UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406 1415

August 8, 2007

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

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000286/2007003

Dear Mr. Dacimo:

On June 30, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Unit 3. The enclosed integrated inspection report documents the inspection results, which were discussed on July 13, 2007, with Mr. Anthony Vitale and other members of your staff.

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

Based on the results of this inspection, one inspection finding of very low safety significance (Green) was identified. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. The NRC is treating this violation as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy because of the very low safety significance and because it is entered into your corrective action program. If you contest this non-cited violation, 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 Nuclear Generating Unit 3.

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

NRC Public Document Room or from the Publicly Available Records (PARS) component of 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 by Donald E. Jackson For/

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

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

Enclosure: Inspection Report No. 05000286/2007003
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-286
License No.:	DPR-64
Report No.:	05000286/2007003
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	Indian Point Nuclear Generating Unit 3
Location:	450 Broadway, GSB Buchanan, NY 10511-0249
Dates:	April 1, 2007 through June 30, 2007
Inspectors:	D. Jackson, Senior Resident Inspector, IP3 B. Wittick, Resident Inspector, IP3 M. Cox, Senior Resident Inspector, IP2 G. Bowman, Resident Inspector, IP2 S. Barr, Senior Emergency Preparedness Specialist, Region I J. Noggle, Senior Health Physicist, Region I T. Nicholson, Nuclear Regulatory Research R. Cureton, Emergency Preparedness Specialist, Region I T. Walker, Senior Project Engineer, Region I B. Haagensen, Operations Engineer, Region I B. Welling, Senior Project Engineer, Region I T. Wingfield, Resident Inspector, Region I G. Smith, Senior Security Inspector, Region I J. Teator, Senior Investigator, Region I
Others:	J. Williams, Groundwater Specialist, U.S. Geological Survey
Approved by:	Eugene W. Cobey, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000286/2007-003; 04/01/2007 - 06/30/2007; Indian Point Nuclear Generating Unit 3; Event Followup.

The report covered a three-month period of inspection by resident and region-based inspectors. One Green finding was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process (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 4, dated December 2006.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. The inspectors identified a finding of very low safety significance (Green), in that, Entergy failed to identify in the corrective action program an adverse condition associated with the 'B' phase high voltage bushing on the 31 main transformer (MT) that was discovered during testing. The data from that test indicated potential degradation of the 'B' phase high voltage bushing. As a result, this condition was not adequately evaluated before placing the transformer back in service, and the bushing subsequently failed. The transformer failure was entered into their corrective action program. Entergy replaced the 31 main transformer and conducted a root cause analysis associated with the failure.

The inspectors determined that this finding was more than minor because it is associated with the equipment performance 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 as well as power operations. Specifically, Entergy did not place this issue in the corrective action process, and as a result, did not conduct an adequate evaluation of a degraded condition associated with the 'B' phase high voltage bushing on 31 MT. Subsequently, the bushing failed during power operation and resulted in a reactor trip, an explosion in the transformer yard, and the declaration of a notification of an unusual event. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This finding was determined to be of very low safety significance because, while it was a transient initiator that resulted in a reactor trip, it did not contribute to the likelihood that mitigation equipment or functions would not be available.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution, because Entergy failed to promptly identify an adverse condition in the corrective action program in a timely manner commensurate with its safety significance. (Section 4OA3)

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and Entergy's actions are described in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Indian Point Nuclear Generating Unit 3 began the inspection period returning to full power after completion of refueling outage 3R14. On April 3, 2007, operators initiated a manual reactor trip due to a loss of speed control of the only operating main boiler feed pump. Entergy returned the unit to power on April 4, 2007. On April 6, 2007, during power ascension with the unit at approximately 91 percent power, the unit tripped automatically as a result of a main generator lockout and main turbine trip. The cause of the event was the failure of the 'B' phase high voltage bushing on the 31 main transformer. As a result of this event, a notification of an unusual event (UE) was declared due to the report of an explosion associated with the bushing failure. Following repair activities on the main transformer, Entergy returned the plant to full power on May 5, 2007, and continued to operate the plant 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)

a. Inspection Scope

The inspectors reviewed Entergy's adverse weather procedures, operating experience, corrective action program (CAP), Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), operating procedures, and applicable plant documents to determine the types of adverse weather challenges to which the site is susceptible.

The inspectors performed plant walkdowns and reviews to verify that plant features and procedures for operation and continued availability of the ultimate heat sink during adverse weather were appropriate, including equipment availability for performance of the reactor shutdown function under the weather conditions assumed prior to shutdown. The intake structure, fire suppression system, and control building ventilation system are risk-significant systems that are required to be protected from adverse weather conditions and were selected for inspection. The documents reviewed during this inspection are listed in the Attachment. Collectively this inspection represented one inspection sample of risk-significant systems.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

Partial Walkdown (71111.04Q - 4 samples)

a. Inspection Scope

The inspectors performed four 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 UFSAR, and system drawings 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. The documents reviewed during this inspection are listed in the Attachment. The inspectors performed partial walkdowns of the following systems, which represented four inspection samples:

- Diesel driven fire pump and motor driven fire pump following maintenance and testing;
- Containment spray system following maintenance and testing;
- Auxiliary feedwater system following replacement of 31 auxiliary boiler feedwater pump minimum flow throttle valve (BFD-33), and 33 auxiliary boiler feedwater pump minimum flow throttle valve (BFD-35); and
- 32 emergency diesel generator (EDG) with 33 emergency diesel generator out of service.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 10 samples)

a. Inspection Scope

The inspectors conducted tours of the ten areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that combustibles and ignition sources were controlled in accordance with Entergy's administrative procedures; fire detection and suppression equipment was available for use; passive fire barriers were maintained; and compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with Entergy's fire plan. The inspectors used procedure ENN-DC-161, "Transient Combustible Program," in performing the inspection. The inspectors evaluated the fire protection program against the requirements of License Condition 2.H. The documents reviewed during this inspection are listed in the Attachment. This inspection satisfied ten inspection samples of fire protection tours.

The areas inspected included:

- Fire Zones 1, 1A, 2, 2A, 58A;
- Fire Zone 264;
- Fire Zone 265;
- Fire Zone 390;
- Fire Zone 14;
- Fire Zones 5, 6, 7, 8, 17A, 18A, 19A, 20A;
- Fire Zones 3, 4, 8A, 9A, 10A, 11A, 12A, 15A, 16A;
- Fire Zones 26A, 27A, 28A, 29A, 30A;
- Fire Zones 5A, 61A, 62A, and 68A; and
- Fire Zones 90A, 91A.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

a. Inspection Scope

The inspectors reviewed the Indian Point Unit 3 Individual Plant Examination (IPE) of External Events and the UFSAR concerning external flooding events. The inspection included a walkdown of accessible areas of the plant to detect potential susceptibilities to external flooding and to verify the assumptions included in the site's external flooding analysis. The inspectors also reviewed relevant abnormal operating and emergency plan procedures. This inspection was conducted during a period of severe weather in mid-April 2007. The documents reviewed during this inspection are listed in the Attachment. This inspection represented one inspection sample of external flood protection.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On May 14, 2007, the inspectors observed licensed operator simulator training to assess operator performance during several scenarios to verify that operator performance was adequate and evaluators were identifying and documenting crew performance problems. The inspectors evaluated the performance of risk significant operator actions, including the use of emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, the implementation of appropriate actions in response to alarms, the performance of timely control board operation and accurate control manipulation, and the oversight and direction provided by the shift manager. The inspectors also reviewed simulator fidelity with respect to the actual plant. Licensed operator training was evaluated against the requirements of 10 CFR 55, “Operator's Licenses.” The documents reviewed are listed in the Attachment. This observation of operator simulator training represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 1 sample)

a. Inspection Scope

The inspectors reviewed performance-based problems involving selected structures, systems, or components (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on:

- Proper Maintenance Rule scoping;
- Characterization of reliability issues;
- Changing system and component unavailability;
- 10 CFR 50.65 (a)(1) and (a)(2) classifications;
- Identifying and addressing common cause failures;
- Trending of system flow and temperature values;
- Appropriateness of performance criteria for SSCs classified (a)(2); and
- Adequacy of goals and corrective actions for SSCs classified (a)(1).

The inspectors reviewed system health reports, maintenance backlogs, and Maintenance Rule basis documents. The inspectors evaluated the maintenance program against the requirements of 10 CFR 50.65. The documents reviewed during this inspection are listed in the Attachment. The following maintenance rule sample was reviewed and represented one inspection sample:

- Vapor containment pressure relief system.

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 reviewed planned or emergent activities to verify that the appropriate risk assessments were performed prior to removing equipment from service for planned work. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The documents reviewed during this inspection are listed in the Attachment.

The following four emergent activities and one planned activity were observed and treated as five inspection samples:

- Main generator voltage regulator repair, including turbine generator shut down and start up operations;
- 38 service water pump planned maintenance;
- Boric acid flow to the charging pump suction at lowered volume control tank pressures troubleshooting and repair activities;
- 32 central control room (CCR) air conditioning unit out of service during mode change for startup; and
- Circulating water pump standby drive maintenance following trip of 32, 34 and 36 circulating water pumps.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 4 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to assess the acceptability of the evaluations, the use and control of compensatory measures, and compliance with Technical Specifications. The inspectors' review included a verification that the operability determinations were made as specified by ENN-OP-104, "Operability Determinations." The technical adequacy of the determinations was reviewed and compared to the TS, UFSAR, and associated design basis documents. The documents reviewed during this inspection are listed in the Attachment. The following evaluations were reviewed and represented four inspection samples:

- Condition report IP3-2007-02059, EDG valve FCV-1176A failed stroke time test;
- Condition report IP3-2007-02442, BFD-FCV-406D, "33 Auxiliary Boiler Feedwater Pump to 34 Steam Generator Control Valve," will not operate with local regulator controls;
- Condition report IP3-2007-02623, Scaffolding interference with SI-MOV-866B that would have contacted the stem position indicator; and
- Condition report IP3-2007-02724, Small residual heat removal system gas void found during 3-PT-M108.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 8 samples)

a. Inspection Scope

The inspectors reviewed post-maintenance test procedures and associated testing activities for selected risk-significant mitigating systems to assess whether the effect of maintenance on plant systems was adequately addressed by control room and engineering personnel. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with design basis documentation; test instrumentation had current calibrations and the range and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon completion, the inspectors verified that equipment was returned to the proper alignment necessary to perform its safety function. Post-maintenance testing was evaluated against the requirements of 10 CFR 50, Appendix B, Criterion XI, “Test Control.” The documents reviewed during this inspection are listed in the Attachment. The following post-maintenance test activities were reviewed and represented eight inspection program samples:

- Work order IP3-07-19935 and WO IP3-07-19744, 31 and 33 auxiliary boiler feed pump minimum flow throttle valves after replacement;
- Work order IP3-07-20519, Main generator voltage regulator 15 volt power supply after replacement;
- Work order IP3-07-12275, Post work test for SI-MOV-866B after 6-year major planned maintenance;
- Work order IP3-06-22068, Post work test for 31 component cooling water pump mechanical seal replacement;
- Work order IP3-05-22763, Post work test for 31 charging pump discharge check valve repair;
- Work order IP3-06-15512, Post work test for R-11 containment radiation monitor pump replacement;
- Work order IP3-06-17569, IP3-06-17577 and IP3-06-23098, Post work test for 33 EDG following planned maintenance; and
- Work order IP3-07-19329, Post work test for 31 main transformer deluge system after transformer replacement.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20 - 2 samples)

a. Inspection Scope

The inspectors observed plant start up activities, including the approach to criticality associated with two forced outages during the inspection period. In addition, the inspectors observed the main generator synchronization to the electrical grid, and initial power ascension. The documents reviewed during this inspection are listed in the

Attachment. The combined efforts described above represent two inspection program samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 5 samples)

a. Inspection Scope

The inspectors witnessed performance of surveillance tests and/or reviewed test data of selected risk-significant SSCs to assess whether the SSCs satisfied TS, UFSAR, Technical Requirements Manual, and Entergy procedure requirements. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with design basis documentation; test instrumentation had current calibrations and the range and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon surveillance test completion, the inspectors verified that equipment was returned to the status specified to perform its safety function. The inspectors evaluated the surveillance tests against the requirements in TS. The documents reviewed during this inspection are listed in the Attachment. The following surveillance tests were reviewed and represented five inspection samples (one RCS leakage rate sample, one inservice testing sample, and three other surveillance test samples):

- 3-PT-CS-004, "Low Head Injection, Accumulator & Residual Heat Removal Valve Test," Revision 19;
- 3-PT-Q83, "RWST Level Instrument Check and Calibration (LIC-921)," Revision 25;
- 0-SOP-LEAKRATE-001, "Reactor Coolant System Leakrate Surveillance, Evaluation, and Leak Identification," Revision 00;
- 3-PT-Q132, "Emergency Boration Flow Path Valve CH-MOV-333," Revision 2; and
- 3-PT-Q062A, "31 Charging Pump Operability Test," Revision 8.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness (EP)

1EP2 Alert and Notification System Evaluation (71114.02 - 1 sample)

a. Inspection Scope

Region-based specialist inspectors reviewed Entergy's corrective actions related to the existing Indian Point alert and notification system (ANS) failures, and reviewed the progress made in the design and installation of the new siren system. Inspection activities were conducted onsite periodically between April 12 and June 28, 2007. This inspection was conducted in accordance with the baseline inspection program deviation authorized by the NRC Executive Director for Operations (EDO) in a memorandum dated October 31, 2005, and renewed by the EDO in a memorandum dated December 11, 2006.

A new ANS is being installed around the Indian Point Energy Center to satisfy commitments documented in an NRC Confirmatory Order (dated January 31, 2006) that implements the requirements outlined in the 2005 Energy Policy Act. In January 2007, Entergy requested an extension of the deadline for completing the ANS project as described in the Confirmatory Order. The Confirmatory Order set a January 30, 2007, deadline for completing installation. Entergy's extension request cited several issues that were beyond their control as the basis for the delay. On January 23, 2007, the NRC granted Entergy's extension request and established April 15, 2007, as the new installation completion date. Entergy conducted a full-system demonstration test of the new ANS on April 12, 2007, and the results of that test failed to meet the acceptance criteria for the new system. On April 13, 2007, Entergy requested another extension which was subsequently denied. On April 23, 2007, the NRC issued a Notice of Violation and civil penalty for Entergy's failure to comply with the siren operability date in the Confirmatory Order.

The inspectors conducted the following onsite inspection activities during this quarter.

- The inspectors observed the full-volume sounding on April 12, 2007 to meet the April 15, 2007 deadline.

- The inspectors reviewed supplemental bench testing done by Entergy's vendor to verify test results from the degraded battery voltage testing performed in the previous quarter.

- The inspectors observed and inspected the degraded voltage re-test of one of the back-up batteries for the new ANS system. The re-test was done because during the first test there was a problem with the resistive load used for the simulated activation. This testing conducted from May 29, 2007 to June 6, 2007 assured that the battery at the siren would operate at its end-of-life condition after having lost alternating current power for 24 hours.

- During all onsite siren inspection activities, the regional inspectors also reviewed the status of and corrective actions for the current ANS to assure that Entergy was appropriately maintaining the system, including the quarterly full-system growl test of the current ANS conducted on June 28, 2007 to demonstrate its functionality.

b. <u>Findings</u>

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 May 16, 2007. 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 inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151- 2 samples)

a. Inspection Scope

The inspectors reviewed performance indicator (PI) data for the below-listed cornerstones and used Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 4, to verify individual PI accuracy and completeness.

Initiating Events Cornerstone

- Unplanned Scrams With Loss of Normal Heat Removal

Mitigating Systems Cornerstone

- Safety System Functional Failures

The inspectors reviewed data and plant records from April 2006 to March 2007. The records reviewed included PI data summary reports, licensee event reports, operator narrative logs, maintenance rule records, maintenance records and condition reports for affected systems. The inspectors verified the accuracy of the data reported, and interviewed licensee personnel associated with the PI data collection and evaluation.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Routine Problem Identification and Resolution (PI&R) Program Review

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of all items entered into Entergy's CAP. The review was accomplished by accessing Entergy's computerized database for condition reports (CRs) and attending CR screening meetings.

In accordance with the baseline inspection procedure, the inspectors selected CAP items across the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones for additional follow-up and review. The inspectors assessed Entergy's threshold for problem identification, the adequacy of the cause analyses, extent of condition review, operability determinations, and the timeliness of the specified corrective actions. The CRs reviewed are listed in the Attachment.

b. Findings and Observations

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 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 2007 to assess the total number and significance of condition reports written in various subject areas, such as equipment or processes, 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 2006 and the first quarter of 2007 to ensure they were appropriately evaluating and trending identified conditions.

b. Assessment and Observations

No findings of significance were identified.

The inspectors determined that Entergy was appropriately identifying and evaluating trends in identified conditions.

.3 Fitness-For-Duty (FFD) Program (71152)

a. Inspection Scope

The inspector reviewed the actions taken by Entergy in response to an employee displaying unusual behavior. The actions taken by the employee's supervisor and the Fitness-for-Duty personnel in the Medical Department were reviewed along with Entergy's FFD policies and procedures.

b. Findings and Observations

No findings of significance were identified. The inspectors determined that Entergy took appropriate actions in accordance with applicable NRC regulatory requirements and internal FFD policies and procedures.

.4 Annual PI&R Sample Review: Control Room Air Conditioning Unit Performance Issues (71152 - 1 sample)

a. Inspection Scope

The inspectors conducted reviews of problems associated with the performance of the 31 and 32 control room air conditioning units, and the placement of the 32 control room air conditioning unit in a 10CFR50.65 a(1) monitoring status. The inspectors interviewed engineers responsible for the system, reviewed applicable condition reports from 2005 to present, and reviewed the associated engineering evaluations and corrective actions. The documents reviewed during the inspection are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified. The inspectors determined that Entergy's threshold for problem identification was appropriate, and associated causal analyses, extent of condition reviews, and corrective actions were adequate.

.5 Annual Sample: Safety Conscious Work Environment Corrective Actions (71152 - Unit 2: 1 sample, Unit 3: 1 sample)

a. Inspection Scope

On December 21, 2006, the NRC issued a letter [ADAMS Ref. ML063560335] requesting that Entergy provide its plan for evaluating a potential chilling effect onsite and its plan of action for addressing the matter to the NRC. This letter and its enclosure documented the results of problem identification and resolution (PI&R) team inspections at the Indian Point Energy Center (IPEC). The letter stated that the NRC had become aware of incidents where workers perceived that individuals were treated negatively by management for raising issues. As a result of these incidents, some workers expressed reluctance to raise issues under certain circumstances. While most workers made a distinction between nuclear safety issues and other concerns, the teams noted that some of the illustrative examples provided by plant workers could have nuclear safety implications. However, the teams did not identify any more than minor issues which had not been raised. The teams also noted that Entergy had not fully evaluated the results of a 2006 safety culture assessment to understand the causes of negative responses and declining trends related to the safety conscious work environment onsite.

Entergy responded in a letter dated January 22, 2007 [ADAMS Ref. ML070240242]. Based primarily on the results of interviews conducted by an independent assessment team, Entergy reported that a "perception exists within a segment of the IPEC workforce that they may suffer in some way if they were to raise a safety concern." The results of the interviews were consistent with NRC's observations during PI&R inspections and generally consistent with the results of the independent safety culture assessment.

Entergy's letter provided a plan with actions intended to improve the safety conscious work environment (SCWE). Specifically, the plan included corrective actions to improve communications; identify and prevent retaliation, chilling effect, and the perception of retaliation; enhance the corrective action program; enhance the employee concerns program; and improve the broader work environment at IPEC. Entergy also indicated that metrics would be developed to measure performance at achieving the components of a healthy SCWE and an assessment would be conducted to confirm the effectiveness of its actions in early 2008.

The NRC reviewed Entergy’s response and concluded that Entergy’s completed and planned diagnostic activities were reasonable to characterize the challenges to the safety conscious work environment onsite and the planned corrective actions were appropriate. The results of the NRC’s review were documented in a letter to Entergy dated February 26, 2007 [ADAMS Ref. ML070570518]. This letter also stated that the NRC would monitor Entergy's corrective actions through baseline inspection activities.

In June 2007, the inspectors performed PI&R sample inspections on each operating unit to review the status of Entergy's corrective actions related to the SCWE at Indian Point. The inspection included over 50 interviews and discussions with technicians, staff, supervisory and management personnel in a representative cross section of work groups. The inspectors also attended selected meetings and reviewed supporting documentation for corrective actions.

b. Findings and Observations

No findings of significance were identified.

The inspectors concluded that Entergy's progress on corrective actions related to the SCWE was adequate. The inspectors observed that Entergy implemented a number of actions to address previously identified issues affecting the work environment, as revealed in a 2006 safety culture assessment, NRC inspections, and an independent assessment conducted on behalf of Entergy.

Based on interview results and document reviews, the inspectors determined that several actions were effective in communicating the site’s commitment to a safety conscious work environment.

These actions included:

- Site Vice President meetings with small groups;
- Site-wide communications on safety conscious work environment; and
- Changes to site schedules that allowed supervisors and managers to spend more time in the field.

The inspectors identified two corrective actions that were not yet effective. Both of these were associated with Entergy's actions to detect and prevent retaliation, chilling effect, and the perception of retaliation. These items constituted issues of minor significance, because there was no actual impact on the work environment.

- First, the inspectors identified a deficiency in the implementation of the Executive Review Board (ERB), which was established to review proposed personnel actions to ensure: they were not in violation of 10 CFR 50.7 employee protection regulations; they did not involve retaliation; and any potential chilling effect was addressed. Specifically, the inspectors identified that the potential for retaliation or a chilling effect for raising safety issues was not considered for some adverse personnel actions that went before the ERB. In response to this observation, Entergy entered the issue in the corrective action program with an action for the ERB to review the personnel action cases for the potential for retaliation or a chilling effect related to raising safety issues.

- Secondly, the inspectors identified that the Executive Protocol Group (EPG) was not fully meeting its charter in providing advice to senior management on issues that may be related to retaliation or a chilling effect. For example, the EPG had not reviewed a specific event involving an individual who felt reluctant to raise issues based on the actions of a site manager. Additionally, the inspectors observed that the EPG was not reviewing some data and trending information as specified in its process document. For example, the EPG had not reviewed SCWE-related data from condition reports or findings from surveys and assessments. Entergy made several enhancements to the EPG meeting process to incorporate the inspectors' observations.

The inspectors also observed that Entergy's process for tracking and trending CRs with potential SCWE aspects was not timely. Specifically, the review of CRs with SCWE-related trend codes was being performed on a 6-month basis, which may not be timely for management to respond to and mitigate new issues or trends that could affect the work environment.

During interviews with the inspectors, all personnel indicated that they would raise issues that they recognized as a nuclear safety concerns. Some individuals stated they had heard of others who may be hesitant to raise issues, due to events that had happened in the past. A few individuals stated that they may not raise low level issues, because they did not believe the issues would be corrected.

When questioned about the site’s initiatives in the area of SCWE, most individuals were aware of the ongoing efforts. Some believed that the corrective actions were having a positive effect. Others were more skeptical of the corrective actions, based on their observations or what they had heard about statements made by management. Some personnel indicated that they were awaiting a demonstrated commitment to a SCWE, rather than just communications.

The inspectors noted that Entergy has a number of actions planned to continue its progress in improving the SCWE onsite. These actions include:

- Departmental action plans to address the safety culture aspects of a 2007 Entergy Employee Survey;
- A second round of Site Vice President meetings with small groups to continue the dialogue on SCWE;
- Ongoing efforts to conduct facilitated discussions and additional activities to improve the work environment in the Instrumentation and Controls work group; and
- Refresher training on SCWE.

The inspectors observed that Entergy's self-assessment of actions related to SCWE have been self-critical. For example, Indian Point management held a meeting in April 2007, to discuss and take corrective actions for certain events and management behaviors that were not conducive to establishing and maintaining a healthy safety conscious work environment onsite. Additionally, a recent Entergy corporate assessment and a quality assurance audit identified opportunities for improvement in this area.

4OA3 Event Followup (71153 - 2 samples)

.1 Manual Reactor Trip - Loss of 32 Main Boiler Feed Pump Speed Control and (Closed) LER 05000286/2007-001-00

On April 3, 2007, operators performed a manual reactor trip of Indian Point Unit 3 due to a loss of 32 main boiler feed pump speed control while conducting maintenance on the main boiler feed pump speed control system. The loss of speed control caused steam generator levels to lower, such that a manual reactor trip was required by procedures. The loss of speed control was attributed to operators de-energizing a power supply that was thought to provide power to the speed control system for only the 31 main boiler feed pump, which was shut down in preparation for the maintenance evolution.

A discrepancy in the plant drawing being used for developing the blocking points for the associated safety tagging led to a misunderstanding of how power was supplied to the speed control system. In actuality, speed control system power was supplied to both main boiler feed pumps through the circuit that the operators de-energized. In addition, a second power supply that was not working properly was forced to carry load when operators turned off what they thought was the correct circuit breaker. This led to an unexpected speed control problem with the 32 main boiler feed pump, which was the only pump in operation. Thus, both the plant drawing discrepancy and the degraded second power supply contributed to the loss of speed control.

Operators correctly diagnosed the situation and tried to restore main boiler feed pump speed control to normal. They were unsuccessful in this attempt, leading to the need to actuate a manual reactor trip. All systems functioned normally after the trip, and the plant was quickly stabilized in a hot shutdown condition.

Entergy replaced the affected power supplies, established planned maintenance to replace the control system power supplies, performed an extent of condition review, and implemented revisions to system controlled documents.

The inspectors reviewed the licensee event report (LER) and identified no findings of significance or violations of NRC requirements. A finding was not identified because although a performance deficiency did exist associated with plant drawings not being accurate, it would take the failure of another power supply to lead to the loss of the 32 main boiler feed pump. There was no violation of NRC requirements because the affected equipment is not safety-related, and therefore does not fall under requirements of 10 CFR 50 Appendix B. Entergy documented the event and corrective actions in condition report CR-IP3-2007-01775. This LER is closed.

.2 Automatic Reactor Trip - 31 Main Transformer Fire and (Closed) LER 05000286/2007-002-00

On April 6, 2007, while at 91 percent power, the Indian Point Unit 3 reactor automatically tripped due to a main turbine trip and generator lockout caused by an electrical fault in the 31 main transformer. The electrical fault in the 31 main transformer resulted in an explosion originating in the 'B' phase high voltage bushing, which is a integral part of the transformer. The electrical fault and explosion were only evident for a few seconds, and the ensuing fire was extinguished by the fire brigade in about 10 minutes. Operators declared a notification of a UE once it was realized that an explosion had occurred. However, this declaration was delayed due to personnel not immediately making the control room staff aware that an explosion had been observed. Although the explosion was transient in nature, left little evidence that it had occurred, and quickly became observable as a fire in the 31 main transformer, there were some Entergy personnel that were aware that an explosion had taken place. A number of these people did not contact the control room with their observation because the fire was quickly announced by the control room staff, and these personnel felt that the added communication with the control room would not be desirable as the control room was already taking actions to mitigate the event.

The inspectors confirmed that the shift manager made a timely and appropriate event classification once he was made aware that an explosion had occurred. Entergy documented this concern in the corrective action program as CR-IP3-2007-02036, and determined, as a part of their review, that additional site staff training is necessary to sensitize plant staff that the shift manager needs to be made aware of observations such as an explosion so that event classification can occur. Entergy's current training program meets the requirements of their emergency plan. The inspectors determined that operator actions after the reactor trip were in accordance with station emergency operating procedures, and the plant responded as expected to the reactor trip.

The inspectors identified a performance deficiency, in that, plant staff did not immediately make the shift manager aware of their observation that an explosion was observed from the 31 main transformer bushing. The Indian Point Emergency Plan Event Classification Guide requires that the Shift Manager declare a notification of a UE upon receiving a report from plant personnel of an observation of an explosion within the protected area of the plant. Inherent in this requirement is that when personnel observe an explosion in the protected area of the plant they promptly report the observation to the central control room. The inspectors determined that the performance deficiency was of minor safety significance because its occurrence would not lead to a significant event, nor could it become a more significant safety concern. In addition, no performance indicator would be affected by the deficiency. Finally, the performance deficiency did not affect the Emergency Preparedness cornerstone objective because Entergy provided adequate measures to protect public health and safety.

Entergy replaced and tested the 31 main transformer and associated bushings; tested and inspected the 32 main transformer, unit auxiliary transformer, and high voltage equipment; developed plans to establish testing acceptance criteria and data trending; and performed an extent of condition review. Entergy documented the failed component and corrective actions in condition report CR-IP3-2007-01834.

The inspectors reviewed the LER and identified no violations of NRC requirements. This LER is closed.

Findings

Introduction. The inspectors identified a finding of very low safety significance (Green), in that, Entergy failed to identify in the corrective action program an adverse condition associated with the 'B' phase high voltage bushing on the 31 main transformer (MT) that was discovered during testing. The data from that test indicated potential degradation of the 'B' phase high voltage bushing. As a result, this condition was not adequately evaluated before placing the transformer back in service, and the bushing subsequently failed.

Description. On April 6, 2007, the 'B' phase high voltage bushing on 31 MT failed while the unit was at approximately 91 percent power, in power ascension. The failure resulted in an explosion in the main transformer yard, a turbine trip, a reactor trip, and the declaration of a notification of a UE. A notification of a UE indicates a potential degradation in the level of safety of the plant, and that no release of radioactive material requiring offsite response or monitoring is expected unless further degradation occurs.

Following this failure, the inspectors reviewed maintenance activities associated with the 31 MT that were performed during a plant outage which occurred between March 6, 2007, and March 30, 2007. A power factor test was performed on March 27, 2007. This test is commonly used to determine the insulation integrity of high voltage equipment. The results from that test indicated potential degradation of the 'B' phase bushing. The nameplate power factor ratings and the most recent power factor test results are shown in the table below. The power factor test result on the 'B' phase bushing was identified by the vendor performing the test as requiring further evaluation, and the site engineering staff was notified.

Bushing Power Factor In Percent (%) By Phase

	A	B	C
Bushing Name Plate Rating	.44%	.30%	.43%
Test Results From 1999	.48%	.54%	.49%
Test Results From March 2007	.53%	**1.43%**	.53%

The engineering staff identified this as a potential adverse condition but did not place this issue into the corrective action program. Entergy's engineering staff reviewed the results of the test, and contacted an Entergy transmission and distribution system expert to determine the significance of the test results. Engineering personnel determined that the data was not representative of insulation degradation that would result in premature failure based on past operational history, recent thermography, and the work performed during the refueling outage. They concluded the 'B' phase bushing could be replaced during the next refueling outage. The transformer was returned to service following this determination.

The inspectors reviewed the transformer maintenance history, applicable operating experience, Entergy's initial evaluation of the identified condition, and industry standards for power factor testing acceptance criteria. The inspectors also reviewed Entergy's root cause evaluation of the failure.

The inspectors determined that, during Entergy's initial evaluation of the test results, the Indian Point Energy Center system engineer did not have complete information on the power factor testing acceptance criteria. In addition, the evaluation did not include a review of past operating experience specific to this particular bushing design, a General Electric Type U bushing. The inspectors noted that there was significant industry experience with failures of this particular bushing design. Several sources have provided power factor acceptance limits specific to this design. The inspectors evaluated acceptance criteria provided by General Electric (GE), Doble Engineering, and ABB, in addition to generic criteria provided in Institute of Electrical and Electronics Engineers (IEEE) standards.

Specific to the Type U bushing design, the GE criteria states that if the power factor exceeds 3.0 percent, the bushing needs to be replaced. If the value is between 1.0 percent and 3.0 percent, it is in a "region of concern," but there is little risk of failure if the capacitance change is less than 5.0 percent. A bushing in this "region of concern" should be monitored on an annual basis. Doble Engineering recommends replacing a bushing if the power factor exceeds 1.5 percent, or if it exhibits a sudden increase in value beyond 1.0 percent. A bushing with a power factor above 1.0 percent, or less than 1.0 percent but exhibiting a sudden increase, should be considered questionable and retested within six months. The capacitance recommendation is the same as GE's. ABB recommends replacement if the power factor doubles the nameplate value, or the capacitance increases to 110 percent of the nameplate value.

The inspectors found that Entergy relied upon the recommendation of their transmission and distribution system expert who determined that, based on the power factor number measured on March 27, 2007, the bushing would function properly until the next plant refueling outage. Based on interviews conducted with an Indian Point Energy Center system engineer, the inspectors determined that the transmission and distribution system expert had requested the previous test data for comparison; but, it was not made available. The inspectors determined that this was a necessary piece of information, given the available operating experience and the testing results, for the expert to assess the condition of the bushing. Therefore, the conclusion that the bushing was in acceptable condition was made without all the necessary information to provide a sound engineering justification, because no comparison to the previously conducted test result could be performed. Specifically, interpretation of the results depends primarily on comparing previous results with current test results. In addition, Doble Engineering and ABB acceptance criteria are dependent on the change in power factor over time.

Entergy's root cause evaluation stated that the power factor test met the GE acceptance criteria, therefore the bushing condition was satisfactory and the failure was the result of a random failure. The inspectors noted that the test results did meet the acceptance criteria provided by GE; however, these criteria had not been substantially modified since being established in 1979. Since that time, several other vendors have provided acceptance criteria which incorporate more recent test and failure data, both generically and associated with this particular bushing design. Acceptance criteria from Doble Engineering was provided in 1985; IEEE industry standards were dated 1995 and 2000; and ABB provided standards in 1998. Based on any of these criteria, with the exception of the GE criteria, the bushing should have been replaced prior to placing the transformer back into service. The previous power factor test was performed in 1999 and the results are listed in the table above. The test data from 2007 showed a significant increase for the 'B' phase bushing (from 0.54 percent to 1.43 percent) and would have led to a replacement of the bushing based on Doble Engineering and ABB acceptance criteria.

The inspectors determined that the numerical value for the bushing power factor of 1.43 percent would not always require replacement. However, given the significant rise since the last test, the industry experience associated with failures of this particular bushing design, and the basis for the various acceptance criteria, the inspectors determined that a thorough evaluation should have resulted in the replacement of the bushing prior to returning the transformer to service. While the inspectors determined that the bushing would not have required replacement based on the GE acceptance criteria, as stated by Entergy, these criteria do not appear to take into account the significant operating experience and data gathered since 1979. In addition, a facility within the Entergy fleet has used the Doble Engineering criteria as the standard for replacement of a bushing, therefore it would be reasonable to assume that the same criteria would be considered at Indian Point Energy Center. On February 15, 2007, a notice was received by the staff at Indian Point Energy Center describing a concern identified at the Grand Gulf Nuclear Station. This notice discussed the industry issues with the GE Type U bushing and stated that the station had used the criteria specified by Doble Engineering in their evaluations which led to bushing replacement. Grand Gulf Nuclear Station's engineering evaluation of the issue provided a basis for the use of these criteria. The bushing associated with the Grand Gulf Nuclear Station transformer was replaced based on its power factor being greater than 1.5 percent. However, since Entergy used the Doble Engineering criteria to make this determination, they should have reached the same conclusion to replace the bushing if they had power factor test results and history similar to that of Indian Point due to the criteria recommending replacement of a bushing with a power factor of greater than 1.5 percent, or that exhibits a sudden increase and is greater than 1.0 percent.

Analysis. The inspectors determined that the failure to identify, in the corrective action program, the adverse condition of the 'B' phase high voltage bushing on 31 MT is a performance deficiency, because it is contrary to the requirements of Entergy's procedure EN-LI-102, "Corrective Action Process." This procedure requires employees to initiate a condition report for all adverse conditions. 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 is associated with the equipment performance 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 as well as power operations. Specifically, Entergy did not place this issue in the corrective action process, and as a result, did not conduct an adequate evaluation of a degraded condition associated with the 'B' phase high voltage bushing on 31 MT. Subsequently, the bushing failed during power operation and resulted in a reactor trip, an explosion in the transformer yard, and the declaration of a notification of a UE. 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." This finding was determined to be of very low safety significance because, while it was a transient initiator that resulted in a reactor trip, it did not contribute to the likelihood that mitigation equipment or functions would not be available.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution, because Entergy failed to promptly identify an adverse condition in the corrective action program in a timely manner commensurate with its safety significance. (P.1(a))

In response to the inspectors' initial conclusion, Entergy provided further information which the staff subsequently reviewed. Entergy stated that the transmission and distribution system expert could make an adequate recommendation concerning the bushing without reviewing the 1999 power factor test results. The NRC staff disagrees with this conclusion, and considers the rapid change in power factor from test to test to be relevant in determining whether or not the bushing should remain in service.

In addition, Entergy asserted that the operating experience associated with the Grand Gulf Nuclear Station transformer was fundamentally different; in that, the power factor testing result was greater than 1.5 percent and required bushing replacement. The NRC staff acknowledged this fact. However, the staff concluded that Entergy utilized the Doble Engineering criteria to replace the bushing at the Grand Gulf Nuclear Station; and had the same criteria been utilized at Indian Point Energy Center, it is reasonable to conclude that bushing replacement would have occurred in this circumstance because the Doble Engineering criteria also recommends bushing replacement if a sudden increase in power factor occurs between tests, if above 1.0 percent power factor.

Furthermore, Entergy stated that utilizing their corrective action process would not necessarily have led to a different decision on their part. The NRC staff disagrees with this conclusion and believes that implementation of the guidance in Entergy procedures EN-LI-102, "Corrective Action Process," EN-OP-104, "Operability Determinations," and ENN-DC-115, "Engineering Request Response Development," would have resulted in a determination that the bushing should be replaced prior to returning it to service.

Evaluation. No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance, because the failure to enter the degraded condition into the corrective action program or adequately evaluate the condition occurred on a non-safety-related system. **(FIN 05000286/2007003-01, Failure to Identify in the Corrective Action Process, or Adequately Evaluate a Degraded Condition Associated with a High Voltage Bushing on a Main Transformer)**

4OA5 Other Activities

.1 Groundwater Contamination Investigation

a. Inspection Scope

Continued inspection of Entergy's plans, procedures, and characterization activities affecting the contaminated groundwater condition at Indian Point, relative to NRC regulatory requirements, was authorized by the NRC Executive Director for Operations in a Reactor Oversight Process deviation memorandum dated October 31, 2005 (ADAMS Accession number ML053010404) and renewed on December 11, 2006 (ADAMS Accession number ML063480016). Accordingly, continuing oversight of Entergy's progress has been conducted throughout this quarterly inspection report period consisting of onsite inspections, independent split sample analyses of selected monitoring well samples, review of action plan completion status, and periodic communications with Federal, State, and local government stakeholders.

Inspectors conducted an onsite review of tracer test sampling results on May 9 and 10, 2007. New York State Department of Environmental Conservation officials observed and participated in the proceedings. The onsite meeting provided for an independent hydrology review of Entergy's tracer test findings and associated re-evaluation of the current site groundwater model.

b. <u>Findings and Observations</u>

No findings of significance were identified.

The objective of the tracer test, as mentioned above, was to identify groundwater flow and direction by injecting fluorescent tracer dye into a subsurface location representing the source of leakage, and tracking its natural groundwater migration as it was intercepted by existing monitoring wells and storm drain locations. The fluorescein dye was injected into a specially designed tracer injection co-located near monitoring well MW-30, adjacent to the Unit 2 spent fuel pool (SFP). On February 8, 2007, the tracer test began with injection of approximately 200 gallons of dye at a subsurface elevation equivalent to the bottom of the Unit 2 spent fuel pool. The natural groundwater migration of this tracer has been tracked for approximately 13 weeks by measuring the dye content in either charcoal samplers or water samples collected at selected onsite monitoring wells and storm drain locations.

The tracer test was designed as an analogue to the Unit 2 SFP leakage. Entergy's hydrology consultant, GZA, described (through its visualizations) how the tracer entered the unsaturated zone above the local water table similar to the abnormal releases from the Unit 2 SFP, and moved horizontally to adjacent wells before moving vertically into the saturated zone. GZA also noted the roles of backfills which provide preferential paths to the storm drains as was demonstrated from tracer material observed in the manholes near the Unit 2 SFP.

GZA indicated that its preliminary assessment considered flow and transport in the Inwood Marble formation to be dominated by porous media flow conditions, and that the fractures were so numerous and interconnected at the site scale that it may not be reasonable to single out and ascribe parameters for fracture flow and transport modeling. The U.S. Geological Survey (USGS) indicated the possibility that analysis of borehole data (e.g., downhole logging data), pump test and ambient flow results, and observed fracture orientations and spacing using the WELLCAD code could provide insights to discern the presence of significant fracture zones, and their transmissivities (i.e., flow parameters). To this end, NRC staff is working with the USGS to accomplish an independent analysis considering an alternative conceptual model of flow and transport. Additional review and evaluation is expected to ascertain if there could be any significant difference in groundwater flow that would affect the overall assessment of public dose.

GZA noted that it was in the process of modifying its dose assessment model to factor in more realistic, site-specific conditions and parameters that were revealed from the recovery well RW-1 pump test and subsequent tracer test results. GZA, USGS, and NRC staff agreed that it was important to effectively consider the groundwater recharge zones and net flow discharge zones, and couple the information with the data developed from the pumping and tracer test; and the transmissivity values for the fracture zone as derived from WELLCAD modeling results. Such effort is expected to provide a more refined estimate of groundwater effluent release and dose assessment.

NRC, USGS, Entergy, and GZA staff discussed the development of a site-wide, long-term monitoring program plan to be linked to the dose assessment model. The plan would identify which existing wells and manhole sampling locations could provide the best performance indicators of the groundwater flow system behavior, and provide early detection of any abnormal radiological releases from onsite structures, systems, and components.

Based upon the technical discussions, current remediation strategies include the continued processing of the Unit 1 spent fuel pool utilizing filter/demineralization processes; the eventual removal of the spent fuel to dry cask storage; and subsequent draining of the Unit 1 spent fuel pool. Such activities are planned to be accomplished by Entergy in 2008. Currently, Entergy has no plans for further pumping tests using RW-1 since it was demonstrated that pump-out of the groundwater through this location will result in cross-contamination of groundwater in the vicinity of Unit 2. Entergy indicated that the groundwater conditions would continue to be evaluated for remediation, as necessary, upon completion of the Unit 1 spent fuel pool activities.

Monitoring for tracer material is expected to continue through July 2007, and sampling results will be reported to the NRC and NYS DEC. GZA agreed to provide well logging, pumping test, and fracture characterization data for USGS's WELLCAD modeling. Follow-on technical meetings will focus on GZA's final monitoring report which incorporates their new dose assessment model; USGS's WELLCAD analyses; and development of a site-wide groundwater monitoring plan.

The NRC monitoring well 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. The NRC's assessment of Entergy's sample analytical results data indicated that their analytical contractor continued to report sample results that were comparable with the NRC's analytical results. Information to date continues to support that the estimated radiological release fraction through groundwater is negligible relative to NRC regulatory limits.

The NRC's ORISE/ESSAP sample results are available in ADAMS under the following accession numbers: ML071900438, ML071900442, ML071900445, ML071900447, ML071900448, ML071900456, ML071900458, ML071900462. To date, sample results from site boundary wells and offsite environmental groundwater sampling locations have not indicated any detectable plant-related radioactivity.

.2 (Closed) URI 05000286/2006301-01, Examination Development Issue

a. Inspection Scope

In response to a notification by the licensee that a potential compromise in examination security may have occurred, the NRC initiated an investigation (Office of Investigations, 1-2007-003). This investigation included reviewing licensee procedures, training records, and interviews with applicants, trainers, and supervisors. The investigation assessed whether a compromise had occurred and, if substantiated, determined the extent of the compromise, and gathered information to support potential enforcement actions.

This issue was initially documented as URI 05000286/2006301-01, Examination Development Issue. The requirements of 10 CFR 55, "Operator's Licenses," and the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, were used as criteria.

b. Findings

A licensee-identified violation is documented in section 4OA7.

4OA6 Meetings, including Exit

Exit Meeting Summary

On July 13, 2007, the inspectors presented the inspection results to Mr. Anthony Vitale and other Entergy staff members, who acknowledged the inspection results presented. Entergy did not identify any material as proprietary.

4OA7 Licensee-Identified Violations

The following Severity Level IV violation was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG 1600, for being dispositioned as a non-cited violation.

Prior to administering the 2006 initial licensed operator NRC examination, Entergy informed the NRC that regulations and guidelines regarding examination security may not have been followed. Specifically, a training supervisor was directing training to be conducted for examination topics that were not previously covered during the applicants' training. After receiving this report, the NRC, in parallel with Entergy, conducted an investigation to determine the nature and extent of the issue. The NRC determined that the extent of the compromise was ultimately limited to two questions on the written examination and one job performance measure (JPM). To ensure the integrity of the written examination, these two questions and twenty three others were removed from the examination. These questions were replaced with other randomly selected test items that were provided by the NRC. The compromised JPM was replaced. Based upon the replaced JPM, the nature of the operating examination, and the security arrangements, the NRC did not consider the operating examination to be compromised. The examination was determined to be valid and was administered. The investigation continued to gather information to support potential enforcement actions.

Following the administration of the examination, the NRC further investigated the personnel and events surrounding this issue and determined that the training supervisor had misinterpreted NRC guidance regarding what was, and what was not, appropriate activities for a person in his position. Regardless of his understanding, and although his actions were identified and corrected prior to the administration of the examination, the NRC concluded that the supervisor's actions were a violation of NRC requirements as stated below. NRC regulations prohibit facility licensees from engaging in any activity that could compromise the integrity of any examination required by 10 CFR 55, "Operator's Licenses."

This finding was determined to be more than minor because the failure to administer an equitable and consistent licensed operator qualification examination had the potential to cause a credible impact on safety since operators could have been considered for licensing without demonstrating an adequate level of knowledge. This finding was considered as traditional enforcement because the issue had the potential for impacting the NRC's ability to make a licensing decision to permit individuals to operate the controls of a nuclear power plant. This finding was determined to be a Severity Level IV non-cited violation because no willfulness was involved, it was not repetitive, it was entered into the licensee's corrective action program, and the licensee notified the NRC of this issue.

10 CFR 55.49, "Integrity of Examinations and Tests," states in part that, "applicants, licensees, and facility licensees shall not engage in any activity that compromises the integrity of any application, test, or examination required by this part. The integrity of a test or examination is considered compromised if any activity, regardless of intent, affected, or, but for detection, would have affected, the equitable and consistent administration of the test or examination. This includes activities related to the preparation and certification of license applications and all activities related to the preparation, administration, and grading of the tests and examinations required by this part."

Contrary to the above, Entergy developed and submitted the 2006 Initial Licensed Operator Qualification Examination for NRC review and approval and then subsequently engaged in training activities in a manner which compromised the integrity of the examination. The training activities in question occurred in late August 2006 and throughout September 2006 in the weeks leading up to the examination which was originally scheduled for the weeks of October 23 and 30, 2006. These training activities were identified by the licensee and reported to the NRC. Subsequent investigations by the NRC during the weeks of October 10 through December 15, 2006, determined that a compromise, and thus a violation, had occurred. Entergy provided focused training on examination test items just before the examination was to be administered, thereby undermining the ability of the NRC to infer adequate mastery of the necessary knowledge and abilities for making a licensing decision.

Entergy entered this issue into their corrective action program (CR IP3 2006-02786 and 03108) and immediately initiated a root cause investigation. Entergy's investigation made a determination regarding the extent of the compromise, which corresponded to the results of an independent investigation conducted by the NRC. Because the issue was placed in the corrective action program and compliance was restored before the examination was administered and because the issue was not repetitive nor willful, this violation is being treated as a Severity Level IV non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

F. Dacimo, Site Vice President
J. Comiotes, Director, Nuclear Safety Assurance
A. Williams, Acting Site Operations Manager
A. Vitale, Acting Plant Manager
T. Barry, Security Manager
J. Donnelly, Manager, Maintenance
P. Conroy, Manager, Licensing
B. Sullivan, Emergency Planning Manager
T. Jones, Licensing Supervisor
L. Lee, Systems Engineering Supervisor
T. Orlando, Manager, Design Engineering
P. Cloughhessy, Maintenance Rule Program Coordinator
N. Azevedo, Codes and Fire Protection
S. Verrochi, System Engineering Manager
S. Davis, Superintendent, Operations Training
R. Christman, Training Manager, Indian Point Energy Center
D. Huntington, Senior Instructor
W. Altic, Senior Instructor
S. Joubert, Training Supervisor

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000286/2007003-01	FIN	Failure to Identify in the Corrective Action Process, or Adequately Evaluate a Degraded Condition Associated with a High Voltage Bushing on a Main Transformer

Closed

05000286/2006301-01	URI	Examination Development Issue
05000286/2007-001-00	LER	Manual Reactor Trip Due to Decreasing Steam Generator Levels as a Result of the Loss of Feedwater Flow Caused by the Failure of 32 Main Feedwater Pump Train A Control Logic Power Supply

05000286/2007-002-00	LER	Automatic Reactor Trip Due to a Turbine-Generator Trip Caused by a Fault on the 31 Main Transformer Phase B High Voltage Bushing

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
3-SOP-RW-002, Rev 22: “Intake Structure Operation”
3-SOP-RW-001, Rev 29: “Circulating Water System Operation”
OAP-008, Rev 2: “Severe Weather Preparations”
OAP-48, Rev 4: “Seasonal Weather Preparation”
3-SOP-FP-001, Rev 28: “Fire Protection System Operation”
3-SOP-V-006, Rev 15: “Heating and Ventilation Systems”

Work Orders:

IP3-06-01219	IP3-05-01995	I3-027709969	IP3-06-01230
IP3-06-01320	IP3-04-05227	IP3-05-00179	IP3-05-00187
IP3-04-05232			

Section 1R04: Equipment Alignment

Procedures
COL-FPV-1, Rev 2: “Fire Pump House Verification”
3-COL-FW-2, Rev 29: “Auxiliary Feedwater System”
3-PT-M042B, Rev 4: “Diesel Fire Pump Test”
3-PT-Q117B, Rev 5: “32 Containment Spray Pump Functional Test”
COL-CSV-1, Rev 5: “Containment Spray Verification”

Drawings

9321-F-20193	9321-F-20183	9321-F-20173	9321-F-20413
9321-F-27503	9321-F-27353		

Condition Reports

IP3-2005-05226	IP3-2007-00687

Work Orders

I3-017701087	IP3-06-02130	IP3-04-09148	IP3-07-00257
IP3-06-16687	IP3-06-16638	IP3-05-14887	IP3-04-06137

Section 1R05: Fire Protection

Procedures
ENN-DC-161, Rev 1: “Transient Combustible Program”
SMM-DC-901, Rev 2: “IPEC Fire Protection Program”

Miscellaneous
Pre-Fire Plan 306, Rev 0: “General Floor Plan- Primary Auxiliary Building”
Pre-Fire Plan 264, Rev 0: “Intake Structure - Exterior Buildings”
Pre-Fire Plan 265, Rev 0: “Diesel Fire Pump House - Exterior Buildings”
Pre-Fire Plan 351, Rev 5: “480V Switchgear Room- Control Building“
Pre-Fire Plan 308A, “Volume Control Tank- Primary Auxiliary Building,” Revision 0

Condition Reports
IP3-2007-02302347

Section 1R06: Flood Protection Measures

Procedures
2-AOP-FLOOD-1, Rev 5: “Flooding”
3-AOP-FLOOD-1, Rev 3: “Flooding”
OAP-008, Rev 3: “Severe Weather Preparations”

Section 1R07: Heat Sink Performance

Procedures
EN-DC-147, Rev 2: ”Indian Point Units 2 & 3 Eddy Current Program”
0-HTX-400-GEN, Rev 1: “Eddy Current Inspection of Heat Exchanger Tubes”

Section 1R11: Licensed Operator Requalification Program

Procedures
E-0, Rev 21: “Reactor Trip or Safety Injection”
E-3, Rev 20: “Steam Generator Tube Rupture”

Other Documents
LRQ-SES-37, Rev 8: “MFRV Fails Closed, 33 ABFP Trip, SGTR, Loss of IA To Containment”

Section 1R12: Maintenance Effectiveness

Procedures
ENN-DC-205, Rev 0: “Maintenance Rule Monitoring”
AP-55, Rev 5: “Preventive Maintenance Program”
EN-DC-337, Rev 1: “Living Preventive Maintenance Program”
EN-DC-324, Rev 0: “Preventive Maintenance Process”
EN-LI-102, Rev 8: "Corrective Action Process"

Condition Reports
IP3-2006-02827 IP3-2006-00565 IP3-2006-01001
IP3-2007-01545

Miscellaneous
ENN-MS-S-008, Attachment 7.2, Rev 0: "Maintenance Rule Action Plan for Unit 3 Containment Building Pressure Relief Valve VS-PCV-1190"

Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Procedures
IP-SMM-WM-101, Rev 1: "On-Line Risk Assessment"
IP-SMM-WM-100, Rev 5: "Work Control Process"
EN-MA-125, Rev 2: "Troubleshooting Control"
3-AOP-VAC-1, Rev 4: "Loss of Condenser Vacuum"

Work Orders

IP3-07-00739	IP3-07-20519	IP3-06-21771	IP3-07-21140
IP3-07-00415			

Condition Reports

IP3-2007-02148	IP3-2007-02350	IP3-2007-02357	IP3-2007-02594
IP3-2007-02595	IP3-2007-02312	IP3-2007-02327	IP3-2007-02324

Miscellaneous
System Description 27.2, "Exciter"
Troubleshooting Control Form , "Reactivity Anomaly of the RCS"
Entergy letter NL-05-026, dated February 22, 2005; regarding Alternate Source Term license amendment request.
Entergy letter NL-05-036, dated March 14, 2005; regarding Amendment Request Alternate Source Term.
Entergy letter NL-04-068, dated June 2, 2004; regarding Full Scope Adoption of Alternate Source Term.

Section 1R15: Operability Evaluations

Procedures
IP-SMM-AD-102, Rev 4: "IPEC Implementing Procedure Preparation, Review and Approval"
EN-OP-104, Rev 4: "Operability Determinations"
OAP-026, Rev 0: "Determination of Operability"
EN-LI-102, Rev 8: "Corrective Action Process"
3-PT-Q016, Rev 19: "EDG and Containment Temperature SW Valves SWN-1176 & 1176A and SWN-TCV-1104 & 1105"
3-PT-R090D, Rev 12: "Emergency Local Operation of Auxiliary Boiler Feed Pumps"
3-SOP-ESP-001, Rev 17: "Local Equipment Operation and Contingency Actions"
3-PT-M108, Rev 3: "RHR/SI System Venting"
SI-SOP-SI-001, Rev 38: "Safety Injection System Operation"

Condition Reports

IP3-2005-00695	IP3-2007-02059	IP3-2007-02442	IP3-2007-02441

Drawings
93-13102: Darling Double Disc Gate Valve

Calculations
CN-SEE-03-128-R.1: “Indian Point Unit 3 Containment Spray RWST Alignment Minimum and Maximum Spray Flow”

Miscellaneous
WCAP-16212-P: Rev 0: NSSS and BOP Licensing Report

Procedures
EN-DC-105, Rev 0: “Configuration Management”
ENN-DC-103, Rev 1: “Design Process”
ENN-DC-115, Rev 6: “ER Response Development”
OAP-031, Rev 0: “Control of Operator Aids”
ENN-DC-112, Rev 7: “Engineering Request and Project Initiation Process”
ENN-DC-117, Rev 4: “Post Modification Testing and Special Testing Instructions”
3-OSP-WDS-001, Rev 2: “RCS and Refueling Cavity Cleanup”
OAP-7, Rev 10: “Containment Entry and Egress”
3-AOP-SW-1, Rev 2: “Service Water Malfunction”
EN-LI-102, Rev 8: “Corrective Action Process”

Section 1R19: Post-Maintenance Testing

Procedures
OAP-024, Rev 2: “Operations Testing”
3-SOP-FW-004, Rev 26: “Auxiliary Feedwater System Operation”
3-PT-Q117B, Rev 5: “32 Containment Spray Pump Functional Test”
0-VLV-413-MOV, Rev 2: “Motor Operated Valve Minor Preventive Maintenance”
0-VLV-412-MOV, Rev 2: “Use of Motor Operated Valve Diagnostics”
3-PT-Q088, Rev 15: “Component Cooling Pumps Functional Test”
3-PMP-003-CCW, Rev 0: “Inspection/Repair of the Component Cooling Pump”
0-VLV-420-GEN, Rev 0: “Inspection and Repair of Conval Clampseal Piston Check Valves”
3-PT-Q062A, Rev 8: “31 Charging Pump Operability Test”

Work Orders

IP3-07-19935	IP3-07-19744	IP3-07-00739	IP3-07-20519
IP3-07-12275	IP3-02-22193	IP3-03-23793	IP3-03-10580
IP3-06-12306	IP3-06-22068	IP3-05-00534	IP3-05-01723
IP3-03-19160	IP3-06-11019	IP3-05-21031	IP3-03-03320
IP3-06-21095			

Condition Reports
IP3-2007-02370

Section 1R20: Refueling and Outage

Procedures
3-POP-1.2, Rev 49: “Reactor Startup”
3-SOP-RC-001, Rev 27: “Full Length Rod Control and RPI System Operation”
3-AOP-ROD-1, Rev 01: “Rod Control and Indication Systems Malfunction”

3-POP-1.3, Rev 51: “Plant Startup from Zero to 45% Power”
3-POP-4.2, Rev 23: “Operation Below 20% Przr Level with Fuel in the Reactor”

OAP-007, Rev 10: “Containment Entry and Egress”
3-POP-4.2, Rev 23: “Operation Below 20% Pressurizer Level with Fuel in the Reactor”
3-POP-4.1, Rev 25: “Operation at Cold Shutdown”

Condition Reports
IP3-2007-02099 IP3-2007-01998

Work Orders
IP3-07-00736

Section 1R22: Surveillance Testing

Procedures
SOP-WDS-010, Rev 13: “Monitoring Leaks Within The Containment Building”

Condition Reports

IP3-2005-02985	IP3-2005-03336	IP3-2005-03289	IP3-2005-01896
IP3-2006-03061	IP3-2006-02834	IP3-2007-02338	IP3-2007-02377
IP3-2007-02350	IP3-2007-02357		

Work Orders

IP3-05-16829	IP3-05-15435	IP3-05-22984	IP3-06-17297
IP3-05-22763	IP3-07-13796		

Section 1EP6: Drill Evaluation

Procedures
IP-EP-120, Rev 2: “Emergency Classification”
IP-EP-410, Rev 3: “Protective Action Recommendations”
IP-EP-AD1, Rev 1: “Maintaining Emergency Preparedness”

Condition Reports

IP2-2007-02051	IP2-2007-02053	IP2-2007-02054	IP2-2007-02055
IP2-2007-02056			

Section 4OA1: Performance Indicator Verification

Procedures
EN-LI-114, Rev 2: “Performance Indicator Process”
NEI 99-02, Rev. 4: “Regulatory Assessment Performance Indicator Guideline”
EN-LI-114, Attachment 9.2, Rev 2: “NRC Performance Indicator Technique Sheet”

Condition Reports:
IP3-2007-02552 IP3-2006-00046 IP3-2006-01001

Miscellaneous:
Maintenance Rule Program Quarterly Report, First Quarter 2007

Section 4OA2: Identification and Resolution of Problems

Procedures
EN-NS-116, Rev 2: "Access Authorization Processes"
EN-NS-102, Rev 3: "Fitness for Duty Program"

Miscellaneous
Indian Point Energy Center Quarterly Trend Report- 4th Quarter 2006
Indian Point Energy Center Quarterly Trend Report- 1st Quarter 2007
2006 Unit 3 Annual Report, Central Control Room HVAC
IP3-RPT-HVAC-01904, Rev 0: "Maintenance Rule Basis Document, AFW HVAC, Electrical Tunnel HVAC, Control Building HVAC and Control Room HVAC"

Condition Reports

IP2-2007-00682	IP3-2007-01867	IP3-2007-01870	IP2-2007-01514
IP3-2007-01803	IP3-2006-00511	IP2-2005-03898	IP2-2006-04874
IP2-2006-04280	IP2-2006-04361	IP3-2005-05863	IP2-2007-01039
IP3-2005-00952	IP3-2006-00726	IP2-2006-01213	IP2-2006-00607
IP2-2006-03930	IP3-2006-03931	IP3-2006-02529	IP3-2007-02678
IP3-2007-02682	IP3-2006-00324	IP3-2007-02132	IP3-2006-00029
IP3-2005-05862	IP3-2006-00231	IP3-2006-00313	IP3-2006-00324
IP3-2006-00327	IP3-2006-01616	IP3-2006-01895	IP3-2006-00582
IP3-2006-00362	IP3-2006-03165	IP3-2006-03169	IP3-2006-03330
IP3-2006-03348	IP3-2006-03714	IP3-2006-03717	IP3-2006-03988
IP3-2006-04059	IP3-2006-04083	IP3-2007-01767	IP3-2007-01799
IP3-2007-02095	IP3-2007-02111	IP3-2007-02132	IP3-2007-02224
IP3-2007-02268	IP3-2007-02281		

LIST OF ACRONYMS

ADAMS	agencywide documents and management system
ANS	alert notification system
AFW	auxiliary feed water
CAP	corrective action program
CCR	central control room
CFR	Code of Federal Regulations
CR	condition report
DEC	Department of Environmental Conservation
EDG	emergency diesel generator
ESSAP	Education, Environmental Site Survey and Assessment Program
IMC	inspection manual chapter
IP2	Indian Point Nuclear Generating Unit 2
IP3	Indian Point Nuclear Generating Unit 3
IPE	individual plant examination

LER	licensee event report
MW	monitoring well
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
PARS	publicly available records
PI	performance indicator
RHR	residual heat removal
RW	recovery well
SDP	significance determination process
SFP	spent fuel pool
SI	safety injection
SSC	systems, structures, components
TS	technical specifications
UE	unusual event
URI	unresolved item
UFSAR	updated final safety analysis report
WO	work order