May 14, 2002

Mr. Robert J. Barrett
Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Nuclear Generating Unit 3
295 Broadway, Suite 3
Post Office Box 308
Buchanan, NY 10511-0308

SUBJECT: INDIAN POINT 3 NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-286/02-02

Dear Mr. Barrett:

On March 30, 2002, 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
April 25, 2002, with you and members of your staff.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the inspectors identified one issue of very low safety significance. This issue involved an inadequate risk assessment for planned testing activities.

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

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).

Sincerely,

/RA/

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

Docket No. 50-286
License No. DPR-64

Enclosure: Inspection Report No. 50-286/02-02

Attachment: Supplemental Information

cc w/encl:
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M. Kansler, Senior Vice President and CEO
J. DeRoy, General Manager - Operations
D. Pace, Vice President - Engineering
J. Knubel, Vice President Operations Support
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J. Kelly, Director - Licensing
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-286
License No.	DPR-64
Report No.	50-286/02-02
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point 3 Nuclear Power Plant
Location:	295 Broadway, Suite 3 Buchanan, NY 10511-0308
Dates:	February 17 - March 30, 2002
Inspectors:	P. Drysdale, Senior Resident Inspector L. James, Resident Inspector A. Lohmeier, Senior Reactor Inspector A. Della Greca, Senior Reactor Inspector
Approved by:	Peter W. Eselgroth, Chief Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000286-02-02, on 02/17- 03/30/2002, Entergy Nuclear Northeast, Indian Point 3 Nuclear Power Plant. Resident inspection report.

The inspection was conducted by resident and regional inspectors. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/reactors/operating/oversight.html

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified an inadequate risk assessment that the licensee used to schedule a safety injection (SI) pump surveillance test concurrent with a safety injection actuation logic test. Simultaneous performance of these tests would have reduced the licensee's accident mitigation capability since it would have resulted in the unavailability of two SI trains (1R13).

B. Licensee Identified Violations

None

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Report Details

SUMMARY OF PLANT STATUS

The reactor operated at full power for most of this inspection period. On March 1, 2002, reactor power was reduced to approximately 92% in order to conduct main turbine stop and control valve testing. The reactor was returned to full power the same day.

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

1R04 Equipment Alignment

a. Inspection Scope (71111.04)

- The inspectors verified the availability and correct alignment of the 31 & 32 component cooling water (CCW) pumps while the 33 CCW pump was out of service to replace the inboard and outboard pump seals on March 4, 2002. During a partial system walkdown, the inspectors reviewed protective tagging order (PTO) 02-0108 that the licensee used to isolate the 33 CCW pump from the system. Following the seal replacement and removal of the PTO, the inspectors used check-off list COL-CC-1, "Component Cooling System," to verify the CCW system was restored to its normal operational alignment.

- During the quarterly preventive maintenance (PM) inspection of the 31 emergency diesel generator (EDG) on March 19, 2002, the inspectors performed a partial system walkdown of 32 and 33 EDGs (fuel oil, air start, and ventilation subsystems). The inspectors reviewed PTO 02–0149 during the walkdown, and verified the specified hold tags were properly hung and that the alignment of system components properly isolated the 31 EDG from service. Following the PM and removal of the PTO, the inspectors used COL-EL-5, "Diesel Generators," to verify the 31 EDG was fully restored to service.

- During corrective maintenance (CM) to repair a leaking pump case vent plug on the 32 safety injection (SI) pump on March 21, 2002, the inspectors performed a partial system walkdown of the 31 and 33 SI pumps to verify their proper configuration and alignment for automatic operation. During the walkdown, the inspectors used PTO 02-0144, which designated hold tags on the 32 SI pump, and drawings 9321-F-27353, "Safety Injection System," Sh.1 and 9321-F-27503, "Safety Injection System," Sh.2, to verify it was properly isolated for the repairs.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope (71111.05Q)

The inspectors conducted fire protection tours in the fire zones listed below to evaluate the existence of potential fire hazards and to verify that fire protection equipment was staged appropriately. The inspectors also observed if the licensee 1) controlled transient combustibles in accordance with fire protection procedure FP-9 "Control of Combustibles;" 2) controlled ignition sources in accordance with FP-8, "Controlling of Ignition Sources;" and 3) provided the fire protection equipment specified in the Pre-Fire Plans (PFPs) listed below. In addition, the inspectors assessed the general material condition of the fire protection equipment and fire protection barriers.

- On March 5 - 7, 2002, the inspectors performed a fire protection walk-through of the cable spreading room (Fire Zone 11) in the control building, using PFP-27, "Cable Spreading Room/Battery Rooms - Control Building." The inspection also included the upper cable tunnel (Fire Zone 60A), and the lower cable tunnel (Fire Zone 7A), using PFP-32, "Upper Electrical Tunnel," and PFP-30, "Lower Electrical Tunnel," respectively.

- On March 14, 2002, the inspectors performed a fire protection walk-through of the Condensate Pump area (Fire Zone 41A) using PFP-35, "Condensate Pumps - Turbine Building."

- On March 15, 2002, the inspectors performed a fire protection walk-through of the central control room (CCR) (Fire Zone 15), using PFP-28, "Control Room - Control Building."

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope (71111.11)

On March 11, 2002, the inspectors observed two plant simulator exercises conducted by Operations Crew "B" at the start of an extended three-week course of licensed operator periodic requalification training. Prior to observing the exercises, the inspectors reviewed the plant issues matrix (PIM) for Indian Point 3 Nuclear Power Plant to identify potential weaknesses in operator performance. The inspectors also reviewed the planned scenarios as documented in Lesson Nos. LRQ-SES-15, "Loss of all AC Power;" and LRQ-SES-16, "Uncontrolled Depressurization of all Steam Generators," to determine if they contained 1) clear event descriptions with realistic initial conditions; 2) clear start and end points; 3) clear descriptions of visible plant symptoms for the crew to recognize; and 4) clear expectations of operator actions in response to abnormal conditions. In addition, the inspectors reviewed the Shift Manager's list of specific training objectives for crew performance based on his judgement of areas that needed enhancement.

During the simulator exercises, the inspectors evaluated the crew's performance for the clarity and formality of communications; the correct use and implementation of emergency operating procedures (EOPs) and off-normal operating procedures (ONOPs); the ability to properly interpret and verify alarms, and to take timely control board operation and manipulation; the ability to take timely actions in a safe direction based on transient simulator conditions. The inspectors also evaluated the control room supervisor's ability to exercise effective oversight and control of the crew's actions.

The Operations Manager and the Shift Manager conducted post-scenario critiques with the simulator instructors after each exercise. The Operations Manager determined that the crew passed the simulator examination, but did not meet the requirements for briefs and updates as set forth in operations directive OD-12, "EOP and ONOP User's Guide," and OD-13.1, "Operations Briefs and Updates during EOPs and ONOPs." DER 02-00774 was subsequently written to address this deficiency. Following the critiques, the inspectors discussed the DER and other minor crew performance deficiencies with the Operations Manager.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspectors independently reviewed the licencee's implementation of the Maintenance Rule (MR) for five structures, systems, and components (SSCs), with respect to 1) scoping in accordance with 10CFR50.65; 2) characterizing failed SSCs as a functional failures (FFs), maintenance preventable functional failures (MPFFs), or repetitive maintenance preventable functional failures (RMPFFs); 3) safety significance classifications; 4) the proper 10CFR50.65 (a)(1) or (a)(2) classifications for the SSCs; 5) the appropriateness of performance criteria for SSCs classified as (a)(2) or the appropriateness of goals and corrective actions for SSCs classified as (a)(1).

- Boric Acid Heat Trace (BAHT) system; in MR Status (a)(1) during the fourth quarter of 2001. The licensee monitored the system's performance for reliability (RMPFF) associated with Westinghouse W-2 switches on circuit 53, and developed a replacement schedule for these switches.

- Engineered Safeguards Initiation Logic; in MR status (a)(1) during the fourth Quarter of 2001. The licensee monitored the system's performance for reliability (RMPFF) associated with multiple failures of Westinghouse type W-2 switches, and developed replacement schedule for these switches.

- Instrument Air (IA) System (31, 32, 33 IA Compressors, and 31, 32, 33 IA Dryers); in MR status (a)(2) during the fourth quarter of 2001. The licensee monitored the system in accordance with the MR for unavailability and reliability associated with oil leaks and vibration of the 33 instrument air compressor, and operation of the 31 & 32 IA compressors and dryers.

- Component Cooling Water (CCW) System, in MR status (a)(2) during the fourth quarter of 2001.

- Residual Heat Removal (RHR) System, in MR Status (a)(2) during the fourth quarter of 2001.

The inspectors reviewed the 24 month system unavailability data from the third quarter of 1999 through the fourth quarter of 2001; and also reviewed the licensee's records of unavailability and compared the results with the appropriate Maintenance Rule performance criteria for system operation over the 24 months. In addition, the inspectors reviewed the balance between improvement of availability and reliability in accordance with the performance and condition monitoring requirements of 10CFR50.65 (a)(3). Related DERs over the year 2001 for the systems were reviewed to ensure that significant issues in the maintenance rule scope were not omitted, and that corrective actions were planned to resolve the problems.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control

a. Inspection Scope (71111.13)

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

- On February 26, 2002, the licensee noticed an unidentified substance below the 34 reactor coolant pump (RCP) on the 46 ft. elevation inside containment (DER 02-00601). The following day, the licensee performed a video inspection of the area and discovered water and boron deposits on the floor (DER 02-00642). Upon further investigation, the licensee noted a large accumulation of boron on the packing gland of the reactor coolant system (RCS) 34 loop outboard drain valve (RC-515B) and an active leak of approximately 1 drop per two to three seconds. The licensee's evaluation concluded that both the inboard and outboard 34 RCS loop drain valves (2 inch manual valves RC-515A and RC-515B) had leaked past their seats and pressurized the 34 intermediate loop level indication column, causing it to leak. The leakage was not large enough to be detected by the containment leakage detection systems, and was significantly below the maximum unidentified and identified leakage allowed by the Technical Specifications.

  During March 1 - 22, 2002, the licensee conducted weekly video inspections of these valves to monitor the status of the leak. The inspectors reviewed these tapes and discussed the condition of the valves with operations and engineering personnel. The licensee developed action plan IDSE-APL-02-004 on March 6, 2002, to identify those actions necessary to evaluate the long term effects of the leakage on the loop drain isolation valves, and on the adjacent piping and piping supports. There were no carbon steel parts on the RC-515A & B valves; however, boron was accumulating on adjacent carbon steel pipe supports. The inspectors attended engineering department meetings to review the effects of boric acid on pipe supports. Engineering Memo IP-DEM-002-003, March 15, 2002, addressed these boron deposits, and concluded that the current temperature conditions were not high enough (>100F) to warrant concern before the next outage.

On March 15, the licensee observed that the leakage had increased, and that actions should be planned to stop it. Consequently, action plan SPOS-APL-02-001 was developed to provide a series of progressive actions to address the leakage which included manually tightening the valve stems. On March 22, a nuclear plant operator entered the 46 ft. elevation of containment, and rapidly tightened both of the RC-515A & B valve handles approximately 1/4 turn. The inspectors reviewed a subsequent video inspection, which revealed that his actions stopped the leakage. The licensee planned subsequent video inspections on a monthly basis to confirm that the leakage had not resumed.

- Technical Specification surveillance tests 3PT-Q116A, "31 Safety Injection Pump Functional Test;" and 3PT-M14B, "Safety Injection System Logic Functional Train B." On the morning of March 4, 2002, prior to the performance of these tests, the inspectors noted that the licensee had scheduled them to be performed concurrently. The inspectors questioned the licensee about concurrent performance of these tests with respect to the impact of the planned testing on the mitigation capability of the safety injection (SI) system.

b. Findings

The inspectors reviewed the risk assessment for the week of March 4 - 8, 2002, which included the conduct of tests 3PT-Q116A and 3PT-M14B concurrently. The risk assessment did not identify that their simultaneous performance would have made two SI trains unavailable for automatic operation (i.e., inoperable). This condition would have required operators to enter Technical Specifications paragraph 3.0.3, which requires initiation of a plant shutdown within one hour if two SI pumps are not restored to operability. The inspectors also noted that the risk assessment incorrectly concluded that 1) the scheduled activities would not result in a more restrictive limiting condition for operation (LCO) action statement when compared to the activities being performed separately; 2) the scheduled activities did not result in two or more Maintenance Rule risk significant systems/trains/channels assigned to the high head safety injection system being unavailable concurrently; and 3) that the scheduled activities did not result in the unavailability of two safety injection components (i.e., pumps).

This issue is considered to be more than minor since it had a credible impact on safety in that the risk assessment did not identify that the simultaneous performance of the SI tests would have made two SI trains unavailable for automatic operation (i.e., inoperable). Removing two safety injection pumps from service would reduce the mitigation capabilities of the SI system, and is contrary to Technical Specification 3.5.2

which requires two SI pumps to be operable at all times when the plant is above hot shutdown. This issue was determined to be of very low safety significance (Green) by the Safety Determination Process (SDP) because the licensee did not actually remove two safety injection trains from service and they remained available for automatic injection.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspectors reviewed four Action Commitment Tracking System (ACTS) Items and three Westinghouse Nuclear Safety Advisory Letters (NSALs) on potentially degraded or non-conforming conditions that raised questions on equipment operability. The inspectors reviewed the resulting evaluations for technical adequacy, whether or not continued operability was justified, and to what extent other existing systems adversely impacted the affected system or compensatory actions. The following ACTS Items and NSALs were evaluated:

- ACTS Items 02-51566 and 02-61681: NSAL 02-03, “Steam Generator Mid-deck Plate Pressure Loss Issue,” February 15, 2002

  NSAL 02-03 reported that previously unaccounted for pressure drops across the steam generator (SG) mid-deck plate could potentially impact the low-low level trip setpoint uncertainty calculation as a bias in the indicated high direction. Westinghouse did not maintain the setpoint uncertainty calculation of record for Indian Point 3, and requested that the licensee evaluate any unaccounted for differential pressure across the SGs to determine if there could be an affect on the low-low level trip setpoint calculation. However, Westinghouse indicated that the additional differential pressure (psid) that should be added to the Indian Point 3 setpoint calculation was 0.0 psid. Consequently, the licensee concluded that this NSAL did not impact the existing uncertainty calculation.

- ACTS Item 02-589: NSAL 02-04; “Maximum Reliable Indicated Steam Generator Water Level,” February 19, 2002.

  NSAL 02-04 reported that Westinghouse steam generator (SG) narrow range level instrument uncertainty calculations were potentially in error since they did not reflect the void content of the two phase mixture above the mid-deck plate. Consequently, the SG level high-high trip setpoint could be set at a level above the level assumed in the level safety analysis limit. Since Westinghouse did not maintain the setpoint of record for Indian Point 3, they recommended that the licensee review the void fraction above the mid-deck plate to determine its affect on the SG level uncertainty calculation. The NSAL indicated that Indian Point 3 should use a value of 15% for the level associated with the minimum differential pressure across the level transmitter. This level was subtracted from 100% for the high level trip to be used for the analytic limit (i.e., 85%). However, since 85% was the same value used in the current safety analysis for the high level trip setpoint at Indian Point 3, the licensee concluded that this NSAL had no impact.

- ACTS Item 02-61627: NSAL 02-05, "Steam Generator Water Level Control System Uncertainty Issue," February 20, 2002.

  NSAL 02-05 reported that SG water level control system uncertainties of $\pm$5% or $\pm$7% of span assumed in the SG water level safety analysis may not be bounding. Recent uncertainty calculations based on revised process measurement accuracies have resulted in significant increases in the control system uncertainties for model 54F SGs. Westinghouse did not identify any changes necessary in the specific error values for the Indian Point 3 SGs (model 44F). However, the licensee noted that Indian Point 3 changed the level control system error to $\pm$10% in 1996 to account for a 24 month operating cycle and other factors, and that Westinghouse had used this value in a 1998 safety evaluation. Therefore, the licensee concluded that the NSAL had no impact on the SG water level control uncertainty calculations at Indian Point 3.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope (71111.16)

The inspectors performed a review of operator work-arounds, both transient and non-transient, to determine the cumulative effect 1) upon the reliability, availability, and potential for mis-operation of a system; 2) upon initiating event frequencies; and 3) upon the operator's ability to respond in an appropriate and timely manner to plant transients. This review include the operator work-around list, control room deficiencies list, Central Control Room turnover sheets, and system operating procedure SOP-SD-01, "Work Control Process." In addition, the inspectors reviewed the work control and DER programs to assess the open problem identification tags (PIDs), work requests (WRs) and DERs for potential operator work-around consideration.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modification

a. Inspection Scope (71111.17)

The inspectors reviewed Design Change Package (DCP) 00-03-018, "Replacement of Station Battery 31 and Station Battery 32," to verify that the design and licensing bases, and system performance capabilities associated with the batteries would not be degraded through the modification. DCP 00-03-018 included removal of the old 32 battery cells, modification of the existing 32 battery racks to accommodate the different dimensions of the new 32 batteries, installation of the new 32 battery cells, and replacement of the 32 battery charger fuse clips. In order to replace the 32 batteries with the plant at power, the old 31 batteries were installed as the temporary 32 battery during the changeout. This supplied continuous DC power to the 32 battery loads when the old 32 battery was removed.

During February 17 - 21, 2002, the inspectors reviewed the licensee's documented lessons learned from the 31 battery replacement, observed portions of the 32 battery replacement, and reviewed post-modification test data of both the temporary and new 32 batteries (WR 94-01648-26).

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope (71111.19)

The inspectors 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 following PMT activities were evaluated:

- WR 01-04788-00: PMT on the 34 containment fan cooler unit (FCU) following PM inspection and flow switch replacement; February 26, 2002.

  WR 01-01410-00: PMT following replacement of four FCU ventilation damper air cylinders.

- WR 02-00509-00: PMT following seal replacements on the 33 CCW pump; March 5, 2002. The test instructions were incorporated into the WR for the seal replacement, and involved an inspection for zero leakage at all system joints at system pressure for 10 minutes. The licensee performed surveillance test 3PT-Q088, "Component Cooling Pumps Functional Test," to accomplish the PMT.

- WR 01-04043-02: PMT using surveillance 3PT-M79A, "31 EDG Functional Test," following a quarterly PM on the 31 EDG; March 20, 2002. The work request also prescribed an inservice leak test on the fuel injection covers, the crankcase covers, and the jacket water and lube oil heat exchangers.

  The inspectors reviewed completed maintenance procedure GNR-004-ELC, "Emergency Diesel Generator Quarterly Inspection." The PM involved inspections of several EDG subsystems, including fuel injection, fuel oil storage and supply, air intake and exhaust, lube oil, engine cooling water, and air start. Inspection of the engine crankcase and heat exchangers were also performed.

WR 02-00955-02: PMT for inservice leak test of the 31 EDG air start motors associated with all inlet air louvers. The PMT also tested the operation of all ventilation exhaust fans, and inlet and outlet louvers.

WR 01-03440-01: PMT on the 31 EDG air start supply strainer.

- WR 01-03420-03; PMT following corrective maintenance to repair leakage on the casing vent plug of the 32 safety injection pump; March 22, 2002.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspectors observed portions of the following surveillance tests and reviewed the surveillance test procedures to assess whether 1) the test preconditioned the component(s), 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 was adequate with proper calibration, 5) the test was performed in the proper sequence, and 6) the test equipment was removed following testing.

- 3PT-Q132, "Emergency Boration Flow Path Valve CH-MOV-333;" March 11, 2002.
- 3PT-Q016, EDG and Containment Temperature Service Water Control Valves, SWN-1176, 1176A, & TCV-1104 and TCV-1105;" February 26 - 27, 2002.
- 3PT-Q120B; "32 Auxiliary Boiler Feedwater Pump Functional;" March 14, 2002.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23A)

The inspectors reviewed the temporary modification package TM 97-05176-02, which provided control circuit power and channel separation to the 33 auxiliary boiler feedwater pump (ABFP). The temporary modification was installed to separate the common power source for each motor-driven ABFP controller circuit so that a single failure of instrument bus 33 would not result in a loss of runout protection for both motor-driven ABFPs. The inspectors verified that this temporary modification did not adversely affect the safety function of the auxiliary feedwater system. The inspectors reviewed the temporary modifications and associated nuclear safety evaluation against the system design bases documents, including the Final Safety Analysis Report and the Improved Technical Specifications.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope (71114.06)

As part of implementation of Inspection Procedure 71114.06, “Drill Evaluation,” for Indian Point 2 Nuclear Power Plant (documented in NRC Inspection Report 50-247/02-02), the inspectors observed the licensee’s accountability drill on March 8, 2002, and initiated an Unresolved Item (URI) associated with the licensee’s failure to complete site-wide (Units 2 and 3) full personnel accountability within 30 minutes. In addition, the inspectors reviewed the revised procedures for both Indian Point 2 and 3 to ensure that the changes made did not reduce the effectiveness of the Emergency Plan.

b Findings

During a review of the Indian Point 3 Emergency Plan Implementing Procedures (EPIPs), the inspectors identified a change in EPIP 1050, “Accountability,” that appeared to reduce the effectiveness of the Emergency Plan (DER 02-00962). The March 6, 2002 revision of EPIP 1050 did not require the dispatch of a security guard to the onsite assembly area during off normal hours to maintain the accountability of individuals evacuated from the protected area. Without an individual in the onsite assembly area to be the point of contact with the Lead Accountability Officer, the licensee reduced its ability to maintain accountability of the individuals evacuated to the assembly area and to evacuate those individuals from the site. In accordance with 10 CFR 50.54q, the licensee must obtain prior NRC approval before decreasing the effectiveness of the Emergency Plan. This issue is unresolved pending the resolution of DER 02-00962 and NRC review of the follow-up accountability drill scheduled for April 8, 2002. **(URI 05000286/02-02-01)**

4. OTHER ACTIVITIES (OA)

40A1 Performance Indicator Verification

a. Inspection Scope (71151)

The inspectors reviewed the licensee’s data for the following performance indicators (PIs) reported to the NRC for the third and fourth quarters of 2001 against the applicable criteria specified in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 1. The review included all of the raw data compiled for each month of the year 2001, and verification that the submitted data was accurate.

- Unplanned scrams per 7000 critical hours.

- Scrams with loss of normal heat removal.

- Unplanned power changes per 7000 critical hours.

- Safety system unavailability (SSU):

  1) Emergency AC Power System
  2) High Head Safety Injection (HHSI) System
  3) Auxiliary Feedwater (AFW) System
  4) Residual Heat Removal (RHR) System

For the Emergency AC Power and AFW systems, the inspector also reviewed the NRC PI Data Submittals for the third and fourth quarters of 2001, and compared the measured unavailability with the 2001 data collection reported in the 2001 with the World Association of Nuclear Operators (WANO) data submittals.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

a. Inspection Scope (71153)

(Closed) Licensee Event Report (LER) 1998-001-01: Potential Failure or Inadvertent Operation of Fire Protection Systems, Caused by Personnel Error in Design, Could Cause a Loss of Cable Spreading Room Cooling Placing the Plant Outside Design Basis

The licensee submitted this LER Supplement to the NRC on March 19, 2002, and the inspectors performed an in-office review to assess its accuracy and completeness. The supplement was an update to the original LER (1998-001-00) issued in March 1998 which indicated that the analysis of the root cause was still under evaluation, and that corrective actions to address the event were still in progress. Based upon the inspectors review, the root cause analysis was completed and corrective actions were entered into the licensee's corrective action system. The stated corrective required installation of a plant modification and those actions appeared to be adequate. This item is closed.

(Closed) Unresolved Item (URI) 05000286/19990901: Degraded Grid Voltage Relays Setting

The inspectors determined that no violation of NRC requirement occurred in conjunction with the setting of the degraded grid voltage relays.

During a review of the Indian Point 3 voltage study, NRC inspectors observed that the licensee had not calculated the voltage drop from the motor control center buses to the motor starters. The inspectors questioned what assurance the licensee had that under degraded grid voltage conditions, the voltage at the starters would be sufficiently high to start and accelerate the safety-related motors.

The inspectors also observed that the licensee had not performed an accuracy calculation of the voltage sensing circuit. Therefore, the inspectors questioned how the relay setting adequately met the Technical Specification (TS) requirements regarding minimum bus voltage (414V). In response to NRC questions, the licensee re-evaluated the results of their degraded grid voltage study, IP3-CALC-EL-01972, Rev. 1, and concluded that the TS process limit of 414V as a minimum voltage was acceptable. Also, in another calculation, IP3-CALC-ESS-03154, Rev. 0, the licensee determined that the current relay setting of 422V was acceptable and that the minimum bus voltage of 414V stated in the TS was reasonable. The inspectors reviewed the bases of the above calculations and concluded that the results were reasonable. This item is closed.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Annual Assessment Meeting Summary

On April 11, 2002, the NRC met with the licensee in a public forum, and presented the results of the NRC's annual assessment of Entergy's performance for the period of April 1 through December 31, 2001. The licensee responded to the NRC's findings and highlighted those areas where improvements were planned during the next assessment period.

Exit Meeting Summary

On April 25, 2002, the inspectors presented the inspection results to Mr. R. Barrett and other Entergy staff members who acknowledged the inspection results presented. The inspectors verified with Entergy personnel that no materials evaluated during the inspection were considered proprietary.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

R. Barrett	Vice President, Operations - IP3
R. Burroni	I&C Manager
R. Cavalieri	Site Planning and Outage Services Manager
J. Comiotes	Director, Nuclear Safety Assurance
J. DeRoy	General Manager of Plant Operations
J. Donnelly	Licensing Manager
M. Gillman	Operations Manager
F. Inzirillo	Emergency Planning Manager
J. Perrotta	Quality Assurance Manager
K. Peters	Corrective Actions and Assessment Manager
M. Smith	Director, IP-3 Engineering
A. Vitale	Maintenance Manager
C. Welling	Radiation Protection and Chemistry Manager

b. List of Items Opened, Closed, and Discussed

Opened

05000286/02-02-01	URI	Potential failure to obtain NRC approval prior to reducing the effectiveness of the Emergency Plan.

Closed

1998-001-01	LER	Potential Failure or Inadvertent Operation of Fire Protection Systems, Caused by Personnel Error in Design, Could Cause a Loss of Cable Spreading Room Cooling Placing the Plant Outside Design Basis
05000286/99-09-01	URI	Degraded grid voltage relay settings

c. <u>List of Acronyms</u>

ABFP	auxiliary boiler feedwater pump
AC	alternating current
ACTS	Action Commitment Tracking System
AFW	auxiliary feedwater
BAHT	boric acid heat trace
CCR	central control room
CCW	component cooling water
CFR	Code of Federal Regulations
COL	Check-Off List
DCP	design change package
DER	Deviation/Event Report
EDG	emergency diesel generator
EOP	emergency operating procedure
EPIP	Emergency Plan Implementing Procedure
FCU	fan cooler unit
FP	Fire Protection
FSAR	Final Safety Analysis Report
FF	functional failure
HHSI	high head safety injection
IA	instrument air
IP	inspection procedure
IP3	Indian Point 3
IR	inspection report
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MPFF	maintenance preventable functional failure
MR	Maintenance Rule
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSAL	Nuclear Safety Advisory Letter
OD	Operability Determination
OD	Operations Directive
ONOP	Off-Normal Operating Procedure
PFP	Pre-Fire Plan
PI	performance indicator
PID	problem identification tag
PIM	Plant Issues Matrix
PMT	post-maintenance test
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RMPFF	repetitive maintenance preventable functional failure
SDP	Significance Determination Process
SG	steam generator
SI	safety injection
SOP	system operating procedure
SSCs	Structures, Systems, and Components

SSU	Safety System Unavailability
SW	service water
TM	temporary modification
TS	Technical Specifications
URI	Unresolved Item
WANO	World Association of Nuclear Operators
WR	Work Request