United States Nuclear Regulatory Commission Official Hearing Exhibit	
In the Matter of:	Entergy Nuclear Operations, Inc. (Indian Point Nuclear Generating Units 2 and 3)

ASLBP #: 07-858-03-LR-BD01
Docket #: 05000247 | 05000286
Exhibit #: NRC000092-00-BD01 **Identified:** 10/15/2012
Admitted: 10/15/2012 **Withdrawn:**
Rejected: **Stricken:**
Other:

NRC000092
Submitted: March 30, 2012

January 31, 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 2 - NRC INTEGRATED INSPECTION REPORT NO. 05000247/2006005

Dear Mr. Dacimo:

On December 31, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Unit 2. The enclosed integrated inspection report documents the inspection results, which were discussed on January 10, 2007, with Mr. Keith Polson 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, five findings of very low safety significance (Green) were identified. Three of these findings were also determined to be violations of NRC requirements. However, because of their very low safety significance, and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a written 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. 220555-0001; 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 2.

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

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report No. 05000247/2006005
w/ Attachment 1: Supplemental Information
w/ Attachment 2: Mitigating System Performance Index Verification

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

REGION I

Docket No.:	50-247
License No.:	DPR-26
Report No.:	05000247/2006005
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	Indian Point Nuclear Generating Unit 2
Location:	450 Broadway, GSB Buchanan, NY 10511-0305
Dates:	October 1, 2006 through December 31, 2006
Inspectors:	M. Cox, Senior Resident Inspector, Indian Point 2 G. Bowman, Senior Resident Inspector, Indian Point 3 B. Wittick, Resident Inspector, Indian Point 3 C. Hott, Resident Inspector, Indian Point 2 S. Barr, Senior Emergency Preparedness Specialist, Region I D. Jackson, Senior Project Engineer, Region I C. Long, Project Engineer, Region I J. Lilliendahl, Reactor Inspector, Region I S. Schneider, Senior Resident Inspector, Millstone K. Mangan, Senior Reactor Engineer, Region I K. Young, Senior Reactor Engineer, Region I J. Noggle, Senior Health Physicist, Region I J. Kottan, Senior Health Physicist, Region I T. Nicholson, Senior Technical Advisor, RES
Approved by:	Eugene W. Cobey, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000247/2006005; 10/01/2006 - 12/31/2006; Indian Point Nuclear Generating Unit 2; Evaluations of Changes, Tests, or Experiments; Maintenance Effectiveness; Maintenance Risk Assessment and Emergent Work; Access Control to Radiologically Significant Areas; and ALARA Planning and Controls.

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

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. The inspectors identified a Green non-cited violation (NCV) of Title 10 of the Code of Federal Regulations (CFR), Part 50.65(a)(4), because Entergy did not adequately assess and manage the risk of on-line maintenance activities while operating with a degraded steam inlet valve on one of Entergy's two main boiler feed pumps (MBFP). Specifically, from November 16 through 21, 2006, the degraded condition of the 21 MBFP increased the likelihood of a reactor trip, but was not assessed or included in the plant's on-line risk model. Entergy entered this issue into their corrective action program and properly assessed 21 MBFP risk on November 21, 2006.

The inspectors determined that this finding was more than minor because Entergy failed to consider risk significant structures, systems, components, and support systems that were unavailable during the performance of on-line maintenance. Specifically, Entergy failed to assess the increase in online risk from the increased likelihood of a reactor trip due to the 21 MBFP degraded condition. The inspectors evaluated this finding using IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," and determined that this finding was of very low safety significance because the finding resulted in an increase in the incremental core damage probability of less than $1x10^{-6}$ (actual increase was approximately $2x10^{-8}$).

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not provide complete and accurate procedures, in that, the online risk assessment procedure did not require degraded equipment that impacted risk to be assessed or managed. (Section 1R13)

Cornerstone: Mitigating Systems

Green. The inspectors identified a Green finding, in that, Entergy's corrective actions were inadequate to resolve a deficiency associated with the gas turbine 1 (GT-1) starting diesel. This deficiency was identified following a failure of GT-1 to start on

February 7, 2005, and resulted in three subsequent failures. A corrective action was written to correct the deficient condition following the initial failure and was closed on June 22, 2005, with no actions taken based on a senior management decision to cancel preventive maintenance activities on the gas turbines due to pending system retirement. Entergy entered this issue into their corrective action program and installed a modification to the coolant system to prevent further trips due to this condition.

The inspectors determined that this finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, it impacted GT-1 reliability, in that, the deficiency resulted in multiple failures to start on demand after the condition was identified and the action to correct the condition was closed without being implemented. 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," and determined that a Phase 2 evaluation was required because the finding represented an actual loss of safety function of a non-Technical Specification required train of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours. The inspectors used the Risk-Informed Inspection Notebook for Indian Point Nuclear Generating Unit 2, to conduct the Phase 2 evaluation. The inspectors determined that 65 hours of unavailability were caused by the additional failures of GT-1 due to the starting diesel coolant system deficiency. The inspectors conservatively equated this cumulative unavailability time to the total exposure time and used an initiating events likelihood of less than three days. The Phase 2 approximation yielded a result of very low safety significance (Green).

The inspectors determined this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that equipment and resources were available and adequate to assure reliable operation of GT-1. Specifically, Entergy did not minimize long-standing equipment issues and maintenance deferrals associated with the gas turbine system. (Section 1R12)

Cornerstone: Barrier Integrity

Severity Level IV. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59, "Changes, Tests and Experiments," for failure to obtain a license amendment pursuant to 10 CFR 50.90 prior implementing a change to alter the requirements of a shutdown fission product barrier. The inspectors reviewed Safety Evaluation 04-0732-MD-00-RE R1, "Installation of a Temporary Roll-up Door on the Containment Equipment Hatch," to determine if the conclusion that a licensee amendment was not required was correct. Entergy concluded that the roll-up door was equivalent to the closure plate and, therefore, adequate to close containment as required by the action statement. The inspectors found that the door was not designed to be air-tight; therefore, any radioactive release inside containment would bypass the roll-up door. The inspectors concluded that the roll-up door did not meet the design or licensing basis of the closure plate as described in the Updated Final Safety Analysis Report (UFSAR) and previously approved license amendments. Consequently, Entergy incorrectly concluded that a license amendment pursuant to 50.90 was not required prior to implementing the change. Entergy entered the issue into their corrective action program to evaluate and correct.

The inspectors determined that Entergy changed the requirements for the shutdown fission product barrier (containment) prior to receiving NRC approval. As a result, traditional enforcement was used to evaluate the issue because the deficiency affected the NRC's ability to perform its regulatory function. The severity level of the violation was determined to be Severity Level IV in accordance with example D.5 of Supplement 1 of the NRC Enforcement Policy. Additionally, the issue was determined to be of very low safety significance (Green) based on the low decay heat levels at the time the roll-up door was credited in accordance with the significance determination process described in Inspection Manual Chapter (IMC) 0609 Appendix H, "Containment Integrity." (Section 1R02)

Cornerstone: Occupational Radiation Safety

Green. A Green, self-revealing NCV of 10 CFR 20.1501 with respect to 10 CFR 20.1902(b) was identified, in that, Entergy failed to survey radiological condition changes after a plant manipulation that was likely to cause a change in radiological conditions, and this led to the failure to post a plant area as a high radiation area. As a result, two workers were allowed access to an unsurveyed and unposted high radiation area.

The finding is more than minor because it is associated with the Occupational Radiation Safety cornerstone attribute of exposure control and affected the cornerstone objective, because not establishing radiological conditions and commensurate controls after changing plant radiological conditions prior to allowing access to the affected areas can cause increased personnel exposure. The inspectors evaluated this finding using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," and determined that it was of very low safety significance (Green) because it did not involve ALARA planning and controls, an overexposure, a substantial potential for overexposure, or an impaired ability to assess dose. This issue was entered into Entergy's corrective action program and training was provided to the radiation protection staff.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not use a conservative assumption in the decision-making process, in that, the watch radiation protection technician did not question the radiological conditions of the pipe chase area after a change of plant conditions had occurred and did not require a survey of the pipe chase area before authorizing access to personnel. (Section 2OS1)

Green. A self-revealing finding was identified that involved inadequate modification planning and construction preparations relative to a Unit 2 containment sump strainer modification that resulted in significant unplanned collective exposure (93.7 person-rem compared to a work activity estimate of 10.9 person-rem). Specifically, the actual job site conditions for installation of the containment sump modification were not adequately evaluated with respect to the radiological impact of increased occupancy in high dose rate work areas. This unplanned additional in-field high radiation work resulted in significant unintended exposure that could have been avoided. This issue was entered into Entergy's corrective action program so that lessons learned could be incorporated into the Unit 3 containment sump modification.

The inspectors determined that this finding was more than minor because it was similar to examples 6.a and 6.b of IMC 0612, Appendix E, "Examples of Minor Issues," in that, the issue involved actual collective exposure greater than 5 person-rem and was greater than 50 percent above the estimated or intended exposure; and the majority of the dose overrun was due to activities within Entergy's control. The inspectors evaluated this finding using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," and determined that the finding was of very low safety significance (Green) because it involved an ALARA planning issue, and the 3-year rolling average collective dose for Unit 2 was less than 135 person-rem (73 person-rem average annual exposure for 2003 through 2005).

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not adequately incorporate job site conditions in the work control planning process. (Section 2OS2)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Indian Point Nuclear Generating Unit 2 began the inspection period operating at full power and remained at or near full power until a reactor trip occurred on November 15, 2006. The reactor trip occurred during troubleshooting activities associated with the main generator voltage regulation system. The plant returned to full power on November 17, 2006 and remained at full power until the operators commenced a plant shutdown on November 30, 2006, to repair a steam leak. The plant returned to full power on December 1, 2006, and continued to operate 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 the readiness of risk-significant systems for extreme weather conditions. The inspectors reviewed Entergy's adverse weather procedures, operating experience, corrective action program, Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), operating procedures, staffing, and applicable plant documents to determine the types of adverse weather challenges to which the site is susceptible.

Additionally, the inspectors evaluated implementation of the adverse weather preparation procedures and compensatory measures for the affected conditions before the onset of and during adverse weather conditions. The inspectors conducted walkdowns of plant equipment and reviewed operating procedures to ensure that equipment important to safety would not be adversely affected by severe weather conditions. The documents reviewed are listed in Attachment 1. The following risk-significant systems that were required to be protected from adverse weather conditions were selected and collectively they represent one inspection sample of risk-significant systems:

- Emergency diesel generators, emergency diesel generator lube oil system, and 480 volt switchgear.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02 - 18 Samples)

a. Inspection Scope

The inspectors reviewed six safety evaluations completed during the previous two year period. The safety evaluations were completed by Entergy to evaluate if proposed changes to the facility or procedures described in the UFSAR, or changes to tests or experiments not described in the UFSAR required NRC approval prior to implementation in accordance with the requirements of 10 CFR 50.59. The safety evaluations reviewed were distributed among the Initiating Event, Mitigating System, and Barrier Integrity cornerstones. The inspectors reviewed the selected safety evaluations to verify that Entergy had appropriately concluded that the changes and tests could be accomplished without prior NRC approval in accordance with 10 CFR 50.59 and, if prior approval was required, it was obtained prior to implementing the change. Additionally, the inspectors verified that safety issues pertinent to the changes were properly resolved or adequately addressed. The following safety evaluations were reviewed:

- Develop New Fuel Design - Westinghouse 15x15 Upgraded Fuel Design;
- Indian Point Unit 2 Cycle 17 Reload Core Design Change;
- Operation of Feedwater Bypass BFP-90 Series and 417L Series Valves at Stretch Power Uprate;
- Installation of a Temporary Roll-up Door on the Containment Equipment Hatch;
- Increase in Tave from 562 degrees to 565 degrees F; and
- Electrical Separation Design Criteria.

The inspectors also reviewed 12 screened "out of scope" evaluations for changes, tests and experiments for which Entergy determined that safety evaluations were not required. This review was performed to verify that Entergy's threshold for performing safety evaluations was consistent with 10 CFR 50.59. In addition, the inspectors reviewed the administrative procedures that were used to control the screening, preparation, and issuance of the safety evaluations to ensure that the procedure adequately covered the requirements of 10 CFR 50.59. The listing of screened-out evaluations and documents reviewed is provided in Attachment 1.

b. Findings

Introduction: The inspectors identified a Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," for failure to obtain a license amendment pursuant to 10 CFR 50.90 prior implementing a change altering the requirements of the containment equipment hatch. Entergy incorrectly credited the use of a roll up door in place of the containment equipment hatch or closure plate to fulfill the requirements of TS action statements.

Description: The inspectors reviewed safety evaluation 04-0732-MD-00-RE R1, “Installation of a Temporary Roll-up Door on the Containment Equipment Hatch,” to determine if the conclusion that a license amendment was not required was correct. Entergy’s safety evaluation was performed to assess the adequacy of using a roll up door to meet the requirements of the TS action statements 3.9.4.A.4 and 3.9.5.B.3. The action statements are applicable in Mode 6 (Refueling Operations) and state that if the residual heat removal (RHR) system is not operable (Loss of decay heat removal event) “Ensure equipment door or closure plate is properly installed” within four hours. Entergy concluded that the roll-up door was equivalent to the closure plate and, therefore, adequate to close containment as required by the action statement. The TS basis and the Abnormal Operating Procedure (AOP) AOP-RHR-1, "Loss of RHR," were changed to allow the use of the roll-up door in place of the closure plate.

The inspector's review noted that the closure plate was added to the TS by License Amendment 103 (November 1985). This license amendment and NRC safety evaluation stated that the temporary hatch was able to withstand 3 pounds per square inch differential across the door without failure. The closure plate requirements were placed in the UFSAR as a result of the amendment. The inspectors noted that the amendment was used to change TS 3.9.1 related to a fuel handling accident and did not address requirements for loss of decay heat removal. The inspectors reviewed License Amendment 211 (June 2003) which replaced the unit's TS with Improved Technical Specifications (ITS). The standard ITS had credited only the containment equipment hatch to complete action statements 3.9.4.A.4 and 3.9.5.B.3, but the amendment added the use of the closure plate to meet the requirements. Entergy discussed in the amendment request that this change was acceptable based on previous licensing basis. The change was approved by the NRC. The inspectors determined that the only licensing and design basis for the closure plate was based on License Amendment 103 and, therefore, any change that affected the action statement must meet this design requirement.

The inspectors reviewed the basis for considering the roll-up door as equivalent to the closure plate. The modification description stated that the door would serve no safety function and was designed to be fire retardant and weather proof. Additionally, it would limit inflow and outflow of air from the containment building. The inspectors observed that Entergy concluded that the roll-up door was acceptable because operators would take mitigating actions to combat the loss of decay heat removal and, therefore, the containment structure would not be required to prevent a radioactive release to the public. The inspectors concluded that because the door was not designed to be air-tight any radioactive release inside containment would bypass the roll-up door and escape to the environment. In addition, the door was not designed to hold pressure in containment (such as the 3 psid design basis pressure described in License Amendment 103) that would result from the boiling of coolant during the event.

<u>Analysis:</u> The performance deficiency associated with this finding is that Entergy changed the requirements for a fission barrier (containment) without prior NRC approval. Entergy incorrectly concluded that a license amendment was not required prior to changing the definition of terms in the TS action statements. Because the safety evaluation concluded that a license amendment was not required, Entergy did not allow the NRC to review the change prior to implementation. As a result, traditional enforcement applies because the failure to submit a license amendment affected the NRC's ability to perform its regulatory function. In accordance with the NRC Enforcement Policy Supplement 1, example D.5, violations of 10 CFR 50.59 that result in conditions having very low safety significance (Green) by the significance determination process, are considered Severity Level IV. The inspectors evaluated the significance of this finding using the significance determination process described in Inspection Manual Chapter (IMC) 0609, Appendix H, "Containment Integrity," because the finding represented a potential open pathway in the physical integrity of reactor

containment. The finding was considered a Type B finding (related to a degraded condition that has potentially important implications for large early release frequency) during shutdown conditions. The inspectors determined that this finding was of very low safety significance (Green) because Entergy credited the roll up door as a containment boundary after more than eight days had elapsed following plant shutdown.

Enforcement: 10 CFR 50.59 states that a license amendment pursuant to 50.90 is required prior to implementing a change when a design basis limit for fission product barriers, as described in the UFSAR is exceeded or altered. Contrary to this requirement, on October 27, 2004, Entergy changed the requirements for the fission barrier (containment) via their 50.59 Safety Evaluation process which was credited and used during the refueling outage in April 2006 as a method to meet the TS action statements 3.9.4.A.4 and 3.9.5.B.3 without prior approval by the NRC. Entergy was operating in Mode 1 and was in compliance with TS when this issue was identified (TS only applies in Mode 6). Additionally, Entergy entered the issue into their corrective action program (IP2-2006-06405) with a planned action to revise plant procedures to prevent recurrence. Therefore, the finding is being treated as a Severity Level IV, non-cited violation consistent with Section V1.A.1 of the NRC Enforcement Policy: **(NCV 05000247/2006005-01, Inadequate Containment Closure Equipment)**

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

.1 Quarterly Inspection

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 in order to verify that the alignment of the available train was proper to support its required safety functions. The inspectors also reviewed applicable condition reports and work orders to ensure that Entergy had identified and properly addressed equipment discrepancies that could potentially impair the capability of the available train, as required by 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action.” The documents reviewed are listed in Attachment 1. The inspectors performed a partial walkdown on the following systems which represented four inspection samples:

- Gas turbine 3 while gas turbine 1 was out of service for maintenance;
- 21 and 22 coolant charging pumps while 23 coolant charging pump was out of service;
- 23 emergency diesel generator following restoration from maintenance; and
- 21 and 22 auxiliary boiler feed pumps while 23 auxiliary boiler feed pump was out of service.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Inspection

a. Inspection Scope

The inspectors performed a complete system alignment inspection on the auxiliary feedwater system (AFW) to determine whether the system was aligned and capable of providing feedwater to the steam generators for decay heat removal in accordance with the design basis requirements and TS. The inspectors reviewed operating procedures, surveillance test results, piping and instrumentation drawings, equipment lineup checkoff lists and the UFSAR to determine if the system was aligned to perform its safety function. In addition the inspectors walked down all accessible system components to verify alignment and evaluate material condition. The inspectors reviewed a sample of condition reports and work orders written for deficiencies associated with the auxiliary feedwater system to ensure they had been evaluated and resolved consistent with the requirements of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action.” The documents reviewed are listed in Attachment 1. The walkdown of the auxiliary feedwater system represents one semi-annual inspection sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 16 samples)

a. Inspection Scope

The inspectors conducted a tour of the 16 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.k. The documents reviewed are listed in Attachment 1. This inspection represented 16 inspection samples for fire protection tours. The areas inspected included:

- Fire Zone 262;
- Fire Zone 1;
- Fire Zone 1A;
- Fire Zones 31A and 23A;
- Fire Zones 74A and 74B;

- Fire Zone 370;
- Fire Zones 55 and 55A;
- Fire Zone 62A;
- Fire Zone 25;
- Fire Zone 10;
- Fire Zones 5, 5A, 6 and 6A;
- Fire Zones 8, 8A, 9A, 10A and 11A;
- Fire Zone 15;
- Fire Zone 14;
- Fire Zones 11, 12, 13, and 24; and
- Fire Zones 2 and 2A.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 2 samples)

a. Inspection Scope

The inspectors reviewed Indian Point Nuclear Generating Unit 2's Individual Plant Examination of External Events and the UFSAR concerning both internal and external flooding events. The inspection included a walkdown of accessible areas of the plant to look for potential susceptibilities to external and internal flooding and to verify the assumptions included in the site's flooding analysis. The inspectors also reviewed relevant abnormal operating and emergency plan procedures. The documents reviewed are listed in Attachment 1. These activities represented two internal flooding inspection samples. The following areas were reviewed:

- Turbine building, 15 foot elevation; and
- Primary auxiliary building, 15 foot elevation.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A - 1 sample)

a. Inspection Scope

The inspectors performed a review of the 21 emergency diesel generator (EDG) jacket water and lube oil coolers to verify that Entergy is maintaining the heat exchangers in accordance with their commitments to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," concerning heat exchanger inspection and testing. The inspectors reviewed recent visual inspection reports and eddy current results to validate that the inspections and testing are in accordance with approved plant procedures and industry guidance and that acceptance criteria were appropriate. The heat exchangers were walked down to observe their material condition and verify the expected system indications. The documents reviewed are listed in Attachment 1. The inspection of the 21 EDG heat exchangers constituted one inspection sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On November 14, 2006, the inspectors observed licensed operator simulator training to assess operator performance during an emergency planning exercise 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 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, "Operators' Licenses." The documents reviewed are listed in Attachment 1. This observation of operator simulator training constituted one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 3 samples)

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 in accordance with 10 CFR 50.65;
- 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 are listed in Attachment 1. The following three maintenance rule samples were reviewed:

- Main feedwater regulating and low flow bypass valves;
- 440 volt alternating current (VAC) system; and
- Gas turbine system.

b. Findings

Introduction. The inspectors identified a Green finding, in that, corrective actions were inadequate to resolve a deficiency associated with the gas turbine 1 (GT-1) starting diesel. This deficiency was identified following a failure of GT-1 to start on February 7, 2005, and resulted in three subsequent failures, the last occurring August 6, 2006.

Description. On February 7, 2005, GT-1 failed to start during routine maintenance due to a trip of the starting diesel on low coolant level in its associated heat exchanger. This failure was documented in CR IP2-2005-00698 and an apparent cause evaluation was conducted. It was found that a design deficiency associated with the starting diesel heat exchanger made it difficult to ensure a proper coolant system purge following maintenance. This resulted in a void space of approximately nine gallons after the system was filled. A corrective action was written to ensure that steps were added to either the work order step list associated with filling and venting of the heat exchanger, or to operations procedures to provide additional guidance to properly purge the system after maintenance. This corrective action was closed on June 22, 2005, with no actions taken based on a senior management decision to cancel preventive maintenance activities on the gas turbines due to pending system retirement.

The inspectors reviewed condition reports related to the gas turbine system and determined that GT-1 had failed to start due to low starting diesel coolant level three additional times following the original identification of the problem. These failures occurred on April 14, 2005, March 22, 2006, and August 6, 2006. In each instance a condition report was generated but no additional corrective actions were taken to correct this specific problem.

Analysis. The inspectors determined that failure to provide adequate corrective actions to correct the identified condition was a performance deficiency and did not meet the requirements of Entergy procedure EN-LI-102, “Corrective Action Process.” 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.

This finding was determined to be more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, it impacted GT-1 reliability, in that, the deficiency resulted in multiple failures to start on demand after the condition was identified and the action to correct the condition was closed without being implemented. The gas turbines are credited as an alternate electric power source for both station blackout and Appendix R fire scenarios. 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," and determined that a Phase 2 evaluation was required because the finding represented an actual loss of safety function of a non-Technical Specification required train of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours.

The inspectors used the Risk-Informed Inspection Notebook for Indian Point Nuclear Generating Unit 2, to conduct the Phase 2 evaluation. The inspectors determined that 65 hours of unavailability were caused by the additional failures of GT-1 due to the starting diesel coolant system deficiency. The inspectors conservatively equated this cumulative unavailability time to the total exposure time and used an initiating events likelihood of less than three days. The Phase 2 approximation yielded a result of very low safety significance (Green). The most dominant accident sequence involved a loss of off-site power, combined with the subsequent failure of two emergency diesel generators, and the failure to restore power within 5 hours. Entergy has placed this issue in their corrective action program (CR-IP2-2006-06842).

The inspectors determined this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that equipment and resources were available and adequate to assure reliable operation of GT-1. Specifically, Entergy did not minimize long-standing equipment issues and maintenance deferrals associated with the gas turbine system.

Enforcement. No violation of regulatory requirements occurred. The inspector determined that the finding did not represent a noncompliance since the gas turbine system is a non-safety related system. Entergy entered this issue into the corrective action program (CR-IP2-2006-6842) and installed a modification to the coolant system to prevent further trips due to this condition: **(FIN 05000247/2006005-02, Failure to Implement Corrective Actions to Correct a Degraded Condition Which Impacted Gas Turbine #1 Reliability and Availability)**

c. Unresolved Item

Introduction. The inspectors identified an Unresolved Item (URI) associated with acceptability of the gas turbine system performance under the Maintenance Rule. Inspectors identified that Entergy may not have monitored the gas turbine system in a manner to provide reasonable assurance that the system could perform its intended safety function of mitigating the effects of a station blackout (SBO). Additional information is needed to determine whether the gas turbine system performance has impacted overall system functionality, and has met the assumptions in licensing basis documentation.

Description. The inspectors identified the URI during a routine Maintenance Rule inspection on the gas turbine system. Gas turbines 1 and 3 (GT-1 and GT-3) are credited in Entergy's analysis to cope with station blackout and Appendix R fire scenarios to ensure safe shutdown of the reactor. The system is classified as risk-significant in accordance with Entergy's Maintenance Rule program. This system has been in a category (a)(1) monitoring status since the inception of the Maintenance Rule in 1996.

An (a)(1) action plan was established to improve overall system performance. However, Entergy may not have provided justifiable (a)(1) goals for maintenance preventable functional failures (MPFF's) and Entergy may not have appropriately classified repeat maintenance preventable functional failures (RMPFF's). Specific to reliability, the goal was set as less than or equal to five MPFF's and no RMPFF's in a 24 month rolling cycle. The number of allowable MPFF's was calculated under the assumption that there would be, on average, 82 start demands during the 24 month cycle. The inspectors reviewed the operating history over the last three years and determined that the number of start demands averaged 38 during the 24 month cycle. The inspectors need more information to evaluate Entergy's goals for MPFF's to determine their adequacy.

Additional information is required to evaluate Entergy's implementation of the Maintenance Rule as it pertains to the gas turbine system. Actual unavailability and reliability information is needed to evaluate the gas turbine system performance and to assess whether performance of the system is bounded by the Station Blackout / Appendix R commitments, and assumptions in the design basis. This issue will be treated as a URI pending additional licensee input and inspector evaluation of gas turbine system performance: **(URI 05000247/2006005-03, Assess Reliability / Unavailability of the Gas Turbine System and Impacts on Functionality)**

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

a. Inspection Scope

The inspectors reviewed the following five activities to verify that the appropriate risk assessments were performed prior to removing equipment for 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 are listed in Attachment 1. The following activities represent five inspection samples:

- WO IP2-06-01073, generex alarm card calibration;
- WO IP2-06-32508, 21 main boiler feed pump steam admission valve;
- WO IP2-04-31641, turbine first stage pressure calibration;
- WO IP2-06-33010, 22 steam generator blowdown line repair; and
- WO IP2-05-19600, 21 boric acid transfer pump maintenance.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR Part 50.65(a)(4), because Entergy did not adequately assess and manage the risk of performing online maintenance activities while operating with a degraded steam inlet valve on 21 main boiler feed pump (MBFP).

Description. During the plant startup that occurred on November 16, 2006, operators attempted to start the 21 MBFP on high pressure steam. During the startup, 21 MBFP was unable to reach full speed. Entergy subsequently determined that the high pressure steam inlet valve would not fully open. Entergy has two turbine-driven MBFPs that are each designed to support 70 percent of full power operation. Both MBFPs are required to operate at full power. The turbines have a high pressure steam inlet valve which is opened to rotate the turbine during startup and a lower pressure reheat steam inlet valve that is opened to rotate the turbine during normal operations. High pressure steam is required during startup because reheat steam, which is extracted from various stages of the main turbine exhaust, is not available until main turbine startup occurs. Subsequently, if the plant trips, reheat steam is also lost and the 21 MBFP would not have been available to supply water to the steam generators as designed due to the degraded high pressure steam inlet valve. Entergy determined that the degraded condition of 21 MBFP was a direct result of maintenance which was conducted during the refueling outage that ended in May 2006. Entergy also concluded that the 21 MBFP could perform its full power function since adequate reheat steam was available at full power. However, Entergy determined that 21 MBFP would not be able to perform its function on a loss of 22 MBFP because less reheat steam is available at the resultant lower power levels. Recognizing this, Entergy implemented a temporary procedure change to manually trip the reactor upon a loss of 22 MBFP.

The inspectors reviewed procedure SOP-SD-09, “Online Risk Assessment Process.” The inspectors determined that operating at full power with 21 MBFP in its degraded state presented an increased trip risk to the plant which should have been assessed in accordance with the plant’s online risk assessment process. The inspectors questioned Entergy about the potential effect to online risk, and Entergy subsequently included the risk increase associated with the degraded MBFP in their online risk model on November 21, 2006, and entered the issue into their corrective action program (CR-IP2-2006-06670).

Analysis. The inspectors determined that Entergy’s failure to adequately assess and manage online maintenance activities associated with the degraded 21 MBFP was a performance deficiency which was reasonably within Entergy’s ability to foresee and prevent. The inspectors determined that this finding impacted the Initiating Events Cornerstone. Traditional enforcement does not apply since there was no actual safety consequence or potential for impacting the NRC’s regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy procedures.

The inspectors determined that this finding was more than minor because Entergy's risk assessment failed to consider risk-significant systems, structures, components, and support systems that were unavailable during on-line maintenance. Specifically, the increased reactor trip risk due to the 21 MBFP degraded condition was not included in Entergy's on-line risk assessment from November 16 through 21, 2006. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, Appendix K, "Maintenance Risk Assessment And Risk Management Significance Determination Process." The increase in the likelihood of an initiating event was determined to be very low because the probability of failure for an operating turbine is low. The inspectors determined that this finding was of very low safety significance because the finding resulted in an increase in the incremental core damage probability of less than $1x10^{-6}$ (actual increase was approximately $2x10^{-8}$).

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not provide complete and accurate procedures, in that, the online risk assessment procedure did not require degraded equipment that impacted risk to be assessed or managed.

Enforcement. 10 CFR Part 50.65(a)(4), requires, in part, that Entergy assess and manage the increase in risk that may result from proposed maintenance activities. Contrary to the above, from November 16 through 21, 2006, Entergy did not adequately assess and manage the increased risk of operating with the degraded 21 MBFP at full power prior to performing online maintenance. Entergy properly assessed and incorporated into their risk model the increased risk of operating the degraded 21 MBFP on November 21, 2006. Because this issue is of very low safety significance and is entered into Entergy's corrective action program (CR-IP2-2006-06670), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **(NCV 05000247/2006005-04, Inadequate Risk Assessment for 21 MBFP steam inlet valve)**

1R15 Operability Evaluations (71111.15 - 4 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to assess the acceptability of the evaluations; when needed, 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 procedure 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 are listed in Attachment 1. The following evaluations were reviewed and represent four program samples:

- CR IP2-2006-01587, vapor containment sump level instrumentation and reactor coolant system leakage detection capability;
- CR IP2-2006-06732, service water pump area discharge check valve;
- CR IP2-2006-06839, containment bypass leakage through main steam system; and
- CR IP2-2006-06960, vapor containment sump due to check valve back-leakage

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A - 1 sample / 71111.17B - 8 samples)

1. Annual Inspection

a. Inspection Scope

The inspectors reviewed a modification associated with replacement of the Indian Point Nuclear Generating Unit 2 toxic gas monitoring system. This modification was implemented using ER-06-2-020, "IP2 [Indian Point Unit 2] Toxic Gas Monitors Replacement," to address the obsolescence of system components. The inspectors reviewed the modification package to ensure it was technically adequate and conducted walkdowns of the modification to verify it was completed in accordance with the design. The UFSAR was reviewed for the system design and licensing basis. The inspectors reviewed calculations for the seismic mounting of the new components and the changes to electrical power loading for the safety related inverters and batteries. The inspectors evaluated their observations against the requirements of 10 CFR Part 50.59, "Changes, Tests, and Experiments;" 10 CFR Part 50, Appendix B; and Technical Specifications. The documents reviewed are listed in Attachment 1. The review of this modification represents one sample.

b. Findings

No findings of significance were identified.

2. Biennial Inspection

a. Inspection Scope

The inspectors reviewed eight risk-significant plant modification packages selected from the design changes performed on systems associated with the Initiating Events, Mitigating Systems and Barrier Integrity cornerstones within the past two years. The inspectors reviewed the selected modifications to verify that the design bases, licensing bases, and performance capability of the risk significant structures, systems and components had not been degraded as a result of the modifications. Additionally, the inspectors assessed whether the modifications had adversely affected the availability, reliability, or functional capability of the system or associated interface systems. The following modifications were selected for review and represent eight inspection samples:

- Replace Bergen Paterson hydraulic snubbers with Lisega snubbers;
- EDG service water piping replacement;
- Power uprate setpoint changes;

- Replace steam flow instrument loop power supplies with combined direct current power and scaling modules;
- Roll up door outside 95' elevation equipment hatch;
- Replacement of 118 VAC instrument bus 22 breakers;
- Replacement of 118 VAC instrument buses 21 and 24 breakers; and
- Replacement of reactor coolant system narrow range temperature measuring system.

For the modifications selected, the inspectors verified that systems potentially affected by the modification remained consistent with the design and licensing basis. The inspectors reviewed a variety of parameters to determine if the modification had impacted either of these bases. The parameters reviewed included electrical, steam, fuel or air requirements; replacement component and materials compatibility and qualification; adequate heat removal capacity; automatic and manual control signal for startup, shutdown and control; external and internal hazards protection such as flooding, fire, freeze protection, high energy line break and missile; pressure boundary and ventilation boundary integrity; structural integrity; process medium design parameters such as voltage, current fluid flow and pressure; and potential failure modes. The parameters were reviewed to verify that they were technically appropriate and consistent with the UFSAR and associated design basis documents.

The inspectors reviewed the post-modification testing, functional testing, and instrument calibration records to determine readiness for operations. This review included verifying that the modification did not create unintended system interactions; SSC performance characteristics were not affected by the modification; original modification design assumptions were correct; and the modification test acceptance criteria were appropriate and had been met. Additionally, the inspectors verified that the timing sequence was correct and response time limits had not been exceeded.

The inspectors also reviewed the affected procedures, drawings, design basis documents, supporting calculations, analysis and relevant UFSAR sections to verify that the affected documents had been appropriately updated. Additionally, the inspectors verified affected normal, abnormal, and emergency operating procedures, testing and surveillance procedures had been updated; as required. The inspectors reviewed operator training records to assess if appropriate training had been conducted on the modification. The inspectors verified that necessary TS changes had been identified and, if NRC approval was required, it was obtained prior to performing the modification.

The inspectors reviewed selected condition reports associated with the modification process and design change notices that were issued during the installation. The inspectors verified that the problems associated with the installation were adequately resolved and that conditions adverse to quality identified by Entergy's processes had been appropriately corrected. The documents reviewed are listed in Attachment 1.

b. <u>Findings</u>

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed seven 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 are listed in Attachment 1. The following post maintenance test activities were reviewed and represent seven inspection program samples:

- WO-IP2-06-29695, gas turbine 1 following maintenance;
- WO-IP2-05-19407, 21 boric acid transfer pump following maintenance;
- WO-IP2-05-20791, 23 coolant charging pump following fluid drive replacement;
- WO-IP2-06-27470, isolation valve seal water filter 5804 following replacement;
- WO-IP2-06-27219, 21 emergency diesel generator following air start motor replacement;
- WO-IP2-06-32689, auxiliary feed water valve 405D following maintenance; and
- WO-IP2-06-33033, non-destructive testing of valves MS-86-1 and MS-68-2 installation welds.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors observed and reviewed activities during two Indian Point Nuclear Generating Unit 2 forced outages. The first outage occurred between November 15 and 17, 2006, following a reactor trip due to a problem with the generator excitation system. The second outage occurred between November 30 and December 1, 2006, following a plant shutdown to correct a steam leak on the 22 steam generator blowdown piping. The documents reviewed are listed in Attachment 1. The following activities were reviewed for each forced outage which represented two inspection program samples:

- The inspectors reviewed outage schedules and procedures, and verified that TS required safety system availability was maintained, shutdown risk was considered, and that contingency plans existed to restore key safety functions such as electrical power and containment integrity, as required.

- The inspectors conducted a containment walkdown on the initial entry into the vapor containment and conducted a closeout tour of the containment prior to plant restart.

- The inspectors observed portions of the reactor startup following the outage, and verified through plant walkdowns, control room observations, and surveillance test reviews that safety-related equipment required for mode change was operable, that containment integrity was set, and that reactor coolant boundary leakage was within TS limits.

b. <u>Findings</u>

No findings of significance were identified.

1R22 <u>Surveillance Testing</u> (71111.22 - 3 samples)

a. <u>Inspection Scope</u>

The inspectors witnessed performance of surveillance tests and/or reviewed test data of selected risk-significant systems, structures, and components to assess whether the systems, structures, and components 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 are listed in Attachment 1. The following surveillance tests were reviewed and represented three inspection program samples:

- SOP-LEAKRATE-001, “RCS Leakrate Surveillance, Evaluation and Leak Identification,” Revision 0;
- 2PT-Q034, “22 ABFP,” Revision 22; and
- 2PT-Q28B, “22 RHR Pump,” Revision 17.

b. <u>Findings</u>

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

A region-based specialist inspector conducted an inspection of Entergy's corrective actions related to the existing Indian Point Alert and Notification system (ANS) failures, and also reviewed the progress made in the design and installation of the new siren system. The inspection was conducted onsite October 3 through 6 and November 13 through 17, 2006, per the baseline inspection program deviation authorized by the NRC Executive Director of Operations in a memorandum dated October 31, 2005.

The inspector was onsite the first week of October to assess Entergy's response to the September 19, 2006, loss of siren event which occurred as the result of the computer software database failing to reconnect following a preventive maintenance reboot of the siren system computer. This event involved a failure of the automatic startup sequence following the reboot, and although the automatic startup failed, manual rebooting of the ANS computer remained available and maintained the ANS functional. The inspector reviewed aspects of the event to determine if the failure met the criteria of a significant finding, as defined in NRC Inspection Manual Chapters (IMCs) 0609, Appendix B, "Emergency Preparedness Significance Determination Process," and 0612, "Power Reactor Inspection Reports."

On October 6, 2006 Entergy and the NRC conducted a public meeting in Buchanan, New York, during which Entergy discussed additional corrective actions to be taken to assure the proper operation and maintenance of the existing siren system and the progress in the design and installation of the new siren system. Entergy submitted a letter to the NRC on October 18, 2006, documenting these additional corrective actions. The inspector reviewed the planned corrective actions to verify they were appropriate to address the siren failures which had occurred.

The inspector returned to the site in November to assess Entergy's compliance with and implementation of the corrective actions. The inspector observed the biweekly re-boot of the current system's control computer and reviewed the log books of the technicians responsible for the "around-the-clock" monitoring of the current system. The inspector also reviewed the circumstances of a November 9, 2006, event that involved the loss of the Entergy's ability to actuate 13 of 156 sirens for approximately 30 minutes, due to a maintenance technician opening the antenna connection on a specific siren. The inspector reviewed the condition report for the event and discussed it with members of the Indian Point emergency preparedness staff, to determine if this failure met the criteria of a significant finding, as defined in NRC IMC 0609, Appendix B, and IMC 0612.

The inspector interviewed the senior project manager and the nuclear information technology manager for the new siren system to understand Entergy's progress towards meeting the milestone dates required by the NRC's Confirmatory Order dated

January 31, 2006. While on site, the inspector reviewed the progress of Entergy's installation of the new siren system components, especially to understand Entergy's plans for addressing the remaining challenges in pole/siren and radio communication tower installation. The inspector also reviewed Entergy's progress in obtaining Department of Homeland Security approval of the Indian Point Energy Center Prompt Alert and Notification System Design Report.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

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

a. Inspection Scope

During December 20 through 29, 2006, the inspector conducted the following activities to verify that Entergy was properly implementing physical, engineering, and administrative controls for access to high radiation areas, and other radiologically controlled areas, and that workers were adhering to these controls when working in these areas. Implementation of the access control program was reviewed against the criteria contained in 10 CFR 20, site Technical Specifications, and Entergy's procedures. The documents reviewed are listed in Attachment 1.

(1) There were no radiation work permits for airborne radioactivity areas with the potential for individual worker internal exposures of >50 mrem committed effective dose equivalent (CEDE).

(2) During 2006 there were no internal dose assessments for any actual internal exposures greater than 50 mrem CEDE.

(3) Entergy's physical and programmatic controls for highly activated materials stored underwater in the spent fuel pools were reviewed and evaluated through walkdowns and observations in these areas.

(4) A review of Entergy's radiation protection program self-assessments and audits during 2006 was conducted to determine if identified problems were entered into the corrective action program for resolution.

(5) Seventeen condition reports associated with the radiation protection access control and as low as reasonably achievable (ALARA) areas between January 2006 and December 2006, were reviewed and discussed with Entergy staff to determine if the follow-up activities were being conducted in an effective and timely manner commensurate with their safety significance.

(6) Based on the condition reports reviewed, repetitive deficiencies were screened to determine if Entergy's self-assessment activities were identifying and addressing these deficiencies.

(7) There was one occupational exposure performance indicator incident reported during the current assessment period. This was associated with installation of the lower core barrel assembly during the Spring 2006 Unit 2 refueling outage and it was determined that there were no overexposures or substantial potential for overexposures.

(8) There were no significant dose gradients requiring relocation of dosimetry for the radiologically significant jobs observed during this inspection.

(9) Changes to the high dose rate, high radiation area and very high radiation area procedures since the last inspection in this area were reviewed, and management of these changes was discussed with the Radiation Protection Manager.

(10) Controls associated with potential very high radiation areas that included reactor core flux monitor calibration thimble withdrawal and coordination with plant operations prior to allowing personnel entry into the reactor cavity sumps was discussed with duty watch radiation protection technicians.

(11) All accessible locked high radiation area entrances were verified to be locked through challenging the locks or doors.

(12) Several radiological condition reports (see Section 4OA2) were reviewed to evaluate if the incidents were caused by radiation worker errors, determine if there were any trends or patterns, and if Entergy's corrective actions were adequately addressing these trends.

(13) Radiation protection technicians work performance was evaluated with respect to their knowledge of the radiological conditions, the specific radiation protection work requirements and radiation protection procedures.

(14) Several radiological condition reports (see Section 4OA2) were reviewed to evaluate if the incidents were caused by radiation protection technician errors and determine if there were any trends or patterns and if Entergy's corrective actions were adequately addressing these trends.

b. Findings

Introduction. A Green, self-revealing NCV of 10 CFR 20.1501 with respect to 10 CFR 20.1902(b) was identified, in that, Entergy failed to survey radiological condition changes after a plant manipulation that was likely to cause a change in radiological conditions, and this led to the failure to post a plant area as a high radiation area. As a result, two workers were allowed access to an unsurveyed and unposted high radiation area.

Description. On August 27, 2006, two nuclear plant operators (NPOs) were briefed by radiation protection personnel prior to entering the Unit 1 42 foot elevation pipe chase (a posted radiation area), where dose rates were expected to be between 5 and 10 mrem/hour. Upon entering the area, the NPOs noticed that their electronic dosimeters indicated dose rates between 75 and 90 mrem/hour. Subsequently, the individuals left the area and reported the condition to radiation protection personnel. The two workers received exposures of 26 and 13 mrem for this entry.

A follow-up survey of the Unit 1 42 foot elevation pipe chase revealed a liquid waste transfer pipe measuring 1300 mrem/hour at contact and between 150 and 250 mrem/hour at 30 centimeters through a large portion of the pipe chase. Entergy subsequently re-posted the area and controlled it as a high radiation area. The inspector confirmed that in addition to an incorrect briefing on radiological conditions, the workers were not on a radiation work permit that allowed access to high radiation areas and the area they were permitted entry into was not posted or controlled as a high radiation area.

The inspector determined that during the previous week, a waste transfer of sludge occurred that plugged the Unit 2 filter demineralizer system and required corrective maintenance. This activity involved the transfer and relocation of highly radioactive material in the associated filter demineralizer system piping and indicated a change of plant radiological conditions had occurred prior to the August 27, 2006, entry to the pipe chase area.

Entergy entered this issue into their corrective action program (CR-IP2-2006-05143). Entergy immediately posted the area as a high radiation area, and provided training to radiation protection personnel about the event, and the operation of the liquid waste processing system.

Analysis. The failure to survey an area of the plant after a known change of plant radiological conditions had occurred prior to allowing personnel access to an unposted high radiation area is a performance deficiency and resulted in an inappropriate radiological briefing of the workers; the use of an inappropriate radiation work permit; and providing access of personnel to an unposted high radiation area. Entergy procedure, EN-RP-101,"Access Control for Radiologically Controlled Areas," Revision 1, requires that specific monitoring and radiological controls for access to radiologically controlled areas shall be made by radiation protection personnel. In this instance, no specific monitoring of the Unit 1 42 foot elevation pipe chase had been performed after the waste transfer event had occurred and no high radiation area radiological controls had been established prior to allowing personnel access.

The finding is more than minor because it is associated with the Occupational Radiation Safety cornerstone attribute of exposure control and affected the cornerstone objective, because not establishing radiological conditions and commensurate controls after changing plant radiological conditions prior to allowing access to the affected areas can cause increased personnel exposure. The inspectors evaluated this finding using IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," and determined that it was of very low safety significance (Green) because it did not involve ALARA planning and controls, an overexposure, a substantial potential for overexposure, or an impaired ability to assess dose. This issue was entered into Entergy's corrective action program and training was provided to the radiation protection staff.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not use a conservative assumption in the decision-making process, in that, the watch radiation protection technician did not question the radiological conditions of the pipe chase area after a change of plant conditions had occurred and did not require a survey of the pipe chase area before authorizing access to personnel.

Enforcement. 10 CFR 20.1501 states, in part, that licensee's shall make such surveys that may be necessary to comply with 10CFR 20.1902(b) (posting of high radiation areas), and are reasonable under the circumstances to evaluate the magnitude and extent of radiation levels and the potential radiological hazards. Contrary to this requirement, on August 27, 2006, Entergy failed to survey the Unit 1 42 foot elevation pipe chase after a known change of plant radiological conditions had occurred and allowed access to an unposted high radiation area. Because the failure to survey and post a high radiation area while allowing personnel access, was determined to be of low safety significance (Green) and was entered into Entergy's corrective action program as CR-IP2-2006-05143, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy, NUREG-1600. **(NCV 05000247/2006005-05, Failure to survey and provide access to an unposted high radiation area)**

2OS2 ALARA Planning and Controls (71121.02 - 7 samples)

a. Inspection Scope

During December 20 through 29, 2006, the inspector conducted the following activities to verify that Entergy was properly maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and Entergy's procedures. The documents reviewed are listed in Attachment 1.

(1) Site specific source term trends in collective exposures and source-term were reviewed, indicating an increasing trend reflecting higher than average pressurized water reactor radiation levels and an increasing trend in collective exposures for Unit 2. Unit 3 exposure and source-term reflect lower than average PWR collective exposures and source-term.

(2) The collective exposure results from the Spring 2006 Unit 2 refueling outage were compared to the applicable ALARA planning dose estimates and evaluated for any dose overruns and applicable causes.

(3) The assumptions and basis for the 2007 annual exposure estimates was reviewed based on applicable procedures. These estimates included both dose rate and man-hour estimate calculations.

(4) Source-term data was reviewed to assess an increasing trend from 2003 through 2006. Interviews were conducted with the ALARA supervisor and the Radiation Protection Manager relative to reactor water chemistry and source-term controls being evaluated to reduce occupational exposure.

(5) There were three declared pregnant workers during 2006 and their exposure records and monitoring control records were reviewed.

(6) The ALARA program self-assessments and audit were reviewed to determine if Entergy's overall audit program scope and frequency met the requirements of 10 CFR 20.1101.

(7) With respect to the condition reports reviewed (see Section 4.02), any repetitive deficiencies that were identified were reviewed with respect to Entergy's self-assessment and audit program identification and resolution.

b. Findings

Introduction. A Green self-revealing finding was identified relative to the collective exposure performance of a Spring 2006 Unit 2 containment sump strainer modification that resulted in 93.7 person-rem compared to a work activity estimate of 10.9 person-rem.

Description. The Unit 2 containment sump strainer modification dose overrun was primarily due to inadequate work activity planning. Lack of in-field walkdowns prior to designing the modification resulted in a modification design that did not fit the actual plant conditions. This resulted in a significant amount of as-found interferences that required removal and reinstallation. The differences in as-found dimensions resulted in a significant amount of "fit-up" problems which required additional in-field high radiation area work. This unplanned additional in-field high radiation work resulted in significant collective exposures that could have been avoided had sufficient job planning and preparation occurred.

Analysis. The inspectors determined that the inadequate modification planning and construction preparations that resulted in significant unplanned collective exposure was a performance deficiency which was reasonably within Entergy's ability to foresee and prevent. Specifically, the actual job site conditions for installation of the containment sump modification were not adequately evaluated with respect to the radiological impact of increased occupancy in high dose rate work areas. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC’s regulatory function and the finding was not the result of any willful violation of NRC requirements or Entergy’s procedures.

The inspectors determined that this finding was more than minor because it was similar to examples 6.a and 6.b of IMC 0612, Appendix E, "Examples of Minor Issues," in that, the issue involved actual collective exposure greater than 5 person-rem and was greater than 50 percent above the estimated or intended exposure; and the majority of the dose overrun was due to activities within Entergy's control. The inspectors evaluated this finding using IMC 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” and determined that the finding was of very low safety significance (Green) because it involved an ALARA planning issue, and the 3-year rolling average collective dose for Unit 2 was less than 135 person-rem (73 person-rem average annual exposure for 2003 through 2005). This issue was entered into Entergy's corrective action program including lessons learned for the Unit 3 containment sump modification (CR-IP2-2006-02344).

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not adequately incorporate job site conditions in the work control planning process.

Enforcement. No violation of regulatory requirements occurred. The inspector determined that the finding did not represent a noncompliance because the ALARA requirement in 10 CFR 20.1101, does not require every work activity to achieve its ALARA performance goal. Although the Unit 2 containment sump strainer modification was not performed in accordance with the ALARA collective exposure objective, there was no programmatic indication of pervasive collective dose overruns; therefore, no violation of the ALARA rule was identified. Entergy entered this issue into the corrective action program (CR-IP2-2006-02344). **(FIN 05000247/2006005-06, Unit 2 containment sump strainer modification collective exposure overruns due to inadequate mod preparation)**

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151 - 3 samples)

.1 Mitigating Systems Cornerstone

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. The documents reviewed are listed in Attachment 1.

Mitigating Systems Cornerstone

- Safety System Functional Failures.

The inspectors reviewed data and plant records from March 2004 to September 2006. The records reviewed included PI data summary reports, licensee event reports, operator narrative logs, and maintenance rule records. The inspectors verified the accuracy of the number of critical hours reported, and interviewed the system engineers and operators responsible for data collection and evaluation.

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector reviewed implementation of Entergy's Occupational Exposure Control Effectiveness PI program. Specifically, the inspector reviewed CRs, and radiological controlled area dosimeter exit logs for the past four calendar quarters. These records were reviewed for occurrences involving locked high radiation areas, very high radiation areas, and unplanned exposures against the criteria specified in Nuclear Energy Institute 99-02, to verify that all occurrences that met the criteria were identified and reported. The documents reviewed are listed in Attachment 1.

b. Findings

No findings of significance were identified.

.3 Radiological Environmental Technical Specifications/ Offsite Dose Calculation Manual - Radiological Effluent Occurrences

a. Inspection Scope

The inspector reviewed a listing of relevant effluent release reports for the past four calendar quarters, for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences per site that exceed 1.5 mrem/quarter whole body or 5.0 mrem/quarter organ dose for liquid effluents; and 5.0 mrads/quarter gamma air dose, 10.0 mrad/quarter beta air dose, and 7.5 mrads/quarter for organ dose for gaseous effluents. The documents reviewed are listed in Attachment 1. The inspector reviewed the following documents to ensure that Entergy met all requirements of NEI 99-02:

- Monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases;
- Quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases; and
- Dose assessment procedures.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152 - 4 samples)

.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 corrective action program. The review was accomplished by accessing Entergy’s computerized database for CRs and attending CR screening meetings.

In accordance with the baseline inspection modules, the inspectors selected corrective action program 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, and operability determinations, and the timeliness of the specified corrective actions. These CR’s reviewed are listed in Attachment 1.

b. Findings

No findings of significance were identified.

.2 PI&R Annual Sample: Procedural Adequacy Issues (71152 - 1 sample)

a. Inspection Scope

The inspectors conducted a review of CR IP2-2006-03930, which identified a number of issues involving procedural adequacy identified by the NRC during 2006. The inspectors evaluated the corrective actions associated with this condition report to determine if the scope of the actions was sufficient to correct the identified issue and if the actions taken were effective. In addition the inspectors reviewed action plans associated with procedural improvements in Operations, Maintenance and the Instrumentation and Control (I&C) departments. The documents reviewed are listed in Attachment 1.

b. Findings and Observations

The inspectors determined that the scope of the actions plans was sufficient to address the identified concerns. The inspectors also determined that the plans for procedural improvements were adequate; however, the action plan to address procedural concerns associated with the I&C department lacked sufficient detail. No specific goals were established for the procedural improvements in this area and no specific details were provided as to how the project would be accomplished. In addition the inspectors found the time line for project completion for all actions was by end of 2007; however, sufficient resources had not yet been provided in any of the affected departments to ensure this completion date could be met.

.3 PI&R Annual Sample - Selected Issue Follow-up Inspection - Technical Support Center Diesel Fails to Start (71152 - 1 sample)

a. Inspection Scope

The inspectors reviewed Entergy's actions to resolve problems with the performance of the technical support center (TSC) diesel generator that occurred during the Northeast Blackout on August 14, 2003, and during subsequent testing of the TSC diesel on August 30, 2003. The failure of the TSC diesel to start on a loss of the normal power supply was previously reviewed and determined to be a Green Finding (NRC Inspection Report 05000247/2003013, Finding 04). The inspectors reviewed Entergy's root cause analysis and specification of corrective actions as documented in CR-IP2-2003-05475 since this finding was issued. In addition, the inspectors reviewed the issues and corrective actions described in CR-IP2-2003-05199, which documented the failure of one of two supply breakers from outside power to trip during this event. The inspectors also interviewed Design Engineering personnel, the root cause investigator, and Corrective Actions and Assessment (CA&A) personnel. Additional documents reviewed during the inspection are listed in Attachment 1.

b. Findings and Observations

The inspectors determined that Entergy’s evaluation and determination of corrective actions for this issue took an extended period of time to complete. This was mitigated since a temporary power source was installed for the duration of the evaluation, corrective actions and subsequent testing. Although the final evaluation and corrective actions were adequate, the inspectors identified several instances where corrective action responses did not provide clear and complete documentation of the actions taken to address the concerns. In one instance, the inspectors identified that a contributing factor to the TSC diesel failure was a lack of an effective preventive maintenance program; however, no evidence of a corrective action specifying improved preventative maintenance activities could be found in the corrective action documents. The inspectors determined that an evaluation of the preventative maintenance program had been completed and improvements were made; however, this had been performed outside of the corrective action process.

.4 PI&R Annual Sample - Selected Issue Follow-up Inspection - Review of Corrective Actions Associated with Five Risk Assessment NCVs issued to Indian Point Energy Center in 2006 (71152 - 1 sample)

a. Inspection Scope

The inspectors conducted a review of the effectiveness of corrective actions associated with the five NCVs issued to Indian Point Energy Center in 2006 for inadequate risk assessments during maintenance. These included NCVs:

- 50-286/2006-002-01, "Failure to Perform a Risk Assessment for Emergent Work on the IP3 Appendix R EDG;"

- 50-286/2006-002-02, "Failure to Perform a Risk Assessment for Emergent High Wind Conditions During 33 EDG Planned Maintenance;"
- 50-286/2006-003-01, "Failure to Perform a Risk Assessment for Emergent Work Performance at IP3 of N-42 Axial Offset Calibration;"
- 50-247/2006-002-04, "Failure to Risk Assess Scaffolding Construction in the Cable Spreading Room Resulting in an IP2 Reactor Trip;" and
- 50-247/2006-003-07, "Failure to Assess Maintenance Activities at IP2 on Valve SI-869A."

The inspectors interviewed the planning and operations personnel responsible for performing risk assessments, reviewed condition reports from 2006 to present which documented the issue, assessed Entergy's threshold for problem identification, the adequacy of the cause analyses, extent of condition review, and corrective actions. The documents reviewed during the inspection are included in Attachment 1.

b. Findings and Observations

No findings of significance were identified. Corrective actions have been implemented by operations to standardize risk assessment practices between Indian Point Units 2 and 3 watchstanders and to reinforce that operations watchstanders are responsible for risk assessing off-hours emergent work. However, the inspectors identified recent issues which demonstrate that problems associated with performing risk assessments for emergent work and schedule changes still exist. The inspectors identified that Entergy's corrective actions for previous NCVs did not consistently address all causal factors. Specifically, the corrective actions did not address the established administrative controls which would have required risk assessments to be performed or revised for schedule changes or emergent work. The inspectors discussed this observation with Entergy management and reviewed the adequacy of Entergy's corrective actions to address this concern.

.5 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 third and fourth quarters of 2006 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 second and third quarters of 2006 to ensure they were appropriately evaluating and trending identified conditions.

b. Findings

No findings of significance were identified.

.6 Occupational Radiation Safety Cornerstone

a. Inspection Scope

The inspector reviewed 17 condition reports associated with the radiation protection program that were initiated between January and December 2006. The inspector verified that problems identified by these condition reports were properly characterized in the corrective action program, and that applicable causes and corrective actions were identified, commensurate with the safety significance of the radiological occurrences. The documents reviewed are listed in Attachment 1.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153 - 1 sample)

Main Turbine Trip due to Generex Voltage Regulator system Troubleshooting

a. Inspection Scope

The inspectors observed control room personnel response to an unexpected reactor and main turbine trip on November 15, 2006, that resulted from troubleshooting on the Generex voltage regulator system for the main turbine generator. The inspectors observed Entergy's post-trip response in the control room to verify that plant equipment response was as expected, and to ensure that operating procedures were being appropriately implemented. The inspectors attended post-trip review and forced outage meetings, and discussed the event and corrective actions with plant management. The purpose of the reviews was to confirm that Entergy had taken appropriate corrective actions prior to commencing restart activities. The documents reviewed are listed in Attachment 1.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Temporary Instruction (TI) 2515/169, Mitigating Systems Performance Index (MSPI) Verification

a. Inspection Scope

The inspectors completed TI 2515/169, “Mitigating Systems Performance Index Verification.” The purpose of this inspection was to verify that Entergy had correctly implemented the MSPI guidance for reporting unavailability and unreliability of the monitored safety systems. On a sampling basis, the inspectors verified that Entergy had correctly identified surveillance tests or evolutions which would not be included in the MSPI due to short duration or operator recovery credit. For each MSPI system, the inspectors independently determined baseline planned unavailability hours to confirm that these hours were correctly translated into the MSPI basis document. On a sampling basis, the inspectors reviewed operator logs, condition reports, and maintenance records to verify that Entergy had accurately determined actual planned and unplanned unavailability, and system failure data. The documents reviewed are listed in Attachment 1.

b. Findings

No findings of significance were identified.

Per TI 2515/169-05 reporting requirements, Attachment 2 to this report documents additional information pertaining to the inspectors review.

.2 (Closed) URI 05000247/2000004-01, Adequacy of Hemyc Cable Wrap Fire Barrier Qualification Test and Evaluation.

Inspection Report 05000247/2000 documented the potential inadequacy of Hemyc fire barrier wrap material at Indian Point Nuclear Generating Unit 2. The issue was unresolved pending further NRC review to determine whether the qualification tests of the Hemyc fire wrap systems were acceptable. In subsequent NRC fire tests, results indicated that Hemyc/MT materials cannot be routinely relied upon as one hour fire barriers. The NRC staff has completed a significant effort informing industry of the concerns associated with these materials by issuing Information Notice (IN) 2005-07, "Results of Hemyc Electrical Raceway Fire Barrier System Full Scale Fire Testing," and Generic Letter (GL) 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations." As required by GL 2006-03, Indian Point Unit 2 has responded appropriately to the NRC concerns by identifying all applications of Hemyc/MT materials, implementing compensatory measures as appropriate and initiating corrective actions to resolve as necessary. Therefore, the NRC staff has determined that there was no performance deficiency associated with the issue and this unresolved item (URI) is closed.

.3 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 of Operations in a Reactor Oversight Process deviation memorandum approved October 31, 2005 (ADAMS Accession Number ML053010404). Accordingly, continuing oversight of licensee progress has been conducted throughout this inspection period consisting of onsite inspections, frequent review of licensee performance, progress and achievements, and periodic communications with Federal, State, and local government stakeholders.

An inspection was conducted during November 13 through 17, 2006, that focused on the Unit 1 spent fuel pool (SFP) leak to evaluate any prior opportunities of discovery or licensee deficiencies in mitigation of the current Unit 1 source of groundwater contamination on site. The inspection included a review of the performance of the Unit 1 SFP, a review of Unit 1 SFP radionuclide data, SFP leak rate calculations, and modifications to the Unit 1 SFP leak groundwater drainage system. The inspection also included review of the construction and floor plan drawings of the Unit 1 facility, physical inspection of areas and facilities, and sampling data as appropriate.

The inspections also verified licensee groundwater contamination assessment and monitoring commitments identified in Entergy's March 24, 2006 letter (NL-06-033). In addition, the NRC staff reviewed Entergy’s groundwater sampling program. The NRC Staff, with New York State Department of Environmental Conservation officials, observed groundwater sampling and protocols relative to chain-of-custody verification. Throughout the inspection period, the NRC continued to split samples of offsite, site boundary, and other selected monitoring wells with Entergy and New York State Department of Environmental Conservation to verify and confirm the accuracy of the licensee’s analytical results.

During onsite inspection activities, NRC staff met with Entergy to review the results of its pumping test using recovery well 1 (RW-1), adjacent to the Unit 2 SFP. The short-term pumping test was conducted to develop detailed information on groundwater flow characteristics relative to the application of possible containment and recovery of the contaminated groundwater in the vicinity of the Unit 2 SFP. An important part of the analysis was to determine the appropriate pumping rate in RW-1 to create a groundwater capture zone in and around the Unit 2 SFP which would not affect the groundwater migration of Strontium-90 (SR-90) contaminated groundwater in the vicinity of the Unit 1 SFP.

NRC staff reviewed Entergy’s long-term groundwater protection program, which outlines the identification and application of certain indicator monitoring wells and boundary wells to support its groundwater radiological environmental monitoring program. The objectives of the monitoring activities are to:

- Detect and quantify potential release of licensed radioactive material to adjacent properties via groundwater;
- Detect and quantify release of licensed radioactive materials to the Hudson River via groundwater;
- Provide leak detection capabilities for potential sources of groundwater contamination such as the Unit 2 SFP;
- Detect and quantify any new or emergent sources of groundwater contamination, such as a spill or leak from a radioactively contaminated component or system; or change in the site hydrology that mobilizes or exposes radioactive contamination sequestered in the soil or bed rock;
- Verify the accuracy of the characterization and hydrology of existing groundwater contamination (e.g., locations, depths, radionuclides of concern, radionuclide concentrations and migration or transfer rates are as predicted); and
- Monitor and evaluate the effectiveness of remediation or intervention actions.

b. Findings and Observations

No findings of significance were identified.

The NRC samples were analyzed by the NRC's contract laboratory, the Oak Ridge Institute for Science and Education, Environmental Site Survey and Assessment Program (ORISE/ESSAP) radioanalytical laboratory. NRC's assessment of the licensee's sample analytical results data generally indicated that the licensee's analytical contractor continued to report sample results that were consistent with NRC's analytical results. However, a discrepancy was identified with regard to certain strontium-90 (Sr-90) sample analyses. Specifically, Entergy's analytical sample results for 14 samples from 7 on-site monitoring wells, which were collected from August 1, 2006 through September 18, 2006, were not consistent with NRC sample results. In this case, the NRC identified and confirmed that the licensee's contractor reported Sr-90 groundwater concentrations that ranged from approximately 10 percent to 50 percent lower than indicated by NRC's results. NRC confirmed that its analytical results were comparable to analytical results reported by the New York State Department of Environmental Conservation.

The licensee generated a condition report in accordance with its internal corrective action program and initiated an investigation of the processes and protocols applied by its contracted analytical laboratory relative to the Sr-90 discrepancy. As part of its investigation, Entergy required its contractor to conduct its own internal investigation. In the interim, Entergy contracted the services of another independent laboratory. Aspects of this matter, including quality assurance protocols, were previously discussed in NRC Inspection Report 05000247/2006-003.

Upon completion of its investigation, Entergy concluded that, based on the information provided by their contract laboratory, the cause for the data disparity was inconclusive. Accordingly, Entergy terminated its contract with the affected contractor and initiated a new contract with a different analytical laboratory. Subsequently, the NRC analyzed additional monitoring well samples to verify the reliability of the groundwater sample database; and continues to split samples the licensee and the State of New York for selected monitoring wells.

The NRC's ORISE/ESSAP sample results are available in ADAMS under the following Accession Numbers: ML070110548, ML070110559, ML070110561, ML070110577, and ML070110602. To date, sample results from site boundary wells and offsite environmental groundwater sampling locations have not indicated any detectable plant-related radioactivity.

NRC's review of Entergy's "Pumping Test Report," which included input from New York State and U.S. Geological Survey hydrology experts, identified some differences in the interpretation of certain technical data relative to radionuclide migration. Specifically, Entergy interpreted the groundwater flow system as being fully confined and acting as a porous media. However, upon close inspection of the data, the monitoring well responses did not appear to be uniform during the pumping period, allowing the possibility that the groundwater flow system could also be viewed as indicating dual permeability properties, which may be indicative of a combination of porous media and a fracture flow system. In addition, the report provided data indicating that one of the Unit 1 monitoring wells, where Sr-90 had been detected (MW-53), indicated a substantial reduction in water level during the test which could be indicative of a possible connection to the Unit 1 Sr-90 contaminated groundwater plume. Accordingly, Entergy is considering additional pump testing, using lower flow rates over longer time periods, to more firmly establish the steady-state conditions necessary to ensure an adequate capture zone for the Unit 2 SFP while avoiding cross-contamination from the adjacent Sr-90 contaminated groundwater plume.

Entergy's pump test provided important and valuable information relative to the effect that application of the RW-1 recovery well may have on groundwater, and useful insights for possible groundwater contamination remediation strategies. The effort also provided insights for other areas that could be evaluated to assist in understanding of significant fracture flows. For example, integrated analysis of the groundwater flow system, using cross-sections between the Indian Point Units (North to South) and projecting East to the Hudson River may provide plots of encountered fracture zones, hydraulic gradients, flow directions in both the horizontal and vertical directions. Additionally, the discussions identified information from the geologic logs, cores, geophysical surveys and groundwater flow and quality data from each monitoring well that could be used in constructing cross-section diagrams of various fracture zones. Such effort would be useful for the identification of indicator and boundary monitoring wells, performance indicators, and frequency of required observations in support of the "Long-Term Groundwater Monitoring Protection Program." At present, there is still uncertainty in the vertical flow and transport conditions, and whether fracture zones or fracture sets control radionuclide concentration transport observed in the monitoring wells.

The new protocols for the groundwater sampling procedure were expected to enhance the integration and comprehensiveness of analyses. In particular, measurement to be made at the time of sampling such as turbidity, dissolved oxygen, pH, specific conductance, temperature, and depth to water following the sampling would provide valuable information in interpreting the monitoring well data.

4OA6 Meetings, including Exit

Exit Meeting Summary

On January 10, 2007, the inspectors presented the inspection results to Mr. Keith Polson and other Entergy staff members, who acknowledged the inspection results presented. Entergy did not identify any material as proprietary.

Public Meeting On Alert and Notification System Sirens

On October 6, 2006, the NRC held a public meeting where Entergy provided an update on the status of the installation of the new siren system being installed. They also provided a review of corrective actions taken and planned to improve the performance of the existing siren system.

ATTACHMENT 1: SUPPLEMENTAL INFORMATION
ATTACHMENT 2: MITIGATING SYSTEM PERFORMANCE INDEX VERIFICATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

F. Dacimo, Site Vice President
P. Rubin, General Manger of Plant Operations
E. O'Donnell, U2 Operations Manager
B. Christman, Manager of Training and Development
S. Davis, Superintendent of Operator Training
A. Singer, Operations Training Supervisor
D. Eccleston, Senior Operations Instructor
E. Goetchius, Senior Operations Instructor
D. Huntington, Senior Operations Instructor
R. Robenstein, Simulator Support Supervisor
J. Gullick, Senior Simulator Specialist
J. Rowland, Senior Simulator Specialist
T. Beasley, System Engineer
J. Kayani, System Engineer
B. Meek, Maintenance Supervisor
J. Bubniak, Senior Engineer
R. Scalone, Performance Engineering Supervisor
N. Azevedo, Code Program Supervisor
D. Gaynor, Senior Engineer
R. Lee, Senior Design Engineer
G. Dahl, Licensing Engineer
R. Mann, Programs and Components Engineer
T. Pepe, Programs and Components Engineer
J. Joy, Programs and Components Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000247/2006005-03	URI	Assess Reliability / Unavailability of the Gas Turbine System and Impacts on Functionality (Section 1R12)

Opened and Closed

05000247/2006005-01	NCV	Inadequate Containment Closure Equipment (Section 1R02)
05000247/2006005-02	FIN	Failure to Implement Corrective Actions to Correct a Degraded Condition Which

		Impacted Gas Turbine #1 Reliability and Availability (Section 1R12)
05000247/2006005-04	NCV	Inadequate Risk Assessment for 21 MBFP steam inlet valve (Section 1R13)
05000247/2006005-05	NCV	Failure to survey and provide access to an unposted high radiation area (Section 2OS1)
05000247/2006005-06	FIN	Unit 2 containment sump strainer modification collective exposure overruns due to inadequate mod preparation (Section 2OS2)

Closed

05000247/2000004-01	URI	Adequacy of Hemyc Cable Wrap Fire Barrier Qualification Test and Evaluation

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

OAP-48, “Seasonal Weather Preparation,” Revision 3
2-SOP-11.5, “Space Heating And Winterization,” Revision 31

Condition Reports

IP2-06-05455	IP2-06-05351	IP2-06-05702
IP2-06-05438	IP2-06-04676	IP2-06-00058

Section 1R02: Evaluations of Changes, Tests, or Experiments

Safety Evaluations

04-0732-MD-00-RE, Installation of a Temporary Roll-up Door on the Containment Equipment Hatch, Rev. 1
04-1269-MD-00-RE, Westinghouse 15x15 Upgraded Fuel Design, Rev. 0
04-1311-MD-00-RE, IP2 Cycle 17 Reload Core Design Change, Rev. 1
04-1649-EV-00, Increase in Tave from 562 Degrees to 565 Degrees F, Rev. 1
05-1209-PR-00-RE, Electrical Separation, Design Criteria, Rev. 0
ECL-IP2-05-26384, Operation of Feedwater Bypass BFP-90 Series and 417L Series Valves at Stretch Power Uprate, Rev. 1

10 CFR 50.59 Screened-Out Evaluations

04-1050-PR-00-RS, Setting of Pressurizer Safety Valves
04-0436-MD-00-RS, Replacement of RCS Narrow Range Temperature Measuring System
05-0735--D-00-RS, IP2 EDG Service Water Piping Replacement
05-1133-PR-00-RS, Flow Test for Underground Service Water Line 409
2-AOP-CMT-R02, Loss of Containment Integrity, Rev. 2
2-E.C.-004-FIR, IP2 Casualty Cable Installation, Rev. 0
2-SYS-006-GEN, Installation, Control, and Removal of Support Electrical and Mechanical Equipment Required for Plant 480 V Bus Outage
2-TAP-002-EDG, Removal and Installation of Service Water Drain Line on Emergency Diesel Generator Heat Exchanger, Rev. 7
CR-IP2-2005-04614, Compensatory Measures to Change Duration of Switch Manipulation for Valve 888A Opening from CFR, Rev. 7
ER-04-2-236, IP2 - Replace Bergen Paterson Hydraulic Snubbers with LISEGA Snubbers - Phase 1, Rev. 7
ER-05-2-067, Cable Separation Modification for Outside VC (Vapor Containment)
ER-05-2-114, Removal of Bellows Seal Assembly & Extension Bonnet, and Installation of Live Loaded Graphite Packing for Pzr Spray Valves NCV-455A and NCV-455B, Rev.7

Condition Reports

2004-06285	2005-02493	2006-06254	2006-06405
2005-01572	2005-02642	2006-06372	
2005-02344	2005-05914	2006-06386	

Miscellaneous

0-RES-401-GEN, LISEGA Snubber Installation and Removal, Rev. 0
2-AOP-RHR-1, Loss of RHR, Rev. 4
2-SOP-11.1, Ventilation System Operation, Rev. 47
2-SOP-17.31, Refueling Operations Surveillance, Rev. 27
2R17 Refueling Outage Schedule Risk Assessment Report, performed February 14, 2006
EN-LI-100, Process Applicability Determination, Rev. 2
EN-LI-101, 10 CFR 50.59 Review Program, Rev. 2
EN-LI-102, Operating Plant Changes and Modifications, Rev. 1
ENN-NDE-10.03, VT-3 Examination, Rev. 1
GL 88-17, Loss of Decay Heat Removal Response
Indian Point 2 Licence Amendment No. 103
Indian Point 2 Improved Technical Specification Conversion Project – Section 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level
Indian Point 2 Improved Technical Specification Conversion Project – Section 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level, Rev. 2
IP-CALC-04-01292, Reactor Containment Building – Temporary Equipment Hatch Cover
IP-SMM-OU-104, Shutdown Risk Assessment, Rev. 2
IP-TCS-92-030, Analysis of Conditions Resulting from a Break in the 4" Steam Supply Pipe to the Auxiliary Feed Pump Turbine, performed January 18, 1992
IP2-04-12018, Roll Up Door Outside 95’ El. Equipment Hatch

IP3-CALC-VC-02347, Vapor Containment Pressurization Due to Loss of RHR Cooling During Midloop Operation, Rev. 2
LEC-948-R1, Seal Life Evaluation of Bergen-Paterson Snubbers Indian Point Nuclear Energy Center - Unit 2, Rev. 0
NEI 96-07, Guidelines for 10 CFR 50.59 Implementation, Rev. 1
NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management, Dec. 1991
Regulatory Guide 1.187; Guidance for Implementation of 10 CFR 50.59 , Changes, Tests, and Experiments; Nov. 2000
Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No 103 to Facility License No. DRP-26

Section 1R04: Equipment Alignment

Condition Reports

IP2-2006-06755	IP2-2006-02046	IP2-2006-06227	IP2-2006-04720
IP2-2006-06749	IP2-2006-01544	IP2-2006-05635	

Procedures
2-SOP-27.3.1.3, “23 Emergency Diesel Generator Manual Operation,” Revision 13
2-COL-21.3, “Steam Generator Water Level and Auxiliary Boiler Feedwater,“ Revision 29

Drawings
9321-F-2030-39, “Flow Diagram Fuel Oil TO Diesel Generators”
9321-H-2029-49, “Flow Diagram Starting Air to Diesel Generators”
9321-F-2028-36, “Flow Diagram Jacket Water to Diesel Generators”
9321-F-2018, “Flow Diagram Condensate and Boiler Feed Pump Suction,” Revision 141
9321-F-2019, “Flow Diagram Boiler Feedwater,” Revision 113

Section 1R05: Fire Protection

Procedures
ENN-DC-161, “Transient Combustible Program,” Revision 1

Condition Reports

IP2-2006-06494	IP2-2006-04946	IP2-2006-07003

Section 1R06: Flood Protection Measures

Condition Reports
IP2-2006-02256

Procedures
2-AOP-FLOOD-1, “Flooding,” Revision 1
2-ARP-004, “Waste Disposal Panel,” Revision 2

Miscellaneous

Consolidated Edison Letter, July 14, 1980, “Response to NRC’s May 20, 1980 Request for Additional Information Concerning the Effects of Flooding due to Failure of Non Seismic Class I Equipment.
Westinghouse Letter, May 8, 2006. “Revised RHR Motor Information”

Section 1R07: Heat Sink Performance

Program Documents
SEP-SW-001, Generic Letter 89-13 Service Water Program, Rev. 0
0-HTX-400-GEN, Eddy Current Inspection of Heat Exchanger Tubes, Rev. 1
IP3-RPT-UNSPEC-03499, Indian Point Units 2 & 3 Eddy Current Program, Rev. 1

Test and Inspection Results
21EDLC-21EDJC Visual Inspection Report, Dated 11/29/06, 6/15/06, and 12/28/05

Section 1R11: Licensed Operator Requalification Program

Procedures
E-0, “Reactor Trip or Safety Injection,” Revision 47
E-3, “Steam Generator Tube Rupture,” Revision 45
2-AOP-SG-1, “Steam Generator Tube Leak,” Revision 45

Simulator Test Documentation
ILO-S-033, Revision 0, “ILO Integrated Simulator Scenario Unit 2"

Condition Reports

IP2-2006-06611	IP2-2006-06119	IP2-2006-05933

Section 1R12: Maintenance Effectiveness

Condition Reports

IP2-2005-00698	IP2-2006-01299	IP2-2006-04720	IP2-2006-06607
IP2-2005-01450	IP2-2006-01367	IP2-2006-04739	IP2-2006-06509
IP2-2005-02338			

Drawings
A260589-01, “Gas Turbine #1 Flow Diagram, Lube Oil System”
B262047-01, “Gas Turbine #1 Lube Oil System Schematic”

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Maintenance Rule Action Plan, “Action Plan to Remove the Gas Turbines From (a)(1) Status,” 08/12/2005

Maintenance Rule Basis Document, “Gas Turbines,” Revision 3
Maintenance Rule Basis Document, “440 VAC Electrical Distribution System,” Revision 2

Procedures
2-COL-31.1, “Gas Turbines,” Revision 8

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Condition Reports

IP2-2006-06670	IP2-2006-06078	IP2-2006-06362	IP2-2006-06865
IP2-2006-06777	IP2-2006-06247	IP2-2006-06839	IP2-2006-06885
IP2-2006-06925			

Drawings
IP2-SOD-041, “Steam Dump System,’ Revision 1

Procedures
2-PC-R19, “Turbine First Stage Pressure,” Revision 20
2-PT-R002A, “Containment Sump Pumps and Instrumentation,” Revision 16
0-TUR-402-MFW, ”Main Boiler Feed Pump Turbine Inspection,” Revision 1
SOP-WDS-010, “Monitoring Leaks within Containment”, Revision 13

Work Orders

IP2-04-31641	IP2-05-01374	IP2-06-32508	IP2-06-00768
IP2-01-23477	IP2-06-10811	IP2-06-01073	IP2-06-33033
IP2-04-34108			

Miscellaneous
Operators Risk Report, 11/1/06
IP2 Unit Log, November 01, 2006
EN-OP-111, Attachment 9.2, “Increased Sump Pumpout Frequency Action Plan”

Section 1R15: Operability Evaluations

Calculations
“Calculation to Establish Maximum Air Leakage into S/G at VC Design Pressure (47 psig)”
FMX-00236-01, “Service Water Header Low Pressure Alarm Setpoint,” Revision 1
FIX-00072-00, “Service Water Header Low Pressure Alarm Setpoint and Loop Uncertainty Calculation,” Revision 0

Procedures
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Condition Reports

IP2-2006-06839	IP2-2006-06712	IP2-2001-00707
IP2-2006-06732	IP2-2006-06717	IP2-2006-06960

Miscellaneous
EN-MA-125, Attachment 9.3, “Troubleshooting Plan for Check Valves 1758 and 1760,” Revision 2

Section 1R17: Permanent Plant Modifications

Modifications

ER-03-2-208, Replacement of 118V AC Instrument Buses 21 and 24
ER-03-2-216, Replace Steam Flow Instrument Loop Power Supplies with Combined DC Power and Scaling Modules, Rev. 0
ER-03-2-217, Power Uprate: Setpoint Changes
ER-04-2-005, Replacement of the RCS Narrow Range Temperature Measuring System
ER-04-2-044, Roll Up Door Outside 95’ El. Equipment Hatch
ER-04-2-236, IP2 - Replace Bergen Paterson Hydraulic Snubbers with LISEGA Snubbers - Phase 1, Rev. 7
ER-05-2-059, IP2 EDG Service Water Piping Replacement, Rev. 0
ER-05-2-064, Replace Westinghouse Circuit Breakers (22)
ER-06-2-020, IP2 Toxic Gas Monitors Replacement, Rev 0
ELMP-2006-006, IP2 Toxic Gas Monitor Electrical Load Modification Package, Rev. 0
IP2 Toxic Gas Monitors Replacement Electrical Separation Review, Dated 3/21/06

Condition Reports

2004-05957	2004-06837	2005-02493	2006-06372
2004-06776	2005-01572	2005-02642	2006-06386
2004-06818	2005-02344	2005-05914	

Calculations
IP-CALC-06-00069, Evaluation of Mounting of New Components for IP2 Toxic Gas Monitor Replacement, Rev. 0
EGP-00012-04, IP2 - DC Load Study Battery 22 Calculation, Rev.

Drawings
227011, Toxic Gas Monitor Pump, Rev. 6
226927, CCR Toxic Gas Monitors Channel #1, Rev. 4
226928, CCR Toxic Gas Monitors Channel #2, Rev. 4
138932, Electrical Heating and Ventilating Control Scheme #1, Rev. 18
IP2-S-000256, CCR Ventilation Power Distribution, Rev. 9
B251249, Chlorine Monitoring Panel, Rev. 1
B251247, Ammonia Monitoring Panel, Rev. 1

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2-SOP-11.1, Ventilation System Operation, Rev. 47
30-1146, Type 30 Hydraulic Snubber BP Replacement, Rev. 0
CN-PAFM-04-45, IP2 RTD Replacement Project - RCL Piping Stress Evaluation, Rev. 1

END-DC-112, Engineering Request and Project Initiation Process, Rev. 7
END-DC-116, Engineering Request Response Installation, Rev. 5
END-DC-117, Post Modification Testing and Special Testing Instruction, Rev. 4
END-LI-102, Operating Plant Changes and Modification, Rev. 1
END-NAE-10.03, VT-3 Examination, Rev. 1
FCN IPP0-40543, Field Change Notice for Fast Response RTD Thermowell Installation
IP-RCS-92-030, Analysis of Conditions Resulting from a Break in the 4" Steam Supply Pipe to the Auxiliary Feed Pump Turbine, Performed 1/18/92
IP2-04-11142, RCS Tavg and ΔT Instrumentation Channel Calibration, Including RTD Time Constant Testing
IP2-SSO-03-8, Over Temperature Delta T Reactor Trip and OverPower Delta T Reactor Trip Setpoints Uncertainty Calculations, Rev. 4
IP2-SSO-03-20, Steam Flow in Two Steamlines, High ESFAS Safety Injection Actuation Setpoint Uncertainty Calculations, Rev. 1
IP2-SCS-03-29, Steamline Pressure, Low ESFAS Steamline Isolation Actuation Setpoint Uncertainty Calculations, Rev. 1
IP2-SCS-03-16, Plant Operability/Margin to Trip for Indian Point 2 Stretch Power Uprate Project, Rev. 1
IP2-Nuclear HVAC DBD, CFR HVAC, Rev. 1
IP2-RPT-03-00030, IP2 Replacement of RCS Resistance Temperature Detectors, Rev. 1
IPT0401R0, AMS Test Report for Response Time Testing of Primary Coolant RTDs at Indian Point 2, Rev. 0
LEC-948-R1, Seal Life Evaluation of Bergen-Paterson Snubbers Indian Point Nuclear Energy Center - Unit 2, Rev. 0
NEI 96-07, Guidelines for 10 CFR 50.59 Implementation, Rev. 1
PI-900307-04, PCI Welding Procedure for Removal of RTDs and Installation of RTD Thermowell, Rev. 3
PQE-31A.1, EQ Evaluation for (Lewis) Silicone Rubber Insulated and Jacketed Instrumentation Cable, Rev. 0
PQE-37-3A1, EQ Evaluation for (Raychem/UE&C, Crouse-Hinds) Cable Repair Kits, Rev. 0
PQE-49.2, EQ Evaluation for (EGS) Quick Connectors, Rev. 0
PQE-54.1, EQ Evaluation for RTD Assemblies, Rev. 1
Regulatory Guide 1.187; Guidance for Implementation of 10 CFR 50.59 , Changes, Tests, and Experiments; dated November 2000

Section 1R19: Post-Maintenance Testing

Condition Reports

IP2-2003-05475	IP2-2006-06514	IP2-2006-06473
IP2-2006-06632		

Drawings
9321-F-2746-42, “Flow Diagram, Isolation Valve Seal Water System”

Miscellaneous
UFSAR Section 6.5, “Isolation Valve Seal Water System,” Revision 17
IP2-AFW-DBD, “Auxiliary Feed Water Design Basis Document,” Revision 1

Procedures
2-PT-M021A, “Emergency Diesel Generator 21 Load Test,” Revision 1
2-PT-Q013, “Inservice Valve Tests,” Revision 38

Work Orders

IP2-05-28068	IP2-06-31591	IP2-06-25887	IP2-06-32705
IP2-06-27470	IP2-05-20791	IP2-06-32689	
IP2-06-27219			

Section 1R20: Refueling and Other Outage Activities

Procedures
2-POP-1.2, “Plant Startup - Mode 3 to Mode 2,” Revision 50
2-POP-1.3, “Plant Startup - Mode 2 to Mode 1," Revision 74

Condition Reports

IP2-2006-06362	IP2-2006-06885
IP2-2006-06777	

Section 1R22: Surveillance Testing

Procedures
2-PT-Q028B, “22 Residual Heat Removal Pump,” Revision 17
0-SOP-LEAKRATE-001, “RCS Leakrate Surveillance, Evaluation, and Leak Identification,” Revision 0
2-PT-Q034, “22 ABFP,” Revision 22

Condition Reports

IP2-2006-06514	IP2-2006-06667

Drawings
A251783-28, “Flow Diagram, Auxiliary Coolant System Residual heat Removal Pumps”
A235296-68, “Flow Diagram, Safety Injection System”
UFSAR Figure 6.2-7, “Residual heat Removal Pump Performance,” Revision 17A

Section 2OS1: Access Control to Radiologically Significant Areas

Condition Reports:

CR-IP2-2006-02429	CR-IP2-2006-00928	CR-IP2-2006-00709
CR-IP2-2006-05143	CR-IP2-2006-04361	CR-IP2-2006-02818
CR-IP2-2006-02358	CR-IP2-2006–02344	CR-IP2-2006-04168
CR-IP2-2006-02933	CR-IP2-2006-01889	CR-IP2-2006-02905
CR-IP3-2006-01982	CR-IP2-2006-04502	CR-IP3-2006-02672
CR-IP2-2006-05070	CR-IP3-2006-01715	

Miscellaneous
2R17 Refueling Outage Report
Indian Point Energy Center Five-Year ALARA Plan 2006-2010
Post 2R17 Review of Indian Point Unit 2 Outage Dose Reduction - Westinghouse Customer 1st
Indian Point Energy Center Radiation Protection Excellence Plan 2006-2007
QA-14-2006-IP1, IPEC Radiation Protection Program QA audit, 2/6-3/3/06
QS-2006-IP-006, RP and radworker practices during 2R17, 6/2/06
QS-2006-IP-018, Outage Management, Maintenance, RP, Supplemental Employees during 2R17, 6/12/06
QS-2006-IP-23, Followup of Corrective Actions in Response to Marginally Effective Radiation Protection Performance during 2R17, 8/16/06
EN-RP-101, Rev. 1, Access Control for Radiologically Controlled Areas
Self-Assessment: Control of Contamination and Radioactive Material, 9/11-15/06
Snap Shot Self-Assessment: Exposure Reduction through Permanent Scaffold and Shielding, 9-10/06

Section 2OS2: ALARA Planning and Controls

Condition Reports:

CR-IP2-2006-02429	CR-IP2-2006-00928	CR-IP2-2006-00709
CR-IP2-2006-05143	CR-IP2-2006-04361	CR-IP2-2006-02818
CR-IP2-2006-02358	CR-IP2-2006–02344	CR-IP2-2006-04168
CR-IP2-2006-02933	CR-IP2-2006-01889	CR-IP2-2006-02905
CR-IP3-2006-01982	CR-IP2-2006-04502	CR-IP3-2006-02672
CR-IP2-2006-05070	CR-IP3-2006-01715	

Miscellaneous
2R17 Refueling Outage Report
Indian Point Energy Center Five-Year ALARA Plan 2006-2010
Post 2R17 Review of Indian Point Unit 2 Outage Dose Reduction - Westinghouse Customer 1st
Indian Point Energy Center Radiation Protection Excellence Plan 2006-2007
QA-14-2006-IP1, IPEC Radiation Protection Program QA audit, 2/6-3/3/06
QS-2006-IP-006, RP and radworker practices during 2R17, 6/2/06
QS-2006-IP-018, Outage Management, Maintenance, RP, Supplemental Employees during 2R17, 6/12/06
QS-2006-IP-23, Followup of Corrective Actions in Response to Marginally Effective Radiation Protection Performance during 2R17, 8/16/06
EN-RP-101, Rev. 1, Access Control for Radiologically Controlled Areas
Self-Assessment: Control of Contamination and Radioactive Material, 9/11-15/06
Snap Shot Self-Assessment: Exposure Reduction through Permanent Scaffold and Shielding, 9-10/06

Section 4OA1: Performance Indicator Verification

Procedures
EN-LI-114, Revision 1: “Performance Indicator Process”
NEI 99-02, Rev. 4: “Regulatory Assessment Performance Indicator Guideline”

LERs

LER-2004-004	LER-2005-002	LER-2006-001	LER-2006-002

Section 4OA2: Identification and Resolution of Problems

Procedures
0-SYS-014-GEN, Rev 5: “Scaffolding Construction and Control”
OAP-008, Rev 2: “Severe Weather Preparations”
IP-SMM-WM-100, Rev 5: “Work Management Process”
IP-SMM-WM-101, Rev 1: “On-Line Risk Assessment”
EN-WM-101, Rev 0: “On-Line Work Management Process”

Condition Reports

IP2-2006-06701	IP3-2006-00245	IP2-2006-01012	IP2-2006-03930
IP2-2006-01011	IP2-2006-01014	IP2-2006-01013	IP2-2006-01026
IP2-2006-03382	IP2-2006-05316	IP2-2006-06272	IP2-2006-01027
IP2-2006-03374	IP2-2006-01043	IP2-2006-00619	IP2-2006-01834
IP3-2006-01093	IP2-2006-01644	IP2-2006-04861	IP3-2006-03481

Work Orders
IP2-06-15098

Miscellaneous
Licensee Event Report #2006-001-00, “Manual Reactor Trip Due to Multiple Dropped Control Rods Caused by Loss of Control Rod Power Due to Personnel Error”
Indian Point Energy Center Maintenance Consolidation Action Plan
Indian point Energy Center Maintenance Excellence Plan
Operations Department Procedure Action Plan

Procedures
0-MD-402, “Maintenance Procedure Development and Feedback Administrative Directive,” Revision 2

Section 4OA3: Event Followup

Condition Reports
IP2-2006-06657
IP2-2006-06658
IP2-2006-06659
IP2-2006-06666
IP2-2006-06667

Procedures
IP-SMM-OP-105, “Post Transient Evaluation, “ Revision 4

Miscellaneous
Control Room Operator Logs, November 15, 2006

Section 4OA5: Other Activities

Condition Reports

IP2-2006-07108	IP2-2006-00076	IP2-2006-00302	IP2-2006-01804
IP2-2006-00003	IP2-2006-00108	IP2-2006-00664	IP2-2006-01857
IP2-2006-00051	IP2-2006-00159	IP2-2006-00688	IP2-2006-02489
IP2-2006-00058	IP2-2006-00202	IP2-2006-01053	IP2-2006-02397
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IP2-2006-06487	IP2-2006-00287		

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PI Data Summary Reports - 1st Quarter 2002 through 3rd Quarter 2006
Consolidated Data Entry, "MSPI Derivation Report," September 2006
MSPI Basis Document, "Mitigating Systems Performance Index (MSPI)," Revision 5
ENN-LI-114, "NRC Performance Indicator Technique Sheet, " Revision 1
NEI 99-02, "Mitigating Systems Performance Index," Revision 4
Modification FPX-95-72783-F, Curtain drain and sphere foundation sumps
Self-assessment on groundwater monitoring program, July 10-21, 2006
Self-assessment snapshot on groundwater workshop 2006, February 14, 2006
Groundwater sampling procedure, O-CY-2775
Standard Operating Procedure Pumping Test, October 11, 2006
Long Term Groundwater Protection Plan
IPEC Groundwater Dose Calculations, December 2006
Pumping Test Report, December 8, 2006
Offsite Dose Calculation Manual draft, December 2006

Procedures
ENN-DC-171, "Screening and Functional Failure Determination," Revision 0
2-PT-2M2, "Reactor Protection Logic Train "A" Actuation"
2-PT-2M4, "SI Logic Train Testing "A""
2-PT-Q13, "In-Service Valve Test"
2-PT-V24 "822 A/B Stroke Test"
2-PT-Q30A, "21 CCW Pump"

LIST OF ACRONYMS

ABFP	auxiliary boiler feed pumps
ADAMS	agency wide document and management system
ALARA	As Low as is Reasonably Achievable
AOP	abnormal operating procedure
CAP	corrective action program
CCP	coolant charging pumps
CEDE	committed effective dose equivalent
CFR	Code Of Federal Regulations
CR	condition report
DEC	Department of Environmental Conservation
DHR	decay heat removal
EDG	emergency diesel generator
EOF	emergency operations facility
EOP	emergency operating procedure
ESSAP	Environmental Site Survey and Assessment Program
GL	generic letter
gpd	gallons per day
IMC	inspection manual chapter
IPE	individual plant examination
IPEC	Indian Point Energy Center
LER	Licensee Event Report
MBFP	main boiler feed pumps
MPFF	maintenance preventable functional failures
MSPI	mitigating systems performance index
MW	monitoring well
NCV	non-cited violation
NPO	nuclear plant operator
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OP	operating procedure
ORISE	Oak Ridge Institute for Science and Education,
PCB	Poly-chlorinated biphenyls
PI	performance indicator
PI&R	problem identification and resolution
PM	preventive maintenance
RETS	radiological effluents technical specifications
RHR	residual heat removal
RMPFF	repeat maintenance preventable functional failures
RO	reactor operator
RP	radiation protection
RW	recovery well
SDP	significance determination process
SE	safety evaluations
SFP	spent fuel pool
SRO	senior reactor operator
SSC	structure, system, and component

TS	technical specifications
TSC	Technical Support Center
UFSAR	Updated Final Safety Evaluation Report
WO	work order

ATTACHMENT 2

SUPPLEMENTAL INFORMATION

MITIGATING SYSTEM PERFORMANCE INDEX VERIFICATION

Question 1: For the sample selected, did the licensee accurately document the baseline planned unavailability hours for the MSPI systems?

Answer: Yes

Comments: The inspectors identified several examples where Entergy over-counted unavailability hours, resulting in a non-conservative determination of planned unavailability. These errors were determined to be non-significant and were corrected by Entergy. They did not result in a change in index color.

Question 2: For the sample selected did the licensee accurately document the actual unavailability hours for the MSPI systems?

Answer: Yes

Comments: The inspectors identified one example where unavailability hours were incorrectly counted. This error was non-significant and did not result in a change in index color.

Question 3: For the sample selected, did the licensee accurately document the actual unreliability information for each MSPI monitored component?

Answer: Yes

Comments: None

Question 4: Did the inspector identify significant errors in the reported data, which resulted in a change to the indicated index color?

Answer: No

Comments: None

Question 5: Did the inspector identify significant discrepancies in the basis document which resulted in (1) a change to the system boundary; (2) an addition of a monitored component; or (3) a change in the reported index color?

Answer: No

Comments: None