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

May 10, 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. 05000247/2007002

Dear Mr. Dacimo:

On March 31, 2007, 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 April 4, 2007, with Mr. James Comiotes 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, four 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 were 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
Projects Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report No. 05000247/2007002
w/ Attachment: Supplemental Information

cc w/encl:
G. J. Taylor, Chief Executive Officer, Entergy Operations
M. Kansler, President, Entergy Nuclear Operations, Inc.
J. T. Herron, Senior Vice President for Operations
M. Balduzzi, Senior Vice President, Northeastern Regional Operations
W. Campbell, Senior Vice President of Engineering and Technical Services
C. Schwarz, Vice President, Operations Support (ENO)
K. Polson, General Manager Operations
O. Limpias, Vice President, Engineering (ENO)
J. McCann, Director, Licensing (ENO)
C. D. Faison, Manager, Licensing (ENO)
R. Patch, Director of Oversight (ENO)
J. Comiotes, Director, Nuclear Safety Assurance
P. Conroy, Manager, Licensing
T. C. McCullough, Assistant General Counsel, Entergy Nuclear Operations, Inc.
P. R. Smith, President, New York State Energy, Research and Development Authority
P. Eddy, Electric Division, New York State Department of Public Service
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
D. O'Neill, Mayor, Village of Buchanan
J. G. Testa, Mayor, City of Peekskill
R. Albanese, Four County Coordinator
S. Lousteau, Treasury Department, Entergy Services, Inc.
Chairman, Standing Committee on Energy, NYS Assembly
Chairman, Standing Committee on Environmental Conservation, NYS Assembly
Chairman, Committee on Corporations, Authorities, and Commissions
M. Slobodien, Director, Emergency Planning
B. Brandenburg, Assistant General Counsel
Assemblywoman Sandra Galef, NYS Assembly
County Clerk, Westchester County Legislature
A. Spano, Westchester County Executive

R. Bondi, Putnam County Executive
C. Vanderhoef, Rockland County Executive
E. A. Diana, Orange County Executive
T. Judson, Central NY Citizens Awareness Network
M. Elie, Citizens Awareness Network
D. Lochbaum, Nuclear Safety Engineer, Union of Concerned Scientists
Public Citizen's Critical Mass Energy Project
M. Mariotte, Nuclear Information & Resources Service
F. Zalcman, Pace Law School, Energy Project
L. Puglisi, Supervisor, Town of Cortlandt
Congressman John Hall
Congresswoman Nita Lowey
Senator Hillary Rodham Clinton
Senator Charles Schumer
G. Shapiro, Senator Clinton's Staff
J. Riccio, Greenpeace
P. Musegaas, Riverkeeper, Inc.
M. Kaplowitz, Chairman of County Environment & Health Committee
A. Reynolds, Environmental Advocates
M. Jacobs, Director, Longview School
D. Katz, Executive Director, Citizens Awareness Network
S. Tanzer, The Nuclear Control Institute
K. Coplan, Pace Environmental Litigation Clinic
M. Jacobs, IPSEC
D. C. Poole, PWR SRC Consultant
W. Russell, PWR SRC Consultant
W. Little, Associate Attorney, NYSDEC
M. J. Greene, Clearwater, Inc
R. Christman, Manager Training and Development
J. Spath, New York State Energy Research, SLO Designee

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W. Little, Associate Attorney, NYSDEC
M. J. Greene, Clearwater, Inc
R. Christman, Manager Training and Development
J. Spath, New York State Energy Research, SLO Designee

Distribution w/encl:
S. Collins, RA
M. Dapas, DRA
J. Lamb, RI OEDO
J. Lubinski, NRR
M. Kowal, NRR
J. Boska, PM, NRR
J. Hughey, NRR
E. Cobey, DRP
D. Jackson, DRP
M. Cox, DRP, Senior Resident Inspector - Indian Point 2
G. Bowman, DRP, Resident Inspector - Indian Point 2
R. Martin, DRP, Resident OA
Region I Docket Room (w/concurrences)
ROPreports@nrc.gov

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-247
License No.:	DPR-26
Report No.:	05000247/2007002
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	Indian Point Nuclear Generating Unit 2
Location:	450 Broadway, GSB Buchanan, NY 10511-0249
Dates:	January 1, 2007 through March 31, 2007
Inspectors:	M. Cox, Senior Resident Inspector, Indian Point 2 D. Jackson, Senior Resident Inspector, Indian Point 3 G. Bowman, Resident Inspector, Indian Point 2 C. Hott, Resident Inspector, Indian Point 2 B. Wittick, Resident Inspector, Indian Point 3 S. Barr, Senior Emergency Preparedness Specialist, Region I T. Nicholson, Senior Technical Advisor on Radionuclide Transport J. Noggle, Senior Health Physicist, Region I
Other:	J. Williams, Groundwater Specialist, U.S. Geological Survey
Approved by:	Eugene W. Cobey, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000247/2007002; 01/01/2007 - 03/31/2007; Indian Point Nuclear Generating Unit 2; Operability Evaluations, Permanent Plant Modifications, Problem Identification and Resolution.

The report covered a three-month period of inspection by resident and region-based inspectors. Four 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 10 CFR 50, Appendix B, Criterion III, "Design Control," in that, Entergy did not appropriately incorporate design requirements into an operating procedure used to establish adequate component cooling water (CCW) flow to the reactor coolant pump (RCP) thermal barriers. Specifically, the flow specification in the CCW operating procedure did not incorporate the calculated design flow requirements to bound allowable CCW temperature limits. Entergy entered this issue into their corrective action program and will be evaluating the flow requirements specified in procedure 2-SOP-4.1.2, "Component Cooling Water System Operation," to ensure that they bound the allowed plant operating limits.

The inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, Entergy did not incorporate design flow requirements necessary to assure adequate cooling water flow to the RCP thermal barriers into the plant operating procedures which establish the required flow. On a loss of seal injection, the procedure did not ensure that the heat removal capability was adequate to prevent a rise in seal temperature which would require the RCP to be stopped with a subsequent reactor trip. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This finding was determined to be of very low safety significance because it would not result in exceeding the Technical Specification limit for identified reactor coolant system leakage and would not have likely affected other mitigating systems resulting in a loss of their safety function. The inspectors found that the procedurally established nominal flow band would have assured adequate cooling of the RCP thermal barriers for the highest CCW supply temperature recorded over the previous year.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the operating procedure used to set the flow rate of cooling water to the RCP thermal barriers was not adequate to make certain that sufficient cooling water was available to assure the components could perform their design function. (Section 1R15)

Green. The inspectors identified a Green, NCV of 10 CFR 50 Appendix B, Criterion XI, "Test Control," in that, Entergy did not establish appropriate testing to assure adequate component cooling water (CCW) flow to the reactor coolant pump thermal barriers. Specifically no preventive maintenance activities or functional checks were conducted for the individual flow meters. It was determined that the rotameters on 21 and 23 RCP were not indicating correctly and that actual CCW flow to the thermal barrier heat exchangers was less that the design requirements for CCW temperature. Entergy entered this issue into their corrective action program (CR-IP2-2007-00783 and 00955), adjusted individual cooling water flow within the nominal band using ultrasonic flow meters, wrote work orders to replace the faulty flow meters, and is conducting an evaluation to determine the appropriate test requirements for the flow indicators.

This inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, Entergy's test program did not assure that all testing required to demonstrate that the RCP thermal barriers will perform satisfactorily in service because no testing was performed to ensure the accuracy of the individual flow meters used to establish the required cooling water flow. Consequently, it was identified that two individual flow indicators did not read correctly and the CCW flow to two RCP's was not sufficient to assure adequate cooling in the event that seal water was lost based on the flow requirements established in design calculations. On a loss of seal injection, the cooling water flow would not ensure that the heat removal capability was adequate to prevent a rise in seal temperature which would require the RCP to be stopped with a subsequent reactor trip. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This finding was determined to be of very low safety significance because it would not result in exceeding the Technical Specification limit for identified reactor coolant system leakage and would not have likely affected other mitigating systems resulting in a loss of their safety function. (Section 1R15)

Cornerstone: Barrier Integrity

Green. The inspectors identified a Green, NCV of 10 CFR 50.65(a)(2) because Entergy did not demonstrate that the performance or condition of the containment hydrogen monitoring system was being effectively controlled through the performance of appropriate preventive maintenance such that the system remained capable of performing its intended function. The inspectors identified that both channels of the containment hydrogen/oxygen (H_2/O_2) analyzers had been out of service since September 7, 2006, due to compressor seal leakage. The inspectors determined that the H_2/O_2 analyzers are within the scope of Entergy's Maintenance Rule program since

they are used in the emergency operating procedures. The inspectors noted that, based on the significant unavailability time of both trains, the system should have been in 10 CFR 50.65(a)(1) status with an action plan to improve system performance back to an (a)(2) status. Entergy entered this issue into their corrective action program and changed the priority of the work orders to perform repairs on the H_2/O_2 analyzers.

This inspectors determined that this finding affected the Barrier Integrity cornerstone and was more than minor since it was similar to Example 7.b in IMC 0612, Appendix E, "Examples of Minor Issues." Specifically, Entergy failed to demonstrate effective control of the performance of the H_2/O_2 analyzers and did not place the system in (a)(1) status. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The finding required further evaluation through IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," because it resulted in an actual reduction in the defense-in-depth for the hydrogen control function of the reactor containment. The inspectors determined that this finding was of very low safety significance because it did not affect core damage frequency and the H_2/O_2 analyzers are not important to large early release frequency.

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 to assure reliable operation of the H_2/O_2 analyzers. Specifically, Entergy did not minimize long-standing equipment issues and maintenance deferrals associated with the containment hydrogen monitoring system. (Section 4OA2)

Cornerstone: Emergency Preparedness

Green. The inspectors identified a Green finding because Entergy failed to take adequate corrective actions for an issue associated with monitoring of service water intake bay level. This deficiency could have prevented identification of entry conditions for an emergency action level. Entergy entered this issue into the corrective action program as CR IP3-2007-00453, and initiated several corrective actions, including plans for enhanced monitoring of service water bay levels, backwashing of trash racks, procedural upgrades, correction of service water bay level instrumentation modification installation, development of modifications for enhanced service water level monitoring equipment, and enhanced inspection and cleaning of intake structure trash racks.

The inspectors determined that this finding was more than minor because it was associated with the Emergency Preparedness cornerstone attribute of facilities and equipment; and, it affected the cornerstone objective of ensuring that a licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, inadequate monitoring of service water intake bay level could have resulted in failure to declare a notification of unusual event (UE). The inspectors reviewed the EAL entry criteria and determined that this performance deficiency did not affect Entergy's ability to declare any event higher than a UE. The inspectors evaluated this finding using IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process," Sheet 1, "Failure to Comply," and determined that it was of very low safety significance because the

declaration of a UE based on low service water bay level could have been missed or delayed, consistent with the example provided in the appendix.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not implement effective corrective actions for a previously identified issue associated with inadequate monitoring of service water intake bay level. (Section 1R17)

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 February 28, 2007. The reactor was manually tripped following failure of the main feedwater pump suction pressure transmitter, which caused a loss of feedwater flow. The plant returned to full power on March 1, 2007, and remained at 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 - 2 samples)

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. 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:

- primary water storage tank;
- refueling water storage tank; and
- fire water storage tank.

Additionally, the inspectors evaluated implementation of the adverse weather preparation procedures and compensatory measures before the onset of, and during adverse weather conditions. Specifically, the inspectors evaluated Entergy's preparations following a heavy snow warning on February 13, 2007. 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. This inspection satisfied one inspection sample for the onset of adverse weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q - 3 samples)

a. Inspection Scope

The inspectors performed three partial system walkdowns to verify the operability of redundant or diverse trains and components during periods of system train unavailability or following periods of maintenance. The inspectors referenced the system procedures, the UFSAR, and system drawings to verify that the alignment of the available train supported 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 during these inspections are listed in the Attachment. The inspectors performed a partial walkdown on the following systems which represented three inspection samples:

- 21 and 22 containment spray pumps following testing;
- 21 and 22 emergency diesel generators during maintenance and testing on 23 emergency diesel generator; and
- 21 and 23 auxiliary boiler feedwater pumps during testing on the 22 auxiliary boiler feedwater pump.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 10 samples)

a. Inspection Scope

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

- Fire Zone 1;
- Fire Zones 27A and 33A;
- Fire Zone 650;
- Fire Zone 3 and 3A;

- Fire Zone 14;
- Fire Zones 11, 12, 13, and 24;
- Fire Zones 5, 6, and 7;
- Fire Zones 23A, 24A, 25A, and 26A;
- Fire Zone 332A; and
- Fire Zone 2 and 2A.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

a. Inspection Scope

The inspectors reviewed selected risk-significant plant design features and Entergy's procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors selected the 480 volt switchgear room for review. The inspectors reviewed flood analysis and design documents, including the Individual Plant Examination and the UFSAR, engineering calculations, and abnormal operating procedures. The inspection included a walkdown of accessible areas of the plant to look for potential susceptibilities to internal flooding and to verify the assumptions included in the site's flooding analysis. The documents reviewed during this inspection are listed in the Attachment. These activities represented one internal flooding inspection sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the 21 component cooling water heat exchanger to verify that Entergy was maintaining the heat exchanger in accordance with their commitments to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The inspectors reviewed recent visual inspection reports and eddy current results to verify that the inspections and testing were in accordance with approved plant procedures and industry guidance and that acceptance criteria were appropriate. The inspectors conducted a walk down of the heat exchanger to observe its material condition and verified the expected system indications. The documents reviewed during this inspection are listed in the Attachment. The inspection of the 21 component cooling water heat exchanger represented one inspection sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On March 23, 2007, the inspectors observed licensed operator simulator training to verify that operator performance was adequate and that 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 during this inspection are listed in the Attachment. This observation of operator simulator training represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 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 during this inspection are listed in the Attachment.

The following Maintenance Rule samples were reviewed and represent two inspection samples:

- Intake structure; and
- Control building floor drains.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed maintenance 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 during this inspection are listed in the Attachment. The following activities represent seven inspection samples:

- Work order (WO) IP2-07-34280, 21 residual heat removal pump breaker failure and extent of condition review;
- Electrical feeder outages for switch yard work;
- WO IP2-06-15853, 22 auxiliary feedwater pump test with gas turbine 1 out of service for maintenance;
- WO IP2-07-10997, 22 lighting bus transfer switch maintenance;
- Condition report (CR) IP2-2007-00971 and 00972, fuel pin failure during inspection;
- CR IP2-07-01333, central control room toxic gas monitoring system alarm; and
- CR IP2-2007-00571, breaker 9 failure to open for fault isolation.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to assess the acceptability of the evaluations, the use and control of compensatory measures, and compliance with TS. The inspectors review included a verification that the operability determinations were performed in accordance with 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 during this inspection are listed in the Attachment. The following operability evaluations were reviewed and represent five inspection samples:

- CR IP2-06-07188, NUS controllers following 10 CFR 21 notification;
- CR IP2-07-00980, 22 auxiliary boiler feedwater pump following surveillance test failure;
- CR IP2-07-00745, component cooling water flow to reactor coolant pump (RCP) thermal barriers;
- CR IP2-06-07120, 22 emergency diesel generator following maintenance; and
- CR IP2-07-00117, ultra-low sulfur fuel oil for emergency diesel generators.

b. Findings

1. Introduction: The inspectors identified a Green, non-cited violation (NCV) of 10 CFR 50 Appendix B, Criterion III, "Design Control," in that, Entergy did not appropriately incorporate design requirements into an operating procedure used to establish adequate component cooling water (CCW) flow to the RCP thermal barriers. Specifically, the flow requirements established by the procedure did not incorporate the calculated flow necessary to bound allowable CCW temperature limits.

   Description: During an evaluation of an operability concern associated with CCW flow to the RCP thermal barrier heat exchangers, the inspectors reviewed operating procedure 2-SOP-4.1.2, "Component Cooling Water System Operation." This procedure specified a minimum required cooling water flow of 13 gallons per minute (gpm) to each RCP with a nominal flow range of 25 to 30 gpm and stated that the minimum and nominal flow requirements were derived from calculation WCAP-12312, "Safety Evaluation for an Ultimate Heat Sink Temperature Increase to 95 °F at Indian Point Unit 2."

   The inspectors reviewed WCAP-12312 and identified that the minimum required CCW flow to the thermal barrier heat exchangers was temperature dependent. The 13 gpm minimum specified in procedures 2-SOP-4.1.2 was only valid if the CCW supply temperature was less than or equal to 70 degrees Fahrenheit (°F). The inspectors noted that the allowable limit for CCW supply temperature was 70 - 110 °F. The inspectors also determined that, based on the calculated values for minimum flow requirements, the nominal flow band in the procedure did not bound the flow required to assure adequate thermal barrier cooling for the allowable CCW supply temperature range. If CCW flow was set at 25 gpm, as allowed by the procedure, adequate cooling would not be assured if CCW supply temperature exceeded 103 °F.

   The RCP thermal barriers are designed to protect the pump seals from high temperature conditions. High pressure seal injection water is introduced just above the thermal barrier. A portion of this water flows down the RCP shaft through the thermal barrier where it acts as a buffer to prevent hot reactor coolant from entering the bearing and seal section of the pump. If seal injection is lost, the thermal barrier is designed to minimize the heat flow to the pump lower radial bearing and seal package by cooling the reactor coolant passing upward through it to an acceptable temperature to prevent seal

damage. In the event that both seal cooling and CCW flow to the thermal barriers are inadequate, the seal temperature would rise until it reached a setpoint requiring the RCP be stopped, and a reactor trip be initiated.

The inspectors reviewed operator logs dating back to January 1, 2006, and determined that the maximum CCW supply temperature during the time period was 92 °F, which would require 20 gpm to assure adequate cooling water to the thermal barrier heat exchangers. The inspectors noted that the minimum flow of 13 gpm specified in the procedure was used as part of an evaluation to justify operability when a low flow condition was identified in condition report IP2-2007-00745.

Analysis: The inspectors determined that the failure to incorporate design basis information into operating procedures required to assure adequate cooling water flow to the thermal barriers is a performance deficiency and does not meet the requirements of 10 CFR 50, Appendix B, Criterion III, "Design Control." Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy's procedures.

The inspectors determined that this finding was more than minor because it was associated with the Equipment Performance attribute of the Initiating Events cornerstone; and' it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, Entergy did not incorporate design flow requirements necessary to assure adequate cooling water flow to the RCP thermal barriers into the plant operating procedures which establish the required flow. Consequently, the nominal flow band established by the procedure did not bound the flow required to assure adequate seal cooling over the allowable CCW supply temperature range. On a loss of seal injection, the procedure did not ensure that the heat removal capability was adequate to prevent a rise in seal temperature which would require the RCP to be stopped with a subsequent reactor trip and could result in seal damage due to high temperatures. In addition, the minimum flow requirement specified in the procedure was non-conservative and was used, in part, as a basis for operability when degraded cooling water flow was identified. The inspectors evaluated the significance of this finding using Phase 1 of Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This finding was determined to be of very low safety significance because it would not result in exceeding the TS limit for identified reactor coolant system leakage and would not have likely affected other mitigating systems resulting in a loss of their safety function. The inspectors found that the procedurally established nominal flow band would have assured adequate cooling of the RCP thermal barriers for the highest CCW supply temperature recorded over the previous year.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the operating procedure used to set the flow rate of cooling water to the RCP thermal barriers was not adequate to make certain that

sufficient coolant water was available to assure adequate cooling of the RCP seals if seal water was lost.

Enforcement: 10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that applicable regulatory requirements and design basis for safety-related structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, prior to February 20, 2007, Entergy failed to incorporate design basis information into operating procedures required to assure adequate cooling water flow to the RCP thermal barriers. Specifically, Entergy did not incorporate design flow requirements necessary to assure adequate cooling water flow to the RCP thermal barriers into the plant operating procedures which establish the required flow. Entergy entered this issue into their corrective action program (CR IP2-2007-00587 and -00745) and a corrective action was implemented to evaluate the requirements specified in procedure 2-SOP-4.1.2, "Component Cooling Water System Operation," to ensure that procedural flow requirements bound the allowed plant operating limits. Because this issue is of very low safety significance and is entered into Entergy's corrective action program, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC enforcement manual. **(NCV 05000247/2007002-01, Failure to Incorporate Design Basis Information into Procedures to Assure Adequate Cooling Water Flow to the RCP Thermal Barriers)**

2. Introduction. The inspectors identified a Green, NCV of 10 CFR 50 Appendix B, Criterion XI, "Test Control," in that, Entergy did not establish appropriate testing to assure adequate component cooling water (CCW) flow to the reactor coolant pump thermal barriers. Specifically no preventive maintenance activities or functional checks were conducted for the individual flow meters, which are used to established the required flow rate.

   Description. On February 8 through 20, 2007, the inspectors reviewed Entegy's actions associated with inconsistent flow measurements between the indicated combined CCW flow to the reactor coolant pump (RCP) thermal barrier heat exchangers as read on flow meter FIC-625, and the individual flows as read on the local flow rotameters. When the condition was first identified on February 9, 2007, the combined flow indicator read 75 gallons per minute (gpm) and the sum of the individual flows was 94 gpm. The indication on FIC-625 was verified accurate with an ultrasonic flow measuring device. Following adjustments to increase flow, the difference between combined and the sum of the individual flows increased to 25 gpm. Entergy determined this condition did not adversely impact component operability since the minimum flow requirement per RCP was 13 gpm per procedure 2-SOP-4.1.2, "Component Cooling Water System Operation." The licensee determined that with a total combined flow of 77 gpm there was still, on average, 19 gpm per pump and therefore the minimum flow requirement was met. On February 20, 2007, Entergy performed ultrasonic flow measurements on the individual cooling lines to each RCP. It was determined that the flow meters on 21 and 23 RCP were not indicating correctly. The actual flow was 12.5 gpm with an

indicated flow of 22 gpm for 21 RCP, and an actual flow of 17 gpm with an indicated flow of 27 gpm for 23 RCP.

The inspectors reviewed Entergy's analysis for operability and determined that the minimum requirement of 13 gpm was not appropriate since the minimum flow required to ensure adequate cooling is temperature dependent. CCW cooler outlet temperature is normally maintained between 80 and 90 degrees Fahrenheit. For that temperature band, a minimum flow of 19 gpm would be required to ensure adequate thermal barrier cooling. In addition, the inspectors reviewed the work history associated with the individual flow meters, and determined that these indicators were not in a preventive maintenance program and no functional or channel checks were performed on these instruments. No method was established to assure the accuracy of the individual flow measuring devices. During CCW flow balancing, these indicators are used to establish the required design flow to ensure adequate cooling for the CCW thermal barriers.

Analysis. The inspectors determined that the failure to establish testing required to assure adequate cooling water flow to the thermal barriers to ensure they could perform satisfactorily when required was a performance deficiency and did not meet the requirements of 10 CFR 50 Appendix B, Criterion XI, "Test Control." 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 inspectors determined that this finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, Entergy's test program did not assure that all testing required to demonstrate that the RCP thermal barriers will perform satisfactorily in service because no testing was performed to ensure the accuracy of the individual flow meters used to establish the required cooling water flow. Consequently, it was identified that two individual flow indicators did not read correctly and the CCW flow to two RCP's was not sufficient to assure adequate cooling in the event that seal water was lost based on the flow requirements established in design calculations. On a loss of seal injection, the cooling water flow did not ensure that the heat removal capability was adequate to prevent a rise in seal temperature which would require the RCP to be stopped with a subsequent reactor trip. The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." This finding was determined to be of very low safety significance since it would not result in exceeding the Technical Specification limit for identified reactor coolant system leakage and would not have likely affected other mitigating systems resulting in a loss of their safety function. Entergy performed and evaluation which determined that the maximum temperature at the seal in conjunction with a loss of seal water, given the as found flow conditions and the maximum CCW temperature over the last year of operation. They determined the condition would not have resulted in the RCP seals reaching a temperature that would adversely impact seal performance.

Enforcement. 10 CFR 50 Appendix B, Criterion XI, “Test Control,” requires, in part, that a test program be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, prior to February 20, 2007, Entergy failed to establish testing to assure the accuracy of the CCW individual flow indicators for RCP thermal barrier heat exchangers, which are used to establish the minimum required cooling water flow to assure the thermal barriers will perform satisfactorily in service. Specifically, no preventive maintenance or functional checks were performed on the individual flow indicators to validate the accuracy of the installed instrumentation. Entergy entered this issue into their corrective action program (CR-IP2-2007-00783 and 00955), adjusted individual cooling water flow within the nominal band using ultrasonic flow meters, wrote work orders to replace the faulty meters, and is conducting an evaluation to determine the appropriate test requirements for the flow indicators. Because this issue is of very low safety significance and is entered into Entergy’s corrective action program, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC enforcement manual. **(NCV 05000247/2007002-02, Failure to Establish Testing to Assure Adequate Cooling Water Flow to the RCP Thermal Barriers)**

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

.1 Service Water Intake Bay Level Monitoring

a. Inspection Scope

The inspectors reviewed modification documents and reviewed the installation and testing of modifications to the Indian Point Nuclear Generating Unit 3 service water bay in accordance with modification ER-05-25451, "Mounting of Permanent Service Water Bay Level Indication." The modifications added level indicators to the Indian Point Unit 2 and Indian Point Unit 3 service water bay to provide low water level indications in support of Emergency Action Level criteria. The modification to install a post with calibrated level markings was completed under work order IP3-05-25367. The review of this modification represented one inspection sample.

b. Findings

Introduction. A Green, self-revealing, finding was identified because Entergy failed to take adequate corrective actions for an issue associated with monitoring of service water intake bay level. Specifically, Entergy’s daily performance of intake bay level measurements could have prevented identification of entry conditions for an emergency action level (EAL) under the Emergency Plan.

Description. In November 2005, NRC inspectors identified a Green NCV because Entergy did not have adequate indications available to determine if the entry condition for a notification of unusual event (UE) had been met. Specifically, EAL 8.4.3 requires

declaration of a UE if service water intake bay level reaches 4 feet 5 inches below mean sea level. At the time, Entergy did not have an established means to measure intake bay level, or any instrumentation available to plant operators to assess intake bay level, as required by 10 CFR 50.47(b)(4). The NRC issued NCV 05000247/2005005-05, "Inadequate Equipment to Assess Threshold for Emergency Action Level 8.4.3." In response, Entergy entered the issue into the corrective action program and installed a level measuring device in the service water intake bay.

On February 5, 2007, Indian Point Units 2 and 3 experienced low levels in the service water intake bay due to a combination of debris clogging of the intake trash racks and an unusually low tide. Operators were alerted to this condition because the Indian Point 3 non-safety-related screen wash pumps had tripped due to low suction pressure, resulting in a control room alarm. Indian Point Unit 3 operators responded to the intake bay area, observed the installed, intake bay level measuring device, and determined that the entry conditions for a UE were met. Indian Point Unit 3 operators declared a UE at 7:07 a.m. on February 5, which was terminated at 10:14 a.m. when water level increased above the UE entry conditions. Indian Point Unit 2 also experienced lower than normal service water intake bay levels, but did not meet the entry conditions for a UE.

Following the February 2007 UE, the inspectors reviewed Entergy's corrective actions from the November 2005 NCV. The inspectors reviewed Entergy's method of monitoring service water intake bay level, and reviewed alarm response and abnormal operating procedures associated with service water system. The inspectors determined that while Entergy had installed a measuring device, it was not used in a manner to provide assurance that the entry conditions for a UE would be identified in a timely manner. Specifically, while the device was used to measure intake level as a part of operator rounds, the readings were not trended and were only recorded once per day with no time specified for when intake bay level should be measured. As a result, the readings could potentially be taken during periods of high tide, which could mask subsequent low level conditions in the service water intake bay. Additionally, the inspectors reviewed both alarm response procedures and abnormal operating procedures, and identified that existing plant procedures did not provide sufficient guidance to operators to identify and mitigate low level conditions in the intake bay. Plant procedures did not direct the operators to check service water intake bay level following the trip of screen wash pumps, required no specific actions if service water bay level was low out of specification on operator logs, and provided no actions to assist operators in mitigating a low level condition, once identified. These issues were also identified by Entergy during their root cause investigation of the February 2007 UE.

Entergy procedure EN-LI-102, "Corrective Action Process," requires that corrective actions address the cause or resolve the deficiency associated with an adverse condition. Attachment 9.2 of EN-LI-102 provides examples of adverse conditions, and includes actual or potential NRC violations, as well as conditions which could negatively impact reliability or availability. The inspectors determined that Entergy's actions to address the previous NCV did not appropriately correct a condition adverse to quality, as required by EN-LI-102.

Analysis. The inspectors determined that Entergy's failure to take adequate corrective actions for the improper monitoring of service water intake bay level was a performance deficiency. This issue was reasonably within Entergy's ability to foresee and prevent, given that the issue had been identified and documented in a condition report and the corrective action requirements were addressed in Entergy procedure EN-LI-102. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy procedures.

The inspectors determined that this finding was more than minor because it was associated with the facilities and equipment attribute of the Emergency Preparedness cornerstone; and, it affected the cornerstone objective of ensuring that a licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, inadequate monitoring of service water intake bay level could have resulted in failure to declare a UE. The inspectors reviewed the EAL entry criteria and determined that this performance deficiency did not affect Entergy's ability to declare any event higher than a UE. The inspectors evaluated this finding using IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process," Sheet 1, "Failure to Comply." Section 4.4 of IMC 0609, Appendix B, provides examples for use in assessing emergency preparedness findings. One example of a Green finding states, "The EAL classification process would not declare any alert or notification of unusual event that should be declared." Since the declaration of a UE based on low service water bay level could have been missed or delayed, this finding was considered consistent with the example provided and was therefore determined to be of very low safety significance (Green).

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not implement effective corrective actions for a previously identified issue associated with inadequate monitoring of service water intake bay level.

Enforcement. Because this finding is associated with a non-safety-related service water intake bay level monitoring function, no violation of regulatory requirements occurred. Entergy entered this issue into the corrective action procedure as CR IP3-2007-00453, and initiated several corrective actions, including plans for enhanced monitoring of service water bay levels, backwashing of trash racks, procedural upgrades, correction of service water bay level instrumentation modification installation, development of modifications for enhanced service water level monitoring equipment, and enhanced inspection and cleaning of intake structure trash racks. **(FIN 05000247/2007002-03, Inadequate Corrective Actions for Failure to Appropriately Monitor Service Water Intake Bay Level)**

.2 Unit 2 Containment Sump Modification during Spring 2006 Outage

a. Inspection Scope

The inspectors previously reviewed a modification to upgrade the containment and recirculation sumps. This modification was implemented using engineering request (ER) 04-2-234, "IP2 Recirculation Sump and Vapor Containment Sump Strainer Upgrade," to address concerns associated with pressurized water reactor containment sump clogging. This inspection was documented in Inspection Report 05000247/2006003. Subsequently, Entergy identified a number of instances where weld data sheets for the modification were missing, and the inspectors reviewed Entergy's disposition of this issue.

b. Findings

No findings of significance were identified.

c. Unresolved Item

Introduction: The inspectors identified an unresolved item associated with retention of weld data sheets for the Indian Point Unit 2 containment and recirculation sump upgrade. This issue is unresolved pending completion of Entergy's evaluation of this issue.

Description: During the Spring 2006 outage, Entergy completed a partial modification to install upgraded sump strainers in response to Generic Safety Issue 191, which was associated with debris-induced clogging of pressurized water reactor sumps. Prior to restart from the Spring 2006 outage, Entergy identified several instances where weld data sheets were missing for the sump modification. Entergy formed a reconstitution engineering team to recover the missing data sheets or disposition the missing data through engineering evaluation. This effort was completed and Entergy determined that the sump was operable prior to restart.

On January 22, 2007, Entergy learned that additional weld records for the sump strainer installation were potentially missing, and initiated an independent review into eight of the 63 completed work packages associated with the strainer modification. The review identified additional missing weld records which were lost, misplaced, or discarded, but which had not been identified or evaluated during the previous reconstitution effort. Entergy initiated CR IP2-2007-00699 on February 8, 2007, to document the results of the independent review and initiate corrective actions. Entergy completed an engineering review of the newly identified missing information and concluded that the sumps remained operable. Additional actions planned by Entergy include a review of the remaining containment sump work packages and a visual inspection of safety-related welds with missing weld data.

This issue is unresolved pending the completion of Entergy's review and NRC's subsequent evaluation. **(URI 05000247/2007002-04, Containment Sump Modification Missing Weld Data)**

1R19 Post-Maintenance Testing (71111.19 - 3 samples)

a. Inspection Scope

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

- WO IP2-07-12346, gas turbine 1 following corrective maintenance;
- WO IP2-06-25127, 23 emergency diesel generator following maintenance; and
- WO IP2-06-14865, 21 auxiliary boiler feedwater pump following maintenance.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors observed and reviewed activities during one Indian Point Nuclear Generating Unit 2 forced outage. The outage occurred between February 28 and March 1, 2007, following a reactor trip due to failure of the main feedwater pump suction pressure transmitter. The documents reviewed during this inspection are listed in the Attachment. The following activities were reviewed for the outage, which represented one inspection sample:

- 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 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. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors witnessed performance of surveillance tests and/or reviewed test data of selected risk-significant structures, systems and components to assess whether the they 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. Following the test, the inspectors verified that equipment was properly aligned to perform its safety function. The inspectors evaluated the surveillance tests against the requirements in TS. The documents reviewed during this inspection are listed in the Attachment. The following surveillance tests were reviewed and represented six inspection samples:

- 2-PT-M7, “Analog Rod Position Functional,” Revision 28;
- 2-PT-M021C, “Emergency Diesel Generator 23 Load Test,” Revision 13;
- 2-PT-Q56A and -Q56B, “6.9 kilovolt Undervoltage Relays Functional Test” and “6.9 kV Underfrequency Relays Functional Test,” Revision 3;
- 2-PT-V72, “IST (In Service Test) Relief Valve Tests,” Revision 0;
- 2-PT-Q29C, “23 Safety Injection Pump,” Revision 16; and
- 2PT-Q034, “22 ABFP(Auxiliary Boiler Feed Pump),” Revision 22.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

The inspectors reviewed the temporary modification TM-07-2-007, “Defeat of Gas Turbine 1 Lube Oil Sump Trip.” The inspectors assessed the adequacy of the 10 CFR 50.59 evaluations for this temporary modification and verified that the installation was consistent with the modification documentation, the drawings and procedures were

updated as applicable, and the post-installation testing was adequate. The documents reviewed during this inspection are listed in the Attachment. This inspection satisfied one inspection sample for temporary modifications.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Evaluation (7111402 - 1 sample)

a. Inspection Scope

Region-based specialist inspectors evaluated Entergy's corrective actions related to the existing Indian Point alert and notification system (ANS) failures, and reviewed the progress made in the design and installation of the new siren system. Inspection activities were conducted onsite throughout the quarter between January 16 and March 28, 2007. This inspection was conducted in accordance with the baseline inspection program deviation authorized by the NRC Executive Director of Operations (EDO) in a memorandum dated October 31, 2005, and renewed by the EDO in a memorandum dated December 11, 2006.

A new ANS is being installed around the Indian Point Energy Center to satisfy commitments documented in a NRC Confirmatory Order dated January 31, 2006, that implements the requirements outlined in the 2005 Energy Policy Act. In January 2007, Entergy requested an extension of the deadline for completing the ANS project as described in the Confirmatory Order, which set a January 30, 2007, deadline for completion of the installation. Entergy's extension request cited several issues that were beyond their control as the basis for the delay. On January 23, 2007, the NRC granted Entergy's extension request and established April 15, 2007, as the new installation completion date.

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

- Assessed Entergy's progress with the new ANS to validate Entergy's justification for the extension of the original Confirmatory Order deadline (January 16, 2007)

- Observed the first full-volume sounding of the new sirens (February 15, 2007)

- Reviewed Entergy's acceptance testing process for transfer of the ANS subsystem components from the vendor to Entergy (February 27-28, 2007)

- Observed and inspected the degraded voltage testing of the back-up batteries for the new ANS as described in the Test Plan for Indian Point Emergency Notification System in accordance with NRC Order EA-05-190 (dated July 5, 2006)

Note- This testing assured that the batteries at the central control units, the simulcast towers, and the sirens, would operate at their end-of-life condition following a loss of AC power for 24 hours. The inspectors observed the discharge of the batteries at one of the siren locations and at one of the simulcast towers, and observed the subsequent testing of the siren system with the batteries in the degraded condition (March 12-14, 2007).

- Observed and inspected full-volume sounding of the new sirens (March 21, 27, and 28, 2007)

During the onsite inspections cited above, the inspectors also reviewed the status of, and corrective actions for, the current ANS to assure that Entergy was appropriately maintaining the system.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors observed an emergency preparedness drill conducted on January 24, 2006. The inspectors used NRC Inspection Procedure 71114.06, "Drill Evaluation," as guidance and criteria for evaluation of the drill. The inspectors observed the drill and critiques that were conducted from the participating facilities on-site, including the Indian Point Unit 2 plant simulator, and the emergency operations facility. The inspectors focused the reviews on the identification of weaknesses and deficiencies in classification and notification timeliness, quality, and accountability of essential personnel during the drill. The inspectors observed Entergy's critique and compared the licensee's self-identified issues with the observations from the inspectors' review to ensure that performance issues were properly identified. The observation of the drill represented one inspection program sample.

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 - 7 samples)

a. Inspection Scope

On March 19 through 22, 2007, the inspectors 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, Technical Specifications, and Entergy's procedures.

(1) Radiation work permits were reviewed that provide access to exposure significant areas of the plant including high radiation areas. Specified electronic personal dosimeter alarm set points were reviewed with respect to current radiological condition applicability and workers were queried to verify their understanding of plant procedures governing alarm response and knowledge of radiological conditions in their work area.

(2) 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.

(3) Between March 19 through 22, 2007, the following, radiologically-significant work activities were selected; the radiological work activity job requirements were reviewed; and work activity job performance was reviewed with respect to the radiological work requirements:

- Refueling activities;
- Containment sump modification;
- 33 and 34 reactor coolant pump seal replacement activities;
- Reactor cavity drain down and reactor vessel head reinstallation; and
- 31, 32, 33, and 34 steam generator primary manway insert maintenance.

(4) During observation of the work activities listed in (3) above, the adequacy of surveys, job coverage and contamination controls were reviewed.

(5) There were no significant dose gradients requiring relocation of dosimetry for the radiologically significant work activities listed in (3) above.

(6) During observation of the work activities listed in (3) above, radiation worker performance was evaluated with respect to the specific radiation protection work

requirements and their knowledge of the radiological conditions in their work areas.

(7) During observation of the work activities listed in (3) above, radiation protection technician work performance was evaluated with respect to their knowledge of the radiological conditions, the specific radiation protection work requirements and radiation protection procedures.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

During March 19 through 22, 2007, the inspectors 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.

(1) The following highest exposure work activities for the Spring 2007 Unit 3 refueling outage were selected for review:

- Refueling activities;
- Containment sump modification;
- 33 and 34 reactor coolant pump seal replacement activities;
- Reactor cavity drain down and reactor vessel head reinstallation; and
- 31 through 34 steam generator primary manway insert maintenance.

(2) With respect to the work activities listed in (1) above, these job sites were observed to evaluate if surveys and ALARA controls were implemented as planned.

(3) With respect to the work activities listed in (1) above, radiation worker and radiation protection technician performance was observed during the performance of these work activities to demonstrate the ALARA principles.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151 - 3 samples)

a. Inspection Scope

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

Initiating Event Cornerstone

- Unplanned Scrams per 7000 Critical Hours
- Unplanned Transients per 7000 Critical Hours

Barrier Integrity Cornerstone

- Reactor Coolant System Activity

The inspectors reviewed data and plant records from January 2006 to December 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.

4OA2 Identification and Resolution of Problems (71152 - 2 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 to 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 causal analyses, extent of condition reviews, operability determinations, and the timeliness of the

specified corrective actions. The CRs reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 PI&R Annual Sample Review: Maintenance Rule Scoping for Emergency Operating Procedure Equipment (71152 - 1 sample)

a. Inspection Scope

The inspectors conducted a review of CR IP3-2006-00254, which identified that a thorough review of Maintenance Rule scoping of SSCs was required to determine applicability for both Indian Point Units 2 and 3. This was done to ensure that SSCs used in the EOPs were properly scoped following the identification that the control rod drive fans were not within Entergy's Maintenance Rule program, as required. The inspectors evaluated the extent of condition review as well as the adequacy and effectiveness of the associated corrective actions. The inspectors reviewed the EOPs and cross-referenced to Maintenance Rule SSCs to determine whether any components had been improperly assessed. In addition, the inspectors reviewed applicable engineering requests and documentation to support the review.

b. Findings and Observations

Introduction: The inspectors identified a Green, NCV of 10 CFR 50.65(a)(2) because Entergy did not demonstrate that the performance or condition of the Indian Point Unit 2 containment hydrogen monitoring system was being effectively controlled through the performance of appropriate preventive maintenance, such that the system remained capable of performing its intended function.

Description: The inspectors identified that both channels of the containment hydrogen/oxygen (H_2/O_2) analyzers had been out of service since September 7, 2006, due to compressor seal leakage. Both had open work orders for repair, but they were classified as elective maintenance instead of corrective maintenance. One channel was scheduled to be worked the week of May 7, 2006, and the other had not been scheduled. The inspectors noted that a monthly calibration check is performed on both channels to ensure functionality, but these checks had been deferred since both channels were inoperable.

The hydrogen analysis function of the H_2/O_2 analyzers is used to evaluate the Indian Point Unit 2 containment atmosphere and assess the degree of core damage during a beyond design basis accident. If an explosive mixture that could threaten containment integrity exists during a beyond design basis accident, then other severe accident management strategies would need to be considered. The hydrogen monitoring function is needed to evaluate containment atmospheric conditions and implement appropriate strategies for severe accident management. The NRC authorized the

removal of the H_2/O_2 analyzers from Entergy's TS in April 2005, since the equipment is not required to mitigate design basis accident's, is not risk-significant, and does not meet the definition of a safety-related component. However, since hydrogen monitoring is required to diagnose the course of beyond design basis accidents, the safety evaluation approving the removal of the components from TS required that Entergy make a regulatory commitment to maintain the functionality of the hydrogen monitoring system. Entergy committed to include the hydrogen monitors in a preventive maintenance program to assure they are maintained reliable and functional.

The inspectors determined that the H_2/O_2 analyzers are within the scope of Entergy's Maintenance Rule program since they are used in the emergency operating procedures. The system was classified by Entergy as being in a Maintenance Rule (a)(2) status. This classification requires performance of the system to be effectively controlled through preventive maintenance, such that the system remained capable of performing its intended function. Based on the significant unavailability time of both trains, the inspectors noted the system should have been in 10 CFR 50.65(a)(1) status with an action plan to improve system performance back to an (a)(2) status.

Analysis: The inspectors determined the failure to demonstrate effective control of the performance and condition of the H_2/O_2 analyzers, or put the system in Maintenance Rule (a)(1) status, was a performance deficiency. Entergy did not meet the requirements of 10 CFR 50.65(a)(2), which specifies that monitoring of structures, systems, or components (SSCs) as specified in (a)(1) is not required when it is demonstrated that performance is being effectively controlled through appropriate preventive maintenance. 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 inspectors determined that this finding affected the Barrier Integrity cornerstone and was more than minor since it was similar to Example 7.b in IMC 0612, Appendix E, "Examples of Minor Issues." Specifically, Entergy failed to demonstrate effective control of the performance of the H_2/O_2 analyzers and did not place the system in (a)(1). The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The finding required further evaluation through IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," since it resulted in an actual reduction in the defense-in-depth for the hydrogen control function of the reactor containment. The inspectors determined that this finding was of very low safety significance because it did not affect core damage frequency and the H_2/O_2 analyzers are not important to large early release frequency.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not ensure that equipment and resources were available to assure reliable operation of the H_2/O_2 analyzers. Specifically, Entergy did not minimize long-standing equipment issues and maintenance deferrals associated with the containment hydrogen monitoring system.

Enforcement: 10 CFR 50.65(a)(1) requires, in part, that licensees monitor the performance or condition of SSCs within the scope of the rule as defined by 10 CFR 50.65(b) against licensee-established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. 10 CFR 50.65(a)(2) states, in part, that monitoring as specified in 10 CFR 50.65(a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. Contrary to the above, prior to February 6, 2007, Entergy failed to demonstrate that the performance or condition of the containment H_2/O_2 analyzers was being effectively controlled through the performance of appropriate preventive maintenance, and had not monitored the performance of the system against established goals. Specifically, both channels of the H_2/O_2 analyzers had been out of service since September 7, 2006, which demonstrates that the system's performance was not being effectively controlled through preventive maintenance, and goal setting and monitoring had not been not implemented as required. Entergy entered this issue into their corrective action program (CR IP2-2007-00783 and -00955) and changed the priority of the work orders to perform repairs on the H_2/O_2 analyzers. One channel was brought back to service on February 23, 2007. In addition Entergy is reviewing the current methodology used to identify functional failures associated with structures, systems, and components covered by the Maintenance Rule program. Because this issue is of very low safety significance and is entered into the Entergy's corrective action program, this violation is being treated as an NCV consistent with Section VI.A.1 of th3e NRC Enforcement Policy: **(NCV 05000247/2007002-05, Failure to Move Containment Hydrogen Analyzers to 10 CFR 50.65(a)(1) Status)**

.3 PI&R Annual Sample - Aggregate Impact of Operator Workarounds (71152 - 1 sample)

a. Inspection Scope

The inspectors conducted a review of the aggregate impact of operator burdens and workarounds. The inspectors reviewed Entergy's implementation of procedures OAP-45, "Operator Burden Program," Revision 1 and PL-163, "Operations Expectations and Standards," Revision 2. The inspectors conducted control room walkdowns and interviewed plant operators to determine the impact of deficiencies on operator response to plant events. The inspectors verified that operator workarounds and burdens were appropriately entered into the corrective actions program and were dispositioned commensurate with their safety significance.

b. Findings and Observations

No findings of significance were identified. The inspectors determined that, in general, Entergy was appropriately entering issues that represented operator workarounds and burdens into the corrective action program. Issues that could impact operator response during plant events were appropriately prioritized and corrective actions were timely. However, the inspectors identified one example where Entergy's actions for a degraded

condition were inconsistent with the guidance in OAP-45. Specifically, the inspectors identified that operation with the main generator voltage regulator in manual control was not classified as an operator burden or workaround, even though adjustments were required several times a day and operation with the voltage regulator in manual could complicate operator response to certain plant transients. In addition, while operators were aware that additional actions would be required to prevent a generator trip following a main turbine runback, the impact on overall plant risk had not been assessed. These issues were evaluated by the inspectors and determined to be minor because operators were familiar with the actions necessary to prevent a generator trip and subsequent testing in the simulator demonstrated that it was likely operators would be successful.

.4 Occupational Radiation Safety Cornerstone

a. Inspection Scope

The inspector reviewed two CRs associated with the radiation protection program that were initiated between September and October 2006. The inspector verified that problems identified by these condition reports were properly characterized in Entergy's event reporting system, and that applicable causes and corrective actions were identified, commensurate with the safety significance of the radiological occurrences.

b. Findings and Observations

No significant findings or observations were identified.

4OA3 Event Followup (71153 - 4 samples)

.1 Manual Reactor Trip Due to Failure of the Main Feedwater Pump Suction Pressure Transmitter

a. Inspection Scope

The inspectors observed control room personnel response to an unexpected manual reactor trip on February 28, 2007, that resulted from failure of the main feedwater pump suction pressure transmitter. Failure of this transmitter caused an automatic runback of both main feedwater pumps. 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 these reviews was to confirm that Entergy had taken appropriate corrective actions prior to commencing restart activities. The documents reviewed are listed in Attachment.

b. Findings

No findings of significance were identified.

.2 (Closed) Licensee Event Report (LER) 05000247/2006002-00, Technical Specification Prohibited Condition for Two Inoperable Channels of Post-Accident Monitoring Instrumentation.

On July 24, 2006, Entergy determined that post-accident monitoring instrument LT-3300 was inoperable when it was noted that containment sump level indication unexpectedly changed when containment pressure changed. Subsequent evaluation determined that the instrument had been inoperable since May 16, 2006. Level transmitter LT-940 had previously been declared inoperable, therefore two channels were inoperable for a period of time greater than that allowed by Indian Point Unit 2 Technical Specifications. Entergy determined that troubleshooting and repair of LT-3300 at power could not be performed due to the component's location. A work order was initiated to repair the instrument during the next refueling outage. Subsequent to the event, a Technical Specification amendment was approved which allows continued operation with the failed channel, provided a report is submitted to the NRC pursuant with TS 5.6.6. This report was submitted and evaluated by the NRC staff. The inspectors reviewed LER 05000247/2006002-00, the associated condition report (CR IP2-2006-04402), and Entergy's causal analysis. No findings of significance or violations of NRC requirements were identified. This LER is closed.

.3 (Closed) LER 05000247/2006003-00, Manual Reactor Trip Due to a Mismatch Between Reactor Power and Turbine Load Caused by Cycling of Steam Dump Valves After a Power Reduction for Loss of Heater Drain Tank Pumps.

On August 23, 2006, control room operators manually tripped the reactor due to a mismatch between reactor power and turbine load. Power had been reduced from 100 percent to 77 percent following loss of both heater drain tank pumps. Operators were in the process of further reducing power to less than 50 percent, due to reactor core axial flux difference exceeding Technical Specification limits, when they identified a significant reduction in turbine load with no operator action. Operators determined that they did not have adequate control of the power reduction and initiated a manual reactor trip. Entergy determined that the mismatch between reactor power and turbine load was due to cyclic operation of the high pressure steam dump valves, which had been improperly calibrated. At the time of the reactor trip, this issue was reviewed by the NRC and two Green findings were identified in Inspection Report 05000247/2006004. The inspectors reviewed LER 05000247/2006003-00, the associated condition report (CR IP2-2006-05066), and Entergy's causal analysis. No additional findings of significance or violations of NRC requirements were identified. This LER is closed.

.4 (Closed) LER 05000247/2006004-00, Automatic Actuation of Both Motor Driven Auxiliary Feedwater Pumps Due to Trip of 21 Main Feedwater Pump Caused by High Vibrations.

On August 24, 2006, during plant startup following a manual reactor trip, the 21 main feedwater pump tripped due to high vibrations. At the time of the trip, both main feedwater pumps were isolated, the 22 main feedwater pump was shutdown, and the motor-driven auxiliary feedwater pumps were being used to feed the steam generators.

The trip of the 21 main feedwater pump resulted in an automatic actuation signal to the motor-driven auxiliary feedwater pumps, but because they were already running, there was no impact on plant operation. Entergy determined that the main feedwater pump vibrations were due to a procedural inadequacy which allowed the pump to be operated at its critical speed for an excessive period of time. Entergy entered this issue into the corrective action program (CR IP2-2006-5098) and revised the main feedwater system operating procedure to prevent recurrence. The inspectors reviewed LER 05000247/2006004-00, Entergy's causal analysis, and the associated corrective actions. No findings of significance or violations of NRC requirements were identified. This LER is closed.

4OA5 Other Activities

Groundwater Contamination Investigation

a. Inspection Scope

Continued inspection of Entergy's plans, procedures, and characterization activities regarding 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 dated October 31, 2005 (ADAMS Accession Number ML053010404) and renewed on December 11, 2006 (ADAMS Accession Number ML063480016). Accordingly, continued oversight of Entergy's progress has been conducted throughout this quarterly inspection period, consisting of on-site inspections; independent split sample analyses of selected monitoring well samples; frequent review of Entergy's performance, progress, and achievements; and periodic communications with Federal, State, and local government stakeholders.

The inspectors conducted an on-site review of tracer test sampling and waterloo sampler maintenance from February 26 to March 2, 2007. A teleconference was held on March 21, 2007, to discuss Entergy's preliminary data and interpretations of their groundwater tracer study, which began on February 8, 2007. The NRC team included representatives from the NRC's Region I office, as well as the NRC's Office of Nuclear Regulatory Research, the U.S. Geological Survey's New York Science Center, and the New York State Department of Environmental Conservation (NYS DEC). The teleconference provided for an independent hydrology review of Entergy's initial tracer test findings and associated re-evaluation of the current site groundwater model.

The tracer test objective uses groundwater tracing techniques by injecting fluorescent tracer dye into a ground location representing the source of leakage and tracks the natural groundwater progress as it is intercepted by existing monitoring wells and storm drain locations. This process better characterizes groundwater flow directions and flow rates in areas identified as being affected by water contaminated with strontium and tritium. The fluoresceine dye was injected into a tracer injection well next to existing monitoring well 30 (MW-30), which is adjacent to the Unit 2 spent fuel pool (SFP). On February 8, 2007, the test began with injection of approximately 200 gallons of dye at a

three gallons per minute at a ground elevation equivalent to the bottom of the Unit 2 SFP. The natural groundwater flow of this tracer test is expected to be tracked for approximately 13 weeks by measuring the dye content in charcoal and water samples taken at selected, on-site monitoring wells and storm drain locations.

Initial results indicated that dye tracer was detected within four hours of injection at shallow sampling levels of MW-31 and MW-32. After one day, tracer was detected at deeper levels within MW-31 and in recovery well 1 (RW-1). Direct water sampling was conducted in surrounding wells with carbon sampling devices in outer wells. Once the fluoresceine dye was detected in the carbon sampling devices, direct water sampling was performed to determine the dye concentration. Arrival times and concentrations of the dye were identified in the down-gradient wells and storm drains [e.g., manholes (MH-5 and later MH-6)] as the tracer test progressed. Ozark Underground Laboratory is analyzing the tracer samples and will be reporting their results to Entergy.

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. The NRC's assessment of Entergy's sample analytical results data generally indicated that their analytical contractor continued to report sample results that were consistent with NRC's analytical results.

The NRC's ORISE/ESSAP sample results are available in ADAMS under the following Accession Numbers: ML070940618, ML070940504, ML070940515, ML070940534, ML070940546, and ML070940574. To date, sample results from site boundary wells and off-site environmental groundwater sampling locations have not indicated any detectable plant-related radioactivity.

NRC's assessment of Entergy's interim tracer test results from February 8 to March 9, 2007, which included input from NYS DEC and U.S. Geological Survey hydrology experts, indicated that an additional complexity to the site groundwater model has been observed with some preferential fracture flow observed in the unsaturated zone (above the water table), as well as a general groundwater flow towards the Hudson River. Additional information will be obtained as the 13 week tracer test progresses to help clarify these initial observations in a later NRC review. Ultimately, clarification of groundwater flow rates of contaminants off-site toward the Hudson River is the focus of this NRC hydrology assessment. Together with monitoring well sample data, an accurate assessment of Entergy's effluent release reports and public dose assessments will result from these efforts.

Entergy and their contractors pointed to the preliminary nature of their data and interpretation. They agreed to provide timely data transfer with a technical meeting in May to review all of the tracer data, arrival times and concentrations. No further

pumping in RW-1 or other tracer tests will occur until the data has been reviewed and analyses have been conducted.

Remaining activities identified include: (1) completion of the direct sampling of the tracer in the monitoring wells; (2) preparation and analysis of breakthrough curves (tracer clearance rates) for the tracer at the monitoring wells differentiated by depth; (3) analysis of the breakthrough curve "tails" to determine the nature of groundwater flow (i.e., fracture flow or porous media flow); and (4) correlation of the earlier RW-1 pump test data with the tracer test data to further clarify and corroborate the groundwater flow model using these two independent tests utilizing different measurement parameters. Additional evaluation will continue as the tracer test concludes in May 2007 to assess the site groundwater contaminant flow direction and flow rate of the effluent groundwater releases to the Hudson River.

4OA6 Meetings, including Exit

Exit Meeting Summary

On April 4, 2007, the inspectors presented the inspection results to Mr. James Comiotes and other Entergy staff members, who acknowledged the inspection results presented. Entergy did not identify any material as proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

V. Andreozzi, Electrical Design Engineering Supervisor
N. Azevedo, Code Program Supervisor
J. Baker, Shift Manager
T. Beasley, System Engineer
C. Braun. Switchyard Coordinator
K. Brooks, Shift Manager
B. Christman, Manager of Training and Development
P. Cloughessy, System Engineer
P. Conroy, Director of Nuclear Safety Assurance
F. Dacimo, Site Vice President
R. Hansler, Reactor Engineering Superintendent
T. Jones, Licensing Supervisor
J. Kayani, System Engineer
S. Manzione, Component Engineering Supervisor
B. McCarthy, Shift Manager
B. Meek, Maintenance Supervisor
G. Mosher, System Engineer
E. O'Donnell, Indian Point Unit 2 Operations Manager
T. Orlando, Director of Engineering
D. Parker, Maintenance Superintendent
J. Pineda, System Engineer
K. Polson, General Manger of Plant Operations
B. Ray, Maintenance Superintendent
B. Sullivan, Emergency Planning Manager
P. Studley, Planning, Scheduling, and Outage Manager
M. Vasely, Balance of Plant System Engineering Supervisor
S. Verrochi, System Engineering Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000247/2007006-04	URI	Containment Sump Modification Missing Weld Data (Section 1R17)

Opened and Closed

05000247/2007002-01	NCV	Failure to Incorporate Design Basis Information into Procedures to Assure Adequate Cooling Water Flow to the RCP Thermal Barriers (Section 1R15)
05000247/2007006-02	NCV	Failure to Establish Testing to Assure Adequate Cooling Water Flow to the RCP Thermal Barriers (Section 1R15)
05000247/2007002-03	FIN	Inadequate Corrective Actions for Failure to Appropriately Monitor Service Water Intake Bay Level (Section 1R17)
05000247/2007002-05	NCV	Failure to Move Containment Hydrogen Analyzers to 10 CFR 50.65 (a)(1) Status (Section 4OA2)

Closed

05000247/2006002-00	LER	Technical Specification Prohibited Condition for Two Inoperable Channels of Post-Accident Monitoring Instrumentation (Section 4AO3.2)
05000247/2006003-00	LER	Manual Reactor Trip Due to a Mismatch Between Reactor Power and Turbine Load Caused by Cycling of Steam Dump Valves After a Power Reduction for Loss of Heater Drain Tank Pumps (Section 4OA3.3)
05000247/2006004-00	LER	Automatic Actuation of Both Motor-Driven Auxiliary Feedwater Pumps Due to trip of 21 Main Feedwater Pump Caused by High Vibrations (Section 4OA3.4)

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

IP2-2006-05455	IP2-2006-05351	IP2-2006-05702
IP2-2006-05438	IP2-2006-04676	IP2-2006-00058

Section 1R04: Equipment Alignment

Condition Reports

IP2-2006-06755	IP2-2006-06227	IP2-2006-07199
IP2-2006-06749	IP2-2006-05635	IP2-2006-07329
IP2-2006-02046	IP2-2006-04720	IP2-2007-01340
IP2-2006-01544	IP2-2006-07100	

Procedures

2-COL-21.3, "Steam Generator Water Level and Auxiliary Boiler Feedwater," Revision 29
2-PT-W1. "Emergency Diesel Generator," Revision 20
2-SOP-27-3.1.1, "21 Emergency Diesel Generator Manual Operation," Revision 14
2-SOP-27.3.1.3, "23 Emergency Diesel Generator Manual Operation," Revision 13
2-COL-10.2.1, "Containment Spray System," Revision 18
PT-Q35B, "22 Containment Spray Pump Test," Revision 14

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

Work Orders

IP2-2006-23407

Section 1R05: Fire Protection

Procedures

ENN-DC-161, "Transient Combustible Program," Revision 1

Condition Reports

IP2-2006-02072	IP2-2006-06494	IP2-2006-07003
IP2-2006-04906	IP2-2006-04946	

Section 1R06: Flood Protection Measures

Condition Reports

IP2-2002-04034	IP2-2006-02256	IP2-2007-00309
IP2-2003-06975		

Drawing

9321-F-4011, "Control Building"
9321-F-4002, "Control Building"
9321-F-4001, "Control Building"

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"
NRC Letter dated December 18, 1980 and Enclosed Safety Evaluation Report, "Susceptability of Safety-Related Systems to Flooding from Failure of Non-Category 1 Systems for Indian Point Nuclear Power Station Unit 2"
ER IP2-04-35340

Section 1R07: Heat Sink Performance

Program Documents

EN-DC-147, "Indian point Units 2 & 3 Eddy Current Program," Revision 2

IP2-EC-Guideline N0. 1, "IP2 Eddy Current Program Tube Plugging Criteria for BOP Heat Exchangers," Revision 1

Test and Inspection Results

Preliminary Report of Eddy Current Inspection - 21 CCW, 2/23/05 and 1/8/07

Condition Reports

IP2-2007-0011

Work Orders

IP2-05-12720

Section 1R11: Licensed Operator Requalification Program

Procedures

2-AOP-INST-1, "Instrument/Controller Failures," Revision 3
2-AOP-LICCW-1, "Leakage Into CCW System," Revision 2
2-AOP-ROD-1, "Rod Control and Indication Systems Malfunctions," Revision 3

Simulator Test Documentation

12SX-ILO-AOP001, "AOP Simulator Exam," Revision 1

Section 1R12: Maintenance Effectiveness

Condition Reports

IP2-2001-07205	IP2-2005-01450	IP2-2006-04739
IP2-2002-04034	IP2-2005-02338	IP2-2006-05827
IP2-2003-05474	IP2-2006-01299	IP2-2006-06509
IP2-2003-05771	IP2-2006-01367	IP2-2006-06607
IP2-2005-00698	IP2-2006-04720	

Drawings

A260589-01, "Gas Turbine #1 Flow Diagram, Lube Oil System"
B262047-01, "Gas Turbine #1 Lube Oil System Schematic"

Miscellaneous

Unit 2 Gas Turbines System Health Report 3rd Quarter 2006

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

Work Orders

IP2-2001-22494	IP2-2003-05195	IP2-2003-18686

Miscellaneous

Expert Meeting Panel Minutes Dated October 15, 2001; December 7, 2001; March 21, 2006; April 11, 2006; and September 28, 2006

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Condition Reports

IP2-2007-00013	IP2-2007-00971	IP2-2007-01335
IP2-2007-00022	IP2-2007-00972	IP3-2005-05389
IP2-2007-00205	IP2-2007-00975	IP3-2007-00453
IP2-2007-00561	IP2-2007-01333	IP3-2007-00459

Drawings

A208088, "One Line Diagram of 480V SWGRS 21 & 22 Bus 2A, 3A, 5A &6A,' Revision 42
B225137, "Elementary Wiring Diag of Residual heat Removal Pumps # 21 & 22," Revision 10
A250907, "Electrical Distribution and Transmission System," Revision 23
2003MD0684, "Indian Point & Buchanan System Ties," Revision 30
9321-F-20123, "Intake Structure General Arrangement," Revision 10
9321-3204, "120 VAC Distribution Panel 21"

Procedures

2-ARP-043, "Accident Assessment Panel 1," Revision 29
2-PC-Q4A, "Toxic Gas Monitor Calibration (Channel 1)," Revision 1
2-PC-Q4B, "Toxic Gas Monitor Calibration (Channel 2)," Revision 1
2-PT-Q13, "Inservice Valve Tests," Revision 39
2-PT-Q34, "22 Auxiliary Feed Pump," Revision 22
2-PT-Q34B, "PCV-1310A and PCV-1310B Nitrogen Supply," Revision 5

3-AOP-SW-1, “Service Water System Malfunction,” Revision 2
EN-LI-102, “Corrective Action Process,” Revision 8
IP-SMM-WM-100, “Work Management Process,” Revision 6
IP-SMM-WM-101, “On-Line Risk Assessment,” Revision 1

Work Orders

IP2-2006-10193	IP2-2006-32028	IP2-2007-10997
IP2-2006-11676	IP2-2006-32029	IP2-2007-34280

Miscellaneous

21 RHR Pump Breaker Extent of Condition Action Plan, dated 1/4/07
IP2-NUCLEAR HVAC DBD, “Design Basis Document for CCR HVAC System,” Revision 1
Operators Risk Report, 1/31/07
Operators Risk Report, 1/11/07
Operators Risk Report, 3/05/07
Operators Logs, 1/31/07
NF-IP-06-53, “Indian Point 2 EOC 17 Post Irradiation Inspection Report”

Section 1R15: Operability Evaluations

Calculations

WCAP-12312, “Safety Evaluation for an ultimate heat Sink Temperature Increase to 95°F at Indian Point Unit 2,” Revision 2

Procedures

2-PT-R22A, “Steam Driven Auxiliary Feed Pump Full Flow,” Revision 13
2-PT-Q34, “22 Auxiliary Feed Pump,” Revision 21
2-SOP-4.1.2, “Component Cooling System Operation,” Revision 30
2-PC-2Y625, “Reactor Coolant Pump Thermal Barrier Component Cooling Header Flow,” Revision 1
3-PT-Q120B, “32 ABFP (Turbine Driven) Surveillance and IST,” Revision 10
SOP 31.1.4, “Gas Turbine 1 Fuel Oil Receipt,” Revision 9
0-CY-1810, “Diesel Fuel Oil Monitoring,” Revision 5
0-CY-3320, “Relative Density / Specific Gravity of Fuel Oil,” Revision 1

Condition Reports

IP2-2006-07112	IP2-2006-07148	IP2-2007-00745
IP2-2006-07120	IP2-2007-00117	IP2-2007-00922
IP2-2006-07123	IP2-2007-00587	IP2-2007-00980
IP2-2006-07125	IP2-2007-00722	

Work Orders

IP2-1999-07992 IP2-2007-12445

Miscellaneous

10 CFR Part 21 Report 21-06-03
IP2-AFW DBD, "Auxiliary Feedwater System," Revision 1
UFSAR Section 4.2.2.4, "Reactor Coolant Pumps," Revision 20
MPR-2980, "Evaluation of Ultra Low Sulfur Diesel Fuel for Use in EDGs," Revision 0

Section 1R17: Permanent Plant Modifications

Modifications

ER-05-28310, Detail for SW Bay Level Measurement Device

Condition Reports

IP2-2007-00690

Work Orders

IP2-05-28235

Drawings

9321-F-20123-10, "Service Water Bay Elevation Diagram," Revision 10

Procedures

EN-DC-105, "Configuration Management," Revision 0
ENN-DC-103, "Design Process," Revision 1
ENN-DC-115, "ER Response Development," Revision 6
OAP-031, "Control of Operator Aids," Revision 0

Section 1R19: Post-Maintenance Testing

Condition Reports

IP2-2006-07329 IP2-2007-00397

Miscellaneous

SYS-APL-07-001, “Action Plan - Gas Turbine System Reliability,” Revision 0

Procedures

2-PT-M021C, “Emergency Diesel Generator 23 Load Test,” Revision 13

Work Orders

IP2-2005-01180 IP2-2006-25127 IP2-2007-12346

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
2-POP-2.1, “Operation at Power, Mode 1,” Revision 46

Condition Reports

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

Section 1R22: Surveillance Testing

Procedures

2-PT-M7, “Analog Rod Position Functional,” Revision 27
2-PT-Q027B, “23 Auxiliary Feed Pump,” Revision 14
2-PT-Q34B, “PCV-1310A and PCV-1310B Nitrogen Supply,” Revision 5
2-PT-M021C, “Emergency Diesel Generator 23 Load Test,” Revision 13
2-PT-Q029C, “23 Safety Injection Pump,” Revision 16

Condition Reports

IP2-2005-02167 IP2-2006-01027

Drawings

329489, “AFW Pump 23,” Revision 0

Calculations

FIX-00069, "Auxiliary Feedwater Pump Instrument Accuracy For Recirculation Flow Control and ASME Section XI Testing," Revision 3

Work Orders

IP2-2006-01361 IP2-2006-13589

Section 1R23: Temporary Modifications

Procedures

2-SOP-31.1.2, "Gas Turbine 1 Local Operations," Revision 25

Drawings

A260589, "Gas Turbine #1 Flow Diagram Lube Oil System," Revision 1
9975-296, "Diagram - Functional Block," Revision A
NA-8-F-918, "Electric Board - Turbine Duplex Control Board," Revision 13

Miscellaneous

TM-07-2-007, "Removal of Lube Oil Sump Low Level Switch Signal"

Section 2OS1: Access Control to Radiologically Significant Areas

Condition Reports

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

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, 2/6/06 - 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 "Access Control for Radiologically Controlled Areas," Revision 1
Self-Assessment: Control of Contamination and Radioactive Material, 9/11/06 - 9/15/06
Snap Shot Self-Assessment: Exposure Reduction through Permanent Scaffold and Shielding

Section 2OS2: ALARA Planning and Controls

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LIST OF ACRONYMS

ABFP	auxiliary boiler feed pump
ADAMS	agency wide document and management system
ALARA	as low as is reasonably achievable
ANS	alert and notification system
AOP	abnormal operating procedure
CCW	component cooling water
CFR	Code of Federal Regulations
CR	condition report
EAL	emergency action level
EOP	emergency operating procedure
ESSAP	Environmental Site Survey and Assessment Program
EDO	Executive Director of Operations
GL	generic letter
gpm	gallons per minute
IMC	inspection manual chapter
IST	in service test
LER	Licensee Event Report
MH	manhole
MW	monitoring well
NCV	non-cited violation
NRC	Nuclear Regulatory Commission
NYS DEC	New York State Department of Environmental Conservation
ORISE	Oak Ridge Institute for Science and Education
PARS	Publically Available Records System
PI	performance indicator
PI&R	problem identification and resolution
RCP	reactor coolant pump
RW	recovery well
SDP	significance determination process
SFP	spent fuel pool
SSC	structure, system, and component
TS	Technical Specifications
UE	notification of unusual event
UFSAR	Updated Final Safety Evaluation Report
WO	work order