U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/84-34

Docket No. 50-219

License No. DPR-16 Priority -- Category C

Licensee: GPU Nuclear Corporation
100 Interpace Parkway
Parsippany, New Jersey 07054

Facility Name: Oyster Creek Nuclear Generating Station

Inspection At: Forked River, New Jersey

Inspection Conducted: November 18 - December 31, 1984

Inspectors: W. H. Bateman 2/6/85
W. H. Bateman, Senior Resident Inspector date

for E L Conner 2/6/85
J. Wechselberger, Resident Inspector date

W. Baunack 2/6/85
W. Baunack, Project Engineer date

Approved by: E L Conner 2/6/85
E. L. Conner, Chief, Reactor Projects Section 1A date

Inspection Summary: During this report period, the resident inspectors closely followed continuing startup activities and the licensee's efforts to return the electromatic relief valves (EMRV) to operation. Also, facility tours were conducted; the QA 1984 annual assessment was attended; and radiation protection, physical security, licensee event reports, surveillance testing, and periodic reports were reviewed.

One violation for failure to adhere to radiological procedures was identified. Also, the inspector developed some concerns with the large number of deficiency tags present in the control room and with some pipe supports.

The inspection involved 157 inspector hours.

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DETAILS

1. Review of Previous Inspection Findings

(Closed) Inspector Follow-up Item 84-32-02: The licensee seal welded the retainer to the valve cage on all five electromatic relief valves (EMRVs). This prevented steam from leaking past the retainer threads which allowed the pilot valve to vent the under disc chamber. On November 23, 1984, all five EMRVs successfully fulfilled the requirements of Procedure 602.4.003, Electromatic Relief Valve Operability Test.

(Closed) Violation 84-18-01: A contract guard carried a package into the protected area without the package being searched. When apprised of the situation, a site protection officer searched the package and discovered the contract guard's lunch. The personnel involved received a reprimand. A security directive was issued reminding all site security personnel of the requirement to search packages entering the protected area. The contract guard resigned the following day. The resident inspectors have not observed a similar situation during their routine tours.

(Open) Inspector Follow-up Item (84-18-02). The inspector reviewed all of the licensee's video tapes associated with the IRM/SRM drytube cracking problem and the analysis conducted by Technical Functions. The licensee replaced one fuel channel that was adjacent to a cracked dry tube; this fuel channel had exhibited cracking indications and will be further examined by the licensee to determine their cause. This item remains open, pending completion of the licensee's investigation.

The following open items from the Oyster Creek Outstanding Items List were reviewed and evaluated during this inspection period. For the most part these are old items which had not been documented in any previous inspection report. These items are being closed out for the following reasons.

(Closed) Inspector Follow-up Item 78-BU-2A (IFI), "A Typical Weld Material In Reactor Pressure Vessel Welds". This item is a duplicate of outstanding item 78-BU-12.

(Closed) Inspector Follow-up Item 78-SB-07 (IFI), "Check Drywell Temperature History for Adherence to Design Spec and to Temperature Capabilities of Electrical Cable". This item has been resolved by the licensee's responses to Bulletin 79-01B, Environmental Qualification of Class IE Equipment.

(Closed) Inspector Follow-up Item 78-35-03 (IFI), "Seismic Qualification of the Gould Batteries Used in Mod Are Required to be Quality Test Review Required". This item is closed by the licensee's responses to bulletin 79-01B, Environmental Qualification of Class IE equipment.

(Closed) Inspector Follow-up Item 78-35-04 (IFI), "125 VDC Battery C Mod Not Completed Required Further Inspection and System Testing". Routine inspections have shown the modification has been completed and the battery is in service.

(Closed) Inspector Follow-up Item 79-BO-22 (IFI), "Pipe Support Base Plate Design Using Concrete IEB 79-02, Revision 2". This item is a duplicate of outstanding Item 79-BU-02.

(Closed) Inspector Follow-up Item 79-SB-03 (IFI), "Examine Isolation Capability of Sensors/Probes in Event of Gross Fuel Failure (RE: RHR Conductivity Probe at PB Has Manual ISOL Only and Could Dump Hot Coolant". The facility being constructed in accordance with Part 50, Appendix A, Criterion 55 (Isolation of Instrument Lines) and the evaluations required by 0737 relative to leakage path out of the containment serve to close this item.

(Closed) Inspector Follow-up Item 81-LO-65 (LII), "Limitorque Operators For Two Isolation Valves For ISO Condenser System Found to Have Defects Which Affect Operability Of One Valve". This item is one of two Licensee Event Reports (LER) on the Outstanding Item List and is separately tracked by the residents, as are all other LER's.

(Closed) Inspector Follow-up Item 81-36-03 (LII), "Reactor Water Level Instrumentation For One Channel in Both RPS Systems Were Inoperable". Same as item above.

(Closed) Inspector Follow-up Item 83-02-01 (IFI), "Spent Fuel Pool Inventory Procedure to Control Items Placed In The Pool Excepting Fuel". This item is not identified in Inspection Report 83-02. This item was written following an inspection relative to an allegation dealing with a lost source in the spent fuel pool. An inspection conducted on January 2-4, 1985 shows the licensee has prepared a procedure (Procedure No. 1002.5, Fuel Pool Material and Inventory Control, Revision 0, September 21, 1983) governing items and materials which may be placed in the spent fuel pool and includes provisions for inventory control of all items stored in the pool.

2. Electromatic Relief Valve Testing

At the beginning of this report period the facility was shutdown, conducting electromatic relief valve (EMRV) repairs. The EMRVs had not successfully passed Procedure 602.4.003, Electromatic Relief Valve Operability Test.

On November 4, 1984, the "A", "C" and "D" EMRVs performed in accordance with the requirement of the valve operability surveillance, but "B" and "E" EMRVs failed to open. The licensee shutdown the reactor in accordance with the requirements of the Technical Specifications. The licensee found the pilot valve stem travel to be incorrect on the "A", "B" and "E" EMRVs and to be correct on the "C" and "D" EMRVs. The proper pilot valve adjustments were made and a reactor startup commenced.

The EMRV operability test was conducted again on November 9 and 10. The "A" EMRV operated as required. The "E" opened and appeared not to reseat completely as indicated by the downstream temperature device and the acoustic monitor. The "B" EMRV failed to open. On November 10, 1984 at 0200, a controlled reactor shutdown was commenced. The licensee initiated an investigation to determine the cause of the valve failures.

The licensee's investigation revealed that steam leakage past the retainer threads caused the valves to malfunction. The retainer threads into the valve cage and forms the lower chamber underneath the main valve disc. This lower chamber is vented by the pilot valve allowing system pressure to open the main valve. The licensee determined that seal welding the retainer to the valve would eliminate steam leakage past the retainer threads and thus allow the valves to function properly.

This problem has been experienced by other nuclear utilities and a similar solution of seal welding the retainer was used to resolve the problem. The licensee's Procedure 123, Operating Experience Assessment and Implementation, assigns the responsibilities of reviewing industry events to Plant Engineering. The function of the Plant Engineering group is to review events and recommend changes "to plant design and/or plant operation." The licensee conducted a complete review of industry events after the EMRVs failed to operate as required. This type of review is required by Procedure 123 on a continuing basis to benefit from industry events and to avoid experiencing similar problems. Had industry problems with EMRVs been evaluated prior to the EMRV overhaul conducted during the outage, two facility shutdowns possibly could have been prevented.

In addition to the steam leakage past the retainer threads, "E" EMRV failed to reseat properly. Upon investigating the "E" EMRV, the licensee found that seat angle on the body seal ring for the main disc was improper. The main disc was designed to seat at a 45 degree angle on the seal ring and not a 54 degree angle seal ring angle as found. This 54 degree angle prevented the "E" EMRV from reseating properly.

On November 23, 1984 the licensee performed Procedure 602.4.003. The "A" EMRV was tested properly but the "B" EMRV acoustic monitor responded, indicating approximately 20% after the transient had steadied out. The "C" EMRV was tested satisfactorily, but the "C" EMRV acoustic monitor did not respond. The licensee made a drywell entry to investigate the erroneous acoustic monitor responses. The licensee determined that one acoustic monitor cable was cut and that one acoustic monitor had lost the required sensitivity.

Five EMRVs were tested after the acoustic monitors were repaired and were declared operable after successfully passing the valve operability test.

3. Facility Tours

The inspectors toured various areas of the plant on a routine basis. Not all areas of the plant were accessible for inspection because of the plant's operating status. The control room was toured daily during which time log books were reviewed, equipment status was inspected, and shift turnovers were observed. In addition startup activities were monitored during the backshift. No unacceptable conditions were identified.

The inspectors observed that deficiencies existed with several alarms, instruments, and equipment items both in the control room and in the power plant and expressed their concern to plant operations management regarding the large quantity of deficiencies and the elapsed time since many of them were posted. Operations responded by stressing to Maintenance and Construction the importance of performing the work required to clear the deficiencies but little net progress was noticed during this report period.

The inspectors reviewed a number of the deficiencies and determined they were of minor importance. This item will continue to be reviewed for the next few months to insure the net impact of the deficiencies has no adverse affect on plant operation.

The inspectors also observed apparent discrepancies with several Containment Spray (CS) system pipe supports. In particular, several variable load pipe supports in the CS system did not appear to have hot and cold setpoint markings, have load settings that were at the design setpoint, be not properly attached to the building structure, or have load scales either painted over or missing thus making it difficult to determine if the supports were properly set. Additionally, expansion anchor bolts installed to fasten the baseplate of several CS dead weight type supports to the torus room floor were improperly installed in that they were skewed beyond an acceptable installation angle and the nuts lacked full thread engagement. These observations were pointed out to the licensee who in turn evaluated them to be acceptable as-is. The licensee did make an internal recommendation to correct load scale discrepancies where load scales were missing or painted over. The inspectors had no additional questions at this time.

4. QA 1984 Annual Assessment

The inspector attended the 1984 annual QA assessment meeting held during this report period. In attendance at this meeting were various site managers, corporate managers, and the Executive Vice President of GPU Nuclear Corporation. The purpose of the assessment was to summarize the accomplishments of 1984 and part of 1983, to present areas where improvements are needed, and to set goals for 1985.

The presentation of the assessment was broken down into five areas that correspond to the five individual QA/QC departments that report to the QA Mod/Ops Manager. These five areas are:

(1) Quality Control
(2) Special Processes and Programs
(3) Quality Assurance Engineering
(4) Site Quality Assurance Audits
(Operations Quality Assurance

Each presentation was made by the manager of the respective department. The accomplishments as presented by these managers were impressive and indicated that major improvements in the overall QA/QC program were made during the assessment period. The areas pointed out for improvement indicated both a self-awareness of weaknesses within each department and a desire to enhance the efficiency and performance in areas that were performing acceptably.

The Executive Vice President actively participated in the assessment by asking perceptive questions and setting dates by which he expects the goals for 1985 to be well underway towards accomplishment.

The inspector considered the QA assessment to be objective and well received by upper level management. No outstanding items were identified.

5. Drywell Entry

On November 23, 1984 the licensee made a drywell entry to investigate the erroneous indication on the electromatic relief valve (EMRV) acoustic monitors. The licensee was performing Procedure 602.4.003, Electromatic Relief Valve Operability Test, when the operators noted that the EMRV acoustic monitors were not responding as required. Plans were made to enter the drywell to determine the cause of the acoustic monitor's aberrant indications.

The appropriate radiological precautions were followed as the licensee personnel entered the drywell at approximately 0732 on November 23, 1984. The licensee personnel exited the drywell at approximately 0742. The radiological control technicians who entered the drywell decided to exit the drywell when they encountered an unexpectedly high radiation field. Although no exposure limits were exceeded, the average dose the personnel received was substantially greater than the amount the Group Radiological Controls Shift Supervisor (GRCSS) had expected based on previous entries.

This entry and previous drywell entries had been made in accordance with Procedure 902.5, Preparation for Drywell Initial Entry and 500 PSIG Inspection or 1000 PSIG Inspection, and Procedure 201.2, Plant Heatup to Hot Standby.

Paragragh 4.13.2 of Station Procedure 201.2, Rev. 16, and 3.12 of Radiological Control Procedure 902.5, Rev. 19, require inserting sufficient control rods so that the Group Shift Supervisor (GSS) can verify the reactor is sub-critical and power level is decreasing, prior to personnel entering the drywell. This requirement was not met because a communications problem between and within radiological control and plant operation personnel resulted in the team making the entry without the GSS's knowledge or permission. The failure to follow procedures is contrary to Technical Specification requirements and is a violation (219/84-34-01).

The major contributing factor to the higher than expected radiation field inside the drywell was the nitrogen-16 gamma radiation from the main steam lines. This radiation field was created as a result of the turbine bypass valves being open for the EMRV testing that had just been terminated and would resume immediately upon repair of the EMRV acoustic monitors. The procedures did not provide adequate guidance to the licensee personnel with regard to bypass valve position for drywell entry. These procedures should provide the guidance necessary to safely perform drywell entries for various plant conditions.

Other discrepancies identified by the inspectors during their review of this event included:

Procedure 902.5 did not specify the plant conditions to include power level limitations nor bypass valve positions necessary for the drywell entry and Procedure 201.2 did not reference Procedure 902.5 in paragraph 4.13.2.

The licensee conducted a critique of the event to determine the root cause and to insure proper corrective action. In addition, a consultant was employed to independently investigate the event and the conditions surrounding and leading up to the drywell entry.

6. Radiation Protection

During entry to and exit from radiation controlled areas (RCA), the inspector verified that proper warning signs were posted, personnel entering were wearing proper dosimetry, that personnel and materials leaving were properly monitored for radioactive contamination and that 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 to verify that personnel complied with the requirements of applicable RWP's and that workers were aware of the radiological conditions in the area.

7. Physical Security

During daily entry and egress from the protected area, the inspector verified that access controls were in accordance with the security plan and that security posts were properly manned. During facility tours, the inspector verified that protected area gates were locked or guarded and that isolation zones were free of obstructions. The inspector examined vital area access points to verify that they were properly locked or guarded and that access control was in accordance with the security plan.

8. Review of Licensee Event Reports (LERs)

The inspector reviewed LERs submitted to NRC:R1 to verify that the details were clearly reported, including the accuracy of the description and corrective action adequacy. The inspector determined whether further information was required, whether generic implications were indicated, and whether the event warranted onsite follow-up. The following LERs were reviewed:

84-24: The reactor low-low water level sensor (RE02C) switch failed to trip at the setpoint required by the procedure. There are two switches associated with each RE02 instrument. An AC switch for the Reactor Protection System and a D. C. switch which actuates core spray. During Procedure 619.3.004, Reactor Low-Low Water Level Functional Test, the level sensor did not trip as required. The problem stems from not being physically able to place both switches at the exact same setting. All switch settings were found to be in compliance with the requirements in technical specifications. The licensee will revise the procedures to properly calibrate the reactor water low-low water level sensor switches.

This LER is closed.

84-25: Torus water was injected into the reactor vessel during the performance of 610.3.005, Core Spray System Instrument Channel Calibration and Test. This occurred as a result of extensive revisions to the procedure and personnel error. The following corrective action was initiated:

- -- Operations personnel will be counseled to control tests closely.
- -- Guidelines will be developed to determine which procedures must be accomplished in the as written sequence.

This LER remains open.

84-27: The liquid poison system boron concentration was found to be 10.1%. Technical specifications require the minimum concentration to be 10.3%. The licensee immediately commenced a reactor shutdown and made all the necessary notifications. Additional sodium pentaborate was added and the plant shutdown terminated. Operation personnel had filled the liquid poison tank with water from 74% level to 96% level. Plant chemistry personnel had assumed the water addition would be from 80% to 91% level, the normal tank operating level band. This resulted in the boron concentrate being below the minimum specified concentration. The licensee has issued a memorandum providing operations personnel temporary guidance on filling the liquid poison tank. In addition, the licensee will implement procedural controls to verify tank levels with chemistry personnel prior to adding water and to sample after the water addition.

This LER is remains open.

84-28: The electromatic relief valves (EMRVs) did not operate as required by Procedure 602.4.003, Electromatic Relief Valve Operability Test. Steam leakage past the retainer treads prevented the valve from operating as designed. The licensee seal welded the retainer to the valve body to eliminate the steam leakage path. In addition, the licensee revised the maintenance procedure 702.1.007, Electromatic Relief Valve Removal, Disassembly, Repair, Reassembly and Installation, to incorporate additional post-maintenance testing to provide assurance of valve operability. Section 2 and Inspection Report 50-219/84-32 provide additional information on this topic.

This LER is closed.

9. Surveillance Testing

The inspector reviewed the following surveillance tests to determine if the tests were included on the master surveillance schedule, if the test was technically adequate and if it was performed at the required frequency.

620.3.003 APRM Surveillance Test and Calibration, Revision 10, 9/22/84.

620.3.001 LPRM Test and Calibration (Front Panel Test), Revision 6, 10/6/84. Short Form 22504 dated 11/28/84, was written to correct auxiliary downscale lights for 5 LPRM's. The audible alarm was found to be operable for all five LPRM's in question.

604.3.009 Drywell and Torus Oxygen Analyzer Calibration, Revision 6, 9/8/84

No unacceptable conditions were identified.

10. Review of Periodic and Special Reports

Upon receipt, periodic and special reports submitted by the licensee pursuant to Technical Specification 6.9.1 were reviewed by the inspector. This review included the following considerations: the report includes the information required to be reported to the NRC; planned corrective actions are adequate for resolution of identified problems; and that the reported information is valid. The November Monthly Operating Report was reviewed by the inspector.

No unacceptable conditions were identified.

11. Exit Interview

At periodic intervals during the course of this inspection, meetings were held with senior facility management to discuss the inspection scope and findings. A summary of findings was presented to the licensee at the end of this inspection.

The following licensee personnel were present at the exit interview:

J. Barton, Deputy Director
P. Fiedler, Vice President/Director - Oyster Creek
S. Fuller, Operations Quality Assurance Manager
D. Holland, Oyster Creek Licensing Manager
J. Sullivan, Jr., Plant Operations Director

At the end of the exit interview the licensee personnel present were asked if any of the topics discussed contained proprietary information. The licensee did not consider any information discussed to be of a proprietary nature.