November 10, 2003

Mr. Fred Dacimo
Site Vice President
Entergy Nuclear Northeast
Indian Point Energy Center
295 Broadway, Suite 1
Post Office Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 2 - NRC INTEGRATED INSPECTION REPORT 050000247/2003011

Dear Mr. Dacimo:

On September 27, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at the Indian Point Nuclear Generating Unit 2 (Indian Point 2). The enclosed integrated inspection report documents the inspection findings, which were discussed on October 8, 2003, with Mr. John Ventosa 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, the inspectors identified four findings of very low safety significance (Green). Three of the findings were determined to be violations of NRC requirements. However, because of the very low safety significance and because the issues have been addressed and entered into your corrective action program, the NRC is treating these three findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-001; with copies to the Regional Administrator, Region 1; the Director, Office of Enforcement; and the NRC Resident Inspector at Indian Point 2.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial power nuclear power plants during calender year 2002 and the remaining inspection activities for Indian Point 2 were completed in January 2003. The NRC will continue to monitor overall safeguards and security controls at Indian Point 2.

In accordance with 10 CFR 2.790 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). Should you have any questions regarding this report, please contact Mr. David Lew at 610-337-5120.

Sincerely,

/RA/

Brian E. Holian, Deputy Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000247/2003011
w/Attachment: Supplemental Information

cc w/encl:
G. J. Taylor, Chief Executive Officer, Entergy Operations
M. R. Kansler, President - Entergy Nuclear Operations
J. Herron, Senior Vice President and Chief Operating Officer
C. Schwarz, General Manager - Plant Operations
D. Pace, Vice President, Engineering
R. Edington, Vice President, Operations Support
J. McCann, Manager, Nuclear Safety and Licensing
J. Kelly, Director, Nuclear Safety Assurance
J. Comiotes, Director, Nuclear Safety Assurance
C. Faison, Manager, Licensing
H. Salmon, Jr., Director of Oversight
J. Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc.
P. Smith, Acting President, New York State Energy, Research and Development Authority
J. Spath, Program Director, 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
T. Walsh, Secretary, NFSC, Entergy Nuclear Operations, Inc.
D. O'Neill, Mayor, Village of Buchanan
J. G. Testa, Mayor, City of Peekskill
R. Albanese, Executive Chair, Four County Nuclear Safety Committee
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
P. Rubin, Manager of Planning, Scheduling & Outage Services
Assemblywoman Sandra Galef, NYS Assembly
C. Terry, Niagara Mohawk Power Corporation
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
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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
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Senator Charles Schumer
J. Riccio, Greenpeace
A. Matthiessen, Executive Director, Riverkeepers, Inc.
M. Kapolwitz, Chairman of County Environment & Health Committee
A. Reynolds, Environmental Advocates
M. Jacobs, Director, Longview School
D. Katz, Executive Director, Citizens Awareness Network
P. Gunter, Nuclear Information & Resource Service
P. Leventhal, The Nuclear Control Institute
K. Copeland, Pace Environmental Litigation Clinic
R. Witherspoon, The Journal News
W. DiProfio, PWR SRC Consultant
W. Poole, PWR SRC Consultant
W. Russell, PWR SRC Consultant
W. Little, Associate Attorney, NYSDEC

Distribution w/encl:
H. Miller, RA/J. Wiggins, DRA (1)
R. Laufer, NRR
D. Lew, DRP
H. Nieh, RI EDO Coordinator
P. Habighorst, SRI - Indian Point 2
M. Cox, RI - Indian Point 2
W. Cook, DRP
T. Jackson, DRP
P. Milano, PM, NRR
G. Vissing, PM, NRR (Backup)
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	05000247/2003011
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point Nuclear Generating Unit 2
Location:	Buchanan, NY
Dates:	June 29, 2003 - September 27, 2003
Inspectors:	P. Habighorst, Senior Resident Inspector L. James, Resident Inspector M. Cox, Resident Inspector W. Cook, Senior Project Engineer J. McFadden, Health Physicist J. Laughlin, Emergency Planning Specialist A. Burritt, Senior Resident Inspector, Limerick T. Fish, Senior Operations Engineer T. Jackson, Project Engineer P. Patniak, Senior Engineer, NRR P. Drysdale, Senior Resident Inspector, Indian Point Unit 3 R. Berryman, Resident Inspector, Indian Point Unit 3
Approved by:	David C. Lew, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000247/2003-011; 06/29/2003 - 09/27/2003; Indian Point Nuclear Generating Station, Unit 2; Maintenance Effectiveness, Maintenance Risk Assessment/Emergent Work, and Personnel Performance During Non-Routine Plant Evolutions or Events.

The report covered a three-month period of inspection by resident, region-based inspectors, and a headquarters-based inspector. Three Green non-cited violations (NCVs), one Green finding, and one unresolved item were identified. 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: Mitigating Systems

- Green. The inspector identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V. In November 2002, a maintenance work instruction to install a 21 emergency diesel generator (EDG) service water supply flexible coupling did not include critical installation steps per the vendor manual. This resulted in a significant service water leak from the expansion joint on August 14, 2003.

  This finding is greater than minor since if left uncorrected, it could be a more significant safety concern as this type of flexible coupling is used on all three EDGs. The inspectors determined that the expansion joint leakage was of a very low safety significance since it did not adversely impact service water cooling to the emergency diesel generator or the overall service water system cooling capability, did not impact equipment and functions associated with internal flooding in the diesel generator room, and did not result in a loss of service water or emergency power safety function that contributed to internal flooding initiated events. (Section 1R12)

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion III, involving the design change package (DCP-200105716-I) to replace a pressurizer level recorder which did not contain accurate design details. As a consequence, during installation of the design change an unintended plant transient challenged operators.

  This finding is greater than minor based upon NRC Manual Chapter 0612, Appendix E, example 4.b. This finding is of very low safety significance. The finding did contribute to the likelihood of a reactor trip; however, it did not impact the availability of mitigation equipment, increase the likelihood of a primary or secondary system LOCA, or increase the likelihood of an internal fire or flood. (Section 1R13)

- <u>Green</u>. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion V, involving an incomplete procedure for restoring to service the 22 seal injection filter from maintenance. The consequence was an approximate 70 gallon per minute chemical volume and control system leak through an open vent valve which lasted for approximately two minutes before operators identified and shut the vent valve.

  This finding is more than minor since it adversely impacted the Mitigating System Cornerstone objective of safety system capability and availability with respect to the attributes of configuration control and procedural quality. The inadequate restoration procedure resulted in a significant chemical and volume control system leak (the capacity of one coolant charging pump) that degraded normal charging flow and emergency boration capability for a short period of time. The finding is of very low safety significance since it did not result in a loss of emergency boration safety function. (Section 1R14)

- <u>Green</u>. The inspectors identified a finding involving inadequate corrective actions associated with multiple leaks on a six-inch fire header in the Unit 1 turbine building. On September 10, 2003, an 80 gallon per minute fire header leak occurred that operators isolated by depressurizing the entire fire water suppression system at Unit 2 for approximately three hours. This leak occurred approximately one foot from a similar through-wall leak which occurred on July 16, 2003.

  This performance issue is considered more than minor based on example 4.f. in MC 0612 Appendix E. The performance finding involves the Mitigating Systems Cornerstone objective of fire suppression system availability to respond to fires. The finding is very low risk significance based upon the results from the fire protection risk significance screening methodology (FPRSSM). The finding impacts both manual suppression capability and automatic suppression capability. (Section 1R13)

B. <u>License-Identified Violations</u>

A violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program.

- TS 3.2.B.3 requires one flow path from the boric acid storage tank to the reactor coolant system. Contrary to this, the licensee determined that between April 22 and 26, 2001, both trains of the boric acid storage system were inoperable for 54.3 hours. This licensee identified finding is of very low safety significance because a loss of safety function for emergency boration did not exist since the refueling water storage tank was available to the suction of the charging pumps.

REPORT DETAILS

<u>Summary of Plant Status</u>

The unit began the inspection period at 100% power. On August 3, 2003, a reactor trip loss of normal power event occurred due to off-site electrical disturbances on the 345 Kilovolt (KV) system. On August 14, 2003, a loss of all off-site power occurred that resulted in a reactor trip/turbine trip. Both of the above reactor trips will be evaluated and documented in a special team inspection report (50-247/2003-013). Unit 2 returned to full power operations on August 17th at 12:58 a.m. The unit remained at full power throughout the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity and Emergency Planning

1R01 Adverse Weather Protection

a. Inspection Scope (71111.01)

The inspector evaluated Entergy's preparations in response to Hurricane Isabel between September 16-19, 2003. The inspectors evaluated the following documents associated with licensee controls and actions in response to high winds.

- Technical Specification 4.12.a
- Past corrective actions from Hurricane Floyd (CRs IP2-CR-199907062 and IP2-CR-199907424
- Entergy's Severe Weather Action Plan
- Abnormal Operating Instruction (AOI) 28.0.7, "Severe Weather" Revision 13
- Individual Plant Examination of External Events, Section 6, "High Winds, Floods and Others"

The inspector performed site and switch yard walkdowns to identify potential missile hazards, availability of mitigating equipment, fuel tank levels for gas turbines and emergency diesel generators, and potential security response vulnerabilities. The inspector also verified the appropriateness of the licensee's response to control room operator contingency briefings on emergency operating procedures, emergency action levels, and abnormal operating instructions. The inspector reviewed scheduled surveillance and maintenance activities to ensure maximum availability of mitigating equipment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope (71111.04Q and 71111.04S)

Partial System Walkdowns: The inspectors performed system walkdowns during periods of system train unavailability in order to verify that the alignment of the available train was proper to support the safety function. The inspectors also reviewed licensee identified equipment discrepancies that could potentially impair the functional capability of the available train.

On September 2, 2003, the inspector performed a partial system walkdown of the nitrogen back-up to instrument air system. The inspector used the check off list (COL) 29.2 and procedure OASL 15.26, “Component Status Control and Position Verification,” during the walkdown to verify general condition of the system and correct system alignment.

On September 9, 2003, a partial system alignment check of the 23 emergency diesel generator lube and fuel oil support systems occurred during preventative maintenance on the 21 emergency diesel generator. The inspector used check off list (COL) 27.3.1, “Diesel Generators,” to check for correct valve and power alignments.

Complete System Walkdown: The inspector performed a walkdown of accessible portions of the central control room (CCR) heating ventilation and air conditioning (HVAC) system to verify proper system alignment and identify any discrepancies that may adversely impact the function of the system. The inspector also verified that the licensee had properly identified and resolved equipment problems that could impact the availability and functional capability of this accident mitigation system. The inspector selected the CCR HVAC system based upon its importance to operator and plant safety. The inspector reviewed a number of documents listed in the attachment to this report, to confirm system availability and functional capability.

b. Findings

No findings of significance were identified.

1R05 Fire Protection - Plant Tours

a. Inspection Scope (71111.05)

The inspector toured areas important to plant safety and risk based upon a review of Section 4.0, “Internal Fires Analysis,” and Table 4.6-2, “Summary of Core Damage Frequency Contributions from Fire Zones,” in the Indian Point 2 Individual Plant Examination for External Events (IPEEE). The objective of this inspection was to determine if the licensee had adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, and had adequately established compensatory measures for degraded fire protection equipment. The inspector evaluated conditions related to: 1) licensee control of transient combustibles and ignition sources; 2) the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and 3) the fire barriers used to prevent fire damage or fire propagation. The areas reviewed were:

- Fire Zone 11, 21 Battery Room
- Fire Zone 12, 22 Battery Room
- Fire Zone 59A, Primary Auxiliary Building, Fuel Storage Building, and Containment Ventilation System Area
- Fire Zone 23, Auxiliary Boiler Feed Pump Room
- Fire Zone 10, Emergency Diesel Generator Building
- Fire Zone 1A, Electrical and Piping Tunnel, Piping Penetration Area
- Fire Zone 2A, Primary Water Make-up Pump Room
- Fire Zone 2, Containment Spray Pump Area

Reference material consulted by the inspector included the Fire Protection Implementation Plan, Pre-Fire Plan, and Station Administrative Orders (SAOs)-700, "Fire Protection and Prevention Policy," SAO-701, "Control of Combustibles and Transient Fire Load," SAO-703, "Fire Protection Impairment Criteria and Surveillance," and Calculation PGI-00433, "Combustible Loading Calculation." The inspector identified a number of minor items related to drawing errors in the pre-fire plan sketch, and penetration drawing errors and housekeeping concerns. The condition reports are identified in the attachment to this report.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope (71111.06)

The inspector reviewed and toured various elevations in the Primary Auxiliary Building (PAB) containing equipment used to mitigate and detect an internal flood within the building. The plant areas selected contained risk significant equipment based on the Individual Plant Examination of External Events (IPEEE) Section 5.2.2.1. Specifically, internal flood initiations from service water, fire protection line breaks, and refueling water storage tank breaks in the primary auxiliary building contributed approximately six percent of the overall core damage frequency from internal floods. The inspection verified the accuracy of the descriptive text contained in the IPEEE and compared it with actual plant conditions in the PAB.

The inspector reviewed applicable licensee procedures which included actions to mitigate the effects of flooding, and reviewed past pertinent condition reports. The documents reviewed are listed in the attachment to this report.

b. Findings

The inspector observed a number of spliced electrical cables in one of the supporting trays in the piping penetration, located beneath and about six feet from the containment spray piping. The inspector noted that the licensee had previously generated condition report CR-IP2-2002-02474 on March 5, 2002 identifying the presence of the spliced cables and noting the need to determine what cables in the tray had been spliced, since the cables were not identified, and the splices were not recorded in licensee drawings or records.

IPEEE section 5.2.2.1.4, addresses the potential effects of spraying from pipe ruptures in this area of the PAB, and concludes that spraying effects in the pipe penetration area would not be risk significant because “it was noted that there were no cable splices within 10' of any piping.” If the equipment associated with the spliced cables is risk-significant, the presence of cable splices within 10' of the containment spray piping could be contrary to the licensee’s IPEEE described basis. Additionally, the licensee has not yet determined the extent of condition for the cable splices in the area, which raises the possibility of additional splices having been introduced into risk-significant electrical cables in the area which could be within 10' of the numerous pipes running through the area.

Based on CR-IP2-2002-02474, the licensee initiated work order (WO) IP2-03-17236 (scheduled for the week of December 15, 2003) to complete identification of the spliced cables, as well as whether they affect any risk significant cables and equipment. The licensee has not yet performed an extent of condition evaluation for the remainder of this area as described in the IPEEE, including the pipe penetration area. This will remain unresolved **(URI 50-247/03-11-01)** pending completion of the licensee’s evaluation of the identified cable splices and determination of whether there are risk significant cables spliced in the area and potential impact on IPEEE PAB internal flood analysis.

1R11 <u>Licensed Operator Requalification Program</u>

1. <u>Biennial Review by Regional Specialist</u>

a. <u>Inspection Scope</u> (71111.11B)

The following inspection activities were performed using NUREG-1021, Rev. 8, “Operator Licensing Examination Standards for Power Reactors,” Inspection Procedure Attachment 71111.11, “Licensed Operator Requalification Program,” and NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process (SDP),” as acceptance criteria.

The inspector observed the administration of annual operating tests given to one operating crew. The quality of the tests met or exceeded the criteria of the Examination Standards and 10 CFR 55.59. The inspector also observed simulator performance during the conduct of the examinations.

The inspector noted the following exam results for the one crew:

- The crew passed the simulator test.
- All crew members passed the simulator test.
- All individuals passed the walk-through test.

The inspector reviewed the schedule of testing for the remaining operators and crews including administration of the comprehensive written exam. Completion is expected by November 2003.

b. Findings

No findings of significance were identified.

2. Requalification Activities Review by Resident Staff

a. Inspection Scope (71111.11Q)

The inspector observed the performance of an Operating Team "2B" during licensed operator re-qualification training. Specifically, the inspector observed a simulator exam associated with lesson plan ESR-024-005. The inspection was conducted to assess the adequacy of the training, licensed operator performance, implementation of the emergency plan, and the adequacy of the licensee's critique.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness

a. Inspection Scope (71111.12Q)

The inspectors evaluated licensee's work practices and preventive maintenance activities for the component cooling water (CCW) pumps to assess the effectiveness of maintenance activities. The inspectors reviewed the performance history of the CCW pumps to assess the licensee's corrective actions and to evaluate licensee's monitoring, evaluations, and disposition of issues in accordance with station procedures and the requirements of 10 CFR 50.65. The inspectors evaluated system deficiencies over the last four quarters to verify that maintenance preventable functional failures were being properly identified and reviewed. The following documents were reviewed:

- CCW System Health Report - 2nd Quarter 2003
- CCW Performance Criteria / Goal Evaluation
- Condition report (CR)-IP2-2003-4098

On August 14, 2003, flexible coupling (SWN-66-3) on the inlet of the 21 emergency diesel generator (EDG) failed. The failure resulted in significant service water leakage into the EDG room. The licensee repaired the flexible coupling on August 15 under corrective maintenance work order (WO) IP2-03-23710. The inspector reviewed the work package and observed portions of the corrective maintenance to evaluate work practices. The inspectors reviewed the licensee's reliability evaluation of the failed flexible coupling and subsequent emergency diesel generator unavailability to affect repairs under the Maintenance Rule (10 CFR 50.65) program.

b. Findings

Introduction. A Green non-cited violation (NCV) was identified in that the maintenance procedure to install a flexible coupling connection on the safety related portion of the service water system was inadequate and contributed to the coupling's failure. This was determined to be a violation of 10 CFR 50, Appendix B, Criterion V "Instructions, Procedures and Drawings."

Description. Following the loss of offsite power event on August 14, 2003, all service water pumps, by design, de-energized and then restarted on the emergency diesel generators. When service water flow was restored, a system pressure transient occurred which resulted in the failure of the flexible coupling on the inlet of 21 EDG lube oil cooler. A significant leak in the system resulted that required compensatory measures (redirection of the spray flow) to control.

Licensee engineering analysis of the failed coupling showed that the coupling gasket was greater than four degrees off-center when the coupling clamp was tightened down. The maximum deviation from center according to the vendor manual is plus or minus four degrees. It was also found that the inner metal retaining ring ends were butted together instead of sliding across each other as designed. The above installation problems would prevent the coupling gasket from being properly compressed when torqued to the specified value.

The flexible coupling was replaced in November 2002 under work order #IP2-02-57692. The inspector's review of work order #IP2-02-57692 identified that the step list for the installation did not have any requirements to verify that the gasket was properly centered. The inspector determined that this would be a critical characteristic as documented in the vendor's document to ensure proper installation of the flexible coupling. On August 15, 2003, the 21 EDG was taken out of service and the flexible coupling was replaced.

Analysis. The inspectors determined that this was a performance deficiency since incomplete work instructions in November 2002, contributed to the failure of the service water flexible coupling. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy's procedures.

This finding is greater than minor since if left uncorrected, it could be a more significant safety concern. The failed coupling was on the service water inlet side of the EDG coolers and below the deck grating. The same type of flexible coupling is used on the outlet side of the EDG coolers and above the deck grating. Without verifying the proper installation of the coupling it is reasonable to believe this failure could occur on the outlet side of the coolers. The coupling is in a location in which the spray could wet important electrical components. The finding affects the Mitigating Systems Cornerstone attribute of availability since the 21 EDG was unavailable for repairs for 6.5 hours. The inspectors conducted a MC 0609, Phase 1 SDP screen and determined that the flexible coupling leakage was of a very low safety significance (Green) since it did not adversely impact service water cooling to the emergency diesel generator, did not impact equipment and functions associated with internal flooding in the diesel generator room, and did not result in a loss of service water or emergency power safety function that contributed to internal flooding initiated events. This finding is also related to the cross-cutting area of human performance in that maintenance planning failed to adequately identify and incorporate into the work package critical characteristics required for proper installation of a service water flexible coupling.

Enforcement. 10 CFR 50, Appendix B, Criterion V states, in part, that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. Contrary to this, in November 2002, the procedural steps for the installation of a flexible coupling were not adequate to verify that the component was properly installed. Specifically, the installation instructions did not verify that the gasket was properly centered and was within the manufacturer's specification. Because this violation is of very low safety significance and has been entered in the licensee's corrective actions program (IP2-CR-2003-5200) this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: **(NCV 50-247/03-11-02)**

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

a. Inspection Scope (71111.13)

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

- Work Order (WO) IP2-03-22771, Troubleshoot and Repair Intermittent Overpower Differential Temperature Rod Block Alarms
- WO IP2-03-17890, Pressurizer Level Recorder Replacement
- WO IP2-03-03145, Fire Header Leak

b. Findings

1. Pressurizer Level Recorder Replacement

Introduction. A Green NCV was identified in that inadequate design controls for a pressurizer level recorder replacement resulted in an inadvertent letdown isolation and securing of pressurizer heaters. This was determined to be a violation of 10 CFR 50, Appendix B, Criterion III "Design Control."

Description. On August 8, 2003, Instrumentation and Controls technicians were assigned to replace the central control room pressurizer level recorder in accordance with design change package DCP-200105716-I. The purpose of this design change was to replace the recorder instrumentation in the control room with a modern digital design. Work Order IP2-03-17890 step 60.3.1, required the technicians to place an electrical jumper wire between two terminals lugs to maintain the level current loop. When this step was performed letdown flow was immediately isolated and the pressurizer heaters were automatically secured. Modification installation work stopped and control room operators took actions in accordance with abnormal operating instruction (AOI) 3.1, "Chemical and Volume Control System Malfunction," to restore pressurizer level and pressure control. Pressurizer level rose six percent above the normal band, for approximately nine minutes and the pressurizer high level alarm actuated. The system was restored to normal in thirty-two minutes. During the transient reactor coolant system (RCS) pressure did not deviate more than ten psi from normal.

Work Order IP2-03-17890 was written assuming the installed pressurizer level recorder input was a current signal. An error by design engineering in the design change documentation concluded that the recorder input was a current signal. Licensee subsequent analysis showed the actual recorder input to be a voltage signal. By placing the jumper between the input terminals, the voltage input was shorted causing the indicator pressurizer level signal to go to zero. This resulted in an automatic letdown isolation and securing of pressurizer heaters.

Analysis. The inspectors determined that this was a performance deficiency since the design change documentation did not accurately reflect existing equipment conditions. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function and the finding was not the result of any willful violation of NRC requirements or Entergy's procedures. The finding was determined to be greater than minor based on example 4.b. of IMC 0612, Appendix E. The finding affected the Initiating Event cornerstone, since it contributed to the increased likelihood of a reactor trip. However, the inspectors conducted a MC 0609, Phase 1 SDP screen and determined that the finding is of very low safety significance (Green), because it did not impact the availability of mitigation equipment, increase the likelihood of a primary or secondary system LOCA, or increase the likelihood of an internal fire or flood. This finding is also related to the cross-cutting area of human performance since design engineering failed to perform a comprehensive review to determine the as-built condition via installation drawings. This error was translated by the work planner in WO IP2-03-17890 installation instructions.

Enforcement. 10 CFR 50 Appendix B, Criterion III states, in part, that design changes shall be subject to design control measures commensurate with those applied to the original design and that design control measures shall provide for verifying or checking the adequacy of design. Contrary to this, on August 8, 2003, design change package DCP-200105716-I did not accurately reflect actual plant conditions and resulted in an unintended plant transient which challenged plant operators during the removal of the installed pressurizer recorder instrument. The failure to maintain proper design control for the pressurizer recorder replacement is of very low safety significance and has been entered into the licensee's corrective actions program (CR-IP2-2003-05052). This violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 50-247/03-11-03)**

2. Fire Header Leak

Introduction. The inspectors identified a Green finding involving inadequate short-term corrective actions associated with multiple leaks on a six-inch fire header in the Unit 1 turbine building. On September 10, 2003, an 80 gallon per minute fire header leak occurred, for which operators isolated by depressurizing the entire fire water suppression system at Unit 2 for approximately three hours.

Description. On September 10, 2003, a leak developed on a six-inch fire header in the non-safety related Unit 1 turbine building at a rate of approximately 80 gallons per minute (gpm). The operators secured the fire water suppression system for both Units 1 and 2 for approximately 3 hours to isolate the six-inch fire header leak and affect temporary repairs.

The fire header leakage spread into portions of the cable spreading room and the 480 volt switchgear room. No safety-related equipment was impacted by the fire header leakage into the room based primarily on the floor design of the Unit 1 turbine building/cable spreading room and operator compensatory measures to minimize water spray.

Inspector follow-up determined that the licensee had a previous self-revealing leak on the same six-inch fire water header on July 16, 2003, as documented in CR IP2-2003-4493. The licensee's evaluation of the leak found an additional leak site. Both through wall minor leaks were evaluated via ultrasonic piping examinations in the localized area and circumferentially around the six-inch header. The licensee concluded that structural integrity of the fire header was maintained at the leak site. On August 18, 2003, CR IP2-2003-4493 was closed to work order IP2-03-3145. The work order was scheduled for permanent repairs during the week of October 27, 2003.

The leak sites were not evaluated by the licensee for potential plant risk from internal flooding or fire suppression system impact to provide a basis for short-term compensatory actions or actions to minimize potential fire header spray to mitigating equipment. On September, 10, 2003, another through-wall fire header leak occurred approximately one linear foot away from the July 16 leak site.

The inspector verified that the licensee properly adhered to administrative requirements during the shutdown of the fire water suppression system and during isolation of the fire header leak. The fire header was repaired with a temporary clamp and the system restored to service. A fourth leak was identified by the licensee during routine Unit 1 supervisory tours on September 29, 2003. This fourth leak was evaluated using ASME Code Case N-513 and Entergy found that the pipe was structurally adequate for projected loads, based upon the ultrasonic pipe thickness measurements. At the end of the inspection period, the licensee implemented a periodic monitoring program of the leak sites on this six-inch fire header.

Analysis. The failure to take appropriate corrective action for identified degraded fire water header piping is considered more than minor by reference to example 4.f in MC 0612, Appendix E. This performance finding involves the Mitigating Systems Cornerstone objective of fire suppression system availability to respond to fires. The finding did not impact equipment necessary to mitigate internal flooding, seismic, or severe weather initiated events. The inspector verified through plant walkdowns, discussions with cognizant personnel, and review of IPEEE section 5.2.2.2 that flood measures between the Unit 1 turbine building and Unit 2 cable spreading room were appropriate to ensure no impact on mitigating equipment. Phase 1 of the SDP screened the finding to NRC IMC 0609 Appendix F, since the fire header leak and subsequent operator actions resulted in a degradation in the fire protection defense-in-depth (fire suppression). The finding is very low risk significance (Green) based upon evaluation results from the fire protection risk significance screening methodology (FPRSSM). The finding impacted both manual and automatic suppression capability for 2 hours and 57 minutes. However, since the three-hour fire barrier separation and one-hour fire barriers for safe shutdown were not adversely impacted and there were no instances of intervening combustibles in the effected fire areas, this finding is of very low safety significance. **(FIN 50-247/03-11-04)**

Enforcement.

No violation of regulatory requirements occurred.

1R14 Personnel Performance During Non-Routine Plant Evolutions and Events

1. Reactor Trips Due to Electrical Grid Disturbances

a. Inspection Scope (71111.14)

On August 3 and August 14, 2003, the plant experienced reactor/turbine trips due to electrical grid disturbances. A special inspection team from the regional office was tasked to evaluate the electrical system design and reliability, operator performance during the transients, and the equipment issues that were identified during these transients. The inspectors did evaluate one emergent work item after the August 14 trip which is documented in Section 1R12 of this report. Between August 14 and August 17 the inspectors provided 24 hours a day site coverage with extended control room observations.

b. Findings

NRC review of licensee performance during the August 3 and August 14, 2003, Unit 2 reactor trips is documented in special inspection report 50-247/2003-013 and 50-286/2003-010.

2. Seal Injection Filter Leak

a. Inspection Scope (71111.14)

On September 20, 2003, leakage from a vent valve in the chemical volume and control system (CVCS) occurred during the restoration of 22 seal injection filter following maintenance. The inspectors reviewed licensee response to verify that appropriate procedures and Technical Specifications had been entered and that the event had been properly classified. The inspectors reviewed the Emergency Action Levels (EAL) to verify that no classification was required and that the licensee's response was appropriate in accordance with EAL 3.1.1. The inspectors also reviewed the work order, tagout, and restoration procedure for the associated maintenance activity.

b. Findings

Introduction. A Green non-cited violation of 10 CFR 50, Appendix B, Criterion V was identified involving an inadequate procedure for restoring the 22 seal injection filter to service following maintenance. The consequence was an approximate 70 gallon per minute (gpm) chemical volume and control system leak through an open vent valve. The leak lasted for approximately two minutes until operators shut the vent valve.

<u>Description.</u> On September 5, 2003, the 22 seal injection filter was removed from service to repair leakage from the filter cover in accordance with work order #IP2-02-02040. The filter was tagged-out in accordance with tagout No. 2-CVCS-22CVCSF and drained using system operating procedure (SOP) 3.1, “Charging, Sealwater, and Letdown Control,” prior to commencing the work. The vent valve for the filter was left in the open position in accordance with SOP 3.1. When the work was completed, operators were dispatched to return the system to service. The restoration guidance in the tagout required the operators to remove the tagout and then use the SOP 3.1 to vent the system and return it to service. Once the inlet valve to the filter was cracked open, water was discharged through the open vent valve at a rate estimated at 70 gpm. This caused a loss in charging and seal injection flow, thus causing system leakage into the charging pump cell at approximately 70 gpm. Operators quickly (within two minutes) shut the filter inlet valve and restored charging and seal injection flow to normal. During the event, control room operators entered abnormal operating procedure (AOI) 3.1, “Chemical and Volume Control System Malfunction,” as required, and subsequently entered the appropriate Technical Specification.

<u>Analysis.</u> The inadequate restoration procedure resulted in a significant chemical and volume control system leak (the capacity of one coolant charging pump) that degraded normal charging flow and emergency boration for a short period of time. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC’s regulatory function and the finding was not the result of any willful violation of NRC requirements or Entergy’s procedures.

This finding is more than minor since it adversely impacted the Mitigating System Cornerstone objective of safety system capability and availability with respect to the attributes of configuration control and procedural quality. Using MC0609, Phase 1 SDP screening, the finding is of very low safety significance (Green) because, in spite of being degraded, this CVCS system condition did not result in a loss of emergency boration safety function, did not result in a loss of safety function of a single train greater than the Technical Specification allowed outage time, did not degrade other mitigation equipment within the charging cell, and did not contribute to an internal flooding core damage sequence. The initiation of emergency boration is a manual safety function and since operators quickly recognized and corrected the equipment configuration error, it did not result in a loss of this safety function. This finding is also related to the cross-cutting area of human performance since operations planning personnel did not identify that the restoration procedure was inadequate as written.

<u>Enforcement.</u> 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” states, in part, that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and shall be accomplished in accordance with those procedures. Contrary to this, on September 5, 2003, SOP 3.1 and equipment tagout No. 2-CVCS-22CVCSF, used for the restoration of the 22 seal injection filter following maintenance, were inadequate to maintain proper configuration control of the chemical and volume control system. Because this violation is of very low safety significance and has been entered in the licensee’s corrective actions program

(CR-IP2-2003-5572), this violation is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy **(NCV 50-247/03-11-05)**.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspectors reviewed the below listed condition reports and associated operability evaluations to ensure that operability was properly justified and that the component or system remained available, without a significant degradation in performance or unrecognized operability issue. The inspectors used Technical Specifications, Updated Final Safety Analysis Report, and design basis documents, as appropriate. The following operability evaluations associated with condition reports were reviewed:

- CR-IP2-2003-04570 - Containment Spray Nozzle Surveillance Failure
- CR-IP2-2003-03394 - Isolation Valve Seal Water Solenoid Valve Failure
- CR-IP2-2003-05219 -Auxiliary Feedwater Suction Piping Water Hammer
- CR-IP2-2003-03035 - 23 Emergency Diesel Generator Pre-Lube Oil Pump
- CR-IP2-2003-04288 - MOV-536 Limit Switch Failure

b. Findings

No findings of significance were identified.

1R17 Permanent Modifications

a. Inspection Scope (71111.17A)

The inspector reviewed plant modification FMX-00-52429-D, “Component Evaluation of Replacement Steam Generators.” This modification involved replacement of the original steam generators with the new steam generators which was completed during the fourteenth refueling outage of Unit 2. The inspection focused on a specific design aspect in regard to the ASME Code reconciliation of the replacement steam generators. The ASME Code of record for the original steam generators was the 1965 Edition, Section III, including all addenda through Summer 1966. The replacement steam generators were fabricated in accordance with the 1980 Edition of the ASME Code, Section III, including all addenda through Winter 1981. The inspector, therefore, verified that the materials used in the fabrication of the replacement steam generators either met or exceeded the material strength properties used in the Stress Report. The inspector verified that implementation of the later Code in construction and use of new materials for tubes (thermally treated Alloy 600) and support plates (SA-240, Type 405) based on industry experience of degradation, exceeded the requirements of the original construction Code.

The licensee subsequently performed a preservice inspection of the replacement steam generators in accordance with the 1989 Edition, ASME Code, Section XI, and pursuant to 10 CFR 50.55a (g).

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope (71111.19)

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

The selected testing activities involved components that were risk significant as identified in the IP2 Individual Plant Examination. The regulatory references for the inspection included Technical Specification 6.8.1.a. and 10 CFR 50, Appendix B, Criteria XIV, "Inspection, Test, and Operating Status." The following testing activities were evaluated:

- PT-M38A for Gas Turbine 1
- WO IP2-03-24684, PWT for Technical Support Center (TSC) Temporary Diesel
  WO IP2-03-14062, PT-Q13 to Stroke Valve 21 Containment Fan Cooler Unit Damper (FCV-21)(CR-IP2-2003-05638)
- WO IP2-03-25737, PWT for Control Rod Urgent Failure
- WO IP2-03-25696, PWT for Gas Turbine 3
- WO IP2-02-46483, PWT for 22 Atmospheric Dump Valve

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

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

- PT-Q89, Control Rod Testing
- PT-Q27A, 21 Auxiliary Boiler Feed Pump
- SOP 1.7, Reactor Coolant System Leak Rate Calculation
- PT-Q29, 21 Safety Injection Pump
- PT-M48, 480V Under Voltage Relay Testing

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23)

Temporary Alteration No: TA-02-2-079, Enclosure of the Lower Leak-off Connection Pipe Cap on valve PCV-455B

The inspector reviewed documentation on Temporary Alteration No: TA-02-2-079 “Enclosure of the Lower Leak-off Connection Pipe Cap on valve PCV-455B.” PCV-455B is one of the two pressurizer spray valves. This valve has two leak-off connections. The upper one collects leakage past the first set of packing rings. The lower leak-off connection is capped and it gets pressurized to reactor coolant system (RCS) pressure in the event the stem bellows developing a leak. The weld between the lower leak-off pipe and the cap developed a leak. The modification involved installing a welded enclosure over the existing piping cap and the leaking weld. The enclosure consists of a cap and a ring fabricated from ASME SA-479 Gr 316L rolled bar stock. The ring was in two halves and was welded to the outside diameter of the leak-off line. The new cap was installed over the existing cap and was welded to the ring.

The inspectors reviewed the temporary design, the engineering design verification, and the 10 CFR 50.59 screening against the applicable design basis documents.

Temporary Alteration No: TA-03-2-190, Provide Temporary Diesel for the Technical Support Center

The inspector reviewed documentation on Temporary Alteration No: TA-03-2-190, “Provide Temporary Diesel for Technical Support Center (TSC).” This modification was installed to provide a reliable source of back-up power to the TSC in the event of a loss of offsite power. The inspectors reviewed the temporary design, the engineering design verification and the 10 CFR 50.59 screening against the applicable design basis documents. The inspectors observed portions of the installation, reviewed the installation documentation (WO IP2-03-24599), ensured the modification was properly tagged and walked down the completed modification to verify the as-built configuration conformed to the design documentation and was in accordance with administrative procedure ENN-DC-136, “Temporary Alteration Control.” The inspectors reviewed the TSC electrical loading to verify the diesel had sufficient capacity to perform its intended function and performed an assessment of the associated breaker and cabling to verify it was adequate for the generators rated capacity.

b. Findings

No findings of significance were identified.

EMERGENCY PREPAREDNESS

1EP2 Alert and Notification System Testing

a. Inspection Scope (71114.02)

Entergy completed an upgrade to the siren activation and verification system in January, 2003. The inspector reviewed Emergency Plan (E-Plan) commitments concerning the Alert Notification System (ANS), reviewed procedure NEM-5.702, Revision 0, “Testing of the Indian Point Siren System,” and siren testing documentation to verify compliance with testing commitments. The inspector interviewed the emergency preparedness (EP) Manager, the Entergy staff responsible for ANS testing, and the contracted system design engineer concerning system design, operation, testing, and maintenance. The contract engineer provided a system demonstration at the control console located in the emergency operations facility, including a partial siren silent test and siren system monitoring capability. The inspector also interviewed the licensee staff responsible for implementation of the tone alert radio program.

The inspector conducted the review in accordance with guidance provided in NRC Inspection Procedure 71114, Attachment 02, “Alert and Notification System Testing.” The applicable planning standard, 10 CFR 50.47(b)(5), and related requirements in 10 CFR 50 Appendix E, Section IV.D. were used as acceptance and reference criteria.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing

a. Inspection Scope (71114.03)

The inspector reviewed the licensee's E-Plan commitments for Emergency Response Organization (ERO) staffing and emergency facility activation. He also reviewed recent quarterly call-in test results and the last two off-hours mobilization drill reports, October 15, 2002 (Unit 3) and June 4, 2001 (Unit 2), to assess Entergy's ability to augment the shift staff with sufficient responders in a timely manner. The inspector reviewed staff depth for key ERO positions to ensure that sufficient numbers of responders were available. The inspector interviewed the EP Manager and the EP staff responsible for ERO augmentation testing to assess testing effectiveness and results. Lastly, the inspector reviewed the licensee's staffing commitments to ensure compliance with NUREG-0654, Table B-1, "Minimum Staffing Requirements for NRC Licensees For Nuclear Power Plant Emergencies."

The inspector conducted this review in accordance with the guidance in NRC Inspection Procedure 71114, Attachment 03, "Emergency Response Organization Augmentation." The applicable planning standard, 10 CFR 50.47(b)(2), related requirements in 10 CFR 50, Appendix E, and the licensee's E-Plan commitments were used as acceptance and reference criteria.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope (71114.04)

The inspector reviewed E-Plan revision 03-01, dated March 6, 2003, and sampled E-Plan implementing procedure changes to verify that the changes had not reduced the effectiveness of the E-Plan. There were no recent Emergency Action Level (EAL) changes to review.

The inspector conducted the review in accordance with NRC Inspection Procedure 71114, Attachment 04. The applicable requirements in 10 CFR 50.54(q), 10 CFR 50.47(b), and 10 CFR 50, Appendix E were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope (71114.05)

The inspector reviewed Audit Report A03-06-I of the Indian Point Energy Center (IPEC) Emergency Planning Program, conducted May 19 through June 12, 2003, by the quality assurance (QA) department to meet the requirements of 10 CFR 50.54(t). The inspector reviewed the audit report to verify that it met NRC requirements, determine QA-identified deficiencies, and if any repeat issues were identified. He also interviewed the audit team leader to discuss finding details. The inspector reviewed a sampling of Condition Reports (CRs) documenting problems and associated corrective actions to assess the licensee's ability to identify and resolve EP issues, and determine if corrective actions were effective to prevent recurrence. Lastly, the inspector reviewed the 2003 focused self-assessment report titled "EP Department Program Performance" for further insights on licensee problem identification and resolution.

The inspector conducted these reviews in accordance with Inspection Procedure 71114, Attachment 05. The applicable planning standard, 10 CFR 50.47(b)(14), and the requirements in 10 CFR Appendix E, Section IV.F.2.g, were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP6 Emergency Plan Drill Evaluation

a. Inspection Scope (71114.06)

On September 24, 2003, the inspectors observed the emergency response organization (ERO) during an announced emergency preparedness training drill at Indian Point Unit 3. The simulated emergency included the activation of the Operations Support Center (OSC), the Technical Support Center (TSC), Emergency Operations Facility (EOF), and the Joint News Center (JNC) after an Alert (simulated) was declared by the control room operators.

The inspectors observed the conduct of the exercise in the control room simulator, TSC, and EOF. The inspectors assessed licensed operators and the ERO staff's adherence to emergency plan implementation procedures and their response to simulated degraded plant conditions. The inspectors verified licensee performance in classification, notification, and protective action recommendations. In addition to the drill, the inspectors observed the licensee's controller critique and evaluated the licensee's self-identification of weaknesses and deficiencies. Condition report CR-IP3-2003-05248 documented that six of eight performance indicator opportunities (classifications, notifications, and protective action recommendations) were successful. The inspectors compared the licensee's identified findings against their observations. The inspectors' review included the following documents and procedures:

- Indian Point Energy Center Emergency Plan
- IP-EP-410, Protective Action Recommendations, Revision 2
- IP-EP-250, Emergency Operations Facility, Revision 0
- Emergency Action Levels
- Condition Report Nos. IP3-2003-05952, IP3-2003-05248, IP3-2003-05249, IP3-2003-05251, IP3-2003-05250, IP3-2003-05279, IP3-2003-05278, and IP3-2003-05255.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope (71121.01)

The inspector reviewed radiological work activities, practices, and procedural implementation during tours and observations of the facilities and inspected procedures, records, and other program documents to evaluate the effectiveness of Indian Point 2 access controls to radiologically significant areas.

On July 9, 2003, the inspector toured and observed work activities in selected portions of the fuel handling building and of the chemical systems building in Unit 1, including the area in the sphere annulus area where the pipe from the north curtain drain was located. The inspector reviewed the work activities on the Unit 1 fuel handling floor involved in characterizing the contents of several spent resin storage vessels. On this date, the inspector discussed with the project leaders the plans for the Unit 1 re-mediation projects involving the tanks and vessels, the north curtain drain water processing, and the spent fuel pool. On July 10, the inspector reviewed the work activities on the Unit 1 fuel handling floor involved in characterizing the contents of several spent fuel pool demineralizer vessels. Also on this date, the inspector toured and observed work activities in the primary auxiliary and fuel storage buildings in Unit 2. The inspector also reviewed the conduct of a pre-job briefing for an entry into the Unit 2 containment at full power. The pre-job briefing covered radiation safety, confined space, and heat stress considerations associated with the planned work evolutions. During the walkdowns, the inspector observed and verified the appropriateness of the posting, labeling, and barricading of radioactive material, radiation, contamination, high radiation, and locked high radiation areas. The inspector reviewed work activities by both radiation workers and radiation protection technicians for compliance with the radiation work permit (RWP) requirements and radiological protection procedures.

At the access control point to the radiologically controlled area (RCA), the inspector observed radiation workers logging into the RCA on RWPs, using electronic dosimeters, and observed radiation workers exiting the RCA and then logging out of their RWPs. The inspector examined the use of personnel dosimetry and the radiological briefings for ingoing radiation workers.

The inspector performed a selective examination of procedures, records, and program documents (see the attachment to this report) to evaluate the adequacy of radiological controls.

The review was against criteria contained in 10 CFR 19.12, 10 CFR 20 (Subparts D, F, G, H, I, and J), Technical Specifications, and site procedures.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope (71121.02)

The inspector reviewed the effectiveness of Entergy's program to maintain occupational radiation exposure as low as is reasonably achievable (ALARA).

The inspector discussed the actual cumulative year-to-date dose results for 2003 for Units 1 and 2 with IPEC radiation protection personnel. These results were tracking at or below the projected values.

On July 9, the inspector discussed the status of the radiation exposure reduction plan, including ALARA program changes, with the Technical Support Manager and the Assistant Radiation Protection Manager.

On July 7, the inspector met with a respiratory protection health physicist and discussed the procedures and processes in place to implement the requirements of 10 CFR 20.1703(f) for standby rescue personnel whenever one-piece atmosphere-supplying suits, or any combination of a supplied-air respiratory protection device and personnel protective equipment, are used from which an unaided individual would have difficulty extricating himself or herself. The health physicist had recently completed a self-assessment report which addressed this issue. This assessment resulted in the identification of procedural improvements which were being tracked by CR-IP3-2003-04012 in the corrective action program.

The inspector performed a selective examination of procedures and program documents (see the attachment to this report) for regulatory compliance and for adequacy of control of radiation exposure.

The review was against criteria contained in 10 CFR 20.1101 (Radiation protection programs), 10 CFR 20.1701 (use of process or other engineering controls), and site procedures.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment

a. Inspection Scope (71121.03)

The inspector reviewed the program for health physics instrumentation and for installed radiation monitoring instrumentation to determine the accuracy and operability of the instrumentation.

During the plant tours described in Section 2OS1 of this report, the inspector reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity and radiation levels, including portable field survey instruments, hand-held contamination frisking instruments, continuous air monitors, and radiation monitors (including whole body friskers, portal monitors, and area monitors). The inspector selectively verified current calibration, source checking, and proper instrument function. The inspector also identified and noted the condition and operability of selected installed area and process radiation monitors and any accessible local indication information for those monitors.

On July 9, the inspector met with the supervisor of the site thermoluminescent dosimetry (TLD) to discuss the National Voluntary Laboratory Accreditation Program (NVLAP) On-Site Assessment Report for Ionizing Radiation Dosimetry conducted between June 25-27, 2003. The inspector also performed a selective examination of procedures and program documents (see the attachment to this report) for regulatory compliance and adequacy.

The review was against criteria contained in 10 CFR 20.1501, 10 CFR 20 Subpart H, Technical Specifications, and site procedures.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope (71122.02)

The inspector reviewed the radioactive material processing and transportation work activities and practices during tours of the facilities, discussed observations and issues with Entergy representatives, and inspected procedures, procedural implementation, records, and other program documents to evaluate the effectiveness of performance in this area.

Radioactive Waste System Walkdown

The inspector walked down selected accessible portions of the station's radioactive liquid and radioactive solid waste collection, processing, and storage systems/locations to verify that the current system configuration and operation agreed with descriptions contained within the Updated Final Safety Analysis Report (UFSAR) and the Process Control Program (PCP). The areas reviewed during the walk downs included buildings/areas within the radiologically controlled and protected areas (including: in Unit 1, the containment, fuel handling, and chemical systems buildings, and, in Unit 2, the primary auxiliary, fuel storage, and maintenance and outage buildings; and the radioactive material storage area outside the Unit 2 equipment hatch) and areas outside the protected area including the old steam generator storage building, and the Yard 8 radioactive material storage area.

During system walk downs on August 19 and 21, and during discussions with radioactive waste processing and shipping personnel, the inspector reviewed the status of non-operational and/or abandoned-in-place radioactive waste process equipment and administrative and physical controls for the systems; the inspector also reviewed the adequacy of any changes to the radioactive waste processing systems since the last inspection in this area and the potential radiological impact and reviewed the current processes for transferring radioactive waste resin and filter cartridges into shipping/disposal containers and for resin de-watering. The inspector reviewed the recent process improvements made for the transfer of spent primary resin from the spent resin storage tank to a transport container.

Waste Characterization and Classification

The inspection included a review of conformance with applicable waste characterization and classification regulations and program procedures. This included a selective review of the radiochemical sample analysis results for each of the tracked radioactive waste streams and the development of scaling factors for difficult to detect and measure radio-nuclides. The inspector also verified that programmatic elements were in place to ensure that determination of waste classification (10CFR61.55) and waste characteristics (10CFR61.56) was adequate and that the waste stream composition data accounts for changing operational parameters.

Shipment Preparation

Based on the scheduled radioactive waste processing and shipment activities, the inspector had limited opportunity to observe shipment preparation from initial packaging through final readiness for shipment. However, on August 19, 2003, the inspector did observe a pre-job brief for the transfer of a high integrity container (HIC) containing spent liquid waste processing resin. Afterwards, the inspector viewed the transfer of this HIC from an onsite storage container to a shielded transport cask on the 70-foot elevation of the Unit 1 fuel handling building. Based on this observation, the review of shipment records, radioactive waste program documents, and shipment preparation procedures, and on discussions with radioactive waste processing and shipping personnel, the inspector was able to assess the adequacy of shipment preparation activities from initial packaging to shipment readiness and to determine that shipping personnel were knowledgeable of NRC and Department of Transportation (DOT) shipping regulations.

Shipping Records

The inspector examined the shipping records for five non-excepted packages including two low specific activity (LSA) type shipments, two surface contaminated object (SCO) type shipments, and one Type B quantity shipment. The inspector reviewed these records for compliance with NRC and DOT requirements, including shipment papers and description requirements, shipper's certification, proper use of forms, package marking and labeling, vehicle placarding, emergency response information, and packaging requirements.

Identification and Resolution of Problems

The inspection included a selective review of audits and self-assessments related to the radioactive waste processing and transportation programs performed since the last inspection in this area. The inspector also reviewed selected condition reports and their corrective actions for issues related to the inspected area. Specifics regarding the corrective action program are addressed in Section 4OA2 of this report.

During the review of the five areas, which are listed above under inspection scope, the inspector performed a selective examination of procedures, records, and documents (see attachment to this report) for regulatory compliance and adequacy.

The above review was against criteria contained in: 10 Code of Federal Regulations (CFR) Part 20: Subpart F (Surveys and monitoring); 10 CFR 20.1902 (Posting requirements); Subpart I (Storage and control of licensed material); Subpart K (Waste disposal); Appendix G to Part 20 (Requirements for transfers of low-level radioactive waste intended for disposal at licensed land disposal facilities and manifests); 10 CFR 61.55, Waste classification; 10 CFR 61.56, Waste characteristics; 10 CFR 61.57, Labeling; 10 CFR 71, Packaging and transportation of radioactive material; 49 CFR Part 172 (Hazardous materials table, special provisions, hazardous-materials communications, emergency response information, and training requirements); 49 Part 173 (Shipper's general requirements for shipments and packaging); 49 CFR Subpart I (Class 7 (radioactive materials); 49 CFR Part 177 (Carriage by public highway); NRC Bulletin 79-19; and site procedures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

a. Inspection Scope (71151)

The inspector reviewed the licensee's performance indicator (PI) data collecting and reporting process as described in procedure ENN-LI-107, "NRC Performance Indicator Process." The purpose of the review was to determine whether the methods for reporting PI data were consistent with the guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guidelines," Revisions 1 and 2. The inspection included a review of the indicator definitions, data reporting elements, calculation methods, definition of terms, and clarifying notes for the performance indicators. Plant records and data were sampled and compared to the reported data. The inspector reviewed the licensee's actions to address and satisfactorily resolve discrepancies in the performance indicator data.

1. Safety System Functional Failures

The inspectors reviewed results for 2nd quarter 2003 for the Mitigating Systems Cornerstone, Safety System Functional Failure performance indicator for IP2 to verify individual PI accuracy and completeness. This inspection also examined data and plant records from the third quarter 2002, including a review of PI Data Summary Reports, Licensee Event Reports, operator narrative logs, and maintenance rule records.

b. Findings

No findings of significance were identified.

2. Safety System Unavailability - High Pressure Safety Injection

a. Inspection Scope

The inspectors reviewed results for 2nd quarter 2003 for the Mitigating Systems Cornerstone, Safety System Unavailability - High Pressure Safety Injection indicator for IP2 to verify individual PI accuracy and completeness. The inspector compared the PI data reported by the licensee to information gathered from the control room logs, condition reports, and work orders for the 3rd, 4th quarters of 2002 and the 1st and 2nd quarters of 2003. The inspectors compared the PI data against the guidance in NEI 99-02.

b. Findings

No findings of significance were identified.

3. Drill and Exercise Performance/Emergency Response Organization Drill Participation/Alert and Notification System Reliability

a. Inspection Scope

The inspector reviewed the licensee's process for identifying the data utilized for the three emergency preparedness PIs, which are: 1) Drill and Exercise Performance (DEP); 2) Emergency Response Organization Drill Participation (ERO); and, 3) Alert and Notification System Reliability (ANS). The inspector also reviewed PI data from the fourth quarter of 2002 through the second quarter of 2003 using the criteria of NEI 99-02. The inspector verified that the raw quarterly data was consistent with the data reported to the NRC.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope (71152)

1. Baseline Procedure Problem Identification and Resolution Review

The inspection included a review of the following issues identified in the corrective action program for the appropriateness and adequacy of event categorization, immediate corrective action, corrective action to prevent recurrence, and timeliness of corrective action: Condition Report Nos. IP2-1999-07062, -07424, 2002-02474, 2003-04098, -5200, -05052, -04493, -05572, -04570, -03394, -05219, -03035, -04288, -05248, -05255, 05952, -05248, -05249, -05251, -05250, -05279, -05278, and -04012. Additional condition reports reviewed as part of baseline inspection procedures are identified in the attachment to this report.

b. Findings

No findings of significance were identified.

2. Annual Sample Review - Emergency Planning Siren Deficiencies

a. Inspection Scope (71152)

The licensee documented approximately 30 CRs related to emergency siren issues between January and July, 2003. Of these, seven issues resulted in reports to the NRC in accordance with 10 CFR 50.72. The inspector selected 10 CRs for detailed review, including all seven which were reported to the NRC, and one which was a roll-up analysis of all siren CRs between January and April. The inspector reviewed the reports to ensure that the full extent of the issues were identified, an appropriate evaluation was completed, and appropriate corrective actions were prioritized and performed. The inspector discussed the reports with the EP Manager, the department corrective action coordinator, and Entergy staff responsible for siren maintenance and testing.

b. Findings

There were no findings identified associated with the sample selected. The inspector noted that Entergy's criterion implementing 10 CFR 50.72 reporting requirements was the failure of ten percent of the sirens in any one of the four counties in the emergency planning zone. The number of sirens by county are: Westchester-77, Rockland-51, Orange-16, Putnam-10. For example, the February 7, 2003 loss of two Orange County sirens (greater than 10% of the 16 county sirens) due to a power outage, resulted in a report to the NRC. Entergy subsequently changed the reporting criterion to be 25% of all 154 sirens, which would have negated five of the seven reports made to the NRC. The inspector determined this reporting change was acceptable and consistent with industry standard practices.

Entergy completed an upgrade to the siren activation and verification system in January 2003. This is a computer-based state-of-the-art system designed by a vendor for the licensee which provides continuous system monitoring and additional parametric information not available with the old system. The inspector determined that Entergy was effective in identifying siren system problems, maintained an appropriately low threshold for documentation in the corrective action program, and issues were prioritized and corrected in a timely manner. Additionally, the vendor performed a detailed analysis of siren system availability, including the server (computer system as a whole), the repeaters (transmitters which send radio signals to the sirens), and the sirens themselves. This analysis concluded at least 98 percent availability since the new siren control system was employed.

3. Annual Sample Review - Radioactive Waste Problems

a. Inspection Scope (71152)

The inspector selected four issues identified in the Corrective Action Program (CAP) for detailed review (CR Nos. CR-IP2-2002-01881, 2002-04408, 2003-02724, and 2003-03750). The issues were associated with errors in procedures, a lack of procedural guidance and training, a missing fence lock, and a lapse in training, respectively. The reports were reviewed to ensure that the full extent of the issues was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized.

b. Findings

No findings of significance were identified.

4. Annual Sample Review - Service Water Expansion Joint Leakage

a. Inspection Scope (71152)

On August 14, 2003, a system pressure transient occurred which adversely impacted the emergency diesel generator (EDG) portion of the service water system. The water hammer resulted in the failure of a gasket in a flexible connection on the discharge of 21 EDG (see report detail 1R12) and damaged a service water pressure gage on the 23 EDG. This event was documented in CR-IP2-2003-05200. As part of the corrective actions and extent of condition review, a walk-down of the service water system was performed. It was reported that no additional leaks or degraded components were found during this review and this was used to support the licensee's operability determination. The inspectors reviewed the CR documentation to evaluate the appropriateness and adequacy of the problem identification and short-term corrective actions, and to ensure it conformed to the licensees procedural guidance in ENN-LI-105, "Corrective Action Program."

b. Findings

No findings of significance were identified.

On August 20, 2003, the inspector performed a walk down of the EDG building and noted a small leak from the 10-inch combined service water discharge header in a location where a ½-inch nipple was attached to the piping. Upon further investigation it was found that the ½-inch nipple was supposed to have additional piping and an isolation valve (SWN-76-1) attached, but this section of piping had fallen off due to corrosion. The nipple is open to the ten-inch pipe but leakage through the hole was minimal due to a build-up of corrosion products in the line and a partial vacuum at that point in the system. The licensee performed an operability determination to ensure the EDG's would still perform their safety function. Since the degraded condition was downstream of the EDG heat exchangers it was determined that there would be no adverse impact to system cooling. During the investigation it was also found that a work order was issued in 1995 to perform a modification that would have replaced the ½-inch drain line with a three- inch line, due to degradation of the original line. The work order was signed-off and drawings revised to reflect the new design, but the modification was not actually performed. The proposed actions under this work order were to replace the ½-inch nipple and weld back on the existing valve. Documentation showed that the last time the missing valve was verified in its proper position was May 2000.

The inspectors determined that this was a performance deficiency associated with less than adequate extent of condition review associated with CR-IP2-2003-05200. However, the inspectors evaluated this finding as minor based on the guidance provided in IMC 0612, Appendix B. The leak and missing valve did not impact equipment operability due to the location downstream of the individual EDG coolers and system pressure at this location. The drainage system in the EDG building is sufficient to remove the water even if the ten-inch line was completely separated. The licensee wrote CR-IP2-2003-05294 based on the inspector's observation. An apparent cause evaluation and system walkdown were performed as part of the corrective actions. During the system walkdown three other minor leaks were identified and appropriately addressed.

5. Annual Sample Review - Auxiliary Component Cooling Water Pump

a. Inspection Scope (71152)

On June 11, 2003, the 21 auxiliary component cooling water pump (ACCWP) failed to meet the minimum acceptance criteria for developed head (differential pressure) specified in the American Society of Mechanical Engineers (ASME) Code, Section XI quarterly surveillance test (3PT-Q31A). The actual pump head was measured at 92.4 ft. and was below the operability range of 93.9 to 109.23 ft. The licensee subsequently declared the pump inoperable and initiated CR-IP2-2003-03688 to investigate the pump's condition. The inspector reviewed the CR and discussed identified corrective actions with the cognizant in-service test (IST) engineer to evaluate the actions taken, which included an apparent cause evaluation and additional pump testing after replacing the installed suction and discharge pressure gauges. The licensee also performed a calibration check on the first set of gauges. The second test with different instruments was performed under the provisions of the Section XI, OM-6, section 6.1, and the pump passed the test acceptance criteria with its developed head at 94.7 ft. The licensee subsequently declared the pump operable. In addition, the licensee noted that the required action range on both the 21 and 22 ACCWPs (0.93 to 1.10 of baseline head) had been established within the IST program under the OM-6 criteria for vertical line shaft pumps. However, after consultation with Westinghouse, the licensee concluded that the ACCWPs should have been characterized as standard centrifugal pumps, which have a larger margin in OM-6 for developed head (0.90 to 1.10 of baseline).

The inspector also reviewed previous test data for the 21 ACCWP with the IST engineer and noted that the differential pressure had fluctuated between 94 and 97 ft. since January 2002. The licensee considered that pump performance had been stable without a degrading trend since 1983, but was considering early pump replacement since its differential pressure was consistently low in the acceptance band.

b. Findings

No findings of significance were identified

4OA3 Event Follow-up (71153)

1. (Closed) URI 50-247/01-09-01: “Design Problem with Source Range Nuclear Instruments counted as a Safety System Functional Failure.”

   On September 5, 2000, the licensee submitted LER 2000-006 as the plant being outside the design basis since maximum postulated control room design temperatures were not considered within the design of the source range detector high flux reactor protection system trip. The licensee submitted a frequently asked question concerning the applicability of this condition as it relates to the NRC PI on Safety System Functional Failure. On December 12, 2002, the final decision was that since the licensee reported the condition under 10 CFR 50.73(a)(2)(ii), it should not be counted as a Safety System Functional Failure unless reported under 10 CFR 50.73(a)(2)(v). Notwithstanding, if this was considered reportable under 10 CFR 50.73(a)(2)(v) it would not have crossed the threshold at the time and would be considered minor. This item is closed.

2. (Closed) Licensee Event Report (LER) 2003-003-00: “Automatic Reactor Trip Initiated by a Main Turbine Trip on Auto Stop Oil.”

   LER 2003-003-00 was previously reviewed in report 50-247/2003-007, report detail 1R14. No findings were identified and this LER is closed.

3. (Closed) Licensee Event Report (LER) 2003-002-00: “Plant in a Condition Prohibited by Technical Specification due to Unavailability of Boric Acid Storage For More Than 48 Hours.”

   On March 21, 2003, the licensee identified that both trains of the boric acid storage system had been inoperable for 102.3 hours between April 22 to April 26, 2001. This is a condition prohibited by TS 3.2.C.2. The apparent cause of this condition was human error during preventative maintenance on 22 boric acid transfer pump discharge diaphragm valve. Specifically, the valves internal fingerplate in the bonnet assembly was installed upside down. Following the removal from service of the 21 boric acid tank for inspection and installation of a modified level indication system. This licensee-identified violation is further documented in report detail 4OA7. This LER is closed.

4. Full Activation Test of Emergency Planning Sirens

   a. Inspection Scope

   Entergy conducted an Alert Notification System (ANS, i.e., sirens) full system activation test on August 26, 2003. During that test, the siren verification system report provided to the four counties indicated that 30 of 154 sirens failed to operate properly (Westchester-15 of 77, Rockland-10 of 51, Orange-3 of 16, Putnam-2 of 10). Entergy's detailed review of the system computer records showed that only three sirens had failed (two in Westchester, one in Orange). However, if an actual event had occurred, the counties would have performed route alerting for each “failed” siren, in response to the feedback information they received. This would have resulted in the needless dispatch of numerous emergency organization resources (i.e., county police and fire personnel) due to inaccurate siren operability information.

The inspector reviewed the August 26 test results and discussed them with the Entergy EP manager. The inspector reviewed Entergy’s short and long-term corrective actions which were/will be taken to resolve issues with inaccurate siren verification system reports provided to the counties. The inspector also interviewed Westchester County officials to discuss siren verification system issues. The inspector visited the Rockland County EOC, observed the operation of the siren control console at the County Warning Point, and interviewed Rockland County officials concerning siren issues.

Entergy conducted another full-system activation test on October 21, 2003 after corrective actions were completed. There were observers at all 154 sirens during this test; one siren failed to operate.

b. Findings

No findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings (71152)

Section 1R12 describes a finding in which a flexible coupling on the service water supply to 21 emergency diesel generator failed due to improper installation. Work Planning failed to place a critical installation characteristic for gasket alignment in the work package. After the failure, analysis identified that the gasket misalignment was greater than that allowed by the vendor

Section 1R13 describes a finding in which a plant transient occurred due to the improper installation of a pressurizer level recorder. The work was performed in accordance with the work package, however the package was planned using inaccurate design modification data. Design engineering failed to accurately identify the recorder input characteristics.

Section 1R14 describes a CVCS system leak and loss of emergency boration capability due to inadequate restoration steps in a maintenance procedure. Operations planning failed to identify that the restoration procedure, as written, would leave a vent valve open when flow was restored.

4OA6 Meetings, Including Exit

The inspectors met with Indian Point 2 representatives at the conclusion of the inspection on October 8, 2003. At that time, the purpose and scope of the inspection were reviewed, and the preliminary findings were presented. The licensee acknowledged the preliminary inspection findings.

The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was reviewed during this inspection.

On July 27, 2003, the NRC Chairman and Regional Administrator visited Indian Point Energy Center for a plant tour, discussions with licensee personnel and observed a portion of a security drill.

4OA7 Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program.

- TS 3.2.B.3 requires one flow path from the boric acid storage tank to the reactor coolant system. TS 3.2.C.2 allows this flow path to be inoperable for a period of up to 48 hours before going to a hot shutdown condition. Contrary to this, the licensee determined that between April 22 and 26, 2001, both trains of the boric acid storage system were inoperable for 54.3 hours. This was identified in the licensee's corrective action program (CR-IP2-2003-01681) and LER 2003-002. Previously, the inspectors documented a very low safety significant finding (reference inspection report 50-247/2003-003) concerning an inadequate post-work test in 2001 on a boric acid discharge valve. A second flowpath was taken out of service for planned maintenance on a tank inspection and upgrade to the tank's level transmitter. Since the inadequate post work test on the discharge valve was not reasonably known by the licensee at the time of planned maintenance on the other train no performance deficiency exists. This licensee identified finding is of very low safety significance because a loss of safety function for emergency boration did not exist since the refueling water storage tank was available to the suction of the charging pumps.

SUPPLEMENTAL INFORMATION

Key Points of Contact

Licensee Personnel:

A. Burger	Radioactive Waste Operator
K. Cullen	Radiation Protection Technician
R. Deschamps	Radiation Protection Coordinator
R. DeCensi	Technical Support Manager and Radiation Protection Manager
M. Donegan	Unit 1 Project Coordinator
C. English	Unit 1 Project Coordinator
R. Fuchek	Radiation Protection Supervisor
D. Gately	Assistant Radiation Protection Manager
L. Glander	Radiation Protection Supervisor
J. LePere	Waste Services Engineer
R. LaVera	ALARA/Planning Supervisor
L. Menoscal	System Engineer
T. Phillips	Radiological Engineer
R. Rodino	Radiation Protection Supervisor
W. Scholtens	Radioactive Waste Specialist
R. Tagliamonte	Radioactive Waste Supervisor
R. Vogle	Integration Specialist
L. Cortopassi,	Training Manager
R. Christman,	Superintendent, Operations Training
A. Singer	IP2 LOR Program Administrator
J. Comiotes	Director, Nuclear Safety Assurance
F. Dacimo	Vice President, Nuclear Operations
A. Iavicoli	QA Lead Auditor
F. Inzirillo	Emergency Planning Manager
T. Jones	Licensing Engineer
J. McCann	Licensing Manager
J. Perrotta,	QA Manager
C. Schwarz	General Manager, Plant Operations

Westchester County:

A. Sutton, Deputy Commissioner, Westchester County Department of Emergency Services

Rockland County:

D. Greeley, Assistant Director, Fire and Emergency Services

List of Items Opened, Closed, and Discussed

Opened

05000247/2003011-01	URI	Unresolved pending completion of the licensee's evaluation of the identified cable splices and determination of whether there are risk significant cables spliced in the area and potential impact on IPEEE PAB internal flood analysis.

Opened/Closed

NCV 50-247/03-11-02	NCV of 10 CFR 50, Appendix B, Criterion V in November 2002, the procedural steps for the installation of a flexible coupling were not adequate to verify that the component was properly installed. Specifically, the installation instructions did not verify that the proper centering of the gasket was within the manufacturer's specification.
NCV 50-247/03-11-03	NCV of 10 CFR 50 Appendix B, Criterion III a design change package DCP-200105716-I did not accurately reflect actual plant conditions and resulted in an unintended plant transient which challenged plant operators during the removal of the installed pressurizer recorder instrument.
NCV 50-247/03-11-05	NCV of 10 CFR 50 Appendix B Criterion V that on September 5, 2003, SOP 3.1 and equipment tagout (2-CVCS-22CVCSF) to restore the 22 seal injection filter were inadequate to maintain proper configuration control of the system.

Closed

FIN 50-247/03-11-04	The performance finding involved inadequate short-term corrective actions associated with fire leaks on a fire header in the Unit 1 turbine building.
URI 50-247/01-09-01	Design Problem with Source Range Nuclear Instruments counted as a Safety System Functional Failure
LER 50-247/03-03-00	Automatic Reactor Trip Initiated by a Main Turbine Trip on Auto Stop Oil
LER 50-247/03-02-00	Plant in a Condition Prohibited by Technical Specification due to Unavailability of Boric Acid Storage For More Than 48 Hours

List of Documents Reviewed

Section 1R04

Technical Specifications for the CCR HVAC, Section 3.3.H
Maintenance Rule Background Document for CCR HVAC
Outstanding elective and corrective maintenance activities associated with CCR HVAC
Outstanding control room deficiencies associated with CCR HVAC
System Engineering Health Reports for CCR HVAC for 3rd quarter of 2001
Design Basis Document for CCR HVAC
LERs 50-247/99-009 and 50-247/87-007
Condition Reports CR-IP2-2001-02715, CR-IP2-2001-06105, CR-IP2-2001-07536, CR-IP2-2002-02699, CR-IP2-2002-03461, and CR-IP2-2003-00070.

Section 1R06

Procedure AOI 28.0.6, Nuclear Side (Outside Containment) Flooding
WO IP2-03-14062
CR-IP2-2003-05638
CR-IP2-2002-02474
CR-IP2-2003-03035
Indian Point 2 Individual Plant Examination of External Events (IPEEE) Section 5.0

Sections 1EP2, 1EP3, 1EP4, and 1EP5

IP-EP-AD4, "Conduct of Drills and Exercises," Revision 0.
IP-EP-AD5, "Emergency Preparedness Performance Indicator Program," Revision 1.
IP-EP-AD9, "Notification Systems Testing and Maintenance," Revision 1.
NEM-5.702, "Testing of the Indian Point Siren System," Revision 0.

Section 2OS1

SAO-213, Containment entry, egress, and inspection, Rev. 4
SAO-302, Radiation work permit (RWP) program, Rev. 20
RWP 032028, Nonoutage vapor containment entries, Rev. 00
RWP 031203, Unit 1 project, Task 14 - Vessel work and inspections, Rev. 00
Radiological survey/job coverage record on July 9, 2003 for the catwalk in the below-floor spent fuel pool demineralizer valve gallery in the Unit 1 fuel handling building (70-foot elevation)
Project plan for phase 1 of Unit 1 remediation
Tank inspection timeline report for Unit 1, April 29, 2003
Self-assessment of RWPs and work practices for 2003, IPEC-RP-2003-060, June 12, 2003

Section 2OS2

SAO-302, Radiation work permit (RWP) program, Rev. 20
IP-SMM-RP-101, Temporary shielding program, Rev. 0
IP#1 daily ALARA information dated July 8, 2003
IP#2 daily ALARA information dated July 8, 2003
Presentation package for IPEC ALARA committee meeting on June 19, 2003
Status of radiation exposure reduction plan, LO-IP3LO-2003-00359, July 9, 2003
Bench marking report for visits to Byron, Braidwood, and Callaway plants, LO-IP3LO-00359 CA-00026, June 2003

Section 2OS3

DOS-6.121, Dosimetry laboratory quality manual, Rev. 3
National Voluntary Laboratory Accreditation Program (NVLAP) On-Site Assessment Report for Ionizing Radiation Dosimetry, June 25-27, 2003
NVLAP certificate of accreditation, Entergy Nuclear Northeast, Buchanan, NY, ionizing Radiation Dosimetry, Effective through June 30, 2004
Random instrumentation field calibration check observation for first half of 2003, P-HPS-2003-205, June 15, 2003
IPEC Snapshot self-assessment report, air-supplied respiratory protection equipment, learning organization opportunity - INPO OE 03-16239 “Separation of airline coupling on supplied air hood,” July 3, 2003 (CR-IP3-2003-04012)

Radioactive Waste System Walkdown

- Procedure RW-SQ-4.007, Revision 9, Process control program

Waste Characterization and Classification

- Radio-chemical sample analysis results for radioactive waste streams analyzed in 2002 and 2003 and the development of scaling factors for difficult-to-detect-and-measure radionuclides
- Procedure RW-Q-4.006, Revision 8, 10 CFR 61 Sampling program
- Procedure RW-SQ-4.007, Revision 9, Process control program
- Procedure RW-SQ-4.011, Revision 3, RADMAN program operation

Shipment Preparation

- Radioactive material shipping logs for 2002 and 2003
- Radioactive waste shipping logs for 2002 and 2003
- Procedure RW-SQ-4.000, Revision 14, Shipment final QC inspection
- Procedure RW-SQ-4.105, Revision 12, Survey of radioactive material shipments
- Procedure RW-SQ-4.107, Revision 13, Radioactive shipment preparation
- Procedure RW-SQ-4.109, Revision 10, Radioactive material storage
- Procedure RW-SQ-4.210, Revision 5, Management of solid radwaste
- Procedure RW-SQ-4.303, Revision 14, Shipping cask handling procedure
- Procedure RW-4.304, Revision 16, Dry active waste processing
- Procedure RW-4.500, Revision 6, Decontamination of areas and components

- Procedure RW-SQ-4.700, Revision 14, Spent resin storage tank transfer setup,

Shipping Records

- Shipment No. 03-116, LSA II, liquid waste system resin
- Shipment No. 03-108, Type B package, primary resin
- Shipment No. 03-039, LSA II, dry active waste
- Shipment No. 03-104, SCO II, contaminated long-handled tool
- Shipment No. 03-091, SCO II, contaminated fuel pool cooling equipment

Condition Reports

CR-IP2-2003-00756, concerning the failure of two Orange County sirens during the quarterly growl test.
CR-IP2-2003-01107, concerning the failure of all 154 sirens during cancel testing.
CR-IP2-2003-02126, concerning the loss of 10 out 16 Orange County sirens due to a loss of power to the Harriman Tower repeater site.
CR-IP2-2003-02291, concerning the loss of nine out of 16 Orange County sirens due to a power outage.
CR-IP2-2003-02382, concerning the loss of two out of 10 Putnam County sirens due to the loss of power to repeater R-257 in Rockland County.
CR-IP2-2003-02404, concerning the initiation of 27 CRs dealing with siren failures since January 1, 2003.
CR-IP2-2003-02873, concerning siren communication failures due to the terminal servers not fully responding.
CR-IP2-2003-02910, concerning the loss of all 154 sirens due to the failure of the two terminal servers in the emergency operations facility.
CR-IP2-2003-03103, concerning the loss of three out of 10 Putnam County sirens due to fuse failure at the sirens.
CR-IP2-2003-03892, concerning the loss of all 154 sirens due to computer software memory lock-up.
CR-IP3-2003-02810, concerning Emergency Planning Department's assessment of the NUE declaration at Indian Point 3 on April 29, 2003.
Unit 3 Notification of Unusual Event Report, April 29, 2003.
CR-IP2-2003-05638, Post work test on 21 fan cooler unit outlet valve
CR-IP2-2002-02474, unidentified cable splices
CR-IP2-2003-03035, emergency diesel generator prelube pump circuit breaker operation

Identification and Resolution of Problems

- Nuclear quality assurance independent oversight program assessment report no.01-AR-31-RP, September 2001, Radwaste
- Nuclear quality assurance independent oversight program assessment report no.01-AR-35-RP, January 2002, Radwaste
- Nuclear quality assurance independent oversight program assessment report no.02-AR-06-RP, March 2002, Radwaste
- Draft self-assessment of radioactive shipment documentation, August 2003
- Condition Report Nos. CR-IP2-2002-01881, 2002-04408, 2003-02724, and 2003-03750

List of Baseline Inspections Performed

7111402	Alert and Notification System Testing	
7111403	Emergency Response Organization Augmentation Testing	
7111404	Emergency Action Level and Emergency Plan Changes	
7111405	Corrections of Emergency Preparedness Weaknesses and Deficiencies	
71151	Performance Indicator Verification	
71152	Identification and Resolution of Problems (Sample)	
71111.01	Adverse Weather	1R01
71111.04	Equipment Alignment	1R04
71111.05	Fire Protection	1R05
71111.06	Flood Measures	1R06
71111.11	Operator Requalification	1R11
71111.12	Maintenance Effectiveness	1R12
71111.13	Maintenance Risk Assessment and Emergent Work Activities	1R13
71111.14	Personnel Performance During Non-Routine Plant Evolutions	1R14
71111.15	Operability Evaluations	1R15
71111.19	Post Maintenance Testing	1R19
71111.22	Surveillance Testing	1R22
71111.23	Temporary Plant Modifications	1R23
71121.01	Access Control to Radiologically Significant Areas	2OS1
71121.02	ALARA Planning and Controls	2OS2
71121.03	Radiation Monitoring Instrumentation and Protective Equipment	2OS3
71122.02	Radioactive Material Processing and Transportation	2PS2
71151	Performance Indicator Verification	4OA1
71152	Problem Identification and Resolution Sample	4OA2
71153	Event Followup	4OA3

List of Acronyms

ACCWP	auxiliary component cooling water pump
ALARA	as low as reasonably achievable
ANS	alert and notification system
AOI	abnormal operating instruction
CAP	corrective action program
CCR	central control room
CCW	component cooling water
CFR	Code of Federal Regulations
CR	condition report
CVCS	chemical volume and control system
DEP	drill and exercise performance
DOT	Department of Transportation
EAL	emergency action level
EDG	emergency diesel generator
EP	emergency preparedness
ERO	Emergency Response Organization
FPRSSM	fire protection risk significance screening methodology
gpm	gallons per minute
HVAC	heating ventilation and air conditioning
IMC	Inspection Manual chapter
IPEC	Indian Point Energy Center
IPEEE	Individual Plant Examination for External Events
IST	in-service test
LER	licensee event report
LSA	low specific activity
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
PAB	primary auxiliary building
PCP	process control program
PI	performance indicator
PS	public safety
PWT	post work test
QA	quality assurance
RCA	radiologically controlled area
RWP	radiation work permit
SAO	Station Administrative Orders
SCO	surface contaminated object
SDP	significance determination process
TSC	technical support center
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
WO	work order