July 25, 2002

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 INSPECTION REPORT 50-247/02-04

Dear Mr. Dacimo:

On June 29, 2002, the NRC completed an inspection at the Indian Point 2 Nuclear Power Plant. The enclosed report presents the results of that inspection. The results were discussed on July 3, 2002, with members of your staff.

The inspection was an examination of 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 inspection also reviewed the security program and recent emergency plan document changes. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel. Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green).

The NRC has increased security requirements at Indian Point 2 Nuclear Power Plant in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to monitor overall security controls and will issue temporary instructions in the near future to verify by inspection the licensee's compliance with the Order and current security regulations.

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,

Brian E. Holian, Deputy Director
Division of Reactor Projects

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

Enclosure: Inspection Report 50-247/02-04

Attachment 1 - Supplemental Information

cc w/encl: J. Yelverton, Chief Executive Officer
M. R. Kansler, Senior Vice President and Chief Operating Officer
J. Herron, Senior Vice President
R. J. Barrett, Vice President - Operations
L. Temple, General Manager - Operations
D Pace, Vice President - Engineering
J. Knubel, Vice President Operations Support
J. McCann, Manager, Nuclear Safety and Licensing
J. Kelly, Director of Licensing
C. Faison, Manager - Licensing, Entergy Nuclear Operations, Inc.
H. Salmon, Jr., Director of Oversight, Entergy Nuclear Operations, Inc.
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.
Mayor, Village of Buchanan
R. Albanese, Executive Chair, Four County Nuclear Safety Committee
S. Lousteau, Treasury Department, Entergy Services, Inc.
M. Slobodien, Director Emergency Programs
B. Brandenburg, Assistant General Counsel
P. Rubin, Operations Manager

Assemblywoman Sandra Galef, NYS Assembly
County Clerk, Westchester County Legislature
A. Spano, Westchester County Executive
R. Bondi, Putnam County Executive
C. Vanderhoef, Rockland County Executive
E. A. Diana, Orange County Executive
T. Judson, Central NY Citizens Awareness Network
M. Elie, Citizens Awareness Network
D. Lochbaum, Nuclear Safety Engineer, Union of Concerned Scientists
Public Citizen's Critical Mass Energy Project
M. Mariotte, Nuclear Information & Resources Service
F. Zalcman, Pace Law School, Energy Project
L. Puglisi, Supervisor, Town of Cortlandt
Congresswoman Sue W. Kelly
Congressman Ben Gilman
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, Executive Director, Westchester Peoples Action Coalition
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
S. Richards, NRR (ridsnrrdlpmlpdi)
P. Eselgroth, DRP
P. Milano, PM, NRR
G. Vissing, PM, NRR (Backup)
S. Barber, DRP
W. Cook, DRP
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No. 50-247

License No. DPR-26

Report No. 50-247/02-04

Licensee: Entergy Nuclear Operations, Inc..

Facility: Indian Point 2 Nuclear Power Plant

Location: Buchanan, New York 10511

Dates: May 12 - June 29, 2002

Inspectors: Peter Habighorst, Senior Resident Inspector
Lois James, Resident Inspector
William Raymond, Senior Resident Inspector, Pilgrim
Greg Smith, Security Specialist
Anthony Dimitriadis, Security Specialist
Todd Fish, Operations Engineer
David Silk, Senior Emergency Preparedness Inspector

Approved by: Peter W. Eselgroth, Chief
Projects Branch 2
Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000247-02-04, on 5/12 - 6/29/2002, Entergy Nuclear Operations, Inc.; Indian Point 2 Nuclear Power Plant. Barrier Integrity and Mitigating Systems.

The report covered a seven week period of inspection by resident and region-based inspectors. No findings of significance were identified. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/reactors/operating/oversight.html.

Cornerstone: Barrier Integrity

Green. On May 27, 2002, during surveillance testing of the safety injection discharge motor-operated valve (851B), the valve failed to stroke closed. The initial operability evaluation did not consider the non-automatic containment isolation function for this valve. This event was documented in condition report No. 200205433. The performance issue associated with this finding is a weakness in operator knowledge of multi-function safety system components. This is the second recent example where operators did not consider this function for a safety-related valve. The first example was documented in NRC report 50-247/2002-003, section 1R15. The untimely and incomplete operability assessment for safety injection discharge valve 851B has very low safety significance since the containment isolation valve was restored to an operable status prior to exceeding Technical Specification 3.6.A.3.a.2.d limiting condition for operation.

Cornerstone: Mitigating Systems

Green. On May 17, 2002, multiple grounds on the protective circuit for Unit 1 substation 102NS3 resulted in a loss of the 13.8 kilovolt (kv) lighting and power bus section 3. The consequence of this event was a loss of alternate safe shutdown power to all major alternate safe shutdown pumps and selected instrumentation. At the time, the Unit 2 normal and emergency electrical power supplies were available to supply power to the above stated mitigation equipment and instrumentation. The licensee repaired and restored the 13.8 kv bus section 3 within 30 hours of the fault. The performance issue is inadequate retirement of protective circuits for 440 volt substations (132PC3 and 142PC3) that could impact availability of alternate safe shutdown power supplies. This issue is more than minor since unavailability of alternate safe shutdown equipment for 30 hours is viewed as a precursor to a significant event and the alternate safe shutdown power supplies are a risk-significant maintenance rule system which was unavailable for greater than 24 hours.

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Report Details

SUMMARY OF PLANT STATUS

The plant operated at full power during the inspection period.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness)

1R04 Equipment Alignment

Partial System Walkdowns

a. Inspection Scope (71111.04)

On May 14, 2002, the inspector performed a partial system walkdown of the 21 and 22 trains of the auxiliary feedwater system, when the 23 auxiliary feedwater (AFW) pump was out of service for surveillance testing and predictive maintenance. The inspector observed the conditions in the auxiliary feedwater building to verify no discrepancies would impact the AFW system function. The inspector also reviewed the status of key AFW system components based on check off list (COL) 21.3, “Steam Generator Water Level and Auxiliary Boiler Feedwater," revision 22. The inspector observed the physical condition of the 21 and 22 AFW pumps, reviewed the operations logs, and discussed performance issues with an operator.

On June 20, 2002, the inspector performed a partial system walkdown of the 21 and 23 trains of the charging system. The review was performed while the 22 charging pump was isolated and tagged out (reference tagout 16488) for corrective maintenance. The inspector reviewed the status of equipment alignment based upon COL 3.1, Chemical and Volume Control System,” revision 33, and plant drawing nos. 9321-F-2736-114, “Flow Diagram Chemical and Volume Control System, sheet 1,” and A208168-51, Flow Diagram Chemical and Volume Control System, sheet 2.”

b. Findings

No significant findings were identified.

1R05 Fire Protection

.1 Fire Protection Tours

a. Inspection Scope (71111.05)

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 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 74A, Electrical Penetration Area
- Fire Zone 39A, Turbine Building 33 foot elevation
- Fire Zone 11, Cable Spreading Room

Reference material consulted by the inspector included the Fire Protection Implementation Plan, Pre-Fire Plan, and Station Administrative Orders (SAOs)-700, "Fire Protection and Prevention Policy," SAO-701, "Control of Combustibles and Transient Fire Load," SAO-703, "Fire Protection Impairment Criteria and Surveillance," and Calculation PGI-00433, "Combustible Loading Calculation." The inspector reviewed licensee corrective actions to address fire protection program deficiencies for the above stated fire zones.

b. Findings

No significant findings were identified.

.2 Fire Drills

a. Inspection Scope (71111.05)

On June 20, 2002, the inspector observed an announced fire brigade drill. The drill was in accordance with the pre-planned drill scenario for a 23 emergency diesel generator (EDG) control panel fire. This fire brigade drill was a training drill for a new fire brigade leader with the shift fire brigade leader acting in an advisory role. The purpose of this observation was to evaluate the readiness of the licensee's personnel to prevent and fight fires. The inspector evaluated the following aspects:

- Protective clothing/turnout gear is properly donned.
- Self-contained breathing apparatus (SCBA) equipment is properly worn and used.
- Fire hose lines are capable of reaching all necessary fire hazard locations, are laid out without flow constrictions, and are simulated being charged with water.
- Fire area is entered in a controlled manner.
- Sufficient fire fighting equipment is brought to the scene by the fire brigade.
- Effective smoke removal operations are simulated.
- The fire fighting pre-plan strategies are utilized.
- The licensee's pre-planned drill scenario is followed.
- The drill objectives' acceptance criteria are met.

Minor deficiencies that did not impact the ability to fight a fire were noted during the drill critique and were entered into the Condition Reporting System (CR Nos. 200206616 and 200206187.

b. Findings

No significant findings were identified.

1R06 Flood Protection Measures

a. Inspection Scope (71111.06)

The inspector reviewed and toured areas containing equipment used to mitigate and detect an internal flood on various elevations within the primary auxiliary building.

The plant areas selected contained risk significant equipment based on the Individual Plant Examination of External Events (IPEEE) Section 5.0. Specifically, internal flood initiations from service water, fire protection line breaks, and refueling water storage tank breaks in the primary auxiliary building contribute approximately 6% of the overall core damage frequency from internal floods. The inspection determined whether procedures were adequate, mitigation systems were operable pursuant to technical specification requirements, and sump level instruments were properly tested.

The inspector reviewed applicable licensee procedures and surveillance procedures, which included actions to mitigate the effects of flooding and tours to verify operability of mitigating equipment. The procedures reviewed are listed in Attachment 1.

The inspector reviewed Updated Final Safety Analysis Report section 9.3.3.2.2. The inspector also reviewed safety evaluation NS-2-84-047, Primary Auxiliary Building Flood Alarms, associated with a modification to the sump level switches and alarms.

b. Findings

No significant findings were identified.

1R07 Heat Sink Performance

a. Inspection Scope (71111.07A)

The inspector verified that the licensee's program was adequate to ensure proper heat exchanger performance for the 21 and 22 emergency diesel generator (EDG) jacket water and lube oil coolers. The references used for this inspection included the Emergency Diesel Generator System Health Report, the EDG Basis Document revision 2, and Heat Exchanger Inspection Reports for Work Order Nos. 01-22146, 01-22147, 98-02651 and 98-02653.

The inspector reviewed heat exchanger preventive maintenance (PM) records to verify that the performance monitoring techniques ensuring heat removal capabilities were acceptable. The inspector verified that the inspection results were appropriately compared against established acceptance criteria; the performance monitoring considered the differences between plant conditions and design conditions established in Calculation PGI-00354-02; the frequency of testing and inspections was sufficient; and, the licensee established acceptance criteria for bio-fouling control. The inspector verified that the results were evaluated to ensure proper heat exchanger operation.

The inspector reviewed a sample of corrective action system condition reports related to the selected equipment to verify that identified problems were appropriately resolved

(reference CR Nos. 199808862, 200110318 and 200203601). The inspector also conducted a walkdown of the selected heat exchangers to assess material condition.

b. Findings

No significant findings were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope (71111.11Q)

Background

On May 31, 2002, a staff crew failed the licensee administered final simulator scenario of the accelerated high intensity training program. The staff crew performance was documented in CR No. 200205647. The operators were prohibited from fulfilling licensed duties until successfully completing a two week remediation program. The high intensity training program was part of the training improvement plan that was documented in NRC report 50-247/02-09.

a. Inspection Scope (71111.11Q)

On June 21, 2002, the inspector observed the performance of the staff crew during scenario ESR-022-06. The inspector verified that the scenario met the attributes outlined in Attachment 11 of Inspection Procedure 71111.11. The training staff administered the scenario and evaluated operator performance. The operators passed the evaluation and the inspector's evaluation agreed with that of the facility.

b. Findings

No significant findings were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspector reviewed risk significant equipment problems that were associated with the auxiliary feedwater, service water, 13.8 kilovolt system, and 440 volt systems. The inspector reviewed licensee follow-up actions to assess the effectiveness of maintenance activities. Issues selected for review included licensee identification of any maintenance preventable functional failures and repetitive failures, as well as, problem identification and resolution of any maintenance related issues. The inspector also reviewed system availability, system reliability monitoring, and system engineering involvement. The inspector reviewed the maintenance rule basis documents as listed in Attachment 1. The following specific issues were reviewed:

- CR No. 200111489, Failure of automatic function of auxiliary feedwater flow control valve FCV-406A.

- CR No. 200202589, Failure of valve SWN-7 (service water to turbine lube oil coolers) to fully close due to operator gear stripping.
- CR No. 200205115, Loss of 13.8 KV Alternate Safe Shutdown Bus Section 3.

The inspector reviewed the licensee's evaluations for the deficiencies identified in Condition Report 200205115, which concerned the loss of the power supply for alternate safe shutdown equipment (via substations 12RW3 and 12FD3) for 29 hours and 42 minutes (see report detail 1R13). The inspector reviewed the licensee's corrective actions to assess the impact of the equipment loss on the reliability and availability of the 440 volt system and to assess the event as a maintenance preventable functional failure, since the failure of a "retired" IP Unit 1 protective relaying circuit affected a Unit 2 risk significant system.

b. Findings

No significant findings were identified.

1R13 Maintenance Risk Assessment and Emergent Work Activities

a. Inspection Scope (71111.13)

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

- WO 02-02772, Safety Injection Valve 851B Failed to Stroke (CR 200205369).
- WO 02-02812, Control Room West Wall Inspections (CR 200205807).
- WO 02-45258, Unit 1 Lighting and Power Bus Section 3 Overcurrent Relay Retired Input Bypass (CR 200205115).
- WO 02-00735, Valve MS-41 Main Steam Supply to 22 Auxiliary Feed Pump packing leak.

The inspector reviewed licensee actions to assess, evaluate, and correct the deficiency in Condition Report 200205369, which concerned the failure of safety injection valve MOV-851B to stroke during a surveillance test. Further NRC review of this matter is described in section 1R15 of this report.

On May 17, the inspector observed operators responding to a loss of the Unit 1 13.8 kilovolt lighting and power bus section 3. The inspector observed repair activities and evaluated the licensee's causal evaluation and corrective actions as documented in Condition Report 200205115.

b. Findings

GREEN. On May 17, multiple grounds on the over-current protective circuit for Unit 1 substation 102NS3 resulted in a loss of the 13.8 kv lighting and power (L&P) bus section 3. The consequence of this event was a loss of available power to major alternate safe shutdown (10 CFR 50 Appendix R section III, Specific Requirements) pumps (23 and 24 service water pumps, 23 component cooling water pumps, 21 auxiliary boiler feedwater pump, 23 charging pump, and 21 and 22 residual heat removal pumps). In addition, some instrumentation for alternate safe shutdown was de-energized, including the alternate safe shutdown source range nuclear instrument and reactor coolant system loop 21 and 22 hot and cold leg temperature indicators.

This issue is more than minor since unavailability of alternate safe shutdown equipment for 30 hours is viewed as a precursor to a significant event. The alternate safe shutdown power supply is considered a risk-significant maintenance rule system which was unavailable for approximately 30 hours, greater than guidance contained in NRC Manual Chapter 0609 Appendix A (24 hours). The inspector consulted Region 1 senior risk assessment personnel to further characterize the risk significance of this performance issue. The NRC evaluated all accident sequences in the licensee's Individual Plant Examination for External Events (IPEEE) Table 4.6, in which alternate safe shutdown equipment is assumed for recovery action. The results of the analysis, based upon a duration of 30 hours, resulted in a very low risk significance (5E-7/reactor-year core damage frequency). The dominant sequence is a fire in the control room which results in de-energizing the Unit 2 normal and emergency sources of power.

The licensee repaired and restored the 13.8 kv bus section 3 within 30 hours of the initial fault. The licensee implemented appropriate risk assessments and adhered to administrative requirements identified in station administrative order 703, "Fire Protection Impairment Criteria and Surveillance." The administrative action for alternate safe shutdown components was to restore them to an operable condition within 72 hours. The Unit 2 normal and emergency electrical power supplies were available to supply power to the above stated mitigation equipment and instrumentation. During the course of troubleshooting the electrical grounds, the licensee identified that two additional over-current circuits were energized to long-standing retired 440 volt substations (132PC3 and 142PC3) at Unit 1. The protective circuits supply input into the 13.8KV supply breaker (SB1-3) to L&P bus section 3. No preventative maintenance is performed on the retired energized protective circuits at Unit 1. Restoration of the L&P bus section 3 was accomplished with temporary facility change TFC 2002-044 that permanently disabled the overcurrent protection circuit to substations 132PC3 and 142PC3 and bypassed the grounded sections of protection from substation 102NS3. The licensee plans an extent-of-condition review to evaluate all 440 volt Unit 1 substations by the end of September, and to identify if protective circuits to retired equipment are still energized.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspectors reviewed selected operability determinations to assess the adequacy of the evaluations, the use and control of compensatory measures, compliance with the Technical Specifications, and the risk significance of the issues. The inspectors used the Technical Specifications, Technical Requirements Manual, Final Safety Analysis Report,

and associated Design Basis Documents as references. The specific issues reviewed included:

- CR No. 200203319, Stroke Time for Post Accident Containment Vent Valve E-3.
- CR No. 200205369, Loose Wire in MOV 851B Open Circuit.
- CR No. 200205673, Large Amount of Debris on the Floor of the 480V Switchgear Room.

b. Findings

GREEN. On two occasions between May 27 and 29, 2002, during surveillance testing of safety injection discharge motor-operated valve (851B) the valve failed to stroke closed. The initial operability evaluation performed by operations did not consider the containment isolation function for this valve. Valve 851B has three safety-related functions. The first valve function is to open and direct flow from the high head safety injection pump 22 to the appropriate safety injection header. The second function of valve 851B is to close upon a failure of the 21 safety injection pump to ensure equalization of safety injection flow. The third function of valve 851B is to be remotely closed by operators for containment isolation following a transfer from low head cold leg recirculation to hot leg recirculation, as documented in emergency operating procedure (EOP) ES1.4, "Transfer to Hot Leg Recirculation." For 9.25 hours on May 27, and for approximately 1.5 hours on May 29, operators did not consider the non-automatic containment isolation function and did not enter Technical Specification (TS) 3.6.A.3.a. Following inspector discussions with the on-shift senior reactor operator on May 29, actions were taken in accordance with TS 3.6.A.3.a.

The containment isolation function was restored to an operable status by a change to EOP ES 1-4 that provided guidance to operators to close the valve from its motor control center with the appropriate tools and equipment in the primary auxiliary building. The inspector verified the EOP change and staging of tools and equipment in the primary auxiliary building. This event was documented in CR No. 200205433.

The performance issue associated with this finding was the lack of operator awareness of the multiple safety functions for mitigating systems valves. This was the second recent example of operators not considering all the safety functions for mitigating system valves. The first example was documented in NRC report 50-247/2002-003 section 1R15.

The incomplete operability assessment for safety injection discharge valve 851B has very low safety significance since the containment isolation valve was restored to an operable status prior to exceeding the 36-hour allowed outage time, per TS 3.6.A.3.a.2.d. Consequently, no violation of TS 3.6.A.3.a.2.d occurred.

1R16 Operator Workarounds

a. Inspection Scope (71111.16)

The inspector reviewed the licensee's list of twenty operator burdens as of

June 25, 2002, to assess the cumulative effects on system reliability, availability, and potential for mis-operation of a system. The inspector evaluated the below listed burdens to assess the operator impact to implement abnormal operating procedures or emergency operating procedures:

- CR No. 200109397, Nuclear Detector Rate Comparator Indication.
- CR No. 200109706, Letdown Pressure Regulator Manual Operation.
- CR No. 200110544, Residual Heat Removal Flow Control Valve Position Indication.
- CR No. 200202395, 24 Reactor Coolant Pump Motor Temperature Indication.
- CR No. 200101809, 22 Main Feedwater Pump Steam Stop Valve Control Switch.

The inspector verified that none of the deficiencies were an operator work-around that would require operator action during a transient that was time-dependent and would adversely affect risk or major equipment reliability. The inspector reviewed the cumulative effect of the burdens for either increasing the potential for initiating events or adversely impacting mitigating systems. The inspector verified the condition of the deficiencies and the compensatory measures taken. The inspector used Operations Administrative Steplist (OASL)-15.43, "Operator Burden Program", Revision 0, as a reference for this review.

b. Findings

No significant findings were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope (71111.17A)

The inspection consisted of a review of plant modification FMX-97-12648-M, "Installation of Canopy Seal Clamp Assemblies," and the associated safety evaluation No. 01-0707-M. This modification is being planned for implementation during the upcoming refueling outage as a contingency if a reactor vessel head penetration nozzle for a spare control rod drive mechanism (CRDM) was found to be leaking. Five canopy seal leaks have been repaired at Indian Point Unit 2 since the issue was first identified in the mid 1980s. The most recent repair was performed in 1997. The purpose of this modification is to provide a seal clamp around the leaking canopy seal. This modification has been implemented at a number of other nuclear facilities in the recent past.

The specific inspection attributes evaluated included:

- compatibility of clamp assembly with reactor coolant system.
- environmental qualification and evaluation of seismic interactions with the spare CRDM nozzles.
- verification that reactor coolant pressure boundary was not compromised.
- verification of post-maintenance testing proposed.
- verification of failure modes introduced by the modification.

This contingency modification has the potential to impact the barrier integrity cornerstone.

b. Findings

No significant findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope (71111.19)

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

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

- PWT IP2-02-48106 Replacement of Zone 2 Weld Channel Containment Pressurization system valve 1110-8, performed on June 25, 2002.
- PWT IP2-02-42774 Replace Packing and Change Out Stuffing Boxes for the 22 Charging Pump, performed on June 20-24, 2002.

b. Findings

No significant findings were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspector reviewed surveillance test procedures and observed testing activities to assess whether: 1) the test preconditioned the component tested; 2) the effect of the testing was adequately addressed in the control room; 3) the acceptance criteria demonstrated operational readiness consistent with design calculations and licensing documents; 4) the test equipment range and accuracy was adequate and the equipment was properly calibrated; 5) the test was performed per the procedure; 6) the test equipment was removed following testing; and 7) test discrepancies were appropriately evaluated. The surveillances 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 Specification 6.8.1.a. The following test activities were reviewed:

- PT-Q27B, 23 Auxiliary Feedwater Pump Test, Revision 8 (CR 200204977 and 200204979), performed on May 14, 2002.
- PT-Q13 Data Sheets 174, LCV-1208A, and 176, LCV-108B, Inservice Valve Test, Revision 23, performed on May 22, 2002.
- PT-M93, Fuel Storage Building Filtration System Functional Test, Revision 0, performed on June 13, 2002.

b. Findings

No significant findings were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23)

The inspector reviewed temporary facility change (TFC) package, TFC-2001-100, "Gas Turbine 1 Black-start Diesel Battery Installation." This TFC was prepared to install a temporary replacement to the installed black-start emergency diesel generator battery due to a failure in December 2001. The inspector reviewed the TFC and the associated safety evaluations to verify the facility change did not adversely impact safety system operability and the license requirements, and did not violate 10 CFR 50.59. The inspector performed a field walkdown, reviewed the site procedure for performing temporary facility changes, SAO 206, Revision 22, "Temporary Field Change," and evaluated the comprehensiveness of the licensee's review of impacted procedures and drawings.

On May 20, 2002, the inspectors observed the installation of TFC 2002-033, "FCV-1207 Hot Tap Isolation Valve." TFC 2002-033 installed a hot tap stop valve down stream of flow control valve (FCV)-1207, low pressure steam dump, in order to isolate FCV-1207 from the condenser and perform corrective maintenance on FCV-1207. This TFC was selected for observation due to the potential to degrade main condenser vacuum during the hot tap installation. The inspectors reviewed the temporary facility change package, including the 10 CFR 50.59 screening and Work Order #IP2-02-42003 under which the TFC was installed, against the Updated Final Safety Analysis Report and Station Administrative Order SAO-206, "Temporary Facility Change," Revision 22.

b. Findings

No significant findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope (71114.04)

The inspector conducted an in-office review of licensee submitted changes for the emergency plan-related documents to determine if the changes decreased the

effectiveness of the plan. A thorough review was conducted of documents related to the risk significant planning standards (RSPS), such as classifications, notifications, and protective action recommendations. These changes (final changes) were reviewed against 10 CFR 50.54(q) to ensure that the changes do not decrease the effectiveness of the plan, and that the changes made continue to meet the standards of 10 CFR 50.47(b) and the requirements of Appendix E. These changes are subject to inspections to ensure that the changes continue to meet NRC regulations. The submitted and reviewed documents (Plan and Implementing Procedures) are listed in Attachment 1.

b. Findings

No findings of significance were identified.

3. Safeguards (Cornerstone: Physical Protection)

3PP1 Access Authorization Program

a. Inspection Scope (71130.01)

The below listed activities were conducted to determine the effectiveness of the licensee's behavior observation portion of the personnel screening and fitness-for-duty (FFD) programs as measured against the requirements of 10 CFR 26.22 and the licensee's fitness-for-duty program documents.

Five supervisors representing the Indian Point Units 2 and 3, Maintenance, Operations Procedures, Operations, and Engineering departments were interviewed on May 22, 2002, regarding their understanding of behavior observation responsibilities and the ability to recognize aberrant behavior traits. Two Access Authorization/Fitness-for-Duty self-assessments, two semi-annual fitness-for-duty performance data reports, an audit, and event reports and loggable events for the four previous quarters were reviewed during May 20-24, 2002. On May 22, 2002, five individuals who perform escort duties were interviewed to establish their knowledge level of those duties. Behavior observation training procedures and records were reviewed on May 21, 2002.

b. Findings

No significant findings were identified.

3PP2 Access Control

a. Inspection Scope (71130.02)

The below listed activities were conducted during the inspection period to verify that the licensee has effective site access controls, and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area as measured against 10 CFR 73.55(d) and the Physical Security Plan and Procedures.

Site access control activities were observed, including personnel and package processing through the search equipment during peak ingress periods on June 4 and 5, 2002. On May 22, 2002, testing of all access control equipment including metal detectors, explosive material detectors, and X-ray examination equipment was observed. The Access Control event log, an audit, and three self-assessments were also reviewed.

The inspectors also reviewed the Condition Reports (CRs) generated and entered into the licensee's corrective action program, to address concerns identified during the inspection. The CRs reviewed are identified in the list of documents contained in Attachment 1 of the report.

b. Findings

No significant findings were identified.

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 were consistent with the guidance contained in Nuclear Energy Institute (NEI) 99-02, Revision 1 and Revision 2, "Regulatory Assessment Performance Indicator Guidelines." 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 discrepancies in the performance indicator measurements to verify problems were satisfactorily resolved.

.1 Safety System Unavailability - Auxiliary Feedwater

a. Inspection Scope (71151)

The inspector reviewed maintenance rule tracking, control room logs, and condition reports associated with the auxiliary feedwater system. The inspector reviewed plant data from the 2nd quarter of 2001 through the 1st quarter of 2002 for all three trains of auxiliary feedwater. The inspector also reviewed two specific condition reports (200110095 and 200110097) related to errors in the data provided to the Nuclear Energy Institute prior to issuance to the NRC. The licensee's quality assurance group had identified the errors in the data submittal in response to previous NRC inspection observations of data accuracy.

b. Findings

No significant findings were identified.

.2 Safety System Unavailability - Emergency AC Power

a. Inspection Scope (71151)

In the 1st Quarter Performance Indicator data, the licensee reported that an engineering assessment was ongoing to evaluate the 23 emergency diesel generator (EDG) governor failure modes and effect on fault exposure hours, if any. The inspector reviewed the condition report (CR No. 200203079) and assessment associated with this failure to evaluate the licensee's determination that the EDG was fully functional prior to the repairs to correct the governor failure.

b. Findings

No significant findings were identified.

.3 Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment

a. Inspection Scope

The inspectors reviewed the licensee's programs for gathering and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment Performance Indicators. The review included the licensee's tracking and trending reports, personnel interviews and security event reports for the Performance Indicator data collected from the 2nd quarter of 2001 through the 1st quarter of 2002.

b. Findings

No significant findings were identified.

4OA3 Identification and Resolution of Problems

a. Inspection Scope (71152)

The inspector conducted a problem identification and resolution (PI&R) sample inspection to review Entergy's actions to address problems related to a portion of the Protected Area boundary that may not have had compensatory measures implemented in accordance with the Security Plan, as documented in CR No. 200112339. The inspector verified the appropriateness of compensatory measures and actions taken to count the compensatory hours toward the applicable performance indicator.

b. Findings

No significant findings were identified.

4OA4 Inspection Item Follow-up

.1 (Closed) URI 05000247/2001-04-05: Adequacy of auxiliary feedwater (AFW) alignment and function in response to a feedwater line break. The system response to the postulated event (feed line break) was beyond the design description documented in the

Updated Final Safety Analysis Report. The licensee documented this item in CR No. 200106409 and evaluated the auxiliary feedwater system response to a postulated line break through an engineering analysis (PSA-010830-01, "Feedwater System Pipe Break Analysis). The engineering analysis concluded that decay heat could be removed with the minimum AFW alignment to ensure that the reactor coolant system does not over pressurize or the pressurizer does not go into a water-solid condition. The inspector reviewed the analysis and concluded that there was no violation of NRC requirements. This unresolved item is closed.

.2 (Updated) FIN 05000247/01-013-01: Proposed finding due to crew high failure rate during the 2001 annual requalification simulator examinations. This finding was documented in an October 2001 inspection and initially characterized as a potential Yellow finding, the final safety significance to be determined (TBD). This finding was subsequently evaluated under the significance determination process (SDP) and characterized as (reference NRC to Entergy letters dated December 5, 2001, and February 28, 2002). The 95002 Supplemental Inspection (reference Inspection Report No. 50-247/02-09, dated May 31, 2002), assessed the licensee's evaluation of the crew high failure rates and the corrective actions taken to address this performance issue. As stated in the cover letter to Inspection Report No. 50-247/02-09, this finding remains open until after the completion of Entergy's licensed operator requalification examinations, scheduled for September-October 2002, and further review by the NRC. This item remains open.

4OA6 Meetings, Including Exit

On May 24, 2002, the inspectors presented the inspection results of report details 3PP1, 3PP2, and 4OA3 to licensee representatives. 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.

On July 3, 2002, the inspector presented an overall summary of the inspection results to Mr. Paul Rubin, and other members of the licensee staff, who acknowledged the findings. No material examined during the inspection should be considered proprietary.

ATTACHMENT 1

a. Key Points of Contact

R. Allen	Manager, Regulatory Affair
P. Asendorf	Manager of Security
T. Barry	Security Superintendent
J. Cambigianis	System Engineer
F. Dacimo	Vice President, Operations
R. Depatie	System Engineer
W. Durr	Assistant Operations Manager
T. Foley	System Engineer
W. James	Maintenance and Construction Manager
J. McCann	Manager, Nuclear Safety and Licensing
G. Norton	Control Room Supervisor
P.K. Parker	Maintenance Manager
J. Perrotta	QA Manager
P. Rubin	Operations Manager
V. Sacco	System Engineer
G. Schwartz	Director of Engineering
P. Speedling	Fire Protection Specialist
R. Taylor	QA Engineer
M. Vasely	System Engineering Section Manager
J. Ventosa	System Engineering Manager

b. List of Items Opened, Closed, and Discussed

Closed

URI 50-247/01-04-05	Auxiliary Feedwater Alignment and Function

Discussed

FIN 50-247/01-013-01	Crew High Failure Rate During 2001 Annual Requalification Simulator Examinations

c. List of Documents Reviewed

Procedures

COL 21.3, Steam Generator Water Level and Auxiliary Boiler Feedwater System
COL 3.1, Chemical and Volume Control System
SAO-700, Fire Protection and Prevention Policy
SAO-703, Fire Protection Impairment Criteria and Surveillance
AOI 28.0.6, Nuclear Side (Outside Containment) Flooding
AOI 24.1, Service Water Malfunction
AOI 4.1.1, Loss of Component Cooling Water
ARP SJF Window 4-6, Service Water Header High/Low Pressure

LARP-4, Primary Auxiliary building Sump Pump High Level
EOP ES 1.4, Transfer to Hot Leg Recirculation
OASL 15.43, Operator Burden Program
SAO-206, Temporary Facility Change
SAL-114, Preparation of NRC and WANO Performance Indicators
Emergency Plan, Section 5, Rev 01-02a
IP-1002, Emergency Notification and Communication, Rev. 25
IP-1008, Personnel Radiological Check and Decontamination, Rev. 7
IP-1010, Central Control Room, Rev. 3, 4, 5
IP1011, Joint News Center, Rev. 6, 7
IP-1021, Manual Update, Readout & Printout of PROTEUS Plan Parameter Data, Rev. 6
IP-1023, Operations Support Center, Rev. 17, 18
IP-1024, Emergency Classification, Rev. 9, 10
IP-1027, Personnel Accountability and Evacuation, Rev. 15, 16
IP-1047, Obtaining Offsite Exposure Rates From MIDAS Using a Data Terminal, Rev. 8
IP-1050, Security, Rev. 2, 3
EPMP-EPP-01, Maintenance of Emergency Preparedness, Rev. 14
EPMP-EPP-02, Emergency Equipment Inventories and Checklists, Rev. 25
EPMP-EPP-04, Emergency Exercise/Drill Procedure, Rev. 8
EPMP-EPP-06, Emergency Response Organization Notification Maintenance and Surveillance, Rev. 10

Calculations

PGI-00433, Combustible Loading Calculation
PGI-00354-02, Emergency Diesel Generator Heat Exchanger Performance
PSA-010830-01, Feedwater System Pipe Break Analysis

Drawings

9321-F-2736-114, Flow Diagram Chemical and Volume Control System, Sheet 1
A208168-51, Flow Diagram Chemical and Volume Control System, Sheet 2

Miscellaneous

Indian Point Nuclear Generating Station Unit 2 Maintenance Rule Basis Document, 13.8 KV System, Revision 1
Indian Point Nuclear Generating Station Unit 2 Maintenance Rule Basis Document, Auxiliary Feedwater, Revision 1
Indian Point Nuclear Generating Station Unit 2 Maintenance Rule Basis Document, Service Water, Revision 1
Maintenance Rule Unavailability for 13.8 KV System (4/2000 - 3/2002)
Maintenance Rule Reliability for 13.8 KV System (2001 - 2002)
Maintenance Rule Unavailability for 440 V System (4/2000 - 3/2002)
Maintenance Rule Reliability for 440 V System (2001 - 2002)
4th Quarter 2001 System Health Report for 13.8 KV and 440 V Systems
FMX-97-12648-M, Installation of Canopy Seal Clamp Assemblies
TFC 2001-100, Gas Turbine 1 Black start Diesel Battery Installation
TFC 2001-033, FCV-1207 Hot Tap Isolation Valve

Condition Reports

200206616, Fire Brigade Leader without SCBA donned
200206187, Blocked fire hydrant outside emergency diesel generator building
199808862, Debris on 23 diesel generator jacket and lube oil heat exchangers
200110318, Epoxy broken off on 21 emergency diesel generator jacket water heat exchanger
200203601, 21 and 22 CCW heat exchanger as left performance data
200205647, Staff team "X" failed to complete high intensity training
200111489, Failure of automatic function of auxiliary feedwater flow control valve FCV-406A
200202589, Failure of SWN-7 to fully close due to operator gear stripping
200205115, Loss of 13.8 kv alternate safe shutdown bus section 3.0
200205369, Failure of safety injection MOV 851B to close
200203319, Stroke time for Post accident containment vent valve E-3
200205369, Loose wire in MOV 851B open circuit
200205673, Large amount of debris on the floor of the 480 volt switchgear room
200205433, Incomplete Operability for MOV 851B
200109397, Nuclear detector rate comparator indication
200109706, Letdown pressure regulator manual operation
200110544, Residual heat removal flow control valve position indicator
200101809, 22 main feedwater pump steam stop valve control switch
200202395, 24 Reactor coolant pump motor temperature indication
Condition Reports (cont.)
200110095, Safety system unavailability for auxiliary feedwater error for 3rd quarter data
200110097, Safety system unavailability for auxiliary feedwater does not have data available
200203079, 23 emergency diesel generator test
200112339, Protected area not properly compensated

Sections 3PP1 and 3PP2

Entergy Fitness-for-Duty Program
Fitness-for-Duty Performance Data, January - June 2001; June - December 2001
Self-Assessment - Access Control, March 2002, Report 02-001
QA Surveillance Report - Fitness-for-Duty Program, April 26, 2002
Condition Detail Report 200205250, Individual Failed to Report Immediately for a FFD test
Condition Detail Report 200200272, Non-Notification of Employee No Longer Requiring Photo Badge
Condition Detail Report 200201585, Access not Notified of Employee on Medical Leave; Badge Not Deactivated
Indian Point Nuclear Generating Station Unit 2 Maintenance Rule Basis Document, 13.8 KV System, Revision 1

d. List of Acronyms

AFW	auxiliary feedwater
AOI	abnormal operating instruction
ARP	annunciator response procedure
ATWS	anticipated transient without scram
CFR	Code of Federal Regulations
COL	check off list
CR	condition report
CRDM	control rod drive mechanism
DBT	design basis threat
EDG	emergency diesel generator
EOP	emergency operating procedure
FCV	flow control valve
FFD	fitness-for-duty
IPEEE	individual plant examination for external events
kv	kilovolt
L&P	lighting and power
LARP	local annunciator response procedure
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OASL	Operations Administrative Steplist
PARS	publicly available records
PI	performance indicator
PI&R	problem identification and resolution
PM	preventive maintenance
PWT	post-work test
RSPS	Risk Significant Planning Standard
SAO	station administrative order
TBD	to be determined
TFC	temporary facility change