June 22, 2001

Mr. Robert J. Barrett
Vice President, Operations-IP3
Entergy Nuclear Northeast
Indian Point 3 Nuclear Power Plant
Post Office Box 308
Buchanan, NY 10511

SUBJECT: NRC'S INDIAN POINT 3 INSPECTION REPORT NO. 50-286/01-04

Dear Mr. Barrett:

On May 19, 2001, 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 May 31, 2001, 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.

Based on the results of this inspection, the inspectors identified one finding of very low safety significance (Green) regarding multiple degraded conditions that resulted in a minor secondary plant transient.

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/

Curtis J. Cowgill, Chief
Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report No. 50-286/01-04
Attachment: Supplemental Information
cc w/encl:

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M. Kansler, Chief Operating Officer
J. Knubel, Vice President Operations Support
F. Dacimo, General Manager, Plant Operations
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REGION I

Docket No.	50-286
License No.	DPR-64
Report No.	50-286/01-04
Licensee:	Entergy Nuclear Northeast
Facility:	Indian Point 3 Nuclear Power Plant
Location:	P.O. Box 308 Buchanan, New York 10511
Dates:	April 1 - May 19, 2001
Inspectors:	Peter Drysdale, Senior Resident Inspector Lois James, Resident Inspector John McFadden, Radiation Specialist
Approved by:	Curtis J. Cowgill, Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000286-01-04; on 04/01/01 - 05/19/01; Entergy Nuclear Northeast; Indian Point 3 Nuclear Power Plant. Initiating Events.

The inspection was conducted by resident and regional inspectors. The inspection identified one green issue. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violation. 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/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- Green: The inspectors identified that the existence of concurrent multiple degraded conditions in two service water valves in the turbine main lube oil system increased the frequency of an initiating event (transient - potential turbine trip) and resulted in a challenge to plant operators that required operator action to reduce reactor power below 100%.

  This event was determined to be of very low significance (Green) using the NRC’s significance determination process because it did not contribute to the likelihood of a loss of coolant accident initiator, did not contribute to the likelihood that mitigation equipment or functions will not be available, and did not increase the likelihood of a fire or internal/external flood. (Section 1R13)

B. Licensee Identified Violations

There were no licensee identified violations during this inspection.

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Report Details

SUMMARY OF PLANT STATUS

At the beginning of the inspection period on April 1, 2001, the Indian Point 3 plant was operating at 100% power. On April 24, 2001, the licensee commenced a turbine load reduction in accordance with a planned power coastdown, and on April 27, initiated a plant shutdown from approximately 97% power. At approximately 30% power, the licensee tripped the reactor and initiated a plant cooldown in preparation for refueling outage number eleven (RO-11). The plant remained shutdown in a refueling outage for the remainder of the inspection period.

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

1R04 Equipment Alignment

a. Inspection Scope (71111.04)

On April 17, 2001, the inspectors performed a partial walkdown of accessible portions of the 32 containment spray (CS) train to verify equipment alignment, and to identify any discrepancies that could impact the function or availability of this mitigating system that is important to safety. During this inspection, the 31 CS pump was out of service for planned system preventive maintenance in preparation for the refueling outage. The inspectors completed check-off list COL-CS-1, "Containment Spray System," to verify component position and utilized standard operating procedure SOP-CS-01, "Containment Spray Operation," for system operation knowledge.

On May 16, 2001, the inspectors performed a walkdown of accessible portions of the fuel oil, air-start, ventilation, and electrical subsystems of the 31 and 33 emergency diesel generators (EDGs) to assure that these subsystems were available to support EDG operability. During the walkdown, the 32 EDG was out of service for its eight-year preventive maintenance (PM) inspection. The inspectors utilized the applicable portions of check-off list COL-EL-5, "Diesel Generators," to complete this inspection.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope (71111.05Q)

The inspectors conducted fire protection tours of several plant areas to assess the licensee's capability to protect plant areas and important safety equipment from potential damage from fires. These areas contained vital equipment important for mitigating the consequences of events, and support equipment that is needed to operate other equipment important to safety. The inspectors' tours evaluated barriers used to prevent fires from starting (transient combustibles and ignition sources) and to rapidly detect, control, and extinguish fires that do occur (manual, automatic, and passive suppression and detection systems) against the licensee's procedures and National Fire Protection Association requirements.

On April 1 & 5, 2001, the inspectors toured all three EDG cubicles and all areas in the 480 volt switchgear room. These areas contain vital equipment necessary for safe plant operation, and for supplying vital electrical power to equipment needed to mitigate the consequences of potential design basis events.

On April 9, 2001, the inspectors toured the cable spreading room and electrical tunnels (CSR/ETs). The CSR/ETs contain equipment important to safety, including static inverters and battery chargers, and numerous cables that are needed to operate equipment important to safety. Minor concerns regarding potential transient combustibles were referred to operations management and documented in deviation/event report (DER) 01-01254.

On April 10, 2001, the inspectors toured all areas of the primary auxiliary building (PAB) and the fuel storage building (FSB). These areas contain vital plant equipment necessary to prevent and mitigate the consequences of potential design basis events, and other equipment important to plant safety. Minor discrepancies were brought to the attention of the fire protection department for evaluation and resolution.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspectors reviewed problems involving selected in-scope structures, systems, and components (SSCs) related to the service water system to assess the effectiveness of the maintenance program. 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 service water SSCs classified as (a)(2). The following deficiency was also reviewed:

- During a routine service water pump (SWP) swap on April 16, 2001, and when starting the 35 SWP, the pump exhibited high vibrations, and a burning odor was detected at its 480 volt supply breaker. The breaker's C-phase contactor showed evidence of arcing, and indicated that the contactor had not operated correctly. The licensee concluded that the arcing was caused by a loose locknut holding the contactor in place, and that misalignment of the C-contactor caused the pump vibrations. The inspectors reviewed the maintenance history of the SWPs by examining operability determination OD-01-05; the licensee's response to DER 01-01383 (excessive vibrations on 35 service water pump); and previous DERs on the SWPs. The inspectors also reviewed the Technical Specifications (TS) and the Final Safety Analysis Report (FSAR) to determine design basis functional requirements for the SWPs. The inspectors calculated the maintenance rule unavailability time associated with the SWPs using plant logs that documented entries into limiting conditions for operation (LCOs).

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work

a. Inspection Scope (71111.13)

The inspectors reviewed the maintenance risk assessments and corrective maintenance work packages for the following planned and emergent work. The inspectors also discussed deficient conditions with cognizant engineering and maintenance personnel.

- WR 00-01704-00; Calibration procedure 3PC-R49A, "Steam Generator Blowdown Radiation Monitor Calibration (R-19)." On April 3, 2001, the licensee performed a scheduled calibration of the steam generator blowdown (SGBD) radiation monitor, which placed a block (SGBD valves in "Rad Bypass") into the containment isolation signals to all SGBD isolation valves that would normally close on a high radiation signal from R-19. The calibration was put on hold after the Rad Bypass block was inserted when operations and engineering questioned (DERs 01-01184 and 01-01191) the performance of this calibration with the isolation valves unable to close on isolation signals from a high temperature in the SGBD heat exchanger room, a low-low steam generator level, a loss of normal feedwater, an undervoltage condition on a 480 volt bus (loss of offsite power), or high radiation from R-19. In the event that one of the motor-driven auxiliary feedwater (AFW) pumps were unavailable at the time, the remaining AFW pump could not deliver sufficient flow to provide full decay heat removal with all SGBD valves open.

  The inspectors noted at the time the R-19 calibration was performed, that no AFW pumps were out of service; however, the licensee suspended all 3PC-R49A activities and removed the block from the isolation signals to the blowdown valves. The licensee's work control organization also took measures to ensure that no activities would be scheduled during the R-19 calibration that could affect AFW pump operability. A temporary change was made to the 3PC-R49A procedure to maintain all isolation signal capability to the blowdown valves except from R-19 during its calibration. The inspector reviewed the procedure change after the R-19 calibration resumed, and discussed the temporary equipment configuration and SGBD valve operability with control room operators. The licensee considered that there may have been previous instances when an AFW pump was inoperable during the R-19 calibration, and initiated an action commitment tracking item (ACT-01-56767) to evaluate the potential reportability associated with those instances. The licensee also intended to evaluate the impact on plant risk with an AFW pump out of service while the SGBD valves were in the Rad Bypass mode.
- On April 16, 2001, the licensee determined that the 480 volt supply breaker for the 35 SWP did not function properly and caused excessive vibrations in the pump while it was operating (DER 01-01383). The licensee declared the 35 SWP inoperable and replaced the supply breaker with a spare. The pump was

subsequently retested and returned to service. The inspectors reviewed the results of the licensee's risk assessment for unavailability of the SWP to determine if the incremental conditional core damage probability (ICCDP) was significant and warranted further NRC review. The inspectors further verified that the licensee followed the applicable procedures and appropriately utilized their available risk assessment tools.

- While performing refueling outage preparations on April 18, 2001, a plant worker broke off the MLO cooler service water drain valve SWT-168 and its associated pipe from the cooler, which caused a service water leak (DER 01-01399). Due to a previously identified degraded condition (internal leak) in the 32 MLO cooler isolation valve SWT-4-2 (PID 49001; June 25, 2000), the MLO cooler and service water leak could not be isolated. In addition, temperature control valve SWT-TCV-1102 was in a previously identified degraded condition (binding; PID 49739; June 26, 2000) which prevented the MLO system from properly controlling the temperature and pressure of the oil supplied to the main turbine control system. As a result there was a small uncontrolled perturbation of the first stage turbine pressure and an associated perturbation of reactor power that required operator action.

b. Findings

The existence of multiple degraded conditions (internal leak in valve SWT-4-2 and binding in valve SWT-TCV-1102) had an actual impact on plant safety, and caused a small uncontrolled reactivity change that resulted in unexpected control rod motion. Drain valve SWT-168 and its pipe had been broken off on two previous occasions, but the plant did not experience a transient because the temperature and pressure controls in the MLO system responded normally to maintain a stable turbine inlet pressure (WR 97-04671-81, WR 97-07177-00, WR 99-03335-00). In the current event, the existence of concurrent multiple degraded conditions increased the frequency of an initiating event (transient - potential turbine trip) and resulted in a challenge to plant operators (i.e., required operator action to reduce reactor below 100%).

This event was determined to be of very low significance (Green) using the NRC's significance determination process because it did not contribute to the likelihood of a loss of coolant accident initiator, did not contribute to the likelihood that mitigation equipment or functions will not be available, and did not increase the likelihood of a fire or internal/external flood. **(FIN 50-286/01-04-01)**

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspectors reviewed DERs on degraded or non-conforming conditions that raised questions on equipment operability. The inspectors reviewed the resulting operability determinations (ODs), if applicable, 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 inspectors evaluated the following DERs, ODs, and calculations:

- On April 20, 2001, the 33 EDG east air-start motor did not stop as expected after the engine started due to a sheared fitting and air regulator problems (DER 01-01440). Each EDG has two redundant air-start motors; however, the surveillance test (3PT-M79C) for EDG operability normally started the engine using both the east and west air motors, and did not demonstrate EDG operability with only one air-start motor. A requirement to demonstrate EDG operability with one air-start motor is contained in the Technical Specification Bases, which is only necessary when an air-start motor is not capable of performing its function. Consequently, when the east air-start motor for the 33 EDG was declared inoperable, the licensee conducted a timed surveillance to demonstrate EDG operability using only the west air-start motor.

- On May 16, 2001, large swings in kilo volt-amps reactance (KVARS) and alternating current voltage (VAC) occurred during the 31 EDG eight-hour endurance run (3PT-R160A). The EDG experienced fluctuations in KVARS on the order of 500 KVARs, and VAC oscillations from 498 - 510 VAC (DER 01-02128). The licensee aborted the test and conducted troubleshooting on the voltage regulator and its associated control circuits. Static electrical tests on the regulator indicated that it was not defective, but high contact resistance was detected in the rheostat used for the generator's automatic voltage adjustment.

  After cleaning to restore normal contact resistance in the rheostat, the licensee performed the normal monthly EDG surveillance test (3PT-M79A) at the same generator power rating . During the test, the output voltage and KVARS remained within specification and stable. Based on these test results, the licensee generated OD 01-016, indicating that the stable voltage and KVARS confirmed EDG operability. The OD also indicated the voltage and KVAR fluctuations during the endurance run did not make the EDG inoperable. The inspectors discussed these conditions with electrical engineering personnel who indicated that the oscillations were experienced during EDG operation in the "parallel" mode and were indicative of grid conditions while the EDG was connected to, and subjected to, the plant's main output transmission circuit. When operated in the "unit" mode (feeding only a safeguards bus in the plant) the EDG would not have experiences these fluctuations.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope (71111.11)

The purpose of this review was to verify that modifications performed during increased risk-significant configurations did not place the plant in an unsafe condition, and to verify that the design bases, licensing bases, and performance capability of risk significant SSCs had not been degraded through modifications

- The inspectors reviewed the Reasonable Assurance of Safety (RAS) 01-3-028, “Indian Point 3 Central Control Room Air Conditioning Damper A and B Replacement During RO-11," and observed field work to replace the dampers on April 30, 2001 (WR 99-03633-01). The purpose of the RAS was to ensure that the licensee met NRC Safety Evaluation Report statements regarding the ability of sensing equipment at IP3 to detect a loss-of-coolant accident at IP2.

- The inspectors reviewed design change package DCP-00-3-069-CVCS, Rev.1, “Replacement of RCP No 1 seal flow transmitters FIT-156A, FIT-157A, & FIT-158A & Power supplies FQ-156A, FQ-157A, & FQ-158A,” and the nuclear safety evaluation written to support the modification. The design change was necessary to replace unreliable flow meters/transmitters for the 31 reactor coolant pump (RCP) with a different model and manufacturer. During the RO-11 outage, the licensee used the modification package to replace the flow meter/transmitters on all four RCPs. The inspectors observed portions of WRs 00-3283-08 and 00-3283-13 for installation of the transmitter power supplies in the control room, reviewed the post-installation testing requirements, and performed a post-installation walkdown of the meter/transmitters inside containment.

- The inspectors reviewed DCP-00-3-005-FW, “Install Auto-Closure Feature for Main Feedwater Motor-Operated Valves BFD-5's and BFD-90's,” and its associated nuclear safety evaluation (NSE). The design change installed isolation signals to the feedwater regulating valve (FRV) and bypass valve block valves (BFD-5-1 through BFD-5-4 and BFD-90-1 through BFD-90-4). These signals would cause the valves to close and isolate main feedwater following a main steam line break (MSLB) inside containment. The modification was necessary to correct a non-conservative assumption in the MSLB analysis for IP3. The analysis underestimated the actual unisolable volume of feedwater piping following a MSLB with the failure of one FRV to close on a feedwater isolation signal. The BFD-5 and BFD-90 series valves did not previously close on a feedwater isolation signal, and their closure will assure that the unisolable volume is within accident limits. The isolation signal was the same feedwater isolation signal which affected auto-closure of the FRVs and bypass valves.

The DCP also specified modifications necessary to upgrade the motors and actuator gear sets for all eight of the motor-operated valves (MOVs) to provide higher torque and thrust capabilities for MSLB conditions, and to meet the standards of the licensee's Generic Letter 89-10, "Motor-Operated Valves," program. The inspectors reviewed calculations IP3-CALC-FW-03296 through -03299 (BFD-5 series) and IP3-CALC-FW-03307 through -03310 (BFD-90 series) which provided detailed engineering analyses for determining the minimum torque and thrust necessary for each MOV under the 89-10 program requirements. The inspectors also reviewed WR 99-05022-00, which installed the modifications and directed static testing of all eight MOVs.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope (71111.19)

The inspectors reviewed post-maintenance test procedures for the 31 EDG 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 post-maintenance test activities were observed and evaluated:

- 3 PT-R160A, "31 EDG Capacity Test," performed on May 15 and 18, 2001
- 3PT-M079A, "31 EDG Functional Test," performed on May 16, 2001

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope (71111.20)

The inspectors evaluated licensee outage activities to ensure that licensee considered risk in developing outage schedules; adhered to administrative risk reduction methodologies developed to control plant configuration; had developed mitigation strategies to losses of key safety functions, and adhered to their operating license and Technical Specifications. The inspectors focused primarily on shutdown activities were risk could be high if vital systems, structures, and components were not be available as a result of maintenance activities.

Outage Plan and Risk Assessment

Prior to the plant shutdown, the inspectors reviewed the detailed refueling outage schedule and work activities, and administrative procedure AP-9.2, “Outage Risk Assessment,” in order to evaluate the licensee’s guidance used to assess the plant risk represented by the proposed outage schedule, and the criteria used to assure that scheduled activities conform to safe shutdown provisions designed to ensure that reactor core integrity would not be compromised during the outage. The inspectors also reviewed the licensee’s “Risk Assessment for the RO-11 Outage Schedule,” that was produced by an independent risk assessment team of individuals involved in the development of the outage schedule.

The initial assessment was completed in February 2001 based on the proposed outage schedule, and was subsequently revised in April 2001 after the team made 26 formal recommendations for schedule changes to provide for additional defense-in-depth measures that would maintain specific system and plant configurations below defined risk thresholds. The inspectors noted that the assessment adhered to administrative risk reduction methodologies outlined in AP-9.2, and incorporated site specific lessons from previous outages. The final assessment report contained daily risk profiles that ranked each of five safety functions (reactivity, core cooling, power availability, containment integrity, and RCS inventory) according to a color hierarchy (green, yellow, red). During the outage, the licensee implemented defense-in-depth contingency procedures for each safety function that placed specific restrictions on outage operations to ensure that the overall daily risk did not exceed the yellow criteria. The inspectors periodically monitored the daily risk profiles to ensure that they matched the existing plant equipment configuration.

<u>Reactor Shutdown</u>

On April 27, 2001, the inspectors observed the reactor shutdown from the central control room (CCR) and verified that the technical specification cooldown rate limits were satisfied. The operators used plant operating procedure POP-3.1, “Plant Shutdown from 45% Power” to lower turbine load to approximately 250 megawatts, at which time the operators tripped the reactor and entered emergency procedures E-0, “Reactor Trip or Safety Injection” and ES-0.1, “Reactor Trip Response.” The inspectors noted that plant operators remained attentive and alert to changing plant conditions, and that plant equipment functioned as anticipated.

<u>Establish and Monitor Residual Heat Removal (RHR) Cooling</u>

After the primary plant entered mode 3 operations, the licensee placed the 32 RHR pump into service to continue the plant cooldown. The inspectors reviewed system operating procedure SOP-RHR-1, “Residual Heat Removal System,” and observed RHR system parameters to verify the system was functioning properly, and that operators maintained RHR flow to the reactor core at $\geq$1000 gpm with fuel in the vessel.

Clearances

The inspectors verified that tags and clearances were properly installed and removed and that equipment was appropriately configured to support the function of the clearance. Proper use of clearances and tagging ensured that maintenance activities were conducted in a safe environment and that inventory in controlled systems are not lost. The inspectors observed the removal of one protective tag out (PTO) and one caution tag out (CTO) to ensure that the licensee followed administrative procedure AP-10.1, “Protective Tagging.” The inspectors also observed the work control center and the performance of the field support supervisor who approves tag outs, dispatches personnel to install or remove the tags, and provides final disposition of removed tags. Further, the inspectors reviewed plant configurations caused by the clearance tags to ensure that the licensee maintained the minimum required boron injection pathway.

Outage Configuration Management

During the outage, the inspectors reviewed the daily plant configuration risk assessments, which listed the equipment out of service and the contingency procedures in effect to assure that critical safety functions would be maintained. The inspectors also performed control room and plant system walkdowns on a sampling basis to verify that available equipment was maintained operable in accordance with technical specification requirements. The inspectors held regular discussions with control room operators during changing plant conditions to assess their cognizance of the impact on important safety functions.

The inspectors noted several other system “status control” events that were discovered and documented by the licensee during the outage. The inspectors reviewed these events with operations personnel and verified that the licensee properly entered them into the corrective action program.

Reactor Coolant System (RCS) Instrumentation

During the refueling outage, the normal RCS pressure, and temperature instruments in the control room were used by operators. The RCS level instruments used by operators during the outage were temporary devices installed and configured to provide redundant level indication. The licensee tested and installed the Mansell Level Monitoring System (MLMS), which utilized solid state pressure transmitters, and provided accurate level indication during changing pressure and temperature conditions in the RCS. In addition, the licensee installed an Intermediate Leg Level Indication System (ILLIS) that utilized two separate water columns visible to the operators through a remote television monitor in the control room. During the special evolution to vacuum fill the RCS, the licensee installed an Ultrasonic Level Monitoring System that was installed on an RCS hot leg to ensure that redundant level monitoring was available when the ILLIS was not reliable (i.e., during RCS vacuum fill). The inspectors monitored the status of these systems throughout the outage to assure that the level indication provided to operators remained available and accurate.

Inventory Control

The inspectors verified during periodic control room and plant walkdowns that the makeup sources and flow paths for adding RCS inventory were maintained in accordance with the outage risk assessment. The licensee implemented a special outage system for RCS boundary valve controls to assure that control room authorization was provided prior to changing the configuration of those valves. The inspectors verified that there were adequate controls in place to prevent inventory loss during outage activities that had a potential for RCS losing inventory.

Reduced Inventory; Mid-Loop Operations

The inspectors reviewed plant operating procedures POP-4.1, “Operation at Cold Shutdown,” and POP-4.2, “Operation Below 10% Przr (sic) Level with Fuel in the Reactor,” and system operating procedure SOP-RP-020, “Draining the RCS/Refueling Cavity,” in order to evaluate operator actions to maintain the appropriate RCS levels and inventory for preplanned evolutions. Operators also utilized system operating procedure SOP-RCS-017, “Reactor Vessel Vacuum Refill and Mansell Level Monitoring System Operation,” for special evolutions that employed the MLMS system.

The inspectors also reviewed the licensee’s commitments to Generic Letter 88-17, evaluated the functionality of the redundant level monitoring instruments during reduced inventory operations. The licensee reduced the RCS level below the reactor vessel flange on three occasions during the outage in order to install the refueling cavity seal, to complete a 31 RCP seal replacement, and to prepare for vacuum fill of the RCS at the end of the outage. During the reduced inventory and mid-loop conditions, the inspectors reviewed the configuration of plant systems in accordance with those commitments. During mid-loop operations, the licensee imposed special evolution controls on plant operator activities and placed restrictions on plant personnel access into the control room in order to minimize distractions and the operators’ ability to maintain required reactor vessel level. The inspectors verified on a sampling basis that the minimum reactor cavity level for fuel movement was maintained above 92' 1-1/2" (i.e., ≥23 feet above the top of the reactor vessel flange) in accordance with the Technical Specifications

Electrical Power

The inspectors reviewed the planned outage schedule for removing the 138 kilovolt (KV) offsite power circuits for maintenance. This activity occurred over approximately one week when the plant was in mode 6, and the reactor core was offloaded. During this time, the 13.8KV sources were available and at least two EDGs were operable. The inspectors periodically observed and verified during plant tours that the licensee maintained an enhanced level of protection for electrical power supplies to safety-related equipment in accordance with technical specification requirements.

Reactivity Control

During plant shutdown operations, the inspectors reviewed plant operation procedures, observed operator actions in the control room, and interviewed reactor and senior reactor operators to verify that the licensee was controlling reactivity in accordance with their Technical Specifications and Technical Requirements Manual. Procedures reviewed included:

- Plant Operating Procedure POP-2.3, “Core Operating Limits for Cycle 11"
- POP-4.3, “Operations without Fuel in the Reactor”
- SOP-CVCS-003, “Reactor Coolant System Boron Concentration Control”
- TOP-181, “Aligning Refueling Cavity Transfer Pump for Cavity Cleanup”

Spent Fuel Pool Cooling

The licensee completely off-loaded the reactor core into the spent fuel pool during outage RO-11. Both the primary and backup cooling systems were placed in service prior to and during the core offload. During the fuel transfer period, the inspectors observed operations in the pool and the fuel transfer canal, and monitored pool temperatures. During the core off-load, the pool temperature increased from approximately 78F to 120F. On May 7, 2001, the licensee removed the primary cooling system from service to perform maintenance, and kept the backup system in service to maintain pool cooling. The pool temperature increased to approximately 150F and remained stable with only the backup system in service.

On May 8, 2001, make-up water to the backup cooling system was lost due to a power failure in a temporary trailer that supplied the makeup water. This caused an interruption in cooling to the fuel pool, and pool temperature increased to approximately 155F before the licensee discovered the problem. Once make-up water was restored, normal cooling was recovered, and pool temperature returned to approximately 150F. The inspectors evaluated the licensee’s response to this event (see section A4O3 of this report) and continued to monitor fuel pool temperatures throughout the remainder of the outage.

Refueling Activities

The inspectors observed the core offload and reload from the refueling bridge above the reactor cavity, from the bridge above the spent fuel pool, and from the deck around the spent fuel pool. During the core offload, the inspectors also observed the licensee’s inspection of several fuel bundle top nozzles to evaluate the process and acceptance criteria for top nozzle replacement. During the core reload, a cable became detached from the fuel transfer cart, and a fuel bundle was temporarily held up in the fuel transfer tube until other standby cables were used to complete the transfer into the reactor cavity. The inspectors reviewed the licensee’s evaluation of this condition, which concluded that sufficient flow was available through the transfer tube to cool the bundle, and that the temperature of adjacent concrete was not jeopardized. The inspectors also observed management discussions regarding the resolution of the transfer cart equipment problems, and observed manual manipulation of the transfer cart to restore the cable for normal operation.

The inspectors also reviewed the licensee's DERs written during the outage on the fuel transfer system equipment. The licensee had initiated an equipment failure evaluation and design review as part of the corrective actions for these DERs to resolve the equipment problems noted.

Containment Closure

The inspectors reviewed TS requirements for containment closure during refueling activities, plant restart, and power operations and observed the licensee's process for complying with containment closure requirements. The inspectors reviewed procedure PFM-109, "Containment Leakage Rate Testing Program," to ensure that required containment leak rate tests had been conducted in accordance with station procedures and in compliance with the TS.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspectors 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 was adequate with proper calibration, 5) the test was performed in the proper sequence, and 6) the test equipment was removed following testing.

The inspectors observed portions of the following surveillance tests and Inservice Inspection Tests (ISTs) for the 480 volt safeguards buses, the auxiliary boiler feedwater pumps (ABFPs), and the main steam isolation valves (MSIVs).

- 3PT-M62, "Degraded Grid Undervoltage Test," performed on April 6, 2001.

- 3PT-Q120B; "32 ABFP (Turbine Driven) Surveillance and IST," performed on April 13, 2001, and observed by the inspectors. Flow control valve FCV-405 B and C initially failed the IST acceptance criteria on stroke time (DER 01-01339). As required by the test procedure, the licensee performed an evaluation in accordance with American Society of Mechanical Engineers (ASME) standard OM-10, which concluded that the test results were acceptable based upon repeated satisfactory tests.

- 3PT-CS002B; “MSIV 1-31, 1-32, 1-33, & 1-34 Inservice Test,” initially performed on May 17, 2001, and observed by the inspectors in the control room. Three of the four MSIVs failed to meet all IST requirements. The licensee performed American Society of Mechanical Engineers (ASME) code section OM-10 evaluations, and three of the MSIVs failed to meet all of the IST requirements. The “Fix-It-Now “ team adjusted the valve packing and 3PT-CS002B was performed a second time with satisfactory results.

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 (71121.01)

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

The inspector observed activities at the routine radiologically-controlled-area (RCA) control point on the fourth floor of the Administration Building on a daily basis to verify compliance with requirements for RCA entry and exit, wearing of record dosimetry, and issuance and use of electronic dosimeters. On May 14, 15, and 17, 2001, the inspectors toured various areas in the RCA including the reactor containment, the PAB, the fuel storage building, the radioactive machine shop building (RAMS), and the health physics (HP) count room and observed the radiological controls in place for the ongoing refueling outage (RO) activities. During these tours, the inspectors reviewed the posting, labeling, barricading, and level of access control for locked high radiation areas (LHRAs), high radiation areas (HRAs), radiation and contamination areas, and radioactive material areas.

Using a radiation survey meter, the inspector also verified selected posted dose rates in reactor containment. The inspectors observed pre-job briefings for the reinstallation of conoseals on the reactor head (radiation work permit (RWP) #01-220), for hose down of the reactor-side transfer canal (RWP #01-238), and for inspection of the fuel transfer cart in the reactor-side transfer canal (RWP #01-219). Also, the inspector noted that the health physics remote monitoring station (HPRMS) at the main HP RCA access control point was operational and provided video monitoring, voice communication, and teledosimetry for work activities in the RCA. From this vantage point, the inspector observed the radiological controls practiced for various reactor reassembly tasks, reactor-side transfer canal entries, a high radiation trash movement, and releases of radioactive materials from reactor containment through the equipment hatch.

The inspector selectively examined the following RWPs, survey record, procedures, and other program documents.

- RWP #01-219 Fuel transfer system work
- RWP #01-220 Reactor head mechanical support
- RWP #01-221 Reactor head instrumentation and control support
- RWP #01-238 Reactor cavity/transfer canal decontamination
- Post-decontamination dose-rate survey of the transfer canal for an RWP update dated May 17, 2001 (0230 hours)
- Electronic dosimeter report for a diver involved in a transfer canal diving evolution on May 11, 2001
- Procedure RE-REA-4-1, Rev. 16, Radiation work permit (RWP)
- Procedure RE-ACC-5-1, Rev. 22, Radiologically controlled area (RCA) access control
- Procedure RE-REA-4-1, Rev. 16, Radiation work permit (RWP)
- Procedure RE-INS-7UG-4, Rev. 6, Use of Merlin-Gerin DMC-100 dosimeters
- Procedure RE-INS-7UG-5, Rev. 3, Programming Merlin-Gerin DMC-100 electronic dosimeter using Windows
- Memorandum titled Tritium Exposure During RO-11, dated May 10, 2001
- RO-11 worker contamination statistics, dated May 17, 2001
- Checklist items for the Radiation Protection Program Monitoring Team for refueling outage surveillances
- Reports by the Radiation Protection Program Monitoring Team for refueling outage surveillances for April 28, 29, 30, May 1, 2, 4, 5, 6, 8, 9, 10, 11, 12, and 13, 2001
- Radiological Event Report summary for the first quarter of 2001
- Memorandum titled Adjustments to First Quarter 2001 TLD Reads, dated April 23, 2001
- Memorandum titled Condenser Material Free Release Method, Rev 1, dated May 9, 2001
- Memorandum titled Evaluation of Trailers Temporarily Occupied on Tank Pad, dated April 12, 2001

The inspector reviewed the following six DERs and their associated Action Commitment Tracking System (ACTS) items for appropriateness of categorization, immediate corrective actions, and corrective actions to prevent recurrence and for the timeliness and effectiveness of corrective actions: DERs 01-01110, 01-01495, 01-01604, 01-01653, 01-01771, and 01-01931. These DERs were generated during the period of March to May 2001.

The review in this section 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 (71121.02)

The inspector reviewed radiological work activities and practices during tours of the facilities and inspected procedures, procedural implementation, records, and other program documents to determine the effectiveness of ALARA (As Low As Reasonably Achievable) planning and control.

During the pre-job briefs, the tour of the facilities, and the observations from the HPRMS described in the previous section, the inspector observed the attention paid by radiation protection technicians to dose avoidance in discussions with radiation workers, the directions given by radiation protection technicians to workers during job coverage to maximize dose avoidance, and the use of signs to identify low dose waiting areas and to identify higher dose areas where access time should be minimized.

The inspector selectively examined the following ALARA review packages, procedure, and other program documents.

Pre-job and Ongoing ALARA review packages for

- RWP #01-215, Scaffolding in vapor containment and in primary auxiliary building
- RWP #01-241, Steam generator secondary side hand hole/port/man way cover
- RWP #01-244, Steam generator secondary side sludge lance/bundle flush
- RWP #01-251, Motor-operated valve testing in vapor containment and in primary auxiliary building
- RWP #01-260, Non-regenerative heat exchanger work
- RWP #01-263, Diving operations in vapor containment/fuel storage building
- Procedure RE-REA-4-1, Rev. 16, Radiation work permit (RWP)
- RO-11 outage ALARA reports dated May 9, 15, 16, 17, 2001 (containing top five RWP's by dose, RO-11 dose tracking, RWP dose tracking, and departmental dose breakdown)
- Pre-outage ALARA committee meeting minutes for RO-11 dated April 18, 2001

The review in this section 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.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope (71121.03)

The inspection included the following activities to determine the accuracy and operability of radiation monitoring instruments that are used for the protection of occupational workers. Inspection activities selectively verified the calibration, operability, and alarm set-point (if applicable) of selected instruments and equipment, including the following:

- HP survey equipment including RO2 and RO2A survey meters and teletectors,
- HP monitoring equipment including area air samplers, hand-held friskers, continuous air monitors (CAMs), small article monitors (SAMs), portal monitors, and whole body contamination friskers, and
- HP counting room instrumentation including gas flow proportional and beta counting instruments.

The inspection included a review of the radiation survey meter self-issue point inside the radiation controlled area (at the containment access facility (CAF)), the calibration status of the meters staged for use, the daily source check process, and the process for placing meters out of service.

The inspector selectively examined the following procedures:

- Procedure RE-INS-7-CE-6, Rev. 9, Calibration of the N.E. IPM-7/8 Installed personnel monitor
- Procedure RE-INS-7-CE-8, Rev. 6, Calibration of the N.N.C. Gamma-60 portal radiation monitors
- Procedure RE-INS-7-CF-3, Rev. 10, Calibration of Tennelec LB-5100 Alpha/beta counting system
- Procedure RE-INS-7CF-15, Rev. 4, Interchangeable counting room calibration with HP-300

The review in this part (2OS3) was against criteria contained in Title 10 of the Code of Federal Regulations (CFR) Parts 20.1203 (Determination of external dose from airborne radioactive material), 20.1204 (Determination of internal exposure), Subpart F (Surveys and monitoring), Subpart H (Respiratory protection and controls to restrict internal exposures in restricted areas), and Subpart L (Records) and against criteria contained in 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)

Unplanned Scrams/7000 Critical Hours and Scrams With Loss of Normal Heat Removal

To verify the number of “unplanned scrams per 7000 critical hours” and the “scrams with a loss of normal heat removal” performance indicators (PIs), the inspectors reviewed all the PI data related to scrams reported to the NRC. The unplanned scrams per 7000 critical hours PI was last verified in Inspection Report (IR) 2000-002 dated May 20, 2000, and the scrams with a loss of normal heat removal was last verified in IR 2000-004 dated July 1, 2000. The number of unplanned scrams and the number scrams with a loss of heat removal were obtained by reviewing the licensee’s DER database and by reviewing Licensee Event Reports (LERs). The number of critical hours was obtained by reviewing operator logs. The inspectors used Nuclear Energy Institute (NEI) Report 99-02, “Regulatory Assessment Performance Indicator Guideline” to calculate the PIs.

b. Findings

No findings of significance were identified

4OA3 Event Follow-up

a. Inspection Scope (71153)

On May 8, 2001, with the normal spent fuel pool cooling (SFPC) system out of service for maintenance, spent fuel pool cooling was temporarily lost when the back up spent fuel pool cooling (BUSFPC) pumps tripped as a result of a loss of makeup water to the BUSFPC towers. The temperature in the pool rose from 151 to 155F. The inspectors responded to this event and evaluated the licensee’s immediate actions to restore makeup water to the BUSFPC system. Although makeup was restored within approximately 20 minutes after the loss was discovered, pool cooling was interrupted for approximately 50 minutes. Once restored, the pool temperature gradually dropped to 150F.

During a review of nuclear plant operator (NPO) logs for the BUSFPC system, the inspectors noted that the make up water pressure indicator data was out of the acceptable range during the period May 6-9, 2001, and the licensee did not initiate a DER. In accordance with administrative procedure AP-21, “Conduct of Operations,” a DER should be written after a logged reading is out of specifications for 24 hours. The licensee acknowledged that a DER should have been written, and subsequently initiated DER 01-2435 to document this condition.

b. Findings

Following the initial resident inspector follow-up to this event, the NRC conducted a special onsite inspection the week of May 14, 2001. The results of this special inspection are to be documented in Inspection Report 50-286/01-06.

4OA6 Meetings

Exit Meeting Summary

On June 7, 2001, the inspectors presented the inspection results to Mr. F. Dacimo and other Entergy staff members who acknowledged the inspection results presented. The inspectors asked Entergy personnel whether any materials evaluated during the inspection were considered proprietary. No proprietary information was identified.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

R. Barrett	Vice President, Operations - IP3
J. Barry	Sr. Radiological Engineer
R. Burroni	I&C Superintendent
F. Dacimo	General Manager of Plant Operations
E. Danko	Licensing Engineer
J. Comiotes	Director, Safety Assurance
J. DeRoy	Director, IP-3 Engineering
R. Deschamps	Radiological and Environmental Services Manager
R. LaVera	Sr. Radiological Engineer
D. Mayer	Manager, Health Physics/Chemistry
F. Mitchell	HP General Supervisor
J. Perrotta	Quality Assurance Manager
K. Peters	Corrective Actions/Assessment Manager
P. Rubin	Operations Manager
J. Russell	Special Projects Manager
F. Short	ALARA Engineer
R. Solano	HP Supervisor
J. Stewart	HP Supervisor
A. Vitale	Maintenance Manager
J. Wheeler	Training Manager

b. List of Items Opened, Closed and Discussed

Opened and Closed

FIN 50-286/01-04-01 DER 01-01399 32 main lube oil cooler - service water leak

c. List of Acronyms

ABFP	auxiliary boiler feedwater pump
ACTS	Action Commitment Tracking System
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
BUSFPC	backup spent fuel pool cooling
CAF	containment access facility
CAM	continuous air monitors
CCR	central control room
CFR	Code of Federal Regulations
COL	check-off list

List of Acronyms (cont.)

CS	containment spray
CSR/ET	cable spreading room/electrical tunnel
CTO	caution tag out
DER	deviation/event report
EDG	emergency diesel generator
FIN	Finding
FRV	feedwater regulating valve
FSAR	final safety analysis report
FSB	fuel storage building
HP	health physics
HPRMS	health physics remote monitoring station
HRA	high radiation area
ICCDP	incremental conditional core damage probability
ILLIS	Intermediate Leg Level Indication System
IP2	Indian Point 2
IR	inspection report
IST	in-service test
Kv	kilovolt
LCO	limiting condition for operation
LER	licensee event report
LHRA	locked high radiation area
MLO	main lube oil
MLMS	Mansell Level Monitoring System
MOV	motor operated valve
MSLB	main steam line break
NEI	Nuclear Energy Institute
NPO	nuclear plant operator
NRC	Nuclear Regulatory Commission
NSE	nclear safety evaluation
OD	operability determination
PAB	primary auxiliary building
PI	performance indicator
PM	preventive maintenance
POP	plant operating procedure
PTO	protective tagout
RAMS	radioactive machine shop
RAS	reasonable assurance of safety
RCA	radiologically controlled area
RCS	reactor coolant system
RHR	residual heat removal
RO	refueling outage
RWP	radiation work permit
SAM	small article monitors
SFPC	spent fuel pool cooling
SGBD	steam generator blowdown
SOP	standard operating procedure
SSCs	structures, systems and components
SWP	service water pump
TS	Technical Specifications
WR	work request