U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No:	98-12
Docket No:	50-219
License No:	DPR-16
Licensee:	GPU Nuclear Incorporated 1 Upper Pond Road Parsippany, New Jersey 07054
Facility Name:	Oyster Creek Nuclear Generating Station
Location:	Forked River, New Jersey
Inspection Periods:	December 14 - 18, 1998, January 6 - 7, 1999, January 20-22, 1999
Inspectors:	Leonard Cheung, Senior Reactor Engineer, EPB Ram Bhatia, Reactor Engineer, ESB Jimi Yerokun, Acting Chief, ESB
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EXECUTIVE SUMMARY

Oyster Creek Nuclear Generating Station
Report No. 98-12

An engineering inspection was performed at Oyster Creek Nuclear Generating Station (Oyster Creek) during the periods of December 14 - 18, 1998, January 6 - 7, 1999, and January 20 - 22, 1999. The inspection consisted of the review of licensee's self-assessment of Oyster Creek environmental qualification (EQ) program, the review of licensee's corrective actions for previously identified engineering open items, and the review of licensee's implementation of Generic Letter (GL) 89-10 motor-operated valve (MOV) program.

Engineering

- The licensee showed completed activities required for the NRC to close its review of the GL 89-10 MOV Program. The licensee also completed several modifications that enhanced valve performance and updated the Performance Prediction Methodology (PPM) calculations to reflect industry standards. (Section E1.1)

- The efforts to enhance the MOV program during the 17R refueling outage were good. These efforts included: (1) modifying the reactor water cleanup valves and isolation condenser valves to increase their output capability; (2) increasing the torque switch settings of torus spray valves; and (3) dynamic testing of torus spray valves and shutdown cooling valves. (E1.1)

- The licensee had completed an in-depth self-assessment (audit) of the Oyster Creek EQ program, excluding the EQ process and EQ procedure reviews. The auditors were knowledgeable of EQ requirements and were qualified for their audit functions. The licensee management was actively involved with the audit and had provided appropriate personnel to support the audit. Responses to the auditors' questions were generally prompt, and the resolutions and corrective actions were appropriate. (Section E7.1)

- The corrective actions for design control issue of the Electromatic Relief Valve (EMRV) were comprehensive and in-depth. The corrective actions for the other engineering open items were adequate. (Section E8)

Report Details

Summary of Plant Status

Oyster Creek was at full power during this inspection.

Engineering

E1 Conduct of Engineering

E1.1 GL 89-10, Motor Operated Valve Program Review

a. Inspection Scope (2515/109; 92701)

This inspection was conducted to determine if General Public Utilities Nuclear (GPUN) had completed the actions necessary to satisfy the NRC closure of its review of the Generic Letter 89-10 Motor-Operated Valve (MOV) Program at Oyster Creek. The inspection included a review of GPUN's resolution of those items identified in NRC Inspection Report 50-219/98-07 as remaining to be resolved prior to the NRC closing its review of MOV Program.

b. Observations and Findings

Background

On June 28, 1989, the NRC issued Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," which requested licensees to establish a program to ensure that switch settings for safety-related motor-operated valves (MOVs) were selected, set, and maintained properly. Seven supplements to the GL have been issued to provide additional information and guidance on development of programs. NRC inspections were conducted at Oyster Creek based on guidance contained in NRC Temporary Instruction (TI) 2515/109, "Inspection Requirements for Generic Letter 89-10."

During the last NRC inspection of the MOV Program at Oyster Creek, conducted in June 1998 and documented in NRC Inspection Report 50-219/98-07, the NRC was unable to reach closure regarding the review due to the large portion of MOVs in the program for which modifications and updated calculations remained to be completed. The NRC identified that GPUN had not yet adequately verified the design basis capability of two-thirds of the valves in the program. Specifically, the NRC identified that the following items needed to be resolved prior to closure of the program:

- Modification and completion of the MOV Performance Prediction Methodology (PPM) calculations for Group 1A Reactor Water Clean Up valves (RWCU) V-16-1, V-16-2, V-16-14 and V-16-61 by the Fall 1998 refueling outage (17R).
- Modification and completion of PPM calculations for Group 1C Isolation Condenser valves V-14-36 and V-14-37 by the Fall 2000 refueling outage (18R).
- Replacement of the cables to Group 1C Isolation Condenser Valves V-14-36 and V-13-30 to increase their thrust margin during the 17R.

- Increasing the torque switch settings of Group 3A Containment Spray valves V-21-5 and V-21-11 to increase their thrust margin during 17R.

- Dynamic testing of shutdown cooling valves V-17-19 and V-17-54 to bolster valve factor (Vf) assumptions during 17R.

- Dynamic testing of Group 6 Torus Spray valves V-21-15 and V-21-18 to bolster Vf assumptions during 17R.

Program Status

Group 1A Reactor Water Clean Up valves V-16-1, V-16-2, V-16-14 and V-16-61 Modification and PPM Calculation

These RWCU valves are required to close under blowdown conditions to terminate a high energy line break as discussed in Supplement 3 of GL 89-10. Originally, the licensee assumed a valve factor of 0.59 based on tests performed by the Electric Power Research Institute (EPRI) and other utilities. However, all but one of the 32 tests cited by the licensee were performed under pumped flow (rather than blowdown) conditions, and the information from two industry blowdown tests of Anchor-Darling valves was not included in the data set.

To meet EPRI PPM guidelines, GPUN modified the valve internals and performed PPM thrust calculations for the valves during the 17R refueling outage. The modification, which included disk and stem replacement, was done to allow the valves to perform more predictably and was accomplished under engineering evaluation 0089-98, "Alternate Replacement discs and stems for V-16-1, 2, 14 and 61 for NRC GL 89-10 concerns."

The inspectors reviewed the post-work diagnostic signature analysis report for all four valves comparing the available thrust values with those required for the valves to function. The minimum required thrusts (using the EPRI MOV PPM) for the valves to function under design basis, were documented in calculation C-1302-900-E540-016, "PPM Thrust Calculation for GL 89-10 Gate Valves." The inspectors verified that the available thrust values were sufficiently greater than required, ensuring that the valves would perform their safety functions. For example, for valve V-16-1 a thrust of 17,152 Lbs is required, and the test demonstrated an available thrust of 22,748 Lbs. Weak link analyses (Report R98.062 and Calculation C-1302-215-E310-069) were also completed to determine the maximum allowable actuator thrust for the valves and to ensure that MOV switch settings do not allow excessive loading. The inspectors reviewed the analyses and identified no concerns. The inspectors determined that GPUN had properly modified and completed the EPRI PPM calculations for these Group 1A RWCU valves and considered this item addressed.

Group 1C Isolation Condenser Valves V-14-36 and V-14-37 Modification and PPM Calculation

During the previous NRC inspection in this area, GPUN had stated that the internals of valves V-14-36 and V-14-37 would be modified during the 18R refueling outage to increase their thrust margins. The deferral to the 18R outage was unavoidable because the valve work requires a full core off-load to the spent fuel pool, which could not be performed during the 17R refueling outage. In the interim, GPUN had determined that the MOVs were capable of overcoming a valve factor of 0.79 assuming the running loads and stem friction coefficients measured during the latest static diagnostic tests. During the 17R refueling outage, the licensee replaced the motor power cable for valve V-14-36 to increase the thrust margin of the valve under degraded voltage conditions. The inspector reviewed the work document used for accomplishing this replacement and found it acceptable. The inspector also reviewed the weak link analysis for the valve (as well as for V-14-30), Anchor Darling Analysis Report, "A/D Report R94.101, Seismic Weak Link Analysis Report - 10 inch Class 600 Stainless Steel Flex Wedge Gate Valve," and identified no concerns. The inspector determined that GPUN had appropriately demonstrated the viability of deferring the modification of the valves internals until the 18R outage. Meanwhile, GPUN had demonstrated adequate assurance that the valves would be capable of performing their intended safety function. The inspector considered this adequate for closing the NRC's review of the MOV program.

Groups 2A and 2B Isolation Condenser Valve V-14-30 Cable Replacement

During the previous NRC inspection, GPUN had indicated that they planned to increase the design capability of valve V-14-30 (which had a design margin of six percent) by replacing the motor power cable during the 17R refueling outage. However, in a letter dated September 3, 1998, to the NRC, GPU indicated that they would defer the modification to the 18R refueling outage because a recent assessment indicated that the valve design margin based on preliminary EPRI PPM thrust determination was considered sufficient for cycle 17 operation. The inspectors reviewed the available information that supported the licensee's assertion. The thrust required for the valve to function under design basis accident conditions was reflected in calculation C-1302-211-E540-104, "PPM Thrust Calculation for Isolation Condenser Valves." Calculation C-1302-900-E120-019, "Thrust and Operator Sizing for GL 89-10 Gate Valves," showed that the valve had enough margin available (4.7%), based on close motor gearing capability, for meeting the full design requirements of the GL 89-10 program. The inspectors determined that GPUN had appropriately demonstrated the viability of deferring the cable replacement for valve V-14-30 until the 18R outage. Meanwhile, GPUN had demonstrated adequate assurance that the valve would be capable of performing its design function.

Group 3A Containment (Torus) Spray Valves V-21-5 and V-21-11 Torque Switch Setting Increase

As initially set up, the thrust capabilities of valves V-21-5 and V-21-11 were determined to be marginal. The licensee was able to demonstrate valve operability using standard analytical methods based on measured (versus design basis) stem friction coefficients, packing loads, and rate of loading. To improve the thrust margin, GPUN increased the

torque switch settings of these valves during the 17R refueling outage. The inspectors reviewed the work orders used for the adjustments. In addition, the inspectors reviewed the weak link analysis, C-1302-900-E540-020, "Weak Link Analysis for OC MOVs," which reflected the maximum allowable actuator thrust for the valves to protect against excessive loading by the MOV switch setting. For valves 5 and 11 the weak link was the disc-to-stem T-head with a maximum allowed thrust of 65.949 KIPS (opening) and 29.301 KIPS (closing). The inspectors identified no concerns. The increase in torque switch setting provided additional thrust margin for the valves while still being bounded by the weak link analysis.

Shutdown Cooling Valves V-17-19 and V-17-54 Dynamic Testing

To demonstrate that shutdown cooling valves V-17-19 and V-17-54 had adequate thrust margins, the licensee had used the tested Vf of 0.66 instead of the design basis assumption of 0.8. Using measured (versus design basis) values of rate of loading and packing load in the standard industry equations, the available valve factor had been determined to be 0.78 for these MOVs. Nevertheless, the thrust capability margin of valve V-17-19 had been found to be acceptable (by approximately 17%). During the 17R refueling outage, the licensee bolstered its valve factor assumption by dynamically testing valve V-17-54 using procedure 613.4.002, "Shutdown Cooling Valve Operability and IST Valve Test" (a dynamic test of V-17-19 (pump suction valve) was not practicable). The inspectors reviewed the test result contained in a Gate Valve Test Analysis Data Sheet for valve V-17-54. The maximum closing stroke thrust value was 13,310 Lbs. The results showed that there was ample thrust margin for the V-17-54 valve. The inspectors also verified that valve V-17-19 was a pump suction valve and was not practicable to be tested. Therefore, the inspector concluded that the licensee had adequately addressed this item.

Group 6 Torus Spray Valves V-21-15 and V-21-18 Dynamic Testing

The licensee's 1997 self-assessment team had concluded that the assumed valve factor of 0.6 for torus spray valves V-21-15 and V-21-18 was not supported adequately by the results of in-situ dynamic tests and industry data. The inspectors had determined that the valve capability margins were very low at the current torque switch settings if design basis rate of loading and packing load assumptions were used. However, using measured packing loads, the licensee had demonstrated that an immediate operability concern did not exist with the torque switch settings. To validate that valve factor assumption, GPUN successfully tested both valves dynamically, during the 17R refueling outage. The dynamic tests were accomplished per special procedure 98-002, "Dynamic Testing of Containment Spray Valves." The inspector had no concern in this area and was satisfied that the licensee had accomplished their commitment in this area.

Rate of Loading (ROL)

During the last inspection, the inspectors expressed concern over the fact that the licensee's stroke time calculations did not take into account the rate of loading (ROL) phenomenon. During this inspection, the inspector reviewed revised calculation, C-1302-730-5350-017, "Stroke Time Calculation for GL 89-10 MOVs" and verified that the calculation included considerations of ROL. The calculation also included appropriate consideration for degraded voltages for the DC powered actuators. The inspector also verified that the calculated stroke times were not in conflict with the valves Inservice Test stroke time requirements.

Core Spray Valves Deletion

The inspector reviewed the removal of core spray valves V-20-12 and V-20-18 from the MOV program. These valves are the Core Spray subsystem I and subsystem II discharge valves respectively. The licensee indicated that the basis was that the valves are normally in their safety position (open) and that they had instituted administrative controls for the core spray system to be considered inoperable when the valves are closed. The inspector reviewed station procedure 308, "Emergency Core Cooling System Operation," Revision 62, and verified that the required normal alignment of the valves was in the open position. The inspector also reviewed procedure 610.4.003, "Core Spray Valve Operability and In service Testing," and verified that it clearly states that the core spray system is inoperable when the valves are closed. The inspector was satisfied that there were appropriate administrative controls in place to ensure that the valves are maintained in their safety position or the system declared inoperable when they are not. Therefore, the inspector found no issue with the valves being removed from the MOV Program.

c Conclusions

The licensee completed activities required for the NRC to close its review of the GL 89-10 MOV Program. The licensee also completed several modifications that enhanced valve performance and updated the Performance Prediction Methodology (PPM) calculations to reflect industry standards. (Section E1.1)

The efforts to enhance the MOV program during the 17R refueling outage were good. These efforts included: (1) modifying RWCU Valves V-16-1,-2,-14, -16, and isolation condenser valves V-14-30 and 36 to increase output capability; (2) increasing the torque switch settings of torus spray valves V-21-5 and 11; and (3) dynamic testing of torus spray valves V-21-15, and 18, and shutdown cooling valve V-17-54. (E1.1)

E7 Quality assurance in Engineering Activities

E7.1 Self-assessment of Environment Qualification

a. Inspection Scope (IP 40501)

The licensee conducted a self-assessment of the Oyster Creek environmental qualification (EQ) program from December 14, 1998, to January 21, 1999. The inspectors reviewed the self-assessment process, including scope, plans, and findings; interviewed the assessment team members; and observed the team's walkdown of EQ equipment at the plant site. The inspectors also reviewed the licensee's response to the team's findings and subsequent corrective actions.

b. Observations and Findings

The self-assessment team consisted of a team leader (a contractor) and three team members (two contractors and a GPUN employee). The inspectors reviewed the experience of the team and confirmed that they all had adequate qualifications for their assigned duties. The inspectors also interviewed the team, both individually and in group, and found them knowledgeable in the EQ areas.

The licensee had completed an inspection plan for the assessment, entitled "Oyster Creek Environmental Qualification Vertical Slice Self-assessment," dated December 3, 1998. The inspectors reviewed this assessment plan and found that it did not cover the EQ programmatic aspects such as EQ process and EQ procedure reviews. The licensee explained that the EQ procedure review and the EQ process would not be covered due to the current operating status at Oyster Creek. However, those aspects would be covered in their next self-assessment at Three Mile Island. The licensee expected only limited EQ file updates or additions, and this small quantity could be closely scrutinized. The inspectors determined this to be acceptable.

The licensee selected two accident scenarios for the "vertical slice" audit: 1) small break loss-of-coolant accident (SBLOCA) inside the drywell; and 2) isolation condenser line break outside the drywell. The audit consisted of three tasks: 1) review of accident scenarios and equipment selection; 2) review of EQ files of items selected in task 1; and 3) physical inspection (walkdown) at the plant site of items selected in task 1. The inspectors reviewed the listed attributes for each of the three tasks and found they provided a broad scope of each task. The inspectors also reviewed the "EQ File Release Checklist" that the assessment team used for checking the adequacy of the EQ files, and found the list very extensive and detailed.

The inspectors noted that the assessment team had reviewed 17 EQ files that covered the 17 components selected from the two accident scenarios, and had walked down 12 components that were accessible during plant operation. The review and walkdown resulted in 95 questions. The licensee's response to those questions resulted in 10 Corrective Action Processes (CAP) and one Determination of Operability (DOO) being generated. The inspectors' review of the DOO (Procedure ES-027 did not envelope SBLOCA pressure) did not identify any concerns. However, the inspector' review of the 10 CAPs identified the following:

- CAP 01999-0069 described that the actuators for four motor-operated valves (MOV) (V-20-15, 21, 40 and 41) had been replaced with larger and qualified actuators (for MOV program implementation), that had different model numbers, in December, 1992. However, the EQ files for these valves actuators had not been properly updated to document and demonstrate the qualification status of the installed actuators. The deficiency and planned corrective actions were appropriately documented in and being tracked by CAP 01999-0069 and CAP 01999-0069-01. This non-repetitive, licensee-identified and corrected violation (10 CFR 50.49 (j), EQ records) is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of NRC Enforcement Policy. **(NCV 50-219/98-12-01)**

- CAP 01999-0071 described that the hydrogen/oxygen analyzers were Regulatory Guide 1.97 Category 1 instruments. These analyzers were equipped with sample line heat tracing to maintain the sample from the drywell above the saturation temperature of post-LOCA drywell pressure. Failure of the heat tracing could cause inaccurate hydrogen concentration measurements. The CAP also identified that the analyzers were powered by an emergency power source whereas the heat tracing was not. The licensee later (after additional review) informed the inspectors that the heat tracing was powered by a separate source of emergency power. The licensee evaluated this issue and determined that there was no safety concern even if the heat tracing failed after the accident, because the inaccuracy was on the conservative side, and inadvertent venting of the drywell could not occur. The inspectors reviewed the evaluation and had no further questions.

The inspectors observed the assessment team's walkdown process on January 7, 1999. The inspectors noted that the selected Limitorque motor operator was not opened to examine the control wiring, torque switches, etc inside the valve actuator. The licensee explained that this MOV was in the isolation condenser system, and could not be opened during plant operation. The assessment team completed walkdown of 12 items (the other five items were not accessible during plant operation), resulting in the issuance of two CAPs (01999-0041 and 01998-1827), that pertained to heat tracing issues and the hydrogen/oxygen analyzer qualified life. The inspectors' review of these to CAPs did not identify any concerns.

The inspectors reviewed the list of findings/questions presented to the licensee by the team at the assessment team's exit meeting which was held on January 21, 1999. The inspectors identified four issues that had the potential to become significant concerns if inadequate responses/resolutions were provided. The details and resolutions for these issues were not available for the inspectors' review during this inspection. These issues were:

1) No realistic flood calculations for the corner rooms;
2) No analysis of high energy line break (HELB) at 95 feet elevation;
3) A number of EQ files had inappropriate steps in aging calculations (e.g., first LOCA peak, and energized mode); and
4) Relevant EQ related maintenance requirements were not identified in the EQ files (e.g., Patel connectors, and Rosemount transmitters).

This item was unresolved pending NRC review of the licensee's resolutions of these four issues. **(URI 50-219/98-12-02)**

c. Conclusions

GPUN had completed an in-depth self-assessment (audit) of the Oyster Creek EQ program, excluding the EQ process and EQ procedure reviews. The auditors were knowledgeable of EQ requirements and were qualified for their audit functions. The licensee management was actively involved with the audit and had provided appropriate personnel to support the audit. Responses to the auditors' questions were generally prompt, and the resolutions and corrective actions were appropriate.

E8 Miscellaneous Engineering Issues

Followup on Previously Identified Engineering Open Items (92903)

The inspectors reviewed the licensee's responses and subsequent corrective actions for nine previously identified engineering open items. The results of these reviews are documented as follows.

E8.1 (Closed) EEI 50-219/97421-01013, Emergency Service Water (ESW) Pump Surveillance Test Failure Due to Improper Material

During a previous inspection (97-06), the NRC identified an issue involving inoperable ESW pumps. The licensee determined the cause of the pump failure to be the use of improper material, causing galvanic reaction between cast iron and stainless steel, and resulting in a pump coupling failure. The licensee later identified that ESW Pump "D" had similar problem.

In their December 17, 1997, response letter, the licensee attributed the root cause to be the reviewer's lack of understanding on the impact of material incompatibility, and the lack of procedure guidance in this area.

The licensee replaced both "C" and "D" ESW pumps with all stainless steel spare pumps, and verified that both "A" and "B" pumps were all stainless steel pumps.

During this inspection, the inspectors verified the licensee's completion of the following long term corrective actions by reviewing appropriate documents and training records:

1) The pump vendor (Borg and Warner Industrial Pumps) had issued a 10 CFR 50 Part 21 report to the NRC on September 24, 1997 and a supplement on October 3, 1997, regarding the adverse effect of using incompatible material;

2) The licensee updated Oyster Creek Station Procedure 124.2, "Control of Engineering Directed Alternate Replacements", (Revision 15) to include (in Exhibit 3) guidance on galvanic corrosion prevention;

3) The licensee conducted an engineering review of about 300 warehouse documents searching for equipment that could have incompatible materials. Equipment found to have incompatible materials in the warehouse were properly disposed of.

4) The licensee completed an engineering design change evaluation dated November 13, 1997, for the replacement of cast-iron parts in the ESW pumps with stainless-steel parts. This design change evaluation also included Safety Evaluation SE 000532-022, dated October 30, 1997;

5) Training of engineering personnel (at both the Headquarter Office and the Oyster Creek site) on the adverse effect of using dissimilar metals in components of ESW pumps (attendance sheets indicated training was completed in January 1998).

The inspector determined that the corrective actions were adequate. This item is closed.

E8.2 <u>(Closed) Violation 50-219/98-03-01, Emergency Diesel Generator (EDG) Starter Motor Replacement</u>

During a previous inspection (98-03), the NRC identified an issue involving failure to document a test failure of the EDG starter motor. On June 8, 1998, EDG 1 failed to start during the surveillance test. Subsequently, the licensee initiated the replacement process of the two electric starter motors of the EDG. The licensee obtained two replacement starter motors from the warehouse, one new motor and the other a refurbished one. The refurbished motor failed the pre-installation bench test. The licensee later found that the refurbished motor contained an extra jumper on the external terminals. The licensee removed the jumper and completed the motor replacement. Post maintenance testing was also completed satisfactorily and EDG 1 was returned to operable status. However, the licensee failed to document the deficient condition of the refurbished motor in a Receipt Deficiency Report (RDR) as required by station procedure 125.2.10.

In their response latter dated August 14, 1998, the licensee attributed the violation to be an administration requirement oversight of the maintenance supervisor, because his attention was drawn to the Technical Specifications limited condition for operation (LCO) of the EDG.

The licensee's corrective actions included: 1) completing an RDR for the starter motor jumper issue, 2) determining whether a 10 CFR Part 21 report was required, and 3) conducting a meeting to discuss configuration control requirements among the configuration control manager, electrical maintenance manger, electrical maintenance supervisor, and electrical maintenance staff.

During this inspection, the inspectors reviewed RDR 98-082 dated June 17, 1998, and found that this report adequately documented the motor jumper issue. The inspectors also reviewed a November 24, 1998, memo (No.1950-98-1014), entitled "Potential 10 CFR 21 Notification, Oyster Creek Diesel Generator Starter Motor," and found that this memo properly documented the reason why a 10 CFR Part 21 report was not required for this issue.

The inspectors considered licensee's corrective actions adequate. This item is closed.

E8.3 (Closed) EEI 50-219/98049-01014, Primary Containment Boundary Isolation Using Freeze Seal

During a previous inspection (97-11), the NRC identified an issue involving a failure to complete a safety evaluation for a temporary configuration change. On January 15, 1998, while conducting a Containment Spray and Emergency Service Water Pump System 2 operation test, the licensee identified that the relief valve in the containment spray heat exchanger was leaking (about 2 gpm). Subsequently, the licensee secured the containment spray pump and declared this system inoperable (the other redundant system was operable) and manually closed containment spray isolation valves and opened the pump's circuit breakers. The licensee recognized that the isolation valves had not been leak-rate tested. The licensee replaced the defective relief valve using a freeze seal to provide an isolation boundary. However, the licensee failed to complete a 10 CFR 50.59 safety evaluation for this temporary configuration change, and to ensure that no unreviewed safety questions were involved for this activity.

During this inspection, the inspectors reviewed licensee's response letter dated May 14, 1998. The inspectors noted that the licensee did not mention in the response letter whether they had completed the 10 CFR 50.59 safety evaluation, which was an important corrective action for this violation. In response to the inspectors' request, the licensee provided for the inspectors' review the safety evaluation for JO 521147, dated January 16, 1998. The inspectors found this evaluation adequate.

Following the violation, the licensee formed a Performance Enhancement Review Committee (PERC 98-01), consisting of a chairman and nine members, to review this issue. This effort resulted in the issuance of a lesson learned memo in February, 1998, entitled "Human Performance Eye Opener: The Cost of Overconfidence," that had been distributed to engineering, operations and maintenance personnel, to alert them of potential pitfalls because of overconfidence and lack of attention to detail. The licensee also conducted several informal training sessions, involving maintenance supervisors, operations personnel, and engineering staff, regarding this violation. The inspectors verified these training sessions by reviewing training records and discussion with involved personnel. The inspectors considered the corrective actions adequate. This item is closed.

E8.4 (Closed) Violation 98-80-01, Failure to verify the operability of automatic depressurization system (ADS) as required by the Technical Specifications (TS)

During a previous inspection (98-80), the NRC identified a violation involving a failure to verify the operability of the ADS system during the reactor vessel pressure testing. The Oyster Creek TS required all five Electromatic Relief Valves to be operable when the reactor water temperature was above 212 °F and reactor water pressure was above 110 psig. The TS allowed the valves' pressure relief function to be bypassed during reactor vessel pressure testing. However, the TS also required the high drywell pressure instrument be operable to support the ADS' operability when the valves' relief function was bypassed. The operability of the high pressure instrument could be demonstrated by performing a channel check once a day. However, in October 1996 when the ADS was required to be operable during the reactor vessel testing, the licensee failed to perform the required high drywell pressure channel check.

As a result of this violation, the licensee issued a Licensee Event Report (LER 98-003) on March 31, 1998. The inspectors's review of this LER indicated that the LER had properly documented the causes and corrective actions for this issue. The licensee also issued CAP 01998-0223 to track the corrective actions. During this inspection, the licensee told the inspectors that they had revised the TS to delete the ADS operability requirements during reactor vessel pressure testing because ADS was not required to function under this condition. The inspectors reviewed the latest version (Amendment 199) of the Oyster Creek TS, and found that the ADS operability requirements had been deleted from Section 3.4.B.1.

The inspectors also interviewed the training personnel who provided training to the operators and reviewed an internal memo, which confirmed that the subject discussed in LER 98-003 had been covered during the operator training.

The inspectors also reviewed a recent log (dated January 21, 1999) of the TS log sheet, and confirmed that the high drywell pressure data (although no longer a requirement) was appropriately documented.

The inspectors considered the licensee's corrective actions adequate. This item is closed.

E8.5 (Closed) EEI 98220-01013, Design control for the Electromatic relief valves (EMRV)

During the April 1998 inspection (98-80), the NRC found that a General Electric EQ test document in the Oyster Creek EQ file for the EMRV specified that the EMRV required a minimum voltage of 105 Vdc to operate in a harsh environment. However, the licensee did not have a voltage calculation to demonstrate that the five EMRVs at Oyster Creek had sufficient voltage to operate following a SBLOCA. The licensee later completed a preliminary calculation and found that they did not have sufficient voltage (105 Vdc). Subsequently, the licensee conducted a series of tests at Wyle Laboratories, Huntsville, Alabama, using a spare EMRV. These tests demonstrated that the minimum operable voltage was 80 Vdc under the harsh environment (ADS mode), and 70 Vdc in a mild environment (pressure relief mode). These voltage requirements were documented in a GPU report entitled, "Review of EMRV Test Results from Testing performed at Wyle Laboratories from March 18, 1998 through March 28, 1998," dated April 2, 1998. The inspectors' review of this document indicated that a licensee team of nine engineers were involved with the test-result review. In addition, the inspectors also reviewed the records of Wyle test results, which confirmed the above minimum operable voltage requirements.

The licensee later completed a more detailed calculation and found that, under a harsh environment, three of the five EMRVs did not have 80 Vdc at the EMRV solenoids. The licensee subsequently declared the ADS inoperable, and formed five teams to resolve this issue: the root cause team (to find out root cause and contributing factors); the test and calculation team (to conduct EQ testing and complete voltage-drop calculations); the modification team (to design and implement a plant modification); the extent of issue team (to review other 125 Vdc powered components for similar problems); and the in-line review team (to perform independent reviews of activities completed by the other four teams). During the plant shutdown in March 1998, the licensee completed a plant modification to reduce the voltage drops by increasing the overall cable size for the EMRV solenoids. After the modification, the licensee was able to increase the available voltage at the EMRV solenoids above the required voltage in a harsh environment. The inspectors' review of the modification package for the EMRV cable addition (Modification No. OC-MD-H100-001), including 10 CFR 50.59 Safety Evaluation OC-SE-000642-008, did not identify any concerns. The inspectors also reviewed two calculation packages: 1) C-1302-735-E320-037 entitled "Voltage Drop After Modification to 125 Vdc Operated EMRVs Under ADS Scenario (for harsh environment)", dated April 2, 1998; and 2) C-1302-735-E320-038 entitled "Voltage Drop After Modification to 125 Vdc Operated EMRV's High Pressure Relief Function (for mild environment)", dated April 2, 1998. These calculations showed that the available voltages at the EMRV solenoids were higher than the required voltages with sufficient margins.

The inspectors determined that the licensee had completed comprehensive corrective actions for this item. This item is closed.

E8.6 (Updated) EEI 98220-01023, Environmental Qualification (EQ) of the Electromatic relief valves (EMRV)

During the April 1998 inspection, the NRC found that the EQ of the five EMRVs at Oyster Creek had not been demonstrated as required by 10 CFR 50.49 (f). Specifically, insufficient voltage was available at the EMRV solenoids following a SBLOCA when ADS was required. As discussed in Section E8.5 above, the licensee had completed various corrective actions, including a plant modification, to make the EMRVs qualifiable. The licensee also conducted an EQ program self-assessment as discussed in Section E7 above. However, the licensee had not updated the EQ file for the EMRVs, which is the major document to show that the EMRVs are qualified. In addition, the licensee had not yet received the final EQ test report from Wyle Laboratories, which is the main document to demonstrate the qualification status of the EMRVs. This item remains open pending further NRC's review of the Wyle Laboratories EQ test report, including the temperature and pressure profiles of the post accident environment, and the EQ file.

E8.7 (Closed) Violation 98-80-05, Failure to perform a design verification of the containment spray system (CSS) heat exchanger, and failure to follow station procedure EP-016

During the April 1998 inspection, the NRC identified two cases where the licensee failed to follow their procedural requirements. In the first case, the licensee failed to perform a design verification for the seismic design calculations of containment spray system (CSS) heat exchanger to support the seismic adequacy of safety-related equipment (to address issues of NRC Generic Letter 87-02, Revision 1), as required by Procedure EP-06,"Nuclear Safety/Environmental Determination and Evaluation," Revision 3.03. In the second case, numerous examples of non-compliance with Procedure EP-016 were found. These examples involved the printed names and signatures for engineer/originator, section manager or project manager, responsible technical reviewer, and independent safety reviewer, not being executed.

Following the April 1998 inspection, the licensee completed a re-analysis (Calculation C-1302-241-E540-076, Revision 0) of anchorage of the containment spray heat exchangers on October 26, 1998. The inspectors' review of the analysis indicated that all inputs in this analysis had been design verified. The anchorage for all four CSS heat exchangers was found adequate and were capable of performing the intended design function without a modification. In addition, the documentation associated with this analysis was appropriately signed, approved, and verified by a design engineer, as required by Procedure EP-06.

The inspectors also noted that the licensee had revised the Administrative Procedure 1000-ADM-1216.03,"Regulatory Correspondence Control," and Corporate Calculation Procedure EP-06, to clearly state that any required design verification must be completed prior to release of any calculations.

In response to the second concern, the licensee attributed this violation to a number of individuals who failed to comply with GPUN procedural requirements; and to the administrative procedure that did not contain the requirements which were specified in Procedure EP-06. The inspectors verified that the licensee had appropriately revised the administrative procedure on March 6, 1998, to include these requirements. The licensee also stated that they had completed a self-assessment, that had identified a number of strengths, good practices and opportunities for improvement. This lack of consistency among procedures and the level of compliance to print one's name in addition to signing were both noted as areas of concerns. The staff has been alerted to this effect.

Also, the licensee's safety review training was revised to reflect these procedural changes and to emphasize these concerns. The inspectors reviewed two recently completed safety evaluations and found that the responsible technical reviewer and independent safety reviewer had appropriately made the entries in the signature blocks by printing their names, in addition to signing their names, as required by Procedure, EP-016. The inspectors concluded that the licensee had adequately addressed the issues associated with this violation. This item is closed.

E8.8 (Closed) Violation 98-80-06, Failure to implement adequate corrective actions for improperly stored chain hoist

During the April 1998 inspection, the NRC observed a seismic deficiency with a containment spray system heat exchanger (a chain hoist near the heat exchanger), that had been identified in August 1994, had not been adequately corrected.

In their May 22, 1998, letter in response to this violation, the licensee agreed with this concern. The licensee stated that during a walkdown of Safe Shutdown Equipment List components (for Unreviewed Safety Issue (USI) A-40) in 1994, the CSS heat exchanger chain hoist issue was identified and was written up as a program outlier condition. A work request (WR) was issued and a work order (WO) was generated to resolve the issue. The WR stated that the chain falls be removed or bagged. The bagging option was chosen and the WO was closed out on December 6, 1996, without notifying engineering, and therefore no engineering verification of this work was performed, until March 1998.

During this inspection, the inspectors noted that engineering had reinspected all chain hoist configurations near the four heat exchangers. They determined that all chain hoist slacks had been tied to the railing on the lower ends and bagged on the top areas, and would not impose an interaction hazard to any plant components during a Safe Shutdown Earthquake (SSE). The inspectors verified that the CSS Heat Exchanger Cleaning and Assembly Procedure 200-SMM-3214.02, had been appropriately updated to include controls on storage of chain hoists near and around CSS heat exchangers.

The inspectors considered the licensee corrective actions adequate. This item is closed.

E8.9 (Closed) Violation 98-80-07, Failure to submit changes, tests and experiment reports for 1983 and 1986

In the April 1998 inspection, the NRC identified that the licensee had not submitted changes, tests and experiments (CTE) reports for 1983 and 1986. 10 CFR 50.59(b)(2) requires that the licensee submits a report containing a brief description of any changes, tests and experiments, including a summary of the safety evaluation of each, annually or along with the UFSAR updates as required by 10 CFR 50.71(e) or at such shorter intervals as may be specified in the license.

In the June 15, 1998, response letter, the licensee attributed this violation to be their misunderstanding of 10 CFR 50.59(b)(2) requirement. Following the NRC rulemaking that allowed FSAR updates to be submitted on a schedule consistent with the refueling cycle, GPUN adopted the two year update interval based on the 24 month refueling cycle. At that time, the licensee thought that the 10 CFR 50.59 summary report could also be updated on the two year cycle. However, they did not realize that the report was required to be submitted with the FSAR update or submitted annually instead.

During this inspection, the inspectors noted that the licensee performed a detailed evaluation of all reports made since 1981 through 1998. The licensee determined that all information required to be reported by 10 CFR 50.59 had been reported to the NRC in various reports, except for the information for the period March 1995 through December 1996.

To address the above discrepancy, the licensee submitted 10 CFR 50.59(b) reports to the NRC on June 15, 1998, and February 16, 1998, that covered the period April 1993 to March 1995 and April 1995 to May 1997 respectively. Based on these recent licensee submittals, the licensee now had became in full compliance with the current reporting requirements of 10 CFR 50.59(b) and this has put them on a reporting schedule consistent with 10 CFR 50.71(e) as permitted by 50.59(b). The licensee had scheduled their next report to be submitted in mid June 1999. The licensee stated that they would submit at that time both the CTE reports and FSAR updates, consistent with the NRC requirements of 24 months. This item is closed.

V. Management Meetings

X1 Exit Meeting Summary

On February 4, 1999, the inspectors held a telephone exit meeting from NRC Region I Office with members of GPUN at the Oyster creek Nuclear Generating Station and of GPU Headquarter Office in the Parsippany, New Jersey, to discuss the findings of this inspection. The licensee acknowledged the inspection findings. A list of those present at the exit is shown in the Persons Contacted Section.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

A. Agarwal, Manager, EP & I, Oyster Creek
R. Buhowski, Senior Engineer, EP & I
G. Busch, Manager, Nuclear Safety & Licensing
B. DeMerchant, Licensing Engineer
S. Levin, Director, Operations and Maintenance
R. McGoey, Director, Technical Support
R. Paniker, Manager, EP & I, Headquarter
M. Roche, Vice President and Director, Oyster Creek
A. Rone, Vice President, Engineering
S. Tiwari, NSA Lead Assessor

NRC

J. Schoppy, Senior Resident Inspector

INSPECTION PROCEDURE USED

IP 40501	Licensee Self-Assessment
IP 92903	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

Number	Type	Description
98-12-01	NCV	EQ Documentation of four MOV Actuators
98-12-02	URI	Licensee's Resolutions for Four EQ Self-Assessment Findings

Closed

98-12-01	NCV	EQ Documentation of four MOV Actuators
98-80-01	VIO	Failure to verify ADS operability as required by TS
98-80-05	VIO	Failure to perform a design verification of the CSS heat exchanger; and Failure to follow procedure EP-016
98-80-06	VIO	Corrective actions for improperly stored chain hoist
98-80-07	VIO	Failure to submit CTE reports for 1983 and 1986

98-03-01	VIO	EDG Starter Motor Replacement
97421-01013	EEI	Emergency Service Water Pump Test Failure
98049-01014	EEI	Primary Containment Boundary Isolation Using Freeze Seal
98220-01013	EEI	Design Control for EMRV solenoids

Updated

98220-01023	EEI	EQ for EMRV solenoids

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CSS	Containment Spray System
CTE	Changes, Tests and Experiment
EDG	Emergency Diesel Generator
DOO	Determination of Operation
EEI	Escalated Enforcement Item
EMRV	Electromatic Relief Valve
EPB	Engineering Program Branch
EPRI	Electric Power Research Institute
EQ	Environmental qualification
ESB	Engineering Support Branch
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
GE	General Electric
GL	Generic Letter
gpm	Gallons per Minute
GPUN	General Public Utilities (GPU) Nuclear
HELB	High Energy Line Break
IP	Inspection Procedure
KIPS	1,000 Pounds
Lb	Pound
LCO	Limited Condition for Operation
LER	Licensee Event Report
MOV	Motor-Operated Valve
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
PPM	Performance Prediction Methodology
psig	Pounds per Square Inch
RDR	Receipt Deficiency Report
ROL	Rate of Loading
RWCU	Reactor Water Clean Up

SBLOCA	Small Break Loss-of-Coolant Accident
SSE	Safe Shutdown Earthquake
TI	Temporary Instruction
TS	Technical Specifications
UFSAR	Updated Safety Analysis Report
URI	Unresolved Item
USI	Unreviewed Safety Issue
Vac	Volt, alternating current
Vdc	Volt, direct current
Vf	Valve Factor
WO	Work Order
WR	Work Request