November 13, 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 50-247/01-09

Dear Mr. Dacimo:

On September 29, 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 October 19, 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 the program and controls for Radioactive Material Processing and Transportation. 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.

Based on the results of this inspection, three violations of NRC requirements were identified. One violation concerned several examples of the operators' failure to follow a calorimetric procedure on August 17, 2001, which resulted in a non-conservative adjustment to power range nuclear instruments and increased likelihood of a reactor trip. This was assessed as having very low safety significance. The second violation involved the operators' failure to adequately monitor reactivity parameters and plant conditions that resulted in a minor overpower condition on August 17 and a violation of License Condition 2.C(1). Similar to NRC findings described in Inspection 50-247/00-15, the August 17 overpower reactivity management event was not adequately responded to by the plant staff. Therefore, a third violation of NRC requirements concerned ineffective corrective actions for past events which did not prevent recurrent problems in the area of procedure use, log keeping, and post-evolution debriefs, and contributed to the August 17 overpower and untimely management response. However, because of the very low safety significance (the overpower condition was approximately three percent and lasted for five minutes) and because these licensee-identified violations 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 50-247/01-09

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
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
E. Smeloff, Pace University School of Law
L. Puglisi, Supervisor, Town of Cortlandt

Distribution w/encl: H. Miller, RA/J. Wiggins, DRA (1)
D. Loveless, RI EDO Coordinator
W. Raymond, SRI - Indian Point 2
E. Adensam, NRR (ridsnrrdlpmlpdi)
P. Eselgroth, DRP
P. Milano, PM, NRR
G. Vissing, PM, NRR (Backup)
S. Barber, DRP
L. Harrison, DRP
R. Junod, DRP
R. Martin, DRP
Region I Docket Room (w/concurrences)

DOCUMENT NAME: G:\BRANCH2\IP2\IP20109.WPD
After declaring this document "An Official Agency Record" it **will/will not** be released to the Public. **To receive a copy of this document, indicate in the box:** "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP	RI/DRP		RI/DRP	
NAME	WRaymond/SB for WJR	PEselgroth/PWE		BHolian/BeH	
DATE	11/7/01	11/9/01		11/8/01 11/13/01	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	50-247/01-09
Licensee:	Entergy Nuclear Operations, Inc..
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	August 19 - September 29, 2001
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector John R. McFadden, PhD, Health Physicist Eva Brown, Resident Inspector Ross Telson, Resident Inspector Leonard Prividy, Senior Reactor Engineer
Approved by:	Peter W. Eselgroth, Chief Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000247-01-09, on 08/19-09/29/2001, Entergy Nuclear Operations, Inc., Indian Point 2 Nuclear Power Plant. 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 green and no color issues. The “no color” significance level indicates that the IMC 0609 “Significance Determination Process” does not apply to these findings.

Cornerstone: Initiating Events

Green The operators’ failure to adhere to plant procedures and to adequately monitor plant conditions resulted in an overpower condition on August 17, 2001, and a violation of the License Condition 2.C.(1) thermal power limit. The failure to follow calorimetric and operating procedures was a violation of Technical Specification 6.8.1. The overpower condition impacted the reactor safety cornerstone since it could have caused a reactor trip if not corrected by the operators. This event had very low safety significance, since the overpower condition was minor, existed for a small amount of time, and resulted in no loss of function or availability of mitigation equipment. The violations of License Condition 2.C.(1) and Technical Specification 6.8.1.a are treated as Non-Cited Violations, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368)

Cross-Cutting Issues

No Color The licensee corrective actions in response to past reactivity management and plant events were ineffective in precluding recurrent problems in log keeping, procedural adherence, and post-evolution debriefs. These deficiencies contributed to the August 17, 2001 overpower condition and the subsequent, untimely management review. This is a recurrent example of an issue in problem identification and resolution. The failure to correct conditions adverse to quality is considered a 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, issued on May 1, 2000 (65 FR 25368).

No Color The inspector identified an error in the reactor coolant system (RCS) activity performance indicator (PI) data reported for the second quarter of 2001. Transcription errors and ineffective review contributed to the errant PI data. The errors had minimal significance since the PI remained within the green band. However, previous inspection findings identified errors in reporting Indian Point 2 PI data (reference NRC Inspections 05000247/00-01 and 00-11). This issue has more than minor significance because the failure to accurately report PI data potentially could impact the ability of the NRC to perform its regulatory function. The licensee entered this issue in the corrective action program as Condition Report 200109517.

TABLE OF CONTENTS

Report Details

SUMMARY OF PLANT STATUS

The plant operated at full power during the period. On September 6, 2001, the NRC issued Amendment Numbers 50 and 220 and approved the transfer of the licenses for Indian Point Nuclear Generating Units 1 and 2 from Consolidated Edison Company of New York, Inc. to Entergy Nuclear Indian Point 2, Limited Liability Corporation (LLC) as the owner of Indian Points 1 and 2, and to Entergy Nuclear Operations as the entity to maintain Indian Point 1 and to operate Indian Point 2.

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

1R01 Adverse Weather Protection

a. Inspection Scope (71111.01)

The purpose of this inspection was to review licensee actions per Technical Specification 3.14 from September 10-12, 2001, when Hurricane Erin came within 500 miles of the site. The inspector reviewed licensee actions to monitor plant equipment, assess plant risk, and implement compensatory measures in accordance with Abnormal Operating Instruction (AOI) 28.0.7, Hurricane/Tornado/High Wind/Severe Thunderstorm, Revision 11. The hurricane did not pose a significant threat to the site.

b. Issues and Findings

No significant findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope (71111.04)

On September 11, 2001, the inspector performed a partial walkdown of the 21 and 23 emergency diesel generators. The review was conducted to verify support systems and component alignments were proper. The inspector evaluated the impact on system function from outstanding equipment deficiencies and area housekeeping issues. The licensee was performing scheduled maintenance on the 22 emergency diesel generator at the time. The inspector reviewed licensee actions to expedite recovery of the 22 emergency diesel generator as part of the measures to assure these station emergency power supplies were in an operationally ready status.

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:

- Fire Zone 361, 13.8 kilovolt (kV) Light and Power Auxiliary Bus Room
- Fire Zone 14, 480 volt (V) Switchgear Room

Reference material consulted by the inspector included Con Edison’s Fire Protection Implementation Plan, Pre-Fire Plan, and Station Administrative Order (SAO)-700, “Fire Protection and Prevention Policy,” SAO-701, “Control of Combustibles and Transient Fire Load,” SAO-703, “Fire Protection Impairment Criteria and Surveillance,” and Calculation PGI-00433, “Combustible Loading Calculation.” The regulatory basis for the inspection included license condition 2.K, Technical Specification (TS) 6.8.1.e, and Branch Technical Position (BTP) 9.5-1, Appendix A.

b. Issues and Findings

No significant findings were identified.

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 200109310, 200109464 and 200109475.

.2 Alternate Safe Shutdown Cables

a. Inspection Scope (71111.05Q)

The inspector performed a partial walkdown of the 440 volt ac and the 13.8 kilovolt power cables that support alternate safe shutdown operations in accordance with 10 CFR 50 Appendix R. The primary focus of the walkdown was to determine whether cables routed through manholes or underground conduits were submerged, and if the observed conditions were within the cable design parameters. The inspector reviewed the following reference material:

- Abnormal Operating Instruction (AOI) 27.1.9, “Control Room Inaccessibility Safe Shutdown Control,”

- Plant drawings A250907-19, 138327-9, 138159-13, 138146-8, A141119-11, 138410-11, A140966-25, 138370-1, 138476-1,and 140925-12
- Plant Modification EGP-88-01469-E , “Replacement of 13.8 KV Feeder to IP-1"
- EDS-261 and 262 Quarterly Preventative Maintenance for Manholes 2 and 3
- Cable Specification EL-17 and 7402XJ-2

b. Issues and Findings

No significant findings were identified.

1R11 Licensed Operator Requalification

.1 Observation of Simulator Training

a. Inspection Scope (71111.11)

The inspector reviewed licensed operator simulator testing conducted on September 18, 2001, per Lesson No. ESR-500-0101A to assess the adequacy of the training, licensed operator performance, emergency plan implementation, and the adequacy of the licensee’s critique. The training considered lessons learned from operating experiences and included simulator drills for responding to a steam generator tube rupture and a loss of reactor coolant using procedures E-0, E-3, ECA-3.1, AOI 1.2 and SAO 124.

b. Issues and Findings

No significant findings were identified.

The inspector verified that licensee-identified crew or individual weaknesses observed during training were remediated prior to the resumption of licensed duties. Licensee and NRC evaluation of the Licensed Operator Requalification program test results continued at the conclusion of the inspection period.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspector reviewed risk significant test activities on the power operated relief valves and block valves that resulted in the automatic control function of the backup over pressure protection for the reactor coolant system being unavailable. The inspector reviewed operator logs, the reactor coolant system Maintenance Rule background document, emergency Technical Specification Amendment 185 (12/8/1995), Technical Specification Amendment 72 (8/24/1981), Condition Report 199900482, and the following surveillances for the overpressure protection system completed in 2001: PC-R40, “OPS Pressure Indication Calibration inside the CCR,” and PT-V14, “Overpressure Protection System Analog Channel Test.”

b. Issues and Findings

No significant findings were identified.

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

a. Inspection Scope (71111.14)

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.

b. Issues and Findings

No significant findings were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspector 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, emergency operating procedures, system operating procedures, EPRI Nuclear Power Plant Equipment Qualification Review Manual, Updated Final Safety Analysis Report, and associated Design Basis Documents as references. The specific issues reviewed included:

- CR 199908669 and 199704709, NRC Information Notice 88-23 Supplement 5, “Potential for Gas Binding of High Pressure Safety Injection Pumps during a Loss-of-Coolant Accident
- CR 200108053 and 200108179, Motor driven auxiliary feedwater pump oil bearing qualification
- CR 200108752, Operability of High Head Minimum Flow Valves 842 and 843
- CR200108499, Fire System Leaks through Victaulic Couplings
- CR200108518 and 8738, 24 Reactor Coolant Pump Seal Return Flow

b. Issues and Findings

No significant findings were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope (71111.17A)

The inspector reviewed two plant modifications to assess whether 1) the modifications were safely implemented, 2) design inputs were correctly defined, 3) the associated calculations and other supporting documents were correct and clear, 4) adequate post modification testing was performed, and 5) plant procedures and records were being updated to reflect the modified design.

Modification No. MEX-93 09212-M, Revisions, 4, 6, 7 and 8, “Weld Channel System Upgrade”

Changes to the Weld Channel and Penetration Pressurization System (WC&PPS) were implemented to resolve several system air leaks. These changes were as follows:

- In June 1997, leaks were found in weld channel sections W-10 and B-6, which are located beneath the concrete of the containment floor. The leaks were confirmed to be from some tubing associated with the W-10 and B-6 sections. The affected tubing was cut and capped. Similar work was done for weld channel sections B-2 and B-5 in January 1998, and section W-11 in March 2000.
- In January 1998 weld channel section D-2, which is located in the containment dome, also had tubing leaks. The tubing supply to this zone was cut and capped.

The inspector reviewed Condition Reports 199700997 and 200001335 and the accompanying operability evaluations concerning the initial problems found with weld channel sections B-2, B-5 and W-11. Specifically, the inspector reviewed the licensee’s evaluation of these problems regarding the impact on containment integrity. This review included the impact of the retired weld channel sections on the next performance of Procedure PT-3Y1, “Integrated Leak Rate Test (ILRT)”. The licensee issued Condition Report 200108419 regarding the inspector’s review of the WC&PPS modifications. The licensee’s response to this condition report noted that Communications to Staff memo No. 99-0229 was issued to revise Procedure PT-3Y1 to ensure that the retired weld channel sections would be vented to containment to assure the entire containment boundary is tested. The last ILRT was performed in 1991 and prior to any of the WC&PPS sections being retired.

Modification No. FPX-94-10254. Revisions 2 and 4," Nitrogen Supply to Containment"

This modification was implemented to improve the pressure control operation of the nitrogen supply line to the safety injection (SI) accumulator tanks inside containment. New pressure control valves (PCV-942 and PCV-7726) with upstream Y-pattern strainers (ST-275 and ST-276) were installed. Revisions 2 and 4 of the modification were issued to install a flow limiting orifice upstream of PCV-942 and PCV-7726. The orifice was sized to limit the flow through each PCV to 1500 standard cubic feet per minute (SCFM). Without this revision to the initial design, the potential existed for exceeding the maximum SI accumulator relief valve capacity of 1536 SCFM since the new PCV makeup capacity was 2025 SCFM. This problem was identified in Condition Report 199900162, which prompted the orifice design change.

b. Issues and Findings

No significant findings were identified.

The inspector discussed the nitrogen modification with the licensee and reviewed Calculation Number FPX-00307-00, "Orifice Plate Sizing for Nitrogen Supply to Containment." Independent inspector observations concerning unclear documentation of design inputs and assumptions in this calculation had been identified by the licensee previously in Condition Report 200107208. Based upon the inspector's review, the licensee issued Condition Report 200108393 to correct a modification implementation problem. The problem was a failure to establish a preventative maintenance activity for periodically inspecting and cleaning strainers ST-275 and ST-276.

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:

- PT-Q35A, 21 Containment Spray Pump Coupling Inspection and Realignment
- PT-M48, 480 Volt Bus Undervoltage Test for WO 01-23928 (CR 200109163)
- PT-Q33C, 23 Charging Pump Flow Controller Repair
- PMT 23284 Test 1 of 1, "Repair 24 Feedwater Regulating Valve" (CR 200108263)

b. Issues and Findings

No significant findings were identified.

1R20 Refueling and Other Outage Activities

.1 Spent Fuel Pool Activities

a. Inspection Scope (71111.20)

The inspector reviewed the activities related to fuel movements in the Unit 2 spent fuel pool to verify they were in conformance with the applicable procedure and technical specifications. The inspector reviewed the licensee's plan for relocating spent fuel in the pool in light of the limitations due to boraflex degradation (reference NRC Inspection 50-247/00-05), and verified the fuel configuration was consistent with the design basis. The storage pattern fully utilized the newest racks and increased the use of a checkerboard pattern in the older racks, but eliminated the provisions for a full core offload capability. Entergy submitted a proposed change to the operating license (reference Entergy Letter to the NRC dated September 20, 2001) that would allow credit for soluble boron in the pool. Fuel handling operations were reviewed for conformance with the following requirements:

- SOP 17.12, Spent Fuel Assembly Handling Tool, Revision 6
- SOP 17.25, Handling Instructions for Fuel Assemblies and Inserts, Revision 7
- Fuel Handling Data Sheets for Fuel Bundle Q18, P52, P02 (Steps 10, 11, 12)
- Technical Specification 3.8, "Refueling"
- FP-IPP-R15A, Refueling Procedure Cycle VIV-XV, TPC 01-0153
- Work Order NP-01-22447, Work Step List #1, Fuel Handling Controls
- Work Order NP-01-23085, Spent Fuel Crane Testing per PT-R8B
- Radiation Work Permit (RWP) 010135
- NETCO Letter "Report for Criticality Analysis of Interim Loading Pattern", dated September 11, 2001

The inspector reviewed the corrective actions for equipment issues and minor mechanical damage (bent grid straps and cell guides) that were identified while moving fuel, as described in Condition Reports 200108733, 200108778, 290108813. The inspector reviewed the licensee's corrective action for a failure to provide intermittent health physics coverage as required by RWP 010135, as described in Condition Report 200108736. The inspector verified the issues were appropriately documented in the corrective action system for evaluation.

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 a surveillance test procedure and observed the testing activity 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 selection of surveillance tests to be observed were based upon a review of risk significant components as identified in the Indian Point 2 Individual Plant Examination. The regulatory requirements that provided the acceptance criteria for this review were 10 CFR 50 Appendix B criterion V, “Instructions, Procedures, and Drawings,” Criterion XIV, “Inspection, Test, and Operating Status,” Criterion XI, “Test Control,” and Technical Specifications 6.8.1.a. The following test activities were reviewed:

- PT-Q27B, 23 Auxiliary Feed Pump Test (CR 200108603)
- Map 15FC10, Power Distribution and Hot Channel Factor Determination at 99.9% power and 7534.1 MWD/MTU, August 21, 2001 (CR 200106456)
- Map 15FC11, Power Distribution and Hot Channel Factor Determination at 99.9% power and 8519.31 MWD/MTU, September 18, 2001 (CR 200106456)
- PT-3Y15, Fire Loop Flow (CR 200108973, 200108979, 200108981)

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 ($F_{\Delta}H$), which reached its maximum value at about 5500 MWD/MTU and began to decrease. 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.

The inspector reviewed licensee actions in response to a number of condition reports issued during the performance of surveillance PT-3Y15, which measured the flow from the diesel driven fire pump. The test failed because the measured flow of 2,028 gallons per minute (gpm) was below the acceptance criteria of 2,600 gpm (CR 200108973). The flow was less than required because relief valve RV-192 was inadvertently opened and bypassed fire water from the test nozzles. Entergy appropriately declared the high pressure fire header system out of service until RV-192 was isolated, and completed an operability evaluation for the diesel driven fire pump.

b. Issues and Findings

No significant findings were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety (PS)

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope

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

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

The inspection included a selective review of conformance with applicable waste characterization and classification program procedures:

- the radio-chemical sample analysis results for radioactive waste streams,
- the development of scaling factors for difficult to detect and measure radionuclides,
- the methods and practices to detect changes in waste streams as described in the PCP,
- the methods and practices to determine waste classification (10 CFR 61.55) and to determine DOT shipment subtype (49 CFR 473),
- Procedure RW-Q-4.006, Revision 8, 10 CFR 61 Sampling Program,
- Procedure RW-SQ-4.007, Revision 8, Process Control Program,
- Procedure RW-SQ-4.011, Revision 3, RADMAN Program Operation,
- Procedure RW-SQ-4.103, Revision 5, 10 CFR 61 Radwaste Classification,
- Procedure RW-SQ-4.104, Revision 9, DOT Classification of Radioactive Materials.

The inspection included a review of radioactive waste program documents, shipment preparation procedures, and activities for regulatory compliance, including the following:

- current radioactive waste inventory records,
- radioactive material shipping log for 2001,
- radioactive waste shipping log for 2001,
- verification that training was provided in accordance with NRC Bulletin 79-19 and 49 CFR Part 172 Subpart H,
- Procedure RW-SQ-4.000, Revision 13, Shipment Final QC Inspection,
- Procedure RW-4.001, Revision 5, Container Control and Accountability,
- Procedure RW-SQ-4.003, Revision 11, Radwaste Section Responsibilities,
- Procedure RW-SQ-4.105, Revision 10, Survey of Radioactive Shipments,
- Procedure RW-SQ-4.107, Revision 12, Radioactive Shipment Preparation,

- Procedure RW-SQ-4.109, Revision 7, Radioactive Material Storage,
- Procedure RW-SQ-4.201, Revision 9, Replacement of Filters,
- Procedure RW-SQ-4.210, Revision 5, Management of Solid Radwaste,
- Procedure RW-SQ-4.303, Revision 14, Shipping Cask Handling,
- Procedure RW-4.304, Revision 14, Dry Active Waste Processing,
- Procedure RW-4.500, Revision 6, Decontamination of Areas and Components,
- Procedure RW-SQ-4.700, Revision 12, Spent Resin Transfer Setup,
- Procedure RW-S-4.801, Revision 7, Operation of the Portable Demineralizer,
- Procedure RW-SQ-4.202, Revision 7, Operation and Dewatering of Radwaste Demineralizer System.

The inspection involved a review of the following five package shipment records for compliance with NRC and DOT requirements: Shipment Nos. 01-001W, 01-011W, 01-022W, 01-136RM, and 01-143RM.

In the area of identification and resolution of problems, the inspection included a selective review of the following audits, surveillance report, assessments, and Condition Reports (CRs) related to the radioactive material and transportation programs since the previous inspection and a determination if identified problems were entered into the corrective action program for resolution:

- Audit report no. 99-03-E, June 28, 1999, Radwaste Material Packaging and Transport and Process Control Program,
- Audit Report No. 00-03-E, September 25, 2000, Radwaste Material Packaging and Handling and Process Control Program,
- Nuclear Quality Assurance Field Observation Report No. 01-F-172, September 27, 2001, Unit 1 RCA Tour of Radwaste Liquid Processing Facilities,
- Nuclear Quality Assurance Independent Oversight Program Assessment Report No. 01-AR-17-RP, August 2001, Radwaste,
- Preliminary Exit Report for an EPRI Assessment of IP2's Solid Radioactive Waste Program, July 2001,
- Draft report for an EPRI Assessment of IP2's Low Level Liquid Radwaste Program, June 2001, and
- Condition Report Nos. 2001-00691, 2001-00854, 2001-03429, 2001-04128, and 2001-06748.

The above review was against criteria contained in:10 Code of Federal Regulations (CFR) Part 20: Subpart F (Surveys and monitoring); 10 CFR20.1902 (Posting requirements); Subpart I (Storage and control of licensed material); Subpart K (Waste disposal); Appendix G to Part 20 (Requirements for transfers of low-level radioactive waste intended for disposal at licensed land disposal facilities and manifests); 10 CFR 61.55, Waste classification; 10 CFR 61.56, Waste characteristics; 10 CFR 61.57, Labeling; 10 CFR 71, Packaging and transportation of radioactive material; 49 CFR Part 172 (Hazardous materials table, special provisions, hazardous-materials communications, emergency response information, and training requirements); 49 Part 173 (Shippers-general requirements for shipments and packaging); 49 CFR Subpart I (Class 7 (radioactive materials); 49 CFR Part 177 (Carriage by public highway); NRC Bulletin 79-19; and site procedures (cited above).

The inspector also reviewed the storage conditions and procedural controls for the radioactive waste stored onsite which the licensee was currently treating as mixed waste (i.e., waste that contains both hazardous waste and source, special nuclear, or byproduct material). The inspector performed a visual inspection of the waste containers in each of the accessible designated storage locations. (A visual inspection of the containers in the Unit 1 refueling cavity, which were stored there in early 2000, would require the removal of floor plugs. The licensee provided pictures of the containers in the Unit 1 refueling cavity which were taken in March of 2001.) The designated mixed waste storage locations included the following:

- elevation 70, Unit 1 containment, floor area inside hatch and floor area behind steam generators,
- elevation 108, Unit 1 containment, refuel floor area and refueling cavity (also referred to as the reactor internals storage pit),
- elevation 53, storage room, Unit 1 chemical systems building, and
- outside area in diesel generator alley way, portable hazardous material safety storage container.

The inspector discussed the procedural controls and the conduct of the licensee's inspections with licensee management and technical personnel. The inspector reviewed Procedure RW-Q-4.803, Revision 3, Hazardous/non-hazardous/mixed waste storage area specifications and inspections, which required inspections of all mixed waste containers and related equipment (i.e., condition of containers, physical condition of storage area, and condition of safety equipment). The inspector reviewed the inspection records for September 2001.

The inspector selectively verified that radioactive waste was: labeled, posted, and secured in accordance with NRC regulations; and that the quantities of radioactive material, the condition of the storage containers and storage locations, the proceduralized controls, and the monitoring of any potential release pathways met the intent of the guidance in NRC Generic Letter 81-38.

The above-cited review of radioactive materials was against criteria contained in 10 Code of Federal Regulations (CFR) Part 20 Subpart F (Surveys and monitoring), 20.1902 (Posting requirements), Subpart I (Storage and control of licensed material), Generic letter 81-38, Storage of low-level radioactive wastes at power reactor sites, and site procedures (cited above in this section). See Section 4OA5 for a review of the New York State Order regarding the storage of mixed waste.

b. <u>Findings</u>

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 <u>Performance Indicator Verification</u>

The inspector reviewed the licensee's performance indicator data collecting and reporting process as described in procedure SAO-114, "Preparations 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, calculations, definition of terms, and clarifying notes for the performance indicator. The licensee's corrective action program records were also reviewed to determine if any problems with the collection of PI data had occurred.

.1 Safety System Functional Failures

a. Inspection Scope (71151)

The inspector reviewed the Performance Indicator (PI) for Safety System Functional Failures (SSFF). This PI remained in the green band. The inspector reviewed licensee event reports between the 2nd quarter of 2000 until the 2nd quarter of 2001.

b. Issues and Findings

Licensee event report 05000247/2000-006 documented that both source range instrument channel trip setpoints were outside the design basis due to the failure to account for postulated worst case ambient temperatures in the control room. Entergy did not classify this event as a safety system functional failure because the source range high flux trip is not credited in the UFSAR Chapter 14 accident analysis. The source range nuclear instruments are required to be operable per the technical specifications. NUREG-1022, Section 3.2.7, states that a failure of any component listed in the technical specification to perform a safety function, including shutdown of the reactor, is considered reportable under in 10 CFR 50.73(a)(2)(v). Further, if reported under this criteria, the failure would then meet the definition of a safety system functional failure. This item is considered unresolved pending further review by the NRC **(UNR 05000247/01-09-01).**

.2 Scrams with Loss of Normal Heat Removal

a. Inspection Scope (71151)

The inspector reviewed the Performance Indicator (PI) for Scrams With loss of Normal Heat Removal. This PI remained in the green band. The inspector reviewed licensee event reports between the 2^{nd} quarter of 2000 through the 2^{nd} quarter of 2001.

b. Issues and Findings

No significant findings were identified

.3 Reactor Coolant System Leakage

a. Inspection Scope (71151)

The inspector reviewed the Performance Indicator (PI) for Reactor Coolant System (RCS) Leakage for the period from February until June 2001. This PI remained in the green band. The inspector reviewed the completion RCS leak rate determinations per SOP 1.7 to verify the adequacy of the reported PI data. The licensee's corrective action program records were also reviewed to determine if any problems with the collection of PI RCS Leakage data had occurred.

b. Issues and Findings

No significant findings were identified.

.4 Reactor Coolant System (RCS) Activity

a. Inspection Scope (71151)

The inspector reviewed the Performance Indicator for Reactor Coolant System Activity for the period from January through June 2001. This PI remained in the green band.

b. Issues and Findings

(**No Color**) The inspector identified errors in the RCS Activity PI data reported for the second quarter of 2001. The indicator value for RCS dose equivalent iodine-131, expressed as a percent of the 1.0 micro-curie/gram limit, was reported at 0.1 for January through March, increasing to 0.8 for April and May, and returning to 0.1 in June 2001. The RCS chemistry data showed actual dose equivalent iodine remaining below 0.08 through June 8, and increasing to 0.142 after a fuel leak developed which was estimated to be a pin-hole defect in one fuel rod (reference NRC Inspection Report 05000247/01-06, Section 1R15). Further, the RCS chemistry data for April 2001 was reported as the indicator value for May 2001, and the May data was reported for April. Transcription errors and ineffective review contributed to the errant PI data.

The consequence of the specific errors in the second quarter 2001 data was minimal since the PI remained within the green band. The inspector noted that previous inspection findings identified errors in reporting Indian Point 2 PI data (reference NRC Inspections 05000247/00-01 and 00-11). This issue has more than minor significance because the failure to accurately report PI data potentially could impact the ability of the NRC to perform its regulatory function. The licensee issued Condition Report 200109517 to address this matter in the corrective action program.

4OA2 Cross Cutting Issues

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

.1 Radiological Controls for Fuel Movement

a. Inspection Scope (71111.20)

The inspection scope was to review the adequacy of radiological controls for work in the primary auxiliary building on September 12, 2001, including the controls in place during the movement of spent fuel in the spent fuel pool.

b. Issues and Findings

The inspector identified two discrepancies in radiological controls. First, the health physics (HP) technician at check point HP-1 provided an incomplete briefing regarding the radiological conditions in the primary auxiliary building, in that the technician was not aware of new radiological conditions in the safety injection pump room, and was not aware of work activities that were in progress in the spent fuel pool under Radiation Work Permit (RWP) 010135. The incomplete briefing was attributable, in part, to inadequate communications and shift turnovers between members of the HP staff.

A second human performance discrepancy was the failure to provide intermittent health physics coverage of the work in the spent fuel pool as required by RWP 0100135 during the afternoon of September 12, 2001. The assigned HP technician provided continuous coverage as required when spent fuel was moved in the pool, but left the site while activities were in progress to train new operators on use of the fuel handling machine using the dummy fuel assembly. The lapse in HP coverage occurred, in part, because the work activities in the spent fuel pool continued longer than had been communicated to the HP staff.

These issues had minor safety significance because there was no significant loss of worker radiological protection. The licensee addressed these matters in Condition Report 200108736.

.2 Operator Performance During an Overpower Event

a. Inspection Scope (71153)

The licensee returned plant operation to full power on August 17, 2001, following a load reduction to repair a main boiler feedwater pump. The inspector reviewed plant operations resulting in an overpower condition on August 17 and the licensee's response to the event. The overpower condition occurred because of human performance errors in how the operators increased power during a turbine load adjustment, and in the manner in which operators followed plant procedures. A condition report (200108052) was initiated at the time of the overpower condition on August 17 based on plant power reaching 101.5%; however, a root cause investigation team was not established until September 18, 2001, when the licensee became aware that the overpower reached 102.7% of the licensed thermal limit for five minutes. The licensee classified the overpower as a significant reactivity management event (reference SL-1 Report dated October 5, 2001), and reported this matter as Licensee Event Report 05000247/2001-00.

The inspection scope involved interviews with operators, review of calorimetric data for all 2001 power changes, walkthrough of the calorimetric process with a reactor operator, review of operator logs and observation of the corrective action review board's evaluation of the root cause. During the inspection, the inspector identified a number of minor procedural quality items and accuracy of operator aids. The issues were documented in Condition Reports 200109039, 200109065, and 200109070.

b. Issues and Findings

(GREEN) During a power increase from 94% on August 17, 2001, the operators' failed to adhere to system operating procedure (SOP) 15.1, "Reactor Thermal Power Calculation," revision 24, and plant operating procedure (POP) 1.3, "Plant Startup from Zero Power Condition to Full Power Operation," revision 55. The operators' failure to adequately adhere to SOP 15.1 and adequately monitor plant conditions resulted in the following consequences:

- power range nuclear instruments (PRNIs) were adjusted 1.2 to 2% lower than actual calorimetric power;
- the reactor protection system trip setpoint for PRNI N-44 was 109.6%, which exceeded the technical specification 2.3.1.B(1) limit of 109% for approximately 3.5 hours; and,
- reactor thermal power exceeded the License Condition 2.C.(1) limit of 3071.4 megawatts and reached 102.7% of full power for five minutes.

The existing overpower condition was aggravated by an excessive turbine load increase while at full power. The operators failed to perform additional calorimetrics as required by procedures, which could have identified the original errors and non-conservative PRNI adjustments. The operators failed to adhere to administrative orders for reactivity manipulations; specifically, the operators failed to monitor reactor power using all available indications, such as reactor coolant system average temperature and loop differential temperature.

The operator performance issues were more than minor since they had a credible impact on safety during the August 17 overpower condition. The safety significance was that one of the four power range trip setpoints was non-conservative, and thermal power was greater than 102%. Reactor thermal power greater than 102% exceeded the assumption for steady-state power in the accident analyses, as described in Updated Final Safety Analysis Section 14.0.2.1. The licensee's event analysis demonstrated that the consequences of analyzed accidents remained bounding despite the overpower condition. The overpower condition impacted the reactor safety cornerstone since it could have caused a reactor trip. This issue was evaluated in phase 1 of the SDP and was found to have very low safety significance, since the minor overpower condition occurred for a short time (5 minutes), and the condition did not impact the function or availability of mitigation equipment.

Technical Specification 6.8.1.a, requires, in part, written procedures to be implemented for activities referenced in Appendix "A" of regulatory Guide 1.33, Rev. 2. Appendix A includes the requirement for items "1d", "Procedural Adherence," "2f", "Changing Load," and "8.b.(1)(w), "Heat Balance - Flux Monitor Calibrations." Station Administrative Order (SAO)-133, "Procedures, TS, and License Adherence and Use Policy," step 4.1 requires that procedures shall be followed. The inspector identified the following:

- SOP 15.1 Step 2.5 requires that the reactor coolant system average temperature (Tave) be maintained within one degree Fahrenheit of the reference temperature (Tref) during a calorimetric. On August 17, the operators failed to maintain steady state plant conditions with Tave within one degree of Tref during a calorimetric at 92% power. The actual difference between Tave and Tref increased to 1.8 degrees Fahrenheit during the calculation of thermal power. The failure to maintain plant conditions required for the calorimetric resulted in a non-conservative adjustment of the PRNIs.

- The inspector identified that the operators did not adjust the power range instruments as required by the plant procedures. SOP 15.1 Step 2.9 and POP 1.3 Step 4.66.2 require that the power range nuclear instrument (PRNI) gains be adjusted to 2% above the calculated thermal power when the plant is between 70 - 90% power. Contrary to this, on August 17, the operators failed to adjust PRNI to 2% above the calculated power prior to exceeding 90% power.

- SOP 15.1 Step 4.6.4 requires that when the PRNIs are adjusted with reactor power greater than 90% full power, the gain adjustment be limited to one-half of one percent power when indicated and calculated thermal power differ by more than 1%. Contrary to this, on August 17 operators adjusted all four PRNI 's the full difference of 2.9% from 94.1% to 91.2%.

The licensee identified all but one of the procedural violations during the post event review of the August 17 overpower condition (reference Condition Report 200108502). The multiple examples of a failure to adhere to procedures is considered a violation of

TS 6.8.1.a. This violation is treated as a non-cited violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). The licensee entered these issues in Condition Report 200108052. **(NCV 50-247/01-09-02)**

License condition 2.C (1) authorizes the licensee to operate the facility at steady state reactor core power levels not to exceed 3071.4 megawatts thermal (100%). Contrary to this requirement on August 17, 2001, between 2:24 p.m. until 3:00 p.m., the operator caused reactor power to exceed the licensed power limit. Within this same 26 minute time frame, reactor power exceeded 102% of the licensed limit for 5 minutes. The operator failed to monitor reactivity parameters during the power increase as required in SAO-442, "Reactivity Management," Step 4.2.4.c, and OAD-39, "Reactor Power Control," Steps 5.18 and 6.22. Specifically, operators failed to monitor reactor power using all available indications (such as reactor coolant system average temperature and loop differential temperature) during the power increase to preclude a violation of License Condition 2.C.(1). This violation was entered into the licensee's corrective action program as Condition Report (CR) 200108052. This violation is treated as a non-cited violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). **(NCV 50-247/01-09-03)**

.3 Problem Identification and Resolution

a. Inspection Scope (71153)

The inspector evaluated the effectiveness of corrective actions taken in response to a January 2, 2001, turbine trip (reference Inspection 05000247/2000-15, LER 05000247/2001-001, and Condition Report 200100048), as they related to the licensee's identified causes for the overpower event on August 17, 2001 (reference CR 200108502). The inspector also observed short-term corrective actions (an operations stand down on September 21, 2001), reviewed Entergy's root cause investigation and evaluated the team's conclusions, and evaluated the corrective action effectiveness from a previous human performance event.

b. Issues and Findings

(No Color) The licensee corrective actions in response to past reactivity management and plant events (CR 200100048 and CR 200100364) were ineffective in precluding inadequate control room log keeping, procedural adherence, and post-evolution debriefs, which contributed to the August 17, 2001, overpower condition and untimely management response. Past corrective actions included refresher operator training on proper log keeping and shift manager observations of log keeping practices, various operation and station stand downs on procedural adherence, and administrative guidance established for event response review particularly for significant reactivity management events (i.e. unplanned power increase of 2%). For the August 17, 2001, overpower event, no control room log entry indicated any overpower condition or actions to restore power; there were no log entries regarding the problems completing the calorimetric and nuclear instrument calibrations; and, there was no post evolution de-brief between operators to assure adequate communication and understanding of the overpower condition by all members of the shift crew.

The initial and short term licensee response to issues described in CR 200108502 did not assess the significance of the event in a timely manner. Multiple opportunities existed for the licensee to have timely identified the event significance, including: periodic management discussions with the operators, a condition report screening committee meeting on August 20, 2001, and reactor engineering evaluations. A reactor engineering evaluation on September 14 concluded that exceeding 102% full power was a significant reactivity management event, yet upper management was not notified, nor was an investigation team initiated pursuant to SAO-442.

The SL-1 investigation team that began activities on September 18, 2001, adequately evaluated the event and determined logical root and contributing causes. The corrective action review board contributed additional proposed corrective actions to address the initial inadequate review of the August 17 event by the condition report screening committee. This issue is considered more than minor because of an ongoing adverse trend in the problem identification and resolution process as previously documented in NRC Inspections 05000247/01-08, 01-04, 01-03, and 01-02. NRC review of operator actions regarding this overpower event and with other activities where similar operator errors have been observed continued at the conclusion of this inspection period.

The failure to take effective corrective actions associated with procedural adherence, log keeping, and prompt identification of conditions adverse to quality is considered a violation of 10 CFR 50 Appendix B, Criterion XVI. This issue had low actual safety significance because additional corrective actions and effectiveness reviews have been assigned to CR 200108502 that document the above 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 50-247/01-09-04)**

4OA3 Inspection Item Followup (71153)

.1 (Closed) URI 05000247/2001-03-06: Review Changes to the Facility per 10 CFR 50.59. The NRC's review of this area was summarized in NRC Inspection 05000247/01-05. The open NRC concerns in this area, including changes to the reactor protection system per 10 CFR 50.59 and the adequacy of the wiring separation criteria, are tracked per item 05000247/2001-05-02. This item is closed.

4OA5 Other

a. Scope

The inspector reviewed the circumstances surrounding an Administrative Order of Consent between the Consolidated Edison Company of New York (former licensee of the Indian Point Unit 1 and 2 Stations) and the New York State Department of Environmental Conservation, dated September 5, 2001. The Order relates to storage of mixed waste at Indian Point Units 1 and 2.

The inspector reviewed the licensee's waste storage practices for the locations identified in the Order. The risk significance of the findings in this area are discussed in Section 2PS2 of this report.

b. Findings

No significant findings were identified.

4OA6 Meetings

Exit Meeting Summary

On October 19, 2001, the inspector presented the inspection results to Mr. F. Dacimo 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**

J. Cottam	Fire Protection Engineer
K. Cullen	Health Physics Technician
M. Dampf	Radiation Protection Special Projects
M. Donegan	Health Physics Manager
C. English	Radioactive Waste Manager
R. Fucheck	Health Physics Supervisor
E. Libby	Licensed Operator Instructor
M. Miele	Radiation Protection Department Manager
T. McCafferty	System Engineering Manager
M. Miller	Manager, Generation Support
W. Osmin	Reactor Engineer
J. Reynolds	SL-1 Team Leader
J. Rodriguez	Nuclear Production Technician
R. Rose	Security Manager
W. Scholtens	Radioactive Waste Specialist
G. Schwartz	Chief Engineer
W. Smith	Manager, Operations
R. Sutton	Maintenance Rule Coordinator
L. Temple	Plant Manager
J. Touhy	Manager, Design Engineering
M. Vaseley	System Engineer Supervisor
T. Wadell	Manager, Maintenance

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

Opened

05000247/01-09-01	UNR	Reporting Safety System Functional Failures in PI Data

Opened and Closed During this Inspection

05000247/01-09-02	NCV	Several Examples of Failure to Follow Calorimetric Procedure
05000247/01-09-03	NCV	Poor Reactivity Management Caused Violation of Power Limit
05000247/01-09-04	NCV	Inadequate Corrective Actions Contrary to Criterion XVI

Closed

05000247/01-03-06	URI	Changes to the Facility (RPS) per 10 CFR 50.59

c. **List of Acronyms**

AOI	Abnormal Operating Instruction
BTP	branch technical position
CFR	Code of Federal Regulations
CR	Condition Report
DOT	Department of Transportation
EPRI	Electric Power Research Institute
HP	health physics
ILRT	integrated leak rate test
IRC	Incident Response Center
IPEEE	Individual Plant Examination of External Events
KV	kilovolt
LLC	Limited Liability Corporation
NRC	Nuclear Regulatory Commission
PARS	publicly available records
PCP	Process Control Program
PCV	pressure control valve
PI	performance indicators
PS	Public Safety
QC	Quality Control
RCA	Radiologically Controlled Area
RCS	reactor coolant system
RWP	radiation work permit
SAO	station administrative order
SCFM	standard cubic feet per minute
SDP	significance determination process
SI	safety injection
SSC	structure, system and component
TPC	temporary procedure change
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
V	volt
WC&PPS	weld channel and penetration pressurization system