U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.	92-14
Docket No.	50-219
License No.	DPR-16
Licensee:	GPU Nuclear Corporation 1 Upper Pond Road Parsippany, New Jersey 07054
Facility Name:	Oyster Creek Nuclear Generating Station
Inspection Period:	June 7, 1992 - July 18, 1992
Inspectors:	Dave Vito, Senior Resident Inspector Tim Frye, Reactor Engineer John Nakoski, Resident Inspector

Approved By: John Rogge   8/12/92

John Rogge, Section Chief   Date
Reactor Projects Section 4B

Inspection Summary: This inspection report documents the safety inspections conducted during day shift and backshift hours of station activities including: plant operations; radiological controls; maintenance and surveillance; engineering and technical support; emergency preparedness; security; and safety assessment/quality verification.

Results: Overall, GPUN operated the facility in a safe manner. The 'C' electromatic relief valve was inadvertently opened for about eight seconds on July 5, 1992, due to a personnel error by two I&C technicians. A failure to follow procedure caused this personnel error and was identified as a violation.

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EXECUTIVE SUMMARY

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Plant Operations

Operators responded well to an automatic reactor scram on a high-high intermediate range monitor (IRM) signal and to the inadvertent opening of the 'C' electromatic relief valve (EMRV). Response to the concerns of NRC Bulletin 92-01 regarding Thermo-Lag fire barriers was conservative and appropriate.

Radiological Controls

Work on the first phase of the spent fuel pool waste disposal project was performed well.

Maintenance/Surveillance

Poor performance by two I&C technicians resulted in the inadvertent opening of the 'C' EMRV. Work on the 1-2 reactor building closed cooling water (RBCCW) heat exchanger and the inspection of the No. 2 emergency diesel generator were performed well.

Engineering and Technical Support

A plant modification which added pressure gauges between the reactor building-to-torus vacuum breaker valves was properly installed and appropriately documented.

Safety Assessment and Quality Verification

Post-transient review group (PTRG) and independent transient review group (ITRG) efforts in response to the automatic scram caused by faulty IRM bypass switches were good. Operations developed a comprehensive assessment of the plant response to the inadvertent opening of the 'C' EMRV. Maintenance performed an adequate critique of the technician error which caused the opening of the 'C' EMRV. However, the procedure noncompliance which caused this event was identified as a violation.

<u>DETAILS</u>

1.0 OPERATIONS (71707,93702)

1.1 Operations Summary

At the start of the inspection period, the unit was in preparation for startup from a short maintenance outage. The initial startup effort was halted on June 7, 1992, due to substantial packing leakage on 'B' isolation condenser steam inlet valve V-14-33. Startup recommenced on June 10, 1992, after repairing V-14-33. At 11:01 p.m. on June 10, 1992, about 20 minutes after achieving criticality, an automatic scram occurred due to a high-high intermediate range monitor (IRM) signal. There were no indications of an actual reactivity excursion, and the root cause was ultimately determined to be faulty IRM bypass switches. The IRM bypass switches were replaced and startup was again initiated on June 11, 1992.

At 12:15 p.m. on June 13, 1992, with the unit at 90% power, the control room received a report of a forest fire about four miles away from the site. At 5:15 p.m. that day, after learning that the fires could potentially affect the 230kV offsite distribution lines, GPUN began to reduce power to about 40% to get to within the capacity of the turbine bypass valves in anticipation of a possible load rejection. Power ascension was recommenced at 11:00 p.m. on June 13, 1992, after GPUN received a report from the local fire marshall that the fire, although not fully extinguished, was under control and not a threat to the offsite distribution lines. Full power was achieved at 6:30 p.m. on June 14, 1992. The unit operated at or near full power for the remainder of the inspection period.

On July 5, 1992, the 'C' electromatic relief valve (EMRV) was inadvertently opened for about eight seconds due to I&C technician error. Control room operators responded well to the brief transient condition. This issue is discussed further in Sections 1.3 and 3.4 of this report.

1.2 Automatic Reactor Scram on Intermediate Range Monitor (IRM) High-High Signal

At 11:01 p.m. on June 10, 1992, an automatic reactor scram occurred from about 0.5% reactor power when high-high signals were received on four of the eight IRMs. IRMs 11, 12, 13, and 14 are associated with reactor protection system (RPS) channel 1, and IRMs 15, 16, 17, and 18 are associated with RPS channel 2. The IRM scram logic is one out of four in each RPS channel taken twice.

The reactor was in the startup mode and being maintained critical while range correlation was being performed between IRM ranges 6 and 7. This correlation is done because the IRMs switch to a different preamplifier mode when going from range 6 to range 7. Range correlation had been completed on five of the eight IRMs (IRMs 12, 13, 16, 17, and 18), and these IRMs were in range 7. While the operator was preparing to bypass IRM 15 for range correlation, high-high signals were simultaneously received on IRMs 12, 16, 17, and

18, and the full scram occurred. The co...rol rods inserted, and no other engineered safety systems actuated.

The plant was placed in cold shutdown, and a post-transient review group (PTRG) was convened to assess the cause of the scram. The results of the assessment of plant parameters and conditions before and after the event indicated that an actual reactivity excursion had not occurred. There was no change in reactor water temperature or other indication of a cold water addition. The control room operators were not moving control rods while the IRM range correlation was being performed. Rod worth minimizer data showed that there had been no control rod movement for more than seven minutes prior to the scram signal. There were no control room alarms received or IRM/SRM recorder increases indicated before the scram. The IRMs which provided the scram signals were all in range 7; the three IRMs which were in range 6 did not provide any of the scram signal inputs.

PTRG concluded that the scram had been caused by electronic noise in the IRM bypass circuits. IRM bypass switch sensitivity has been noted in the past to have caused half scrams when switching from range 6 to range 7. GPUN had planned to replace the switches during the upcoming refueling outage to resolve the switch sensitivity problem. However, this occurrence demonstrated that the noise signal generated in the bypass switch circuit could be fed back through the IRM circuitry, causing an upscale trip on an IRM in the other RPS channel, completing the scram logic. After the scram, both IRM bypass switches were replaced with new switches of similar type. Appropriate post-maintenance testing was done on the new switches to verify that the noise problem had been eliminated. GPUN also checked the effect of SRM drive motion on the IRMs and found that it did not contribute to the electronic noise problem. GPUN is investigating replacement of the older type switches with new switches which will be less susceptible to electronic noise.

The inspectors reviewed the plant data related to the transient and reached the same conclusions as the PTRG. The inspectors concluded that the event was of low safety significance because of the low power level and the fact that no actual reactivity excursion occurred to cause the scram. The inspectors found the PTRG and subsequent Independent Transient Review Group (ITRG) reviews to be comprehensive and accurate.

1.3 Operator Response to Inadvertent Opening of Electromatic Relief Valve due to Technician Error

With the plant at 100% power on July 5, 1992, the 'C' electromatic relief valve (EMRV) was inadvertently opened for about eight seconds due to a personnel error by two I&C technicians. The technicians were testing the pressure sensor for the 'C' EMRV instead of the 'B' EMRV which had been taken out of service for testing by the control room operators as planned. When the technicians used the test equipment to raise the pressure on the 'C' EMRV pressure sensor with the control switch for the 'C' EMRV still in AUTO, the relief valve opened. The control room operators quickly realized the technician error and closed the 'C' EMRV manually by taking the control switch to the OFF position. The plant responded as designed and within about two minutes plant parameters had returned to normal 100% steady state conditions. The I&C technicians were directed by the control room to back out of the surveillance procedure and return the 'C' pressure sensor to a normal condition. The 'B' and 'C' EMRV control switch positions were then returned to the AUTO position from the OFF position. Operations reviewed the plant response to the event. Maintenance performed a critique to assess the root cause of the event and to recommend appropriate corrective actions (see Section 3.4).

Operations performed a comprehensive review of the plant response. The initial observed reactor pressure and main steam flow decreases correlated to that expected for an open EMRV. The response of reactor water level and the feedwater control system was also as expected. Reactor power initially decreased from 100% to about 97% due to the void increase caused by the pressure reduction. After closing the 'C' EMRV, power momentarily increased to about 104% in response to the pressure increase and then returned to the 100% power level as operating parameters stabilized. One channel of torus water temperature indication increased from 79 degrees F to 80 degrees F. The other torus water temperature indication remained constant at 79 degrees F. Following the transient, downcomer temperatures were plotted over the next two hours to assure that the EMRV had seated. As required by technical specifications, a torus to drywell vacuum breaker operability test was performed with satisfactory results.

As a result of this event, a 4-hour NRC notification was made due to the inadvertent actuation of an engineered safety feature (the EMRVs provide the automatic depressurization system (ADS) function). The licensee determined that the technical specifications regarding ADS were complied with during this event. Tech. spec. 3.4.B notes a 72-hour action statement if one EMRV is inoperable. While the control switches for both the B and C EMRVs were in the OFF position for a short period of time after the operators closed the C EMRV, only the ADS function for the 'C' EMRV was inoperable because of the connection of test instrumentation by the I&C technicians. Putting the remote manual control switch in the OFF position defeats the overpressurization function of an EMRV, but not the ADS function. As such, the ADS function of the B EMRV was not rendered inoperable. The inspectors concluded that the control room operators had responded well to the transient and that the operations department post-event assessment was appropriate.

1.4 Licensee Response to Bulletin 92-01

NRC Bulletin No. 92-01, "Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage," was hand delivered by the inspectors to the manager of plant operations at 11:00 a.m. on June 25, 1992. After review by the licensee, at 5:00 p.m. on June 25, the Thermo-Lag fire barriers affected by the bulletin were declared inoperable and compensatory measures were taken as directed by technical specifications. Hourly fire watch patrols were established for the 15 specific areas in the reactor building and turbine building containing Thermo-Lag as identified by procedure 645.6.028, Rev 2, "Thermo-Lag Envelope System Fire Barrier Surveillance."

By June 26, it was established that there were no affected wide cable trays as described in the bulletin. At this time the licensee was still evaluating specific conduit sizes and configurations employing the Thermo-Lag insulation. In addition, a closed circuit camera and monitor were setup to monitor the TIP shield room due to the high radiation levels in this area.

On July 1, the reactor building equipment drain tank (RBEDT) room was removed from the hourly fire watch tours. This was in response to an evaluation by Technical Functions that the Thermo-Lag insulation in the RBEDT room provided adequate fire protection for the affected conduit. This evaluation was based on a verification by the licensee that the insulation in this room was installed in strict compliance with the vendor installation instructions, trained and certified installers and quality control inspectors were used exclusively during the Thermo-Lag installation, adequate quality control measures were taken during installation, and a review of material receipt inspections to verify material quality. Also it was determined that the fire load in the RBEDT room was not sufficient to produce a fire of the magnitude required to jeopardize the insulation. The RBEDT room is provided with ionization type fire detectors to provide early warning of a fire condition. The inspector reviewed the Oyster Creek Fire Hazards Analysis Report and verified that the fire loading in the RBEDT was essentially 0 BTU/hr, excluding transient fire loads, and that the room was equipped with a fire detection system.

On July 8 and July 10, the inspector walked down all accessible areas of the plant which contained the Thermo-Lag insulation. No obvious material or installation defects were noted in the insulation. The inspector also observed the areas for combustible materials which could create a fire hazard. Control of combustible material and ignition sources in all areas was good.

On July 16, a plant review group (PRG) was convened to discuss the actions taken to date to verify the adequacy of the Thermo-Lag insulation, status of the fire watches, and the future course of action. It was determined by the PRG that based on review and evaluations by Technical Functions of the installation procedures, area fire loading, fire detection and suppression systems present, and fire endurance test data available, the insulation in all but three areas provided an adequate fire barrier to support operability. Two of the areas secured contained Thermo-Lag applications not covered by Bulletin 92-01; one area contained a stairwell enclosure and the others contained drywell electrical penetrations. The three areas for which the hourly fire watches would continue were in the 480 volt switchgear rooms "A" and "B." The adequacy of the Thermo-Lag fire barrier was considered indeterminate based in part on the high fire loading in these rooms.

The licensee's response to Bulletin 92-01 has been conservative and appropriate. The licensee quickly evaluated the bulletin and established the appropriate compensatory measures in a timely manner. They have maintained the hourly fire watch patrols on the three remaining areas and the PRG and operations department displayed a questioning attitude with regard to securing the fire watches. They required strong evidence and justification that the installed Thermo-Lag insulation provides adequate fire protection or is not an application addressed by Bulletin 92-01 before considering securing any of the fire watches. The licensee will address compliance with 10 CFR Part 50, Appendix R requirements as further conclusions are reached during the Bulletin 92-01 review process regarding necessary corrective actions for specific fire barrier configurations.

1.5 Hourly Fire Watch Tours

On July 9, 1992, the licensee determined that several hourly fire watch tours performed by an equipment operator (EO) during day shift on July 9 had not been properly completed. The tours included the turbine building mezzanine, 480V switchgear room hallway, and in response to Bulletin 92-01, the 14 remaining areas in the reactor building and turbine building containing the Thermo-Lag fire barriers. The discrepancy was discovered by the control room operators during a conversation with the EO. It was determined that the EO had only been touring the turbine building mezzanine, 480V room hallway, and two other areas in the turbine building. Four of the hourly fire watch tours performed on dayshift, July 9, were performed by this EO and were, therefore, incomplete.

The control room operators immediately brought the situation to the attention of the group shift supervisor. The EO was then relieved of his fire watch duties. Operations management, after discussion with the EO, had the EO's site access revoked. In addition, the EO had submitted his resignation, previous to this event, effective July 15, 1992. His resignation was accepted effective immediately on July 10, 1992.

Since the issuance of the bulletin on June 25, 1992, the hourly fire watches had been performed by contract personnel except for the Thursday fire watches which were performed by Oyster Creek EOs. The cause of this particular incident was in part due to an ineffective turnover of fire watch responsibilities from the 12-8 shift fire watch contractor to the dayshift EO. The EO was apparently unaware of all of the areas in the reactor building and turbine building for which he was responsible. Also, he did not carefully read the fire watch tour sheet and understand the tour areas for which he was signing. However, there did not appear to be any deliberate or willful intent on the part of the EO to falsify the fire watch tour sheet. Operations committed to review the vital access security logs starting from June 25, 1992, to verify that all required Thermo-Lag insulation fire watch tours were adequately performed. Operations also reviewed all required Thermo-Lag insulation areas with the other individuals who were performing the fire watches. All of these individuals were fully aware of the areas required to be toured.

1.6 Fire Drill

On June 16, 1992, the inspector observed a fire drill. The drill consisted of a simulated fire in a shed near the engineering office building. The ignition source was unknown to the responding fire brigade.

The fire brigade response was timely and appropriate. Members were adequately dressed, including self contained breathing apparatus (SCBA). The critique of the drill was good, and one individual who had some minor difficulty donning the SCBA received effective coaching from the drill observer. The inspector verified that the brigade members were currently qualified. The fire fighting strategy was appropriate, and the brigade leader adequately controlled the effort. Security and emergency medical personnel also responded to the drill in a timely manner.

1.7 Facility Tours

The inspectors observed plant activities and conducted routine plant tours to assess equipment conditions, personnel safety hazards, procedural adherence, and compliance with regulatory requirements. Tours were conducted of the following areas:

- control room
- cable spreading room
- diesel generator building
- new radwaste building
- old radwaste building
- transformer yard
- intake area
- reactor building
- turbine building
- vital switchgear rooms
- access control points

Control room activities were found to be well controlled and conducted in a professional manner. Inspectors verified operator knowledge of ongoing plant activities, equipment status,and existing fire watches through random discussions.

2.0 RADIOLOGICAL CONTROLS (71707)

During entry to and exit from the radiologically controlled area (RCA), the inspectors verified that proper warning signs were posted, personnel entering were wearing proper dosimetry, personnel and materials leaving were properly monitored for radioactive contamination, and monitoring instruments were functional and in calibration. Posted extended Radiation Work Permits (RWPs) and survey status boards were reviewed to verify that they were current and accurate. The inspector observed activities in the RCA and verified that personnel were complying with the requirements of applicable RWPs and that workers were aware of the radiological conditions in the area.

2.1 Spent Fuel Pool Waste Disposal Project

On July 7, 1992, the inspector observed the initial loading of 3 underwater vacuum system filters from the spent fuel pool to an 8-120B cask. This work is part of the first phase of the spent fuel pool waste disposal project. Currently, there are 78 used underwater vacuum system filters being stored in the spent fuel pool. Approximately 60 filters will be transferred to the first cask for shipment offsite. The filters are transferred, three at a time, by lowering a shielded filter transfer bell into the spent fuel pool and loading the filters into the transfer bell. The transfer bell is raised above the surface of the spent fuel pool and allowed to drain. The transfer bell is then moved over the cask and the filters lowered from the bell into the high integrity container inside the cask.

During the transfer of the filters to the cask, the inspector observed that all personnel involved were in compliance with the radiation work permit (RWP) requirements. Radiation surveys of the transfer bell and general area were performed as required during filter movement by the radiological controls technicians. Smear surveys of the work area for hot particles after filter transfer were also observed. The inspector reviewed the RWP and the ALARA review prepared for this job. The inspector noted that a pre-job briefing had been held and properly documented with additional personnel briefed at a later time. The ALARA review contained expected radiation levels during the filter transfer and provided appropriate actions for personnel to take if these levels were exceeded. The ALARA review also provided actions to be taken in the event of a dropped filter or area radiation monitor alarm.

All personnel questioned were very knowledgeable of the operation and the hazards associated with this job. The filter transfer effort appears to have been well planned with good communications and coordination demonstrated by the workers.

3.0 MAINTENANCE/SURVEILLANCE (62703,61726)

3.1 Surveillance Observations

The inspectors observed selected surveillance tests to determine whether properly approved procedures were in use, appropriate approval was obtained and prerequisites satisfied prior to beginning the test, test instrumentation was properly calibrated and used, radiological practices were adequate, technical specifications were satisfied, and personnel performing the tests were qualified and knowledgeable about the test procedure. The following surveillance test activities were observed.

636.2.012, Rev. 1, Diesel Generator Batteries Service Test

636.2.002, Rev. 21, Six Month Diesel Generator Inspection

No significant problems were noted during the observation of these surveillances. The surveillances were adequately controlled and conducted.

3.2 No. 2 Diesel Generator Inspection and Breaker Replacement

On June 22, 1992, the inspector observed the six-month diesel generator inspection per Station Procedure 636.2.002, Rev. 21, on emergency diesel generator (EDG) No. 2 and the replacement of the 4160V output circuit breaker per work request (WR) #757343. In addition to performing a periodic inspection of the EDG, procedure 636.2.002 exercises the fast start logic and bypass circuits for emergency starting operations. The 4160V output circuit breaker was replaced with a breaker returned from General Electric (GE) after overhaul. This work was performed as part of a program developed by the licensee to systematically replace all 4160V breakers with GE overhauled breakers in response to the prop spring failure issue discussed in Inspection Reports 50-219/92-07 and 50-219/92-08.

During the EDG inspection, the electricians displayed a good understanding of the scope of the work to be accomplished. They thoroughly read and reviewed each procedure step to ensure that questions were resolved prior to performance. All data and required signatures were promptly recorded as required. Good communications were maintained with the control room to keep the operators informed of impending EDG starts. The inspector verified that all required signatures were obtained prior to commencing work, the required tagout for inspection was properly executed, and the test equipment used for the inspection was calibrated.

Concurrent with the EDG inspection, the inspector also observed the replacement of the 4160V output breaker. The workers involved displayed good work practices in handling the breakers and in working inside the breaker cabinet. Breaker replacement was well coordinated with the ongoing EDG inspection. Overall, both maintenance activities were well performed.

3.3 RBCCW Heat Exchanger Cleaning and Inspection

On June 24 and June 25, 1992, the inspector observed the cleaning of the 1-2 reactor building closed cooling water (RBCCW) heat exchanger. This work was performed per job order (JO) #39723. The inspector reviewed the work package and verified that the maintenance was being performed in accordance with the instructions. Adequate controls were in place to contain residual water in the heat exchanger to prevent damage to surrounding equipment. Equipment used during the maintenance was verified to be in current calibration. The inspector concluded that the work was well performed.

3.4 Maintenance Critique of Technician Error Which Caused Inadvertent Opening of 'C' EMRV

A critique was performed by the maintenance department to assess the root cause and recommend appropriate corrective actions following the inadvertent opening of the 'C' EMRV caused by two I&C technicians on July 5, 1992. The root cause assessment found that the technicians had missed several opportunities to identify and correct their personnel error. The technicians went to the wrong EMRV pressure sensor despite procedural guidance on its location. The technicians did not verify the instrument they were testing by looking at the calibration sticker on the sensor housing. The technicians did check the voltage on the switch contacts in the sensor box in an attempt to verify that the control switch had been turned off. However, due to an apparent improper connection of the voltage meter contacts, the meter reading was about 6 millivolts dc instead of what it should have read (120 vdc). The licensee noted that the location of the switch contacts in the sensor box can make connection of the voltage meter contacts difficult.

The two I&C technicians have been excluded from work on safety systems until they complete a requalification program and receive approval from I&C supervision. The requalification program includes a self-checking training session and a requirement that the technicians requalify on specific on-the-job training surveillances. The technicians are also being required to develop and conduct a training session on how to avoid this type of incident in the future. To address to the hardware problem with access to the switch contact terminal points within the sensor box, an engineering work request was to be developed to assess whether the terminal points could be moved to facilitate connection of the voltage meter contacts.

The inspectors concluded that the maintenance department followup of this event was appropriate and that the root cause assessment was thorough. However, the performance of the technicians was considerably below the level expected by management. Their performance was indicative of a lack of self-checking, contrary to the practice actively promoted by GPUN, and a lack of attention to detail of some routine craft practices. The failure of the technicians to follow procedure 602.3.014, Rev. 0, "Electromatic Relief Valve (EMRV) Pressure Sensor/Pilot Valve Control Relay - Test and Calibration," step 6.3 for the location of the B EMRV was identified as a violation (50-219/92-14-01).

4.0 ENGINEERING AND TECHNICAL SUPPORT (71707,37828)

4.1 Reactor Building-to-Torus Vacuum Breaker Minor Modification

The inspector reviewed minor modification package 130-92 which performed a corrective change to install pressure gauges between the reactor-building-to-torus vacuum breaker valves. The modification package was verified to have been prepared in accordance with procedure 125, revision 11, "Conduct of Plant Engineering." All reviews and approvals required by procedure 125 were obtained. The modification package contained adequate instructions for the installation of components and equipment. The engineering evaluation clearly described the purpose of the modification and its effect on system integrity and operation.

A technical review/safety review determination and safety evaluation were prepared for this modification and were reviewed by the inspector. These were verified to have been prepared in accordance with procedure 130, revision 5, "Conduct of Technical Review and Safety Review by Plant Review Group." The modification package was reviewed by a responsible technical reviewer and a safety evaluation was prepared and reviewed by an independent safety reviewer as required by the determinations. The safety evaluation was well prepared and provided thorough justification that the installation of the modification did not involve an unreviewed safety question or require a technical specification change.

The inspector verified that the equipment was installed in accordance with the direction of the modification package. Instrumentation was verified to be currently calibrated. Components were labeled and valves positioned in accordance with the modification package. Overall, the modification package preparation and equipment installation was well performed.

5.0 OBSERVATION OF PHYSICAL SECURITY (71707)

During routine tours, inspectors verified that access controls were in accordance with the Security Plan, security posts were properly manned, protected area gates were locked or guarded and that isolation zones were free of obstructions. Inspectors examined vital area access points and verified that they were properly locked or guarded and that access control was in accordance with the Security Plan.

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (40500, 71707)

6.1 In Office Review of Licensee Event Reports

NRC inspectors reviewed the following LERs and verified appropriate reporting, timeliness, complete event description, cause identification, and complete information. In addition, the need for further on site review was assessed.

LER NO.	DESCRIPTION
92-006	Electromatic Relief Valve High Pressure Relief Setpoint Exceeded Technical Specification Limit Due to Drift
92-007	Reactor Scram Caused by Failed IRM Bypass Switch During Plant Startup
92-008	Technical Specification Required Shutdown Due to Isolation Condenser Valve Inoperability

The LERs accurately reflected the sequence of events, cause of the failures, safety significance, and appropriate corrective actions. The subject of LER 92-007 is discussed in more detail in Section 1.2 of this report.

7.0 REVIEW OF PREVIOUSLY OPENED ITEMS (92701, 40500)

(Closed) Open Item 50-219/90-23-06. This item tracked GPUN's response to one weakness and ten observations that remained open after an Integrated Performance Assessment Team Inspection (50-219/87-24) and subsequent NRC review in Inspection Reports 50-219/88-13 and 50-219/90-23.

(Closed) (observation) page 6. Management attention in educating the non-licensed worker level understanding of risk importance and in establishing operations as a lead proponent of that philosophy is needed.

(Closed) (observation) page 43. Improvement is needed in the areas of risk perspective and understanding the design basis of the plant.

The lesson plan development guidelines used by maintenance and equipment operator training personnel for updating and developing lesson plans have been revised to include a review of FSAR design basis when appropriate. The inspector reviewed several maintenance worker and equipment operator safety related system lesson plans with training personnel and noted that they included thorough discussions of the system design basis. Included in these discussions are system operating parameters and the accidents and abnormal operating occurrences whose consequences these systems are designed to mitigate.

In addition, the licensee's basic principles trainer (BPT) has been incorporated into equipment operator (EO) and maintenance worker training. Since January 1992, EOs have been participating in BPT exercises with licensed operators. This has allowed the EOs to walk through the emergency operating procedures, abnormal operating procedures, and P&IDs with the instructors so that they have a better understanding of why they are performing certain actions when requested by the control room. The BPT has been used for maintenance worker training so that these workers can see how the failure of different components affect system response to accidents and transients.

(Closed) (observation) page 10. The plant review group (PRG) is apparently under utilized by virtue of the fact that the group essentially meets only to review LERs and very few other plant events.

(Closed) (observation) page 40. The PRG is infrequently used during the review of complex safety issues.

The inspector discussed this item with the Manager, Nuclear Safety and reviewed PRG meeting minutes for the past year. In addition to licensee event report and technical specification amendment reviews, the PRG is being effectively used in a consulting role by operations department management. It was noted that the PRG has met frequently to discuss such issues as complex operability issues and technical specification applicability for which the operations department could not make easy, clear-cut determinations. The PRG had also met several times to discuss deviation reports involving significant issues such as reportability and system status. The PRG has also convened to discuss responsible technical reviews and independent safety reviews for more complex safety issues. A PRG meeting regarding the determination of operability of Thermo-Lag fire barriers was observed by the inspector during this inspection period and was found to be thorough and safety conscious.

(Closed) (observation) page 40. Training given to the responsible technical reviewer (RTR) and the independent safety reviewer (ISR) consisted of only a four-hour oral presentation with no method to gauge the effectiveness of the training. There is no formal training given to personnel who prepare and present safety evaluations.

Safety review process training was revised and a pilot training course was presented on June 29 and June 30, 1988. The course was endorsed by the General Office Review Board (GORB) and is required to be taken by all qualified ISRs and RTRs. While not required to be taken by safety evaluation (SE) preparers, the course was endorsed and recommended for SE preparers by the President, GPU Nuclear, in July 1990. The course includes case studies to give the student practical experience in safety determination and SE preparation. The first case study consists of a review of a prepared safety determination for errors. The second case study requires the preparation of a safety determination and SE. This is graded as Pass/Fail. In addition, a 20 question True/False test is given to test the knowledge of the students.

(Closed) (weakness) page 29. The licensee's matrix type organization and complexity of the MCF work planning procedures places an extraordinary paperwork burden on the people performing work. The individual job foreman becomes heavily involved in administrative duties which detract for time spent actually supervising work activities.

In response to this item, the Maintenance, Construction, and Facilities (MCF) division established a "Work Simplification Committee" to reduce the paperwork burden on job supervisors and planners. A major effort of this committee was to streamline the job package planning, review, and approval process. This has been accomplished through the continuing implementation of the licensee's Generation Maintenance System 2 (GMS2) computer system.

Job planners are now able to create job packages electronically in GMS2 as opposed to having to write them out by hand. These packages can then be reviewed and approved in the computer system, saving the planners and supervisors time in obtaining these concurrences. This saving is particularly noteworthy in the area of quality control (QC) review, approval, and job closeout review. GMS2 has also improved the scheduling of preventive maintenance (PM) and surveillances. PM and surveillance frequencies and due dates are contained in the GMS2 data base and weekly schedules are automatically generated by the system. Production planning is enhanced by being able to list all job packages outstanding for a particular system so that job supervisors can ensure that all required work on a system is recognized and performed. The GMS2 data base also provides valuable information to the supervisor on component stock numbers, vendor information, and availability. Component material history has also been incorporated into the system to allow supervisors easy access to component failures and trends. Also facilitating the accomplishment of work onsite in the last year was the incorporation of the work scheduling function within the Oyster Creek organization. Additional improvement in reducing the paperwork burden is expected as process reengineering project (PREP) initiatives are implemented. The licensee's goal for PREP is to make the process used for the performance of maintenance and modification activities more efficient.

In early 1990, the organizations responsible for maintenance activities at Oyster Creek were reorganized with the MCF department replaced by several different organizations. The maintenance responsibilities of MCF were assumed by the new maintenance department. This maintenance department now reports directly to the Vice President and Director, Oyster Creek, as opposed to the MCF department reporting to a corporate vice president. The plant material department was eliminated and responsibilities transferred to other departments. The responsibility of preventive maintenance program scope, definition as well as implementation, was given to maintenance. The responsibilities for predictive maintenance once performed by plant material were given to plant engineering.

(Open) (observation) page 38. Licensee needs to clarify the FSAR with regard to commitments to Regulatory Guides.

Update 6 to the FSAR dated December 1991 clarified Oyster Creek's commitment to Regulatory Guide 1.105. However, the licensee has yet to review the FSAR regarding how commitments to other Regulatory Guides are documented. The licensee has stated that a proposed change to the FSAR will be initiated to clarify the Regulatory Guide commitments in the FSAR. This issue will be reviewed when the proposed change is initiated and evaluated.

This observation will remain open and will be tracked as open item 50-219/92-14-02.

(Closed) (observation) page 41. The disposition of certain hydraulic control units (HCU) bolting discrepancies were not well documented and that better documentation and traceability of the disposition of various evaluations as they are performed would lead to more efficient resolutions of questions in the future.

For the HCU issue, it was subsequently determined that the 1986 evaluation of the mounting discrepancies in question was an analysis of the existing conditions only. Since the repairs were performed at a later time, the disposition of these repairs was not included in the 1986 evaluation. These repairs were documented in drawings 3D-225-38-100, 3D-225-38-1001, and installation spec OC-IS-323-408-001.

Deficiencies or discrepancies which render the quality of structures, systems, components, equipment, or material unacceptable or indeterminate are documented by the licensee on material nonconformance reports (MNCRs). During the engineering evaluation of these MNCRs, the need for safety evaluations, fire hazard analysis, and design verifications are determined and noted on the MNCR. If one of these evaluations is required, it must be performed and will be referenced on the MNCR. In addition, any technical corrective actions required to assure that all documents reflect final MNCR resolution are identified on the form. These MNCRs are entered into the licensee's computer data base upon final disposition. MNCRs for each system can be retrieved and the deficiency disposition then determined from the forms.

(Closed) (observation) page 43. If management insists on a more universal understanding of their philosophy and offers consistent support, certain problems can be eliminated.

(Closed) (observation) page 43; Management goals are not as well understood at lower levels of the organization; promulgation of these goals to the lower levels of the organization could better aid in their accomplishment.

The Vice President and Director, Oyster Creek, has been conducting "all hands" meetings with site personnel to review the status of ongoing issues and discuss corporate policy. These meetings are routinely held two to three times a year and before and after major plant evolutions such as refueling outages. Procedural compliance has been discussed at these meetings with emphasis placed on the need to stop an activity and, if necessary, modify a procedure prior to continuing.

In early 1988, the Oyster Creek Management Team developed the 1988 Oyster Creek Nuclear Generating Station Objectives and Goals. These goals and objectives were presented to and discussed with first line supervisors during March and April of 1988. Included in these station goals and objectives were expectations and standards for all employees. In addition to this, GPUN management down through first line supervisors were presented with and discussed the GPU Nuclear 1988 Corporate Objectives and Goals during management meetings in May 1988. The development and presentation to employees of the station and corporate goals and objectives has continued on an annual basis.

The operations department initiated a plant operations self assessment (POSA) program in early 1989 to identify weaknesses and establish corrective actions to such concerns as procedural compliance, personnel error, and the promulgation of management goals to lower levels of the organization. Corrective actions included instilling an error free mentality in all personnel and establishing self-checking training for operators. Operations standards have been developed, outlining management expectations for such topics as procedural compliance, equipment monitoring, log keeping, and tours. Operations management holds weekly meetings with shift crews to discuss plant status, operator concerns, and plans for continued improvement in the conduct of operations.

(Closed) (observation) page 44. It would be prudent to have continuing feedback from reviews of the long term programs to determine what additional real time improvements need to be applied until each new program is fully and successfully implemented.

Management meetings are conducted every other week at Oyster Creek, during which the status of ongoing programs such as PREP and implementation of POSA corrective actions may be updated and reviewed. These meetings are held at both the department manager level and with the Vice President and Director and his staff.

8.0 EXIT MEETINGS (40500,71707)

8.1 Preliminary Inspection Findings

A verbal summary of preliminary findings was provided to John Barton, Vice President and Director Oyster Creek, and other senior licensee management on July 27, 1992. During the inspection, licensee management was periodically notified verbally of the preliminary findings by the resident inspectors. No written inspection material was provided to the licensee during the inspection. No proprietary information is included in this report. The inspection consisted of normal, backshift, and deep backshift inspection; 36 of the direct inspection hours were performed during backshift periods, and 14 of the hours were deep backshift hours.

8.2 Attendance at Management Meetings

Mr. A. Randolph Blough, Branch Chief, NRC Region I Projects Branch No. 4, and Mr. John F. Rogge, Section Chief, NRC Region I Projects Section No. 4B, visited the Oyster Creek site on June 12, 1992. After a tour of the site with the NRC resident inspectors, a presentation was made by GPUN to familiarize Messrs. Blough and Rogge with the site and to provide the licensee's self-assessment of performance in various functional areas.