inspection for the RPV penetration nozzles at Unit 2 for the life of the Order.

The inspector reviewed Entergy's inspection methods to detect evidence of leakage and/or cracking of RPV head penetration (control rod drive mechanism, in-core instrumentation and the vessel head vent) nozzles. The inspector also reviewed the examination procedures to determine whether they provided adequate guidance and acceptance criteria to perform the examinations. Entergy performed a visual examination of selected samples of penetrations to evaluate the integrity of vessel head and penetration intersections to confirm the absence of flaws and boric acid deposits. Also, the inspector observed the liquid penetrant examination (PT) examination of penetration nozzles 71, 83 and 91. Entergy performed appropriate examinations for indications of boric acid leaks from pressure-retaining components above the reactor pressure vessel head. The inspector interviewed examination personnel, data analysts, and engineering personnel; and reviewed training and qualification records to verify that licensee personnel were properly trained to perform the visual inspection of the reactor vessel head and the use of eddy current testing (ECT) and ultrasonic testing (UT) of the head penetrations.

The inspector selected three reactor vessel head nozzle penetrations to observe and evaluate the effectiveness of the ECT and UT test methods to detect flaws and to identify a leak path from a failure in the vicinity of vessel head penetrations shrink fit location. The inspector reviewed the UT and ECT reports and visual data displays for penetrations 77, 86, and 97. This data was compared with the reports obtained from the previous inspection during refueling outage 2R15 in 2002. The inspector observed the reactor vessel head had been cleaned of dirt, debris, minor boron deposits, insulation, significant oxidation, and any material that could adversely affect viewing of all penetrations (360 degrees around the circumference of the nozzle). The inspector determined that the examination procedures used required that anomalies, deficiencies, and discrepancies identified during the examination process be evaluated and documented in accordance with Entergy's corrective action program.

b. Findings

No findings of significance were identified.

.2 (Closed) URI 05000247/2005-05-01 Emergency Diesel Generator Building Flooding

In November 2005, NRC inspectors identified questions associated with the potential vulnerability of the normal and emergency 480 volt power sources to flooding in the EDG building. Approximately 30 to 50 oil absorbing pads of varying sizes were found on the floor underneath all three EDGs. During a postulated flooding event, these pads could be swept into the five building drainage sumps, preventing water from being drained from the building. With the building drains clogged, water level would rise, and potentially impact the availability of the EDGs and normal 480 volt power sources.

Entergy reviewed the condition for past operability of the EDGs and normal 480 volt power sources and determined that sufficient time existed for operations department personnel to isolate a leak of water into the room before affecting vital power systems. The service water system and the fire protection system in the EDG building are analyzed such that they will maintain integrity during a seismic event. Entergy generated a calculation that shows that there is sufficient time to isolate water flowing into the EDG building from an actuation of portions of the fire protection system before the EDGs or normal 480 volt power sources would be negatively impacted.

NRC Region I inspectors performed a review of Entergy's analysis and determined that it was satisfactory. Entergy also posted signs warning personnel not to leave debris on the floor of the EDG building to ensure that debris that could block the EDG building drainage system does not accumulate. The inspector toured the EDG building and did not find any debris in the vicinity of the EDG building drains. This issue is closed.

.3 (Closed) URI 05000247/2004-002-05 Reference Temperature Miscalibration

In February 2004, NRC inspectors identified questions associated with a miscalibration of the reactor coolant reference temperature identified during a reactor downpower. Entergy determined that this was due to an incorrectly calibrated controller which had been adjusted in May 2003. The calibration error resulted in average reactor coolant temperature being high and outside its normal band whenever reactor power was between 0 percent and 82 percent.

Entergy reviewed the condition for past operability and evaluated the potential effects on plant transients analyzed in Chapter 14 of the UFSAR. Entergy determined that there was no impact on initiating event frequency due to delayed steam dump actuation during a runback transient. In addition, a calculation was performed which determined that operation at the elevated reference temperature was acceptable with respect to the licensing basis analysis and the conclusions in the UFSAR remained valid.

The inspectors performed a review of Entergy's analysis and calculation CN-TA-06-32, "IP2 [Indian Point Unit 2] - Evaluation of Tavg [average temperature] Miscalibration at MUR [measurement uncertainty recapture] Conditions," and determined that their analysis was satisfactory. This issue is closed.

.4 Implementation of Temporary Instruction (TI) 2515/165 - Operational Readiness of Off-site Power and Impact on Plant Risk

a. Inspection Scope

The objective of Temporary Instruction (TI) 2515/165, "Operational Readiness of Off-site Power and Impact on Plant Risk," was to gather information to support the assessment of nuclear power plant operational readiness of off-site power systems and impacts on plant risk. The inspector evaluated Entergy's procedures against the specific off-site power, risk assessment, and system grid reliability requirements of TI 2515/165.

The information gathered while completing this TI was forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation on April 3, 2006.

b. Findings

No findings of significance were identified.

.5 Implementation of TI 2515/166 - Pressurized Water Reactor Containment Sump Blockage

a. Inspection Scope

Temporary Instruction (TI) 2515/166 was issued to support the NRC review of licensee activities in response to NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors." The TI required inspectors to verify that the licensee's proposed actions in response to Generic Letter 2004-02 were implemented and programmatically controlled. Additionally, the TI required inspectors to verify the licensing bases had been updated to reflect the changes to the plant.

The inspectors reviewed Entergy's responses to the Generic Letter, the plant design change package for the modifications to the internal recirculation and vapor containment sump strainer assemblies, and a sample of the procedures affected by the modifications to determine if the licensee implemented the plant modifications and procedure changes committed to in the response to the Generic Letter. The inspectors also observed portions of the strainer installations in both the internal recirculation and vapor containment sumps while modification work was in progress to ensure the work was being performed in accordance with plant procedures. In addition, the inspectors reviewed a sample of the post modification test procedures to ensure adequacy of the testing.

The inspectors reviewed the UFSAR and plant Technical Specifications to determine if Entergy updated the licensing bases to reflect the corrective actions taken in response to Generic Letter 2004-02. The inspectors also reviewed design change package ER-04-2-234, "IP2 [Indian Point Unit 2] Recirculation Sump and VC [vapor containment] Sump Strainer Upgrade," to verify that the existing design bases, licensing bases, and performance capability of the system had not been degraded by the modification. During this evaluation, the inspectors reviewed the design inputs, assumptions, selected procedure changes, 10 CFR Part 50.59 documentation, and a sample of associated calculations to verify implementation of the identified Generic Letter corrective actions.

The inspectors noted that the modification to the vapor containment sump, as described in Entergy's response to the Generic Letter, was not fully implemented by the end of the refueling outage. Specifically, the portion of the modification located outside the containment crane wall was not installed. Therefore, TI 2515/166 remains open pending additional inspection of the modification to the vapor containment sump, and verification that the licensing bases has been updated.

b. Findings

No findings of significance were identified.

.6 Groundwater Contamination Investigation

a. Inspection Scope

Continued inspection of the groundwater contamination investigation at Indian Point Energy Center was authorized by the NRC Executive Director of Operations in a Reactor Oversight Process deviation memorandum dated October 31, 2005 (ADAMS accession number ML053010404). Accordingly, an on-site inspection was conducted from June 12 through 16, 2006, to evaluate the latest geo-technical and hydrological data and evaluations of groundwater contamination on-site that were determined and compiled by Entergy and its contractors. This inspection consisted of a review of representative records, observations of activities, interviews with personnel, and independent analyses of water samples. The following observations were made:

On-site Groundwater Flow Model

(1) Entergy is progressing on the development of two current alternative conceptual models (ACM) relative to the observations and measurements of strontium-90 and tritium in site monitoring wells. Based on extensive contaminant sampling and observations of established monitoring wells, including wells associated with the decommissioned Unit 1 and its auxiliary buildings, Entergy and their contractors have initiated activities to depict strontium-90 and tritium plumes. The inspectors verified that these conceptual models considered all available groundwater sampling information, including information developed from sampling of various sumps and locations associated with Units 1, 2, and 3; and other on-site and off-site sampling locations.

The inspectors confirmed that Entergy and its contractors are continuing efforts to develop and refine the site conceptual models as new information becomes available. Entergy plans to test and verify the integrity of these models by the addition of new monitoring wells and eventual tracer-flow tests.

(2) The inspectors reviewed the efforts of Entergy’s contractor, ABS Consulting (ABS), relative to the development of graphic visualizations of the site characterization model that is being created to depict plumes; potential sources; sub-surface ground components such as structures, bedrock formations, soil, fractured rocks; ground water flowpaths; drainage pathways; and groundwater characterizations. These visualizations include fence diagrams, cross-sections, and three-dimensional computer portrayals and animations. Such graphical presentations and visualizations are considered valuable in communicating and displaying information necessary for the understanding of the relationships between the engineered structures and subsurface foundations, and the site hydrology and geo-technical features.

Entergy informed the NRC that its contractors, ABS and GZA GeoEnvironmental, Inc. (GZA), will coordinate the enhancement of these visualizations to include the Inwood Marble fracture characterization, groundwater levels, monitoring well locations, and anthropogenic features to understand the site, the ground water flow paths and connectivity features, and contaminant sources.

(3) The inspectors verified that Entergy is planning to perform at least two dye-tracer tests. The first test is planned to be near the site of the observed leak from the spent fuel pool of Unit 2. Preparations are in progress to prepare monitoring wells for the performance of the test.

As part of the preparation for this type of testing, Entergy intends to establish an on-site laboratory facility to conduct analysis necessary to the performance of the dye-tracer studies. Such arrangement allows more timely and direct analysis of the monitoring well samples required for this testing.

(4) NRC and U.S. Geological Survey geo-technical specialists agreed with the assessment of Entergy's specialists that the underlying Inwood Marble bedrock is highly fractured, weathered, and distressed by blasting. GZA considers the rock to be best represented as a porous medium, particularly with regard to backfill and blasted rock aggregate. The inspectors confirmed that Entergy's geo-technical evaluations will include geophysical logging with acoustic and visual televiewers, flow meters, and physical inspection of recovered rock cores to further characterize the condition of the bedrock and other foundation features. Based on technical discussions with NRC and U.S. Geological Survey representatives, Entergy and its contractors agreed to incorporate additional geophysical logging of selected monitoring wells to enhance the understanding and development of the site conceptual model.

Based on discussions with the inspection team, Entergy and its contractors agreed to apply enhanced methods, as appropriate, to compile and depict the extensive data (i.e., geophysical, hydraulic, core, and tracer test information) that has been and is expected to be acquired as site characterization efforts continue.

(5) The inspectors observed a number of rock cores and noted regions of potential fracture flow, including iron-stained fractures, clay in-filling and weathering features in marble fractures, and indications of previous fault movement. The team recommended that the cores be stored in a more secure manner to avoid damage and be available for reference, as necessary. Entergy agreed to take appropriate action.

(6) In discussions with the inspection team, GZA emphasized that anthropogenic pathways may exist, especially in areas containing large amounts of backfill. GZA commented that there were no accurate records of this backfill during plant construction, but it is likely that large amounts of mixed backfill, especially blasted rock, were placed near the foundations of major structures. This backfill is highly permeable and permits flow as high as 300 feet per day, which is much higher

than that of the rock itself, which ranges from 0.001 feet per day (in most rock), to 22 feet per day where open fractures were encountered. The mean value of all measurements in fractured rock was about 0.5 feet per day.

GZA indicated that large parts of the screen well structures may contain large amounts of backfill. Accordingly, additional wells are being considered to explore this area. Entergy indicated that ground penetrating radar will be applied in an attempt to characterize the extent and type of the backfill in various locations.

The inspection team confirmed that GZA has made effective use of results from pressure transducers in characterizing the permeability and porosity of the rock or backfill. The inspection team also reviewed data that depicted tidal influence on ground water behavior and hydraulic gradients that affect the discharge canal area. The pressure transducers were also useful for determining the effects of rainfall on the site; especially in the understanding of the relationship of the storm drain systems to ground water behavior relative to local precipitation.

(7) The installation of new wells, geophysical surveys of all available wells, examination and analysis of recovered cores, flow-rate studies, pump tests, and tracer tests are expected to enhance the understanding of the site hydrology and the site conceptual model. While visualizations and conceptual modeling is being employed, GZA indicated that it is also considering the application of special computer codes to assist in development of the site characterization.

Dose Assessment

The inspectors reviewed Entergy's dose assessment method which considers the water balance as derived from total rainfall in the Indian Point watershed area, and partitioning groundwater infiltration and storm water. The assessment addresses the approximate flows of groundwater and storm water into various areas of the site. This is the same dose assessment method that was reviewed and documented in Special Inspection Report 05000247/2005-011, but was updated on April 24, 2006, to reflect the most recent monitoring well data acquired since that time.

The selection of well samples was also modified to utilize monitoring wells in closest proximity to the Hudson River or discharge canal, to more accurately represent off-site releases. The highest concentrations of tritium, strontium-90, and nickel-63 radionuclides currently detected in the applicable proximity wells were used in the revised calculation. The inspectors confirmed the total annual dose to the maximum exposed person as 2.5×10^{-3} mrem (total body) and 1.1×10^{-2} mrem (organ, i.e, adult bone). This represents 0.08% and 0.11% of the respective regulatory specifications as expressed in 10 CFR Part 50, Appendix I, and the ODCM. The inspectors verified that the groundwater flow model was reasonable given the current understanding of the site hydrology characteristics and determined that the sample concentrations and dose estimating methods were correct and in accordance with Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

Split Sample Results and Laboratory Quality Control

During this inspection period, the NRC continued to split samples with Entergy in order to verify the quality and accuracy of its analysis of samples collected from various monitoring wells by assessing the comparability of analytical results of a wide range of nuclides, e.g., tritium, strontium-90, gamma emitters, beta emitters, and various transuranics.

The NRC samples were analyzed by the NRC's contract laboratory, the Oak Ridge Institute for Science and Education-Environmental Site Survey and Assessment Program (ORISE/ESSAP) Radioanalytical Laboratory. NRC's assessment of the licensee's sample analysis, based on comparability evaluation in accordance with NRC Inspection Procedure 84750, "Radioactive Waste Treatment, and Effluent and Environmental Monitoring Inspection," Section 03.09, "Confirmatory Measurements Program," demonstrated that Entergy and its contractors are providing radiological analysis comparable to results from independent NRC analysis of the samples from the same source. The NRC ORISE/ESSAP sample results are available in ADAMS, retrievable with the use of accession number ML061880387.

The inspectors also reviewed Entergy's program for the quality assurance of radio-analytical measurements. For measurements made on-site, Entergy maintains an intra-laboratory program and an inter-laboratory program. The intra-laboratory program consists of the use of radioanalytical instrumentation quality control checks, control charts to track instrument performance, sampling and analytical replicates, traceable calibration standards, and spiked samples. The on-site inter-laboratory program consists of participation in a performance evaluation program with an independent outside laboratory. The licensee also conducted internal audits that included the radioanalytical laboratory.

For the offsite laboratories, the licensee submitted both spiked and split samples to the laboratories for analysis to verify analytical ability. Entergy also maintained current copies of the offsite laboratories analytical procedures, and conducted external audits of the offsite laboratories.

Both the NRC contract laboratory (ORISE/ESSAP) and the Entergy's offsite laboratory (Teledyne Brown Engineering, Inc., Knoxville, TN) participated in a performance evaluation program sponsored by the Department of Energy's Radiological and Environmental Sciences Laboratory. The data from this performance evaluation program, "Mixed Analyte Performance Evaluation Program (MAPEP)," is publically available and may be found at: http://www.inl.gov/resl/mapep.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On June 28, 2006, the inspectors presented the inspection results to Mr. Fred Dacimo and other Entergy staff members, who acknowledged the inspection results presented. Entergy did not identify any material as proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

N. Azevedo, Code Program Supervisor
T. Barry, Security Manager
T. Beasley, System Engineer
T. Carson, Maintenance Manager
L. Cerra, Design Engineer
T. Chan, System Engineer
J. Comiotes, Director of Nuclear Safety Assurance
P. Conroy, Licensing Manager
F. Dacimo, Site Vice President
G. Dahl, Licensing Engineer
R. Hansler, Reactor Engineering Supervisor
F. Inzirillo, Emergency Planning Manager
M. Johnson, System Engineer
T. Jones, Licensing Supervisor
J. Kayani, System Engineer
M. Kempski, System Engineer
L. Lee, System Engineering Supervisor
R. Lee, Senior Design Engineer
G. O'Donnell, Indian Point Unit 2 Operations Manager
T. Orlando, Systems Engineering Manager
S. Petrosi, Design Engineering Manager
J. Raffaelle, Design Engineering Supervisor
P. Rubin, General Manager of Plant Operations
P. Schoen, Indian Point Unit 2 Assistant Operations Manager
J. Ventosa, Director of Engineering
A. Vitale, Site Operations Manager
J. Whitney, System Engineer
S. Wilkie, Fire Protection Engineer
A. Williams, Indian Point Unit 2 Operations Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000247/2006003-01	NCV	Inadequate Procedure for Placing RHR Pump Suction Pressure Gauges in Service
05000247/2006003-02	NCV	Failure to Identify Degraded Residual Heat Removal Pump Cell Fire Door

05000247/2006003-03	FIN	Inadequate Corrective Actions for Degradation of Service Water Piping
05000247/2006003-04	NCV	Failure to Follow Plant Procedures for Implementation of Compensatory Measures
05000247/2006003-05	NCV	Inadequate Post-Work Test on 21 Emergency Diesel Generator
05000247/2006003-06	NCV	Inadequate Procedure for Venting the Reactor Vessel Head While Shutdown
05000247/2006003-07	NCV	Failure to Assess the Risk of Maintenance Activities on Valve SI-869A
05000247/2006003-08	NCV	Inadequate Surveillance Test Procedure for Emergency Diesel Generators
05000247/2006003-09	NCV	Failure to Implement Procedural Requirements Associated with Core Support Barrel Replacement
05000247/2006003-10	NCV	Inadequate Survey During Core Barrel Replacement Caused Unintended Exposure
05000247/2006003-11	FIN	Inadequate Procedure for Placing Standby Main Lube Oil Cooler in Service

Closed

05000247/2005005-01	URI	Emergency Diesel Generator Building Flooding
05000247/2004002-05	URI	Reference Temperature Miscalibration
05000247/2005001-01	LER	Technical Specification Prohibited Condition Due to Exceeded the Allowed Completion Time for One Inoperable Train of ECCS Caused by an Inoperable Auxiliary Component Cooling Water Check Valve
05000247/2005002-00	LER	Technical Specification Prohibited Condition Due to Exceeding the Allowed Completion Time for One Inoperable Train of ECCS Caused by an Inoperable Safety Injection Pump
05000247/2005003-00	LER	Automatic Start of Both Motor Driven Auxiliary Feedwater Pumps Due to Trip of 22 Main Feedwater Pump Caused by Low Lube Oil Pressure Due to Inadequate Procedure

05000247/2005004-00	LER	Automatic Start of Both Motor Driven Auxiliary Feedwater Pumps Due to 22 Steam Generator High-High Level Signal Caused by Personnel Error

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

OAP-48, "Seasonal Weather Preparation," Revision 1
2-SOP-11.1, "Ventilation System Operation," Revision 47

Condition Reports

IP2-06-03351
IP2-06-03357

Miscellaneous

IP2-AFW-DBD, "Design Basis Document for Auxiliary Feedwater System," Revision 1
IP2-MS-DBD, "Design Basis Document for Main Steam System," Revision 1

Work Orders

IP2-05-22876
IP2-06-02602

Section 1R04: Equipment Alignment

Procedures

2-COL-21.3, "Steam Generator Water Level and Auxiliary Boiler Feedwater," Revision 28
2-SOP-4.2.1, "Residual Heat Removal System Operation," Revision 59
2-OSP-4.3.1, "Support Procedure - Spent Fuel Pit Cooling," Revision 4

Condition Reports

IP2-04-03089
IP2-04-03253
IP2-04-03389
IP2-04-03664
IP2-04-04793
IP2-04-05476
IP2-04-06150
IP2-04-06448
IP2-04-06741
IP2-05-01748
IP2-05-02340
IP2-05-03608
IP2-05-03947
IP2-06-01984
IP2-06-02256
IP2-06-02339
IP2-06-02400
IP2-06-02714

Drawings

235296, “Flow Diagram - Safety Injection System, Figure 6.2.1,” Revision 63
251783, “Flow Diagram - Residual Heat Removal Pumps, Figure 9.3-1,” Revision 28
9321-F-2019, “Flow Diagram - Boiler Feedwater, Figure 10.2-7,” Revision 112
9321-2722, “Water System Nuclear Steam, Sheet 1,” Revision 112

Miscellaneous

Indian Point 2 Response to Generic Letter 88-17, “Loss of Decay Heat Removal”, January 4, 1989
Indian Point 2 Response to Generic Letter 88-17, “Programmed Enhancement Recommendations”, February 3, 1989

Work Orders

IP2-01-19877	IP2-02-45059	IP2-06-19025
IP2-02-02037	IP2-06-18705	IP2-06-19026

Section 1R05: Fire Protection

Procedures

PT-Q41, “Fire Doors in the PAB,” Revision 6

Condition Reports

IP2-04-00605	IP2-04-05843	IP2-06-00411
IP2-04-05597	IP2-04-04793	IP2-06-02934
IP2-04-01311	IP2-05-00719	

Work Orders

IP2-06-11679	IP2-06-21569

Section 1R07: Heat Sink Performance

Calculations

FMX-00102-00, “EDG JW and LO HXs Performance,” Revision 0
FMX-00295-00, “Tube Plugging Limits for EDG LO and JW Coolers,” Revision 0
IP2-EC-GUIDELINE No 1, “Indian Point Unit 2 Tube Plugging Criteria,” Revision 1
IP3-CALC-MULT-00734, “Minimum Thread Engagement Calculation,” Revision 0
PGI-00087-00, “EDG LO Cooler Sizing,” Revision 0
Technical Manual 1118-1.2, “CCW HX Specification Data Sheet,” Revision 0

Condition Reports

IP2-1998-06103
IP2-2002-02401
IP2-2003-03987
IP2-2004-01328
IP2-2004-01663
IP2-2004-01758
IP2-2004-01781
IP2-2004-01790
IP2-2004-01820
IP2-2004-01836
IP2-2004-02241
IP2-2004-04198
IP2-2004-06150
IP2-2004-06162
IP2-2006-00392
IP2-2006-00544
IP2-2006-02614
IP2-2006-03916*
IP2-2006-03917*
IP2-2006-03929*
IP2-2006-03941*
IP2-2006-03962*
IP2-2006-03964*
IP2-2006-03965*
IP2-2006-03974*

* Identified as a result of this inspection

Drawings

9321-F-2033-78, “Service and Cooling Water,” Revision 78
9321-F-2722-113, “Nuclear Service Water System, Sheet 1,” Revision 113
A209762-66, “Nuclear Service Water System, Sheet 2,” Revision 66

Procedures

0-MS-411, “Torquing of Mechanical Fasteners,” July 28, 2005
2-AOP-CCW-1, “Loss of Component Cooling Water,” Revision 1
2-AOP-SW-1, “Service Water Malfunction,” Revision 3
2-SOP-24.1, “Service Water System Operation,” Revision 56
2-SOP-27.3.1.2, “22 EDG Manual Operation,” Revision 17
2-SOP-4.1.2, “Component Cooling System Operation,” Revision 30
STR-B-001-A, “Zurn Service Water Strainers Maintenance,” Revision 6

Miscellaneous

04-0646-PR-00-RS, Elimination
ConEd Letter, S. Bram to NRC, dated February 2, 1990, Response to Generic Letter 98-13
ConEd Letter, S. Bram to NRC, dated July 24, 1991, Implementation Status of Generic Letter 98-13
IP3-LO-2006-00026, Indian Point Unit 2 Ultimate Heat Sink Self Assessment, March 30, 2006
SEP-SW-001, “Generic Letter 89-13: Service Water Program,” Revision 0

System Health Reports

Unit 2 BOP Heat Exchangers
Unit 2 CCW System, First Quarter 2006
Unit 2 EDG System, First Quarter 2006
Unit 2 Service Water System, First Quarter 2006

Test Results

0-HTX-400-GEN, “Eddy Current Inspection Report of 21CCHX,” February 22, 2005
2-PT-Q026B, “22 Service Water Pump Quarterly Surveillance,” June 26, 2005
PD04248.01, “Eddy Current Inspection Report of EDG 22,” January 29, 1999
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IP2-03-27738
IP2-04-09484
IP2-04-13051
IP2-04-13053
IP2-04-15918
IP2-04-15920
IP2-04-26550
IP2-04-26552
IP2-05-02425
IP2-05-12641
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IP2-99-08133

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ENN-NDE-9.31, “Magnetic Particle Examination (MT) for ASME Section XI,” Revision 0
ENN-NDE-9.41, “Liquid Penetrant Examination (PT) for ASME Section XI,” Revision 0
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IP2-06-02349	IP2-06-02335	IP2-06-02133
IP2-06-01858	IP2-06-02515	IP2-06-02156
IP2-06-01859	IP2-06-02556	IP2-06-02383
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IP29701, “Indication Data Report Ultrasonic Test of CRDM Penetration #97"
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2-AOP-FW-1, “Loss of Main Feedwater,” Revision 7
2-AOP-INST-1, “Instrument/Controller Failures,” Revision 2
2-AOP-SG-1, “Steam Generator Tube Leak,” Revision 4
2-POP-3.1, “Plant Shutdown - Mode 1 to Mode 3,” Revision 51
E-0, “Reactor Trip or Safety Injection,” Revision 47
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0-AOP-SEC-3, “Event Contingency Actions,” Revision 0

2-AOP-FLOOD-1, “Flooding,” Revision 3
2-ARP-14, “PAB Flooding,” Revision 0
2-PT-R7A, “Motor Driven Auxiliary Feed Pumps Full Flow,” Revision 19
2-PT-R22A, “Steam Driven Auxiliary Feed Pump Full Flow,” Revision 13
2-SOP-ESP-1, “Local Equipment Operation and Compensatory Measures,” Revision 1
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PT-Q34A, “22 Auxiliary Boiler Feed Pump Steam Supply Valves,” Revision 4
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9321-F-2722, “ Flow Diagram Service Water System Nuclear Steam Supply Plant Sheet 1 of 2,” Revision 112
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2-PT-R6, “Main Steam Safety Valve Setpoint Determination,” Revision 25
2-SOP-15.1, “Reactor Thermal Power Calculation,” Revision 51

PT-V15B, “OPS Logic Actuation of PORVs PCV-456 and PCV-455C,” Revision 1

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FPX-00063-00, “Replacement of EDG Air Start Motors,” Revision 0
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ENTGIP057-C-303, “Containment Building VC Sump Strainer Sections and Details,” Revision 2
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TA-06-2-063, “Protection of RHR Pumps from potential flooding due to degraded service water line

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2-PT-R2A, “Containment Sump Pumps and Instrumentation,” Revision 15
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2-PT-R11, "Sensitive Leak Rate Test - Type B," Revision 18

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2-NF-220, “Fuel Assembly Categorization,” Revision 0
2-PI-M2, “Containment Building Inspection,” Revision 19
2-POP-1.1, “Plant Restoration - Mode 5 to Mode 3,” Revision 73
2-POP-1.2, “Plant Startup - Mode 3 to Mode 2,” Revision 47
2-POP-1.3, “Plant Startup - Mode 2 to Mode 1,” Revision 72
2-POP-3.3, “Plant Cooldown - Mode 3 to Mode 5,” Revision 68
2-PT-V53D, “Mode Change Checklist - Mode 4 to Mode 3,” Revision 3
2-PT-V53E, “Mode Change Checklist - Mode 3 to Mode 2,” Revision 4
2-REF-003-GEN, Section 2.1, “Manipulator Crane and Fuel Transfer System Checkouts,” Revision 1
2-SOP-1.1, “Filling and Venting Reactor Coolant System,” Revision 52
2-SOP-1.2, “Draining Reactor Coolant System,” Revision 42
2-SOP-1.4.1, “Overpressure Protective System Operation,” Revision 19
2-SOP-4.2.1, “Residual Heat Removal System Operation,” Revision 59
2-SOP-4.2.2, “Operation With Reduced Reactor Coolant System Inventory,” Revision 18
2-SOP-17.31, “Refueling Operation Surveillance,” Revision 27
IP-SMM-OU-102, “Startup Management,” Revision 1
OAP-7, “Containment Entry and Egress,” Revision 6

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IP2-06-18959

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2-PT-10Y1, “Integrated Leak Rate Test,” Revision 1
2-PT-R7A, “Motor Driven Auxiliary Feed Pumps Full Flow,” Revision 19
2-PT-R22A, “Steam Driven Auxiliary Feed Pump Full Flow,” Revision 13
2-PT-R26A, “Local IVSW Test (Water),” Revision 10

2-PT-R26A - DS29, “21 Spray Pump Discharge Valve 869A,” Revision 10
2-PT-Q27B, “23 Auxiliary Feed Pump,” Revision 14
IP-SMM-WM-100, “Work Management Process,” Revision 4
IP-SMM-WM-101, “On-Line Risk Assessment,” Revision 0
2-PT-R014, “Automatic Safety Injection System Electrical Load And Blackout Test,” Revision 21
2-PT-84A,B and C, 21,22 and 23 “EDG 8 Hour Load Test,” Revisions 2,1 and 8
2-PT-R013, “Safety Injection System,” Revision 27

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IP2-04-05718	IP2-06-01989	IP2-06-03085
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Section 1EP6: Emergency Plan Drill

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IP-EP-410, “Protective Action Recommendations,” Revision 3
IP-EP-AD1, “Maintaining Emergency Preparedness,” Revision 1
IP-EP-AD13, “IPEC Emergency Plan Administrative Procedures,” Revision 2

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IP2-2005-03296
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IP2-2005-04105
IP2-2005-04131
IP2-2005-04150
IP2-2005-04152
IP2-2005-04262
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IP2-2006-01241
IP2-2006-01243
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IP2-2006-02005
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IP3-2005-03944
IP3-2005-04010
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IP3-2005-05457
IP3-2006-00432

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IP2-04-06467
IP2-04-06468
IP2-06-01011
IP2-06-01012

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9321-F-2735, "Flow Diagram - Safety Injection System, Sheet 1," Revision 136

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EN-LI-101, “10 CFR 50.59 Review Program,” Revision 2
EN-LI-100, “Process Applicability Determination,” Revision 6
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2-SOP-21.4, “Main Boiler Feed Pump Lube Oil System,” Revision 12 and 13

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Section 4AO5: Other Activities

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2-AOI-4.2.3, “Transfer To Cold Leg Recirc,” Revision 6
2-AOP-13.8KV-1, “Loss of Power to Any 13.8 kV Bus,” Revision 2
2-AOP-138KV-1, “Loss of Power to 6.9 kV Bus 5 and/or 6,” Revision 4
2-ARP-SBF-1, “CCR Safeguards,” Revision 36
2-COL-5.1.2, “Unit 2 Liquid Waste Disposal System,” Revision 10
2-ES-1.3, “Transfer To Cold Leg Recirculation,” Revision 25
2-OSP-1.3, “Support Procedure - Reactor Coolant Pump Startup and Shutdown,” Revision 3
2-PT-R16, “Recirculation Pumps,” Revision 16
2-SOP-5.1.4, “Waste Holdup Tank Inleakage,” Revision 12
2-SOP-10.1.1, “Safety Injection Accumulators and Refueling Water Storage Tank Operations,” Revision 49
2-SOP-27.1.1, “Operation of 345 kV and 138 kV Components,” Revision 18
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ER-04-2-234, “IP2 Recirc Sump and VC Sump Strainer Upgrade,” Revision ERCN IP2-06-21324
IP-SMM-OP-104, “Off-site Power Continuous Monitoring and Notification,” Revision 3
IP-SMM-WM-101, “On-Line Risk Assessment,” Revision 0
OAD-37, “Guidelines for Performing Risk Assessment,” Revision 14

LIST OF ACRONYMS

2R15	Unit 2 15th refueling outage
2R17	Unit 2 17th refueling outage
ABS	ABS Consulting
ACM	alternative conceptual models
ADAMS	agency-wide documents and management system
ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
CAP	corrective action program
CCW	component cooling water
CFR	Code of Federal Regulations
CR	condition report
CRDM	control rod drive mechanism
CVCS	chemical and volume control system
ECT	eddy current testing
EDG	emergency diesel generator
ESSAP	environmental site survey and assessment program
GPR	ground penetrating radar
GZA	GZA GeoEnvironmental, Inc.
IMC	inspection manual chapter

IP2	Indian Point Unit 2
ISI	inservice inspection
kV	kilovolts
kW	kilowatts
LER	licensee event report
LLD	lower limit of detection
MAPEP	mixed analyte performance evaluation program
MT	magnetic particle testing
NCV	non-cited violation
NDE	non-destructive examination
NRC	Nuclear Regulatory Commission
ODCM	off-site dose calculation manual
ORISE	Oak Ridge Institute for Science and Education
PAB	primary auxiliary building
PARS	publically available records systems
PI	performance indicator
PQR	procedure qualification record
PT	liquid penetrant test
QA	quality assurance
RCA	radiologically controlled area
RCS	reactor coolant system
REMP	radiological environmental monitoring program
RHR	residual heat removal
RMS	radiation monitoring system
RPV	reactor pressure vessel
RVLIS	reactor vessel level indicating system
RWP	radiation work permit
SDP	significance determination process
SEMO	State Emergency Management Office
SFP	spent fuel pool
SG	steam generator
SGTL	steam generator tube leak
SGTR	steam generator tube rupture
SSC	systems, structures, and components
Sr-90	strontium-90
TI	Temporary Instruction
TLD	thermoluminescent dosimeter
TS	Technical Specification
USGS	United States Geological Survey
UT	ultrasonic test
VC	vapor containment
WO	work order
WPS	weld procedure specification