December 12, 2000

Mr. Robert J. Barrett
Vice President Operations
Entergy Nuclear Operations, Inc.
Indian Point 3 Nuclear Generating Unit 3
P.O. Box 308
Buchanan, NY 10511

SUBJECT: NRC'S INDIAN POINT 3 INSPECTION REPORT NO. 05000286/2000-007

Dear Mr. Barrett:

On November 18, 2000, the NRC completed an inspection at the Indian Point 3 nuclear power plant. The enclosed report presents the results of that inspection. The results were discussed on December 1, 2000, with you and other 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. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

No findings of significance were identified.

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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

James C. Linville, Chief
Projects Branch 6
Division of Reactor Projects

Docket No.05000286
License No. DPR-64

Enclosure: Inspection Report No. 05000286/2000-007

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	05000286
License No.	DPR-64
Report No.	05000286/2000-007
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point 3 Nuclear Power Plant
Location:	P.O. Box 308 Buchanan, New York 10511
Dates:	October 1 - November 18, 2000
Inspectors:	Peter Drysdale, Senior Resident Inspector Lois James, Resident Inspector Kenneth Jenison, Senior Project Engineer John McFadden, Health Physicist
Approved by:	James Linville, Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000286/2000-007; on 10/01 - 11/18/00; Entergy Nuclear Northeast; Indian Point 3 Nuclear Power Plant. Licensee identified violations.

The inspection was conducted by resident inspectors, a regional radiation specialist, and a regional senior project engineer.

A. Inspector Identified Findings

None.

B. Licensee Identified Violations

Violations of very low significance which were identified by the licensee have been reviewed by the inspector. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in section 40A7 of this report.

TABLE OF CONTENTS

ATTACHMENT

Report Details

SUMMARY OF PLANT STATUS

The Indian Point 3 plant remained at full power during October 1 - 24, 2000. On October 25, 2000, the plant was taken offline to conduct repairs on the 34 instrument bus static inverter. Plant power was raised to 50% on October 27 and repairs were completed on two control rod drive mechanism (CRDM) fans and motors. Full power operations have been maintained since October 28, 2000.

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

1R01 Adverse Weather Protection

a. Inspection Scope (71111.01)

The inspector reviewed Operations Directives OD-08, "Guidelines for Severe Weather," and OD-27, "Operations Shift Schedules." The inspector also accompanied operations personnel during shift rounds to observe actions taken to assure that low ambient temperature conditions were monitored, and that the effects of low air temperatures on plant equipment were minimized.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope (71111.04)

During November 16-17, 2000, the inspector performed a partial walkdown of the safety-related instrument air (IA) system using Checkoff List, COL-IA-1, "Instrument Air System," and system flow diagrams 9321-F-20353 and -20363. During this inspection, the 32 IA compressor was out of service for preventive maintenance and overhaul, and the 32 IA compressor closed cooling supply valve SOV-1178 was being replaced. The inspector verified the valve lineup in the 31 & 33 IA trains, and verified the 32 train isolation valve positions specified in protective tagout (PTO) 00-1533. The inspector also reviewed System Operating Procedure SOP-IA-1 "Instrument Air System Operation," to confirm that the system was operated in accordance with its normal operating procedure while the 32 compressor was out of service. Following the 32 compressor overhaul, the inspector reviewed PTO 00-1581 which applied three test and measurement tags during hydrostatic testing of the compressor after cooler. The inspector also reviewed problem identification (PID) tag 44648 and deviation/event report (DER) 00-02519, which documented a cylinder head leak on the 31 IA compressor. The inspector discussed with the shift manager the potential operational significance of the head leak since it had grown progressively worse since the condition was identified on October 13, 2000.

Following the performance of surveillance tests affecting auxiliary feedwater (AFW) system valves and the functioning of the AFW system during a recent plant shutdown, the inspector verified that selected AFW system valves and electrical components were

appropriately aligned in accordance with appropriate plant design drawings and instrumentation diagrams.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope (71111.05Q)

The inspector conducted tours of the plant to verify the availability and material condition of fire protection and suppression equipment in the plant equipment areas listed below. The inspector also examined the programmatic controls for combustible and flammable material, and referred minor concerns regarding potential transient combustibles to the fire protection department, operations management, and "area owners" responsible for housekeeping.

- Central Control Room
- Upper and Lower Cable Tunnels
- 31, 32, & 34 Battery Rooms
- Auxiliary Boiler Feed Pump Room

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspector reviewed problems involving selected in-scope structures, systems, and components (SSCs) to assess the effectiveness of the maintenance program. The review included a sample of operating logs, system engineer data, system reports, deficiency reports, availability data, selected surveillance performance data, and selected maintenance-related data. The review focused on proper maintenance rule scoping, proper classification of SSC equipment failures, safety significance classifications, 10 CFR 50.65 (a)(1) and (a)(2) classifications, and performance criteria for SSCs classified as (a)(2). The inspector reviewed NYPA's scoping documents, deficiency/event reports (DERs), and completed work orders. The following SSCs were reviewed:

- 33 steam generator atmospheric relief valve (MS-PCV-1136 and -1137) unavailability for seat leak repairs, packing adjustment, and stroke time testing; MS-PCV-1136 failed its in-service test (IST) acceptance test, DER 00-02937

- Auxiliary Feedwater System

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work

a. Inspection Scope (71111.13)

The inspector reviewed the maintenance risk assessments and corrective maintenance work packages for the following emergent work, and discussed the deficient conditions with cognizant personnel (system engineers, maintenance technicians, etc.):

- Replacement of the 34 static inverter electrolytic capacitors; WR 00-4777-01; DER 00-02584, and Action Plan IDSE-APL-00-010.
- Replacement of the 31 & 33 control rod drive mechanism fans and motors, and installation of temporary power cables; WR 00-03216-05; DERs 00-00334, 00-01151, 00-02038, and 00-02784.
- Troubleshooting over-pressure delta-temperature instrument spikes in reactor protection system channel, and subsequent replacement of bistable TC-421C/D; DER 00-02362, WR 00-05002-00.
- Packing leak and torque adjustment on the 34 main feedwater regulating valve; DER 00-02971

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (71111.14)

Planned Plant Shutdown on October 25, 2000

At approximately 6:00 p.m. on October 25, 2000, operators commenced a normal plant shutdown for corrective maintenance on the 34 static inverter, a vital battery ground, two Control Rod Drive Mechanism (CRDM) fans and motors, and a pressurizer instrument line leak. The shutdown was administered as an infrequently performed procedure requiring numerous meetings and training sessions, Plant Oversight Review Committee (PORC) approval, and additional staffing.

Prior to the shutdown, the inspector discussed with the operations manager documented plant deficiencies including main feedwater regulating valves, sticking secondary plant check valves, auxiliary feed water (AFW) check valve and flow indication, and main feedwater pump vibration. Each of these issues was known and documented by the licensee.

Control room operators were challenged during the power reduction and shutdown when numerous alarm conditions resulted from problems with controlling levels in the steam generators, feedwater heaters, and the heater drain tank. Also, anomalies occurred in three control room indicators for reactor delta-flux conditions and minor plant load fluctuations. At approximately 15% power, the operators were unable to maintain a stable level in the 33 steam generator with its main feedwater regulating valve. The

licensee's investigation found that the No. 33 feed regulating valve (FRV) had stuck open at approximately 12%. When the steam generator level rose, the average reactor coolant system (RCS) temperature (Tave) and pressurizer level dropped, the chemical and volume control system (CVCS) letdown isolated, and operators entered off-normal operations procedure ONOP-CVCS-2, "Reactor Coolant Make-Up Control System Malfunction." While in this condition, operators also entered off-normal procedure ONOP-TG-4, "Turbine Trip Below P-8," because of an oscillating level in the No. 33 steam generator and a continuing plant cooldown. Operators manually tripped the main turbine at about 9:42 p.m., but plant power was below the automatic turbine/reactor trip permissive (P-8), so a reactor trip did not result. Following the turbine trip, operators regained stable plant conditions. The reactor was taken subcritical at 10:18 p.m. and the reactor trip breakers were opened at 10:47 p.m.

Plant Restart on October 27, 2000

During the period of October 26-27, 2000, the inspector observed operator action during plant restart following the October 25 planned, forced outage. The plant restart was delayed several times to resolve the following equipment deficiencies:

- Main Boiler Feed Pump (MBFP) Control Room Indications - During the maintenance outage, the licensee replaced one of the two MBFP control room display panels because the digital indication was fading. While increasing the speed of MBFP No. 32, the corresponding control room indicated speed was erratic and jumped from 49% speed to 90% speed. The inspector observed the licensee's reinstallation of the indication panel removed during this outage.

- Individual Control Rod Indicators (ICRI) - Two of the individual control rod drive indicators malfunctioned. ICRI F-10 failed to indicate and required replacement. The ICRI F-10 failure was entered into the licensee's corrective action system, DER 00-02766. ICRI L-13 indication stopped at 60 steps and initially failed to move with the rest of the control bank "A" indicators. The concern with ICRI L-13 was entered into the licensee's corrective action system, DER 00-02768.

- Heater Drain System - Over 75 feedwater heater and 75 heater drain tank level alarms annunciated during a 6 hour period. The licensee entered the heater drain system issues into the corrective action system, DER 00-02778.

During the plant startup, the licensee identified a condition that was not in accordance with Technical Specification (TS) 3.10.3,1 which states that whenever the quadrant power tilt ratio exceeds 1.02, plant power will be restricted, and the nuclear instrumentation high flux trip setpoint will be readjusted within 2 hours. The operations department determined that the quadrant power tilt was 1.03, exceeding the 1.02 ratio limit, at approximately 17%, however, actions to restrict plant power and adjust the high flux trip setpoint were not taken until approximately 45% power. The licensee entered this issue into the correction action process (DER-00-02781) and planned to report it under 10 Code of Federal Regulations 50.73(a)(2)(i)(B), as a condition prohibited by TS. (see Section 4OA7)

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspector reviewed various DERs on degraded or non-conforming conditions that raised questions on equipment operability. The inspector reviewed the resulting operability determinations (ODs) for technical adequacy, whether or not continued operability was warranted, and to what extent other existing degraded systems adversely impacted the affected system or compensatory actions. The following DERs, calculations, and operability evaluations were evaluated:

- OD 00-28, Revised minimum thrust requirements for motor-operated valve CH-MOV-222 (reactor coolant pump seal return containment isolation valve) with valve disks in the non-preferred orientation; DER 00-02338; Revised calculation IP3-CALC-CVCS-01037

- OD 00-029, Degraded Appendix R diesel generator battery voltage; DER 00-02414

- OD 00-030, Degradation of capacitors in the 34 static inverter; DER 00-02683

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope (71111.19)

The inspector reviewed post-maintenance test 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 following surveillance activities were evaluated:

- 3PT-M090, “Appendix R Diesel Generator Functional Test,” following preventive and corrective maintenance on October 6, 2000

- WR 00-04924-00; post-maintenance leak test instructions for pressurizer instrument (PT-451) line leak following preventive and corrective maintenance on October 26, 2000

- WR 00-04777-01 and -02; post-maintenance test instructions following replacement of electrolytic capacitors in the 34 static inverter, which included inverter load testing and infrared thermography

- ENG-259I, “32 Service Water Pump Reference Test;” post-maintenance test for baseline performance of the No. 32 service water pump following replacement on November 2, 2000

- 3PT-CS030, “Atmospheric Steam Dump Valves Stroke Test;“ post-maintenance test following packing adjustments on the Nos. 33 and 34 atmospheric relief valves

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspector reviewed surveillance test procedures and associated testing activities to assess whether (1) the test preconditioned the component(s) tested, (2) the effect of 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 were adequate with proper calibration, (5) the test was performed in the proper sequence, and (6) the test equipment was removed following testing. The inspector also reviewed the licensee’s actions to address Unresolved Item (URI) 05000286/1999-010-01.

The inspector reviewed/observed portions of the following surveillance tests and performed a review of related historical data and surveillance performance.

- 3PT-Q31, “Inservice Inspection Test Liquid Waste Disposal System.” Containment isolation valve WD-AOV-1610 failed its stroke time test on October 9, 2000 (DER 00-2553). Containment integrity was not challenged due to credit taken for the downstream check valve WD-1616. The licensee previously addressed this condition in DER 00-00953.

- 3PT-M79C, “33 EDG Functional Test;” October 31, 2000

- 3PT-Q116B, “32 Safety Injection Pump Functional Test;” November 1, 2000

- 3PT-Q120B, “32 ABFP (Turbine Driven) Surveillance and IST,” November 2-3, 2000

- 3PT-M13B1, “Reactor Protection Logic Channel Functional Test,” November 6, 2000

- 3PT-Q92D, “34 Service Water Pump Functional Test;” November 15-16, 2000

(Closed) Unresolved Item (URI) 1999-009-01, Use of the Plant Computer to Perform Core Thermal Power Calculation

URI 1999-010-01 was opened in January 2000 following NRC concerns that the licensee did not have a formal basis for the use of a non-qualified instrument (i.e., plant computer) in order to verify compliance with the technical specification limit on core thermal power. The licensee subsequently developed an action plan to review all parameters and functions from the plant computer that were used for technical specification compliance, and initiated detailed reviews and

actions to verify that the computer would provide an accurate source of plant process data. The inspector concluded that the licensee's actions to address this item were adequate to confirm that the computer will not introduce an additional level of uncertainty or inaccuracy into the calculations. Based upon the licensee's actions, this item is closed.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23A)

The inspector reviewed the work packages for temporary modification (TM) 00-02778-10 for the 33 control rod drive mechanism (CRDM) fan motor, and TM 00-03215-08 for the 31 CRDM fan motor, which routed temporary power cables from alternate sources to both CRDM fan motors. The licensee's evaluation of the cable ampacities showed that the pre-existing cables did not have adequate ampacity to support continuous operation of four fans at the same time. The inspector also evaluated the licensee's administrative requirements, and the work packages used to install the temporary modifications and to conduct post-installation testing.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope (71121)

The inspector reviewed the procedures, records, and program documents listed below to evaluate the effectiveness of the licensee's access controls to radiologically significant areas. The inspector also observed activities at the routine radiologically-controlled-area (RCA) control point on the fourth floor of the Administration Building on October 3, 4, and 5 to verify compliance with requirements for RCA entry and exit, wearing of record dosimetry, and issuance and use of electronic dosimeters. On October 3, 2000, during a tour of elevations 15, 32, 41, 55, and 73 of the Primary Auxiliary Building (PAB), elevation 80 of the fan house, elevations 55 and 95 of the fuel storage building, and elevations 44, 54, and 73 of the radioactive machine shop (RAMS) building, the inspector reviewed the posting and labeling of radiation and contamination levels and of radioactive material and the radiological control technician activities in support of the active routine radiation work permit (RWP) activities. Additionally, the inspector reviewed the access controls at each high radiation and locked high radiation area. During a tour of the exterior yard within the protected area, the inspector witnessed a radiological control technician performing area surveys at the northeast corner and establishing a radiological control boundary during the passage of a used steam generator from the Indian Point 2 containment to its onsite storage location.

- RE-ACC-5-1, "Radiologically Controlled Area (RCA) Access Control"
- RE-ACC-5-2, "Instructions to Control Point Personnel"
- RE-ADM-1-7, "Health Physics (HP) Access Key Control"
- RE-SUR-6-6, "HP Periodic Task Scheduling"
- RE-INS-7UG-4, "Use of Merlin-Gerin DMC-100 Dosimeters"
- RE-UOE-14-9, "Lost, Damaged, Off-scale, Alarming, or Abnormal-Reading Dosimetric Devices"
- UOE-14-4, "Radiological Event Classification and Investigation"
- IP-3 Personnel Contamination Events Greater Than 5000 NCPM
- Second Quarter 2000 Radiological Event Report Summary
- Quality Assurance (QA) Audit Plan, Audit No. 00-141, September 2000
- DER/ACTS Listing, QA Audit No. 00-141, September 2000
- QA Surveillance Report No. 00-19, Spent Resin Transfer, July 27, 2000
- Annual Radiological and Environmental Services (RES) Self-Assessment Plan - 2000, Status, September 15, 2000
- RES Department Annual Self-Assessment, June 99 to July 00, July 18, 2000

The inspector reviewed the following deviation/events reports (DERs) and their associated Action Commitment Tracking System (ACTS) items for appropriateness of categorization, immediate correction actions, and corrective actions to prevent recurrence and for the timeliness and effectiveness of corrective actions:

DERs 00-00743, 00-00835, 00-01136, 00-01786, 00-01894, 00-01966, 00-02078, 00-02369, 00-02284, 00-02420, 00-02498. These DERs were generated during the period of January to October 2000.

The review was against criteria contained in 10 Code of Federal Regulations (CFR) 10 CFR 20.1301 (Dose limits for individual members of the public), Subpart F (Surveys and monitoring), 20.1601 (Control of access to high radiation areas), 20.1902 (Posting requirements), site Technical Specification 6.12 (High Radiation Area), and site procedures (cited above in this section).

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope (71121)

The inspector reviewed the program documents and records listed below to determine the effectiveness of ALARA (As Low As Reasonably Achievable) planning and control. On October 3 and 5, 2000, the inspector toured various elevations of the primary auxiliary building, the fan house, the fuel storage building, and the RAMS building and observed the use of signs to identify low dose waiting areas and to identify higher dose areas where access time should be minimized.

- RE-REA-4-1, “Radiation Work Permits”
- RE-ALA-2-3, “Temporary Shielding Control”
- Quarterly Review of Station ALARA Program, May 3, 2000
- Quarterly Review of Station ALARA Program, September 12, 2000
- ALARA Performance, September 6, 2000

During the week of October 2 - 6, 2000, the inspector examined the following RWP packages from the last refueling outage (RFO-10) to evaluate the adequacy of the work permit request, radiological surveys of job locations, the pre-job ALARA review, the approved RWP, pre-job briefing records, on-the-job surveys and ALARA reviews, post-job ALARA reviews, documentation of lessons learned, and the comparison of the estimated job dose versus the actual dose incurred for the job.

- RWP 99310, “Construction Work/Other Than Scaffolding in Vapor Containment (VC)”
- RWP 99315, “Scaffolding Work (VC/PAB)”
- RWP 99320, “Reactor Head Work (Mechanical)”
- RWP 99324, “Refueling Activities(VC/PAB)”
- RWP 99336, “In-Service-Inspection Work (VC/PAB)”
- RWP 99337, “Reactor Coolant Pump Motor Work”
- RWP 99351, “Motor-Operated Valve Actuator Testing (VC/PAB)”
- RWP 99358, “Remove/Replace CH-342"

The review was against criteria contained in 10 CFR 20.1101 (Radiation protection programs), 10 CFR 20.1701(Use of process or other engineering controls), and site procedures (cited above in this section).

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

a. Inspection Scope (71151)

Transients per 7000 Critical Hours

The inspector reviewed the licensee's data supporting the performance indicator for the numbers of plant transients, unplanned power changes, and fluctuations in reactor power greater than 20 percent per 7000 hours of critical operation. The inspector used Licensee Event Reports (LERs) for the past year, Nuclear Energy Institute (NEI) 99-02 "Regulatory Assessment Indicator Guideline," deviation/event reports (DERs), and control room operating logs for this review.

Reactor Coolant System Specific Activity

The inspector reviewed the licensee's performance indicator for reactor coolant system (RCS) specific activity for the third quarter 2000. The licensee reported the maximum Iodine-131 activity in microCuries per gram dose equivalent obtained from RCS chemistry sample analyses that decayed for two weeks. The inspector used these two-week analyses to verify that the percent of technical specification (TS) limit was the same or lower than the maximum value reported by the licensee for the months of July, August, and September 2000. In addition to record reviews, the inspector observed a chemistry technician obtain and analyze an RCS sample.

Occupational Exposure Control Effectiveness

The inspector selectively examined records used by the licensee to identify occurrences involving high radiation areas, very high radiation areas, and unplanned personnel exposures for the period from June 5, 2000 to October 6, 2000 against the applicable criteria specified in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 0, to verify that all conditions that met the NEI criteria were recognized and identified as Performance Indicators. The reviewed records included corrective action program records (Deviation and Event Reports) and radiologically- controlled-area (RCA) access-control alarm reports. This examination did not find any problems with the performance indicator (PI) accuracy or completeness.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

a. Inspection Scope (71153)

At approximately 6:00 p.m. on October 25, 2000, plant operators commenced a normal shutdown to perform corrective maintenance. The inspector noted several difficulties during the shutdown evolution:

- Three anomalous indicators occurred in the control room involving feedwater heater level, heater drain tank levels, steam generator levels, reactor delta flux conditions, and minor load fluctuations.

- Operators' inability to maintain level in one steam generator with the 33 main feedwater regulating valve at approximately 10-12% power as indicated by a rise in steam generator level, a drop in average temperature, a drop in pressurizer level, and isolation of the chemical and volume control system.

The licensee convened a Plant Transient Review Group (PTRG) to review the shutdown and equipment problems. The inspector observed the PTRG's data collection, discussion meetings, and presentation to the Plant Manager. In addition, the inspector reviewed the PTRG's final report. The inspector reviewed the following items:

- Main Feedwater Regulating Valves (FRVs) 33 and 34

  During the shutdown, the 33 FRV stuck open at approximately 10-12% and 34 FRV at approximately 2-4%. The opened 33 FRV resulted in the 33 steam generator level increasing approximately 20% in a six minute period to a high of 63%. Also during this time period, the 33 high feed flow mismatch alarm was received. In response to the high level in the 33 steam generator, RCS temperature began to decrease and continued to decrease causing the operating crew to manually trip the turbine to avoid an excessive cooldown.

  During the event investigation, the licensee discovered the high packing torque on the 33 FRV that caused stem friction. The inspector followed the corrective action identified during the investigation including lubricating the stem, adjusting the valve packing, and stroke testing the valve.

- 33 high feed flow mismatch alarm

  Instrument and control technicians completed the surveillance to validate the setpoint and the results were satisfactory in response to the alarm being received during the shutdown. The inspector reviewed the completed surveillance.

- 33 low-flow steam flow transmitter

  Instrument and control technicians completed the surveillance to calibrate this flow transmitter and the results were satisfactory in response to the 33 high feed flow mismatch alarm being received during the shutdown. The inspector reviewed the completed surveillance.

Licensee Event Report Reviews

(Closed) LER 2000-009-00 Plant Outside Design Basis Due to Valve Seat Leakage During Maintenance That Exceeded the External Recirculation Leakage Limits for the Control Room Ventilation System

On August 17, 2000, the licensee submitted LER 2000-009-00 to the NRC to report a condition where the reactor coolant system indicated leakage outside containment

increased to approximately 0.3 gallons per minute through an open path in the chemical and volume control system (CVCS) during maintenance. This leakage exceeded the assumed design basis maximum for the central control room (CCR) ventilation system, and the calculated control room dose following a design basis accident. The licensee identified a degraded CVCS relief valve as the cause of this event, which provided a reverse flow path outside containment and into the primary auxiliary building through a valve bonnet that was removed for maintenance. The licensee's subsequent corrective actions were adequate to stop the leakage, and to restore RCS leakage outside containment to below CCR ventilation limits. The inspector considered that the LER adequately addressed this event, identified the appropriate corrective actions, and satisfied the reporting requirement of 10 CFR 50.73.

(Closed) LER 2000-010-00 Isolation of the Containment Spray System Spray Additive Tank During Preventive Maintenance is a Condition Prohibited by Technical Specifications and Outside Design Basis Caused by Personal Error

On November 13, 2000, the licensee submitted LER 2000-010-00 to the NRC indicating that the containment spray (CS) system sodium hydroxide spray additive tank (SAT) had been discovered isolated from service for approximately eight hours and forty minutes during system preventive maintenance on July 21, 2000. This would have prevented automatic injection of sodium hydroxide following initiation of the CS system if a design basis event had occurred. The licensee discovered this condition during an engineering review of recent system maintenance and unavailability for Maintenance Rule accounting. The licensee reported that the root cause of this event was human error which resulted from a mis-interpretation of technical specification requirements when applying a system protective tagout which isolated the SAT with the plant at full power. This was a licensee identified violation of technical specifications (see 4OA7 below) which required the SAT to be operable when the reactor coolant system temperature is above 200F. The inspector considered that the LER properly identified the causes and corrective actions for this event, and met the reporting requirements of 10 CFR 50.73.

b. Findings

No findings of significance were identified.

4OA5 Performance Indicator Data Collecting and Reporting Process Review

a. Inspection Scope (Temporary Instruction 2515/144)

In accordance with Temporary Instruction (TI) 2515/144, "Performance Indicator Data Collecting and Reporting Process Review," the inspector reviewed the data and reporting process for the licensee's performance indicators regarding (1) Mitigating Systems - Emergency AC Power System Unavailability, and (2) Physical Protection - Protected Area Security Equipment. The inspector reviewed licensee procedures, LIC-SD-02, "Performance Indicator Reporting," and PFM-83, " WANO/NEI Self Assessment Monthly Report," during the review. QA surveillance and data tables submitted through the Performance Group were also used to verify the performance indicator data for the third Quarter of 2000 (July, August, September).

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On December 1, 2000, the inspection results were presented to Mr. F. Dacimo and other Entergy staff members who acknowledged the inspection results presented. The inspector asked Entergy personnel whether any materials evaluated during the inspection were considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

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

- On October 27, 2000, the licensee identified a violation of Technical Specification section 3.10.3.1, which states that whenever the quadrant power tilt ratio exceeds 1.02, plant power will be restricted, and the nuclear instrumentation high flux trip setpoint will be readjusted within 2 hours. The licensee did not take these actions within the required time. This event was entered into the corrective action system (DER 00-02781), formally evaluated by the licensee, and will be reported to the NRC in accordance with 10 CFR 50.73 **(NCV 05000286/2000-007-01)**.

- On October 12, 2000, the licensee identified a violation of Technical Specification section 3.3.B, which provides no allowed outage time for the containment spray (CS) system spray additive tank when the plant is above 200F. The tank was isolated for approximately eight hours and forty minutes during CS system maintenance on July 21, 2000. This event was entered into the corrective action system (DER 00-02603), formally investigated by the licensee, and reported to the NRC in LER 2000-010-00 **(NCV 05000286/2000-007-02)**.

PARTIAL LIST OF PERSONS CONTACTED

R. Barrett	Site Vice President
J. Barry	Sr. Radiological Engineer
R. Burroni	I&C Manager
F. Dacimo	Plant Manager
J. Comiotes	General Manager-Operations
J. DeRoy	Director, IP-3 Engineering
R. Deschamps	Radiological and Environmental Services Manager
A. Grosjean	Senior Emergency Preparedness Engineer.
R. LaVera	Sr. Radiological Engineer
J. LePere	General Supervisor, Waste Management
R. Martin	Emergency Preparedness Engineer
D. Mayer	General Manager-Support Services
F. Mitchell	HP General Supervisor
J. Perrotta	Quality Assurance Manager
K. Peters	Licensing Manager
P. Rubin	Operations Manager
J. Russell	General Manager-Maintenance
A. Vitali	Maintenance Manager
J. Wheeler	Training Manager
M. Wilson	Emergency Preparedness Coordinator

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

NCV 05000286/2000-007-01	Nuclear instrumentation high flux trip setpoint not readjusted within 2 hours after the quadrant power tilt ratio exceeded the technical specification limit of 1.02
NCV 05000286/2000-007-02	Containment spray system spray additive tank isolated for 8 hours and forty minutes with the plant at full power.

Closed

URI 05000286/1999-009-01	Use of the Plant Computer to Perform Core Thermal Power Calculations
LER 2000-009-00	Plant Outside Design Basis Due to Valve Seat Leakage During Maintenance That Exceeded the External Recirculation Leakage Limits for the Control Room Ventilation System
LER 2000-010-00	Isolation of the Containment Spray System Spray Additive Tank During Preventive Maintenance is a Condition Prohibited by Technical Specifications and Outside Design Basis Caused by Personal Error

LIST OF ACRONYMS USED

ACTS	Action Commitment Tracking System
AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
CCR	central control room
CFR	Code of Federal Regulations
COL	checkoff list
CRDM	control rod drive mechanism
CS	containment spray
CVCS	chemical and volume control system
DER	Deviation/Event Report
EDG	emergency diesel generator
FRV	feedwater regulating valve
HP	Health Physics
IA	instrument air
ICRI	individual control rod indication
IR	inspection report
IST	in-service test
LER	Licensee Event Report
MBFP	main boiler feed pump
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OA	other activities
OD	operability determination
OD	Operations Directive
ONOP	Off-Normal Operating Procedure
OS	Occupational Radiation Safety
PAB	primary auxiliary building
PAR	publicly available record
PI	performance indicator
PID	Problem Identification [tag]
PORC	plant oversight review committee
PTO	protective tagout
PTRG	Post-Transient Review Group
QA	Quality Assurance
RCA	Radiologically Controlled Area
RCS	reactor coolant system
RES	Radiological and Environmental Services
RFO	refueling outage
RWP	Radiation Work Permit
SAT	spray additive tank
SSCs	structures, systems and components
Tave	average reactor coolant system temperature
TI	Temporary Instruction
TM	temporary modification
TS	Technical Specifications
URI	unresolved item
VC	vapor containment

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	Radiation Safety	Safeguards
● Initiating Events	● Occupational	● Physical Protection
● Mitigating Systems	● Public	
● Barrier Integrity		
● Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margins and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margins but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.