U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.	92-23
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:	November 3, 1992 - December 14, 1992
Inspectors:	Dave Vito, Senior Resident Inspector John Nakoski, Resident Inspector Suresh Chaudary, Senior Reactor Engineer, DRS

Approved By: John F. Rogge 12/24/92
John Rogge, Section Chief, Reactor Projects Section 4B — Date

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. One non-cited violation dealt with the entry of two contract workers into the drywell without proper dosimetry.

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EXECUTIVE SUMMARY

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Plant Operations

Overall, the plant was operated in a safe manner. Operators performed well in accomplishing a slow, controlled shutdown of the plant for the 14R refueling outage. The shutdown had minimal impact on the intake canal fish population due to the gradual rate of discharge canal temperature reduction. Operations personnel were cognizant of current outage activities and have maintained compliance with prescribed risk management guidelines for plant outage configurations.

Radiological Controls

Two contract workers entered the drywell without proper dosimetry (thermoluminescent dosimeters). However, the exposure of both workers was monitored by self-reading and alarming dosimeters. GPUN took appropriate corrective actions.

Maintenance/Surveillance

An Unusual Event (UE) was declared at 3:05 a.m. on November 15, 1992, after failure of a secondary containment leak rate test. After commencing a controlled plant shutdown in accordance with the technical specifications, GPUN repaired several degraded seals on the reactor building air lock doors and railroad bay door. The UE was terminated at 1:53 p.m. on November 15, 1992, after successful completion of a subsequent secondary containment leak rate test.

The licensee's efforts in the coordination and control of initial 14R refueling outage activities have been good. Reactor vessel disassembly work has been accomplished quickly and effectively. The mechanical stress improvement of the isolation condenser piping welds was well controlled and near completion by the end of the inspection period. GPUN appropriately handled the investigation of the December 2, 1992, failure of the crane at the intake structure and promptly returned the crane to service to support outage activities.

Engineering and Technical Support

GPUN developed an appropriate short-term operator aid to address a design deficiency related to the fuel zone reactor vessel level indication. Use of the operator aid was not necessitated prior to the plant reaching cold shutdown on November 28, 1992. GPUN will assess this issue in detail during the 14R outage and implement a permanent resolution prior to restart.

On November 15, 1992, the licensee physically located the leak in the underground 20" carbon steel service water supply line. GPUN will inspect the piping for degradation and replace the cracked portion of piping with a spool piece when the service water system is taken out of service during the 14R outage. Following review of two completed design basis documents, the inspectors found licensee's methods for tracking and resolving issues identified through the design basis reconstitution process to be appropriate.

Emergency Preparedness

The November 15, 1992, UE resulting from the secondary containment leak rate test failure and the December 11, 1992, due to high tides at the intake structure were properly classified by control room personnel. Offsite notifications were timely and the licensee's actions in response to the UE conditions were appropriate. Also, GPUN conservatively remained in the high tides UE for more than two days.

Physical Security

A security officer at the drywell access point alertly noted that a worker had another worker's thermoluminescent dosimeter and promptly notified radiological controls personnel.

Safety Assessment and Quality Verification

A review of recent QA/QC monitoring documentation demonstrated more use of performance based observation techniques

DETAILS

1.0 OPERATIONS (71707,93702,94703)

1.1 Operations Summary

The inspection period began with the plant operating at full power. Several power reductions were made to shuffle control rods to allow the lowering of recirculation flow while maintaining full power operations. The standby gas treatment system (SGTS) was tested on November 14, 1992, to demonstrate that an appropriate negative pressure could be maintained in secondary containment. The required vacuum could not be established and the licensee commenced a reactor shutdown. An Unusual Event (UE) was declared at 3:05 a.m. on November 15, 1992, after the group shift supervisor (GSS) determined that secondary containment could not be returned to within the technical specification required negative pressure within eight hours of initiating the shutdown. SGTS was successfully retested after repairs were made to the seals on the reactor building airlock and railroad bay doors and the UE was terminated. See section 3.1 for additional information. On November 18, 1992, the unit commenced an end of core life coastdown. While not all of the control rods were fully withdrawn, the licensee determined that power reductions required to further withdraw the control rods would not have improved the overall plant capacity factor. The 14R refueling outage started on November 28, 1992. Reactor power was at about 96% when the licensee commenced the plant shutdown on November 27, 1992. See section 3.2 for discussion of 14R refueling outage status.

On December 9, 1992, the licensee identified a hole in the service water line downstream of the reactor building closed cooling water (RBCCW) system heat exchangers. The hole was at the inner arc of a piping elbow located just inside the reactor building. The piping configuration produced a low pressure area in the pipe elbow causing air to be drawn into the line, rather than causing service water to flow out of the hole. The licensee developed a plan to inspect other service water elbows and tee connections downstream of the RBCCW heat exchangers. The service water piping upstream of the heat exchangers has been subject to periodic hydrostatic tests and was, therefore, excluded from the licensee inspection plans. A temporary patch has been placed over the hole. The licensee plans to permanently repair the failed elbow during the current refueling outage.

A UE was declared at 9:50 a.m. on December 11, 1992, due to high tides at the intake structure. The licensee remained in the UE condition until 9:15 p.m. on December 13, 1992. Also on December 11, 1992, a reactor scram signal was generated when the control room operator moved the reactor mode switch from the shutdown position to the refuel position. GPUN reported this engineered safety feature (ESF) actuation to the NRC. The mode switch circuitry has a two second time delay that allows the mode switch to be taken out of shutdown and placed into refuel without causing a scram signal. It appears there was a delay in the makeup of the refuel position contacts that caused the two second timer to time out inputting a scram signal to the reactor protection system. The licensee is evaluating the cause of the event.

Overall, the licensee continued to operate the unit in a safe and conservative manner.

1.2 Reactor Shutdown for 14R Refueling Outage

On November 27, 1992, at 9:00 a.m., the licensee started a plant shutdown to begin the 14R refueling outage. Before the shutdown began, the licensee had developed a power descension plan to limit the change in discharge canal temperature to about 1°F/hr and to minimize the crud (radioactive corrosion and wear products) burst associated with a quicker reactor shutdown. The power descension plan provided the group shift supervisors (GSS) with detailed guidance that established a shutdown rate to about 30 MWe/hr and a reactor coolant system cooldown rate to the cold shutdown condition to about 25°F/hr.

As a result of the slow and deliberate shutdown and cooldown, the impact on the fish population in the discharge canal was minimal. No noticeable fish kill was observed during or following the shutdown. Transportation of crud in the reactor coolant system was minimized as noted by the minimal changes in radiation readings on the reactor recirculation system piping. While decreasing reactor power, reactor water cleanup (RWCU) system flow was maintained as high as possible to further reduce the radioactive material in the reactor coolant.

The inspector observed control room operators (CRO) reduce reactor power from about 96% to about 60% on November 27, 1992. The GSS and CROs were observed to refer to the detailed power descension plan provided by operations management during power reduction. The initial shutdown rate was controlled at 30 MWe; however, about 1:30 p.m. on November 27, 1992, the shutdown rate was increased to about 35 MWe. The licensee increased the shutdown rate after ensuring that the decrease in discharge canal water temperature was less than 1°F/hr. Actions recommended by the power descension plan were taken when the proper plant conditions were established. Good support was provided to the CROs by core engineering personnel while power was being reduced. Cold shutdown conditions were achieved at 3:33 p.m. on November 28, 1992. No significant difficulties were noted by the licensee or the inspector during the shutdown. Overall, the actions observed by the inspector during the shutdown were well controlled and efficiently conducted.

1.3 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
- intake area
- reactor building
- turbine building
- vital switchgear rooms

- old radwaste building
- transformer yard
- access control points
- drywell

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.

On November 22, 1992, during a heavy rain, the inspectors observed the leak tightness of the compartments that house the emergency diesel generators (EDG) and related switchgear. The EDG building is a seismic concrete structure designed to protect the EDGs against wind and tornado loads, flooding, externally generated missiles, and seismic events. However, the roof of the EDG building contains a 10 ft by 16 ft opening to facilitate ambient air intake to the diesel engines. As such, rainfall does fall through the opening onto portions of the EDG enclosures and EDG switchgear compartments within the structure. The inspectors found no signs of water intrusion in either the EDG enclosures or the switchgear compartments. The EDG building drains were also functioning properly.

On December 1 and 2, 1992, the inspectors toured the drywell. The licensee was in the process of installing scaffolding and shielding, and removing insulation before drywell work could start. Based on the inspectors' observations, good controls have been established to process personnel into and out of the drywell through the drywell access on the 23 ft elevation of the reactor building. GPUN has provided valets to help personnel put on their protective clothing before entering the drywell and to assist in the removal of protective clothing when exiting the drywell. This has improved the consistency with which protective clothing is used and has helped minimize skin and clothing contaminations that occur due to poor suit up or unsuiting practices. Security control for access to the drywell was good.

During the drywell tours, the inspectors noted that work was being performed efficiently. Work efforts appeared to be well coordinated. The inspectors concluded that the licensee has effectively coordinated initial outage work in the drywell.

The inspectors continued their inspection of scaffolding installed at various locations in the plant in support of refueling outage work. On one occasion prior to shutdown for the outage, the inspectors questioned whether the licensee had considered the effect of scaffold boards on the ability of sprinkler heads and deluge nozzles to achieve their appropriate water spray pattern when activated. Particularly, the inspectors noted scaffold boards that passed between cable trays in the overhead above the reactor building closed cooling water (RBCCW) heat exchangers on the 51 ft elevation of the reactor building. In response, operations and fire protection engineering personnel performed a walkdown of installed scaffolding to assess the effect of scaffold boards on fire suppression equipment. Operations management concluded that the installed scaffold boards did not appear to directly affect fire suppression equipment spray patterns. However, the scaffold boards above the RBCCW heat exchangers were removed as a precautionary measure. Operations management also acknowledged that the existing guidance in procedure 105.2, "Scaffold Erection and

Walkthrough Documentation," was not very prescriptive and left the criteria for scaffold inspection up to the individual performing the inspection. By the end of the inspection period, the licensee had developed a temporary change to procedure 105.2 that added a scaffold inspection checklist to be used by operations department personnel when inspecting completed scaffold installations. The inspector reviewed the checklist and concluded that it provided for a comprehensive assessment of the effect of installed scaffolding on nearby equipment. This checklist was to be permanently included in the procedure. Installation of scaffolding in the drywell in support of initial outage work has been performed well.

The inspectors also noted a significant amount of fire retardant treated wood used in the scaffolding installation. The inspectors questioned the licensee about the additional fire loading that could be attributed to the large amount of fire retardant treated wood in the plant. In response the fire protection engineer stated that procedure 120.5, revision 4, "Control of Combustibles," did not require fire retardant wood to be considered in heat load calculations. When asked to discuss the technical justification for this procedural guidance, the fire protection engineer did not have a clear understanding of the basis. The inspectors reviewed the National Fire Protection Association (NFPA) Fire Protection Handbook, Fifteenth Edition, Section 11, Chapter 6, and procedure 120.5, revision 4, stated that fire retardant treated wood can prevent the spread of flame from the immediate area with a substantial reduction in fuel and smoke contribution. This appeared to support the procedure statement not requiring fire retardant wood to be considered as a transient combustible.

A concern was raised during a tour of the 95 ft elevation of the reactor building with fire barrier for cable penetrations near the northwest corner. Several cable penetrations in the floor did not have any fire barrier material, while others did. The inspectors questioned the fire protection engineer on the need for fire barrier material in these penetrations. The fire protection engineer stated that the 95 ft elevation floor was not considered a fire boundary, but was a division between two fire zones and, therefore, no fire barrier material was required in the cable penetrations between the 75 ft elevation and 95 ft elevation. A fire area consists of a building area separated by rated fire barriers from other building areas, while a fire zone is a subdivision of a fire area in which the fire suppression systems have been designed to combat particular types of fires. As such, fire zones do not require a rated fire barrier between locations. The inspector reviewed the fire protection plan and fire hazards analysis and found that the 95 ft elevation and 75 ft elevation were separate fire zones in one fire area.

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 Drywell Entry Without Proper Dosimetry

On December 7, 1992, the licensee identified that two contract (Catalytic) carpenters had entered the drywell without the proper personal dosimetry. One worker did not have a thermoluminescent dosimeter (TLD) and the second had the TLD of the first worker. Based on the licensee's review, the second worker inadvertently picked up the first worker's TLD before entry into the drywell. The first worker failed to remove his TLD from his security badge before putting protective clothing on. A security guard at the drywell access noted that the TLD of the second worker was on the first worker's badge at the drywell entrance and notified radiological controls (radcon).

When notified, radcon contacted both workers, had them check for dosimetry and exit the drywell. Both workers have been restricted from radiologically controlled areas (RCA) until the issue has been resolved. Both workers were wearing 0-200 mRem and 0-500 mRem self reading dosimeters (SRD) and alarming dosimeters. Based on the 0-200 mRem SRD, the first worker had received a 23 mRem exposure and the second worker had received 110 mRem. The alarming dosimeters read 36 mRem and 106 mRem respectively. Both workers' total exposure remained within the Oyster Creek administrative exposure limits

Radcon issued two dosimetry investigation reports (DIR 92-117 and 92-118) documenting the event. Immediate corrective actions were to have both workers exit the drywell and restrict their access to the RCA. Additional corrective actions were to brief both licensee supervisors and employees and contractor supervisors and employees on the need to be aware of dosimetry requirements. Also, the licensee has directed that the valets assigned to assist personnel entering the drywell to check the workers for proper dosimetry.

The inspector reviewed DIRs 92-117 and 92-118, reviewed procedure 9310-ADM-4241.07, revision 7, "Personnel Dosimetry Requirements," discussed the event and corrective actions with radcon personnel, and reviewed the drywell access controls. Sections 7.2.2 and 7.3.1 of procedure 9310-ADM-4241.07 require that personnel entering the RCA shall wear TLDs at all times while in the RCA. The access controls established for entry into the drywell have radcon personnel question the workers at two distinct times to ensure the proper dosimetry is being worn before entry into the drywell is made. The DIRs accurately reflect the events as they occurred and document the corrective actions and worker exposures.

The inspector concluded that the licensee had adequately documented the event and performed appropriate corrective actions. It did not appear to the inspector that the TLDs were intentionally swapped. The significance of the event was minimized since the workers' exposure had been monitored by both the 0-200 mRem SRDs and the alarming dosimeters. Based on the corrective actions, the limited significance of the event, and the licensee's performance in the radiological controls area, the violation of procedure 9310-ADM-4241.07, section 7.2.2 and 7.3.1, is not being cited as allowed by 10 CFR Part 2, Appendix C (1992).

3.0 MAINTENANCE/SURVEILLANCE (62703,71707)

3.1 Unusual Event Declared due to Secondary Containment Leak Rate Test Failure

On November 14, 1992, during the performance of Procedure 665.5.002, "Secondary Containment Leak Rate Test," GPUN found that the technical specification requirement for secondary containment integrity was not being met. After initiating one train of the standby gas treatment system (SGTS), only 0.211 inches of water vacuum could be maintained in the reactor building. Technical Specification (TS) 4.5.K requires that at least 0.25 inches of water vacuum be maintained with an SGTS filter train flow rate of not more than 4,000 cubic feet per minute. After recording the test failure at about 5:30 p.m. on November 14, 1992, control room operators entered TS action statement 3.5.B.1.1 which allows four hours to restore secondary containment integrity or bring the reactor mode switch to shutdown within the following 24 hours. The operators commenced a controlled plant shutdown at 8:45 p.m. on November 14, 1992.

After several reactor building walkdowns, the licensee determined that the most likely cause of reactor building leakage was degradation of the seals around the reactor building personnel airlock doors and the reactor building railroad bay doors. Efforts were then focused on replacing those seals that appeared to be degraded on the airlock doors and the inner railroad bay door. Although repairs were in progress, an Unusual Event (UE) was declared at 3:05 a.m. on November 15, 1992, in accordance with Oyster Creek Emergency Action Level Category N, after the group shift supervisor (GSS) made a determination that secondary containment would not be returned to the TS required negative pressure within eight hours of initiating the shutdown. The door seal repairs continued until about 10:30 a.m. when another secondary containment leak rate test was performed. With both railroad bay doors closed, 0.274 inches of water vacuum were maintained. With the inner railroad bay door closed and the outer railroad bay door open, 0.264 inches of water vacuum were maintained. With the inner railroad bay door open and the outer railroad bay door closed, 0.241 inches of water vacuum were maintained. Based on these test results, GPUN locked closed the inner railroad bay door and declared that secondary containment integrity had been reestablished. The UE was terminated at 1:53 p.m. on November 15, 1992, with the plant at about 76% power. GPUN then continued its effort to reduce the reactor building leakage by replacing the seals on the outer railroad bay door. On November 16, 1992, a successful secondary containment leak rate test was conducted with the inner railroad bay door open and the outer railroad bay door closed.

The inspectors concluded that the licensee responded promptly to the degraded secondary containment condition and took appropriate actions to resolve it. The Unusual Event condition was properly classified by the operations shift crew, and the event notifications were timely.

3.2 14R Refueling Outage

On November 28, 1992, the licensee started the 14R refueling outage. The major activities of this outage include refueling, maintenance of the high pressure and A low pressure turbines, drywell corrosion mitigation, hydraulic control unit (HCU) repair, control rod drive (CRD) replacement, incorporation of the station blackout (SBO) and primary containment hardened vent modifications, mechanical stress improvement of the isolation condenser welds, and a large number of valve and valve operator maintenance activities.

The licensee has done more preparation work for the 14R refueling outage than for any previous outage. All procedural instructions for each planned outage activity were developed before the outage began. Coordination between the various onsite and contract organizations performing work during the outage has been good. Use of the plant risk management configurations has provided clear guidance as to what equipment and systems are required to remain available for a specific time in the outage. Use of an outage coordination center by plant operations has minimized the number of personnel requiring access to the control room.

The inspectors attended various outage planning, coordination, status, and scope control meetings. Inspectors observed activity in the control room and around the plant during normal plant tours. Overall, the outage has progressed well. No significant problems have been identified with the control of outage activities. The licensee has remained committed to their risk management plan. This was specifically demonstrated on one occasion when the licensee maintained the control rod drive system in service as the second source of reactor vessel water supply while feedwater check valve local leak rate testing was completed, even though this delayed maintenance work on the HCUs.

The inspectors concluded that the licensee's early efforts in the coordination and control of the 14R refueling outage activities have been good.

3.3 Reactor Vessel Head Detensioning

On December 1, 1992, the inspector observed the detensioning of the reactor vessel head. GPUN has purchased a carousel fixture from ABB Atom that allows the simultaneous use of four stud detensioners to reduce the time required to perform reactor vessel head detensioning. Use of the carousel also allowed the refuel floor overhead crane to be available for other operations during reactor vessel detensioning. General Electric (GE) personnel operated the stud detensioning equipment.

Preoperational testing was performed on the Biach tensioner units used to detension the reactor vessel studs before placing the tensioners on the carousel. When the tensioners were installed, the licensee identified the need to modify the hydraulic lines to clear interferences with the rigging and air driven operators. A minor delay was encountered when retesting the tensioners after modifying the hydraulic lines due to leaking fittings. The inspector observed portions of the retesting and corrective actions to repair the leaking hydraulic lines. The licensee had taken the necessary steps to ensure personnel safety while pressurizing the hydraulic lines. Overall, the retesting and repairs were adequately controlled. While working on the Biach tensioner units, the licensee proceeded with other work on the reactor disassembly schedule.

The reactor vessel detensioning proceeded very well. Use of the carousel improved the ability to setup and detension the reactor vessel studs. In addition to the carousel, the licensee had installed air driven operators to the stud detensioners to allow quicker setup. The GE crew worked with the equipment very well, demonstrating good coordination between the work director and the workers installing the Biach stud tensioners.

The inspector concluded that the reactor vessel disassembly work had been controlled well. The licensee's purchase of the carousel and the installation of the air driven operators to the Biach tensioners demonstrated their desire to minimize the exposure received during reactor vessel disassembly. The new equipment effectively cut the time to detension the reactor vessel head in half. Performance of other work in the reactor vessel disassembly schedule while repairing the tensioners demonstrated the licensee's commitment to complete the 14R refueling outage on schedule in a safe manner.

3.4 Mechanical Stress Improvement Process on Isolation Condenser Welds

GPUN has contracted O'Donnell & Associates, Inc., to perform a mechanical stress improvement process (MSIP) on the isolation condenser pipe welds performed during the last refueling outage. The MSIP is the second mitigation technique used to address concerns with intergranular stress corrosion cracking (IGSCC). The piping installed during the 13R refueling outage was IGSCC resistant material.

MSIP is a process that uses a clamp to compress the pipe material near the area of the weld. By compressing the pipe material, the weld area is changed from being in tension to being in compression. By changing from tension to compression, a condition required to propagate IGSCC flaws has been eliminated. To determine when the adequate compression has been obtained, the pipe circumference is measured before MSIP and again after MSIP. The difference is then compared to a specification determined before the process is applied. Typical changes in circumference range between 1/4 inch to 3/4 inch.

The inspectors observed installation of the MSIP clamps, discussed the process with licensee and O'Donnell personnel, and observed mechanical stress improvement on an isolation condenser weld. Clearly established circumference change specifications were provided at the job site. The O'Donnell workers appeared familiar with the installation and use of the MSIP clamps. Work on the MSIP for the isolation condenser welds was proceeding well.

The inspectors concluded that the use of the MSIP on the isolation condenser welds was being well controlled and efficiently performed.

3.5 Intake Structure Crane Failure

At 10:15 a.m. on December 2, 1992, a cable used to raise and lower the 130 ft boom of a mobile crane at the Oyster Creek intake structure failed. The boom was raised to about a 70 degree angle from the ground surface at the time it fell. The top of the boom struck the redundant fire pump house structure and also damaged the discharge and recirculation lines of the redundant fire pump. The redundant fire pump and motor and the 350,000 gallon redundant fire water tank were not damaged. Although the redundant fire water pump and tank are not safety-related, they provide a convenient means of backup fire suppression capability should the main fire pumps be unavailable. Specifically, Technical Specification 3.12.B.3.a requires that in the event of failure of the main fire pumps, backup fire suppression capability shall be provided within 24 hours. GPUN is expediting its efforts to repair the redundant fire water pump system. No personnel injuries occurred as a result of the crane failure. No damage to safety-related or important to safety equipment occurred. No load was being carried by the crane hook when the event occurred.

After the crane failure, GPUN brought in an independent investigator from the Crane Investigation and Certification Bureau (CICB). The CICB investigator concluded that during some period of operation between the last monthly crane inspection on November 18, 1992, and the operation of the crane on December 2, 1992, the load must have been taken off the boom (i.e., by touching the top of the boom to the ground), causing a mislay of the cable on the cable drum. Crane operator inspections of the crane before use on November 20, November 21, and December 2, 1992, did not identify the cable mislay. The cable mislay caused deformation of the cable and ultimately caused the cable failure.

After the CICB investigation, the crane boom and cable were replaced and the crane was recertified. By December 7, 1992, the crane was recertified for use. The inspectors concluded that GPUN appropriately handled the investigation of this event and promptly returned the crane to service to support outage activities.

4.0 ENGINEERING AND TECHNICAL SUPPORT (71707,37700,40500)

4.1 Fuel Zone Level Indication Concern

On November 19, 1992, GPUN identified a design deficiency related to the fuel zone reactor vessel level indication system after findings made during the testing of the site-specific simulator by Exitech, the vendor for the simulator nuclear steam supply system NSSS model. The problem dealt with possible errors in observed level indication after the fuel zone level instrument switches from its narrow range to its wide range detectors. After an initial engineering evaluation, GPUN provided guidance to the control room operators which described the problem and the potential magnitude of the indicated level errors. A graph of indicated wide range fuel zone vs. actual level, assuming the worst case error, was provided to the control room operators for use as an operator aid until the plant was placed in cold shutdown for the forthcoming refueling outage. Use of the operator aid was not necessitated prior to the plant reaching cold shutdown on November 28, 1992. GPUN has stated that this issue will be assessed in detail during the 14R outage and will be resolved prior to restart.

Background

The fuel zone level instrumentation gives indication of reactor vessel water level within the core shroud. This instrumentation was added to Oyster Creek after a May 2, 1979, incident during which the core was nearly uncovered. The incident occurred because false level indications were being supplied by the existing level instrumentation due to the reactor vessel configuration and the location of the level instrumentation. Specifically, the Oyster Creek reactor vessel is configured with no connection between the annulus outside the core shroud and the below core region and the existing level instrumentation measured level only in the annulus region. The fuel zone level indication system was installed to resolve this problem and to provide a more representative indication of the status of core cooling. This instrumentation provides no control function.

The system consists of four narrow range Rosemount detectors in conjunction with four wide range Rosemount detectors which measure level from -150 inches to +180 inches TAF (top of active fuel). Four of these detectors also supply indication to the remote shutdown panel over the same level range. The level transmitters sense differential pressure between the reference leg and variable leg and convert that signal to a dc signal proportional to vessel level. The narrow range instrument variable leg senses level at the core spray spargers (+55 inches TAF). The wide range instrument variable leg senses level at the liquid poison nozzle (-144 inches TAF). Both the narrow and wide range fuel zone level instruments use the same cold reference leg used by the GEMAC narrow range level instruments, which provides annulus level input to the feedwater control system. The fuel zone level instruments are temperature and pressure compensated based on saturated water and steam conditions in the reactor vessel and the instrument impulse lines to detect flashing. Fuel zone level indication is part of Oyster Creek's Regulatory Guide 1.97 instrumentation.

The fuel zone level instrumentation is operational when the following three conditions are satisfied: 1) the reactor recirculation pumps are tripped; 2) water level is within the range of the instruments; and, 3) there is no flashing detected in the impulse lines. If level is greater than +55 inches TAF and the core spray system parallel isolation valves are closed, the narrow range instrument provides level indication. If level is less than +55 inches TAF or the core spray system parallel isolation valves are open, the wide range instrument provides level indication. If a saturation condition (flashing) is detected in the reference leg, the instruments will turn off.

While fuel zone level is within the range of both the wide and narrow range instruments, a bias (K factor) is applied to the measured wide range level to compensate for instrument inaccuracies caused by physical differences in the size and fluid characteristics of two level columns. The K factor, which is the ratio of the measured narrow range level to the measured wide range level, is multiplied times the measured wide range level. The result becomes the indicated wide range level. If the K factor is greater than 1.25, the fuel zone instrumentation logic assumes that the detectors are failed and turns the indication off. When the narrow range indication switches to the wide range indication, the K factor is fixed at whatever its value was prior to the switchover.

Problem Identified During Simulator Testing

During testing of the Oyster Creek site-specific simulator in October 1992, Exitech personnel raised concerns regarding the validity of the wide range fuel zone level indication provided to the control room operators due to the fixed K factor. After review of this issue by Technical Functions engineering, GPUN concluded that actual level could be ranging more or less and that the compensation provided by the fixed K factor, i.e., indicated level, could be above TAF with the core actually uncovered. The engineering evaluation noted that in large break LOCA conditions, K will be greater than 1.25 and the instruments will shut off. Without level indication, the emergency operating procedures (EOPs) direct the operators to initiate reactor pressure vessel flooding. As such, no corrective actions were determined to be necessary for a situation where the fuel zone level indication has shut off. However, GPUN determined that certain abnormal plant conditions could be postulated in which the wide range refuel zone level instrument will remain operational and that corrective actions to address these situations were necessary.

Corrective Actions

To address the period of time before the plant was to be shut down for the 14R refueling outage (about 1 week), the control room operators were provided with required reading material and an operator aid to clarify their actions in a condition where the wide range fuel zone level instrument was being relied upon for reactor level indication. The required reading material provided a good general description of the issue and provided an operator aid to be used to determine reactor level for EOP actions. The operator aid consisted of a curve of indicated wide range fuel zone level vs. actual level. The actual levels were determined assuming the worst-case fixed K factor (1.25) at the time of the narrow range to wide range switchover. The inspectors reviewed the required reading and the operator aid and found them to be of sufficient detail and clarity. The inspectors interviewed several licensed operators and found their knowledge of the issue and the use of the operator aid to be good.

GPUN has acknowledged that the operator aid is not an adequate long-term fix and has required that a permanent resolution to this issue be developed and implemented prior to restart from the 14R refueling outage. GPUN has designated a team of Technical Functions engineering personnel to review current system design information, related operator/EOP interface conditions, and accident/transient analysis to better define those conditions when the fuel zone level instrumentation will be functioning and in use by control room operators to determine EOP actions. This information will be used to establish updated design criteria and functional requirements for the fuel zone level instrumentation system. It will also be used to provide justification for use of the existing system or recommendations for system modification. GPUN also noted that the EOPs will be reviewed and changed, if necessary, to assure that the operators are provided with the best available information to determine reactor water level.

4.2 Design Basis Document Reconstitution

During this inspection period the inspectors reviewed the licensee's design basis document development efforts. The focus of this inspection was to review the licensee's methods for following up and tracking questions or concerns identified as the result of the design basis reconstitution. The inspectors reviewed two completed design basis documents (DBD) and the licensee's actions to respond to the questions identified during the DBD development. The inspector reviewed the containment spray system DBD (OC-DBD-241) completed in 1989, and the reactor coolant system DBD (OC-DBD-220) completed in 1990.

The containment spray DBD identified nine open items requiring further evaluation by the licensee. Most of the open items related to the single failure vulnerability of the containment spray automatic initiation logic. The auto initiation logic has been schedule to be removed during the 14R refueling outage and addresses the concerns identified by the DBD. In each of the open items the licensee appropriately reviewed the concern, developed corrective actions, and has incorporated the corrective actions into appropriate tracking programs to ensure the actions are completed.

Twenty-three open items were identified by the licensee while developing the DBD for the reactor coolant system. The most significant open items related to the calculations to support various system parameters such as reactor vessel vent and drain line sizes, recirculation discharge bypass line size, electromatic relief valve (EMRV) and safety valve (SV) capacity, and EMRV and tailpipe line flow. The licensee has assigned the responsible organization to develop the calculations to support the as-installed capabilities and resolve the other open items identified in the DBD development.

The inspector concluded that the licensee has appropriately addressed the questions and concerns identified during development of the DBDs reviewed. Overall, the program for DBD development has been used effectively to upgrade plant procedures, identify additional engineering information needs, and identify plant modifications needed to resolve DBD concerns.

4.3 Service Water Leak Repair At Condensate Transfer Pump House

In NRC inspection report 50-219/92-21 the inspectors documented the review of the repair efforts for the service water (SW) line leak near the condensate transfer pump house. During this review it was determined that the safety evaluation (SE-312400-016) for injection of a chemical grout to stabilize the surrounding soil did not adequately address the seismic effects of injecting the grout. As written, the safety evaluation addressed seismic effects by stating they were not a concern, but did not document the basis for this determination.

The inspectors discussed the safety evaluation with Technical Functions (TF) engineering personnel. During the meeting the TF engineers described the basis for considering the injection of the chemical grout not having an effect on the seismic response of the buried SW, emergency service water (ESW), and fire water system piping located near the area being excavated. The seismic response of the buried pipe was determined to be unaffected by the injection of the grout due to the relatively small length of pipe in the excavation area (about 2% of total pipe length), the small change in soil density between the grouted and non-grouted soil, and the conservative assumptions made in the original buried pipe analysis for the ESW piping.

The licensee agreed that the safety evaluation should have provided a basis for the determination that the injection of the chemical grout would have no effect on the seismic response of the buried pipe. A revision to the safety evaluation was made on November 18, 1992, which included the licensee's bases for determining that the seismic response of the buried pipes was not affected.

Based on the discussions with the licensee, review of revision 1 to safety evaluation SE-312400-016, and an inspection of the excavated area near the SW line leak the inspectors concluded that the licensee had appropriately considered the effects of injection of the chemical grout to stabilize the soil near the SW line leak.

On November 14, 1992, the licensee physically located the leak in the underground 20" carbon steel service water supply line. The leak was found in a horizontal section of piping, about 20 ft below the ground surface, close to a flanged elbow that directs the service water flow vertically. The leak was maintained at 10-20 gpm until 6:30 a.m. on November 16, 1992, when the exterior pipe coating in the area of the leak broke loose. The leak rate subsequently increased to between 30 and 50 gpm. GPUN took immediate action to contain the leak with a banded clamp.

The pipe defect was observed to be a four to five inch longitudinal crack near the flanged elbow connection. The crack did not appear to be within the heat affected zone of the weld that attaches the flange to the horizontal section of pipe. The current plan for repair is to cut out the portion of the service water piping that contains the crack and replace it with a spool piece. The repair will not be made until the service water system is taken out of service in the 14R refueling outage. At that time, GPUN will also perform visual inspections and ultrasonic testing (UT) on the exposed portion of the service water piping. Following visual inspection, UT will be done on the areas of the piping that displays degradation of the interior wall coating.

5.0 EMERGENCY PREPAREDNESS (71707)

5.1 Unusual Event Due to Higher Than Normal Tides

On December 11, 1992, 9:50 a.m., GPUN declared an unusual event (UE) when the intake water level increased to greater that 4.5 feet above mean sea level (MSL). The maximum intake level noted was about 5.5 feet above MSL. The entry condition for declaring an alert due to high intake water level is when water level is at the intake grate level (about 6.0 feet above MSL). The UE remained in effect until 9:15 p.m. on December 13, 1992, when the licensee concluded that the water level was remaining consistently below 4.5 feet above MSL. Water level dropped to about 3.25 feet.

The higher than normal tides were caused by a severe winter storm, with strong winds that forced water into Barnegat Bay and the Oyster Creek intake canal. No significant equipment problems were encountered due to the high intake level. Service water, emergency service water, and condensate pumps in the intake area were not affected by the high water level. Of more significance during the storm were the high winds encountered early on December 11, 1992. The maximum sustained winds recorded by the Oyster Creek meteorological tower were 34 mph at the 33 ft elevation and 64.6 mph at the 380 ft elevation at 5:45 a.m., December 11, 1992. Using a strip chart recorder for the 380 elevation a maximum wind gust of 98 mph was observed around the time maximum sustained winds were observed. The high winds did not cause significant damage to the facility. Minor damage was caused to exterior shell of the reactor building upper elevation. No damage to the secondary containment boundary occurred. Later on December 11, 1992, the winds subsided and did not threaten any site structures.

The structural integrity of the upper elevation of the reactor building (119 ft elevation) has been under review by the NRC. Based on the NRC review the existing roof framing of the upper portion of the reactor building is likely to be structurally unstable at tornado-wind velocities in excess of 102 mph. NRC review of the structural integrity of the upper portion of the reactor building was documented in a safety evaluation attached to a December 7, 1992, letter to the licensee. While the peak wind gust during the storm was about 98 mph, the maximum sustained wind observed (64.6 mph at the 380 ft elevation) was well below the velocity at which the reactor building is likely to be structurally unstable. The licensee has committed to improved the structural integrity of the upper portion of the reactor building during cycle 14 (1992 - 1993 time frame) in response to NRC concerns with the existing structural design.

During the storm the licensee entered abnormal procedures 2000-ABN-3200.31, "High Winds," and 2000-ABN-3200.32, "Response to Loss of Intake," to respond to the changing environmental conditions. The inspectors discussed the licensee's actions in response to the severe winter storm and reviewed emergency procedure implementing procedure (EPIP) OC-01, revision 1, "Classification of Emergency Conditions." Based on the EPIP, the licensee appropriately classified the high intake level at the UE level. While strong gusts were encountered, sustained winds were lower than the 74 mph entry level conditions for a UE due to high winds. The inspectors found no significant damage to the buildings on site. Minor damage to the exterior of the reactor building was observed, but did not affect the integrity of the reactor building.

6.0 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 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.

7.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (40500,92700)

7.1 Quality Assurance/Quality Control Performance Based Review

The licensee has started a program to place more emphasis on performance based observational techniques during quality control (QC) and quality assurance (QA) monitoring and auditing activities. To assess the licensee's efforts in this area, the inspectors reviewed QC/QA monitoring reports for July through September 1992, and discussed the efforts in this area with the licensee.

Overall, the monitoring reports reviewed clearly described the purpose, conclusions, and observations made during QA/QC monitor activities. The inspector noted a change in observational methods between August and September 1992. The September 1992 monitoring reports reflected a hands-on, performance based approach to QA/QC monitor activities. QA/QC monitors accompanied equipment operators, radwaste operators, and chemistry technicians while they were performing various tours and tasks. In addition, the QA/QC monitors also reviewed plant records for the periods and areas covered during their observations to ensure that the required information was correctly recorded, and independently verified a number of plant parameters recorded in plant logs. Deficiencies identified during the QA/QC monitor observations were adequately identified in the monitoring reports and were transmitted to the appropriate plant manager/supervisor for corrective action.

The inspector concluded that the licensee's initial efforts to implement a QA/QC monitoring program that uses more performance based observational techniques have been good. The licensee was investigating available training material in performance based observational techniques to improve QA/QC monitor capabilities. The NRC will continue to observe licensee progress in this area.

7.2 Review of Licensee Event Reports

NRC inspectors reviewed the following LER 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
92-012	Electromatic Relief Valve High Pressure Relief Setpoints Exceeded Technical Specification Limit Due to Drift

Based on inspector review of LER 92-012, no additional followup of the event was required. The LER accurately described the event as it occurred, past occurrences, and addressed appropriate corrective actions. The licensee has planned to replace the sensors with a more reliable and accurate device during the 15R refueling outage.

8.0 EXIT MEETING (71707)

8.1 Preliminary Inspection Findings

A verbal summary of preliminary findings was provided to the senior licensee management on December 15, 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; 15 of the direct inspection hours were performed during backshift periods, and 26 of the hours were deep backshift hours.