May 13, 2003

Mr. Fred Dacimo
Vice President - Operations
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Units 1 & 2
295 Broadway, Suite 1
Post Office Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT 2 - NRC INTEGRATED INSPECTION REPORT 50-247/03-03

Dear Mr. Dacimo:

On March 29, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at the Indian Point 2 Nuclear Power Plant. The enclosed integrated inspection report documents the inspection findings, which were discussed on April 9, 2003, with yourself and other members of your staff.

The inspections 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), all of which were determined to be violations of NRC requirements. However, because of their very low safety significance and because the issues have been addressed and entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the receipt of this letter, 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 the Indian Point 2 facility.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders (dated February 25, 2002, January 7, 2003, and three on April 29, 2003) and several threat advisories to licensees of commercial power reactors to strengthen licensee security capabilities, improve security force readiness and training, and enhance access authorization. The NRC also issued Temporary Instruction 2515/148 on August 28, 2002, that provided guidance to inspectors to audit and inspect licensee implementation of the interim compensatory measures (ICMs) required by the February 25th Order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) '02, and the remaining inspections are scheduled for completion in CY '03. Additionally, table-top security drills were conducted at several licensees to evaluate the impact of expanded adversary characteristics and the ICMs on licensee protection and mitigative strategies. Information gained and discrepancies identified during the audits and drills were reviewed and dispositioned by the Office of Nuclear Security and Incident Response. For CY ‘03, the NRC will continue to monitor overall safeguards and security controls, conduct inspections, and resume force-on-force exercises at selected power plants. The Indian Point site will receive a “pilot” force-on-force exercise this summer. Should threat conditions change, the USNRC may issue additional Orders, advisories, and temporary instructions to ensure adequate safety is being maintained at all commercial power reactors.

The inspectors reviewed eight effectiveness reviews associated with the Fundamental Improvement Plan (FIP). The FIP was an improvement plan initiated in early 2002 in response to the NRC’s classification of Indian Point Unit 2 as a multiple degraded cornerstone column facility. The effectiveness reviews evaluated the quality of corrective actions in specific areas and concluded if those actions had improved performance. Specific areas evaluated included: management observation of work activities; operator burden and work-down curve for temporary alterations; review of the design basis initiative project; optimization of the preventive maintenance program; corrective actions in monitoring the work control process; equipment reliability actions; work management self-assessments; and corrective action effectiveness reviews for condition reports. The inspectors concluded that the effectiveness reviews were not uniformly self-critical or consistent with recent NRC assessments or performance metrics. For example, the effectiveness review for the design basis initiative project primarily focused on the quality of action plans and efficiencies of plan implementation, instead of the quality of the engineering staff’s products or recently developed design information “road maps.” However, the inspectors did note that the 2003 Indian Point Business Plan does provide appropriate actions to support improvements in the key areas addressed in the FIP. Your attention to the quality of self-assessments remains an important element to continued station improvement.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room **or** from the Publicly Available Records (PARS) component of the NRC’s document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). Should you have any questions regarding this report, please contact Mr. Peter Eselgroth at 610-337-5234.

Sincerely,

/RA/

Brian E. Holian, Deputy Director
Division of Reactor Projects

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

Enclosure: Inspection Report 50-247/03-03
W/Attachment: Supplemental Information

cc w/encl: G. J. Taylor, Chief Executive Officer, Entergy Nuclear
M. R. Kansler, President - Entergy Nuclear Northeast

J. Herron, Senior Vice President, Indian Point Energy Center
C. Schwarz, General Manager - Plant Operations
D. Pace, Vice President, Engineering
J. Knubel, Vice President, Operations Support
J. McCann, Manager, Nuclear Safety and Licensing
J. Kelly, Director, Nuclear Safety Assurance
C. Faison, Manager, Licensing
H. Salmon, Jr., Director of Oversight
J. Fulton, Assistant General Counsel, Entergy Nuclear Operations, Inc.
W. Flynn, 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, Operations Manager
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
T. Judson, Central NY Citizens Awareness Network
M. Elie, Citizens Awareness Network
D. Lochbaum, Nuclear Safety Engineer, Union of Concerned Scientists
Public Citizen's Critical Mass Energy Project
M. Mariotte, Nuclear Information & Resources Service
F. Zalcman, Pace Law School, Energy Project
L. Puglisi, Supervisor, Town of Cortlandt
Congresswoman Sue W. Kelly
Congresswoman Nita Lowey
Senator Hilary Rodham Clinton
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

Distribution w/encl: H. Miller, RA/J. Wiggins, DRA (1)
H. Nieh, RI EDO Coordinator
P. Habighorst, SRI - Indian Point 2
R. Laufer, NRR
P. Eselgroth, DRP
P. Milano, PM, NRR
G. Vissing, PM, NRR (Backup)
W. Cook, DRP
R. Junod, DRP
R. Martin, DRP
Region I Docket Room (w/concurrences)

DOCUMENT NAME: C:\ORPCheckout\FileNET\ML031340032.wpd
After declaring this document "An Official Agency Record" it **will** be released to the Public. **To receive a copy of this document, indicate in the box:** "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP	E	RI/DRP	E
NAME	PHabighorst/PWE for		PEselgroth/PWE		BHolian/BEH	
DATE	05/13/03		05/13/03		05/13/03	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	50-247/03-03
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	December 29, 2002 - March 29, 2003
Inspectors:	Peter Habighorst, Senior Resident Inspector Lois James, Resident Inspector Jason C. Jang, Senior Health Physicist (1/6-1/10/03) William Cook, Senior Project Engineer (3/10-3/14/03) David Silk, Emergency Preparedness Specialist (1/6-29/03) Daniel Barss, Emergency Preparedness Specialist (1/6-29/03), Thomas Burns, Senior Reactor Engineer, DRS (2/10-2/13/03) John R. McFadden, Health Physicist, DRS (2/24-2/28/03) Leonard Cheung, Sr. Reactor Engineer, DRS (11/4-11/9/02)
Approved by:	Peter W. Eselgroth, Chief Projects Branch 2 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000247-03-03, on December 29, 2002 - March 29, 2003, Entergy Nuclear Operations, Inc.; Indian Point 2 Nuclear Power Plant; Maintenance Risk Assessment/Emergent Work, Post Maintenance Testing; and Radioactive Material Processing and Transportation.

The report covered a twelve-week period of inspection by resident, region-based, and headquarters-based inspectors. Four Green non-cited violations (NCVs), and two unresolved items were identified. The significance of the findings are indicated by their color (Green, White, Yellow, Red) in accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July, 2000.

A. NRC- Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. On February 7, 2003, a self-revealing finding involved inadequate emergent work instructions that resulted in an electrical short during replacement of the 22 steam generator low level bistable. The electrical short caused a breaker trip on circuit 10 of instrument bus 21and the resultant loss of electrical power to the pressurizer level and reactor coolant system pressure control channels (failed low). The inadequate work instructions is considered a non-cited violation of 10 CFR 50 Appendix B, Criterion V, since the instructions did not account for consideration of performing this replacement with the circuit de-energized or the proximity to other reactor protection system relays.

  The performance issue is more than minor since the operators were required to take action to restore reactor coolant system pressure and pressurizer level to preclude a reactor trip. The finding involves the initiating events cornerstone in that it increased the likelihood of upset in plant stability and it involves human error during the planning of an emergent work activity. This finding is considered to be of very low safety significance in that in accordance with NRC Manual Chapter 0609, Appendix A, the finding did not contribute to the likelihood of a secondary or primary LOCA initiator and it did not contribute to either a reactor trip or mitigation system unavailability. (Section 1R13)

Cornerstone: Mitigating Systems

- Green. A self-revealing event was identified on February 26, 2003, when operators observed no boric acid flow to the reactor vessel via the No. 22 boric acid transfer pump (BATP). It was determined that during preventative maintenance activities in March 2001, the post-work test on the No. 22 BATP outlet valve to the boric acid filter stop was inadequate to identify that the valve finger plate was installed upside down. This finding is considered a non-cited violation of 10 CFR 50 Appendix B, Criterion V. This event is considered more than minor because the improperly installed valve plate affected the availability of one train of emergency boration. This is considered to be of very low risk significance in accordance with NRC MC 0609 Appendix A, since the emergency boration function was not lost due to this performance issue. (Section 1R19)

- Green. The inspectors identified that ineffective corrective actions resulted in repetitive surveillance test failures of the 23 emergency diesel generator between December 2001 and February 2003. This finding is considered a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI. The finding is more than minor because the surveillance test failures impacted the availability of one train of emergency AC power source. This finding was of very low risk significance because the repetitive failures did not result in an actual loss of function for the emergency AC power. (Section 1R13)

Cornerstone: Public Safety

- Green. A self-revealing non-cited violation of 10 CFR 71.12 was identified for failure to comply with shipping cask package procedures. On February 6, 2003, a CNS 8-120 B cask was received from the Indian Point Energy Center at a consolidation facility in South Carolina with a bolt missing on the primary lid's pressure test port in violation of the cask use and maintenance procedures. This finding was more than minor in that it was associated with the Public Radiation Safety Cornerstone's attribute of procedures for transportation packages. The finding affected the associated cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive materials contained in an NRC-approved Type B package released into the public domain. The finding was determined to be of very low safety significance in that the finding did not involve exceeding transportation radiation limits, there was no breach of the package during transit, and the issue was a Certificate of Compliance maintenance/use performance deficiency. (Section 2PS2)

REPORT DETAILS

SUMMARY OF PLANT STATUS

1. REACTOR SAFETY

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

1R04 Equipment Alignment

a. Inspection Scope

.1 Partial System Walkdowns

On January 23, 2003, the inspector performed a partial system walkdown of the 22 auxiliary feedwater pump (AFWP) train while the 23 AFWP train was out of service for preventive maintenance. The purpose of this walkdown was to verify equipment alignment and identify any discrepancies that could adversely impact the function of the steam driven auxiliary feedwater pump train. The inspector observed the physical condition of the system pump and valves and reviewed the operations logs. The inspector used check-off lists (COLs) 21.3, “Steam Generator Water Level and Auxiliary Boiler Feedwater” and 18.1, “Main and Reheat Steam,” for this walkdown and reviewed the design basis document for the auxiliary feedwater system and Technical Specification 3.4 to verify the valve positions, as defined in the COL, were appropriate.

On February 25, 2003, the inspector performed a system walkdown of the fuel oil portion of the emergency diesel generator (EDG) system while the 21 EDG was out of service for preventive maintenance. The purpose of this walkdown was to verify equipment alignment and identify any discrepancies that could adversely impact the operation of the remaining EDGs and thereby increase risk. The inspector observed the physical condition of the fuel oil system pumps and valves. The inspector used COL 27.3.1, “Diesel Generators,” for this walkdown and reviewed the design basis document for the diesel generator fuel oil system.

On March 11, 2003, the inspector performed a system walkdown of the 21 containment spray system while the 22 containment spray pump was out of service for planned testing and preventative maintenance. The purpose of this walkdown was to verify equipment alignment and identify any discrepancies that could adversely impact the function of the containment spray system and thereby increase risk. The inspector observed the physical condition of the containment spray pump and valves. The inspector used COL 10.2.1, “Containment Spray System,” system operating procedure (SOP) 10.2.1, “Containment Spray System Operation,” and plant drawings 9321-F-2735-130 and A225296. Minor deficiencies involving plant labeling and plant drawing errors were provided to the licensee and addressed via the corrective action process. The inspector’s review of an operability determination for the containment pressure instruments is documented in report section 1R15.

b. Findings

No findings of significance were identified.

.2 Full System Alignment

a. Inspection Scope

The inspector performed a walkdown of accessible portions of the engineered safeguards features system (ESFS) to verify electrical separation between channels 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 ESFS based upon its importance to plant safety and risk. This system is in the top twenty systems at the unit based upon risk achievement worth (which measures the relative risk of systems based on IPE data). The inspector reviewed the following documents to confirm system availability and functional capability:

- The Technical Specifications for the ESFS, Section 3.5.3
- Maintenance Rule Background Document for ESFS
- Outstanding elective and corrective maintenance activities associated with ESFS
- Outstanding control room deficiencies associated with ESFS
- Last completed Technical Specification surveillances: PC-R4, Pressurizer Pressure; PC-R4-1, Pressurizer Pressure Transmitters; PT-Q55, Pressurizer Pressure; PC-R29, Main Steam Line Flow Instrumentation - CCR; PT-Q63, Steam Flow/Feedwater Flow Mismatch Bistables; and PC-R32-1, Main Feedwater Flow - Transmitters;
- System Engineering Health Reports for ESFS for 4th quarter of 2002
- SE-350 Attachment 8.2, System Monitoring Basis Document for ESFS
- Design Basis Document for Engineered Safeguards Feature System
- Abnormal Operating Instruction 10.1.4, Safeguards Relays DC Power Failure
- Condition Report IP2-2003-01026

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Fire Zone Tours

a. Inspection Scope

The inspector toured the 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 63A, circulating water pumps area
- Fire Zone 610, Unit 1 screenwell room
- Fire Zone 14, auxiliary feedwater pump room
- Fire Zone 23, 480V switchgear room
- Fire Zone 6A, Waste Storage and Drumming Station
- Fire Zone 74A and 74B, Electrical Penetration Areas of the Fan House
- Fire Zone 7A, 80-foot elevation of Primary Auxiliary Building
- Fire Zone 43A, 15-foot elevation of Turbine Building

Reference material consulted by the inspector included the Fire Protection Implementation Plan, Pre-Fire Plan, and Station Administrative Order (SAO)-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 associated condition reports for these minor errors are identified in the Attachment to this inspection report.

b. Findings

No findings of significance were identified.

.2 Lack of Cable Separation in Fire Areas F and J

a. Inspection Scope

On February 6, 2003, Entergy identified that the routing of charging pump power supply and control cables do not meet the cable separation criteria specified in 10 CFR 50 Appendix R, Section III.G.1. This was reported to the NRC via 10 CFR 50.72(b)(3)(ii)(B) (Event Notification 39571) as an unanalyzed condition and documented in condition report IP2-2003-00765. A postulated fire in Fire Area F (fire zones 6, 7 and 7A) would disable all three charging pumps. The postulated fire could result in the loss of: the 23 charging pump alternate power feed transfer switch; 23 charging pump alternate power feed; the alternate/normal power feed cable between the transfer switch and the pump; 22 charging pump power feed cable; local/remote control cabling (disabling remote operation of all charging pump breakers); and the control cables and pneumatics for the 22 and 23 charging pumps.

This design vulnerability was identified by Entergy during a re-baseline analysis to validate compliance with 10 CFR 50 Appendix R. Entergy's review was being performed under a design basis initiative project (DBI-PI-1) in response to previously identified concerns about the analysis. Efforts include the location of all Appendix R credited equipment, power, control and instrument cables, power sources, local controls and indication, and other features by fire area and fire zone.

On February 12, 2003, Entergy identified that the normal and emergency power supplies to the six service water pumps were routed through manhole 23 (Unit 2 turbine building under the 15 elevation) and are completely separated from the fire area except for manhole 23. The alternate safety shutdown power cables run approximately 200 feet to the south. The alternate power supply cables for two of the six service water pumps are routed unprotected from Fire Area J through Fire Area A at the south end of the Unit 1 turbine building. The vulnerability is a postulated fire in Fire area J that could result in a complete loss of power to all service water pumps. This was reported to the NRC as an undated to Event Notification 39571. Condition Report IP2-2003-867 documented this deficiency.

The inspector reviewed and verified that licensee compensatory measures for the vulnerabilities in Fire Areas F and J were consistent with SAO-703, "Fire Protection Impairment Criteria and Surveillance," Addendum I, item 9. The inspector walked down the areas to confirm Entergy's conclusions regarding the fire vulnerability. The inspector reviewed the following documents:

- Abnormal Operating Instruction (AOI)27.1.9, Control Room Inaccessibility Safe Shutdown Control
- OASL 15.11, Attachment 1, 480 Volt DB-50 Breaker Operations
- Individual Plant Examination for External Events (IPEEE)

b. Findings

The licensee-identified postulated fire vulnerabilities within Fire Zones F and J are considered unresolved **(URI 50-247/03-03-01)**. A preliminary assessment of these cable separation discrepancies identifies them as low safety consequence, based upon: previous inspection review of the alternate safe shutdown capability and fire protection program implementation (reference inspection report No. 50-247/2000-004, dated May 17, 2001); low combustible loading in the affected fire zones; appropriate compensatory measures in place until final resolution; and low risk significance based upon current IPEEE analysis of the affected fire zones. This issue will remain unresolved pending the completion of the NRC/industry review and resolution of issues affecting safe shutdown associated circuits and manual actions. As discussed above, these issues have been placed into the licensee's corrective action process.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed Entergy's external flood analysis, flood mitigation procedures, and design features to verify whether they were consistent with the IP2's design requirements. The inspectors walked down selected external plant areas, including areas for large on-site tanks that contain equipment important to safety. The inspectors evaluated the condition and adequacy of mitigation equipment to assess whether the flood protection design features were adequate and operable. During the walk downs, the inspectors also evaluated whether there were any unidentified or unanalyzed sources of flooding. The specific areas included:

- Service Water Strainer Pit
- Unit 2 Condensate Storage Tanks
- Refueling Water Storage Tank

The inspectors reviewed Entergy's flood mitigation procedures, flood alarm response procedures, and selected preventive maintenance tasks and surveillance tests for the sump pump in the service water strainer pit to evaluate whether component functionality was routinely verified. In addition, the inspectors reviewed Entergy's corrective action program to verify whether previous flood related issues had been appropriately identified, evaluated, and resolved. The following procedures were included in the review:

- Individual Plant Examination for External Events (IPEEE) Section 6.0 External Flooding
- Updated Final Safety Analysis Report (UFSAR) Section 2.5 Hydrology
- AOI 28.0.6, "Nuclear Side (outside containment) Flooding"
- AOI 28.0.4, "Flooding - Conventional Plant"
- AOI 24.1, "Service Water Malfunction"

b. Findings

No findings of significance were identified.

1R11 Operator Requalification Inspection

a. Inspection Scope

On March 3, 2003, the inspector observed the performance of an operating crew during licensed operator re-qualification training. Specifically, the inspector observed simulator "as-found" exams associated with lesson plan SS.700.032. The inspection was conducted to assess the adequacy of the training, licensed operator performance, emergency plan implementation, and the adequacy of the licensee's critique.

b. Findings

No findings of significance were identified. The operating crew satisfactorily completed the simulator scenario critical task. The crew critique and the evaluators assessment were consistent with the inspectors observation of crew performance. Performance issues associated with the crew involved abnormal operating instruction and emergency operating procedure inconsistent usage and adherence, and recognition of applicable Technical Specifications associated with an instrument failure. The crew initiated a performance improvement plan to improve use of emergency operating procedure and abnormal operating procedure attachments.

1R12 Maintenance Effectiveness

.1 22 Component Cooling Water

a. Inspection Scope

The inspectors evaluated Entergy's work practices and preventive maintenance activities for the 22 component cooling water (CCW) pump to assess the effectiveness of maintenance activities. The inspectors reviewed the performance history of the 22 CCW pump to assess the adequacy of the licensee's corrective actions and to evaluate Entergy's monitoring, evaluations, and disposition of issues in accordance with station procedures and the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance." The inspectors reviewed the following documents associated with the system design and licensing basis:

- Maintenance Rule Basis Document for Component Cooling Water, Revision 1
- System Health Report for the Component Cooling Water System, 4^{th} quarter 2002
- Design Bases Document for the Component Cooling Water System, Revision 0
- System Engineering Procedure SE-303, Maintenance Rule Performance Criteria Development/ Monitoring, Revision 0
- UFSAR Section 9.3, Auxiliary Cooling System
- Technical Specification 3.3.E, Component Cooling System
- Condition Report Nos. CR-IP2-2002-11242, CR-IP2-2003-00515, CR-IP2-2003-01217

b. Findings

No findings of significance were identified.

.2 Boric Acid Transfer Pumps

a. Inspection Scope

The inspectors evaluated Entergy's work practices and preventive maintenance activities for the 21 and 22 boric acid transfer pumps (BATPs) to assess the effectiveness of maintenance activities. The inspectors reviewed the performance history of the 21 and 22 BATPs to assess Entergy's problem identification and the adequacy of the licensee's corrective actions to evaluate whether monitoring, evaluations, and dispositioning of issues were completed in accordance with station procedures and the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance." The inspectors reviewed the following documents associated with system design and licensing basis:

- Maintenance Rule Basis Document for the Chemical and Volume Control System
- System Health Report for the Chemical and Volume Control System, 4th quarter 2002
- Design Basis Document for the Component Cooling Water System, Revision 0
- System Engineering Procedure SE-303, Maintenance Rule Performance Criteria Development/ Monitoring, Revision 0
- UFSAR Section 9.2, Chemical and Volume Control System
- Technical Specification 3.2, Chemical and Volume Control System
- Condition Report Nos. IP2-2003-01564, IP2-2003-01121, IP2-2003-00831, IP2-2001-12846, IP2-2002-00441, and IP2-2002-02248

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Activities

a. Inspection Scope

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, took actions to minimize the probability of initiating events, and maintained the functional capability of mitigating systems. The inspector observed and discussed risk management with maintenance and operations personnel for the following activities:

- Surveillance test PT-Q55, during which pressurizer level bistable PC-457A failed to trip (IP2-2003-00981)
- Surveillance test PT-M21C (EDG 23 Load Test), during which the EDG load rapidly increased to 2300 kW and then rapidly decreased to 0 kW (IP2-2003-00570)
- 22 Battery Charger silicon-controlled rectifier replacement (IP2-2003-00732)
- Emergent maintenance to replace the 22 steam generator level bistable (IP2-2003-00788)

b. Findings

.1 23 Emergency Diesel Generator Load Oscillations

Introduction: A Green Finding was identified for ineffective corrective actions that resulted in repetitive surveillance test failures of the 23 emergency diesel generator, (non-cited violation of 10 CFR 50 Appendix B, Criterion XVI).

Description: The inspector identified that during a period between May 28, 2000 and February 6, 2003, the licensee did not adequately correct load oscillations during surveillance testing on the 23 emergency diesel generator.

On January 29, 2003, the 23 emergency diesel generator (EDG) was started for its quarterly surveillance. While the 23 EDG was incrementally being loaded past 500kW, the load rapidly increased to 2300 kW. The nuclear plant operator attempted to lower the load using the governor raise/lower switch which resulted in the load rapidly decreasing to 0 kW and the EDG tripped on reverse power. This issue was documented in CR-IP2-2003-00570 and required a root cause analysis. Troubleshooting identified that the motor operated potentiometer (MOP) may have had a dead spot, which could explain why the 23 EDG load oscillations were intermittent. The MOP was replaced and the 23 EDG was declared operable. Since failure analysis on the MOP would take several weeks, the licensee increased the surveillance frequency on the 23 EDG to provide confidence that the EDG was operable. On February 6, 2003, the 23 EDG again experienced load oscillations during the surveillance testing.

Following the January 29, 2003 load oscillations, the inspectors reviewed the past work orders (WOs) and condition reports (CRs) associated with the 23 EDG and identified several CRs and WOs documenting load oscillations between May 2000 and February 2003. The work orders documenting load issues associated with the 23 EDG were:

- May 28, 2000, (IP2-2000-03978) 23 EDG increased rapidly from about 750 kW and stayed at 2300 kW for about one minute. The EDG was manually unloaded and tripped on reverse power. Corrective maintenance involved replacement of the unit parallel relay; additionally, contact resistance on the unit/parallel switch was checked.
- December 13, 2001, (IP2-2001-12264) surveillance test PT-M21C on 23 EDG was aborted due to load swings with the governor. Corrective maintenance involved the removal and replacement of the motor operated potentiometer (MOP).
- March 23, 2002, (IP2-2002-03079) the MOP for the 23 EDG would not increase or lower generator voltage, the EGB governor was replaced.
- October 8, 2002, (IP2-2002-09072) the operator tripped the 23 EDG during PT-M-21C surveillance test due to unexpected response of the governor load control, the EGA governor was replaced.
- January 29, 2003, (IP2-2003-00570) the load on the 23 EDG increased uncontrollably to 2300 kw and then decreased to 0 kW. Corrective maintenance involved replacement of the motor operated potentiometer.
- February 6, 2003, (IP2-200300758) load oscillations were observed on the 23 EDG and the licensee replaced the unit parallel relay and associated cables.

The review of the work history between May 2000 and February 2003 highlighted that the major components that could cause the load swings were replaced at least once and, in the case of the motor operated potentiometer, three times. In response to previous condition reports, the licensee used industry operating experience to determine the most probable cause of the load swings and justified continued EDG operation. The inspector concluded that the responses to these condition reports did not include an evaluation of the 23 EDG operating history. Condition Report IP2-2003-00570 required a full root cause which included listing all the potential causes along with the past condition reports and work history. This root cause was thorough and logical, and although no specific cause of the load oscillations was determined, it provided confidence that the complete spectrum of possibilities and the complete history were considered.

<u>Analysis:</u> The failure to correct the multiple generator load oscillations was more than minor because the surveillance test failures associated with the load oscillations impacted the availability of mitigating equipment. The inspectors also determined that this finding was able to be assessed using the Significance Determination Process because the finding was associated with the availability of a system or train in a mitigating system. The inspectors conducted a Phase 1 Significance Determination Process screening and determined that the failure to adequately address the cause of multiple surveillance test failures of the 23 EDG due to load oscillations was of very low risk significance because this finding did not represent an actual loss of safety function.

The inspectors confirmed that the last two surveillances that demonstrated automatic start and loading of the 23 emergency diesel generator during accident conditions did not result in load oscillations.

Enforcement: Criterion XVI of 10 CFR Part 50, Appendix B requires that in the case of significant conditions adverse to quality, that measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition. The licensee's failure to adequately correct the load oscillations associated with the 23 emergency diesel generator between May 2000 and February 2003 was considered a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI in accordance with Section VI.A.1 of the NRC's Enforcement Policy **(NCV 50-247/03-03-02)**. This issue was entered into the licensee's corrective action program as IP2-2003-00570.

.2 Maintenance Instruction Deficiency

Introduction: A Green NCV was identified for inadequate emergent maintenance instructions that subsequently resulted in operator action to restore pressurizer level and pressure based upon de-energized control channel instruments.

Description: On February 7, 2003, inadequate emergent work instructions in work order 03-12284 resulted in an electrical short during replacement of the 22 steam generator low level bistable. The electrical short caused the trip of circuit 10 on instrument bus 21. Operators were subsequently challenged to restore pressurizer level and reactor coolant system pressure due to the controlling channels failing low, based upon a loss of the instruments' 120 volt power source. Specifically, prior to operator actions to shift control channels, reactor coolant system pressure decreased approximately 60 psig (resulting in entry into TS 3.1.G.b) and pressurizer level increased 3% above the program band.

Analysis: This performance issue is more than minor since the operators were required to take action to restore reactor coolant system pressure and pressurizer level to preclude a reactor trip. The performance issue involved the initiating events cornerstone in that it increased the likelihood of upset in plant stability; the attribute was human error during the planning of an emergent work activity. In accordance with NRC Manual Chapter 0609, Appendix A, this finding is considered to be of very low safety significance. This conclusion is based upon the finding not contributing to the likelihood of a secondary or primary LOCA initiator and not contributing to either a reactor trip or to mitigation system unavailability.

Enforcement: The inadequate emergent maintenance instruction is a violation of 10 CFR 50 Appendix B, Criterion V, in that the instructions were not appropriate to the circumstances. Specifically, on February 7, 2003, the instructions for work order 03-12284 did not include information to minimize the likelihood of an electrical short near energized reactor protection system bistables and did not have maintenance work prerequisites to minimize plant transient impact, if the affected circuit were to be inadvertently de-energized. The performance issue was considered a non-cited violation of 10 CFR 50 Appendix B, Criterion V in accordance with Section VI.A.1 of NRC's

Enforcement Policy **(NCV 50-247/03-03-03)**, and was documented in Entergy's corrective action program under CR No. IP2-2003-00788. Short-term corrective actions included lessons-learned with both operations and instrument & control department personnel, work planners review of the apparent cause report for IP2-2003-00788, and development of a procedure to replace reactor protection system bistables.

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

.1 Heater Drain Pumps Tripping due to Leak Repair on the Heater Drain Tank Level Column

a. Inspection Scope

On January 9, 2003, both heater drain tank pumps tripped causing the reactor operators to decrease reactor power from 100% to 80% to maintain steam generator level. Prior to the heater drain tank pumps tripping, a steam leak repair had just been completed on the heater drain tank level transmitter (LT-1127). The inspectors observed operator response, reviewed operator logs, interviewed cognizant personnel, and reviewed the licensee's root cause analysis report.

b. Findings

No findings of significance were identified.

.2 22 Main Condensate Pump Motor Failure

a. Inspection Scope

On March 3, 2003, at 8:42 p.m. the control room received a call from the on-shift chemistry technician that the 22 condensate pump motor was on fire. The control room sounded the fire alarm and dispatched the fire brigade. The fire was extinguished four minutes later. Operators entered into AOI 21.1.1, "Loss of Feedwater," and reduced reactor power to 90 percent. Operators had to restore a number of non-safety related pumps and fans in response to a voltage perturbation on the 6.9 KV bus which resulted from a phase-to-phase fault within the 22 condensate motor. This event was documented in Condition Report IP2-2003-1264.

The inspector reviewed the following documents in response to this plant transient:

- Condensate motor maintenance work history and corrective action program history
- Protective relay test results for the 22 condensate motor breaker
- AOI 21.1.1, "Loss of Feedwater"
- Emergency Action Levels

The inspector evaluated equipment response, operator response to the transient, fire brigade response, and short-term corrective actions taken by Entergy. The cause of the motor failure was a phase-to-phase power supply cable fault. The fire was confined to the cable insulation within the splice box where the fault originated.

b. Findings

No findings of significance were identified.

.3 Loss of Control Power to 22 Main Transformer Auxiliaries

a. Inspection Scope

On March 19, 2003, operators responded to a loss of control power to the 22 main transformer auxiliaries. The cause of the loss of control power was a failure of the 480 volt/120 volt transformer. Operators entered into AOI 27.1.7 and reduced thermal power to 78 percent. Transformer oil temperature reached 109 degrees centigrade (one degree from a required turbine trip per AOI 27.1.7) before the operators installed a mechanical blocking device on the auxiliary contactors and re-energized the transformer auxiliaries. Transformer oil temperatures decreased to normal values after auxiliaries were restored.

The inspectors evaluated operator response to this event, reviewed the adequacy of AOI 27.1.7 guidance for coping with this transient, and verified short-term corrective action involving the temporary alteration/blocking device (see report detail 1R23). Entergy documented the following condition reports involving the loss of control power: IP2-2003-1635; IP2-2003-1638; IP2-2003-1640; IP2-2003-1652; and IP2-2003-1673.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the below listed Condition Reports (CRs) 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.

- IP2-2003-01202, No. 21 accumulator gas and fluid piping ISI inspections.
- IP2-2003-00857, 480V breaker No. 2053-002.
- IP2-2003-00938, 480V breaker for No. 22 BATP motor disconnect at MCC No. 27.
- IP2-2003-00992, EDG building HVAC damper air supply pressure regulating valve PRV-5469.

- IP2-2003-01211, No. 21 EDG miscellaneous mounting bolts and brackets not fully secured.
- IP2-2003-01406, EDG fuel oil day tank drain valve leakage.
- IP2-2003-01517 and IP2-2000-09073, Impulse line tubing for the containment pressure transmitters 1814B and 1814C are bent.

b. Findings

No findings of significance were identified.

1R17 Permanent Modifications

Station Auxiliary Transformer Load Tap Changer Modification

a. Inspection Scope

The licensee recently completed two preliminary calculations, FEX-00143-01, "IP2 Load Flow Analysis of the Electrical Distribution System," and FEX-00181-00, "Evaluation of the Load Tap Changer Operation of the Station Auxiliary Transformer Following Fast-Transfer." These preliminary calculations showed that under certain expected grid voltage conditions, coincident with a plant event which required safety injection (SI), the safety buses at the plant could separate from the off-site power supplies. During normal plant operation, four of the 6.9 kV buses are supplied by the unit auxiliary transformer (UAT). With a unit generator trip (30 seconds after the SI initiation), the supply for these four buses would be fast-transferred to the station auxiliary transformer (SAT). The SAT has an automatic load tap changer (LTC) which maintains the secondary voltage at 7.1 kV. With the existing LTC operation, the fast transfer could cause the secondary voltage to drop below the degraded voltage setpoint for more than 10 seconds, resulting in a separation of the safety buses from the off-site power. This issue was documented in CR 2002-07918, dated August 21, 2002. The licensee found that the design basis calculation (EGP-001100-00) which evaluated compliance with degraded grid requirements and compliance with 10 CFR 50, Appendix A, General Design Criteria 17, had a number of non-conservative assumptions. For example, the calculation did not account for: the fast bus transfer 30 seconds after a safety injection signal, instrument tolerances for the degraded voltage relays, and the neutral position of the LTC.

The licensee initiated design change package (DCP) 02-2-005, "SAT Load Tap Changer SI Modification," to improve the SAT secondary voltage response following an SI initiation. The design of this modification was completed on October 3, 2002, and the modification was to be implemented during the November 2002 outage. The modification involved the addition of a bypass circuit (for a duration of 40 seconds only) in the LTC control circuitry that is activated by the SI signal. The purpose of the modification was to quickly raise, following the two second time delay of the LTC and within the time limit of 40 seconds, the SAT secondary voltage to a maximum of 7350 V or to the highest step of the LTC before the fast-transfer takes place (30 seconds after SI initiation). At the time of the inspection (week of November 4, 2002), the relay box of the bypass circuit was installed, field wiring was in-progress, and the pre-installation calibrations of the voltage sensing and time delay relays were completed.

The inspector selected DCP 02-2-005 for review because preventing a degraded voltage condition at the safety-related buses contributes significantly to the prevention of core damage. The inspector reviewed the design features (including voltage settings and time delay settings of the voltage sensing and time delay relays) to verify that the design requirements were met. The inspector also reviewed the modified schematic diagrams, with the added bypass circuit to the LTC control circuitry, to verify the adequacy of the new design. In addition, the inspector reviewed the bench and functional testing requirements and verified that appropriate acceptance criteria were specified.

b. Findings

At the end of the inspection period, Entergy was in the process of completing the reconstitution of the electrical calculations to support the degraded grid analysis. Entergy has open corrective action assignments to evaluate the implications on past operability of the system and to evaluate reportability per 10 CFR 50.73. The inspector considers this issue unresolved, pending NRC review of the final electrical calculations and implications on historical plant risk of off-site power source reliability. **(URI 50-247/03-03-04)**

1R19 Post Maintenance Testing

a. Inspection Scope

The inspector reviewed post-work test (PWT) procedures and associated testing activities to assess whether: 1) the effect of testing in the plant had been adequately addressed by control room personnel; 2) testing was adequate for the maintenance 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 IP2's 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:

- WO IP2-03-10372 and IP2-02-3307, PWT to verify 22 service water pump strainer functions properly, performed on January 21, 2003.
- WO IP2-03-10335 and IP2-03-010189, PWT using PT-Q34 for 22 auxiliary feedwater pump, performed on January 10, 2003.
- CR-IP2-2003-01121, boric acid flow was not verified during blended makeup on February 26, 2003, and WO-IP2-00-13327, preventive maintenance on the boric acid transfer pump 22 outlet to boric acid filter stop, performed in March 2001.

- WOs IP2-03-14057 and IP2-02-33275, PWT using PT-M38A for gas turbine No. 1 and IP2-03-04664, black start diesel tripped on high cooling water temperature during PT-M38A, performed on March 21 and 22, 2003.
- WO IP2-03-15081, PWT to verify functionality after replacing oil in the “B” stator water cooling pump, performed on March 25, 2003.
- WO IP2-03-10989, PWT following replacement of silicon controlled rectifier on the 22 battery charger (CR 2003-00732), performed on February 4, 2003.

b. Findings

Introduction: A Green NCV was identified for an inadequate post-maintenance test on the 22 boric acid transfer pump outlet to the boric acid filter stop (valve No. 370) in 2001.

Description: A self-revealing event was identified on February 26, 2003, when operators observed no boric acid flow to the reactor core via the 22 boric acid transfer pump (BATP) using blended makeup. During preventive maintenance activities in March 2001, the post-work test on valve No. 370 failed to identify that an internal plate to the valve’s diaphragm was installed upside down. The consequence of the finger plate installed upside down resulted in the diaphragm being cut and subsequently causing the valve to hydraulically lock. This degraded valve condition was not identified during surveillance testing because flow does not pass through the valve during periodic testing.

Analysis: The performance deficiency is considered more than minor because the improperly installed valve plate was not identified during the post work test and adversely impacted the availability of a single train of emergency boration. This finding is considered to be of very low risk significance in accordance with NRC MC 0609 Appendix A, since the emergency boration safety function was not lost due to this performance issue.

Enforcement: This finding is a violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” in that the post-work test prescribed in work order No. IP2-00-13327 did not adequately verify that the valve was properly reinstalled after preventive maintenance. The performance issue was considered a non-cited violation of 10 CFR 50 Appendix B, Criterion V in accordance with Section VI.A.1 of NRC’s Enforcement Policy **(NCV 50-247/03-03-05)**, and was documented in Entergy’s corrective action program under CR No. IP2-2003-01121.

1R22 Surveillance Testing

a. Inspection Scope

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 Indian Point 2 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-Q48, AMSAC Logic, revision 5, performed on January 22, 2003
- PT-M21A, Emergency Diesel Generator 21 Load Test, revision 7, performed on January 30, 2003
- SOP 1.7, Reactor Coolant System Leakage Surveillance, revision 35 N-1, performed on February 12, 2003
- PT-Q35B, No. 22 Containment Spray Pump Test, revision 12, performed on March 11, 2003
- PT-Q29C, 23 Safety Injection Pump, revision 13, performed on February 21, 2003.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23)

The inspector reviewed the below listed temporary alteration (TA) to ensure that: the TA was appropriately evaluated by Entergy in accordance with 10 CFR 50.59; the TA did not adversely impact the safety function or operation of the system/component modified; and that the TA was appropriately installed in accordance with administrative procedure ENN-DC-136, “Temporary Alteration Control.” One minor drawing issue was identified by the inspector and entered into the licensee’s corrective action process for resolution. The following TA was reviewed:

- TA-03-2-089, Temporary power to 22 main transformer auxiliary panel.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

During a combined in-office and on-site inspection during January 6 - 29, 2003, the inspectors reviewed recent changes in the EP area. Specifically, Unit 2 and Unit 3 emergency plans were combined into a common Emergency Plan. Also, certain implementing procedures for the Emergency Operation Facility and the Joint News Center are now common to both units. A thorough review was conducted of aspects of the plan related to the risk significant planning standards (RSPS), such as classifications, notifications, and protective action recommendations. A general review was conducted for non-RSPS portions. These changes were reviewed against 10 CFR 50.54(q) to ensure that the changes did not decrease the effectiveness of the plan, and that the changes continued to meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E. For areas for which minor clarifications would enhance the new Emergency Plan, the licensee generated CRs. All of the changes made to the Emergency Plan or associated implementing procedures are subject to future inspections to ensure that the result of the changes continue to meet NRC regulations.

b. Findings

No findings of significance were identified.

1EP6 Emergency Plan Drills

a. Inspection Scope

On February 26, 2003, the inspectors observed Entergy's emergency response organization during an announced emergency preparedness training drill at Indian Point Unit 2. The simulated emergency included the activation of the Operations Support Center (OSC), the Technical Support Center (TSC) and Emergency Operations Facility (EOF) after an Alert (simulated) was declared by the control room operators. The control room simulator was used for the exercise, in addition to the OSC, TSC, EOF, and the Joint News Center (JNC).

The inspectors observed the conduct of the exercise in the control room simulator, OSC, TSC, and EOF. The inspectors assessed licensed operator and the licensee's adherence to emergency plan implementation procedures, and their response to simulated degraded plant conditions to identify weaknesses and deficiencies in classification, notification, and protective action recommendation activities. In addition to the drill, the inspectors observed the licensee's controller critique to evaluate the licensee's self-identification of weaknesses and deficiencies. 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, revision 03-01
- Implementing Procedure (IP)-1002, “Emergency Notification and Communication”
- IP-1010, “Central Control Room”
- IP-1023, “Operations Support Center”
- IP-1030, “Emergency Operations Facility”
- IP-1035, “Technical Support Center”
- Technical Support Center Drill Log
- Operation Support Center Drill Log
- Condition Report Nos. IP2-2003-01346, IP2-2003-01145, IP3-2003-00937, IP2-2003-01350, IP2-2003-01349, and IP2-2003-01347

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS) and Public Safety (PS)

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspector reviewed radiological work activities and practices and procedural implementation during tours and observations of the facilities. Additionally, the inspector reviewed procedures, records, and other program documents to evaluate the effectiveness of access controls to radiologically significant areas. Further, the inspector observed activities at the routine radiologically-controlled-area (RCA) access control point (HP-1) on several occasions to verify compliance with requirements for RCA entry and exit, dosimetry placement, and issuance and use of electronic dosimeters.

On February 24, 2003, the inspector discussed the status of the radiation protection organization and procedures for Units 2 and 3 with the radiation protection manager (RPM). The inspector also discussed the radiological work activities and dose projections for the upcoming week for each unit.

On February 25, 2003, the inspector reviewed the radiation work permit (No. 032028) used for non-outage containment entries at Unit 2. Also, the inspector discussed with the RPM the methods for the segregation of high level and low level dry active radioactive waste in work areas during outages with radioactive waste personnel and radiation protection technicians from both units. In addition, the inspector discussed survey frequencies and posting and their use for briefings of radiation workers with a Unit 2 radiation protection supervisor.

On February 26, 2003, the inspector toured and observed work activities in selected portions of the fuel handling building and 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. Also, the inspector toured and observed work activities in various elevations in the primary auxiliary building in Unit 2. During these tours and observations, the inspector reviewed, for regulatory compliance, the performance of the radiation workers and radiation protection technicians and the posting, labeling, barricading, and level of radiological access control for locked high radiation areas (LHRAs), high radiation areas (HRAs), radiation and contamination areas, and radioactive material areas. The inspector reviewed and observed work activities for compliance with the radiation work permit (RWP) requirements.

The inspector performed a selective examination of procedures, records, and other program documents (reference the List of Documents Reviewed Attachment) to evaluate the adequacy of radiological controls. This 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

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

On February 24, 2003, the inspector discussed the Unit 2 and 3 cumulative dose results for 2002, and the cumulative dose estimates for 2003, with the technical support manager. Further, the inspector discussed the integration of the ALARA planning process into the station's work planning and control process with an ALARA radiological engineer.

On February 26, 2003, the inspector examined the pre-job ALARA review for 2R15 RWP No. 025226, involving outage valve work and the in-progress and post-job ALARA reviews for the same job, to assess the effectiveness of the radiological controls. The inspector also reviewed the post-job ALARA review for 2R15 RWP No. 025206, involving outage radiological waste support.

The inspector performed a selective examination of procedures, records, and documents (reference the List of Documents Reviewed Attachment) for regulatory compliance and for adequacy of control of radiation exposure. This 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

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

During the plant tour on February 26, 2003, 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; installed radiation monitors; whole body friskers; portal monitors; area monitors, and process monitors. The inspector conducted a review of the instruments observed in the toured areas, specifically for verification of current calibrations, appropriate source checks, and proper function.

The inspector performed a selective examination of documents (reference the List of Documents Reviewed Attachment) for regulatory compliance and adequacy. This 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.

2PS1 Gaseous and Liquid Effluents

a. Inspection Scope

The inspector reviewed the following documents to evaluate the effectiveness of the licensee's radioactive gaseous and liquid effluent control programs. The requirements for radioactive effluent controls are specified in the Technical Specifications and the Offsite Dose Calculation Manual (TS/ODCM):

- the 2001 Radiological Annual Effluent Release Report, including projected public dose assessments;
- review of the current ODCM (Revision 6, October 28, 1999), including technical justifications for Revision 7;
- selected 2002 analytical results for charcoal cartridge, particulate filter, and noble gas samples;
- implementation of the compensatory sampling and analysis program when the effluent radiation monitoring system (RMS) was out of service;
- implementation of IE Bulletin 80-10;
- selected 2002 radioactive liquid and gaseous release permits;
- associated effluent control procedures, including analytical laboratory procedures;
- calibration results for chemistry laboratory measurement equipment (gamma and liquid scintillation counters);

- implementation of the measurement laboratory quality control program, including effluent intra-laboratory and inter-laboratory comparisons and control charts;
- the 2001 and 2002 NQA Audits (Audit Nos. 01-AR-21-RP, August 1-8, 2001 and 02-AR-14-RP, April 11-17, 2002) of the implementation of the radioactive liquid and gaseous effluent control program and the ODCM;
- Radiation Monitoring System Reliability Plan;
- the most recent Channel Calibration and Channel Functional Test results for the radioactive liquid and gaseous effluent radiation monitoring system (RMS) and its flow measurement devices as listed in Tables 4.10-2 and 4.10-4 of the Technical Specifications (TS):

  Unit 1 RMS:

  - Service/River Water Liquid Radiation Monitor (R-51);
  - Liquid Discharge Radiation Monitor (R-54);
  - Secondary Boiler Blowdown Effluent Line (R-52);
  - Sphere Foundation Sump Discharge Monitor (R-62); and
  - Stack Vent Noble Gas Monitor (R-60).

  Unit 2 RMS:

  - Waste Disposal Liquid Effluent Line (R-48);
  - Component Cooling Water Radiation Monitor (R-47);
  - Steam Generator Blowdown Effluent Line (R-49);
  - Service Water System Effluent Line Monitors (R-46/53);
  - Component Cooling Service Water Heat Exchangers (R-39/40);
  - Plant Vent Noble Gas Monitors (R-44 and R-27); and
  - Large Gas Decay Holding Tank Monitor (R-50).

Units 1 & 2 Flow Rate Measurement Devices:

- Stack Vent Flow Rate Monitor;
- Plant Vent Flow Rate Monitor;
- Liquid Effluent Line Flow Rate Monitor; and
- Steam Generator Blowdown Effluent Line.

- the most recent surveillance testing results for the following air treatment systems listed in the following TS:

  - TS 4.5.D Containment Fan Cooler System (air flow tests for five Fan Cooler Units);
  - TS 4.5.E Control Room Air Filtration System (system flow rate, laboratory test in accordance with ASTM D3803-1989, in-place testings for HEPA and charcoal filters, pressure drop test, visual inspection);
  - TS 4.5.F Fuel Storage Building Air Filtration System (system flow rate, laboratory test in accordance with ASTM D3803-1989, in-place testings for HEPA and charcoal filters, pressure drop test, visual inspection); and
  - TS 4.5.G Post-Accident Containment Venting System (system flow rate, laboratory test in accordance with ASTM D3803-1989, in-place testings for HEPA and charcoal filters, pressure drop test, visual inspection).

The inspector also toured and observed the following activities to evaluate the effectiveness of the licensee’s radioactive gaseous and liquid effluent control programs:

- availability of radioactive liquid/gaseous effluent RMS to determine the equipment material condition; and
- operability of air cleaning systems to determine the equipment material condition.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

The inspection consisted of a review of Condition Report IP2-2003-00771, identified in the Corrective Action Program (CAP), for the appropriateness and adequacy of event categorization, immediate corrective action, corrective action to prevent recurrence, and timeliness of corrective action.

b. Findings

Introduction. A Green self-revealing non-cited violation (NCV) of 10 CFR 71.12, which addresses general licenses for NRC-approved packages, was identified for failure to comply with the package procedures relating to the use and maintenance of the packaging.

Description. Condition Report IP2-2003-00771 (dated February 6, 2003), documented that a shipment of filters, packaged in a CNS 8-120 B cask (an NRC-approved Type B shipping container), left Indian Point 2 on February 3, 2003, and arrived at a consolidation facility in South Carolina on February 5, 2003. During the unloading process, facility personnel discovered that a bolt, that was to be installed in the primary lid's air pressure test port, in accordance with procedure TR-OP-035, "Handling Procedure for CNS 8-120 B," was missing. The bolt had not been installed during the re-assembly of the cask for shipment.

Analysis. This finding constituted a performance deficiency in that it resulted in a requirement not being met that was reasonably within Entergy's ability to foresee and correct and that should have been prevented. This finding was more than minor in that the finding was associated with the Public Radiation Safety Cornerstone's attribute of procedures for transportation packages. The finding affected the associated cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain. Specifically, a procedure/document, relating to the preparation, use, and maintenance of an NRC-approved Type B packaging was not properly implemented, and the package was shipped on a public highway. Following the Public Safety Significance Determination Process, the issue was determined to be of very low safety significance in that the finding did not involve exceeding transportation radiation limits, there was no breach of the package during transit, and it was a Certificate of Compliance (CoC) maintenance/use performance deficiency.

Enforcement. 10 CFR 71.12(a) and (c), together with paragraphs 9(i) and 9(iii) of the NRC Certificate of Compliance No. 9168 and procedure TR-OP-035, "Handling Procedure for CNS 8-120 B," require that a bolt shall be installed and torqued into place in the primary lid's air pressure test port prior to shipment. Contrary to these requirements, on February 3, 2003, a bolt was not installed and torqued into place in the primary lid's air pressure test port prior to shipment.

Entergy actions associated with this finding are documented CR IP2-2003-00771. Because this self-revealing violation was of very low safety significance and because Entergy entered these issues into its corrective action program, this violation is being treated as a self-revealing NCV, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 50-247/03-03-06)**

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

The inspector reviewed the licensee's performance indicator (PI) data collecting and reporting process as described in procedure SAO-114, "Preparation of NRC and WANO Performance Indicators." The purpose of the review was to determine whether the methods for reporting PI data are 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 Reactor Coolant System Leakage

a. Inspection Scope

The inspector reviewed the performance indicator (PI) for reactor coolant system (RCS) leakage for the period from January - December 2002. This PI remained in the Green band. The inspector reviewed the completed SOP 1.7 RCS leak rate surveillance determinations to verify the adequacy of the reported PI data. The inspector observed licensee performance of the leak rate surveillance on February 12, 2002. The licensee's corrective action program records were also reviewed to determine if any problems with the collection of PI RCS leakage data had occurred. The inspectors compared the PI data against the guidance contained in NEI 99-02.

b. Findings

No findings of significance were identified.

.2 Scrams Per 7,000 Critical Hours

a. Inspection Scope

The inspector reviewed Entergy's PI data for Unplanned Scrams Per 7,000 Critical Hours to verify whether the PI data was accurate and complete. The PI remained in the green band for the four quarters of 2002. The inspector reviewed operator logs, licensee event reports, and monthly operating reports to compare PI data reported by the licensee. The inspectors compared the PI data against the guidance contained in NEI 99-02.

b. Findings

No findings of significance were identified.

.3 Residual Heat Removal System Unavailability

a. Inspection Scope

The inspector reviewed Entergy's Performance Indicator (PI) data for Residual Heat Removal (RHR) Safety System Unavailability to verify whether the PI data was accurate and complete. The inspectors compared the PI data reported by the licensee to information gathered from the control room logs, condition reports, and work orders for the four quarters of 2002. In addition, the inspectors interviewed the system engineers. The inspectors compared the PI data against the guidance contained in NEI 99-02.

b. Findings

No findings of significance were identified.

.4 Emergency AC Power System Unavailability

a. Inspection Scope

The inspector performed a review of the 2002 quarterly performance indicator data submitted by the licensee for the safety system unavailability of the emergency AC power system (emergency diesel generators) to determine its accuracy and completeness. The inspector researched the control room operating logs and the condition reporting system to identify when the emergency diesel generators were out of service during the period of review. The control room operating logs were also reviewed to determine the number of hours the EDGs were required to be operational. The inspector used the guidance provided in NEI Report 99-02 to calculate the ratio of the number of hours the emergency AC power system was unavailable to the number of hours the emergency AC power system was required.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Baseline Procedure Problem Identification and Resolution Review

a. Inspection Scope

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. IP2-2002-10106 and -10795 and 2003-00328, -00400, -00401, -00751, -00771, -00826, 2001-09250, 2001-09604, 2001-12738, 2001-11035, 2001-02811, 2001-11958, 2001-11056, 2002-11130, 2001-07475, 2001-11614, 2002-01208, 2002-02184, 2002-08402, 2002-11130, 2001-09250, 2001-09604, 2001-12738, 2001-11035, 2001-02811, 2001-11958, 2001-11056, 2002-11130, 2001-07475, 2001-11614, 2002-01208, 2002-02184, 2002-08402, 2002-00504, 2002-03729,2002-

03245,2002-10605, 2002-05578, 2002-50779, 2002-04353, 2002-05578, 2002-07617, 2002-10071, and 2002-10605

b. Findings

No findings of significance were identified.

.2 Adverse Trend in Residual Heat Removal Boron Concentration, IP2-2002-03035

a. Inspection Scope

The inspector selected CR No. IP2-2002-03035 for detailed review. The condition report was associated with an adverse trend (decreasing) in residual heat removal (RHR) boron concentration. The CR was reviewed to ensure a complete and accurate identification of the issue, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized. The inspector evaluated the report against the requirements of the licensee's corrective action process (CAP) as delineated in Site Administrative Procedure ENN-LI-1102, Revision 2, "Corrective Action Process."

b. Findings

There were no findings associated with the sample CR reviewed; however, the inspector found that the licensee had not confirmed that the adverse trend was corrected. The licensee performed an apparent cause evaluation and concluded that the most likely source of the leakage into the RHR system and resulting boron dilution was occurring through valves 730 and 731. The apparent cause evaluation was performed through a review of past valve leakage history in the RHR system. Based on this determination, preventive and corrective maintenance was performed on these valves and the CR was closed. The sample for boron analysis is taken from the system when the RHR pumps are run during their scheduled surveillance test to assure thorough mixing of the water within the system. The inspector determined that the adequacy of the apparent cause determination and the success of corrective actions cannot be confirmed until subsequent quarterly pump testing is completed. Since the apparent cause evaluation appeared to be reasonable and the corrective actions were timely, relative to the significance of the condition, no violation of regulatory requirements was identified.

.3 Loss of Security Perimeter Lighting, IP2-2002-11628 and 2002-11629

a. Inspection Scope

The inspector selected CR Nos. IP2-2002-11628 and -11629 for a detailed review. The CRs were associated with a loss of perimeter lighting on December 27, 2002. The inspector observed that Entergy's short-term compensatory actions were consistent with the Security Contingency Plan, observed short-term corrective actions to repair and restore the lighting, and confirmed long-term work orders existed to improve the reliability and material condition of the perimeter lighting.

b. Findings

No findings of significance were identified.

.4 Communication failures between the Emergency Operations Facility computer system and the field sirens, IP2-2003-01107

a. Inspection Scope

The inspectors selected CR No. IP2-2003-01107 for detailed review. The CR was associated with communication failures between the emergency operations facility (EOF) computer system and the field sirens that alert and notify the public in case of an event at Indian Point. The report was reviewed to ensure that the full extent of the issue was identified, an appropriate evaluation was performed, and that appropriate corrective actions were specified and prioritized. The inspectors evaluated this CR against the requirements of 10 CFR 50.47, "Emergency Plans," and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."

b. Findings

No findings of significance were identified.

.5 Fundamental Improvement Plan Effectiveness Reviews

a. Background and Scope

In January 2002, Entergy provided the NRC with a copy of the fundamental improvement plan (FIP) consistent with the NRC's action matrix for a multiple degraded cornerstone facility. The fundamental improvement plan documented corrective action plans and effectiveness reviews associated with five key areas involving human performance, design control/licensing basis, equipment performance/work management, problem identification and resolution, and licensed operator performance. By letter dated August 28, 2002, the NRC removed Indian Point Unit 2 from the repetitive multiple degraded cornerstone column of the action matrix. Entergy continued to implement actions associated with the FIP until the end of calendar year 2002.

The inspection scope was to review the eight effectiveness reviews conducted within the FIP. The effectiveness reviews included: management observation of work activities; operator burden and work-down curve for temporary alterations; review of the design basis initiative project; optimization of the preventive maintenance program; corrective actions in monitoring the work control process; equipment reliability actions; work management self-assessments; and corrective action effectiveness reviews for condition reports.

The inspector reviewed the FIP expectation, reviewed the completed effectiveness evaluations, and discussed the evaluations with individuals assigned to perform the reviews. The inspector compared the results and conclusions of the individual effectiveness review with applicable performance metrics maintained by Entergy and using past applicable NRC assessments in performance (inspection report findings and ROP-3 end-of-cycle assessments).

b. Findings

No findings of significance were identified. The inspector concluded that selected effectiveness reviews were not consistently self-critical. For example, the effectiveness review for the design basis initiative project primarily focused on the quality of action plans and efficiencies of plan implementation instead of the quality of the output products and the amount of use of this information by design and system engineering staffs. The NRC's evaluation also found that the work control effectiveness reviews did not explore the quality of post-work tests, but rather highlighted processing problems associated with post-work testing.

The NRC noted that the 2003 Indian Point Business plan provided adequate actions to support improvements in the five key areas from the FIP. Further, self-assessments are currently part of the business plan and areas within the plan focus on departmental self-assessments and monitoring the quality of those assessments through a self-assessment review board.

4OA3 Event Follow-up

.1 Licensee Event Report (LER) 2002-002-00 "Restoration of Previously Isolated Portion of Weld Channel and Containment Penetration Pressurization System,"

(Closed) Licensee Event Report (LER) 2002-002-00, "Restoration of Previously Isolated Portion of Weld Channel and Containment Penetration Pressurization System," dated August 7, 2002. On June 8, 2002, the licensee confirmed through testing that a previously retired (because of supposed leakage) section of Zone W-11 of the weld channel and containment penetration pressurization system (WCCPPS) was leak-tight and should be restored to an operable status. The licensee concluded that the cause of the premature retirement of this section of Zone W-11 was an inadequate leak test of the subject weld channel and a poor assumption that leakage was in a section of the weld channel embedded in concrete and, therefore, not capable of being repaired.

In addition to the broad corrective action taken under the FIP, dated January 25, 2002, to address the human performance weaknesses which contributed to this event, the operations manager deleted the operations department troubleshooting procedure (DAD-40) in favor of the work control department's procedure governing troubleshooting and repairs. The work control department procedure implements a more thorough planning, review, approval, and closeout process which ensures a higher probability of a satisfactory troubleshooting result. This LER is closed.

4OA5 Review of Institute of Nuclear Power Operations (INPO) Evaluation Report

The inspectors reviewed the final report of an INPO Evaluation conducted in February 2002. The inspectors identified no new findings of significance.

4OA6 Meetings, Including Exit

The inspectors met with Indian Point 2 representatives at the conclusion of the inspection on April 9, 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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy:

W. Axelson	Support Supervisor
M. Dampf	Health Physics Manager
R. Deschamps	Radiological Protection Superintendent
R. Decensi	Technical Support Manager
R. Fucheck	HP Supervisor
D. Gately	Radiation Protection Coordinator
R. LaVera	Radiological Engineer
R. Majes	Radiological Engineer
R. Richards	HP Supervisor
R. Rodino	Radiological Engineer
W. Scholtens	Waste Management Contractor
R. Solanto	HP Supervisor
J. Stewart	HP Supervisor
R. Tagliomonte	Waste Management Supervisor
N. Azevedo	ISI Supervisor
W. Axelson	Support Supervisor
T. Burns	Environmental Supervisor
J. Comiotes	Director, Nuclear Safety Assessment
L. Cortopassi	IP3 Training Manager
F. Dacimo	Vice-President
M. Dampf	Health Physics Manager
S. Davis	IP2 Licence Operator Requalification Training Supervisor
R. Decensi	Technical Support Manager
R. Deschamps	Radiological Protection Superintendent
J. DeRoy	General Manager Plant Operations, IP3
K. Finucan	Emergency Planning Staff
K. Finvean	Reactor Vessel Head Inspection, Assistant Project Manager
R. Fucheck	HP Supervisor
D. Gately	Radiation Protection Coordinator
M. Gillman	IP3 Operations Manager
L. Glander	Dosimetry Supervisor
J. Goebel	Reactor Vessel Head Inspection, Project Manager
F. Inzirillo	Manager Emergency Planning
T. Jones	Nuclear Safety and Licensing
R. LaVera	Radiological Engineer
R. Majes	Radiological Engineer
J. McCann	Nuclear Safety and Licensing Manager
F. Mitchell	HP Supervisor
D. Pace	VP - Engineering - ENN
J. Perrotta	Quality Assurance Manager

R. Penny	Manager, Engineering Programs
R. Richards	HP Supervisor
K. Richett	HP Technician
R. Rodino	Radiological Engineer
R. Sachatello	Radiological Consultant
C. Schwarz	General Manager, IP2 Plant Operations
G. Schwartz	Chief Engineer
H. Salmon	Quality Assurance Director
M. Smith	Director of IP3 Engineering
R. Solanto	HP Supervisor
S. Stevens	HP Technician
J. Stewart	HP Supervisor
D. Sullivan-Weaver	Emergency Planning Staff
J. Tuohy	Design Engineering Manager
J. Wheeler	Site Training Manager
F. Wilson	Superintendent, Operations Training
M. Wilson	Emergency Planning Staff

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

LER 50-247/ 2002-002-00	Restoration of previously isolated portion of weld channel and containment penetration pressurization system

Open/Closed

50-247/03-03-02	NCV	Ineffective corrective actions associated with the 23 EDG load swings between May 2000 and February 2003
50-247/03-03-03	NCV	Improper emergent work package instructions for 22 steam generator level bistable replacement
50-247/03-03-05	NCV	Post-work test inadequate for 22 boric acid transfer pump boric acid filter stop valve
50-247/03-03-06	NCV	Failure to comply with packaging procedures

Opened

50-247/03-03-01	URI	Lack of cable separation in fire areas F and J, postulated fire compromising associated circuits.
50-247/03-03-04	URI	Electrical calculation reconstitution to support offsite power design basis (SAT load tap changer).

LIST OF DOCUMENTS REVIEWED

Sections 1EP4, Emergency Action Level and Emergency Plan Changes

Emergency Plan for Indian Point Unit Nos 1 and 2, Rev 01-02a
Indian Point 3 Emergency Plan, Rev 46
Indian Point Energy Center Emergency Plan, Rev 02-01
IP-EP-251, Alternate Emergency Operations Facility, Rev 0
IP-EP-255, Emergency Operations Facility Management and Liaisons, Rev 1, 2
IP-EP-310, Dose Assessment, Rev 0
IP-EP-410, Protective Action Recommendations, Rev 0
IP-EP-510, Meteorological, Radiological & Plant Data Acquisition System, Rev 0
IP-EP-520, Modular Emergency Assessment & Notification System (MEANS), Rev 0
IP-EP-610, Emergency Termination and Recovery, Rev 0
IP-EP-620, Estimation of Total Population Exposure, Rev 0
IC/EALs, Initiating Conditions & Emergency Action Levels, Rev 9 (IP3)
IP-1001, Determining the Magnitude of Release, Rev 17, Void (IP3)
IP-1002, Emergency Notification and Communication, Rev 27, 28 (IP2)
IP-1003, Obtaining Meteorological Data, Rev 18, Void (IP3)
IP-1004, MIDAS Computer System, Rev 16, Void (IP3)
IP-1010, Central Control Room, Rev 6, 7, 8 (IP2)
IP-1013, Protective Action Recommendations, Rev 8, (IP2)
IP-1015, Radiological Monitoring Outside the Protected Area, Rev 10 (IP2)
IP-1017, Protective Action Recommendations for the Offsite Population, Void, Rev 13 (IP3)
IP-1019, Coordination of Corporate Response, Rev 11 (IP2)
IP-1021, Manual Update, Readout & Printout of Proteus Plant Parameter Data, Canceled (IP2)
IP-1023, Operation Support Center, Rev 19 (IP2)
IP-1026, Emergency Data Display, Rev 1 (IP2)
IP-1027, Personnel Accountability Rev 17 (IP2)
IP-1027, Emergency Personnel Exposure, Rev 13, (IP3)
IP-1030, Emergency Operations Facility, Rev 8 (IP2)
IP-1030 Emergency Operations Facility, Rev 7 (IP2)
IP-1035, Technical Support Center, Rev 17 (IP2)
IP-1050, Security, Rev 4 (IP2)
IP-1050, Accountability, Rev 28 (IP3)
IP-1054, Search and Rescue Teams, Rev 11 (IP3)
IP-2001, ED, POM, Shift Managers Procedure, Rev 16, 17 (IP3)
IP-2003 CR Watch Chemist, Rev 6 (IP3)
IP-2200, Emergency Activation of the Operations Support Center, Rev 7 (IP3)
IP-2201 OSC Manager, Rev 9 (IP3)
IP-2204, OSC Team Leader, Rev 3 (IP3)
IP-2209, OSC HP Technician, Rev 5
IP-2301, Emergency Director, Void (IP3)
IP-2302, EOF Technical Advisor & Information Liaison, Rev 10 (IP3)
IP-2310, EOF Onsite Radiological Communicator, Rev 4 (IP3)

IP-2400, Emergency Activation of the Alternate Emergency Operations Facility, Void (IP3)
IP-2500, Security Emergency Activation Responsibilities, Rev 12 (IP3)
IP-2600, Emergency Termination & Transition to Recovery, Rev 4, Void (IP3)
IP-2601, Recovery Manager, Rev 1, Void (IP3)
IP-2602, Development of Recovery Plan, Rev 1, Void (IP3)
IP-2603, Recovery Support Group Manager, Rev 1
10CFR50.54(q) review for IPEC Emergency Plan
10CFR50.54(q) review for IP-EP-410
EP-AD-02, Emergency Planning Controlled Documents, Rev 3
EP-AD-03, Emergency Response Organization Training Program, Rev 1
Emergency Response Training Program Curriculum, Rev 16
TNG-AD-18, Emergency Response Training Program, Rev 8
QA-AD-3, IP3 Audit Program, Rev 5

Section 2OS1, Access Control to Radiologically Significant Areas

- SAO-302, Rev. 18, Radiation work permits (RWP) program
- HP-SQ-3.002, Rev. 16, Equipment and materials release requirements
- RW-SQ-4.109, Rev. 10, Radioactive material storage
- RWP 032009, Rev. 00, Assessments in RCA
- RWP 032028, Rev. 00, Non-outage vapor containment-all groups
- Radiation Protection performance goals for 2003
- Unit 2 housekeeping and area decon plan, February 25, 2003
- Quality assurance assessment report no. 02-AR-33-RP, November 11 - 15, 2002
- Continuing training - 2003, Session 1, Radiation Protection Alignment Session

Section 2OS2, ALARA Planning and Controls

- SAO-303, Rev. 11, ALARA program
- SAO-305, Rev. 10, Station ALARA committee
- IP1 and 2 Daily ALARA information for week of 02-16-2003
- IP1 and 2 Weekly exposure trend for 2003
- Pre-job, In-progress, and Post-job ALARA Reviews No. 02-013 (RWP 025226) for Outage valve work for 2R15
- Post-job review (ALARA review 02-013) for outage radioactive waste support (RWP 025206) for 2R15
- Indian Point Energy Center/Radiation protection/2R15 outage ALARA review
- Indian Point Energy Center/Radiation protection/Strategic plan for exposure reduction, 2003 - 2008
- IPEC ALARA committee meeting presentation handout for January 28, 2003

Section 2OS3, Radiation Monitoring Instrumentation and Protective Equipment

- Entergy South automated contamination monitor configuration, December 3, 2002

Section 2PS2, Radioactive Material Processing and Transportation

- RW-SQ-4.303, Rev. 14, Shipping cask handling procedure
- Procedures, license, and safety analysis report for the CNS 8-120B Type B radioactive waste shipping cask, USA/9168/B(U)

Condition Reports Generated During this Inspection

IP3-2003-00480	Addresses minor issues associated with the development of the new plan. Items include the handling of the lead accountability officer function, description of the Safety Team Lead position, inclusion of core exit thermal couples as instrumentation used, specifying a two hour joint news center activation goal, distribution and updating of EALs to State and County locations, clarification of drill applicability for the two units, and removal of extraneous information from Table B-1.
IP3-2003-00493	Addresses review of Table B-1 staffing for a dual unit site with regards to Technical Specification staffing requirements.
IP3-2003-00457	Addresses need to update emergency plan regarding KI when decision is finalized by the State.
IP2-2003-01515	Scaffolding in the pipe penetration area near the containment pressure transmitters not meeting station expectations.
IP2-2003-01520	Discrepancy between drawing and field conditions for 21 containment spray pump discharge header drain
IP2-2002-09231	Pre-fire plan sketch for Fire Zone 90A in error
IP2-2003-00567	Fire Zone 6A discrepancies noted
IP2-2003-01673	Failure to incorporate a drawing revision to a non-operations critical drawing
IP2 2003-01409	Unavailability time incorrect for the 22 CCW pump maintenance rule
IP2 2003-01037	Valves labels in auxiliary feedwater room do not match the check off list
IP2 2003-01161	Valve labels in the emergency diesel generator room do not match the check off list

Condition Reports

IP2-2002-09152, IP2-2002-09054, IP2-2002-06818, IP2-2002-04701, IP2-2001-08308, IP2-2001-05461, IP2-2001-02536,

LIST OF BASELINE INSPECTIONS PERFORMED

71111.04	Equipment Alignment	1R04
71111.05	Fire Protection	1R05
71111.06	Flood Protection 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
71111.17	Permanent Modifications	1R17
71111.19	Post Maintenance Testing	1R19
71111.22	Surveillance Testing	1R22
71111.23	Temporary Plant Modifications	1R23
71114.04	Emergency Action Level and Emergency Plan Changes	1EP4
71114.06	Emergency Planning Drills	1EP6
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 USED

AFWP	auxiliary feedwater pump
ALARA	as low as reasonably achievable
AOI	abnormal operating instruction
BATP	boric acid transfer pump
CAP	corrective action program
CCR	central control room
CCW	component cooling water
CFR	Code of Federal Regulations
CNS	chem nuclear systems
COC	certificate of compliance
COL	check off list
CR	condition report
DBI	design basis initiative
DCP	design change package
EAL	emergency action level
EDG	emergency diesel generator
EOF	emergency operations facility

EP	emergency preparedness
ESFS	Engineered Safeguards Features System
FIP	fundamental improvement plan
HRA	high radiation area
ICMs	Interim Compensatory Measures
INPO	Institute of Nuclear Power Operations
IP	Indian Point
IP2	Indian Point Unit 2
IPEC	Indian Point Energy Center
IPEEE	individual plant examination for external events
ISI	inservice inspection
JNC	joint news center
kV	kilo-volt
Kw	kilo-watt
LER	licensee event report
LTC	load tap changer
MOP	motor operated potentiometer
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
ODCM	offsite dose calculation manual
OSC	operations support center
OS	occupational safety
PI	performance indicator
PM	post maintenance
PS	public radiation safety
PT	penetrant testing
PWT	post-work test
RCA	radiologically controlled area
RCS	reactor coolant system
RHR	residual heat removal
RMS	radiation monitoring system
RPM	radiation protection manager
RPS	reactor protection system
RSPS	risk significant planning standard
RV	reactor vessel
RWP	radiation work permit
SAO	station administrative order
SAT	station auxiliary transformer
SDP	significance determination process
SI	safety injection
SOP	system operating procedure
TA	temporary alteration
TI	temporary instruction
TSC	technical support center
TS	technical specifications

UFSAR	Updated Final Safety Analysis Report
V	volt
WCCPPS	weld channel and containment penetration pressurization system
WO	work order