U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/85-19

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: June 3 - 30, 1985

Inspectors: W Baunack for — 7/24/85
W. H. Bateman, Senior Resident Inspector — date

W Baunack for — 7/24/85
J. F. Wechselberger, Resident Inspector — date

W. Baunack — 7/24/85
W. H. Baunack, Project Engineer — date

Ronald R. Bellamy / for — 7/24/85
H. J. Bicehouse, Radiation Specialist — date

Approved by:

HB Kister — 7/24/85
H. B. Kister, Acting Chief, Reactor Projects Section 1A — date

Inspection Summary:

Routine onsite inspections were conducted by the resident inspectors and two region based inspectors (221 hours) of activities in progress including plant operations, physical security, radiation control, housekeeping, chemistry, surveillances, and hanger inspections. The inspectors also attended a management meeting to discuss the status of the Bulletin 79-14 inspection and reviewed licensee action taken to address previous inspection findings, Bulletins, and Circulars. In addition, the inspectors made routine tours of the control room and followed up the plant's response during and after the reactor scram from full power operation.

8508080403 850726
PDR ADOCK 05000219
Q PDR

Results:

No violations were identified and no new items were opened. Fourteen open items were closed. The plant personnel involved in the response to the reactor scram were observed to have performed in accordance with procedures.

DETAILS

1. Summary of Plant Activities

At the beginning of this report period, the plant was operating at 640 MWe. One of the major concerns was an abnormally high trunion room temperature and an entry was made into the room to attempt identification of the problem. It was not successful. Cycling of V-14-35 ('B' Isolation Condenser isolation valve) continued to be satisfactorily performed on a weekly basis; however, a higher than normal closing current was observed during one of the closing cycles. It is planned to disassemble and inspect the valve during the October outage. The valve continues to leak by the seat, thus causing an Isolation Condenser shell temperature in the 190°F range. The annual emergency drill was conducted June 5, 1985 and was successful. Power output during the month was consistently high except when the plant was shutdown during the recovery period after the scram on June 12.

Hanger inspections continued and although many discrepancies were identified, none were determined to affect system operability. Several supports were characterized as questionable and corrections were made to them within a 72-hour licensee imposed time limit. Because of the many pipe support discrepancies identified, and a concern over the potential affects of these discrepancies on safety-related piping system performance during a seismic event, the NRC requested GPUN to write a letter justifying continued operation of Oyster Creek. This action was completed and the licensee's rational was deemed acceptable by the NRC.

The reactor recirculation pump MG set brushes were observed to be arcing. Corrective action involved one at a time removal from service of the five recirc pumps to fix the problem. During this activity, the recirc pump discharge valve on each of two recirc loops failed to close. The valves were eventually closed by manipulating the torque switch contacts. The problem was considered to be either dirty contacts or the result of electrically backseating the valves. It was also observed during this report period that some of the recirc pump motor stator thermocouple readings were erratic. This concern was under investigation at the end of the report period.

Problems were encountered with safety relief valve position indication devices in the drywell and compensatory measures were taken as required by Tech Specs. At the end of the report period, the 'A' electromatic relief valve thermocouple was declared inoperable and the licensee was pursuing an emergency Tech Spec change to allow continued plant operation. The 'C' ESW pump failed its surveillance and placed the plant in a limiting condition for operation. The problem was determined to be a stuck open check valve. The #2 diesel generator was declared inoperable because of a failed current transformer. The problem was fixed before a plant shutdown would have been required by Tech Specs.

On June 12, with the plant at full power, the electric pressure regulator (EPR) failed causing one bypass valve to open. This caused pressure to decrease to the MSIV trip set point, causing MSIV closure followed by a reactor scram. (This event is discussed in paragraph 10.3 of this report.) The plant was restarted on June 18. During the shutdown, the trunion room high temperature problem was determined to be caused by a damaged fan which was replaced. Other work performed included EPR, SRV position indication, V-16-14 (Reactor Water Cleanup System inlet isolation valve), and scram discharge volume (SDV) drain valve repairs.

Secondary side steam leaks continued throughout the report period with repair efforts only partially successful. The reactor building roof continued to leak each time it rained. GPUN has gone out for bids to repair the roof. Drywell bulk temperature increased to greater than 150°F and efforts were underway to investigate the cause of the high temperature.

The spent fuel shipments from West Valley continued with only two shipments remaining at the end of June.

2. Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item (219/84-09-06): Verification that pipe support rod end bushings are staked per licensee commitment to NRC IE Circular 81-05

A visual inspection was performed of the ball bushing race after removal of the strut rod from the clevis. This inspection indicated staking had been performed.

(Closed) Violation (219/81-05-02): Failure to conduct annual procedure review as required by Technical Specification 6.8.2 and Administrative Procedure 107.

This item was previously reviewed during Region I Inspection 84-28 and was kept open pending the establishment of procedurally required records of procedure reviews. During this inspection, the inspector verified by direct observation of documentation that the Safety Review Manager is now maintaining records of procedure reviews as required by Station Procedure 107, Procedure Control.

(Closed) Violation (219/82-23-05): Failure to calibrate the integrated thermal monitoring system.

Periodic routine thermal monitoring system calibrations did not include calibration of the thermal sensor. The licensee committed to implement a program of replacing the temperature sensors monthly with vendor calibrated sensors as soon as the spare sensors were received. Before this program could be implemented, the calibration requirements were eliminated by the issuance of Technical Specification Amendment 66.

(Closed) Inspector Followup Item (219/82-29-04): The licensee to conduct an extensive feedwater (FW) system valve overhaul program during the next refueling outage.

The inspector selected a number of important FW valves and reviewed their maintenance history to determine the work performed on these valves during the 1983 - 1984 outage. The following is a list of the valves selected and the work performed on them:

Valve	Work Performed
1D11A F.W. Control Valve	Rebuilt
1D12A F.W. Control Bypass Valve	Rebuilt
1D11B F.W. Control Valve	Rebuilt
1D11C F.W. Control Valve	Rebuilt
1D12C F.W. Control Bypass Valve	Rebuilt
V-2-7 F.W. String "A" Inlet	Overhauled
V-2-8 F.W. String "B" Inlet	Overhauled
V-2-9 F.W. String "C" Inlet	Overhauled
V-2-10 F.W. String "A" Outlet	Overhauled
V-2-11 F.W. String "B" Outlet	Rebuilt
V-2-12 F.W. String "C" Outlet	Overhauled

In addition, the inspector discussed the performance of the FW system valves with an operating shift. The operators felt the FW system valves were currently performing much better than they had been in the past. The inspector had no further questions relating to this matter.

(Closed) Inspector Followup Item (219/83-22-01): Licensee to perform an investigation into the cause of the air intrusion into the fuel pool while backwashing a fuel pool filter.

An evaluation in the form of an event critique was performed immediately following the occurrence. In addition, this evaluation was continued by observing equipment performance during a subsequent fuel pool filter backwash. The cause of the event was attributed to operator error. Although the procedure used provided adequate instruction for the safe backwash of the fuel pool filter, it has been modified to clarify instructions in an effort to prevent a recurrence of this event.

(Closed) Violation (219/84-28-01): Instrument root valves left closed following maintenance.

The licensee's corrective action consisted of instructing Maintenance and Construction (M&C) production personnel not to alter any valve positions unless they are within the tagged out boundary or covered by the maintenance Short Form or procedure. The inspectors verified these instructions had been given to M&C personnel.

3. Management Meeting

On June 13, 1985 a meeting was held in Region I, involving NRC and GPUN management personnel, to give GPUN an opportunity to justify continued operation of Oyster Creek. The meeting was a followup to an earlier meeting held May 20, 1985 to discuss the concerns raised by the results of NRC inspection 85-14. Between the two meeting dates, GPUN conducted pipe support inspections to compare as-built pipe support and piping isometric drawings to the actual in plant as-built conditions. These inspections disclosed many discrepancies which precipitated a NRC concern as to whether or not safety-related piping systems could perform as designed if subjected to design seismic loading.

At the June 13 meeting, the NRC requested that GPUN justify continued operation and discuss the action they intend to take to identify and resolve old issues that were not satisfactorily resolved when JCP&L was the licensee. GPUN presented the results of their preliminary hanger inspections and stated none of the discrepancies affected the ability of safety-related systems to perform satisfactorily during a seismic event. The statement was based on engineering judgement and involved a commitment to perform either minor rework, repair, or perform a confirmatory analysis if required. NRC Licensing questioned GPUN as to the validity of the data used in the Systematic Evaluation Program (SEP) for Oyster Creek. GPUN stated they would forward the updated as-built information to NRC Licensing when scheduled inspections were completed but, that as of the date of the meeting, no significant discrepancies had been identified. NRC Licensing reserved judgement on this position pending evaluation of the updated as-built drawings.

As regards the identification and solution of problems inherited from JCP&L, GPUN stated that programs already in existence to accomplish this will continue. The programs include:

- -- Updating as-built drawings
- -- Updating FSAR
- -- Updating vendor document control programs
- -- Use of Field Change Requests to identify and resolve problems

GPUN also stated they had no plans to review action taken to close older NRC Bulletins, but they may reassess this position after work is complete on Bulletins 79-02 and 79-14.

At the conclusion of the meeting, the NRC requested GPUN perform the following actions:

1) Prior to restart of the plant from the unscheduled June 12 scram, submit a justification for continued operation (JCO) of Oyster Creek. The NRC stated the JCO should state the basis for GPUN's conclusion that the piping systems will, despite the large number of identified discrepancies, respond as designed to a seismic event.

2) Submit to NRC Licensing updated pipe and pipe support data for the SEP systems.

3) Commit to completing Bulletin 79-14 prior to restart from the October 1985 mini-outage.

GPUN agreed to the above items. The JCO was presented to Region I and was acceptable. A copy is attached to this Inspection Report.

4. Plant Operation Review

4.1 Routine tours of the control room were conducted by the inspectors during which time the following documents were reviewed:

-- Control Room and Group Shift Supervisor's Logs;

-- Technical Specification Log;

-- Control Room and Shift Supervisor's Turnover Check Lists;

-- Reactor Building and Turbine Building Tour Sheets;

-- Equipment Control Logs;

-- Standing Orders; and,

-- Operational Memos and Directives.

The reviews indicated that the logs were generally complete. Control room housekeeping and behavior were observed to be acceptable.

4.2 Routine tours of the facility were conducted by the inspectors to make an assessment of the equipment conditions, safety, and adherence to operating procedures and regulatory requirements. The following areas were among those inspected:

-- Turbine Building

-- Vital Switchgear Rooms

-- Cable Spreading Room

-- Diesel Generator Building

-- Reactor Building

The following items were observed or verified:

a. Fire Protection:

-- Randomly selected fire extinguishers were accessible and inspected on schedule.

-- Fire doors were unobstructed and in their proper position.

-- Ignition sources and combustible materials were controlled in accordance with the licensee's approved procedures.

-- Appropriate fire watches or fire patrols were stationed when equipment was out of service.

b. Equipment Control:

-- Jumper and equipment mark-ups did not conflict with Technical Specification requirements.

-- Conditions requiring the use of jumpers received prompt licensee attention.

-- Administrative controls for the use of jumpers and equipment mark-ups were properly implemented.

c. Vital Instrumentation:

-- Selected instruments appeared functional and demonstrated parameters within Technical Specification Limiting Conditions for Operation.

d. Housekeeping:

-- Plant housekeeping and cleanliness were in accordance with approved licensee programs.

5. IE Bulletins and Circulars

a. Bulletins

Licensee actions concerning the following IE Bulletins were reviewed by the inspector to verify that: the Bulletin was forwarded to appropriate on-site management; a review for applicability was performed; information discussed in the licensee's reply was accurate; corrective action taken was as described in the reply; and the reply was within the time period described in the Bulletin.

-- IEB 78-03, Potential Explosive Gas Mixture Accumulations Associated with BWR Offgas System Operations. To comply with the Bulletin, Task No. 3208007.02 was initiated by GPUSC to perform a study of the offgas system. An inter-office memo dated January 25, 1979 documented a "Ventilation Study of the Oyster Creek Offgas System". This study made certain

recommendations for additional hydrogen monitoring and ventilation system improvements. Engineering Requests were issued in 1979 to install the equipment recommended. The equipment was verified to have been installed during the 1983 - 1984 outage.

This item is closed.

-- IEB 79-11, Faulty Overcurrent Trip Device in Circuit Breakers for the Engineered Safety Systems. This Bulletin described certain problems identified in Westinghouse DB-50 and DB-75 circuit breaker time delay dashpot end caps. A licensee review determined Westinghouse switchgear is not utilized in any safety related systems at Oyster Creek.

This item is closed.

-- IEB 80-13, Cracking in the Core Spray Spargers. This Bulletin described certain instances of cracking identified in Core Spray spargers. One of the events described occurred at Oyster Creek in 1978. By letter dated June 27, 1980 the licensee forwarded to NRR the inspection and evaluation of Core Spray sparger piping performed during a 1980 outage. By letter dated July 2, 1980 to NRR, additional information related to the Core Spray sparger cracking was provided. Also, by letter dated May 13, 1983, to Region I, in accordance with IE Bulletin 80-13 and Section 6.9.3 (3)e of the Oyster Creek operating license and Technical Specifications, the results of the inspection performed during the 1983 refueling outage were reported. The current license requirements relating to Core Spray sparger inspections are contained in License Amendment No. 70. All actions specified by the Bulletin have been taken and reported by the licensee.

This item is closed.

-- IEB 80-25, Operating Problems with Target Rock Safety-Relief Valves at BWRs. The licensee does not use Target Rock safety-relief valves at Oyster Creek. Also, though not indicated in the licensee's response to the Bulletin, facility Procedure 602.4.003, Electromatic Relief Valve Operability Test, requires "If an EMRV fails to function as designed, excepting for pressure setpoint requirements, and the cause of the malfunction is not clearly determined, understood or corrected, perform the following prior to returning the valve to service:

1) Remove the valve from service, disassemble, inspect and adjust in accordance with Procedure 702.1.007, "Electromatic Relief Valve Removal, Disassembly, Repair, Reassembly and Installation".

This item is closed.

-- IEB 83-06, Nonconforming Material Supplied by Tube-Line Corporation. The licensee responded to this Bulletin by letter to Region I, dated November 23, 1983. Two flanges in the augmented fuel pool cooling system were identified as supplied by Tube-Line Corporation. These were evaluated to be adequate for use. The inspector verified the licensee's response by reviewing the following documentation: 1) Internal response to licensing action item 83153.1, which included a list of vendors which was reviewed for Tube-Line material supplied, 2) Results of chemical and physical tests performed on a sample from the heat from which the two flanges were made, 3) MNCR 83-102 issued for the two flanges in use, and 4) Justification for "use-as-is" for the two flanges. No problems were identified.

This item is closed.

-- IEB 79-15, Deep Draft Pump Deficiencies. This Bulletin described certain manufacturing deficiencies identified in deep draft pumps and requested certain information from the licensee. The licensee responded to the Bulletin by letter dated September 11, 1979. The deep draft pumps utilized in safety-related applications were identified as 1) four emergency service water pumps (ESW), 2) two service water pumps, 3) two diesel driven fire pumps, and 4) one fire pond pump. A second fire pond pump was installed December 30, 1980.

The ESW pumps and the diesel driven fire pumps are normally idle and in a standby status. They are only run for periodic surveillance testing each month for short periods of time (less than one hour per month).

The maintenance history of the safety-related deep draft pumps was reviewed during this inspection. The following identifies the significant maintenance performed to date on the various deep draft pumps:

ESW Pump 1-1:
11/28/78 - Pump inspected and rebuilt
2/08/82 - Pump inspected and rebuilt

ESW Pump 1-2:
1980 - Pump inspected and rebuilt

ESW Pump 1-3:
11/24/78 - Pump inspected and rebuilt
11/20/81 - Pump inspected and rebuilt
1983-84 outage - Pump overhauled

ESW Pump 1-4:
1980 - Pump inspected and bottom case replaced.

Service Water Pump 1-1:
1978 - Replaced old pump with new pump
9/05/80 - Pump inspected and rebuilt
8/21/82 - Pump rebuilt
1/15/83 - Pump rebuilt

Service Water Pump 1-2:
5/15/81 - Pump overhauled
1983-84 outage - Pump rebuilt

Diesel Fire Pump 1-1 and 1-2:
No major maintenance performed.

Fire Pond Pump 1-1 and 1-2:
No major maintenance performed.

The inservice test (IST) data for the pumps was also reviewed. Inservice testing was first initiated in December 1982, but the facility was shutdown February 1983 to October 1984. IST data was again recorded from October 20, 1984 to June 1985. Due to the long shutdown, only a limited amount of IST data is available. The results of the IST data review is as follows:

ESW Pumps:
Three of the four pumps are currently operating in the alert range. However, the flow instrument (sonic flow detector) is in question. The performance of the pumps is being closely monitored and evaluated.

Service Water Pumps:
All IST data is in the acceptable range.

Diesel Driven Fire Pumps:
All IST data is in the acceptable range.

Fire Pond Pumps:
No IST data recorded.

The diesel driven fire pumps have the capability to provide a source of backup water to the Core Spray System, the Condensate Storage Tank, the Isolation Condensers and may also supply backup cooling to the main plant air compressors. However, no credit is taken in the Safety Evaluation for their use.

The ESW pumps appear not to have had any prolonged period of operation other than the routine monthly surveillance (approximately 15 minutes per month). Consequently, their ability to perform long term cooling in the event they are required following an accident has not been proven. This was discussed with the licensee. The licensee calculated that, in the event of an accident, the ESW pumps would be required for a

period of approximately 72 hours. To prove the pumps capability to perform, the licensee will operate one pump for 48 hours and the other 3 pumps 12 hours each during the October 1985 outage.

This item remains open pending review of the prolonged test run data.

-- IEB 84-03, Refueling Cavity Water Seal. The licensee inadvertently submitted an incomplete response to this Bulletin. A supplemental response will be provided.

This item remains open.

b. Circulars

Licensee actions concerning the following IE Circulars were reviewed to verify that the Circular was received by licensee management, that a review for applicability was performed, and that action taken or planned is appropriate:

-- IEC 78-11, Recirculation M-G Set Overspeed Stops. This Circular was reviewed during Region I Inspection 81-01. At that time the installation of recirculation pump M-G set overspeed stops was under review. The requirements of this Circular were not necessary at that time. On August 27, 1984 Amendment 75 to the Oyster Creek Operating License was issued. This Amendment addressed the MCPR limits associated with core flows.

This item is closed.

-- IEC 79-19, Loose Locking Devices on Ingersol-Rand Pump Impellers. This Circular was reviewed by the licensee and it was determined to be applicable to the four main and four booster core spray pumps. A job order was issued in 1981 and the locking devices were installed on all 8 Core Spray pumps during the 1983-1984 outage. Also, Procedure 710.1002, "Core Spray and Core Spray Booster Pump Inspection and Maintenance" was revised to reflect the installation of the impeller lock nut devices.

This item is closed.

-- IEC 81-08, Foundation Material. This Circular recommended certain action for construction permit holders and was not addressed to operating facilities. Although this Circular did not apply to Oyster Creek, it was routed for review. Also, as a result of this Circular, a memorandum was addressed to the Manager, Radwaste Operations dated May 28, 1985 which discussed the survey data and survey interval for the Radwaste Building. The inspector had no further questions relating to this Circular.

This item is closed.

6. Observation of 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.

A moderate loss of physical security was reported to the NRC by the licensee. It involved the discovery of a guard who was asleep at his assigned post. The guard was found asleep by a member of the site security force who was making rounds. The sleeping guard was immediately replaced and subsequently terminated. The guard had been at his assigned post for approximately 1 1/2 hours. A search of the protected and vital areas of the plant was performed and a determination made that there were no unauthorized entries.

The inspectors considered the licensee's corrective action appropriate and had no further questions.

7. Radiation Protection

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 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 RWPs and that workers were aware of the radiological conditions in the area.

During this report period, the licensee identified a contaminated box stored on the Forked River site. The box was approximately 40 feet long, triangular in shape (with the base about 1 foot long and height about 1 1/2 feet), and contained a new SRM/IRM dry tube. The contamination involved fixed contamination on the outside of the box with a maximum value of .08 mr/hr and smearable contamination on the inside surface with a maximum reported value of 6000 dpm/100 cm^2. The dry tube itself was found not to be contaminated. Research back through the records indicated the dry tube may have been unused from the original construction days of Oyster Creek. The licensee explained the contamination probably resulted from storage of the box on the Oyster Creek site during early years of plant operation. No one could reconstruct where the box had been stored at Oyster Creek prior to shipment to the Forked River site, but some

people recalled it had been stored near the old radwaste building for a long period of time. The licensee stated the box was moved from Oyster Creek to Forked River sometime in late 1979 or early 1980 and that the sensitivity of radiological survey instruments used in surveys at that time may not have been sensitive enough to detect the low levels of contamination.

Prior to realizing the box was contaminated, an individual burned (with an oxy-acetylene torch) the bolts off the cover at one end to look inside the box. Shortly after he did this (within 1 day), the box was determined to be contaminated. The individual was given a whole body count the following day whereby it was determined he did not incure any uptake of radioactive material. Soil samples were taken in the area the box had been stored and radioactive Cobalt-60 and Cesium-137 were found in small amounts where the bolts had been burned off. The area was cleaned up. (Cobalt-60 was 5.39 E^{-5} micro curies per gram and Ce-137 was 1.53 E^{-4} micro curies per gram.)

GPUN radiation control personnel responded promptly and properly to the discovery of the slightly contaminated box. It was subsequently returned to Oyster Creek for disposal. Site radcon personnel intend to walk down the Forked River site to verify no additional contaminated material is stored there.

8. Return of Spent Fuel From West Valley

During this report period, additional shipments of spent fuel were received onsite from West Valley, New York. The resident inspectors observed spent fuel receipt at Oyster Creek and the handling and unloading of spent fuel from the TN-9 spent fuel shipping casks. Radiation Control (radcon) personnel were observed to be knowledgeable and in control of radcon related activities. Once reaching the refueling floor, the TN-9 casks were moved and unloaded and spent fuel stored in the spent fuel pool in accordance with controlling procedures. It is anticipated that the final shipments of spent fuel will be completed in July 1985.

9. Pipe Hanger Inspections

During this report period, pipe and pipe support inspections continued. Many discrepancies continued to be identified but very few were considered significant by the licensee. The overall scope of pipe and supports requiring inspection continued to increase and stood at 467 at the end of this report period. The inspectors continued to review a sample of the MNCRs generated to ensure completeness and reasonable technical dispositions. Because of the unscheduled reactor scram and subsequent drywell entry during this report period, pipe and supports inside the drywell were inspected on portions of the Recirculation, Main Steam, Feedwater, and Core Spray Systems. The licensee stated the inspection results did not disclose any serious discrepancies. Based on the unacceptability of certain discrepancies, e.g., a missing shear lug or

cracked hanger clamp, the licensee self-imposed various time limits within which they must fix the discrepancies. The NRC resident inspectors will continue to follow the results of the pipe and pipe support inspections until they are completed.

10. Followup of Operational Events

10.1 During this report period, the licensee declared the thermocouple on the outlet of 'A' EMRV inoperable because of erratic temperature readings. This placed them in a 7-day limiting condition for operation. It was subsequently determined that a drywell entry would be necessary to fix the problem. In an attempt to avoid a plant shutdown, the licensee requested that NRC Licensing grant them an emergency change to the Tech Specs to allow use of the thermocouple in the combined EMRV discharge header. NRC Licensing granted the emergency changes until the next time the drywell is opened at which time the thermocouple must be repaired. The resident inspectors agreed with the temporary solution to the problem.

10.2 Because of the warm summer weather, drywell bulk temperature has increased to the recently changed higher limit of 150°F. (The previous limit was 135°F.) When drywell bulk temperature exceeds 150°F the plant emergency procedures require that all available cooling in the drywell be in operation. At one point during this report period, the 150°F limit was exceeded and the last of 5 drywell coolers was put on line. It had no affect on the drywell temperature and was subsequently secured. The emergency procedures do not require any further action until drywell temperature reaches 281°F. The NRC inspectors questioned the licensee as to the method of calculation of the drywell bulk temperature and learned that the temperature of the gas contained in the large spherical volume at the bottom of the drywell is not factored into the calculation. This substantially cooler gas volume is not factored into the calculation because of the lack of a reliable temperature indication. The NRC inspectors also determined that the FSAR assumes a maximum drywell temperature of 135°F at the start of a LOCA for design accident calculations and that GPUN did not notify NRC Licensing of the increase in the assumed drywell temperature to 150°F at the start of a design basis accident.

In response to the inspectors' concerns regarding the inherent inaccuracy in the drywell bulk temperature calculation, the licensee stated they would pursue other methods to arrive at a formula to more accurately determine the temperature. Because of the deviation between the FSAR and the plant procedures and the failure to notify NRC Licensing of the change in assumed drywell temperature, the licensee stated they would submit a Licensee Event Report to identify the issue and that a more realistic temperature calculation might solve the whole problem.

The inspectors will continue to follow licensee action to address the drywell bulk temperature problem.

10.3 On June 12, with the plant at full power, the electric pressure regulator (EPR) failed. Its failure resulted in one turbine bypass valve opening. This caused a pressure decrease and subsequent MSIV closure and reactor scram. The plant parameters responded as expected to this event, but because the scram dump volume drain valves (V-15-121 and V-15-134) failed to properly close, the Reactor Water Cleanup System isolated, and the reactor vessel water level just after the scram was unknown, the scram recovery was more difficult than would have been expected. Plant operations personnel and most plant equipment responded admirably during the scram recovery which contributed to minimizing the consequences of the aforementioned problems. During the scram recovery, a fire protection sprinkler system initiated and sprayed water on safety-related equipment. This event resulted in declaration of an Unusual Event because the operability status of wetted down pieces of safety-related equipment was unknown. The scram occurred at 0938 and the Unusual Event was declared at 1035 and secured at 1322.

A discussion of the various problems encountered follows:

-- The failure of the EPR was the initiating event. The licensee inspected the EPR to determine the cause of its failure and issued a report documenting their findings. The report stated the pilot spool inside the EPR servo valve (MOOG valve) was stuck in a position whereby it was porting oil to the hydraulic actuator in the valves open direction. This caused the turbine control valves and one bypass valve to open. The cause of the stuck pilot spool was stated to be impurities in the hydraulic oil. The corrective action taken included replacement of associated oil filters, flushing hydraulic lines, rebuilding the MOOG valve, and performing an open loop EPR transient response test. The report stated the problem could occur again but was improbable.

-- Just after the scram and subsequent to reactor water level oscillation due to rapid pressure changes, the indicated reactor water level was pegged high on the normally used Yarways and GEMACS. This resulted in an unknown water level in the reactor and a concern that initiation of an isolation condenser could result in water hammer problems if the water level was at or above the reactor vessel penetration for the isolation condenser inlet line. The operators elected to manually bypass automatic initiation of the Isolation Condenser system until they were sure of the reactor water level. The refueling zone level recorder was placed in service after which time it was determined that water level was just at the top of the Yarways and GEMACS and well below the isolation condenser steam inlet

line. When this determination was made, the operators commenced to use the isolation condensers to control primary system pressure and temperature.

-- Prior to the scram the Reactor Water Cleanup (RWCU) system had been out of service for demineralizer resin replacement. Resin replacement had just been completed and the system was ready to be put back in service. When the problem with high reactor vessel water level was realized, attempts were made to put the system in service to drain water from the vessel. These attempts initially failed because V-16-14 (RWCU inlet isolation) failed to open due to a tripped breaker. The valve was subsequently opened and the RWCU system put in service. Troubleshooting of the problem after the event determined that a gearing change in the Limitorque valve operator was necessary. This was accomplished prior to restart.

-- The reactor scrammed because the MSIVs tripped closed. This scram signal is presently designed so that it cannot be bypassed until plant pressure decreases to 600 psig independent of the position of the Mode Selector Switch. This results in the scram discharge volume drain valves acting as containment isolation valves until the scram can be reset at less than 600 psig. During this event, the two series drain valves failed to stay fully closed, thus, resulting in leakage of hot reactor coolant into the Reactor Building Equipment Drain Tank (RBEDT). This hot coolant flashed to steam and rose out floor drains that also drain into the RBEDT. The rising steam from the elevation 23' floor drains, combined with smoke from hot blistering paint on the SDV piping, caused a smoke detector on elevation 51' in the cleanup demineralizer area to activate the local sprinkler system. The water from the sprinkler system wetted down safety-related instrument racks and electrical and mechanical equipment in the southwest area of 51' and 23' elevations. The southwest portion of elevation 23' was inaccessable for a short period of time due to escaping steam. Subsequent to closure of the SDV drain valves, the condensing steam caused surface contamination problems in the southwest elevation 23' area. An attempt was made to determine the amount of water which drained through the two scram discharge volume drain valves into the RBEDT. Records of tank pumpout rates are maintained for each shift. However, the number of variables associated with this data made it impossible to make an accurate determination.

Subsequent investigation of the failure of the SDV drain valves to fully close disclosed the following:

(1) V-15-134 closing springs were undersized thus failing to hold the valve closed against reactor coolant system pressure.

(2) V-15-121 valve stroke was 1/8" short of being fully closed.

Repairs were made to both valves. A review of the maintenance history cards on these valves indicated the following:

(1) Surveillance Test 619.4.022, Scram Discharge Volume Vent and Drain Valve Functional Test was last successfully performed on October 21, 1984 following maintenance.

(2) Valve V-15-121 failed to close during a surveillance test on August 17, 1984. Failure was attributed to the mechanical override being positioned so that the air operator could not close the valve. The mechanical override was repositioned and the surveillance was repeated and the valve passed. The operating procedure was revised to require mechanical override position verification.

(3) Valve V-15-134 failed to close during surveillance testing on August 17, 1984. The valve was found to be binding due to brittle shaft packing. The valve and operator were overhauled and the packing replaced. The surveillance was repeated and the valve passed.

A review of the inservice test (IST) data for the scram discharge volume drain valves V-15-121 and V-15-134 was performed. Valve V-15-134 showed some erratic stoke times for some tests performed during 1984. This data was evaluated and retests were performed as required. Data recorded for the 10 tests performed during 1985 was stable. Valve V-15-121 test data showed stable performance for all 21 tests performed during 1984 and 1985. Valve inservice test data showed no indication of unsatisfactory performance.

In summary, the SDV drain valves failed to close properly upon receipt of a scram signal. Both of the series valves leaked and permitted some unknown volume of hot reactor coolant to pass through the SDV system and into the reactor building. A review of maintenance records and IST results did not identify any precursors to the leakage problems. However, NRC Unresolved Item 84-09-08 questioned the rational for not including these valves in the 10 CFR 50 Appendix J local leak rate test program. This issue will be followed up in a subsequent inspection.

The licensee conducted a post trip review (PTR) meeting to review the events, responses, and pertinent data files to determine the problems that arose and to suggest a course of corrective action. The PTR identified and/or concluded the following:

(1) The EPR failed;

(2) V-15-121 and 134 failed to close;

(3) V-16-14 breaker tripped when attempting to place the RWCU system in service;

(4) "A" EMRV appeared to not fully reseat;

(5) Low vacuum scram relay 2K11 picked up 10 seconds after its three associated scram relays;

(6) There was an operator action problem regarding control of reactor vessel water level just after the scram;

The PTR group recommended that the identified equipment failures be corrected and basic principle trainer instructions be designed to address reactor vessel level control during isolation scram events.

The NRC resident inspector was in the control room for most of the scram discovery period and observed the response of the operators and plant equipment to the event. The inspector concluded that both operators, and plant equipment, with exceptions as noted above, performed adequately. Followup inspections by a region based inspector of the radiological aspects of the event concluded:

-- Emergency Preparedness (EP)

The licensee implemented EP procedures to monitor potential offsite releases. Following declaration of an Unusual Event at 1035 on 6/12/85, the licensee acted in accordance with the Oyster Creek emergency plan. Offsite dose projection responsibilities were turned over to Radiological Engineering with assistance and support of Radiological Controls personnel. A two-person team provided onsite radiation level readings and air samples approximately 1/4 mile downwind from the Reactor Building vent discharge point. "Dose Projections" were made at 1048, 1124, and 1245 based on data supplied by the onsite team. All dose projections were less than 0.1 mrem (minimum dose projection level). The licensee continued to monitor possible plume readings until about 1500 on 6/12/85 when the onsite team was recalled.

No violations were noted. The licensee appeared to have implemented adequate precautionary measures to monitor potential releases from the vents for the isolation condensers. All stack monitoring instruments remained virtually unchanged.

-- Environmental Monitoring

Soil and water samples taken immediately below the vents and contamination smears of the access road were made by the licensee. No activity attributable to the steam venting from

the isolation condensers was found in any of the samples. Residual levels of cobalt-50 and cesium-137 were detected in the soil samples but comparison to levels before the event showed the residual levels to be statistically unchanged by the event.

No violations were noted in the environmental monitoring program indicated by the licensee in response to the event. There appeared to be no contamination spread as a result of the event.

-- Radiation Protection

Radiation levels and potential in-plant airborne contamination were monitored closely. Radiological Controls personnel monitored area radiation monitoring system readouts, took adequate surveys of radiation, contamination and airborne radioactivity levels and controlled access to Reactor Building areas affected by the event.

No violations were noted. The licensee appeared to have adequately controlled personnel exposures during the event and the initial recovery efforts.

11. Emergency Planning

On June 6, 1985, the GPU Nuclear Sea-Breeze Study Program was reviewed and the data collection sites visited by an NRC Region I based inspector. The program is designed to address post 'TMI Action Plan' requirements (0737, Sup 1), specifically, accident assessment of radiological airborne effluents (Part 50, Appendix E).

The licensee plans to include considerations of sea/land breeze effects on atmospheric transport and diffusion (in the vicinity of the site) as part of the upgraded emergency response program. The program's scope includes data collection, data analysis and interpretation of results. Ultimately, the OCNGS Emergency Plan Implementing Procedures will be supplemented with the program findings.

The data collection phase of the study is currently operational. Three sites were chosen in the vicinity within 10 miles of the OCNGS. The sites are located in approximately a straight line southeast to northwest from Barnegat Light, New Jersey to west of the plant on Lacey Road. The sites provide for continuous meteorological data that is collected in 15 minute averages and stored digitally on magnetic tape. This data (wind speed, wind direction and a measure of vertical mixing depth) will be collected from late May 1985 until early October 1985. Surveillance is performed each Monday, Wednesday, and Friday at each monitoring location and daily operability checks are performed remotely. Strip charts are used at the measurement locations as a back-up to the computerized data loggers.

After the data collection period, a GPU meteorologist will analyze the data, identify, classify and group sea/land breeze episodes. This information will be further refined and proceduralized for use in protective action decision making.

The inspector has verified the start of the data collection phase of the study by a visit to each measurement location and an independent check of instrument operability, maintenance logs and remote interrogation capability.

12. 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 licensee stated that of the subjects discussed at the exit interview, no proprietary information was included.

Attachment

GPU Nuclear

GPU Nuclear Corporation
100 Interpace Parkway
Parsippany, New Jersey 07054-1149
(201) 263-6500
TELEX 136-482
Writer's Direct Dial Number

201-299-6797

June 14, 1985

Dr. Thomas E. Murley, Administrator
Region 1
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Dear Dr. Murley:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Preliminary Response to IE Inspection 85-14

During IE Inspection 85-14, several discrepancies between drawings which were updated as a result of IE Bulletin 79-14, and the inspector's field observations were identified. At a meeting with members of your staff on May 20, 1985, we committed to reinspect those systems accessible during operation to ascertain the extent that drawings reflect actual field conditions and determine if seismic analyses, which were based on these drawings, were still valid. On June 13, 1985, we presented our preliminary results of that reinspection. This letter complies with your request that we provide a written response outlining the bases for continued operation of the Oyster Creek Nuclear Generating station until the reinspection is completed.

We have concluded that the major technical requirements of IE Bulletin 79-14 have been addressed and satisfied through a review of documentation relating to the 1979/1980 inspection. This documentation review has determined that seismic concerns were addressed and corrected during the 1979/1980 effort.

~~8507030198~~ 3pp.

With regard to IE Bulletin 79-02, our review of the existing documentation has concluded that the technical requirements of the Bulletin such as base plate flexibility, factor of safety, cyclic loads and load testing have been addressed and show evidence of being satisfied. The existing test results will be summarized and statistics developed to document confidence levels.

The 1979/1980 inspection contained all the attributes required by IE Bulletin 79-14. The 1985 inspection included all the required 79-14 attributes plus additional attributes which surpass the requirements of the bulletin. Based on our 1985 inspection results, as of June 13, 1985, we have seen numerous minor deviations from the engineering drawings, primarily with regard to hangers. Many of these nonconformances are being corrected as they are found and the remainder are being evaluated by engineering for impact on system seismic capability. An engineering assessment is being performed in two steps. Each Material Nonconformance Report (MNCR) is first individually evaluated for conditional release. A system reevaluation is then conducted in order to assess the aggregate effect of all the MNCR's issued against that system. No variances have resulted in any safety related system being declared inoperable. The 1985 inspection effort, as of June 13, 1985, confirms Bulletin 79-14 conclusions reached in 1979/1980, and it does not change our conclusions about the overall seismic acceptability of the affected piping systems.

Because of the plant trip on June 12th, we are now in the process of performing inspections in the drywell. As of today we would expect to complete the 79-14 inspections of the accessible portions of one Main Steam Line, one equivalent Recirculation Piping Loop, one Main Feedwater Line, and one Core Spray Line. We will continue to inspect until the plant is ready for restart. Very preliminary inspection results from within the drywell, current as of June 14th, show a much reduced number of deviations between inspection attributes and drawings. No significant or major defects have been found to date. The reinspection effort will be completed during our next scheduled outage of October, 1985. If system reanalysis is required, the results will be reported to you as they become available.

In 1978 and 1979, a probabilistic risk analysis of the Oyster Creek Nuclear Generating Station was performed. This analysis, entitled the "Oyster Creek Probabilistic Safety Analysis" (OPSA) estimated earthquakes of 0.22g or greater to have a mean frequency of approximately 1.15×10^{-4}/yr. The OPSA

Dr. Thomas E. Murley, Administrator
June 14, 1985
Page Three

study calculated a core melt frequency of approximately 1.5×10^{-4}/yr for all seismic related events and approximately 2.8×10^{-6}/yr due to seismic-induced LOCAs (seismic pipe breaks which is the 79-14 issue). This equates to approximately 2% of the seismic core melt frequency, and is about 1% of the total core melt frequency. This results from the relatively high seismic capacities of the piping systems.

In summary, our bases for concluding there is reasonable justification for continuing Oyster Creek operation pending final completion of the inspections are as follows:

1. Inspection to date as previously noted does not change the prior 1979-1980 effort on Bulletin 79-14. Hence we conclude the seismic piping design capability is valid.

2. Very preliminary inspection results from within the drywell, current as of June 14th, show a much reduced number of deviations between inspection attributes and drawings. No major defect has been found.

3. The probability of a design bases or more severe earthquake prior to the completion of 79-14 inspection activities, is extremely remote.

4. The PRA studies show seismic induced LOCA's to be a very small contribution to core melt frequency and inspection results do not change this assessment.

It is our conclusion from all of the above that there is no perceptible change in any seismic induced LOCA risk which would affect public health and safety due to the continued operation of Oyster Creek.

Very truly yours,

P. R. Clark

P. R. Clark
President

PRC:kh
0534U

cc: Mr. John A. Zwolinski, Chief
Operating Reactors Branch No. 5
U. S. Nuclear Regulatory Commission
Washington, DC 20555

NRC Resident Inspector
Oyster Creek Nuclear Generating Station
Forked River, N. J. 08731

0534U