May 14, 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 50-247/02-02

Dear Mr. Dacimo:

On March 30, 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 April 8, 2002, with you and 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 engineering and occupational radiation safety areas. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). On February 25, 2002, the NRC issued an Order to all nuclear power plant licensees, requiring them to take certain additional interim compensatory measures to address the generalized high-level threat environment. With the issuance of the Order, we will evaluate Entergy's compliance with these interim requirements.

Based on the results of this inspection, two violations of NRC requirements were identified related to inadequate corrective actions for repeat failures of a service water valve and not installing locking devices on service water valves. However, because of the very low safety significance and because they have been 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 date of this inspection report, to the Nuclear

Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Indian Point 2 Nuclear Power Plant.

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 50-247/02-02

Attachment 1 - Supplemental Information

cc w/encl: J. Yelverton, Chief Executive Officer
M. Kansler, Senior Vice President and CEO
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
J. Rampe, 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

cc w/encl: 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

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	50-247/02-02
Licensee:	Entergy Nuclear Operations, Inc..
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	February 10 - March 30, 2002
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector Leonard Cheung, Senior Reactor Inspector John McFadden, PhD, Radiation Specialist Todd Fish, Senior Reactor Inspector Jimi Yerokim, Senior Reactor Inspector Mel Gray, Reactor Inspector Lois James, Resident Inspector
Approved by:	Peter W. Eselgroth, Chief Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000247-02-02, on 2/10 - 3/30/2002, Entergy Nuclear Operations, Inc.; Indian Point 2 Nuclear Power Plant. Maintenance Risk Assessment and Cross Cutting Issues.

The report covered a 7 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: Initiating Events

Green. The procedure in use was inappropriate in that it did not require that the 138 kilovolt off-site power system be declared inoperable during scheduled maintenance on the station auxiliary transformer (SAT) tap changer. On February 28, 2002, for approximately 51 minutes, control room operators had placed the SAT tap changer in manual and local control in accordance with system operating procedure (SOP) 27.1.7, "Operation of Main, Station and Unit Auxiliary Transformers," section 4.8. The scheduled maintenance was not intrusive into tap changer operation, however, the licensee had not fully evaluated if the intended function could be maintained with operator compensatory actions to restore the tap changer to automatic. The limiting condition for operation in technical specification 3.7.B.3 for a loss of the 138 kilovolt power system is 24 hours, which was not exceeded during this scheduled maintenance activity. The issue had a credible impact on safety. Inappropriate control of the SAT tap changer impacts the initiating event cornerstone in that a loss of off-site power is more likely following a reactor trip. This issue was determined to be of very low safety significance (Green) using phase one of the SDP because no reactor trip occurred during the inspection period and no mitigating systems were directly impacted by the maintenance on the SAT tap changer.

Cross-Cutting Issues

Green. The manual operator on service water (SW) valve SWN-7 failed on March 9 during operations to swap essential SW headers. SWN-7 is the isolation valve for the service water supply to turbine building loads. The inoperable valve could have resulted in insufficient service water flow for the emergency diesel generators and other safety systems had there been a demand for those safety systems. The operator on SWN-7 failed 6 other times since 1995. Following the early failures, an engineering evaluation determined that the design margin for the gear box in the manual operator was marginally adequate. Engineering work request 12110-99 was issued to replace the gear set on SWN-7 and similar valves with high strength materials. The engineering request was canceled in July 1999 and no action was taken. This issue had very low safety significance since the specific failure on March 9 and corrective actions occurred within the limiting condition for operation for the service water system, and no operating or stand-by mitigating equipment supported by service water was called to perform its intended function. The failure to take adequate corrective action for repeat failures of service water valve SWN-7 was a violation of 10 CFR 50, Appendix B, Criterion XVI. This is being treated as a Non-cited violation.

Summary of Findings (cont'd)

No Color. Two personnel errors were identified, which were examples of the cross cutting issue in human performance. The issues involved the failure to properly maintain locking devices on five service water test stop valves; and, the February 23, 2002, security event involving horseplay by a security officer with a duty weapon. The failure to maintain locking devices on service water valves per the operating procedures was a violation of Technical Specification 6.8.1.a. This is being treated as a Non-cited violation.

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Report Details

SUMMARY OF PLANT STATUS

The plant operated at full power during the inspection period, except for minor power decreases for routine tests and maintenance.

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

1RO1 Adverse Weather Protection

a. Inspection Scope (71111.01)

The purpose of the inspection was to review the design and operation of the intake structure debris removal equipment to verify that debris in the Hudson River water would not impact the function of the service water system or increase the initiating event frequency related to a loss of circulating water pumps. The inspector selected the intake structure for review because it was the 3rd most important event based on risk achievement worth. The inspector walked down accessible portions of the service water and circulating water traveling screens, the service water discharge strainers and the wash water system. The documents referenced by the inspector during this inspection included:

- Individual Plant Examination (IPE) Sections 3.2.15, 3.3.4 and Table 3.2-15B
- Updated Final Safety Analysis Report (UFSAR) Section 9.6.1
- Abnormal Operating Procedure (AOI) 28.0.7, Hurricane, Tornado, High Winds, Severe Thunderstorm
- System Operating Procedures (SOPs) 22.1, Wash Water and Traveling Screen Operation, and SOP 24.1, Service Water System Operation
- Local Annunciator Response Procedure LARP28, Service Water Screen Trouble
- SE-330, Service Water Bay Inspection (May 5, 2000)
- Condition Reports (200122317, and 200111357)
- Plant Drawings B245521-12, Screen Wash System; A250654-00, Service Water Traveling Screens #27 and #28; B250117-01, Unit 2 Screen Wash System Screens #21 thru #26; and, A208368-30, Screen Wash System and Bearing Cooling Water for Circulating and Service water Pumps

b. Issues and Findings

No significant findings were identified.

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope (71111.02)

The inspector reviewed nine safety evaluations associated with the initiating events, mitigating systems, and barrier integrity cornerstones to verify that changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR) were reviewed and documented in accordance with 10 CFR 50.59, and that the safety issues pertinent to the changes were properly resolved or adequately addressed. The safety evaluations were selected based on the safety significance of the changes and the risk to structures, systems and components.

The inspector reviewed 13 screened-out evaluations for changes, tests and experiments for which the licensee determined that safety evaluations were not required. This review was performed to verify that the threshold for performing safety evaluations was consistent with 10 CFR 50.59.

In addition, the inspector reviewed the administrative procedure that was used to control the screening, preparation, and issuance of the safety evaluations to ensure that the procedure adequately covered the requirements of 10 CFR 50.59. The inspectors also interviewed engineering personnel engaged in the preparation and the review of the selected 10 CFR 50.59 safety evaluations.

Finally, the inspector reviewed a sample of condition reports documenting problems identified by the licensee in their corrective action program related to safety evaluations to verify the effectiveness of corrective actions. A listing of the 10CFR50.59 safety evaluations, screen-out evaluations, and condition reports reviewed is provided in section d. of Attachment 1.

b. Issues and Findings

No significant findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope (71111.04)

On February 20, 2002, the inspector performed a partial system walkdown of the 22, 23, and 24 instrument buses and associated static inverters. The review was performed while the licensee investigated number 21 inverter which had swapped to its alternate power supply (reference condition report 200201914). The inspector reviewed the status of key 120 volt AC distribution panels and inverters based upon check off list (COL) 27.1.6, “Instrument Buses, DC Distribution and Public Address Inverter,” revision 17, system operating procedure (SOP) 27.1.6, “Instrument Buses, DC Distribution and PA Inverter,” revision 28, and plant drawing A208502, “118 VAC Instrument buses 21, 22, 23, and 24. The inspector noted minor deficiencies associated with errors in COL 27.1.6, untimely closure of a caution tag on retired equipment, and the lack of a mechanical interlock on the 22 instrument bus distribution panel. Condition reports 200201934, 200201937, and 200202927 were issued to address the deficiencies.

On March 5, 2002, the inspector performed a partial walkdown of the 21 and 23 safety injection (SI) pump trains using Check Off List (COL) 10.1.1, “Safety Injection System.” The purpose of this partial walkdown was to verify the operability and component alignment of the available SI trains while 22 SI pump train was being tested. The inspector reviewed Drawing Number 9321-F-2735-129, “Flow Diagram Safety Injection System,” Updated Final Safety Analysis Report (UFSAR) Section 6.2, “Safety Injection System,” Emergency Procedure E-1, “Loss of Reactor or Secondary Coolant,” Technical Specifications, and COL 10, “Locked Valves” to verify the proper alignment of the components in the SI system.

b. Issues and Findings

No significant findings were identified.

1R05 Fire Protection

.1 Fire Zone Tours

a. Inspection Scope (71111.05Q)

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:

- Zone 099A, Boric Acid Evaporator-Gas Stripper
- Zone 100A, Boric Acid Evaporator Room
- Zone 101A, Boric Acid Evaporator-Control Area
- Zone 105A, Monitor Tank Farm
- Zone 15, Control Room
- Zone 14, 480 Volt Switchgear Room
- Zone 23, 15 ft elevation of Auxiliary Feedwater Pump 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.”

A number of minor issues and procedural deficiencies were independently identified by the inspector that did not significantly impact the ability of the licensee to prevent, promptly detect and suppress fires that do occur, or to protect structure, system, and components (SSCs) important to safety such that a fire would not be able to prevent the safe shutdown of the unit. The observations were entered into the licensee's corrective action program as Condition Reports 200202034, 200202039, 200202038, and 200202261.

b. Issues and Findings

No significant findings were identified.

.2 Degraded Central Control Room Fire Barrier-West Wall

a. Inspection Scope (71111.05Q)

In February 2002, Entergy completed planned inspections of control room walls to locate and seal uncontrolled sources of ventilation in-leakage to support planned modifications and subsequent testing of the control room habitability envelope. In support of this activity, the licensee removed "hauserman" wall panels installed as an interior finish over the control room outer walls. The purpose of this inspection was to review the licensee actions in response to the discovery of degraded fire barriers in the control room walls.

b. Issues and Findings

URI While inspecting the central control room (CCR) walls in February 2002, Entergy identified several minor deficiencies in the CCR fire barriers (reference condition reports (CR) 200201933, 200202092, 200202095, and 200202098) and significant deficiencies in the control room west wall (CR 200202031 and 200202404). Entergy declared the fire barriers inoperable and implemented compensatory measures (fire watches) in accordance with station administrative procedure SAO 703 pending the completion of repairs to restore the barriers. The inspector independently verified the extent and significance of the deficiencies, and confirmed the implementation of the compensatory measures.

On February 22, 2002, Entergy identified that construction deficiencies in the CCR west wall had impaired the fire barrier (CR 200202031). The west wall is a composite wall constructed of 4 inch glazed brick on the turbine building side then a 1 inch air gap, and a 4 inch solid concrete block wall (on the control room side). The upper portions of the west wall on the control room side was poorly constructed with loose blocks, unevenly applied mortar and many holes and gaps that penetrated through the block joints. The licensee identified an unprotected steel column and a steel "C" channel. The construction of the brick wall on the turbine building side was acceptable.

The west wall is required by the Fire Hazard Analysis (FHA) Appendix B, Fire Zone 15, to be a Type I fire barrier having a 3-hour fire rating. The west wall did not have a full 3-hour rating as required by the FHA. Entergy estimated the wall had a 1.4 hour rating by taking credit for the 4 inch brick wall. The concrete block wall needed to be "re-pointed" to establish a continuous block and mortar barrier to restore a three hour fire rating. Entergy completed repairs to the masonry wall and restored the fire barrier to an operable status (reference Work Orders 02-25936 and 02-38933).

Entergy reviewed the wall for seismic interactions, and concluded that this condition did not degrade the structural integrity of the control room or impact equipment in the room. Entergy initiated an evaluation of the security aspects of the wall. Entergy removed other wall panels to expose actual control room wall surfaces and joints and identified no other substandard construction in the control room wall. Entergy planned to review additional walls in the plant using the 1980 masonry wall study to identify and inspect walls which are required to be a Type I barrier. The preliminary engineering review determined that the use of interior finish walls was limited to the control room so that walls in other areas of the plant are fully exposed and that the condition of any wall should be readily apparent. This corrective action is tracked through CR 200202031.

Entergy estimated that the west wall was constructed in 1978 as part of a modification to upgrade the wall to a 3-hour fire rating (reference Drawing A214523). The hauserman panels were removed in February 2002 to locate and seal potential sources of ventilation in-leakage to support planned modifications and subsequent testing of the control room habitability envelope. These inspections identified for the first time the deficient construction of the west wall since the 1978 modification. The licensee did not review the wall construction to verify the assigned 3-hour fire rating when the fire hazards analysis was completed for the fire protection program review in 1979. The failure to provide a Type I barrier with a 3 hour fire rating for the CCR west wall was a violation of the fire protection program requirements in License Condition 2.k.

Entergy completed an evaluation using the methodology of Appendix F of NRC Manual Chapter 0609 which credited some fire rating for the west wall and assessed its ability to function for the fire loadings on both sides of the barrier. The preliminary Entergy evaluation concluded the deficiency had very low safety significance. The NRC review of this inspection finding continued at the end of the inspection period, with a preliminarily determination that the issue had moderate safety significance. This matter is unresolved pending completion of the NRC review. The inspector verified control room habitability was assured despite the deficiencies the licensee identified in the control envelope during this period (reference Section 1R19 of this report). (**URI 50-247/02-02-01**).

.2 Hydrogen Storage Locations

a. Inspection Scope (TI 2515/146)

The purpose of the inspection was to verify the licensee was complying with applicable codes and commitments regarding hydrogen storage locations to ensure unrecognized risk-significant conditions do not exist. Specifically, the inspector identified the location where hydrogen is stored on site, and conducted a walk down of plant areas to confirm that the licensee provided greater than 50 feet separation between hydrogen storage and ventilation intakes, and risk-significant tanks or Structures, Systems and Components (SSCs).

The references and licensee commitments used for this inspection included Appendix A to Branch Technical Position 9.5.1; the Indian Point 2 Fire Protection program Plan, Appendix B - Fire Hazards Analysis, Revision 9 dated April 1998; the National Fire

Protection Association Code 50A, " Gaseous Hydrogen Systems;" the NRC Safety Evaluation Report dated January 31, 1979; and Temporary Instruction 2515/146, Hydrogen Storage Locations.

A number of minor deficiencies were identified during the inspection which did not significantly impact the ability of the licensee to protect SSCs important to safety. The licensee entered the observations into the corrective action program as condition reports CR 200201801 and 200201817. Licensee actions continued at the end of the inspection to verify the status (retired) of hydrogen storage systems at Indian Point Unit 1.

b. Issues and Findings

URI No significant findings were identified. The inspector identified hydrogen storage locations that were less than 50 feet from alternate safe shutdown equipment (CR 200203336) and the ventilation intake of a building containing non-safety related equipment (CR 200203337). The licensee had not reviewed either configuration for impact on the fire hazards analysis, and initiated an evaluation. The NRC Fire Protection Safety Evaluation dated January 31, 1979, requires that hydrogen be stored at least 50 feet away from safe shutdown equipment. The storage of hydrogen within 50 feet of alternate safe shutdown equipment was a potential violation of the requirements of License Condition 2.K. The above discrepancies did not present a plant safety concern because the normal power supply to safe shutdown equipment was greater than 50 feet from the hydrogen storage locations. The results of the review were provided to the NRC Office of Nuclear Reactor Regulation for further review and evaluation. This matter is unresolved pending the completion of the NRC review (**URI 50-247/02-02-02**).

1R11 Licensed Operator Requalification Program

.1 Observation of Simulator Training
Background

In response to a Yellow finding based on crew high failure rate (reference NRC Inspection Report 50-247/2001-013), Entergy committed to have the NRC evaluate operators who had been remediated following the requalification exam failures, prior to returning them to shift duties. In January 2002, the licensee informed Region I NRC that they wished to return one such operator to shift duties. This operator had been part of a crew that failed its requalification exam and had been remediated, however he had not returned to licensed duties. Per the commitment, the licensee requested that an inspector evaluate the operator's performance during facility-administered scenarios.

a. Inspection Scope (71111.11

On February 7, 2002, a Region I inspector evaluated the performance of the operator during two scenarios. The inspector verified that the scenarios met or exceeded the scenario attributes outlined in Attachment 11 of Inspection Procedure 71111. The training staff administered both scenarios and also evaluated the operator's performance. For each scenario, the operator was assigned to the role of Shift Manager, which was the watch bill position he was scheduled to fill. The operator passed the evaluation. The facility evaluation agreed with the NRC evaluation.

b. Issues and Findings

No significant findings were identified.

.2 Enhanced Monitoring of Control Room Activities

a. Inspection Scope (71111.20, 71715)

The inspectors continued the implementation of an augmented inspection plan to permit long-term, heightened observation of control room activities and operator performance. The augmented inspection used resident and region-based inspectors. The inspectors monitored control room activities to verify operators, control room supervision and shift managers remained cognizant of plant conditions and work activities in the field. The inspectors verified the Shift Mentors were involved and interacted with the crews to provide real time feedback on performance and management expectations.

The inspectors confirmed the plant was operated safely and that the conduct of operations was generally in conformance with licensee administrative requirements. The inspectors verified the operators followed alarm response and operating procedures. Shift logs were reviewed to confirm the operators properly responded to equipment problems and documented control room activities. Inspector observations of performance deficiencies having minor safety significance were discussed with licensee management.

b. Issues and Findings

No significant findings were identified.

1R12 Maintenance Rule Implementation

.1 480 Volt System

a. Inspection Scope (71111.12)

The inspector reviewed risk significant equipment problems that occurred on the 480 volt system in 2001. The inspector reviewed licensee follow-up actions to assess the effectiveness of maintenance activities. Issues selected for review included licensee identification of any functional failures, 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 following issues were reviewed:

Report No.	Condition Description
200103374	Loss of Bus 3A during the performance of an undervoltage monthly functional test
200109445	Removal of control power fuses for Normal Supply Breakers to buses 2A and 6A
200110348	Cross-connecting of buses 3A and 6A during planned outage resulting in high temperatures on Station Service Transformer

The inspector reviewed the licensee's maintenance rule implementation for the 138 kV system, a safety significant, normally operating system. The inspector verified that the performance criteria were appropriate and the system was classified as Maintenance Rule (a)(1) based on the excessive unavailability of the off site feeders. The inspector used the following reference material and discussed this issue with the system engineer:

- Condition reports (200108388 and 200104619)
- Indian Point Nuclear Generating Station Unit 2 Maintenance Rule Basis Document, 138 kVAC Electrical System (138K), Revision 2
- Maintenance Rule Unavailability for 138k System (4/2000 - 2/2002)
- Maintenance Rule Reliability for 138k System (6/2001 - 2/2002)
- System Description 27.1, Electrical Systems
- 4th Quarter 2001 System Health Report for 138 KV System

b. Issues and Findings

No significant findings were identified.

.2 138 kV System

a. Inspection Scope (71111.12)

The inspector reviewed the licensee's assessment of the 138 kV system which had several Condition Reports (CRs) documenting issues associated with the system during the previous year. The licensee classified the 138 kV system as a high safety significant, normally operating system. The inspector verified appropriateness of the performance criteria for the classification of the system as Maintenance Rule (a)(1).

The inspector used the following reference material and discussed this issue with the system engineer:

- Condition reports (200108388 and 200104619)
- Indian Point Nuclear Generating Station Unit 2 Maintenance Rule Basis Document, 138 kVAC Electrical System (138K), Revision 2
- Maintenance Rule Unavailability for 138k System (4/2000 - 2/2002)
- Maintenance Rule Reliability for 138k System (6/2001 - 2/2002)
- System Description 27.1, Electrical Systems

- 4th Quarter 2001 System Health Report for 138 KV System

b. Issues and 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 Entergy's risk management. 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 discussed risk management with maintenance and operations personnel for the following activities:

- WO 02-25936, Degraded Control Room Fire Barrier-West Wall (CR 200202031)
- WO 02-00080, Work Permit (WP) 60742, Replace Valve Gearing for Service Water Valve SWN-7 (CR 200202591 and 200202589)
- CR 200201685, Failure of Gas Turbine 2 During a Surveillance Test Due to High Vibrations (February 13, 2002)
- Planned Maintenance on 138 KV off-site feeders (95891 and 96951) on March 4, 2002
- WO 02-39477, Gas Turbine #1 Tripping during Periodic Test M38A on March 20, 2002 (CR 200204994)

The inspector reviewed the operations plan for the 138 KV outage and the actions to conduct emergent repairs on service water valve SWN-7 on March 9, 2002. The inspector evaluated the effectiveness of the risk assessments performed for emergent operational activity and verified how the licensee managed the risk in accordance with 10 CFR 50.65(a)(4). These issues were discussed with operations management.

b. Issues and Findings

No significant findings were identified.

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:

- 22 Battery Charger Ground (CR 200202198)
- Control Room Instrument Rack Terminal Lugs (CR 200201551)
- Station Auxiliary Tap Changer Maintenance (CRs 200202112 and CR 200203032)

b. Issues and Findings

No significant findings were identified. A human performance cross-cutting issue is documented in report detail 4OA2.

1R17 Permanent Plant Modifications

a. Inspection Scope (71111.17)

The inspector reviewed ten risk-significant plant modification packages to verify that: (1) the design bases, licensing bases, and performance capability of risk significant structures, systems, or components had not been degraded through modifications; and, (2) modifications performed during increased risk configurations did not place the plant in an unsafe condition.

For the selected modifications, the inspector reviewed the design inputs, assumptions, and design calculations, such as instrument set-point and uncertainty calculations, to determine the design adequacy. The inspector also reviewed: (1) clarification request forms (field changes) that were issued during installations to determine proper installations of the components; and, (2) post-modification testing and instrument calibration records to determine the readiness for operations. Finally, the inspector reviewed the affected procedures, drawings, design basis documents (DBD), and UFSAR sections to verify that the affected documents were appropriately updated.

For the accessible components associated with the modifications, the inspector also walked-down the systems to detect possible abnormal installation conditions.

The following modification packages were reviewed:

FPX-97-12709-F	Installation of Fiber Optic Cable Through Vapor Containment Electrical Penetrations;
FMX-99-12055-M	Containment Recirculation Pumps Replacement;

FIX-97-12592-I	Accumulator Level Setpoint Change and Removal of Current Repeaters;
FPX-97-91612-F	Replacement of Inlet and Outlet Service Water Piping for Iso-phase Bus Heat Exchangers;
FIX-96-12110-1	Replacement of EDG Low Lube Oil Pressure Switches;
FPX-98-13131-F	PORV Nitrogen Accumulator Volume Upgrade;
FMX-97-12705-M	Steam Generator Snubber Reduction;
FPX-98-13139-F	Replacement of Level Instrumentation for Boric Acid Tanks;
FMX-97-12533-M	Replacement of Auxiliary Feedwater Control Valves;
FEX-96-11874-E	Control Room Ventilation Mode Upgrade.

Of the ten plant modifications reviewed, one modification was in the initiating event cornerstone, seven were in the mitigation cornerstone, and two were in the barrier integrity cornerstone.

b. Findings

No significant findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope (71111.19)

The inspector reviewed post-maintenance test (PMT) 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 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:

- PT-EM13, Control Room Pressurization Test (CR 200203166)
- SOP 31.2.2, Gas Turbine 2 Local Operations (CRs 200201685, 200202758 and 200202774)
- PT-M21B, 22 Emergency Diesel Generator Test following preventative maintenance
- WO 02-25827, Safety Injection valve 743, residual heat removal (RHR) minimum flow stop valve

For PT-EM13, the inspector verified that the licensee demonstrated the control room environment would be controlled in accordance with the design requirements with either a positive pressure or acceptable air in-leakage during postulated accident conditions. The inspector verified control room habitability was assured despite the deficiencies the licensee identified in the control envelope during this period (reference Section 1R5.2 of this report). The inspector verified that the licensee took appropriate actions to improve the control room ventilation envelope such as locating and sealing sources of uncontrolled air in-leakage. The inspector verified the testing demonstrated that the effects of the repairs improved the control room ventilation envelope.

b. Issues and Findings

No significant findings were identified.

1R22 Surveillance Testing

.1 Routine Surveillance Test Observations

a. Inspection Scope (71111.22)

The inspector reviewed surveillance test procedures and observed testing activities to assess whether 1) the test preconditioned the component tested, 2) the effect of the testing was adequately addressed in the control room, 3) the acceptance criteria demonstrated operational readiness consistent with design calculations and licensing documents, 4) the test equipment range and accuracy was adequate and the equipment was properly calibrated, 5) the test was performed per the procedure, 6) the test equipment was removed following testing, and 7) test discrepancies were appropriately evaluated. The surveillance observed was 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 activity was reviewed: PT-Q26C, 23 Service Water Pump, 2/19/02.

b. Issues and Findings

No significant findings were identified. An NRC identified human performance error is further documented in report detail 4OA2.

.2 Surveillance of Safety Barriers

a. Inspection Scope (71111.22)

The inspector reviewed surveillance test procedures and observed testing activities used by the licensee to verify the integrity of plant safety barriers. The inspector reviewed the following activities:

- Map 15FC16, Power Distribution and Hot Channel Factor Determination at 99.9% power and 12984.7 MWD/MTU, February 5, 2002
- Map 15FC17, Power Distribution and Hot Channel Factor Determination at 99.8% power and 13968.53 MWD/MTU, March 5, 2002
- Primary to Secondary Leak Rate per IPC-A-110S, Revision 9 (CR)

For core power distribution measurements, the inspector reviewed the licensee's actions to trend the Cycle 15 core peaking factors, including the maximum nuclear enthalpy rise hot channel factor ($F_{\Delta}H$). The inspection verified that the hot channel factors remained within the Technical Specification 3.10.2 limits.

The inspector reviewed licensee actions to monitor primary to secondary system leakage and to report leak rate determinations to plant management. The inspector reviewed the bases for licensee changes to procedures used for leak rate determinations. The licensee changed procedure IPC-A-110-S, Primary to Secondary Leak Rate, Revision 9, to define the value at which steam generator leakage is considered to be reactor coolant leakage; and, the licensee improved the estimate of reactor coolant system total gas activity used in the primary to secondary leak rate calculations.

The inspector noted that condenser air ejector activity slowly increased in proportion to increases in reactor coolant system gas activity caused by a minor fuel defect from what is estimated to be one pin in a twice burned fuel bundle. The inspector verified that the estimated leak rate correlated from condenser air ejector activity remained generally constant, consistent with past values and well below the minimum reliable detection sensitivity of 0.5 gallons per day. The inspector independently reviewed licensee estimates of dose rates from releases via the condenser air ejector pathway, which remained essentially zero with a calculated value that was a very small fraction of the Part 20 limits.

b. Issues and Findings

No significant findings were identified.

1EP6 Drill Evaluation

a. Inspection Scope (71114.06)

During the month of March 2002, the licensee revised the emergency procedures for Units 2 and 3 to require site-wide (Units 2 and 3) personnel assembly and accountability when either unit declares a Site Area emergency. On March 8, 2002, Entergy conducted an accountability drill to demonstrate that onsite radiation workers at the Indian Point Site could be accounted for within 30 minutes as committed by Section 6.4.1.d of the Emergency Plan and IP2 Implementing Procedure 1027 section 5.1.2.f, and IP3 E-Plan implementing procedure 1050. The procedure revisions were issued on March 6, 2002, and an accountability drill was conducted on March 8, 2002. The inspector observed the March 8 drill and reviewed the drill critique report.

b. Issues and Findings

The licensee completed site wide accountability in 38 minutes for this first-time site wide accountability drill. The NRC concluded that the intent of planning standard contained in 10 CFR 50.47(b)(10) was met for this untimely accountability. The basis was minor unit specific coordination issues as identified in Entergy's preliminary critique as documented in CR 200202580. The root cause analysis associated with CR 200202580 was scheduled to be completed within 30 days and a follow-up accountability drill was scheduled for April 8, 2002. This issue is unresolved pending NRC review of quality of the final critique for the March 8 and April 8, 2002 accountability drills and verify the completion of the root cause analysis associated with CR 200202580. The results of the NRC review will be evaluated in the significance determination process in accordance with NRC manual chapter 0609 Appendix B. **(URI 05000247/2002-02-03)**

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope (71121.01)

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

The inspector observed activities at the routine radiologically-controlled-area (RCA) access control point on several occasions to verify compliance with requirements for RCA entry and exit, dosimetry placement, and issuance and use of electronic dosimeters. The inspector toured and observed activities in Unit 2 in the primary auxiliary, fuel storage, and the maintenance and outage buildings. The inspector also toured the fuel handling floor of the fuel handling building and the 33-foot elevation of the annulus area between the containment and the sphere in Unit 1 to observe radiological conditions in the vicinity of the sphere's sand cushion and the annulus moat which receives the discharge from the north curtain drain. The inspector noted that the condition of the Unit 1 north curtain drain was identified by the station as one of their top twelve technical issues. The inspection also included observation of three fenced radioactive material storage areas and the building housing the four retired steam generators outside the protected area. During these observations and tours, the inspector reviewed for regulatory compliance 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 selectively examined the following procedures, records, and other program documents.

- Procedure SAO-140, Indian Point self-assessment program, Rev. 5

- Procedure SAO-300, Radiation protection plan, Rev. 14
- Procedure SAO-301, Personnel dose monitoring program, Rev. 14
- Procedure SAO-302, Radiation work permits (RWP) program, Rev. 17
- Procedure SAO-304, Radiological boundary controls, Rev. 22
- Procedure SAO-315, Radiation protection program review and evaluation, Rev. 15
- Procedure HP-SQ-3.109, Control of high radiation, locked high radiation, special locked high radiation, and very high radiation areas, Rev. 27
- Procedure RS-8.002, Skin dose assessment, Rev. 6
- Skin dose assessment for CR 200200697 (Cl-36 contamination), January 2002
- Quarterly reviews of the radiation protection program per SAO-315 for each of four quarters in 2001
- Procedure QAA-Q-17.201, Independent oversight program, Rev. 2
- Quality assurance audit program/functional area independent oversight plan/radiation protection, February 2002
- Independent oversight schedule for February 2002
- Assessment Report No. 01-AR-34-RP, Radiation Protection, 12/17/01-01/04/02
- Safety evaluation related to amendment number 221 to license number DPR-26

The inspection included a review of six Condition Reports (CRs)(i.e., CR 200111507, 200111690, 200111932, 200200156, 200200528, and 200200697) for the appropriateness and adequacy of event categorization, immediate corrective action, corrective action to prevent recurrence, and timeliness of corrective action.

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

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope (71121.02)

The inspector reviewed the effectiveness of ALARA (As Low As is Reasonably Achievable) planning and control.

The inspector reviewed the projected and actual person-rem results for 2001 and the person-rem estimate and dose budgeting for 2002. The actual cumulative exposure for 2001 was 22.5 person-rem versus the station exposure goal of 23. The actual 22.5 person-rem for 2001 was a historical low for the station. The person-rem estimates for 2002 included 120 for the scheduled Unit 2 refueling outage, 15.37 for Unit 2's routine operations, and 0.63 for Unit 1's routine operations, for a total of 136 person-rem for the year. The inspector also reviewed the following procedures, records, and documents for regulatory compliance and for adequacy of control of radiation exposure:

- Procedure SAO-303, ALARA program, Rev. 11
- Procedure SAO-305, Station ALARA committee, Rev. 9

- ALARA reviews (pre-job and in-progress)(No. 01-012) for RWP 01-02-51, Work on Unit 2 mid cycle outage evolutions except for work in transfer canal and cavity liner, Rev. 3
- ALARA reviews (pre-job, in-progress, and post-job)(No. 01-014) for RWP 01-02-52, Work in Unit 2 transfer canal and cavity liner
- Minutes for station ALARA committee meeting No. 2001-4 on December 5, 2001
- Annual ALARA summary report for 2001

b. Findings

No significant findings were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope (71121.03)

The inspector reviewed the effectiveness of health physics instrumentation, installed radiation monitoring instrumentation, and the program to provide self-contained breathing apparatus (SCBA) to occupational workers. The inspector reviewed the calibration program for health physics instrumentation and for installed radiation monitoring instrumentation to determine the accuracy and operability of the instrumentation.

During plant tours, 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, and continuous air monitors. The inspector conducted a review of the instruments observed in the toured areas, specifically verification of current calibration, of appropriate source checks, and of proper function. The inspector also reviewed activities in the health physics counting room and in the dosimetry office. The inspector evaluated the following procedures for regulatory compliance and adequacy.

- Procedure HP-9.002, Operation of the Eberline AMS-2/AMS-3 air monitors, Rev. 4
- Procedure HP-9.031, Operation of the Johnston Laboratories Triton Model III, Rev. 3
- Procedure HP-9.032, Operation of the Johnston Laboratories Triton Model 955B, Rev. 5
- Procedure HP-9.033, Operation of the Johnston Laboratories Triton Model 1055B, Rev. 4
- Procedure HP-9.038, Operation of the NMC AM-2B and AM-3B CAM, Rev. 4
- Procedure HP-9.593, Calibration and operation of the Eberline gamma tool monitor (GTM), Rev. 1

During plant tours, the inspector identified and noted the condition and operability of selected installed area and process radiation monitors and any accessible local response information on those monitors. The inspector noted that radiation monitor reliability was identified by the station as one of their top twelve technical issues. The inspector also interviewed the system engineer for the installed radiation monitoring system and reviewed for compliance and adequacy the following calibration records for installed radiation monitors.

- Calibration record PC-R15B for containment area radiation monitors R2 and R7, October 2001
- Calibration record PC-EM30 for process radiation monitors R-41 and R-42, January/February 2002
- Calibration record PC-2Y23-47 for liquid process radiation monitor R-47, February 2002

The inspector reviewed the adequacy of the program to provide SCBA for entering and working in areas of unknown radiological conditions. The inspection included a review of the status and surveillance records of SCBA air bottles and of SCBA with air bottles attached, all staged and ready for use in the plant.

The following documents were examined in the course of this review for regulatory compliance and adequacy.

- SCBA inspection/inventory record for January 2002
- SCBA spare tank inspection/inventory record for January 2002

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

b. Findings

No significant findings were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Data Collecting and Reporting

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 NEI 99-02, Revision 1, "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 Scrams Per 7,000 Critical Hours

a. Inspection Scope (71151)

The inspector reviewed the Performance Indicator (PI) for Unplanned Scrams Per 7,000 Critical Hours between the 3rd quarter of 2000 until the 4th quarter of 2001. The PI remained in the green band. The inspector reviewed operator logs, licensee event reports, and monthly operating reports to compare to reported data from the licensee.

b. Issues and Findings

No significant findings were identified.

.2 Emergency AC Power System Unavailability

a. Inspection Scope (71151)

The inspector performed a periodic review of 2nd, 3rd, and 4th quarters of 2001 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 "Regulatory Assessment Performance Indicator Guidance," NEI Report 99-02, Revision 1, to calculate the ratio of the number of hours the emergency AC power system was unavailable to the number of hours it was required.

b. Issues and Findings

No significant findings were identified.

4OA2 Human Performance Cross Cutting Issues

.1 Operator Error During Service Water Surveillance

a. Inspection Scope (71153)

On February 19, 2002, during an observation of the quarterly surveillance test (PT-Q26C) for the 23 service water pump, the inspector noted that non-licensed operators did not maintain manual valve locking devices in accordance with check off list (COL) 24.1.1. The surveillance procedure requires that the five test stop valves on the alternate service water pumps (SWN-501, 502, 503, 505, and 506) be verified closed; they are normally locked closed pursuant to Check off list (COL) 24.1.1, "Service Water and Closed Cooling Water Systems." The operators removed the locking devices and check closed the stop valves. Upon restoration from the surveillance test, the operators did not reinstall the locking devices on the five test valves. The locking devices were not installed until subsequent licensee follow-up actions in response to the inspector's observation.

b. Issues and Findings

NO COLOR. Not properly maintaining locking devices on the five service water test stop valves (SWN-501, 502, 503, 505, and 506) on February 20, 2002 was a violation of Technical Specification 6.8.1.a, that requires, in part, written procedures be implemented for activities referenced in Appendix "A" of Regulatory Guide 1.33, Rev. 2. Appendix A includes the requirement for item "3m", "Operating Procedures for the Service Water System." The issue was more than minor since the locking devices were not installed on five service water valves for approximately 12 hours. If the valves had become repositioned there could have been an impact on both the essential and non-essential service water headers, with a potential to degrade service water flow to the emergency diesel generators and other safety equipment. No actual consequences resulted from this lapse in configuration control. This issue was in the corrective action system as CR 200201923 and 200201955. This violation is being treated as a Non-cited Violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR25368) (**NCV 50-247/02-02-04**). The operator error is an example of the human performance errors as documented in past inspection reports.

.2 Inconsistent application of Technical Specifications during Station Auxiliary Tap Changer Maintenance

a. Inspection Scope (71153)

The inspection scope was to evaluate operator actions in accordance within the Technical Specifications during routine maintenance on the 138 KV station auxiliary transformer tap changer.

b. Issues and Findings

GREEN. The operators did not appropriately consider the 138 kilovolt off-site power system inoperable during scheduled maintenance on the station auxiliary transformer (SAT) tap changer. On February 28, 2002, for approximately 51 minutes, control room operators had placed the SAT tap changer in manual and local control in accordance with system operating procedure (SOP) 27.1.7, "Operation of Main, Station and Unit Auxiliary Transformers," section 4.8. The scheduled maintenance was not intrusive into tap changer operation, however, the licensee had not fully evaluated if the intended function could be maintained with operator compensatory actions to restore the tap changer to automatic. The limiting condition for operation in technical specification 3.7.B.3 for a loss of the 138 kilovolt power system is 24 hours, which was not exceeded during this scheduled maintenance activity.

Past NRC inspections (50-247/99-08, -13, and -14) identified that the SAT tap changer was left in manual for approximately one year, which contributed to a complicated reactor trip on August 31, 1999, resulting in a challenge to the emergency on-site power sources. The licensee corrective actions for the August 1999 event and the associated NRC enforcement actions was to change operating procedures SOP 1.3, SOP 27.1.1, and SOP 27.1.4 to require the SAT tap changer be operated in automatic, and for the operators to enter the TS 3.7.B LCO when the SAT was operated in manual. The SOP changes did not include SOP 27.1.7 section 4.8, which resulted in inconsistent operability guidance for the 138 KV off-site system during scheduled SAT tap changer maintenance.

The inconsistent SOP guidance resulted in inconsistent operator license actions. In January 2002 operators appropriately entered TS 3.7.B.3 during the same preventative maintenance on the SAT. In February 2002, the operators did not appropriately consider the 138 kilovolt off-site power system inoperable during scheduled maintenance on the station auxiliary transformer (SAT) tap changer. The issue had a credible impact on safety. Inconsistent control of the SAT tap changer impacts the initiating event cornerstone in that a loss of off-site power is more likely following a reactor trip. This issue was determined to be of very low safety significance (Green) using phase one of the SDP because no reactor trip occurred during the inspection period and no mitigating systems were directly impacted by the maintenance on the SAT tap changer.

.3 Inappropriate Use of a Security Weapon

a. Inspection Scope (71153)

The licensee responded to a security event on February 23, 2002, involving horseplay by a security officer with a duty weapon (Condition Report 200202061). The inspector responded to the site and reviewed the licensee’s immediate actions with the officers involved with the incident and to address licensee management expectations on the proper use of weapons with members of the guard force. The inspector also reviewed the licensee’s subsequent actions to investigate the incident and take long-term corrective actions. The event was an isolated incident. There were no violations of NRC requirements. The individual who used the weapon inappropriately was remanded to offsite civil authorities and appropriate on-site actions were taken.

b. Issues and Findings

No significant findings were identified.

4OA3 Identification and Resolution of Problems

.1 Problem Resolution for Modifications and Safety Evaluations

a. Inspection Scope (71152)

The inspector reviewed problem reports (condition reports) associated with 10 CFR 50.59 issues and plant modification issues to ensure that the licensee was appropriately identifying, evaluating, and correcting problems associated with these areas. The inspectors also reviewed three self-assessments related to 10 CFR 50.59 and plant modification activities at Indian Point 2.

The inspectors also reviewed the seven condition reports which the licensee issued after the inspectors identified four minor issues in the areas of post-modification testing and documentation update. The review was to determine whether the licensee had appropriately described the problem and entered the problem into their corrective action program.

b. Issues and Findings

No significant findings were identified.

.2 Manual Service Water Valve Maintenance (71152)

a. Inspection Scope

The inspector reviewed the past corrective action associated with a 10 inch butterfly valve (SWN-7), which isolates the service water supply to turbine building loads, provides isolation between the essential and non-essential loads, and preserves service water header independence.

b. Issues and Findings

GREEN. The operator valve gear for SWN-7 failed during operation on March 9, 2002 (reference Section 1R13 above). The sector gear for the valve operator previously failed on April 14, 2000, January 31, 1999, October 21, 1997, May 31, 1996, January 12, 1996, and November 9, 1995. The licensee had evaluated SWN-7 based upon engineering calculation PGI-00384-00, “IP2 Buttery Valve SWN-7 “Manual Actuator Worm Gear Sector Teeth Analysis,” which concluded that the valve has a small design margin for the gear teeth in the manual actuator. The licensee issued engineering work request 12110-99 to replace the gear set in SWN-7 and similar valves with high strength material. On July 12, 1999, engineering work request 12110-99 was canceled and no action was taken on SWN-7. This was a missed opportunity for the licensee to have corrected a design deficiency and prevent repeat failures of a plant safety system component. This issue was determined to be of very low significance since the specific failure on March 9 and corrective actions occurred within the limiting condition for operation for the service water system, and no operating or stand-by mitigating equipment supported by service water was called to perform its intended function.

Contrary to 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” the licensee failed to take appropriate corrective action to preclude repeat failures of service water valve SWN-7. The specific issue was entered into the corrective action program as CR 200202589. The violation is being treated as a non-cited violation consistent with Section VI.A. of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368) **(NCV 50-247/2002-02-05).**

4OA4 Licensee Event Report Reviews and Inspection Item Follow-up

.1 (Closed) LER 05000247/2001-07-00: Automatic Reactor Trip Initiated by a Main Turbine Trip on Auto Stop Oil. The inspector reviewed the information the licensee provided to describe and analyze this event. The event was reviewed in NRC Inspection 50-247/2001-11. The LER accurately described the event. This LER is closed.

.2 (Closed) LER 05000247/2002-01-00: Personnel Error Results in Two Emergency Diesel Generators Inoperable. The inspector reviewed the information the licensee provided to describe and analyze this event. The event was reviewed in NRC Inspection 50-247/2001-14. The LER accurately described the event. This LER is closed.

.3 (Closed) UNR 05000247/2001-08-01: pertaining to the adequacy of procedural actions within abnormal operating instruction (AOI) 28.0.4, “Plant Flooding - Conventional Side,” to mitigate a turbine building flood, and functionality of door flaps to mitigate a potential flood in the primary auxiliary building and auxiliary feed pump building.

The inspector confirmed that additional guidance was added to AOI 28.0.4 to provide improved indicators of a turbine flood in addition to the unit 1 condenser pit sump level alarms. The inspector verified that a log now requires the operator to exercise the door flaps for the auxiliary feed pump and the primary auxiliary buildings monthly. The licensee concluded it had met its commitments to NRC concerning flooding outside containment and the above actions provided necessary improvements. No violations of NRC requirements were identified.

4OA7 Licensee Identified Violations

The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as Non-Cited Violations.

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 2002-002-06	10 CFR 50 Appendix B, Criterion III, “Design Control”

10 CFR 50 Appendix B, Criterion III requires in part, that measures be established for the identification and control of design interfaces and for coordination among participating design organizations. The licensee did not ensure that the pressurizer level instrument drift evaluations were consistently bounded by the assumed instrument uncertainty within the safety analysis for a postulated Loss of Normal Feedwater event

and a Loss of Offsite power event. The licensee documented this issue in condition report 2002000313.

If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region 1: the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Indian Point 2 Station.

4OA8 Meetings

Exit Meeting Summary

Senior NRC and Entergy management met for the IP2 annual performance assessment meeting at 7:00 p.m., March 14, 2002, in Peekskill, NY. The meeting was open to the public. The purpose was to review IP2 performance for the period April 1, 2001 through December 31, 2001. During the meeting, Entergy's management discussed progress in implementing the Fundamentals Improvement Plan (FIP). The meeting provided a useful exchange of information about Entergy's progress to date on the FIP, including a review of its performance indicators.

On April 8, 2002 , the inspector presented the inspection results to Mr. Fred Dacimo, Mr. John Herron, and other members of the licensee staff, who acknowledged the findings. No materials examined during the inspection were considered proprietary.

ATTACHMENT 1

a. Key Points of Contact

R. Allen	Manager, Regulatory Affair
P. Asendorf	Security Manager
T. Burns	Supervisor, Environmental Monitoring
R. Colville	Environmental Monitoring Technician
K. Cullen	Health Physics Technician
F. Dacimo	Vice President, Operations
M. Dampf	Health Physics Manager
M. DiGenova	Senior System Engineer
M. Donegan	Health Physics Manager
N. Ertle	I&C Engineer
D. Gately	Radiation Protection Coordinator
L. Glander	Supervisor, Radiological Support
P. Griffith	Safety Assessment Engineer, Safety Evaluation
V. Jayaraman	I&C Manager
T.R. Jones	Senior Engineer, Nuclear Safety and Licensing
K. Kuran	Specialist, System Engineering
R. Louie	Senior Engineer, Nuclear Safety and Licensing
B. Marguglio	Principal Engineer, Quality Assurance and Licensing
R. Majes	Radiological Support Health Physicist
T. McCafferty	System Engineering Manager
J. McCann	Manager, Nuclear Safety and Licensing
M. Miele	Radiation Protection Department Manager
P. Milewski	Health Physics Technician
M. Miller	Manager, Generation Support
D. Morris	General Manager, Nuclear Quality Assurance and Oversight
V. Nutter	Radiological Support Manager
W. Osmin	Reactor Engineer
P. Parker	Maintenance Supervisor
J. Reynolds	Project Specialist, CAG
P. Rubin	Operations Manager
E. Salisbury	Radiological Engineer
H. Santis	Section Manager, Nuclear Project
G. Schwartz	Director of Engineering
D. Smith	Radiological Assessor, NQA
R. Sutton	Senior Engineer
L. Temple	Plant Manager
J. Tuohy	Manager, Design Engineering
M. Vaseley	System Engineer Supervisor
J. Ventosa	Engineering Manager
E. Woody	I&C Manager
G. Zolotas	Health Physics Technician

b. List of Items Opened, Closed, and Discussed

Opened

50-247/02-02-01	URI	Evaluation of Control Room Degraded Fire Barrier
50-247/02-02-02	URI	Evaluation of Onsite Storage of Hydrogen
50-247/02-02-03	URI	E-Plan drill did not complete worker accountability in 30 minutes

Opened and Closed During this Inspection

50-247/02-02-04	NCV	Failure to Follow Procedures for SW Valve Locking Devices
50-247/02-02-05	NCV	Inadequate Corrective Actions for Repeat Failure of SWN-7
50-247/02-02-06	NCV	10 CFR 50 Appendix B, Criterion III, "Design Control"

Closed

50-247/01-08-01	UNR	Action per AOI 28.0.4 to Mitigate a Plant Flood
50-247/2001-07	LER	Automatic Reactor Trip Due To Main Turbine Trip
50-247/2002-01	LER	Personnel Error Results in Two Inoperable Emergency Generators

c. List of Documents Reviewed

10 CFR 50.59 Safety Evaluations

99-177-MM	Installation of Fiber Optic Cable Through Vapor Containment Electrical Penetrations.
99-223-MM	Containment Recirculation Pumps Replacement.
99-267-MD	Accumulator Level Setpoint Change and Removal of Current Repeaters.
SE-99-190-MD	PORV Nitrogen Accumulator Upgrade, Revision 0.
SE-99-226-MM	Replacement of Inlet and Outlet Service Water piping for Iso-phase Bus Heat Exchangers, Revision 0.
SE-00-073-MM	IP-SG Snubber Reduction with Replacement Series 44F SGs, Revision 1.
SE-99-111-MM	Replacement of Level Instrumentation for Boric Acid Tanks 21 and 22.
SE-99-091-MM	Replacement of Auxiliary Feedwater Control Valves
SE-96-256-MD	IP-CCR Ventilation Mode Upgrade

10 CFR 50.59 Safety Screens

01-0499-EV-00-RS	Ultimate Heat Sink Update.
01-0598-EV-00-RS	Revise Basis for Technical Specification 3.7 to Clarify Emergency Diesel Generator Fuel Oil Discussion.
01-0497–SP-00-AD	Accumulator Level Transmitter Calibration and High/Low Level Setpoints Calculation.
01-0481-SP-00-RS	Auxiliary Feedwater Pump Instrument Accuracy for Recirculation Flow Control and ASME Section XI Testing.

01-0700-EV-00-AD	Revision to UFSAR Section 9.3.1.2.3 to Clarify Minimum Time Required for Moving Spent Fuel.
01-0442-MM-00-AD	Replacement of EDG Low Lube Oil Pressure Switches
01-0874-GM-00-AD	MOV Modification
01-0760-PR-00-AD	Main Steam Isolation Valves
01-0819-SP-00-AD	EDG Building Exhaust Fans Setpoint Change
01-0862-PR-00-AD	EDG Exhaust Fans Functional Test
02-0049-TR-00-AD	Temporary Leak Repair on LCV-1127C
01-0827-SP-00-RS	Setpoint Change for Degraded Voltage Timers
01-0897-MM-00-AD	Functional Removal of the BGIR/BGIS Circuit from EDG Control Circuits

Procedures

SAO-460	10 CFR 50.59 Reviews, Revision 14.
DE-SQ-12.513	Design Verification, Revision 0.
DE-SQ-12.502	Development and Review of Discipline Design Criteria, Revision 1.
DE-SQ-12.503	Obtaining Inputs to Conceptual Designs, Revision 1.
DE-SQ-12.512	Preparation and Approval of Plant Modification Packages and Review of Maintenance and Repair Packages, Revision 4.

Condition Reports

CR No. 200201717	Modification FPX-97-12709-F Revision
CR No. 200201760	Modification FPX-97-12709-F Testing Methodology and Acceptance Criteria
CR No. 200201726	FCV-405C I/P Converter As-found Output > 3X Tolerance
CR No. 200201727	FCV-405D I/P Converter As-found Output > 3X Tolerance
CR No. 200201728	FCV-405B I/P Converter As-found Output > 3X Tolerance
CR No. 200201757	FCV-405B&C Local Position Indicators Not Indicating Fully Closed
CR No. 200201689	Documentation of PORV Accumulator Volume Upgrade Modification
CR No. 200201756	Safety Evaluation Screening for Diesel Fire Water Pump
CR No. 200108406	Design Engineering 2001 Second quarter Self-assessment Findings
CR No. 200201756	Safety Evaluation Screening for Diesel Fire Water Pump
CR No. 200001054	AFW Nitrogen Back-up to Instrument Air Testing
CR No. 199908968	Accumulator Level Instrumentation Issues
CR No. 200006631	Accumulator Level Instrumentation Issues
CR No. 200002328	Emergency Diesel Loading Calculation
CR No. 200004114	21 Recirculation Pump Test Acceptance Criteria
CR No. 200003128	As Found Local Leak Rate Test Results for Electrical Penetration H46
CR No. 200003276	As Left Local Leak Rate Test Results for Electrical Penetration H46

Self Assessments

IP2 Nuclear Licensing Self Assessment -10CFR50.59 Assessor Function, August 2001.

Design Engineering Assessment Second Quarter 2001, August 8, 2001.

Engineering Self Assessment, Design Engineering, System Engineering, Maintenance/Construction, December 15, 2001.

Calculations

FMX-00036-04	Safety Injection Recirculation Pump Available Net Positive Suction Head, February 18, 2000
FIX-00030-02	Nitrogen Back-up System Capacity to Support Critical AFW System Air Users in the Event of a Loss of Instrument Air, March 2, 2000
FIX-00045-01	Accumulator Level Transmitter Re-calibration and High/Low Level Setpoint Change, September 21,1999

d. List of Acronyms

AFW	auxiliary feedwater
ALARA	As Low As Reasonably Achievable
AOI	Abnormal Operating Instruction
CFR	Code of Federal Regulations
COL	check off list
CR	Condition Report
DBD	design basis document
DBT	design basis threat
EDG	emergency diesel generator
FHA	fire hazard analysis
GT	gas turbine
GTM	gamma tool monitor
HRA	High Radiation Area
IPEEE	Individual Plant Examination of External Events
KV	kilovolt
LCO	limiting condition of operation
LHRA	Locked High Radiation Area
MR	maintenance rule
NCV	Non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OS	Occupational Safety
PARS	publicly available records
PMT	post maintenance test
PORV	power operated relief valve
PI	performance indicator
RCS	reactor coolant system
RCA	Radiologically Controlled Area
RHR	residual heat removal
RMS	Radiation Monitoring System
RWP	radiation work permit
SAO	station administrative order
SAT	station auxiliary transformer
SCBA	self-contained breathing apparatus
SI	safety injection
SOP	station operating procedure
SSC	structure, system and component
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report