U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/85-15

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: May 6 - June 2, 1985

Inspectors: E L Conner for 6/21/85
W. H. Bateman, Senior Resident Inspector Date

E L Conner for 6/21/85
J. F. Wechselberger, Resident Inspector Date

Approved by: E L Conner 7/8/85
E. L. Conner, Chief, Reactor Projects Section 3C Date

Inspection Summary: Routine onsite inspections were conducted by the resident inspectors (172 hours) of activities in progress including plant operations, physical security, radiation control, housekeeping, emergency preparedness, chemistry, surveillances, and QC hanger inspections. The inspectors also reviewed licensee action on previous inspection findings, observed TN-9 spent fuel shipping cask receipt and handling activities, and followed licensee action taken to identify and correct a leak of radioactive liquid to the environment from the Augmented Offgas building underground piping. The inspectors also followed up a NRC concern regarding potential for overpressurization of low pressure systems.

DETAILS

1. Summary of Plant Activities

At the beginning of the report period, the plant was operating at 635 MWe and was continuing to experience difficulties staying below the core thermal limits for MAPLHGR. A rod sequence change was performed which alleviated the MAPLHGR concern. Plant power remained at or close to rated output throughout most of the period. The drywell unidentified and identified leakrates remained stable at values well below Technical Specification (TS) limits.

As discussed in Inspection Report 85-13, V-14-35 ('B' Isolation Condenser condensate isolation valve) failed to open during surveillance testing and was declared inoperable. The valve was declared operable after several days of trouble shooting that was inconclusive as to the cause of the valve's failure to open. The surveillance frequency was changed from monthly to weekly. A decrease in V-14-35's closing thrust, as part of the troubleshooting efforts, resulted in leakage past the valve seat and a substantially higher Isolation Condenser shell temperature. The temperature increase has resulted in vapor wisping from the isolation condenser shell side vent on the east side of the reactor building.

Other plant problems included failure of the #2 seal on 'E' recirculation pump, continuing secondary side steam leaks, inconsistent reactor water level indications in the Control Room, and erratic operation of the Augmented Offgas system. A condenser salt water leak occurred in the 'B' north condenser that required a power decrease for several days while the leak was identified and repaired. Concerns for low condenser efficiency were resolved based on the results of a thorough heat balance indicating a condenser efficiency of 90%. Problems persisted with the hydrogen/oxygen monitoring system in that nearly every surveillance test finds a channel inoperable.

The 'A' Emergency Service Water (ESW) pump was declared inoperable when surveillance testing indicated discharge pressure less than the design value. This was subsequently resolved when an interface with the Chlorination system that resulted in bypass of ESW flow was identified. HVAC problems plagued the plant throughout the period; no air conditioning in the office building, trunnion room temperature increase to 160°F (from normal 125°F) with no apparent cause, and feedpump room ventilation inadequacies resulting in power reductions so that feedpump motor stator bearing temperature limits were not exceeded. Efforts are continuing to resolve these problems. A leak of radioactive liquid through the AOG building sump pump discharge pipe into the surrounding dirt was identified and repaired. Approximately 98 cubic feet of dirt was contaminated. A small fire occurred in the turbine building elevator machinery room that did not cause any damage. The fire apparently resulted from a short circuit in an electric motor.

Spent fuel from West Valley continued to be shipped to the site on a somewhat reduced frequency due to problems with the cask trailer. Additionally, a spacer used in the TN-9 cask was inadvertently placed in the spent fuel pool and resulted in one shipment of six elements in lieu of the normal seven.

The annual emergency drill was conducted on June 5, 1985, and was evaluated by NRC observers as a good exercise. The details are included in NRC Inspection Report 85-17.

2. Licensee Action on Previous Inspection Findings

(Close) Inspector Followup Item (219/82-21-01): Non-redundant torus water level piping is not consistent with NUREG 0737 guidelines.

By letter dated April 3, 1985, from P. B. Fiedler to D. G. Eisenhut, GPUN explained they had not identified the deviation from NUREG 0737 requirements in previous correspondence. They also offered a justification for the acceptability of the deviation. In a telephone conversation between the NRC resident inspectors and the licensing project manager, it was explained that NRC licensing had accepted the GPUN position and that a Safety Evaluation (SE) would be forthcoming in June or July of 1985. The SE will document the acceptability of the 27 feet of non-redundant torus water level piping.

(Close) Inspector Followup Item (219/82-21-02): Generic acceptance documentation for qualification of Rosemount transmitters was not available at the corporate office.

In subsequent licensee correspondence it was clarified that the qualification data had been available for at least 1½ years prior to the NRC inspection. The correspondence stated that data was on file at the Cherry Hill office and was listed in the Environmental Qualification Test Report Index. It appears that licensee personnel involved with the NRC inspectors in Inspection 82-21 were not aware of the existence of the data nor did they know how to track it down. The licensee has issued memos to notify personnel of the existence of the Equipment Environmental Qualification File and the Report Index.

(Close) Inspector Followup Item (219/82-21-05): Lack of an active interface between the General Office Review Board (GORB) and the QA Standing Committee.

Amendment No. 69 to GPUN's provisional operating license deleted this TS requirement. In the SE, NRR stated the GORB does not perform a function required by the NRC, therefore, its deletion from the TS is acceptable. It should be noted that prior to deletion of the TS requirement, the licensee did address the lack of an active interface by modifying procedures and changing the membership of the QA Standing Committee to include the plant Vice President and Director and other key management personnel.

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

Followup inspection by the licensee's QC organization resulted in a letter confirming the NRC inspectors' observations that there was no visible evidence that the bushings were staked. This is contrary to a letter sent to GPUN by the strut rod manufacturer wherein it is stated a QC inspection was performed to ensure the bushings were staked. As part of the inspection activity to attempt to close out this item, the NRC resident inspector and the GPUN QC Manager reinspected the particular strut rods identified in Inspection Report 84-09. The conclusion reached was that the strut rod would have to be removed from the clevis to perform a thorough inspection. Because the plant was in an operational status, it was not possible to disassemble the affected supports. This item will remain open pending inspection of the strut rod bushings to determine if the bushings are staked.

(Open) Inspector Followup Item (219/85-06-02): Observe future Isolation Condenser surveillance of valve V-14-35 to verify operability.

At the beginning of this report period, V-14-35 had been declared inoperable because of its failure to open during surveillance testing. Troubleshooting to identify the cause of the problem was in progress, and the licensee was in a seven day TS limiting condition for operation. Troubleshooting during this report period did not identify the cause of the problem, however, the closing thrust of the valve was decreased as a possible fix. At the first opportunity, the licensee intends to disassemble the valve and inspect it as part of the troubleshooting effort. In addition, the surveillance frequency has been increased from monthly to weekly to improve valve operability. Long range plans may involve gradually decreasing the frequency back to the TS required monthly frequency. Based on the reduction of closing thrust and an increased surveillance frequency, the licensee declared the valve operable before a plant shutdown would have been required by TS. The valve passed all weekly surveillances performed during this report period.

As a result of decreasing the closing thrust on the valve, the valve started leaking by the seat. This resulted in an increase in the 'B' Isolation Condenser shell side water temperature into the 190°F range and production of water vapor that appears at the Isolation Condenser vent.

The inspectors will continue to track the performance of V-14-35 and the troubleshooting efforts to determine the cause of the valve's periodic failure to open.

3. Management Meeting

On May 20, 1985, a management meeting was held in the Region I office to discuss NRC concerns regarding an apparent problem with licensee

identification, coordination, technical evaluation, tracking, follow-up, and closeout of several key issues that are the responsibility of the Technical Functions Division. The meeting was precipitated by the findings identified by a NRC inspection conducted the week of May 15 and documented in Inspection Report 85-14. The subject of the NRC inspection was closeout of Bulletins 79-02, 79-04, 79-07, and 79-14. Bulletins 79-02 and 79-14 had a major impact on Oyster Creek, as well as other nuclear power plants, regarding the amount of technical effort required to address the issues raised. The inspection determined that documentation of the technical efforts put forth by the licensee to close these two Bulletins were inadequate. (See Inspection Report 85-14 for details.) Because of the concerns raised by the inspection findings documented in Inspection Report 85-14 and others in the past, a meeting was requested by NRC Region I to offer GPUN management an opportunity to explain their plans and in progress activities to correct the concerns.

NRC explained to GPUN that they were concerned about internal management of Tech Functions and their ability to interface effectively with other divisions within GPUN. Particular examples offered to substantiate this concern included inaccurate as-built drawings, questionable seismic analyses and SEP input data, and lack of adequate preparation and knowledgeable personnel to support an announced NRC inspection. NRC requested that GPUN explain how Tech Functions manages responsibilities and determines the effectiveness of their management.

GPUN Tech Functions presented a summary of their organization and described the many programs both in process and proposed to upgrade the particular areas of as-built drawings, document control, design control, engineering, and vendor documentation. They stated that when the original plant was built in the 1960's, there was no as-built documentation at the time of turnover of the plant to JCP&L. This resulted in a problem with baseline accuracy of all drawings. They explained that drawing control systems in use since GPUN took over the license in 1982 from JCP&L have resulted in accurate representation of post 1981 modifications. They stated that Bulletins 79-02 and 79-14 were considered closed in February 1980 based on walkdowns and engineering evaluations of as found conditions conducted in 1979 and 1980. They agreed the quality of this work was in question based on the results of the NRC inspection and also agreed the validity of piping and support data submitted to NRC licensing as part of the SEP evaluation is in question.

They explained they have been aware of the problem with inaccurate as-built drawings and have had a program underway to correct the problem. The program is expected to cost approximately $800 million and be completed in 1987. It involves walkdowns of mechanical and electrical systems to verify accuracy of existing drawings and making changes when appropriate. GPUN stated, that effective the week of 5/20/85, new walkdowns of all accessible safety-related piping systems would commence to determine the accuracy of drawings used in piping analyses and that this effort would be complete in two to three weeks. It was agreed

between the NRC and GPUN that a followup meeting should be arranged to discuss the results of GPUN's walkdown findings.

GPUN stated they did not know why the results of the original walkdowns of the safety-related piping systems in 1979 and 1980 were not incorporated into as-built drawings despite a commitment in their Bulletin response letter to accomplish this activity. They also stated they were not clear as to why they were not prepared to support announced NRC inspection 85-14, but did state much of the documentation needed to support the inspection was in unmarked boxes turned over to GPUN from JCP&L in 1981-82. GPUN stated a major effort has been underway to sort through the boxes and properly file the information so that it will become identifiable and retreivable.

In summary GPUN responded to the NRC concerns in a general fashion by describing their overall organization and explaining initiatives underway to address them. The GPUN response was in broad terms as was requested by NRC management. It was agreed that future meetings are required to discuss specific concerns after GPUN Tech Functions has had an opportunity to formulate and initiate a plan of action and evaluate its results.

4. Operational Safety Verification

4.1 Control Room Observation

Routinely throughout the inspection period, the inspectors independently verified plant parameters and equipment availability of engineered safeguard features. The following items were observed:

-- Proper Control Room manning and access control;

-- Adherence to approved procedures for ongoing activities;

-- Proper valve and breaker alignment of safety systems and emergency power sources; and,

-- Shift turnover.

4.2 Review of Logs and Operating Records

The inspectors reviewed the following logs and instructions for the period May 6 to June 2, 1985:

-- Control Room and Group Shift Supervisor's logs, all entries;

-- Reactor Building and Turbine Building Tour sheets;

-- Equipment Control Logs;

-- Standing Orders; and

-- Operational Memos and Directives.

The logs and instructions were reviewed to:

-- Obtain information on plant problems and operation;

-- Detect changes and trends in performance;

-- Detect possible conflicts with Technical Specifications or regulatory requirements;

-- Assess the effectiveness of the communications provided by the logs and instructions; and

-- Determine that the reporting requirements of Technical Specifications are met.

The reviews indicated the logs and operating records were generally complete. No inspector concerns were identified.

5. Observation of Physical Security

During daily entry and egress from the protected area, the inspectors 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 from 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.

6. Plant Tours

During the inspection period, the inspectors made frequent tours of the plant areas to make an independent assessment of equipment conditions, radiological conditions, safety, and adherence to 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.

6.1 Radiation Protection

-- Personnel monitoring was properly conducted.

-- Randomly selected radiation protection instruments were calibrated and operable.

-- Radiation Work Permit requirements were being followed.

-- Area surveys were properly conducted and the Radiation Work Permits were appropriate for the as-found conditions.

6.2 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.

6.3 Equipment Controls

-- 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.

6.4 Vital Instrumentation

-- Selected instruments appeared functional and demonstrated parameters within Technical Specification limiting conditions for operation.

6.5 Radioactive Waste System Controls

-- Gaseous releases were monitored and recorded.

-- No unexpected gaseous releases occurred.

6.6 Housekeeping

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

No inspector concerns were identified.

7. Safety System Operability Verification

On a sampling basis, the inspector directly examined selected safety system trains to verify that the systems were properly aligned in the standby mode. This examination included:

-- Verification that each accessible valve in the flow path was in the correct position by either visual observation of the valve or remote position indication;

-- Verification that power supply breakers were aligned for components that must actuate upon receipt of an initiation signal;

-- Visual inspection of the major components for leakage, proper lubrication, cooling water supply, and other general conditions that might prevent fulfillment of their functional requirements; and,

-- Verification by observation that instrumentation essential to system actuation or performance was operational.

During this inspection period, the following system was examined.

-- Accessible portions of the Isolation Condenser system.

No inspector concerns were identified.

8. Emergency Preparedness

During this report period, the licensee conducted the annual emergency drill. The NRC did not actively participate in the drill but did observe licensee performance. The details are described in NRC Inspection Report 85-17. The licensee, FEMA, and the NRC considered the drill a success.

9. 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.

During the unloading operations of shipment #22, a spacer stuck to the bottom of a spent fuel assembly and was inadvertently removed from the TN-9 shipping cask and placed in a spent fuel rack along with the spent fuel assembly. It was not realized the spacer was missing until spent fuel for shipment #24 was being loaded into the cask at West Valley. Upon realization that the spacer was missing, the licensee identified the spent fuel assembly to which the spacer was stuck and removed the spacer. It was then replaced in the appropriate cell of the TN-9 cask used for shipment #24.

Problems with crack propagation of a plate welded to the support I-beam on one of the cask trailers were identified during this report period. This resulted in delays of spent fuel shipments while investigations were conducted to determine the structural integrity of the trailer. The issue was resolved after determining that this problem had no adverse affect.

10. Followup of Operational Events

Water was observed leaking back through a piping penetration in a wall in the Augmented Offgas (AOG) building. The penetration was through the underground section of the wall and the pipe carried the discharge from the AOG building sump. The leaking water was sampled and found to be radioactive. The ground outside the AOG building above the pipe was removed, and a hole in the pipe about the size of a dime was identified. Approximately 98 cubic feet of contaminated dirt had to be removed as part of the cleanup effort. The affected pipe was replaced and the excavation refilled. The hole in the pipe was determined to be caused by external corrosion due to a holiday in the protective wrapping around the pipe.

A system of 13 wells exists onsite to provide a means to monitor the quality of well water. No radioactivity has been detected in any wells. No inspector concerns were identified during inspection of the licensee's corrective action to locate and repair the leak and identify the extent of the contaminated soil.

11. Event "V" Review

A review of the Emergency Core Cooling System (ECCS) was conducted to assess the potential to overpressurize the low pressure portion of the ECCS piping. This overpressurization of low pressure systems has been referred to as "Event V". These types of events have occurred at other facilities and is a concern of the industry. This inspection was conducted in accordance with regional guidance, using information from NUREG/CR-2069, "Summary Report on a Survey of Light-Water-Reactor Safety Systems" and IE Information Notice No. 84-74, "Isolation of Reactor Coolant System from Low-Pressure Systems Outside Containment". In accordance with the guidance provided, the Core Spray system was examined at Oyster Creek. This has been determined to be the only system affected.

11.1 Component Configuration

The following information is provided in response to the inspection guidelines:

-- Component configuration as given in NUREG/CR-2069

Interfacing System: Core Spray System I
Piping Location: In
Number of Penetrations: 2 - 8" in diameter
Component Line-Up

$$\text{RPV-MV-}\begin{bmatrix}\text{CK}\\ \text{CK}\end{bmatrix}\text{I-}\begin{bmatrix}\text{MOV}\\ \text{MOV}\end{bmatrix}\text{MOV-H/L-PRV-CK-P}$$

LO AO* NC NO

Note: *Air Operated

L - Low pressure (PSIG) 400#

H - High Pressure (PSIG) 1250#

Interfacing System: Core Spray System II
Piping Location: In
Number of Penetrations: 2 - 8" in diameter
Component Line-Up

$$\text{RPV-MV-}\begin{bmatrix}\text{CK}\\ \text{CK}\end{bmatrix}\text{I-}\begin{bmatrix}\text{MOV}\\ \text{MOV}\end{bmatrix}\text{MOV-H/L-PRV-CK-P}$$

LO AO* NC NO

-- Testable check valves with an air operator

Core Spray System I
V-20-152 (NZO2C)
V-20-150 (NZO2A)

Core Spray System II
V-20-153 (NZO2D)
V-20-151 (NZO2B)

The air operators for the above listed testable check valves are maintained operable and used to test the opening function of the check valves.

-- Isolation Valves (Parallel)

Core Spray System I
V-20-15
V-20-40

Core Spray System II
V-20-21
V-20-41

-- System Discharge Isolation

Core Spray System I
V-20-12

Core Spray System II
V-20-18

Note: V-20-12 is interlocked with V-20-15 and V-20-40 and V-20-18 are interlocked with V-20-21 and V-20-41. When reactor pressure is greater than approximately 285 psig, this interlock prevents opening one of the parallel isolation valves (normally closed) with the respective system discharge valve open (normally open and de-energized).

11.2 Surveillance Activities

The following surveillance activities that apply to the Core Spray system were reviewed.

-- Procedure 610.3.006 Core Spray Isolation Valve Actuation Test and Calibration.

Test Type

Valve stroking (Pressure switch actuation)
Logic testing (Pressure switch actuation)
accomplished monthly.

Test Conditions

Plant above or below 350 psig.

Summary of Test Steps

1) Isolate pressure sensor from the reactor.

2) Connect voltmeter on sensor to terminals in order to be able to verify trip point for is lation valve (V-20-15, V-20-40, V-20-21, & V-20-41) closure.

3) Connect test pump with a calibrated gauge to the pressure sensor and increase pressure to approximately 400 psig.

4) Jumper auto start relay.

5) Bleed off pressure on pressure sensor until voltmeter deflects and the isolation valve opens.

6) Remove jumper in step 4, verify isolation valves shut.

7) Readjust trip point if found out of specification.

8) Return system to normal.

Precautions/Prerequisites to prevent overpressurization

1) If the pressure switches require readjustment, the jumper across the "auto start" relay shall be disconnected to avoid unnecessary cycling of parallel isolation valves.

2) Obtain permission from the Group Shift Supervisor to commence the test.

3) If reactor pressure is ≤ 350 psig, shut the parallel isolation valves and open the supply breaker. Verify the isolation valves are de-energized by attempting to open.

Acceptance Criteria

This test shall be considered acceptable provided:

1) The trip point of RE17A* is ≥ 294 psig (as found) and 309 ± 5 psig (as left).

2) The trip point of RE17B* is ≥ 291 (as found) and 306 ± 5 psig (as left).

3) The trip point of RE17C* is ≥ 294 psig (as found) and 309 ± 5 psig (as left).

4) The trip point of RE17D* is ≥ 291 psig (as found) and 306 ± 5 psig (as left).

Note: *-For parallel isolation valve actuation to open position.

Regulatory Requirements for Test

None

- Procedure 610.4.003 Core Spray Valve Operability and In-Service Test

Test Type

Valve stroking
Accomplished monthly

Test Conditions

Plant above or below 350 psig.

Summary of Test Steps

1) Unlock and close breaker for V-20-12 (V-20-18), system discharge valve (high-low pressure interface isolation; MOV upstream of parallel isolation valves).

2) Shut and time V-20-12 (V-20-18). Do not reopen at this time.

3) Turn off supply breaker for V-20-12 (V-20-18).

4) Open parallel isolation valves (V-20-15 & V-20-40 or V-20-21 & V-20-41) recording opening times.

5) Shut and time parallel isolation valves.

6) Turn on supply breaker for V-20-12 (V-20-18).

7) Open and time V-20-12 (V-20-18) and then open and lock supply breaker.

Precautions/Prerequisites to prevent overpressurization of low pressure piping.

1) Obtain permission from Group Shift Supervisor to initiate the surveillance.

2) System discharge isolation valves are shut and de-energized prior to testing parallel isolation valves.

Acceptance Criteria

The components tested by this procedure meet TS and In-Service Test requirements for operability if the following criteria are met:

Opening time for parallel isolation valves: ≤ 22 seconds;

Closing time for parallel isolation valves: ≤ 60 seconds;

Closing time and opening time for system discharge valves: ≤ 60 seconds.

Regulatory Requirement:

1) Technical Specifications Sections 3.4.A and 4.4.A.2
2) Inservice Test

-- Procedure 610.4.008 Core Spray Testable Check Valve Operability Test

Test Type

Valve stroking
Accomplished during every refueling shutdown

Test Conditions

Refueling shutdown; reactor coolant system is less than 212°F and vented. Drywell access is available.

Summary of Test Steps

1) Verify the Core Spray parallel isolation valves are shut or the manual isolation valve in the drywell is shut.

2) Establish communication between the control room and an operator in the drywell. The operator in the drywell will observe the pivot arm rotation on each valve.

3) Open and time the testable check valves from the control room.

4) Verify local mechanical and remote (control room) electrical position indications agree.

5) Shut and time the testable check valves.

6) Verify each valve shuts by observing pivot arm rotation. Verify local mechanical and remote electrical position indication agree.

7) Open manual isolation if closed in step 1.

Precautions/Prerequisites to Prevent Overpressurization of Low Pressure Piping

1) Reactor coolant system is less than 212°F and vented.

2) Notify the Group Shift Supervisor when any unsatisfactory results are obtained during the performance of this test.

Acceptance Criteria

1) Valves open and close as specified in procedure. Verified by local position and/or by control room position indication.

2) Local and remote position indications agree.

Regulatory Requirements

1) Technical Specification 3.4.A

2) ASME Section XI

-- Procedure 610.4.011 Core Spray System Testable Check Valve Leakage and In-Service Test

Test Type and Frequency

Leak Testing
Refueling

Test Conditions

Test must be performed prior to exceeding 600 psig reactor pressure and the test differential pressure across the valve must be greater than 150 psig.

Summary of Test Steps

1) Connect test rig upstream of the testable check valves (V-20-153 and V-20-151; V-20-152 and V-20-150) and downstream of parallel isolation valves (V-20-21 and V-20-41; V-20-15 and V-20-40) to a test connection line (V-20-44 and V-20-45; V-20-42 and V-20-43).

2) The test rig consists of a pressure gage, an isolation valve and a 55 gallon drum equipped with a sight gauge and a vent filter.

3) The test connection isolation valves (V-20-44 and V-20-45; V-20-42 and V-20-43) are opened fully and the pressure allowed to stabilize as indicated on the test rig pressure gauge (gauge should indicate reactor pressure).

4) The isolation to the 55 gallon drum (V-1) is slowly opened. As water flow is established pressure should decrease - if not isolate the test rig.

5) When the pressure gauge stabilizes, time the water level increase for one minute.

6) Record data and isolate test rig.

7) Return lineup to normal.

Precautions/Prerequisites to Prevent Overpressurization of Low Pressure Piping.

1) Verify testable check valves are closed using the remote indication.

2) Verify parallel isolation valves are closed using the remote indication.

Note: The test connection line is a 3/4" line with two manual isolation valves in series in the line. This test is conducted with the parallel isolation valves shut prior to exceeding 600 psig reactor pressure.

Acceptance Criteria

1) Leakage rate is ≥ 1 gpm.

2) Leakage rates greater than one (1) gpm, but less than five (5) gpm, are acceptable provided that the leak rate measured in this test does not reduce the margin to maximum permissible leakage rate of 5 gpm by 50 percent or greater.

i.e.,

$$\text{Measured Leakage Rate (this test)} \leq \left[\frac{(5.0 - \text{Previous Leakage Rate})}{(2)} + \text{Previous Leakage Rate}\right]$$

Note: Leakage rates >5 gpm are unacceptable at any time.

Regulatory Requirements

1) Technical Specification 4.3.F
2) ASME Section XI IST Program

11.3 Maintenance Activities

Maintenance activities involving the Core Spray system were reviewed as follows:

The equipment history cards were reviewed for the Core Spray isolation valves (V-20-12, 18, 15, 40, 41, 21, 150, 151, 152, and 153) including both the electrical and mechanical portions. In addition, selected short forms were reviewed to further clarify some equipment history card entries. In general equipment history card entries were first recorded in 1981 for these valves. Therefore, the data is based entirely on entries made over the last four years. The following conclusions were made regarding corrective maintenance performed on these valves:

-- Replacement and adjustment of packing was performed 17 times.

-- Regreasing of limitorque operators was performed for all six limitorque operators on 4/4/84.

-- MOVATS testing was performed on the four parallel isolation valves on 1/11/85.

-- Inspection and repair of a limitorque occured once.

-- The air actuator seal on the testable check valves was replaced on two valves on 5/8/81.

-- One testable check valve failed to stroke on 3/26/84.

-- Rebuilding of a motor was required to be performed 5 times.

-- Six circuit breakers were replaced in March 1982, as a result of an engineering concern. Plant engineering developed a concern as a result of the magnetic breaker tripping upon multiple stroking of a valve. In 1982, General Electric provided their concurrence with Oyster Creek's engineering analysis and recommendation to replace the magnetic only breakers with thermal magnetic breakers.

-- On 1/15/85 an interlock contact failed open for V-20-12; licensee adjusted the contact to correct the problem.

The following preventive maintenance (PM's) routines are conducted every refueling outage:

480V-MCC-V-20-15 Core Spray Isolation Valve
480V-MCC-V-20-12 Core Spray Pump Discharge Valve
480V-MCC-V-20-40 Core Spray Isolation Valve
480V-MCC-V-20-21 Core Spray Isolation Valve
480V-MCC-V-20-41 Core Spray Isolation Valve
480V-MCC-V-20-18 Core Spray Pump Discharge Valve

These electrical PMs are conducted in accordance with Procedure 732.2.004, 480V Motor Control Center Preventive Maintenance. The purpose of this procedure is to provide detailed instructions for

inspection, adjustment, testing, and repair of 480 volt motor control center breakers.

The licensee is developing a program for sampling and greasing Limitorque operators on an every refueling outage basis. This program is presently planned to be implemented by the end of cycle 11 outage. There are no other PM's conducted on the testable check valves, parallel isolation valves or V-20-12(18).

PS-27 A&B located between the parallel isolation valves and V-20-12(18) will initiate a control room alarm at 300 psig indicating pressurization of the piping. PS-27 A&B are calibrated every February at 300 (+0,-8) psig.

Pressure relief valves V-20-24(25) (set to lift at 350 psig) are not routinely checked. These relief valves would relieve any pressure between V-20-12(18) and the pump discharge check valve. These valves were ov rhauled and tested by Wiley labs during the last outage. This overhaul cycle is done in accordance with the requirements of ASME Section XI IW V35.10.

The relays associated with the Core Spray system are D.C. relays which are not normally energized. Preventive maintenance for the Core Spray system relay is performed in accordance with 900I HFA relay inspection (monthly) and 1222E HFA relay inspection (monthly). The interlock contact relay on V-20-15 failed on 1/15/85 preventing the system discharge isolation valve from opening.

The inspector did not identify a standard component replacement policy. Currently, the licensee reviews equipment history cards for repetitive failures. The licensee is planning to implement a computer system to manage many of the maintenance activities including equipment histories, repetitive work orders and NPRDS failure reports.

11.4 QC Coverage

Quality Control (QC) is notified of any important to safety system maintenance activity by the work order requesting the maintenance. The QC Group reviews the work order and the maintenance and construction procedure governing the activity to develop the QC hold points required. In addition, QC insures that Inservice Inspection (ISI) and In Service Test (IST) requirements are identified by plant engineering on the work request. The Core Spray valves of interest (V-20 - 12, 15, 18, 21, 40, 41, 150, 151, 152, 153) are all listed on the ISI/IST list.

QC has developed QC hold/ witness point checklists for maintenance procedures that are repetitive. This hold point list would be reviewed against the maintenance procedure to determine if additional hold points were required or if the hold point list may be used as written. For major maintenace activities, QC engineering would develop an inspection surveillance plan from which QC would develop a hold point checklist to govern the major maintenance action. QC engineering would also develop specific inspection surveillance plans when requested by QC. Industry events and Oyster Creek maintenance experience are evaluated to determine if revisions are necessary to particular hold point checklists.

In addition, corporate QC is presently developing inspection guides for particular types of equipment. For example, an inspection guide will be developed to provide inspector guidance when examining a gate valve or globe valve, etc. These guides will not be specific for any one piece of equipment, but will be written to encompass particular types of equipment. These guides would then be used by site QC to develop hold point checklists.

For the area of interest of this survey, the following QC hold/witness point checklists have been developed:

Procedure 700.1.020 Inspection of Bolted Bonnet Gate Valves [this applies to V-20-12(18), V-20-15(40), V-20-21(41)];

Procedure 700.2.010 Motor Operated Valve Removal, Installation or Inspection (Elect.) (V-20-12, 18, 15, 40, 21, 41);

Procedure 710.1.004 8" Core Spray Check Valve Maintenance (V-20-150, 151, 152, 153); and,

Procedure 710.1.007 Removal/Installation of Core Spray Relief Valves (V-20-24, 25).

11.5 Pre-Maintenance Checks

Pre-maintenance checks and precautions to assure proper removal of equipment from service are delineated in the licensee's procedures. Procedure 105, Conduct of Maintenance, uses a "Short Form" to request maintenance activities. Procedure 108, Equipment Control, governs the release of equipment from operation for maintenance. Short Forms are normally reviewed every day by the Manager Plant Operations. This review is conducted to identify important-to-safety operational limits and special conditions required for maintenance work. The Manager Plant Operations informs the Group Shift Supervisor of the maintenance activities scheduled. The Short Form will indicate whether switching and tagging is required and also requires GSS signature to start work.

Procedure 108 requires completion of the Switching and Tagging Request Form (Form 108-1) to identify the equipment/system to be tagged and the reason for the outage. Procedure 108 requires the GSS to review Form 108-1 for license requirements, station operating conditions and compatability with other equipment outages. The GSS shall specify the testing required prior to releasing the equipment for work. The testing is determined by considering the Technical Specifications, operating requirements, and plant status.

Operations personnel stated that they would not allow more than one maintenance or surveillance activity to be performed on a system at any one time.

11.6 Post Maintenance Testing

Post Maintenance testing and return to service is conducted in accordance with the requirements of Procedure 105. The Job Supervisor is required to determine the necessary post maintenance requirements and to enter them on the Short Form. The Job Supervisor is also responsible to accomplish the post maintenance test requirements prior to turning the system over to operations. Before considering a repaired component as returned to service, the shift supervisor shall demonstrate that the components worked on are operable. System testing shall also be performed as necessary to determine that the system is operable. The shift supervisor will usually specify the surveillance normally conducted on the system to demonstrate system operability.

11.7 Potential Overpressurization Events

Throughout the number of interviews conducted, licensee personnel could not recall an incident that had the potential to overpressurize the low pressure portion of the Core Spray system.

LER 85-002 on the reactor water cleanup system discusses failure of two containment isolation valves in the same penetration. The valves' failure to close upon demand resulted in the system relief valve lifting and passing approximately 1800 gallons of water to the torus before the system was isolated.

11.8 Training and Licensee Review of Industry Events

Selected operations and maintenance personnel were interviewed to determine whether facility personnel are formally trained in and familiar with procedural requirements for proper surveillance and maintenance of the Core Spray isolation valves.

The control room operators training program requires the trainees to perform system surveillances on all the safety systems. This is documented on the control room operator qualification card and is in addition to the formal classroom training on particular system

operation. The shift supervisors that were interviewed stated that they would not allow more than one surveillance to be performed on a system at the same time. The personnel interviewed were familiar with the surveillance requirements of the procedures.

The maintenance personnel interviewed were familiar with the procedures and knowledgable regarding maintenance tasks performed on the Core Spray valves. However, they have not received formal training on the maintenance of these isolation valves. The training they have received was fundamental in nature; supplemented by on the job training. In general, these particular maintenance personnel felt that some training would be beneficial. Some concerns were raised about pre-job training for a particular maintenance action. The usual method employed is to try to determine who has in the past performed this maintenance action and what particular problems they may have encountered. Apparently, on some jobs, it is difficult to obtain the appropriate drawing or vendor manuals associated with a piece of equipment.

The licensee in response to IE Information Notice No. 84-74, "Isolation of Reactor Coolant System from Low-Pressure System Outside Containment", has reviewed the following Oyster Creek surveillance and maintenance procedures associated with the Core Spray System:

610.3.005 Core Spray System Instrument Channel Calibration and Test;

610.3.006 Core Spray Isolation Valve Actuation Test and Calibration;

610.4.003 Core Spray Valve Operability and In-Service Test;

610.4.007 Core Spray System Firewater Valve Test;

610.4.008 Core Spray Testable Check Valve Operability Test;

610.4.011 Core Spray System Testable Check Valve Leakage and In-Service Test;

610.4.012 Core Spray Pump In-Service Test; and

710.1.004 8" Core Spray Check Valve Maintenance.

The licensee's review determined that if procedural compliance was observed in carrying out the surveillance and maintenance actions listed above, that similar events described in the IE Information Notice would not occur at Oyster Creek. In addition, the review recommended that Core Spray system logics be examined with regard to the IE Information Notice.

11.9 Review of Industry Events with regard to the Oyster Creek Core Spray System

The NRC issued an order to Jersey Central Power & Light Company (Oyster Creek) dated April 20, 1981 to measure the leakage past the testable check valves. This was based on a technical evaluation report, concerning primary coolant pressure system pressure isolation valves, completed by the Franklin Research Center. The Franklin Research Center examined the high-low pressure interface between the reactor coolant system and the Core Spray system and concluded that the leakage past the testable check valves should be measured. No other significant recommendations were made concerning the Core Spray system's isolation valves.

The Oyster Creek Core Spray system isolation valve configuration is not of the same arrangement as delineated in NUREG/CR-2069, "Summary Report on a Survey of Light-Water- Reactor Safety System". A combination of at least three failures or errors would be required to overpressurize the low pressure portion of the Core Spray system piping.

In reviewing past industry events and errors that led to initiation of overpressure events, the following problems were reviewed at Oyster Creek to determine if a combination of these failures could lead to an overpressurization of the Core Spray piping.

-- One facility experienced problems with improper maintenance on air solenoid valves. Oyster Creek will usually replace the entire air solenoid valve rather than perform maintenance on the valve. This is not a formalized procedure or requirement, but a preference of the maintenance organization.

-- Another facility reversed the air line on the air actuator for a testable check valve. This was explored from the aspect of physical arrangement of the air lines attaching to the air actuator and from procedural requirements and precautions. The physical arrangement of the air line could not be verified, but discussions with maintenance personnel determined that the air lines for one testable check valve were the exact length required to attach to their respective actuator ports. No particular precautions were delineated in the procedure to insure proper air line connection. The performance of the surveillance procedure for the Core Spray testable check valve requires verification of the local indication with the control room indication, which would preclude the existence of improperly connected air actuator lines during reactor operations.

-- Improper performance of surveillance procedures was another cause of overpressurization events. Surveillances of safety

systems conducted at Oyster Creek are done one at a time. The shift supervisors interviewed stated that they would not allow more than one safety system surveillance to be conducted per system. The parallel isolation valve operability surveillances require V-20-12(18) to be shut and the breaker to be de-energized prior to stroking the parallel isolation valve. This requirement, in addition to the testable check valves, protects the low pressure piping. A pressure switch, located between the parallel isolation valves and V-20-12(18) provides additional assurance. This pressure switch alarms at 300 psig indicating a pressure increase between these valves. One additional protection feature is the interlock between the parallel isolation valves and V-20-12(18). This interlock prevents opening the parallel isolation valves when the reactor pressure is greater than approximately 285 psig and V-20-12(18) is open. As mentioned in Section 11.3, a failure did occur in a relay for this interlock. This is a D.C. relay which is not normally energized, thus the failure rate is expected to be low.

The industry events reviewed do not seem likely to occur at the Oyster Creek Nuclear Generating Station. The procedural requirements and series interlocked isolation valves and testable check valves make an overpressurization event of the Core Spray system piping improbable. A combination of at least three failures or errors would have to occur to initiate overpressurization of the low pressure Core Spray piping.

12. Surveillance Testing

The inspectors reviewed and witnessed performance of the following surveillance test to determine if the test was included on the master surveillance schedule, was technically adequate, and was performed at the required frequency and in accordance with procedure.

-- Isolation Condenser Valve Operability and In-Service Test, 609.4.001, Rev.12

From this review it was determined the surveillance was performed in accordance with the procedure. The frequency of performance of this surveillance was temporarily changed from monthly to weekly because of problems with the performance of V-14-35 as discussed and tracked in NRC Outstanding Item 219/85-06-02. It was observed that V-14-35 was instrumented with accelerometers and thermocouples as part of the troubleshooting efforts and that MOVATS traces were recorded on the valve when it was cycled open and closed.

13. Management Activities

13.1 During this report period, management changed the format and content of the daily report. The change resulted in a much improved report that details significant plant problems and their potential impact on plant activities.

13.2 At the invitation of plant management, the inspectors attended a Project Status Review Meeting. The purpose of the meeting is to review in detail the scheduled work items for upcoming outages and to plan how best to accomplish them. Representatives from all of the affected departments were in attendance at this weekly meeting. The inspectors felt that the meetings are constructive and should result in improved licensee performance in future outages.

13.3 The inspectors were also invited to attend a Preliminary Engineering Design Review (PEDR). This meeting is chaired by Tech Functions and its purpose is to review in detail proposed design changes from such points of view as need, how best to write the design specifications, and selection of the best design and equipment. Attendees at the meeting include departments with engineering interface such as Plant Engineering, Plant Operations, Purchasing, and Startup and Test. The inspectors observed that interplay was active amongst meeting participants and that the particular proposed modification that was the subject of the PEDR was disected in detail. The inspectors felt from their part time observations of this PEDR that the purpose of the meeting was accomplished and that the detailed analysis of the subject matter by the meeting participants cannot help but result in ar improved engineering design, design specification, and technical specification.

14. Pipe Hanger Inspections

In response to the concerns raised in the Management Meeting discussed in paragraph 3 of this report, the licensee commenced a QC inspection of the as-built condition of safety-related piping systems and supports to meet the requirements of NRC Bulletin 79-14. The inspectors observed QC inspection activities and reviewed MNCR's generated as a result of these activities. The QC inspections during this report period identified a substantial number of discrepancies with pipe supports. However, a preliminary engineering judgement review of the discrepancies by Tech Functions pipe stress engineers concluded that none of the discrepancies required a shutdown of the plant due to an unsafe condition. A similar engineering judgement review of the affects on each completely inspected system when, all the discrepancies are considered in bulk, was in progress at the end of the report period. A subsequent computer analysis of the significant discrepancies is planned.

A major problem, although not directly related to the Bulletin, is identifying itself through the inspection activities. In particular, the QC search to determine the most up-to-date drawings for the inspections, is revealing that the configuration control system is either not working and/or is substantially behind in drawing updates. Additionally, it has been determined that MNCRs dispositioned "use-as-is", that reflect as-built conditions different from those shown on the drawings of record, do not get incorporated onto the as-built drawings. QA is aware of these

problems and is taking corrective action. The inspectors are continuing to follow the inspection activities and evaluating, on a sampling basis, the significance of the discrepancies documented by MNCRs. It should be noted that as of the end of this report period, there was about an one to one correspondence between systems and hangers inspected and the number of MNCR's generated.

15. 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.