December 17, 2001

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 50-247/01-10

Dear Mr. Dacimo:

On November 17, 2001, 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 November 28, 2001, 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 engineering and outage activities, and the effectiveness review of corrective actions taken for previously identified problems. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Since September 11, 2001, the Indian Point Nuclear Power Plant has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site. The NRC continues to interact with the Intelligence Community and to communicate information to Entergy Nuclear Operations, Inc. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

The NRC findings this period noted mixed performance in the effective resolution of problems. During the mid-cycle outage, you addressed deficiencies that could have impacted plant operations, which included replacing pressurizer safety valves to stop leakage to the pressurizer relief tank. The NRC also noted that you completed effectiveness reviews to address the specific equipment problems associated with the August 31, 1999 reactor trip. However, your self-assessments also noted the need for continued improvement in correcting problems and resolving underlying causes. Similarly, our inspection findings this period and in recent inspections identified the need for continued improvement in several areas. The operational issues that affected all three auxiliary feedwater pumps during the recent midcycle outage are examples of issues the station continues to face related to equipment reliability. Your actions to improve corrective action effectiveness, human performance and equipment reliability will continue to be reviewed in future inspections consistent with the intent for heightened oversight at Indian Point 2 as noted in the NRC Supplemental Inspection 95003 Multiple Degraded Cornerstone Report 05000247/2001-02.

Based on the results of this inspection, three violations of NRC requirements were identified regarding instances where there was failure to: 1) assure measuring and test equipment were properly calibrated, 2) provide adequate maintenance instructions for repairing safety related pumps, and 3) promptly identify conditions adverse to quality and take effective corrective actions for the recurrent failures of safety injection relief valve SI-855 and the low pressure steam dump valves. However, because of the very low safety significance of these issues as gauged by NRC reactor oversight program criteria, 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.html (the Public Electronic Reading Room).

Should you have any questions regarding this report, please contact Mr. Scott Barber at 610-337-5232.

Sincerely,

/RA/

Brian E. Holian, Deputy Director
Division of Reactor Projects

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

Enclosure: Inspection Report 50-247/01-10

Attachment 1 - Supplemental Information

cc w/encl: J. Yelverton, Chief Executive Officer
M. Kansler, Senior Vice President and CEO
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
T. Walsh, Secretary - NFSC
W. Flynn, President, New York State Energy Research and Development Authority
The Honorable Sandra Galef, NYS Assembly
County Clerk, Westchester County Legislature
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D. Lochbaum, Nuclear Safety Engineer, Union of Concerned Scientists
Public Citizen's Critical Mass Energy Project
M. Mariotte, Nuclear Information & Resources Service
E. Smeloff, Pace University School of Law
L. Puglisi, Supervisor, Town of Cortlandt
The Honorable Sue W. Kelly
The Honorable Benjamin A. Gilman
The Honorable Nita M. Lowey
The Honorable Hilary Rodham Clinton
The Honorable Charles E. Schumer

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	50-247/01-10
Licensee:	Entergy Nuclear Operations, Inc..
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	September 30, 2001 - November 17, 2001
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector Paul Kaufman, Senior Reactor Inspector Leonard Prividy, Senior Reactor Inspector Leonard Cheung, Senior Reactor Inspector Larry Scholl, Senior Reactor Inspector Alfred Lohmeier, Reactor Inspector
Approved by:	G. Scott Barber, Acting Chief Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000247-01-10, on 09/30 - 11/17/2001, Entergy Nuclear Operations, Inc.; Indian Point 2 Nuclear Power Plant. Operability Evaluations, Outage Activities, and Cross-Cutting Issues.

The inspection was conducted by resident and region-based inspectors. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP). This inspection identified three green and two no color issues. The “no color” significance level indicates that the IMC 0609 “Significance Determination Process” does not apply to these findings.

Cornerstone: Mitigating Systems

Green Entergy identified that measuring and test equipment (M&TE) were out of specification, and that condition reports were not consistently initiated to evaluate the impact of the out of specification M&TE on surveillance tests. Entergy’s engineering assessment concluded that the systems impacted by out of specification M&TE were operable. This issue was evaluated in phase 1 of the Significance Determination Process (SDP) and was found to have very low safety significance. A Quality Assurance Audit had previously recognized an inconsistent approach in the control of M&TE. Although a Business Plan performance improvement initiative exists for this area, progress was insufficient to prevent the observed problems. Contrary to 10 CFR 50 Appendix B criterion XII, the licensee had failed to assure that measuring and test equipment used in activities affecting quality were properly calibrated and adjusted to maintain accuracy within limits. This violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A. of the NRC Enforcement Policy.

Green During the plant shutdown for a mid-cycle maintenance outage on October 27, 2001, the operators experienced several problems with the auxiliary feedwater (AFW) system, which caused them to declare two motor driven pumps inoperable. Even though the auxiliary feedwater pumps were subsequently found to have been able to perform their intended safety function, the equipment operating deficiencies had a credible impact on the availability of the auxiliary feedwater system. The issue was evaluated in phase 1 of the SDP and was found to have very low safety significance.

Cross-Cutting Issues

Green The maintenance instructions used to repair the 21 AFW pump on July 16, 2001, were not adequate to pack the pump in accordance with a maintenance standard and vendor instructions. This resulted in poor packing performance and resulted in operators declaring the 21 AFW inoperable during the October 27 shutdown. Further, in 1998 the licensee identified the need to provide instructions on packing pumps to workers, but did not provide adequate information in the maintenance procedures. This issue had a credible impact on safety since a properly packed gland is necessary to ensure reliable AFW pump operation. However, since the maintenance errors did not result in packing failure and a subsequent evaluation concluded the 21 AFW pump could perform its safety function, this issue was determined to have very low safety significance in accordance with a SDP Phase 1 assessment. The failure to provide adequate maintenance instructions for work on safety related equipment was an example of a condition contrary to 10 CFR 50 Appendix B, Criterion V. This violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy.

No Color The results of the 2001 Licensee Operator Requalification (LOR) Program showed a high number of crew and individual failures during the simulator exams. The licensee's preliminary investigation found the exam failures were caused by inadequate corrective actions and insufficient implementation of corrective actions for licensed operator knowledge and performance weaknesses identified during previous year LOR exams. The licensee determined the presently observed performance deficiencies were previously identified but not adequately corrected, aspects of which contributed to degraded performance in two plant reactivity management events and configuration control events in 2001. The inspector noted a root cause of the LOR program results (inadequate corrective actions) was also evident in recent plant events and NRC findings. This was an example of a cross cutting issue regarding human performance and problem resolution. Inspection Report 50-247/01-13 provides additional details regarding licensed operator requalification weaknesses.

No Color The licensee's corrective actions in response to several equipment problems were ineffective. Repetitive failures of safety injection (SI) system relief valve, SI-855, and the low pressure steam dump valves were not prevented. Appropriate analyses were not performed to fully understand the causes for the past failures. In addition, items related to these equipment problems were not entered in the corrective action program for resolution. This is a recurrent example of deficiencies in problem identification and resolution. The failure to correct conditions adverse to quality is considered a Severity Level IV violation of 10 CFR 50, Appendix B, Criterion XVI. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the Enforcement Policy.

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Report Details

SUMMARY OF PLANT STATUS

The plant operated at full power until October 27, 2001, when the unit was removed from service and placed in a cold shutdown condition for a maintenance outage. Plant startup began with heating of the reactor coolant system above 200 degrees Fahrenheit (F) at 4:14 a.m. on November 1. The reactor was taken critical at 9:46 p.m. on November 3, and the plant returned to full power operations on November 7. The plant operated at full power for the remainder of the period.

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

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope (71111.04)

On October 27, 2001, the inspector performed a partial walkdown of the auxiliary feedwater (AFW) system to verify its operability during periods of degraded conditions on the two motor driven auxiliary feedwater pumps. The review was conducted to verify support systems and component alignments were proper. The inspector evaluated the impact of outstanding equipment deficiencies on system function, including the 23 AFW pump vibrations in the alert range (Condition Report 200108603); and excessive steam trap leakage during operation of the 22 AFW pump (Condition Report 200110297). The inspector reviewed the results of ALTRAN Calculation 97217-C-10 which analyzed the auxiliary feedwater pump operation following a postulated high energy line break.

b. Issues and Findings

No significant findings were identified.

1R11 Licensed Operator Requalification

.1 Observation of Simulator Operating Exams

a. Inspection Scope (71111.11)

The inspector reviewed the results of the 2001 Licensed Operator Requalification Program dynamic simulator operating exams, which were conducted from September 5 through October 24, 2001. The licensee identified a high number of crew failures during the simulator tests, and a significant number of individuals also failed. The inspector verified that two of the crews were operating crews, and that the crews and individuals were remediated prior to going back on shift.

The inspector observed the retest of one operating crew on October 19 , 2001, to assess the adequacy of the training, operator performance, and the licensee's critique.

The NRC observed the test of a second operating crew on October 22-24, 2001 to evaluate the adequacy of the training, operator performance, and the licensee's critique (reference NRC Inspection 50-247/01-13). The inspector reviewed the licensee actions following the test of a third crew on November 15, 2001.

The inspector also reviewed the short term corrective actions taken and the licensee bases for continued safe operation, which were provided in the Significance Level 1 Investigation Report for Condition Report 200110167 and in a letter to NRC dated November 5, 2001, respectively. The inspector reviewed the licensee actions to monitor plant startup and operating activities, conduct additional operator assessments, provide senior management presence on shift during startup activities, provide mentors for the Shift Managers, review present and past operator performance on the annual requalification exams, and take specific remediation measures for individual operators.

b. Issues and Findings

Within the scope of this inspection, no significant findings were identified. Inspection 50-247/01-13 provides the NRC findings and assessment of this area. See also Section 4OA2 of this report for a discussion of cross cutting issues related to human performance and problem resolution.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspector reviewed the licensee's assessment of the 23 charging pump, which failed a quarterly technical specification surveillance on October 1, 2001 (reference Condition Report 200109389). The pump failed to achieve the minimum acceptance criteria for flow at the reference speed of 1,108 revolutions per minute. The licensee classified the charging pump as a high safety significant component. The inspector verified that the performance criteria were appropriate, and evaluated the licensee conclusion regarding the function of the emergency boration flow path. The inspector evaluated the potential for similar failures on the remaining charging pumps and evaluated the proposed 50.65(a)(1) corrective actions.

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

- Condition reports (200109389, 200109475, 200106991, and 200106950)
- PT-Q33, "23 Charging Pump", revision 7
- IP-2 Maintenance Rule Basis Document for CVCS, revision 1
- Work Order NP 01-24007, Repairs to 23 Charging Pump
- SE-350, System Monitoring Basis Document, attachment 8.2, revision 0
- Maintenance Rule Unavailability for Charging Pumps (9/1999 - 10/2001)
- Chemical Volume Control System Design Basis Document (Section 5.2.1)
- Emergency Operating Procedure FR-S.1, "Response to Nuclear Generation/ATWS"

The licensee evaluation documented in CR 200109389 concluded that this degraded condition was not a functional failure since the pump did achieve the minimum required flow for emergency boration and was able to maintain pressurizer level during normal plant operations. The apparent cause of the degraded flow was mechanical wear on the internal valves and seats, which were replaced as the short-term corrective action. The licensee initiated corrective maintenance work orders to replace the internal valves and seats for the remaining two pumps since they were last replaced in 1990. The 23 charging pump is in maintenance rule (a)(1) and corrective actions to restore the system to (a)(2) were documented in CR 200106950. NRC Inspection Report 50-247/01-08 (section 1R12) documented past problems with the inability of the 23 charging pump to achieve the required flow.

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 evaluated the effectiveness of the risk assessments performed for emergent operational activities and verified how the licensee managed the risk in accordance with 10 CFR 50.65(a)(4). The inspector verified that the licensee took the necessary steps to plan and control emergent work activities, took actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems. The inspector discussed the risk management with maintenance and operations personnel for the following activity: OTDT#4 Spiking and QM-441A Replacement (WO 01-23580, CR 200110102).

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:

- Emergency Diesel Air Start System Design Deficiency (CR 200110099)
- OTDT#4 Setpoint Operability (CR 200109719, WO 01-23900)
- Reactor Coolant Pump Seal Return Flow (CR 200110125)
- Auxiliary Feedwater Pump Problems During Shutdown for Mid-Cycle Outage

(Condition Reports 200110289, 200110290, 200110293, 200110297)

- Operability Determination 01-006, “Out of Tolerance Measuring and Test Equipment Impacting Reactor Protection and Engineered Safeguards System Functions” (CR 200110493)

Out-of-Calibration of Measurement and Test Equipment

During a recent self-assessment, Entergy noted that a number of measuring and test equipment (M&TE) were found to be out of specification during offsite calibrations by a vendor, and that condition reports were not consistently initiated when deficiencies were identified. Thus, the impact of the out of specification M&TE on surveillance tests had not always been evaluated. The licensee performed an extent of condition review to identify the systems that were impacted by out of specification M&TE, and identified 31 surveillance tests that could potentially impact operability of components in the reactor protection, engineered safety features, safety injection, effluent radiation monitors, and 480 volt systems. The licensee completed an engineering evaluation to determine whether the systems were operable.

The inspector selected a sample of suspect surveillance tests for the safety related equipment listed below and independently determined whether the out of specification M&TE impacted system operability.

- PT-Q62, High Steam Flow Safety Injection Bistable (FC-419A)
- PC-R17A, Safety Injection Accumulator Level Calibration
- PT-Q52, Overpower and Overtemperature Delta Temperature Functional
- PC-EM8, Auxiliary Feedwater Flow Transmitter Calibration
- ICPMs associated with three emergency diesel generators
- 480 Volt Safeguards Breakers Amptector Calibrations
- PC-R1B, Reactor Coolant System average temperature and delta-T average temperature calibration

b. Issues and Findings

(GREEN) Entergy concluded that all systems impacted by out of specification M&TE were operable and able to perform their intended safety function. The basis for operability varied depending on the specific surveillance, but all systems were operable based on one or more of the factors listed below. The inspector independently confirmed the licensee’s bases were acceptable.

- the M&TE was found acceptable during a “spot” check with a more accurate instrument prior to use in the surveillance
- the out of tolerance M&TE instrument range was not the range used in the surveillance
- when corrected for the M&TE error, the “as-left” instrument data was still within the surveillance procedure acceptance criterion
- the M&TE error was bounded by instrument drift analysis
- the M&TE error was bounded by the instrument setpoint error calculation

- the M&TE recommended tolerance was less than the required surveillance tolerance and the actual error between the two tolerances

As documented in Quality Assurance Audit 98-08-F, M&TE and Operation Gauges, the licensee had previously recognized a fragmented and inconsistent approach in the control of M&TE instruments. The 2001 Business plan contained actions to develop a station wide M&TE calibration program staffed by a single group and to develop program procedures for implementation. At the time of the inspection all of the M&TE instrument calibrations were being performed by a vendor offsite. The specific actions within this Business Plan item were in progress but were significantly behind the expected completion date.

10 CFR 50 Appendix B, Criterion XII, requires that measures be established to assure that instruments and other measuring and testing devices used in activities affecting quality are properly calibrated to maintain accuracy within necessary limits. Contrary to this requirement, the measuring and test devices used in thirty one surveillance tests for safety related systems were not in calibration to support the testing. This issue had a credible impact on safety since, if left uncorrected, mis-calibrated measuring and test equipment could affect the operability and availability of mitigating systems. However, since the mis-calibrated measuring and test equipment did not impact the operability of the affected systems, this issue has been determined to have very low safety significance in accordance with the NRC Significance Determination Process, Phase 1 assessment **(Green)**. This violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A. of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368) **(NCV 50-247/01-10-01)**. The licensee initiated condition report 200110493 to address this problem in the corrective action program.

1R19 Post Maintenance Testing

a. Inspection Scope

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:

- PMT-0123900, Overpower Delta-T Testing per PC-R1B (WO 01-23900)
- OAD-40, Troubleshooting 21 Auxiliary Feedwater Pump Following Repacking
- PMT-23580, Overpower Delta-T Testing per PC-Q51 (WO 01-23580)

b. Issues and Findings

No significant findings were identified.

1R20 Plant Shutdown, Maintenance Outage and Startup Activities

.1 Outage Risk Evaluation

a. Inspection Scope (71111.20)

The inspector reviewed licensee plans to conduct a mid-cycle maintenance outage to address conditions that could impact plant operations. Some of the major equipment issues addressed included: replacement of the three pressurizer safety valves; repair of the 22 condenser flexible joint; repair of the main generator hydrogen cooler leak, repair of the individual rod position indication for control rod L-3; preventive maintenance and repair of the electrical system including a ground on the normal supply breaker to safety related Bus 6A, and, numerous secondary plant equipment problems. The replacement of the pressurizer safety valves with valves having a flexible seat eliminated a source of minor leakage from the reactor coolant system that had been observed since the plant started up in January, 2001 (reference Section 1R20.3 below and Inspection 50-247/00-15, Section 1R15). Prior to the mid-cycle outage the inspector independently evaluated outage risk. The inspector referenced the following documents to support the outage risk evaluation:

- Operations administrative directive (OAD)-38, “Outage Risk Management”
- Vertical Outage Schedule as of October 25, 2001
- Abnormal Operating Instruction (AOI) 32.1, “Response to Security Compromise”
- Updated Final Safety Analysis Figures 8.2-6, -9A, -11

b. Issues and Findings

No significant findings were identified.

.2 Plant Shutdown

a. Inspection Scope (71111.20)

The inspector observed control room and plant activities during the plant shutdown. The inspector verified the operators took timely and appropriate actions per procedures E-0 and ES-0.1 when the reactor was manually scrammed at 12:01 a.m. on October 27 as part of the normal shutdown sequence. The inspector observed the operators conduct the shutdown using procedures POP 3.1, (Plant Shutdown, Revision 37), and POP 3.3, (Plant Cooldown, Revision 53). The inspector observed the operators respond to changing equipment conditions with the use of alarm response procedures. The operators experienced several problems with the auxiliary feedwater (AFW) system after the reactor was manually scrammed, as described below. The reactor was cooled down below 350 F using the AFW system until the RHR was placed in service. The plant entered cold shutdown with the RCS less than 200 F at 12:15 p.m. on October 27.

b. Issues and Findings

(GREEN) During the plant shutdown for a mid-cycle maintenance outage, all three AFW pumps (the 21 motor driven pump, the 23 motor driven pump, and the 22 turbine driven pump) were operated and all had operational problems:

- the 21 AFWP was secured after operating for 38 minutes when packing gland temperatures increased to 500 Fahrenheit (F). The operators declared the pump inoperable and entered a 72 hour technical specification limiting condition for operation (CR 200110289);

- the 22 AFW pump was operated after the 21 AFW pump was shutdown and the operator had to add oil to the pump inboard bearing (CR 200110297);

- after running for two hours, the 23 AFW pump outboard bearing heated up to 192 degrees F until the operator valved in supplemental cooling from the city water (CW) system to stabilize temperatures at about 150 degrees F. Since the city water supply lacked seismic qualification, the operator declared the 23 AFW inoperable even though it remained in service, and entered a 12 hour shutdown limiting condition for operation (CR 200110293; and,

- the operators opened the AFW room roll up door after steam leakage from the 22 AFW pump casing and steam traps entered the room from the floor drains (CR 200110297).

The AFW system supplies feedwater to the steam generators to remove reactor decay heat and is the most risk-significant system at Indian Point 2. Entergy's initial risk assessment with the 21 and 23 AFW pumps inoperable placed the plant in an elevated risk condition. NRC resident inspector coverage was maintained through the night during plant cooldown. Since the 22 and 23 AFW pumps provided sufficient decay heat removal, the actual plant risk condition was much less (with only the 21 AFW pump removed from service). Additionally, the operators did not isolate the 21 AFW pump to ensure it remained available. Further, the operators maintained the main feedwater system and condensate pumps available during the plant shutdown to assure a backup means of providing feedwater to the steam generators remained available.

Entergy made a report to the NRC per 50.72(b)(3)(v) on October 27 because both motor driven AFW pumps were declared inoperable (Event No. 38433). Entergy withdrew the 50.72 report on November 5, 2001, after further engineering, maintenance and operability reviews determined that both motor driven auxiliary pumps were functional and capable of performing their safety functions. Entergy concluded based on a review of the "as-found" packing conditions that the 21 AFW pump would have operated satisfactorily and the operators would have used the pump if needed to remove reactor decay heat. Entergy demonstrated that the AFW pump bearing temperatures were acceptable without CW cooling. However, AFW pump operation with CW cooling would reduce bearing temperature and improve pump reliability. Operation of the AFW pumps with CW was acceptable since the portions of the CW system inside the AFW room are seismically qualified, which precludes concerns about flooding or other impact on pump operation. Entergy revised the operating procedures to use city water to supplement bearing cooling on the motor driven AFW pumps, but plans to isolate the cooling supply during the quarterly surveillance testing so as not to "mask" bearing issues. Entergy concluded that the steam leakage into the AFW room would not have prevented operator actions nor adversely impact safety system performance. Finally, Entergy concluded that an oil leak in the sight glass for the 22 AFW pump would not have impacted the pump operability.

Although the auxiliary feedwater pumps were subsequently found to have been able to perform their intended safety function, the event had a credible impact on the availability of the auxiliary feedwater system, which is a mitigating safety system. The impact on availability occurred when the high packing gland temperatures caused the operator to secure the 21 AFW pump. The licensee planned to address the deficiencies needed to support plant operations made evident by the event sequence in the long term corrective actions for Condition Report 200110289. This issue was evaluated in phase 1 of the SDP and was found to have very low safety significance **(Green)**. A cross-cutting issue in the area of human performance related to the maintenance activity to repack the AFW pumps, is documented in Section 4OA2 of this inspection report.

.3 Maintenance Outage Activities

a. Inspection Scope (IP 71111.20)

The inspector reviewed several work activities that occurred during the recent maintenance outage to verify compliance with station procedures. The inspector reviewed these activities as they related to any cross-cutting issues, including problem identification and resolution. Activities reviewed included the installation of modified pressurizer safety valves which used new materials for some internal parts, such as the seating disc. The inspector reviewed the appropriateness of the equivalency determinations of the replacement materials. The inspector also reviewed for adequacy the installation work package, the valve vendor's test procedure, and the results of the replacement valve tests as documented in Wyle Laboratories Test Reports 46124-1 and 43963-0. The inspector reviewed the adequacy of the methods used by the licensee to address minor seat leakage on one of the valves after installation.

The inspector reviewed other work activities from the recent maintenance outage to assess the appropriateness of the licensee's evaluations and corrective actions regarding: (1) repair to two of the six low pressure steam dump valves; and (2) troubleshooting an electrical ground for the number 24 battery charger. The inspector also reviewed records for a repair activity for safety injection (SI) system relief valve (SI-855) (that occurred in August 2000 during the steam generator replacement outage). The work order maintenance history and associated condition reports for the relief valve were reviewed to understand the valve's long term performance.

b. Findings

No findings of significance were identified. A cross-cutting issue associated with problem identification and resolution was identified during the reviews of the equipment problems for SI-855, the low pressure steam dump valves, and the number 24 battery charger ground. This issue is documented in Section 4OA2.

.4 Plant Startup

a. Inspection Scope (71111.20)

The inspector monitored activities to prepare for and complete mode changes from cold shutdown to full power operations. Operator actions were observed to ensure that the operators remained cognizant of system and component conditions, work activities in the field, and expected off normal plant conditions associated with maintenance and testing activities.

The inspector questioned operators on various control room indications and reasons for lit annunciators. Shift logs were reviewed to confirm proper entry into and exit from technical specification limiting conditions for operation for inoperable equipment and documentation of control room activities. Compliance with selected plant technical specifications (TS) and plant procedures were verified.

The inspector reviewed licensee actions to implement enhanced management oversight of shift activities, and to implement a shift mentor program. Shift turnovers and shift briefings were observed to verify the information exchanged was accurate. The inspector observed communications between operators and testing personnel to verify clear identification of the planned effect on plant equipment and to verify appropriate control room indications resulted from test activities. The following activities were reviewed:

- SOP 13.2, Resetting High Flux at Shutdown Alarm
- SOP 3.3, Establishing a Bubble in the Pressurizer (CR 200110507)
- POP 1.1, Plant Restoration from Cold Shutdown to Hot Shutdown
- POP 1.3, Startup from Zero Power Condition to Full Power Operations, Rev 57
- SOP 15.1, Reactor Thermal Power Calculation - Excore Calibration
- SOP 15.3, Quadrant Power Tilt Calculation
- Operator Just in Time Training and Operator Assessments
- Plant Checkoff List (PCO)-2, Plant Heatup Greater than 350 F

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 in the proper sequence, 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 Specifications 6.8.1.a.

The following test activities were reviewed:

- Map 15FC12, Power Distribution and Hot Channel Factor Determination at 99.3% power and 9505.2 MWD/MTU, October 16, 2001
- PT-Q29A, 21 Safety Injection Pump Test (CR200111184)
- PT-R84A, 21 Emergency Diesel Generator 8 Hour Load Test, Revision 1

For core power distribution measurements, the inspector reviewed the licensee’s actions to trend the Cycle XV 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 when core power distribution was analyzed using both the original and revised values for fuel rod pitch.

b. Issues and 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 0, “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 reviewed 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 High Pressure Safety Injection System Unavailability

a. Inspection Scope (71151)

The inspector reviewed the Performance Indicator (PI) for High Pressure Safety Injection System Unavailability. This PI remained in the green band. The inspector reviewed operator logs, maintenance records, and condition reports for the system for the 1st quarter of 2000 and the 1st and 2nd quarters in 2001. The inspector noted that unavailability numbers for trains 1 and 2 in the 2nd quarter 2001 data were reversed. A quality assurance review had previously identified this error, as documented in Condition Report 200110089.

b. Issues and Findings

No significant findings were identified.

4OA2 Cross Cutting Issues

The inspector reviewed events and problems which were indicative of examples of inadequate personnel performance and problem resolution. The items below were addressed in the licensee’s corrective action program.

.1 Recurrent Performance Errors During AFW Pump Packing

a. Inspection Scope

During operation of the 21 auxiliary feedwater (AFW) pump on October 27, 2001, the licensee noted that the inboard packing gland temperature was approximately 200 degrees F. The licensee attempted to loosen the gland to improve packing leak-off, but the adjustment caused the packing gland follower to contact the shaft sleeve. The operator declared the 21 AFW pump inoperable when the gland temperature increased to 500 F. The inspector reviewed the 21 AFW pump packing “as-found” conditions on October 28, the maintenance standards to replace pump packing, the operations administrative directives for packing adjustments, the post-maintenance testing following the July 16, 2001, re-pack of the 21 AFW pump, and past condition reports involving the packing of AFW pumps.

b. Issues and Findings

(GREEN) The “as-found” condition of the 21 AFW pump inboard packing indicated the following:

- the packing rings had square cuts with frayed ends and had various lengths;
- the first ring was longer than the circumference of the shaft sleeve; and,
- the gland follower was 1/8 inch out of parallel with the pump casing with no or little insertion into the stuffing box.

Entergy concluded that the 21 AFW pump would have performed its intended safety function in this condition since the localized close tolerance between the packing gland follower and the shaft sleeve would have increased during pump operation and would have allowed for increased leakoff and cooling of the packing gland.

The 21 AFW outboard and inboard glands were re-packed on July 16 under work order NP-01-21530. The work order contained a packing checklist that referenced maintenance standard MS-005, “Pump and Valve Packing”, and vendor technical manual No. 1076-1.2. The maintenance standard provides instructions to use a mandrel to properly cut the packing material, position of the gland follower in relationship to the stuffing box, and to ensure the first packing ring is properly installed in the stuffing box.

The inspector noted that although MS-005 was referenced in the work order packing checklist, critical points from the standard were not specifically identified in the checklist. The inspector concluded that maintenance personnel had failed to achieve the following conditions per the maintenance standard when the 21 AFW pump inboard gland was packed in July 2001:

- the packing rings did not have diagonal cuts and were not the same length;
- the first packing ring did not have the butt ends firmly together, instead the ring was greater than the circumference of the shaft sleeve; and,
- the gland follower did not enter the stuffing box fully or uniformly; thus, upon adjustment, the gland follower achieved metal to metal contact on the pump’s shaft sleeve.

Further, the inspector noted that Work Order 01-24304 used to replace the inboard packing on October 28, 2001, initially neither referenced the maintenance standard nor included a packing checklist. The Maintenance Manager stopped the job upon discovery that the work order lacked this instruction. The maintenance standard was added to the work package as a reference and the mechanics successfully re-packed the inboard stuffing box on October 29, 2001. The packing gland temperature was between 102 and 113 degrees F after the gland was properly packed.

The inspector reviewed the condition reports to determine the history of past problems with packing the auxiliary feedwater pumps. Condition report (CR) 199810484 identified that the number of packing rings required for the inboard and outboard stuffing box was inconsistent. Specifically, a vendor drawing indicated that 8 rings of packing were required on the inboard gland and 5 rings and a spacer were required on the outboard gland. The licensee contacted the vendor and the vendor recommended 6 rings of packing in each stuffing box with no spacer. The engineering staff updated the vendor manual to indicate that six rings of packing were required in both inboard and outboard stuffing boxes. The licensee found on October 28, 2001, that the 21 AFW pump inboard stuffing box had seven rings installed.

A proposed corrective action for CR 199810484 was for the licensee to incorporate the vendor information regarding packing ring installation in a maintenance procedure. The CR item was closed without incorporating the vendor information in a procedure because the licensee considered the activity to repack a pump as “skill of the craft” for which a packing checklist was appropriate to perform the maintenance. The inspector concluded the licensee corrective actions for CR 199810484 were inadequate, since neither the craft skills nor the maintenance checklists used to repack the 21 AFW pump on July 16, 2001 were sufficient to properly pack the pump and assure the pump would be available when required to operate on October 27, 2001.

10 CFR 50 Appendix B, Criterion V, requires, in part, that activities affecting quality shall be prescribed in documented instructions of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. Contrary to this requirement, the maintenance instructions and checklist in work order NP-01-21530 were not adequate and the 21 AFW pump was not packed on July 16, 2001, in accordance with the reference maintenance standard and vendor instruction. This issue had a credible impact on safety since a properly packed gland is necessary to ensure reliable AFW pump operation. This issue affects the mitigating systems cornerstone since the AFW pumps supply feedwater to the steam generators to cool the reactor. However, since the maintenance errors did not result in packing failure and subsequent pump operation was deemed acceptable given the likelihood of improved gland performance (i.e. opening of clearances between the shaft and packing gland follower), this issue has been determined to have very low safety significance **(Green)** in accordance with NRC SDP Phase 1 assessment. This violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368). **(NCV 50-247/01/010-02)** The licensee initiated condition report 200110167 to address this problem in their corrective action program.

.2 Recurrent Licensed Operator Performance Issues

a. Inspection Scope

The inspection scope was to review the adequacy of licensee actions to address human performance issues.

b. Issues and Findings

(No Color) The results of the 2001 Licensee Operator Requalification (LOR) Program, summarized in Section 1R11 of this report, showed a high number of crew failures during the dynamic simulator exams. Entergy initiated Condition Report 200110167 for this issue, assigned it a significance level 1 (SL1), and initiated a root cause investigation to identify causes and corrective actions. The preliminary results of the SL1 investigation found the 2001 failures were caused by inadequate corrective actions and insufficient implementation of corrective actions for licensed operator knowledge and performance weaknesses identified during previous year LOR exams. The licensee's review of the LOR program results for the previous three years noted conditions were present where insufficient standards or inconsistent implementation of standards resulted in weaknesses in certain aspects of operator performance. The licensee determined the presently observed operator performance deficiencies were previously identified but not adequately corrected, aspects of which contributed to degraded performance in two plant reactivity management events and component misalignment events in 2001. The inspector noted the root cause of the LOR program results, inadequate corrective actions, was also evident in recent plant events and NRC findings (reference Section 4AO2. 1 and 3, and Inspection 50-247/01-09). This was an example of a cross cutting issue regarding human performance and problem resolution.

.3 Problem Identification and Resolution

a. Inspection Scope (IP 71152 and 95002)

The inspectors reviewed several problems which were indicative of an ongoing adverse trend in problem identification and resolution. The first example involved the recurring failures of SI system relief valve SI-855. Discussions were held with engineering and maintenance personnel concerning the past failures of this valve and plans for long term resolution. The review also included the licensee's evaluation of NRC Information Notice 98-23, "Crosby Relief Valve Setpoint Drift Problems Caused by Corrosion of the Guide Ring" for applicability to IP-2. Two other examples involved recurring failures of the low pressure steam dump valves and electrical drawing problems found while troubleshooting a ground associated with the number 24 battery charger. The inspectors discussed with engineering personnel why condition reports (CRs) were not initially issued to capture these problems for appropriate corrective action.

The inspectors also reviewed the adequacy of reviews performed by the licensee to assess the effectiveness of corrective actions to prevent recurrence of problems associated with the August 31, 1999, reactor trip with complications.

b. Findings

(No Color). The licensee failed to implement effective corrective actions for several issues which are noted below. The recurrence of past equipment failures was not prevented, problems were not promptly identified in the corrective action program, and the failures were not properly evaluated by completing appropriate analyses to fully understand the causes. This issue was considered an adverse trend in the corrective action program as documented in NRC inspection reports 05000247/01-09, 01-08, 01-04, 01-03, and 01-02. This issue was determined to be a Non-Cited, Severity Level IV violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action."

Valve SI-855

This valve is a 3/4X1-inch Crosby relief valve that protects the SI pump discharge header from overpressure. NRC Information Notice (IN) 98-23 informed licensees that corrosion of nozzle guide rings for this valve type had caused leakage and setpoint drift problems. The corrective action discussed in the NRC IN was to change the nozzle guide ring material from Type 416 to Type 300 stainless steel. The licensee's evaluation of this NRC IN for applicability to IP-2, which was documented in CR 199805607, indicated that the inservice testing program provided for adequate inspections of SI-855. While corrosion problems had not been experienced, engineering recommended in 1999, based on industry experience, that the nozzle guide rings be changed to Type 300 stainless material at the earliest opportunity. Apparently, this modification was not done.

A review by the inspectors of the maintenance history for SI-855 indicated recurring failures, where the valve lifted prematurely during SI pump quarterly testing. Such premature lifting is undesirable since some SI pump flow intended for core cooling would be diverted from the injection header to the pressure relief tank during accident conditions. The past failures occurred in 1989 (Work Order NP-88-41139), in 1991 (Work Order NP-90-51538), in 1995 (Work Order NP-95-77770), and in 2000 (CRs 200004479 and 200004772). The valve was last repaired in August 2000 when the vendor fabricated the necessary internal parts while onsite. The licensee missed an opportunity at this time to install the Type 300 stainless steel material for the nozzle guide ring. The inspectors also determined that a failure analysis of the valve, which was specified in CR 200004772, was not completed. Apparently, engineering intended to procure a new design relief valve, even though this failure analysis was not completed. The licensee issued CR 200110971 to track the valve modification for improving its performance and CR 200111257 to complete the failure analysis of the valve problems experienced in 2000. The inspectors confirmed with operations personnel that the valve has performed satisfactorily since the last repair.

Low Pressure Steam Dump Valves

The inspectors reviewed work associated with two of the six low pressure steam dump valves which provide overspeed protection for the main turbine following a turbine trip. One of the six valves (FCV-1207) was binding and another valve (FCV-1206) was observed to be sluggish when operated during quarterly stroke testing. During a review of the work package for FCV-1207, the inspectors noted that the workers had identified that there was scoring on the valve plug. The package also indicated that the body-to-bonnet nuts were found to be loose on one side of the valve, possibly causing the scoring on the plug. This condition had not been documented in a condition report nor was there a condition report that tracked the need to complete a failure analysis for the repetitive failures of these valves. The system engineer had identified several similar failures during a review of the corrective maintenance history for the valves. The licensee acknowledged this issue and initiated CR 200111022.

Battery Ground Troubleshooting and Repair

The inspectors reviewed work order NP-01-21915 related to the troubleshooting of an electrical ground associated with the number 24 battery charger. In reviewing the documentation the inspectors noted that there were two entries in the work log which indicated that electrical drawings did not match the plant configuration. However, there was no indication that the discrepancies had been documented in a condition report. The licensee reviewed this issue, confirmed that a condition report had not been issued, and then initiated CR 200111026 to document the discrepancies.

Corrective Action Effectiveness Reviews

Prior to the implementation of the NRC's revised Reactor Oversight Process, the NRC had characterized performance issues associated with the August, 1999, reactor trip with complications as a Yellow finding associated with the mitigating systems cornerstone area (reference the ROP "Feasibility Review," Attachment 7 to SECY 00-0049). The inspectors reviewed the licensee's effectiveness review completed on January 25, 2001, for the corrective actions associated with this event. The effectiveness review was thorough and concluded that the specific equipment problems associated with the event had been effectively addressed. However, the report also noted that additional improvement was necessary to enable the station to demonstrate an ability to more consistently identify, evaluate, understand and resolve risk-significant equipment problems and to improve management technical support to operations during off-normal conditions. The effectiveness review report provided several recommendations which were documented in condition reports and the report also discussed long term actions (Business Plan Items) intended to improve performance in various areas. An additional CR was initiated to track the need to perform an additional effectiveness review once the CRs associated with the recommendations were closed.

In October 2001, another review was completed to assess the effectiveness of corrective actions for the steam generator tube leak event which occurred in February 2000. The scope of this review also included the January 2001 turbine trip event, the August 2001 overpower event and associated aspects of the August 31, 1999 reactor trip. This assessment was also thorough and again cited improvements in various areas but recognized there were areas requiring additional improvement. Fifteen CRs were initiated to document the report recommendations and the CRs were being evaluated at the time of this inspection to determine what additional corrective actions may be required.

The corrective actions for specific causes for problems associated with the August 31, 1999, reactor trip with complications have been adequately addressed and areas requiring additional improvement have been identified and entered into the corrective action program. However, as evidenced by findings associated with the licensee's internal reviews and self-assessments, NRC findings, and findings of other outside organizations, there are several areas which require additional improvement. These areas include corrective action effectiveness, human performance and equipment reliability. Several recent issues exemplify the need for improvement, including:

- The safety injection safety system functional assessment (SSFA) identified problems with training department lessons plans that continued to exist, in part, due to ineffective corrective actions for a previously identified training plan issue.
- As discussed during this inspection, performance problems with the SI-855 relief valve have existed for more than ten years and no root cause analysis had been completed to ensure the recent repairs will effectively resolve the problem. While this issue has not resulted in the inoperability of the safety injection system, premature lifting of SI-855 could result in unnecessary distractions and/or challenges to the operators during an event (e.g. unknown increase in pressure relief tank (PRT) level).
- Problems with the auxiliary feedwater pump packing overheating during the last plant shutdown was an additional example of ongoing equipment performance issues.
- Recent poor performance during operator requalification was another significant example of problems in the human performance area.

The licensee's actions to improve corrective action effectiveness, human performance and equipment performance will continue to be reviewed in future inspections consistent with the intent for heightened oversight at IP-2 as noted in the NRC Supplemental Inspection 95003 Multiple Degraded Cornerstone Report 05000247/2001-02.

Conclusion

The failure to promptly identify conditions adverse to quality and to take effective corrective action as indicated by the recurring failures of valve SI-855 and the low pressure steam dump valves is considered a Severity Level IV violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. This issue was determined to be an adverse performance trend in the cross cutting area of problem identification and resolution. Additional corrective actions have been assigned in condition reports to address these performance issues. 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 FR 25368). **(NCV 50-247/01-10-03)**

.4 Safety System Functional Assessment of the Safety Injection System

a. Inspection Scope (IP 71111.21)

The inspectors reviewed the plan and results for the safety system functional assessment (SSFA) of the SI system recently completed by an independent team contracted by the licensee. Several CRs, which were issued as a result of the team's findings, were reviewed in detail to assess the quality of the SSFA and the licensee's corrective actions. Discussions were held with the licensee to gain their perspective concerning the efforts of the SI SSFA.

b. Findings

The inspectors noted that the SSFA team's inspection plan was detailed and resulted in a thorough effort. The team members were well qualified, having performed similar efforts at IP-2 and other nuclear facilities. More than 40 condition reports were issued during the assessment. For example, CRs 20109352 and 200109358 were issued concerning apparent errors or inconsistencies between SI system design information and operator training lesson plan material. While the licensee properly resolved these problems by correcting the lesson plan material and closing both CRs, the training department recognized that additional work was needed to improve all lesson plan material. The SSFA team considered that the licensee's design basis documentation had improved when compared to the status of the auxiliary feedwater system documentation during a SSFA of that system 16 months prior. The inspectors noted that the SI-855 valve issue, a mechanical area item identified in this report, was not discussed during the SI SSFA.

4OA3 Inspection Item Followup

.1 (Closed) **URI 05000247/2001-005-02a** pertaining to licensee's process to delete RPS wiring separation criteria from the UFSAR. During the March 2001 Reactor Protection System (RPS) wiring separation inspection (Inspection 2001-005), Section 7.2.2.9 of the Updated Final Safety Analysis Report (UFSAR) stated: "Isolation of the reactor protection and engineered safety feature signals in the reactor protection logic racks is achieved by physical separation...The separation is maintained by using separate wireways for safety signals, annunciator signals and computer signals." Before the March 2001 inspection, the licensee had identified many examples where non-safety annunciator signal wiring and safety reactor trip signal wiring were routed in the same wireway, in conflict with the UFSAR statement. These discrepancies were documented in several condition reports (CR).

In March 2001, the licensee generated Safety Evaluation (SE) 99-160-EV to delete the wiring separation criteria in the UFSAR, stating that UFSAR Section 7.2.2.9 was contrary to existing field condition, and a change was required to match the field condition. The SE did not provide an explanation why the UFSAR was incorrect. In addition, during the March 2001 inspection, the inspectors observed that there were annunciator wires that the licensee added (or functionally added to originally spare wires) in a 1982 modification to the P-10 relay contacts, that were routed from the RPS Train A cabinet directly to the Train B cabinet. Deleting the separation criteria in the UFSAR would allow a non-safety wire to be physically in contact (insulated wires bundled together) with train A safety related wires at one end and in contact with Train B safety related wires at the other end. The SE did not address this issue nor discuss the potential of a single credible failure that could affect both train A and train B of the RPS. However, during the March 2001 inspection, the licensee provided information which demonstrated that the wiring configuration described above would not cause an immediate concern that could affect the operability of the RPS: 1) the affected non-safety related wires were protected by Class 1E fuses and breakers, and 2) there were no higher voltage (insulation damaging voltage) cables being routed together with those wires in the cable tray (located in the cable spreading room) outside the RPS cabinets.

This issue was evaluated by the Office of Nuclear Reactor Regulation (NRR). The evaluation concluded that the wiring separation between safety and non-safety wires inside the RPS cabinets was not a design requirement for Indian Point Unit 2, which was based on the design criteria specified in IEEE Standard 279, “Proposed IEEE Criteria for Nuclear Power Plant Protection Systems,” dated August 28, 1968. The result for this issue as described above was that the wiring configuration met the design requirement. Therefore, this item is closed.

.2 (Closed) **URI 05000247/2001-005-02b**, whether nonsafety-related wires that were routed directly from the RPS Train A cabinet to Train B cabinet, without separation from either train A or train B safety-related wires inside the RPS cabinets, met the separation criteria as revised in 1988 for IP2. This issue was evaluated by NRR. The evaluation concluded that this wiring configuration met the design criteria specified in IEEE Standard 279, “Proposed IEEE Criteria for Nuclear Power Plant Protection Systems,” dated August 28, 1968. Therefore, this item is closed.

4OA4 Licensee Event Report Reviews

.1 (Closed) LER 05000247/2001-02-00: Operation in Excess of Maximum Rated Thermal Power. The inspector reviewed the information the licensee provided to describe and analyze this event. The corrective actions for this event were reviewed in NRC Inspection 50-247/01-09. The LER accurately described the event. This LER is closed.

4OA6 Meetings

Exit Meeting Summary

On November 28, 2001, the inspector presented the inspection results to Mr. F. Dacimo and other members of the licensee staff who acknowledged the findings. The inspector asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

a. **Key Points of Contact**

P. Asendorf	Security Manager
W. Axelson	Corrective Action Group
J. Barlok	Plant Engineering
F. Dacimo	Vice-President
R. Depatie	System Engineer
N. Ertle	I&C Engineer
G. Hinrichs	Engineering Assurance
T. Jones	Nuclear Safety and Licensing
T. Klein	Design Engineering
E. Libby	Licensed Operator Instructor
T. McCafferty	System Engineering Manager
J. McCann	Nuclear Safety and Licensing Manager
M. Miller	Manager, Generation Support
W. Osmin	Reactor Engineer
J. Parry	Strategic Planning
B. Ray	System Engineer
G. Schwartz	Chief Engineer
P. Rubin	Operations Manager
A. Sheikh	Design Engineering
R. Sutton	SL-1 Team Leader
L. Temple	Plant Manager
M. Vaseley	System Engineer Supervisor
J. Ventosa	Engineering Manager
T. Wadell	Maintenance Manager
W. Wittich	Supervisor, Mechanical Design
E. Woody	I&C Manager

b. **List of Items Opened, Closed, and Discussed**

Opened and Closed During this Inspection

50-247/01-10-01	NCV	Failure to Control M&TE per Appendix B, Criterion XII
50-247/01-10-02	NCV	Failure to Control Maintenance per Appendix B, Criterion V
50-247/01-10-03	NCV	Failure to Issue Condition Report and Implement Corrective Action as Required by 10 CFR 50, Appendix B, Criterion XVI

Closed

50-247/2001-02-00	LER	Operation in Excess of Maximum Rated Thermal Power.
50-247/00-05-02a	URI	Wiring separation criterion between safety and non-safety wires within the RPS cabinets.

50-247/00-05-02b	URI	Non-safety related wires that were separated from safety related wires and routed directly from RPS Train A cabinets to Train B cabinets.

c. List of Document Reviewed

Condition Reports

199805607, 199904709, 199906643, 200003758, 200004772, 200010860, 200105980, 200105965, 200107324, 200108455, 200109894, 200109938, 200110688, 200111022, 200111026, 200111257

Work Orders

NP-01-23601 Repair low pressure steam dump FCV-1206.
NP-01-22445 Repair low pressure steam dump FCV-1207.
NP-01-21915 Troubleshoot DC ground associated with 24 battery charger.
NP-01-20079 Repair thermo-well (TW-5869) leak.
NP-01-20340 Replace sticky RPS pushbutton.
NP-01-20341 Replace sticky RPS pushbutton.

Self-Assessments

The Effectiveness of Corrective Actions From The Unit Trip of August 31, 1999 dated October 12, 2000
SL-1 Corrective Action Effectiveness Review Report, February 15, 2000 Steam Generator Tube Leak (SGTL) Event, Rev. 0
Effectiveness Review - Trip and Unusual Event 8/31/99 dated January 25, 2001

Procedures

CAG 20.304, Rev. 1, Corrective Action Screening Effectiveness Reviews

d. **List of Acronyms**

AOI	Abnormal Operating Instruction
BTP	branch technical position
CAG	Corrective Action Group
CFR	Code of Federal Regulations
CR	Condition Report
FCV	Flow Control Valve
IEEE	Institute of Electrical and Electronic Engineers
IP	Inspection Procedure
LLC	Limited Liability Corporation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PARS	publicly available records

PI	performance indicators
PRT	Pressure Relief Tank
RCS	reactor coolant system
ROP	Reactor Oversight Program
RPS	Reactor Protection System
SAO	station administrative order
SDP	significance determination process
SE	Safety Evaluation
SI	safety injection
SSFA	Safety System Functional Assessment
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
V	volt