U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/88-17

Docket No. 50-219

License No. DPR-16

Licensee: GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, New Jersey

Facility Name: Oyster Creek Nuclear Generating Station

Inspection At: Forked River, New Jersey

Inspection Conducted: June 13-17, 1988

Inspector: Theodore A. Rebelowski 7/26/88
Theodore A. Rebelowski, Senior Reactor Engineer date

Approved by: N J Blumberg 7/26/88
Norman J. Blumberg, Chief, date
Operational Programs Section
Operations Branch, DRS

Inspection Summary: Routine inspection on June 13-17, 1988 (Report No. 50-219/88-17)

Areas Inspected: The inspection included Previously Identified Inspection Findings; Audit of Safety Parameters Display System and the planned modifications to institute corrective measures to satisfy the Anticipated Transients Without Scram (ATWS), including the Alternate Rod Injection (ARI) System and the Standby Liquid Control System (SLCS).

Results: The licensee's Safety Parameters Display System is operational. Elements that would improve the system such as additional training, completion of trending of the displayed parameters, additional inputs to aid in determining reactor vessel water levels and the use of source range monitors are under review by licensee. The ATWS modifications (ARI and SLCS) scheduled for the next outage were acceptable in the areas of safety evaluations and installation specifications.

DETAILS

1. Persons Contacted

GPU Nuclear Corporation-Oyster Creek Nuclear Generating Station

*R. Barrett, Plant Operations Director
*P. Crosby, Supervisor, Plant Engineering
B. DeMerchant, Licensing Engineer
*E. Fitzpatrick, Vice President and Director, Oyster Creek
*J. Kowalski, Licensing Manager
J. Laden Meyer, Licensing Engineer
*D. MacFarlane, Site Audit Manager
B. Olaf, Computer Applications
*W. Pelenski, Manager, Computer Applications
*A. Rone, Plant Engineering Director
*P. Smith, Senior Engineer Safeguards
J. Sullivan, Plant Operations Director

U.S. Nuclear Regulatory Commission

E. Collins, Resident Inspector
*L. Meyers, Resident Inspector, Peach Bottom N.P.S.
*W. Wechselberger, Senior Resident Inspector

The inspector also held discussions with other licensee personnel during the course of the inspection.

2. Previously Identified Inspection Findings (92703)

2.1 (Closed) Violation (NC4) 50-219/85-35-02: The licensee failed to specify design requirements for partial penetration structural weld used for the installation of instrument racks RK01 and RK02.

The inspector verified that the Field Change Request (FCR) C-039642 issued to General Public Utilities Nuclear (GPUN) drawings SN 15081.02 ES-03 and SN 15081.02-ES04 were corrected and the identified problem welds were documented in a Material Non Conformance Report (MNCR) 85-275. The partial penetration weld repair was completed and verified acceptable by the licensee's Quality Control personnel. Additional corrective actions included inhouse training to site engineers and contracted engineering personnel on the requirements for an adequate weld designs. The inspector observed the configuration of the instrument racks RK01 and RK02. This item is closed.

2.2 (Closed) Violation (NC4) 50-219/85-35-04: The Modification Change Form (MCF) and completion of the Quality Control (QC) inspection of the modifications to the instrument racks RK01 and RK02 was reviewed by the NRC inspectors and a number of discrepancies between procedures/drawing requirements and as-built conditions were identified that were not previously documented by Material Nonconformance Reports (MNCR), Field Change Request (FCR), or any other licensee documentation. The problems and resolution follow:

a) FCRC-039642 to drawings ES-04 and ES-05 specifies partial penetration butt weld, MCF made a seal weld and not the required partial penetration butt weld.

The licensee has determined that the seal weld was substituted for a partial penetration weld due to inadequate guidance provided on the Field Change Report FCRC-039642. Corrective actions included the placement of welded stiffener plate over the identified weld. In addition a technical evaluation dated April 15, 1986, which described the Welding Qualification requirements between American Society of Mechanical Engineers (ASME) and American Welding Society (AWS) codes clarified licensee's welding practices.

b) The licensee substituted a round hole for a slotted hole in joining two structural members with bolts. The Material Non-Conformance Report 85-275 corrects drawings ES-04 and ES-05 to use the round hole. The original slotted holes were for ease of construction to give bolts additional clearance.

c) Bolting details on ES-04 and ES-05 show the use of washer under both the bolt head and nut. Washers were not installed as required. Additionally, due to the approved use of shorter bolts than specified on the material list, examples of partial thread engagement of nuts on bolts were observed.

The licensee's corrective action included review of specifications for structural joints determining that the bolt ends did not require washers. The short bolts identified were due to a material receipt problem. Material Nonconformance Reports 88-273, 85-274, 85-265 documented the problem. The bolts were replaced, properly torqued and proper thread engagement verified. The inspector observed repaired areas and verified the material condition.

d) The licensee placed a smaller sized fillet weld, 1/8" versus 3/16" during steel fabrication as found on the drawing ES-04 Section 1-1.

The placement of this smaller fillet weld was due to the limited accessibility to the structure, as it was previously fabricated off-location versus the modifications in place. The Licensee evaluation of the weld indicated that the bolting was the prime connector and that welds were placed to hold plates together for ease during drilling.

e) During construction, three valve manifold was inverted on the instrument rack. The licensee corrected the reversed manifold. Additional training was given to contractor personnel to aid identification of component locations. A training program was conducted for supervision that addressed the need to maintain job surveillance to prevent this type of misassemble.

These items, a to e, are closed.

2.3 (Closed) Violation (NC4) 50-219/85-35-06: The inspectors review of the prerequisites associated with GPUN procedure A15B-G1136.010, Rev. 0, RK01 Rack Modification - Electrical, and a review of plant conditions to determine if the prerequisites were met, identified a discrepancy in that paragraph 4.7.3 required the closure of the isolation condenser vent valves and main steam isolation valves when secondary containment is required for work at or near the fuel pool. This prerequisite was not met.

The licensee recognized the need to place "special precaution and limitation" warnings in the body of a procedure prior to the procedural step where the conditions should be applied. The Work Management System Manual No. A00-WMS-1220.14 addresses this concern. This item is closed.

2.4 (Closed) Violation (NC4) 50-219/86-12-02: The licensee did not perform a determination of the load carrying capacity of existing non-seismic floor penetrations as required in Procedure ES-014 Piping Design Standard for OCNS. The inspector reviewed licensee's calculation (C1302-251-5320-021/V-1302 251-02) that determined that the penetration sleeve will carry the load impacted by the restraints. Carrying load was 2403 psi versus a criteria of 14401 psi. This item is closed.

2.5 (Closed) Violation (NC4) 50-219/86-12-03: The licensee failed to identify the material of a penetration sleeve prior to a welding attachment. A chemical analysis was performed and annotated on structural weld record. In addition, engineering personnel were advised that material composition must be determined prior to the performance of a welding attachment. This item is closed.

2.6 (Closed) Unresolved Item (219/85-13-04): Pipe hanger support on the Containment Spray System, NQ-2-H39 was not performing its function, i.e., there was approximately a 3/8" gap between the support and bottom of the pipe.

The inspector examined the containment spray piping hanger support NQ-2-H-39 observed that the and the gap had been eliminated by placement of a shim support plate that allows pipe to contact support. In addition, a safety analysis was performed by the licensee of conditions that existed prior to addition of the support plate and the determination was made that the gap did not impair piping function ability and that the adjacent supports were not overstressed. This item is closed.

2.7 (Closed) Unresolved (50-219/86-12-04): A number of concerns were identified by the inspector in his review of the Structural Weld Record Sheets (SWRS). The concerns included Quality Control (QC) signoffs of final acceptance of welds, better job package formats, additional instructions on preparation SWRS, address material requirements and QC annotation of Plant Inspection Reports. The following changes have been made to classify various items to the Welding Manual Procedure, Control of Welding and Brazing No. 6150-QAP-7220.01 Rev. 3.

a) The SWRS has been modified in that additional items appear on each SWRS. Base material, purchase order and heat numbers are recorded with the final acceptance for the total welds requiring signatures by Quality Control;

b) SWRS format is neater and additional sheets can be added to the job package when required. Space on SWRS is adequate;

c) The use of SWRS, has designated instructions for the Preparation and Field Use of Structures Weld Record Sheets (Exhibit 6) E6-1 to E6-6 of Welding Manual No. 6150-QAP-7220.1;

d) The revision of GPUN Welding Manual addresses the requirements to list material traceability when welds are not listed individually on the SWRS and,

e) The SWRS requires QC sign-offs including QC-PIR (Plant Inspection Reports).

This unresolved item is closed.

3.0 Safety Parameter Display System (25005)

3.1 Background

On October 31, 1980, the NRC published NUREG-0737 which identified Item I.D.2, Plant-Safety Parameters Display Console and requested each licensee to describe, install and fully implement this item using the guidance provided by NUREG-0696 Rev. 2 titled, "Functional Criteria for Emergency Response Facilities."

The purpose of the safety parameter display system (SPDS) is to assist control room personnel in evaluating the safety status of the plant. NUREG-0696 final report published in February 1981, described the SPDS. The criteria used in design of systems including function, location size, staffing and display considerations are stated in the report.

NRR/Licensing SPDS Considerations

The licensee has made several submittals over a four year period that addressed schedules for the completion and implementation of SPDS. (See Attachment A). In correspondence from NRR to Oyster Creek N.G.S., dated March 5, 1986, NRR concluded that the documentation of SPDS was acceptable with comments. The Safety Evaluation was issued by NRR and would be confirmed by a post implementation audit. The licensee placed the SPDS on line in December 1987.

3.2 Scope

The inspector's guidance for acceptable systems criteria is addressed in NUREG-0696 Final Report, Section 5.0, Safety Parameter Display Systems (SPDS).

3.3 Findings

Observations and system reviews for the various subsections of NUREG-0696 Section 5 were performed. The subsections titles and number, with the inspector's findings follow.

Subsection 5.1 Functions: The SPDS operator aid has been programmed into one of three monitors in the control room. The grouping of information displayed allows operators to readily determine the plant conditions. The displayed parameters were reviewed by the licensee for Human Factor inputs, in that instruments on control boards can be used to verify SPDS displays. The Emergency Operating Procedures (EOP's) are reflected in the displays systems. A validation and verification program was performed during startup of the SPDS. The SPDS reliability is assured by the use of two (2) redundant computers units. The computer allows the check of sensors on the SPDS which will display the questionable outputs. The various interfaces with non-safety related systems is protected by isolation units.

One area not presently included in SPDS is the ability to automatically compute the displayed parameter trends. The licensee stated that this item would be modified at the next outage.

Subsection 5.2-Location: The SPDS is located on one of three monitors on center console. It is readily accessible to senior operators, shift advisors and shift supervisor.

Subsection 5.3-Size: The SPDS is of a size compatible with the existing space in the control room. It does not interfere with normal movement or full visual access to control board.

Subsection 5.4-Staffing: The design of SPDS is such that no additional personnel are required to monitor the screens during an emergency.

Subsection 5.5-Display Consideration: The SPDS display addresses the five areas of interest. These are related to functions that include the monitoring of Reactivity Control, Reactor Coolant System Integrity, Radioactivity Control, Reactor Core Cooling and Heat Removal and Containment integrity. Each display includes a number of significant parameters for the particular area of interest; in some cases two or more screens are used to aid in following an EOP. The displays are manually selected by the operator.

Two areas not represented on the monitors are (1) the NRC requested source range monitors for reactivity control and (2) no audible notification to alert personnel of a failure of a signal input to SPDS during an operating conditions. Licensee comments indicate that placement of SRM's on screens is under review. The audible alarm has not been addressed.

Section 5.6-Design Criteria: The licensee has addressed the interface of signal conditioners such as isolation devices. The reliability of SPDS is addressed with redundant computers. No technical specifications are necessary based on the licensee's safety analysis and no compensatory measures for the loss of both computers are necessary.

3.4 General Observation and Conclusions

The training provided operators was an informational type of package with training time on console to bring up various screens for EOP training. The validation and verification of SPDS utilized a video licensee's tape that simulated an anomaly. The training department will review the video tape with the intent of presenting the operators with a dynamic training session. The licensee stated that a review of this area of training would be made during the operator requalification program.

The personnel interviewed during inspection demonstrated sufficient depth of knowledge of the SPDS to determine plant condition and implement corrective actions in an efficient manner.

- No audible alarm exists to alert personnel of off normal parameters on SPDS.

- NRR requested the inclusion of Source Range Monitors (SRM's) to be displayed. This item has not been resolved.

- The NUREGS-0696 states that the SPDS shall be capable of presenting magnitude and the trends of displayed parameters. The trending presently does not appear on screen.

- The areas of uncompensated water levels of vessel presently requires calculations to determine true levels and wave motion in suppression chambers need corrective factors. These level calculations should be incorporated in the SPDS program. This item has been identified by the licensee and presently is under review.

The licensee is present SPDS has the ability to aid operators in following the EOP transients. Additional updates of above items will allow a greater flexibility to monitor plant conditions.

3.5 Update of Emergency Operating Flow Charts

While reviewing SPDS parameters to be used by EOPs, the inspector observed that the licensee has developed flow charts for EOPs for use in the Control Room. The flow charts are a duplication of the EOPs and as such must be under administrative control. At the time of the inspection, no procedure exists that would update flow charts if a revision to a EOP is made. This item was brought to the attention of the licensee and remains an unresolved item pending revision of the administrative controls to verify control room flow chart updates. (86-18-01)

4. Anticipated Transients Without Scram Rule (ATWS) 10 CFR 50.62 (25020)

The licensee has scheduled for the next refueling outage, two modifications that will provide additional protection to achieve reactor shutdown. The modifications consist of (1) Alternate Rod Injection System and (2) the Enrichment of Sodium Pentaborate Solution for the Standby Liquid Control System. The following reviews were performed.

4.1 Alternate Rod Injection System (ARIS)

The ARIS modification, MDD-QC643A Rev 2 has been generated to mitigate ATWS, which is an operational event caused by a failure of the Reactor Protective System (RPS) to shutdown reactor. In the event that RPS fails to scram, there was no alternate automatic system to cause control rod injection.

This modification installs a ARI system to cause control rod injection by depressurizing the scram air header.

The inspector reviewed the Modification Design Description which specifies the addition of a new ARI system logic relays, test switches and indicating lights in the rear of Control Panel 8R. Additional changes that include manual initiation, manual reset, loop isolation devices and five new ARI solenoid valves are also added to the scram header. Annunciators are to be added to Control Panel 5F/6F.

The modification package description addresses numerous elements that are to be addressed prior to and during outage changeout. The modification is detailed with clearly defined mechanical and electrical drawings. The work package engineering details were satisfactory. No deficiencies were identified.

4.2 Standby Liquid Control System (SLCS)

The need for additional methods to shut down the reactor in a ATWS event encompassed the need for reduction of risk. The ATWS Rule as specified in 10 CFR 50.62 states that each boiling water reactor must have a SLCS with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute (gpm) of 13 weight percent (wt%) sodium pentaborate solution.

Based on review of licensee documentation a modification will be performed during the next outage. The present solution in SLCS tank is to removed, tank cleaned and a new concentrate solution will be added to conform to the rule. The inspector review included licensee's safety evaluation, installation specification and technical specification submittal. Based on the inspector concerns, the licensee's has under review the examination of the agitation piping, electrical heater element and possible removal of any solidified boron at the tank bottom. In addition the review of Technical Specification submittal is identified the need of for accurate SLCS tank level readouts. The present calibration of the ultrasonic level detector requires removal from the SLCS tank. The licensee has under review the monitoring of levels during pentaborate drainage and refill of SLCS tank. No deficiencies were identified.

5.0 Management Meetings

Licensee management was informed of the scope and purpose of the inspection at an entrance meeting conducted on June 13, 1988. The findings of the inspection were periodically discussed with licensee representatives June 17 during the course of the inspection. An exit was conducted on (see paragraph 1 for attendees) at which time the findings of the inspection were presented.

At no time during this inspection was written material concerning inspection findings provided to the licensee by the inspectors. The licensee did not indicate if any proprietary information was involved within the scope of this inspection.

ATTACHMENT A

A listing of pertinent reference documents is listed by report paragraphs.

Previously Identified Inspection Findings (Paragraph 2)

- Field Change Request (FCR) C-039642. Repairs to Instrument Racks
- Material Non Conformance Report (MNCR) 85-275 Titled Partial Penetration Weld
- Drawings SN15081.02 ES-04 and ES-05
- Material Non Conformance Reports 85-265, 273 and 274 addressed bolting problems
- General Public Utilities Nuclear (BPUN) procedures A15B-G1136.010 Rev. 0, Rack Modification
- Management System Manual No. A00WMS-1220.14 titled Preparation, Review and Approval of Work Procedures
- Seismic Calculation C 1302-251-530-021 V/-1302-251-02
- GPUN Chemical Analysis Report A53 Grade A material
- Welding Manual Procedure, Control of Welding and Brazing No. 6150-QAP-7220.01 Rev. 3

Safety Parameters Display System (Paragraph 3)

- NUREG 0737 Classification of TMI Action Plan Review
- NUREG 0737 Supplement I Clarification of TMI Action Plan Review
- NUREG 0696 Draft Functional Criteria for Emergency Response
- NUREG 0696 Final Functional Criteria for Emergency Response
- Generic Letter 82-33, December 1982
- Licensee submittals of April 2, June 6 and September 1984
- Oyster Creek SPDS-Verification and Validation Report
- On Site Computer Configuration Control - 5000 ADM 7340.02
- NRR Correspondence to GPUN dated July 19, 1984, and March 5, 1986
- Installation Specification for Control Room Console - OC1S-402761-002

Anticipated Transients Without Scram (ATWS) (Paragraph 4)

- Installation Specification for Standby Liquid Control 1.3.328252-001
- Safety Evaluation for the use of Enriched Sodium Pentaborate Solution in the SLCS
- Modification Design Description for Alternate Rod Injection Systems
- Oyster Creek Nuclear Generating Station - Technical Specification (TS) Submittal (TS Change Request No. 162)
- Liquid Poison System Functional Test - No. 612.4.002
- Multi-ranger Programmable Level System (PL-282)
- Temporary Instruction 2500/20 - Inspection to Determine Compliance With The Anticipated Transit With Scram (ATWS Rule 10 CFR 50.62)