May 12, 2005

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

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 2 - NRC INTEGRATED INSPECTION REPORT NO. 05000247/2005002

Dear Mr. Dacimo:

On March 31, 2005 the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Unit 2 (IP2). The enclosed integrated inspection report documents the inspection findings, which were discussed on April 21, 2005, with Mr. Chris Schwarz and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of the inspection, four findings of very low safety significance (Green) were identified. Two of these findings were determined to be violations of NRC requirements. However, because of their very low safety significance, and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCV) consistent with Section VI.A. of the NRC Enforcement Policy. If you contest the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at Indian Point 2.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report No. 05000247/2005002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	05000247/2005002
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Unit 2
Location:	295 Broadway, Suite 3 Buchanan, NY 10511-0308
Dates:	January 1, 2005 - March 31, 2005
Inspectors:	P. Habighorst, Senior Resident Inspector, IP2 M. Cox, Resident Inspector, IP2 J. Schoppy, Senior Project Engineer (Acting), DRP T. Hipschman, Senior Resident Inspector, IP3 P. Finney, Reactor Engineer, DRS J. Noggle, Senior Health Physicist C. Long, Reactor Engineer R. Nimitz, Sr. Health Physicist (In-office review)
Approved by:	Brian J. McDermott, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000247/2005002; 01/01/2005 - 03/31/2005, Indian Point Nuclear Generating Unit 2; Maintenance Risk Assessment and Emergent Work; Fire Protection; Non-Routine Events.

The report covers a 3-month period of inspection by resident inspectors and five regional inspectors. Two Green Non-cited Violations (NCVs), two Green findings, and one licensee identified violation are discussed in this report. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” (SDP). The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green finding associated with ineffective causal analysis for a rod control system problem which resulted in the unexpected insertion of control rod H-8, and power reductions to less than 75 percent, on February 9 and 10. The inspectors determined that the causal analysis was ineffective since it failed to identify that the current traces taken during troubleshooting were ten to fifteen percent below the expected values, even after short-term action to install the original style regulation cards.

  The finding is more than minor since it affected the Barrier Integrity cornerstone objective (fuel cladding). The barrier integrity cornerstone objective provides reasonable assurance that physical design barriers protect the public from radionuclide release caused by accidents or events. This finding impacted the configuration control attribute since it led to the licensee’s inability to maintain the rod alignment criteria prescribed in the Technical Specifications (TS). A Phase 1 SDP screening determined that the inadequate causal analysis and subsequent rod drops were of very low risk significance (Green) since the required actions for rod misalignments prescribed by the TS were performed within the allowed time and in-core flux maps verified that local power limits were met. No violations of NRC requirements were identified. This finding is associated with the cross-cutting area of problem identification and resolution, specifically, an ineffective evaluation of rod control system problems resulted in the unexpected insertion of control rod H-8 and power reductions to less than 75 percent, on February 9 and 10. (Section 1R13)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green non-cited violation of license condition 2.K between November 26, 2004 - March 9, 2005, due to inadequate compensatory actions for a degraded 3-hour rated fire barrier (3M Interam) for penetration H20 concurrent with a degraded hose station nearest to the fire barrier H20. Penetration H20 houses electrical cables needed for the Alternate Safe Shutdown System.

The finding is more than minor since, if left uncorrected, the finding would become a more significant safety concern. The finding affects the Mitigating Systems cornerstone, and its objective of ensuring availability, reliability and capability of systems that respond to initiating events, since both deficiencies contributed to plant risk by decreasing the endurance of the fire barrier and affecting the ability to manually (no automatic suppression capability) fight fires in the electrical penetration room. This issue was of very low risk significance (Green) using phase 1 of the Fire Protection SDP, MC 0612 Appendix F because the barrier was judged to afford greater than 20 minutes of fire endurance protection and low combustible loading was found in the fire area. This finding is associated with the cross-cutting area of human performance (personnel) in that fire protection engineering did not document or implement adequate compensatory measures for the degraded fire barrier and inoperable hose station. (Section 1R05)

- Green. The inspectors identified a Green finding associated with a loss of city water to the primary auxiliary building on January 26, 2005. Specifically, Entergy failed to periodically verify the capability of a backup cooling water supply for the charging pumps, safety injection pumps and the residual heat removal pumps.

  The finding is greater than minor since it affected the Mitigating Systems cornerstone objective of availability of backup cooling to safety pumps in response to a loss of all component cooling water and/or loss of service water event. This finding impacted the procedural quality attribute since no periodic verification existed since 2003 to verify the availability of backup cooling water source, city water. In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the Region I Senior Reactor Analyst (SRA) performed a Phase 3 analysis and determined that this finding was of very low risk significance (Green). No violations of NRC requirements were identified. (Section 1R14)

Cornerstone: Public Radiation Safety

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

  This finding is considered to be more than minor because Entergy failed to meet a waste disposal facility license requirement that was reasonably within its ability to foresee, correct, and prevent. This radioactive material control transportation finding was evaluated against criteria specified in NRC Manual Chapter 0609, Appendix D, and determined to be of very low safety significance (Green) because: 1) no external radiation or contamination limits were exceeded; 2) no package breach was involved; 3) no failure to make a notification was involved;

and 4) although a low-level burial ground non-conformance was involved, burial ground access was not denied and no 10 CFR 61.55 waste classification issue was involved. In addition, although the finding did involve a certificate of compliance issue; the finding was a minor contents deficiency with low risk significance relative to causing a radioactive release to the public or public or occupational exposure. The small quantity of waste material was contained within the NRC approved shipping cask. Entergy temporarily suspended this type of shipment from the Indian Point Energy Center and placed the issue in the corrective action program. (Section 4OA2).

C. Licensee-Identified Violations.

The inspectors reviewed one violation of very low safety significance, which was identified by Entergy. Corrective actions taken or planned by Entergy have been entered into Entergy's Corrective Action Program (CAP). The violation and corrective actions are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Indian Point 2 (IP2) began the inspection period at 100% percent power. On February 9, power was reduced to 73% following the control rod (H-8) insertion into the core. Power ascension was commenced later that day. On February 10, while at 93% power, control rod (H-8) again inserted into the core unexpectedly and required another downpower to 73% (report detail 1R13). Following repairs to the rod control system, operators returned power to 100% on February 12, where it remained for the rest of the quarter.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R01 Adverse Weather Protection (71111.01 - 1 sample of actual adverse weather)

c. Inspection Scope

The inspectors reviewed Entergy procedure OAP-048, Rev. 0, "Seasonal Weather Preparation," and the associated station operating procedures and check-off lists involving cold weather preparations, to verify that these procedures and checklists were completed in accordance with procedural requirements. During the week of January 3, the inspectors performed a risk-informed sample to independently verify that Entergy's actions to assure freeze protection of plant equipment were completed due to the very low ambient temperatures, snow, and icy conditions during that period. The inspectors performed walkdowns of accessible areas of the Unit 2 power plant operating and auxiliary support structures including the auxiliary feedwater building, emergency diesel generators (EDGs), refueling water storage tank, condensate storage tank, and service water (SW) intake structure to assess the adequacy of freeze protection measures. The inspectors also looked for any vulnerable components not previously identified by Entergy.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q - 3 samples)

a. Inspection Scope

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

- On January 19, 2005, the inspector performed a partial system walkdown of Gas Turbine 3 support systems while Gas Turbine 1 was out of service for maintenance associated with the chemical fire suppression system. The inspector reviewed system drawings and COL 31.3 to verify proper alignment of fuel oil, auxiliary power, control switch alignment, battery chargers, and the ansul fire protection system. The inspector also observed the physical condition of the equipment during the verification.

- On March 7, 2005, the inspector performed a partial system walkdown of the essential service water system while the 24 service water pump was out of service for planned maintenance. The inspector reviewed UFSAR section 9.6-1 and UFSAR figure 9.6-1 revision 17D and COL 24.1.2 to verify proper alignment of the essential service water system. The inspector also observed the physical condition of the equipment during the verification.

- On March 10, 2005, the inspector performed a partial system walkdown of the auxiliary feedwater 22 and 23 trains when the 21 auxiliary feedwater pump was out of service for planned maintenance. The inspector reviewed UFSAR drawing 10.2-7 revision 17D and COL 21.3 to verify proper alignment of the auxiliary feedwater trains. The inspector also observed the physical condition of the equipment during the verification.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 8 samples)

a. Inspection Scope

The inspector toured areas that were identified as important to plant safety and risk significance. The inspector consulted the Indian Point 2 Individual Plant Examination for External Events (IPEEE), 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 inspector evaluated conditions related to: 1) control of transient combustibles and ignition sources; 2) the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and 3) the fire barriers used to prevent fire damage or fire propagation. Reference material used by the inspector to determine the acceptability of the observed conditions in the fire zones are referenced in the Supplemental Information attachment at the end of this report. The areas reviewed were:

- Zone 13
- Zone 1A
- Zone 74A
- Zone 74B
- Zone 6A

- Zone 1A
- Zone 2A
- Zone 43A

g. Findings

Introduction. The inspectors identified a Green Non-Cited Violation of license condition 2.K due to inadequate compensatory actions for a degraded 3-hour rated fire barrier (3M Interam) for penetration H20 in fire zone 74B concurrent with a degraded hose station nearest to fire barrier H20. Penetration H20 houses electrical cables needed for the Alternate Safe Shutdown System (ASSS) and the nearest hose station FP-829 was concurrently inoperable. Entergy did not develop or implement an action plan describing interim compensatory measures between November 26, 2004 - March 9, 2005 for multiple inoperable fire protection features that were related to each other.

Description. Addendum I to procedure SAO-703, "Fire Protection Impairment Criteria and Surveillance," provides impairment conditions, required actions, and action times for a 3-hour rated barrier on Alternate Safe Shutdown System conduits and penetration H20 outside containment. Addendum I to SAO-703 requires the fire barrier for penetration H20 to be operable anytime the plant is in MODE 1, 2, 3, or 4. Indian Point Unit 2 entered Mode 4 on November 18, 2004. The 7-day procedural limit for an inoperable fire barrier on penetration H20 (3M Interam E54 Blanket) was exceeded on November 25, 2004. On November 26, 2004, CR IP2-2004-06464 was written stating that penetration H20 was inoperable and gave corrective actions for verifying the fire detectors for fire zone 74A operable within 1 hour and restoring the fire barrier to operable status in 7 days. Step 4.6 of SAO-703 states that, "if a fire protection system or component is impaired for a period time greater than allowed by Addendum I, then a condition report shall be initiated and Fire Protection Engineering shall be responsible for documenting a Plan of Action in response of the CR." According to the procedure, the Plan of Action will include the plans and procedures to be used for restoring the impaired equipment to operable status, the expected return-to-service date, and as applicable, justification for exceeding the Action Time and the recommendation of interim compensatory measures. Entergy closed CR IP2-2004-06464 without the justifications required by SAO-703 and initiated WO IP2-04-33171 to complete the repairs to fire barrier for penetration H20. The fire wrap installation on H20 was completed on March 9, 2005.

On November 4, 2004, hose station FP-829 was declared inoperable with no or inadequate compensatory actions in place. Valve FP-829 could not be cycled during an annual preventative maintenance activity. Hose station FP-829 is the hose station that serves Fire Zones 74A (includes penetration H20), 74B, and 1A. SAO-703 Addendum I allows 14 days to repair hose station FP-829. On November 28, 2004, per SAO-703 step 4.6, Condition Report IP2-2004-06483 was written stating that the 14 days have been exceeded. CR IP2-2004-06483 stated that contingency hoses had been put in place and that the contingency hoses consisted of the hose normally on the reel for FP-829. On March 8, 2005, the inspector identified that additional hoses should be staged at hose station FP-886 and that a deficiency tag should be re-attached to FP-829. In response, Entergy placed two additional 50 ft. hose sections and a 'Y' connector at hose station FP-886 before the end of the shift. Entergy documented this observation in CR IP2-2005-01013.

Entergy's evaluation of these conditions concluded that the combustible loading in fire zone 74A was low, fire detection was available, and manual fire fighting strategies were available though degraded. The inspectors concluded that no interim compensatory actions were documented or taken, such as operator awareness to the implications during Appendix R safe shutdown operations, an assessment in fire brigade strategy changes due to inoperability of hose station FP-829, or no consideration of temporary fire extinguisher placement due to degradation in fire fighting equipment.

<u>Analysis</u>. The performance deficiency involved Entergy's failure to follow procedure requirements regarding evaluations and compensatory measures for degraded fire equipment. Traditional enforcement does not apply because an event did not occur that resulted in an actual safety consequence or impact the NRC's regulatory function, and was not the result of a willful violation of NRC requirements or Entergy procedures. This inspector-identified finding is more than minor since if left uncorrected, the finding would become a more significant safety concern. This finding affects the Mitigating Systems cornerstone and its objective of ensuring availability, reliability, and capability, of systems that respond to initiating events, since both deficiencies contributed to plant risk by decreasing the endurance of the fire barrier and decreasing the ability to manually fight fires in the electrical penetration room (no automatic suppression capability). This issue was determined to be of very low risk significance using Phase 1 of the Fire Protection SDP, MC 0612 Appendix F because the barrier was judged to afford greater than 20 minutes of fire endurance protection and low combustible loading was found in fire area 74A. The primary cause of this finding was associated with the cross-cutting area of human performance (personnel error) in that fire protection engineering did not document or implement an adequate action plan associated with a degraded fire barrier and hose station.

<u>Enforcement</u>. Indian Point Unit 2 operating license condition 2.K states that Entergy Nuclear Operations shall implement and maintain in effect all provisions of the NRC-approved Fire Protection Program Plan as described in the Updated Final Safety Analysis Report. The Indian Point Unit 2 Fire Protection Program Plan was approved by the NRC in a Safety Evaluation Report dated January 31, 1979. The NRC approved the implementation of Section III.G to Appendix R for 10 CFR 50 in a safety evaluation report dated August 22, 1983, which documents that ASSS cables will be protected from other cables. Procedure SAO-703, Rev. 17, "Fire Protection Impairment Criteria and Surveillance" is considered part of the Fire Protection Program Plan under license condition 2.K. Contrary to condition 2.K, Entergy failed to take adequate compensatory actions in accordance with procedure SAO-703. The violation existed for approximately 98 days. Because the failure to implement appropriate interim compensatory measures was entered into Entergy's CAP (reference CR-IP2-2005-01013), this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. The licensee's immediate actions included the placement of additional hoses and a 'Y' connection at FP-886. The licensee's short term corrective actions also installed the remainder of the fire wrap for penetration H20. **(NCV 05000247/2005002-01: Failure to implement adequate interim compensatory measures for fire impairments)**

1R06 Flood Protection Measures (71111.06 - 1 internal sample)

a. Inspection Scope

The inspectors performed a review of major flood sources in the turbine building which could impact the 6.9 kV switchgear. The inspectors selected this sample due to its high contribution to the overall flood induced core damage failure probability. The inspectors reviewed the Final Safety Analysis Report (FSAR) and Individual Plant Examination of External Events (IPEEE) to assess credited mitigating design features. The inspectors then performed an area walk-down to ensure the evaluated design was reflected in the as-built configuration. The condition and adequacy of mitigation equipment was evaluated to assess whether flood protection features were adequate and operable.

The inspectors reviewed abnormal operating procedure AOP-FLOOD-1 to verify operator actions credited in the IPEEE were specified in the procedure and that the procedure could be used to achieve the desired actions. The inspectors reviewed preventive maintenance procedures for flood alarms associated with the lower turbine building elevation and also for rubber boot connections on the circulating water system piping.

b. Findings

No findings of significance were identified.

1R11 Operator Requalification Inspection (71111.11Q - No sample)

Resident Quarterly Review

7. Inspection Scope

The inspectors observed a classroom lecture entitled “On-The-Job-Training and Evaluation” which was presented in accordance with lesson plan JLP-TRNI-OJTTPE. The inspectors evaluated the training environment to ensure it was conducive to learning. The inspectors reviewed the lesson plan to ensure the learning objectives were adequately covered in the body of the lecture. The inspectors also ensured that the lecture was done in accordance with the lesson plans and that the learning objectives were covered in the presentation.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 3 samples)

a. Inspection Scope

The inspectors reviewed the maintenance activities listed below, and recent performance issues with systems and components to assess the effectiveness of Entergy’s Maintenance Rule (MR) program. Using 10 CFR 50.65, “Requirements for

Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," the inspectors verified that Entergy was implementing their MR program in accordance with NRC regulations and guidelines, properly classifying equipment failures, and using the appropriate performance criteria for MR systems in 10 CFR 50.65 (a)(2) status.

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

Main Feedwater System

The inspector performed a review of maintenance issues associated with the main feedwater system between June 18, 2004 - February 18, 2005. The inspector evaluated the MR basis document to determine system boundaries and verified that the system was being properly tracked in accordance with the requirements of 10 CFR 50.65, "Requirements of Monitoring the Effectiveness of Maintenance." The inspector reviewed the Unit 2 Main Feedwater Health Report dated February 23, 2005 and evaluated the system performance monitoring criteria for scope and accuracy. The inspector reviewed CRs for the system and evaluated their proper classification for the MR and compliance with ENN-DC-171, Rev. 0, "Maintenance Rule Monitoring."

Control Room HVAC System

The inspector performed a review of maintenance issues associated with the control room heating and ventilation system between June 18, 2004 - February 18, 2005. The inspector evaluated the MR basis document to determine system boundaries and verified that the system was being properly tracked in accordance with the requirements of 10 CFR 50.65, "Requirements of Monitoring the Effectiveness of Maintenance." The inspector evaluated the system performance monitoring criteria for scope and accuracy. The inspector reviewed CRs for the system and evaluated their proper classification for the MR and compliance with ENN-DC-171, Rev. 0, "Maintenance Rule Monitoring."

Chemical and Volume Control System

The inspector performed a review of maintenance issues associated with the chemical and volume control system (CVCS) since January 2004. The inspector evaluated the MR basis document to determine system boundaries and verified that the system was being properly tracked in accordance with the requirements of 10 CFR 50.65, "Requirements of Monitoring the Effectiveness of Maintenance." The inspector reviewed the quarterly system health inspection report for the 1st quarter of 2004 and evaluated the system performance monitoring criteria for scope and accuracy. The inspector reviewed CRs for the system and evaluated their proper classification for the MR and compliance with ENN-DC-171, Rev. 0, "Maintenance Rule Monitoring."

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 7 samples)

a. Inspection Scope

The inspectors observed selected portions of emergent maintenance work activities to assess Entergy's risk management in accordance with 10 CFR 50.65(a)(4). The inspectors verified that Entergy took the necessary steps to plan and control emergent work activities, to minimize the probability of initiating events, and to maintain the functional capability of mitigating systems. The inspectors observed and/or discussed risk management actions with maintenance and operations personnel. The following planned activities were observed:

- 21 auxiliary boiler feedwater pump (ABFP) preventative maintenance on January 6, 2005.

The following six emergent activities were observed:

- WO-IP2-05-10546, Troubleshoot and repair following 21 Circulating Water Pump (CWP) trip.
- WO-IP2-2005-00004, Troubleshoot and repair following H8 rod drop.
- CR-IP2-2005-00470, Risk assessment and actions associated with identification of gas voids in safety injection system piping. Following the initial assessment this issue was further evaluated during a Safety System Design Inspection conducted during March 2004. This issue will be addressed in inspection report 05000247/ 2005006.
- WO-IP2-10054, Calibration Stator Water Cooling Pressure Switch.
- WO-IP2-13048, Installation of Mechanical Locking Collar on component cooling water (CCW) outlet from Non-Regenerative Heat Exchanger (TCV-130).
- WO-IP2-05-10056, Stator Water Cooling System System Restoration following Water Intrusion into Air System.

b. Findings

Introduction. A self-revealing Green finding was identified due to ineffective causal analysis associated with a fault in the rod control system which led to the unexpected insertion of a single control rod during rod bank motion. This condition required the licensee to reduce power to less than 75 percent on two occasions to comply with their Technical Specifications (TS) associated with rod misalignments.

Description. On November 20, 2004, rod C-7 unexpectedly inserted multiple times during rod withdrawal and subsequent troubleshooting. In addition, rod N-9 which is in the same power cabinet as rod C-7, unexpectedly inserted during rod withdrawal. Entergy established a team to determine the cause of these "rod drops" and was assisted by the vendor. The team concluded that degraded control rod drive mechanism (CRDM) coils in conjunction with new regulation cards installed in the power cabinet during the fall outage were the cause of the dropped rods. It was determined that the new style of regulation cards was more sensitive and less tolerant of the higher resistance associated with the degraded CRDM coils. As a corrective action the original style regulator cards were installed in the power cabinet and the licensee issued work orders to replace the CRDM's for rods C-7 and N-9 during the next refueling outage.

On January 10, 2005, the vendor sent correspondence to Entergy concluding that the most likely cause of the rod drops was due to a higher than normal resistance value on one of the five current feedback resistors. The voltage drop across these resistors is used to control current to the CRDM coils. The voltage drop across each resistor controls current to all the CRDM's based on the resistor with the highest voltage drop. If one of the resistors had a higher resistance, thus higher voltage drop, it would cause all of the CRDM's in the bank to operate with reduced current to the coils. The vendor recommended several courses of action to fix the problem. The licensee reviewed the data from the vendor and determined that their initial assessment, degraded CRDM coils, was still the cause for the rod drops and therefore took no action based on the vendor's recommendations.

During routine rod motion on February 9, 2005, rod H-8 dropped into the fully inserted position in the core. This rod is associated with the same Power Cabinet as rods C-7 and N–9. Reactor power was reduced to less than 75% to comply with the TS action statement associated with misaligned rods. While troubleshooting efforts were ongoing the rod was recovered and aligned with the rest of the rods in its associated bank. On February 10, during power ascension back to 100% full power, rod H-8 again dropped into the fully inserted position.

Following the rod drop on February 10, the licensee implemented one of the corrective actions proposed by the vendor. A modified regulation card was installed in the system. This card was modified to regulate at a higher voltage. This would essentially negate the impact of the high current feedback resistance value and allow the CRDM coils to operate at the design current values. The inspector reviewed current traces to the coils following the modification and found them to be satisfactory. Post work testing was completed satisfactorily and the plant resumed full power operation.

<u>Analysis.</u> The performance deficiency is ineffective causal analysis associated with a rod control fault which resulted in dropping rod H-8 and down powers to less than 75 percent on February 9 and 10. The inspectors determined that the causal analysis in November 2004 was ineffective since it failed to identify that the current traces taken during rod withdrawal were ten to fifteen percent below the expected values, even after installation of the original style regulation cards. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function and the finding was not the result of any willful violation of NRC requirements or Entergy's procedures. The finding was determined to be greater than minor since it affected the Barrier Integrity cornerstone objective, specifically fuel cladding, of providing reasonable assurance that physical design barriers protect the public from radionuclide release caused by accidents or events. This finding impacted the configuration control attribute since it led to the licensee's inability to maintain the rod alignment criteria prescribed in the Technical Specifications.

The TS establishes limits on rod alignment to ensure that the power distribution and reactivity limits defined by the design power peaking are preserved. When the limits are not met the TS actions require thermal power to be reduced to less than 75% so that excessive local linear heat rates caused by the redistribution of flux will not cause the core design criteria to be exceeded and that power distributions that may invalidate safety assumptions do not occur. These limits ensure that departure from nucleate boiling does not occur to prevent damage from excessive heat to the fuel or cladding.

The inspectors conducted a Phase 1 SDP screening and determined that the inadequate causal analysis and subsequent rod drops were of very low risk significance (Green) since the required actions for rod misalignments prescribed by the TS were performed within the allowed time and in core flux maps verified that no local power limits were exceeded. This finding has been placed in Entergy's corrective action program CR-IP2-2005-00568.

This finding is associated with the cross-cutting area of problem identification and resolution in that Entergy personnel did not adequately evaluate the cause of rod control problems in November 2004 or on February 9, 2005, in that, the current traces taken during rod withdrawal were ten to fifteen percent below the expected values and this aspect of the result was not analyzed. (See Section 4OA2).

Enforcement. No violation of regulatory requirements occurred. The inspector determined that the finding did not represent a noncompliance since the inadequate causal analysis occurred on a non-safety related system. **(FIN 05000247/2005002-02: Inadequate corrective actions associated with a rod control failure)**

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

a. Inspection Scope

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

- On January 3, 2005, water was found in control panels and air regulators associated with the main generator's stator water cooling skid (CR-IP2-2005-00020)
- On January 11, 2005, 21 Circulating Water Pump tripped with the 22 circulating water pump out of service (CR-IP2-2005-00123)
- On January 26, 2005, Entergy identified that city water was not available to the primary auxiliary building (CR-IP2-2005-00374)
- On February 9 and 10, 2005, drop of Control Rod H-8 (CR IP2-2005-00568)

b. Findings

Introduction. A self-revealing Green finding was identified associated with a loss of city water to the primary auxiliary building on January 26, 2005. The performance issue was failure to periodically verify the capability of city water backup cooling safety function to the charging pumps, safety injection pumps and the residual heat removal pumps.

Description. During weekly eyewash station testing on the 98 foot elevation of the primary auxiliary building, Entergy identified that city water was not available to the primary auxiliary building and documented this deficiency in condition report (CR-IP2-2005-374). The “city water” system is an on-site potable water supply that has several functions. The last successful eyewash station test in the primary auxiliary building was completed on January 20, 2005. City water was indirectly confirmed through troubleshooting to be available shortly after the unsuccessful eyewash station test. Short-term corrective actions verified city water availability by increased eye wash station testing.

City water comes to the primary auxiliary building via the auxiliary feedwater building, through an 8 inch header underground in the main transformer yard. A 2 inch riser travels outside the wall of the primary auxiliary building and into the primary auxiliary building.

City water provides a safety-function to supply emergency backup cooling to the charging pump lube oil coolers and glycol drive coolers. The city water supply also provides emergency backup cooling to the safety injection pump lube oil and seal coolers and the residual heat removal seal and jacket coolers. The emergency backup is provided to mitigate a complete loss of the normal component cooling water supply to the charging, safety injection and residual heat removal pumps. No actual safety significance existed since component cooling water was available between January 20 and January 26, 2005.

Entergy investigation could not identify the direct cause for the loss of city water to the primary auxiliary building. The most likely apparent cause determination by Entergy was blockage due to internal corrosion products. Entergy did consider other potential sources of blockage such as freezing and a failed component (valve disc separating from the stem). The inspector evaluated the cold weather protection program associated with the 2 inch city water riser section.

The design basis document for City Water section 5.1.1.2 references plant drawing A227781-69 that classifies the supply water to SI, RHR and charging pumps from city water as a safety-related function. Final Safety Analysis Report (FSAR) section 9.6.3 documents backup cooling function for the safety related pump coolers. In 2004, the NRC documented a minor violation (inspection report 05000247/2004003 detail 4OA5) that confirmed completion of an engineering analysis that demonstrated that city water provided an adequate backup to the pump coolers.

At Indian Point Unit 3, in August 1998 flow blockage due to rust and sediment accumulated on the inside of check valve MW-684. This check valve supplies backup cooling to the charging pumps. Blockage occurred on September 11, October 6, and November 10 until the licensee removed the internals of check valve MW-684. Two licensee event reports were prepared for conditions prohibited by TS, since the TS at the time required a verification of backup flow to charging pumps on a monthly basis. The verification of city water to the charging pumps was removed from the TS, however, availability continues to be periodically verified using surveillance procedure 3PT-2Y010, "Emergency City Water to Charging and Boric Acid Transfer Pumps," that implements technical requirements manual 3.1.C.1.9.

Analysis. The performance issue was failure to periodically verify the capability of the city water backup cooling safety function to the charging pumps, safety injection pumps and the residual heat removal pumps. In 2003, Entergy completed an engineering analysis to ensure that city water could provide the backup safety function, however, no actions were taken to periodically verify the capability of this function. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function and the finding was not the result of any willful violation of NRC requirements or Entergy's procedures.

The finding is greater than minor since it affected the Mitigating Systems cornerstone objective of availability of backup cooling to safety pumps in response to a loss of all component cooling water and/or loss of service water event. This finding impacted the procedural quality attribute since no periodic verification existed since 2003 to verify the availability of city water cooling backup.

In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the Phase 1 screen required a Phase 2 analysis because the unavailability of city water back-up to the charging pumps represented a potential loss of a safety function (high pressure injection source). The inspector conducted a Phase 2 Significance Determination Process (SDP) analysis using the following assumptions:

- The unavailability of city water was estimated to be four days (>3 days and less than 30 days);
- Zero (0) mitigation credit was given to the alternate cooling to charging pumps safety function.
- Full mitigation credit for the alternate cooling to safety injection and residual heat removal systems safety function was maintained since several hours were available to perform this function without risk to core damage.
- No operator recovery credit was provided.

SDP worksheets Table 3.11 and 3.12 were used and the sequences involving the affected safety functions were solved. The Counting Rule was applied to the results and the Phase 2 risk approximation yielded a low to moderated safety significance (White) finding. The dominant sequences were:

- LCCW (4) + CWCH1 (0) + CWCH2 (3) = 7
- LCCW (4) + CHCH1 (0) + HPR (3) = 7
- LCCW (4) + CHCH1 (0) + EIHP (3) = 7
- LNSW (4) + CWCH1 (0) + CWCH2 (3) = 7
- LNSW (4) + CHCH1 (0) + HPR (3) = 7
- LNSW (4) + CHCH1 (0) + EIHP (3) = 7

Consistent with IMC 0609, the Region I Senior Reactor Analyst (SRA) performed a Phase 3 analysis to more closely estimate the risk significance of this finding. Using the site specific Spar Model 3.11 and Saphire, the SRA made the following assumptions to evaluate this finding:

- The exposure time used for this assessment was 4 days (96 hours).
- The basic event is operator failure to align city water to the charging pumps.
- The failure probability is the basic event for operator failure to align city water or primary water backup to the SI and RHR systems from $4E^{-3}$ to the base case probability of $6E^{-2}$.
- No additional failures were assumed and routine test and maintenance values were used.

The SRA determined that this finding represented an incremental change in core damage probability of high $1X10^{-7}$ per year, or very low risk significance (Green). The most dominant Phase 3 core damage sequence involved the loss of service water system initiating event and subsequent failure of the reactor coolant pump seals. Consistent with IMC 0609, Appendix A and Appendix H, "Containment Integrity SDP," this finding did not require an external events or large early release frequency (LERF) evaluation. The finding was not significantly impacted by any postulated internal fire scenario and the finding was screened out as a LERF contributor.

Enforcement. No violations of regulatory requirements occurred. Entergy documented the loss of city water in the corrective action program as CR-IP2-2005-374. (**FIN 05000247/2005002-03 Failure to periodically verify the capability of city water backup cooling safety function)** Entergy's planned corrective actions were to verify the adequacy of the design of the heat trace circuit; add a test to verify the operation of the heat trace for the city water; modify appropriate quarterly surveillance tests to verify backup city water flow to the charging pumps, SI, and RHR pumps; and perform a sampling of city water elbows and valves. The inspectors considered the proposed actions reasonable given the possible root causes and the need to periodically verify the capability of the city water supply.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

The inspectors selected operability evaluations that Entergy had generated that warranted review on the basis of potential risk significance. The selected samples are addressed in the CRs listed below. The inspectors assessed the accuracy of the evaluations, the use and control of compensatory measures, if needed, and compliance with the TSs. The inspectors' review included a verification that the operability evaluations were made as specified by procedure ENN-OP-104, "Operability Determinations." The technical adequacy of the evaluations was reviewed and compared to the TSs, Technical Requirements Manual (TRM), FSAR, and associated design basis documents.

- CR IP2-2005-00258, Power Range NI 42 Spike resulting in High Power Channel Trip
- CR IP2-2005-00398, Operability of Safety Injection System due to Nitrogen Gas Intrusion

- CR IP2-2005-00498, UAT/SAT tap changer setting below calculated required value
- CR-2004-6776, Service Water Leaks in Emergency Diesel Generators
- CR IP2-2005-1268, Service Water Leak inlet to 22 Component Cooling Water Heat Exchanger

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16 - 1 sample)

a. Inspection Scope

The inspectors performed a review of operator workarounds and burdens to assess the cumulative effects on system reliability, availability, and the potential for misoperation of a system. The inspectors also toured various areas of the plant to evaluate deficient conditions and their potential impact on operations during Emergency Operating Procedure (EOP) and Abnormal Operating Procedure (AOP) usage. This review included the operator work-around and burden list on March 7, 2005, control room deficiencies list, and system operating procedure SPO-SD-01, “Work Control Process.” In addition, the inspectors reviewed the work control and condition reporting programs to assess the open work request tags and CRs for potential operator workaround consideration. The inspectors used OAP-45, “Operator Burden Program,” Rev. 0 to evaluate plant deficiencies and their effects on plant operation.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 6 samples)

a. Inspection Scope

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

The selected testing activities involved components that were risk significant as identified in the IP2 Individual Plant Examination (IPE). The regulatory references for the inspection included TSs and 10 CFR 50, Appendix B, Criterion XIV, “Inspection, Test, and Operating Status.” The following testing activities were evaluated:

- WO IP2-05-10873, Repair of 22 auxiliary component cooling water pump discharge check valve

- WO IP2-05-11429, Repairs to 23 safety injection pump discharge check valve
- 2-SOP-26.7, Place controller Y-7 to automatic following repairs
- WO IP2-04-35266/CR 2005-1116, Repairs to Foxboro power supply FQ-428A associated with 22 feedwater flow
- WO IP2-05-00558, Service Water Repairs to 22 Component Cooling Water Heat Exchanger
- WO IP2-02-35153, PWT following maintenance on 22 Component Cooling Water Heat Exchanger

1R22 Surveillance Testing (71111.22 - 5 samples)

a. Inspection Scope

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

- 2-PT-Q026E, "25 Service Water Pump," rev. 9 performed on January 5, 2005
- 2-PT-Q030B, "22 Component Cooling Water Pump," rev. 11 on February 7, 2005
- 2-PT-Q29B, "22 Safety Injection Pump," rev. 14 on March 7, 2005
- 2-PT-Q31B, "22 Auxiliary Component Cooling Pump Check Valves" rev. 10 on March 17, 2005.
- PC-EM11-3, "23 AFP Run-Out Protection Instruments" rev. 4 on February 15, 2005.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

The inspectors reviewed a temporary alteration, TA-05-2-027, to install a modified regulation card in Rod Control Power Cabinet 2BD. This card was modified to boost its output by 25% to compensate for a faulty sensing resistor. The increased output of the card would ensure that nominal current was applied to the lift coils of the control rod drive mechanisms supplied by the Power Cabinet. The inspectors reviewed the modification to ensure that there would be no adverse consequences when operating with a single rod on the cabinet and no issues due to interrelations with other components. The inspectors observed the installation and retest, WO IP2-04-34122, to verify the system operated within nominal design parameters once the modification was complete.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6. Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (71121.02 - 6 samples)

a. Inspection Scope

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

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

Indian Point Unit 3 performance of 74 person-rem, ranks in the second quartile of U.S. pressurized water reactors.

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

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

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

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

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

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

b. <u>Findings</u>

No findings of significance were identified.

4. OTHER ACTIVITIES

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

b. Findings

No findings of significance were identified.

2. PI&R Review for IP-71121

a. Inspection Scope

The inspector reviewed 8 CRs that were initiated between December 2004 and January 31, 2005, and were associated with the radiation protection program. The inspector verified that problems identified by these CRs 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.

The inspector reviewed 13 CRs that were initiated between February 01, 2005 and March 20, 2005 and were associated with the radiation protection program. The inspector verified that problems identified by these CRs were properly characterized in the licensee's event reporting system, and that applicable causes and corrective actions were identified commensurate with the safety significance of the radiological occurrences.

b. Findings

No findings of significance were identified.

3. PI&R Annual Sample - 138 KV Voltage Study (71152 - 1 sample)

a. Inspection Scope

The inspectors selected CR-IP2-2004-3473, which identified that the plant was operating at a reactive power level greater than that specified in the design basis voltage analysis. The reactive power level could impact the minimum required off-site voltage. The inspectors reviewed CR-IP2-2004-03473, interviewed personnel, and reviewed associated documents to ensure that the full extent of the issue was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified, prioritized, and implemented. The inspectors evaluated these items against the requirements of Entergy's CAP as delineated in EN-LI-102, "Corrective Action Process."

b. Findings

No findings of significance were identified.

4. PI&R Annual Sample - Barnwell Shipment

a. Inspection Scope (71152 - 1 sample)

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

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

b. Findings

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

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

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

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

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

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

Because this finding was of very low safety significance (Green), and Entergy Indian Point entered this finding into its corrective action program, this violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000247/2005002-04: Failure to transfer waste to a licensed waste disposal facility, in accordance with the provisions of its disposal license)**

5. <u>Cross-Cutting Aspects of Findings (PI&R)</u>

Section 1R13 describes a finding associated with the cross-cutting area of problem identification and resolution. Entergy personnel did not adequately evaluate the cause of rod control problems following the fall refueling outage in November 2004 and on

February 9, 2005, in that, the current traces taken during rod withdrawal were ten to fifteen percent below the expected values and this aspect of the result was not analyzed.

4OA3 Event Followup

5. (Closed) LER 05000247/2004001-00, Manual Reactor Trip Due to Oscillating Feedwater Flow and Steam Generator Level with Flow Perturbations Caused by a Degraded Feed Water Regulating Valve

On September 1, 2004, the operators manually tripped the reactor as a result of oscillating feedwater flow to the 22 steam generator. Entergy determined the cause of the event was a disengaged valve cage on the 22 feedwater regulating valve from the valve body web. A feedwater isolation signal was processed approximately 17 minutes after the manual trip due to 22 steam generator high level from the 22 feedwater regulating valve's failure to close. NRC Inspection report 05000247/2004008, report detail 1R13.1, documented a very low risk significant finding associated with inadequate causal analysis for this event. One new finding associated with this event involves an untimely four hour non-emergency notification in accordance with 10 CFR 50.72(b)(2)(iv)(B). The report was approximately seven hours late. This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement policy. The licensee documented the problem in condition report IP2-CR-2004-04043. This LER is closed.

6. (Closed) LER 05000247/2004002-00, Manual Reactor Trip Due to Decreasing 23 Steam Generator Level Caused by Feedwater Regulating Valve Closure Due to a De-energized Solenoid Operated Valve from Wiring Failure

On September 24, 2004, operators manually tripped the reactor in response to a decreasing 23 steam generator level. The cause of the decreasing steam generator level was a degraded electrical connection for solenoid operated valve FCV-SOV-437-E that caused the associated feedwater regulating valve to close. Corrective actions included replacement of the solenoid and verification of conduit/wiring connection, extent of condition on the remaining three feedwater regulating valves, and engineering review of open modifications involving orientation and configuration of solenoids. NRC Inspection report 05000247/2004008, report detail 1R13.2, documented a very low risk significant finding associated with failure to promptly identify a degraded condition with solenoid valve FCV-SOV-437-E. No new findings were identified in the inspector's review of this LER. The licensee documented the problem in condition report CR-IP2-2004-4522. This LER is closed.

3. (Closed) LER 05000247/2004-003-00, Plant in a Condition Prohibited by Technical Specifications due to Error Making Gaseous Radiation Monitor Inoperable

Entergy identified a violation of the Technical Specifications and reported this condition in accordance with 10 CFR Part 73. The licensee-identified violation is discussed in section 4OA7 of this inspection report. This LER is closed.

4. <u>(Closed) LER 05000247/2004-004-00</u>, Emergency Diesel Generator Actuation from 480 VAC Safety Bus Undervoltage/Blackout Signal Due to Missing Tie Breaker Contact Fingers

On November 9, 2004, two of three emergency diesel generators started during a planned refueling outage evolution to tie 480 VAC safety bus 3A to 480 VAC safety bus 6A. The diesel generators appropriately started in response to an unexpected loss of power to 480 VAC safety bus 6A. The bus lost power because tie breaker 52/3AT6A was not properly reassembled during a prior preventative maintenance activity. Corrective action included installation of the primary contact fingers, revision to the breaker maintenance procedure to include independent verification steps, and lessons learned discussions with maintenance and operations personnel. NRC inspection report 05000247/2004012, report detail 1R13.2, documented a self-revealing Green, non-cited violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings." No new findings were identified in the inspector's review of this LER. The licensee documented the problem in condition report CR-IP2-2004-5927. This LER is closed.

5. <u>(Closed) LER 05000247/2004-005-00</u>, Automatic Reactor Trip Due to Turbine Generator Trip Caused by Low Stator Cooling Water Pressure

On November 26, 2004, an automatic reactor trip occurred due to a main generator/turbine trip resulting from low stator cooling water pressure. The causes of this event were an incorrect pressure switch setpoint due to inadequate testing and setup of the system, inadequate procedure guidance for system start-up, and inappropriately attempting to adjust SCWS control valve. NRC inspection report 05000247/2004012, report detail 1R14.1, documented a self-revealing Green finding involving poor causal analysis associated with the main generator stator water cooling (SWC) system. The ineffective causal analysis was associated with the settings of the generator protection trip pressure switch (63-P79). No new findings were identified in the inspector's review of the LER. The licensee documented the problem in condition report CR-IP2-2004-6467. This LER is closed.

4OA4 <u>Cross-Cutting Aspects of Findings Involving Human Performance</u>

Report section 1R05 documented a finding associated with the cross-cutting area of human performance (personnel) in that fire protection engineering did not document or implement an adequate action plan associated with degraded fire barrier and hose station in the primary auxiliary building.

4OA6 <u>Meetings, including Exit</u>

<u>Exit Meeting Summary</u>

On April 21, 2005, the inspectors presented the inspection results to Mr. C. Schwarz and other Entergy staff members, who acknowledged the inspection results presented. The inspectors asked the licensee what materials examined during the inspection should be considered proprietary. No proprietary information is presented in this report.

4OA7 Licensee-Identified Violation

The following violation of very low safety significance (Green) was identified by Entergy and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, for being dispositioned as an NCV.

- On October 20, 2004, Entergy identified that the containment noble gas radiation monitor alarm setpoint was not consistent with the requirements of TS 3.3.6. Specifically, the alarm setpoint was not adjusted during the implementation of improved Technical Specifications in December 2003. Corrective actions completed by Entergy included reset of the setpoint, extent of condition review of other radiation monitor setpoints as required by TS, TRM, and operational dose control manual, and improvements to the process on controlling radiation monitor setpoints. No new findings were identified in the inspector's review. This finding constitutes a licensee-identified violation of TS 3.3.6. The violation is of very low risk significance since the violation was administrative in nature and the previous setpoints were established to maintain release limits below 10 CFR Part 20. This violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Key Points of Contact

T. Beasley	System Engineer
T. Foley	System Engineer
C. Wend	Radiation Protection Manager
C. Bergeren	In-Service Testing Engineer
J. Boccio	I&C Superintendent
J. Comiotes	Director, Nuclear Safety Assurance
P. Conroy	Manager, Licensing
F. Dacimo	Site Vice President
G. Dahl	Technical Specialist, Licensing
G. Dean	Assistant Operations Manager - Training
R. DeCensi	Technical Support Manager
R. Drake	Supervisor, Mechanical Design Engineering
D. Gately	Assistant Radiation Protection Superintendent
F. Inzirillo	Emergency Planning Manager
T. Jones	Nuclear Safety/Licensing Specialist, Licensing
D. Mayer	Unit 1 Project Manager
B. Meek	System Engineer
V. Myers	Systems Engineering Primary Systems Supervisor
S. Petrosi	Manager, Design Engineering
J. Raffaele	Supervisor, Electrical Design Engineering
P. Rubin	Manager, Site Planning and Outage Services
C. Schwarz	General Manager, Plant Operations
R. Sutton	Systems Engineer
J. Ventosa	Site Operations Manager
C. Wend	Radiation Protection Manager
S. Wilke	Fire Protection Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000247/2005002-01	NCV	Failure to implement adequate interim compensatory measures for fire barrier impairments
05000247/2005002-02	FIN	Inadequate corrective actions associated with a rod control failure
05000247/2005002-03	FIN	Failure to periodically verify the capability of city water backup cooling safety function
05000247/2005002-04	NCV	Entergy IP2 did not properly package radioactive waste for disposal to conform with the waste disposal facility license

Closed

05000247/2004001-00	LER	Manual Reactor Trip Due to Oscillating Feedwater Flow and Steam Generator Level with Flow Perturbations Caused by a Degraded Feed Water Regulating Valve
05000247/2004002-00	LER	Manual Reactor Trip Due to Decreasing 23 Steam Generator Level Caused by Feedwater Regulating Valve Closure Due to a De-energized Solenoid Operated Valve from Wiring Failure
05000247/2004-003-00	LER	Plant in a Condition Prohibited by Technical Specifications due to Error Making Gaseous Radiation Monitor Inoperable
05000247/2004-004-00	LER	Emergency Diesel Generator Actuation from 480 VAC Safety Bus Undervoltage/Blackout Signal Due to Missing Tie Breaker Contact Fingers
05000247/2004-005-00	LER	Automatic Reactor Trip Due to Turbine Generator Trip Caused by Low Stator Cooling Water Pressure

LIST OF BASELINE INSPECTIONS PERFORMED

71111.01	Adverse Weather Protection	1R01
71111.04	Equipment Alignment	1R04
71111.05	Fire Protection	1R05
71111.06	Flood Measures	1R06
71111.11	Operator Requalification Inspection	1R11
71111.12	Maintenance Effectiveness	1R12
71111.13	Maintenance Risk Assessment and Emergent Work Control	1R13
71111.14	Personnel Performance During Non-routine Plant Events	1R14
71111.15	Operability Evaluations	1R15
71111.16	Operator Work Arounds	1R16
71111.19	Post-Maintenance Testing	1R19
71121.02	ALARA Planning and Controls	2OS2
71111.22	Surveillance Testing	1R22
71111.23	Temporary Plant Modifications	1R23
71152	Problem Identification and Resolution	4OA2
71153	Event Followup	4OA3
71154	Cross Cutting Aspects of Findings	4OA4
71122.02	Radioactive Material Processing and Transportation	4OA2

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

2-COL 11.5, Rev. 24, “Space Heating and Winterization”
2-SOP-20.2, Rev. 38, “Condensate System Operation”

Section 1R04: Equipment Alignment

Procedures

COL 31.3, Gas Turbine 3, Revision 5
COL 21.3, Revision 27
COL 24.1.2, Revision 14

Drawings

302783-AA
302776-AA
303222-AA
302775-AA
304122-AA
302777-AA
302774-AA
UFSAR Figure 9.6-1, revision 17D
UFSAR Figure 10.2-7, Revision 17D

Condition Reports

CR-IP2-2004-00862
CR-IP2-2004-1075

Miscellaneous

System Health Report for Gas Turbines - 2nd quarter of 2004
UFSAR Section 9.6.1
UFSAR Section 10.2.6.3

Section 1R05: Fire Protection

Procedures

SMM-DC-901, “IPEC Fire Protection Program Plan,” Rev. 1
ENN-DC-127, “Control of Hot Work and Ignition Sources,” Rev. 1
ENN-DC-161, “Transient Combustible Program,” Rev. 1
Transient Combustible Evaluation No. 04-004, dated 10/21/04
Fire Protection Implementation Plan, Pre-Fire Plans
SOP 29.6.1, Cable Spreading Room Halon Fire Protection System Operation, Revision 9
FP-9, “Control of Combustibles”
FP-8, “Controlling of Ignition Sources”

Fire Protection Implementation Plan, Pre-Fire Plans
SAO - 703, “Fire Protection Impairment Criteria and Surveillance”, revision 17

Miscellaneous

Temporary Alteration 05-2-029, Cable Spreading Room Halon Fire Suppression System
UFSAR Section 9.6.2
Pre-Fire Plan 259, 214, 215
NRC Information Notice 97-48: Inadequate or inappropriate Interim Fire Protection Compensatory Measures
NRC Safety Evaluation Report - Appendix R to 10 CFR 50 Items III.G and III.L (August 22, 1983)

Condition Reports

CR-IP2-2003-2428
CR-IP2-2003-6632
CR IP2-2004-6483
CR IP2-2005-1013
CR IP2-2005-0111
CR IP2-2004-6464

Section 1R12: Maintenance Effectiveness

Condition Reports

CR-IP2-2004-6587
CR-IP2-2004-3669
CR IP2-2004-4096
CR IP2-2004-4145
CR IP2-2004-6472
CR IP2-2004-6515
CR IP2-2004-6512

Work Orders

WO IP2-04-35307
WO IP2-04-35310

Procedures

ENN-DC-143, Revision 0, Unit 2 Main Feedwater Health Report

Miscellaneous

IP2 Feedwater System Unavailability Report dated February 23, 2005
IP CCR-HVAC unavailability Report dated February 17, 2005
Indian Point Nuclear Generating Station Unit 2 Maintenance Rule Basis Document Main Feedwater System, Revision 2

Indian Point Nuclear Generating Station Unit 2 Maintenance Rule Basis Document Heating Ventilation & Air Conditioning for the Central Control Room, revision 2

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Procedures

2-AOP-CCW-1, Loss of CCW, revision 2
2-ARP-SGF, "CCW Discharge Low"
2-OSP-4.1.2, Support Procedure - Component Cooling System Operation, Revision 3
OAP-1, Conduct of Operations, revision 5

Work Orders

IP2-05-13048, Verification of TCV-130 Operating Range (Maximum Flow Rate)

Condition Reports

CR-IP2-2005-868
CR-IP2-2005-567
CR-IP2-2005-1179
CR-IP2-2005-1180

Miscellaneous

Risk-Informed Inspection Notebook For Indian Point Nuclear Generating Station Unit 2, Revision 1
Design Basis Document for Component Cooling Water System, Revision 0
System Description for Component Cooling Water System, Revision 6
ER IP2-05-13407, Hydraulic Flow Calculation
ODMI, Gas In Safety Injection System

Section 1R14: Operator Performance During Non-Routine Evolutions

Condition Reports

CR-IP2-2005-00020
CR-IP2-2005-00034

Drawings

9321-F-2036
OA 94-04, Revision 1, Station Air/Instrument Air

Section 1R15: Operability Evaluations

Calculations

IP-CAL-04-1760

Condition Reports

CR-IP2-2004-06777
CR-IP2-2004-06776
CR-IP2-2004-06741
CR-IP2-2004-6715

Miscellaneous

Action Plan Inspection of Service Water Piping Weld Unit 2 - Service Water Supply Piping to Emergency Diesel Generators

Section 1R16, Operator Work Arounds

Condition Reports

IP2-2004-6126
IP2-2004-0858

Work Orders

IP2-04-10107

Procedures

OAP-045, Revision 0
SPO-SD-01, Work Control Process
Online Operator Work Arounds and Operator Burdens for IPEC, March 7, 2005

Section 1R19, Post Maintenance Testing

Work Orders

WO IP2-05-10873, Repair of 22 auxiliary component cooling water pump discharge check valve
WO IP2-05-11429, Repairs to 23 safety injection pump discharge check valve
2-SOP-26.7, Place controller Y-7 to automatic following repairs
WO IP2-04-35266/CR 2005-1116, Repairs to Foxboro power supply FQ-428A associated with 22 feedwater flow (3/17/05)
WO IP2-05-00558, Service Water Repairs to 22 Component Cooling Water Heat Exchanger
WO IP2-02-35153, PWT following maintenance on 22 Component Cooling Water Heat Exchanger
WO IP2-05-14175
WO IP2-05-00558

Miscellaneous

Ultrasonic Examination Report 05UT114
Ultrasonic Examination Report 05UT108
1992 ASME Code Section XI Division 1, Article IWA-5000, System Pressure Tests

Section 1R22: Surveillance Testing

Procedures

2-PT-Q30B, 22 Component Cooling Water Pump, Revision 11
CEP-IST-4, Standard on Inservice Testing
ASME Section XI OM-6 tables 3a and 3b
2-PT-Q29B, 22 Safety Injection Pump, revision 14
Indian Point Nuclear Generating Station Unit No. 2 Inservice Testing Program Summary for the interval July 1,1994, through May 18, 2005, Revision 2

Condition Reports

CR IP2-2003-3277
CR IP2-2004-6705
CR IP2-2004-5533
CR IP2-2005-920
CR IP2-2004-1517
CR IP2-2005-996
CR IP2-2001-00477

Miscellaneous

Pre-Job Briefing Sheet PT-Q30B, 22 Component Cooling Water Pump

Drawings

9321-F-2735

Section 1EP6: Emergency Plan Drill

Procedures

IP-EP-120, “Emergency Classifications,” Rev. 0
IP-EP-130, “Emergency Notification and Mobilization,” Rev. 2
IP-EP-430, “Site Assembly, Accountability and Relocation of Personnel Offsite,” Rev. 1
IP-EP-220, “Technical Support Center,” Rev. 0
IP-EP-250, “Emergency Operations Facility,” Rev. 3
IP-EP-230, “Operations Support Center,” Rev. 0

Condition Reports

CR-IP2-2005-0122, -0147,-0148,-0150, -0151, -0152, -0153, -0164, -0207, -0208

Miscellaneous

Entergy Nuclear Northeast Indian Point Energy Center, Unit 2, January 12, 2005 Drill Scenario

<u>Section 40A2: Problem Identification and Resolution</u>

<u>Condition Reports</u>

CR-IP3-2004-3114	CR-IP2-2004-5906	CR-IP2-2004-4322
CR-IP2-2004-6271	CR-IP2-2004-5164	CR-IP2-2004-6219
CR-IP2-2004-5179	CR-IP2-2004-6632	

LIST OF ACRONYMS

ABFP	auxiliary boiler feedwater pump
ALARA	as low as reasonably achievable
AOF	abnormal operating procedure
ASSS	alternate safe shutdown system
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
EDG	emergency diesel generator
EOF	Emergency Operations Facility
EOP	emergency operating procedure
EP	emergency preparedness
FSAR	final safety analysis report
IMC	inspection manual chapter
IP2	Indian Point 2
IPE	individual plant examination
IPEC	Indian Point Energy Center
IPEEE	individual plant examination of external events
LERF	large early release frequency
MR	maintenance rule
NCV	non cited violation
PWT	post work test
RTD	resistance temperature detector
SDP	significance determination process
SRA	Senior Reactor Analyst
SW	service water
SWC	stator water cooling
TS	technical specification
TSC	Technical Support Center
WO	work order