U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/84-24

Docket No. 50-219

License No. DPR-16 Priority - Category C

Licensee: GPU Nuclear Corporation
P. O. Box 388
Forked River, NJ 08731

Facility Name: Oyster Creek

Inspection At: Forked River, New Jersey

Inspection Conducted: September 10 - November 30, 1984

Inspectors:	L H Bettenhausen for PCW P. C. Wen, Reactor Engineer	1/9/85 date
	Samuel D Reynolds S. D. Reynolds, Jr., Lead Reactor Engineer, M&PS, EPB	1/8/85 date
Approved by:	L H Bettenhausen L. H. Bettenhausen, Chief, TPS	1/9/85 date

Inspection Summary: Inspection on September 10 - November 30, 1984 (Inspection Report No. 50-219/84-24)

Areas Inspected: Routine, unannounced inspection of startup testing following Cycle 10 refueling. The inspection included licensee action on previous inspection findings, the testing program, precritical tests and power ascension tests. The inspection involved 59 hours onsite and 8 hours in office by two region-based inspectors.

Results: In the areas inspected, no items of noncompliance were identified.

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DETAILS

1. Persons Contacted

E. R. Bujtas, Nuclear Engineer, Oyster Creek Fuel Project
W. J. Enrich, Jr., Senior Engineer
*P. B. Fiedler, Vice President and Director, Oyster Creek
*D. G. Holland, Licensing Manager
*J. R. Molnar, Core Engineering Manager
D. V. Notigan, Engineer
*A. Rone, Plant Operations Engineering Manager
W. Scholtens, QA Lead Monitor
G. Simonetti, QA Lead Monitor
H. S. Sharma, Engineer

USNRC

*C. Cowgill, Senior Resident Inspector
J. Wechselberger, Resident Inspector

*Denotes those present at the exit interview on November 30, 1984.

2. Scram Discharge Instrument Volume (SDIV) Level Sensing Chamber

The inspector reviewed licensee information presented in the following documents concerning radiographic (RT) indications in SDIV Level Sensing Chamber (SDIV-LCS) end cap welds. .

1. GPUN R. Zak to J. Chardos Memo EM-84-999/04301-74 dated 2/22/84.
2. GPUN J. Abramovici to R. Witesel Memo MSS-84-134 dated 3/28/84.
3. GPUN R. Zak to J. Chardos Memo EM-84-1045 dated 4/13/84.
4. GPUN P.O. 201028 and C/N #2 dated 5/14/84.
5. GPUN J. Abramovici to T. Carrie Memo MC-84-2704 dated 8/8/84.

The inspector had telephone discussions with GPUN on 9/6/84 with Messrs. H. Capadano and J. Abramovici to discuss additional details on the SDIV-LCS. The vessel under discussion is a 3/8" wall, 5" diameter, 22" long, 304 pipe with end caps welded on either end. There are 6 vessels (or 12 end cap/shell welds). Five of the welds in three vessels have apparent lack of fusion and/or slag inclusions as determined by re-radiography with IR192.

The approximate circumference is 15". The length of the RT indications are as follows:

1 - one indication 0.5"
2 - one indication 0.6"
3 - one indication 1.0" long and one slag defect 0.3" long
4 - one slag defect 0.3" long
5 - three indications 0.5, 0.6, and 1.0" long

Ultrasonic examination conducted using a qualified procedure with 45°, 60° and 0° scans at 3 times normal sensitivity indicates the defects to be less than detectable by these techniques. The vessels are isolated from the system by manually operated valves.

The FSAR indicates ANSI B31.1-55 is applicable to the SDV system.

GPUN conducted a fatigue (stress) analysis in accordance with the ASME Code, SCIII assuming a (thru the wall) zero defect size with the maximum stress intensity factor of (5). GPUN indicated the vessel passed this engineering evaluation.

The options available for the disposition of these vessels were cutting out the entire units and returning to the vendor, repair welding on site or conducting an engineering evaluation of the indications. The licensee chose the latter option which meets the intent of B31.1 for "applying more complete and rigorous analysis to special and unusual problems" which is stated in the B31.1 "forward" and "introduction".

Based on the licensees foregoing analysis, the ability to isolate the vessels if necessary, the inability to detect the RT indications ultrasonically, and the function of the vessels, the inspector had no further questions regarding this matter.

3. <u>Cycle 10 Reload Safety Evaluation and Core Verification</u>

The Cycle 10 reload contains 200 fresh fuel assemblies (28 ENC Type VB and 172 GE P8X8R) and 360 irradiated fuel assemlies (ENC Type BV) in the core. The characteristics of the Cycle 10 Exxon supplied fuel assemblies (total 28) are the same as used in the Cycle 9 reload fuel. The safety evaluation of the GE supplied reload fuel (NEDO-24195, "General Electric Reload Fuel Application for Oyster Creek") along with the required Technical Specification (TS) change were submitted to the NRC for review. This reload submittal was found acceptable. The licensee independently performed an in-house safety analysis to confirm the GE's calculation results. Both GE and GPU safety analyses show that the Cycle 10 core can be operated and meet the required margins to safety limits.

The inspector reviewed the licensee's own safety evaluation report (TDR-471, "Reload Information and Safety Evaluation Report for Oyster Creek Cycle 10 Reload", Revision 1) to verify the following:

-- overall plant safety margin
-- consistency of the operational parameters
-- implementation of the revised TS changes in station procedures

Shutdown margin was calculated for a full core from the cold Keff with all-rods-out (ARO) and all-rods-in (ARI) configurations using the EPRI-developed NODE-B code. The results indicate that minimum shutdown margin occurs at the beginning of cycle (BOC) and meets the TS requirements. The Standby Liquid Control System shutdown margin was calculated under cold, xenon-free, ARO conditions at the most reactive time. The calculated shutdown margin of 0.043 $\Delta K/K$ is consistent with GE's result of 0.044 $\Delta K/K$ and is well above the TS requirements of 0.010 $\Delta K/K$.

The inspector reviewed the plant operating procedures 1001 series and noted that the operational limits changes associated with the new fuel loading has been incorporated in the procedures. The information used for Cycle 10 startup physics testing was found to be consistent with the values derived from the safety analyses and appropriate TS sections.

The Core Engineering group performed core post-alteration inspection and verification in accordance with procedure 1001.24, Rev. 7. The inspector reviewed the record and verified that the fuel loading agreed with the intended core loading plan.

No discrepancies were identified.

4. Cycle 10 Startup Testing

The unit was started up in late October, 1984 after a 20-month maintenance and modification outage. Initial criticality of cycle 10 was achieved on October 29, 1984. During initial startup, the plant experienced difficulties in Electromatic Relief Valve (EMRV) operability test and was shutdown. After the EMRV problem was corrected, the unit was brought to critical again on November 22, 1984. The power ascension test at 50% power level was completed on November 27, 1984. The remaining power ascension tests will be completed when the unit reaches appropriate power levels. The inspector participated in the NRC augmented shift coverage and observed the activities involved in approaching criticality and power operation during the startup phase. The inspector reviewed selected test programs and their results to verify the following:

-- Procedures were provided with the detailed stepwise instructions, including Precautions, Limitations, and Acceptance Criteria;

-- Technical content of the procedures was sufficient to result in satisfactory calibration and test;

-- Provisions for recovering from anomalous conditions were provided;

-- Methods and calculations were clearly specified and tests were conducted accordingly;

-- Review, approval, and documentation of the results were in accordance with the requirements of the TS and the licensee's administrative controls.

The following tests were reviewed:

4.1 Shutdown Margin (SDM)

The Shutdown Margin Demonstration was performed in accordance with procedure 1001.27, Shutdown Margin Measurement Test, Revision 4.10. The test was performed on July 31, 1984, with moderator temperature of 93°F. A shutdown margin of at least 0.437% ΔK/K was demonstrated with the strongest rod (26-39) fully withdrawn. This measurement of reactivity exceeds the TS required demonstration value of 0.366% ΔK/K (temperature corrected).

A subsequent SDM demonstration was performed to verify that the core remain in subcritical conditions during withdrawal of the control rod diagonally adjacent to a specified fully withdrawn rod. Both A&B sequences (Quarter Core) were demonstrated per procedure 1001.26, Shutdown Margin Demonstration, Rev. 5. The test was completed on August 1, 1984 without the reactor going critical and thus demonstrated the required shutdown margin.

No unacceptable conditions were identified.

4.2 Critical Configuration and Anomaly Check

The inspector reviewed test procedure 1001.2, Estimated Critical Position, Revision 8, and actual critical configuration of November 22, 1984. The estimated critical position (control rod sequence A-1) was group 3 rod (26-31) at position 18. The actual critical position was the same rod at position 20. The inspector verified that the critical rod configuration was within ±1% ΔK/K of the predicted critical pattern.

No unacceptable conditions were identified.

4.3 Core Thermal Power and APRM Calibration

The core thermal power is usually determined by the plant computer. The plant computer monitors the required input parameters and performs a heat balance calculation at one minute intervals. Hand calculation is allowed as a backup when the plant computer is not operating. The inspector noticed that the heat exchange due to CDR flow, cleanup flow, recirculation pump heat and ambient loss were lumped as a single fixed term regardless of power level. The inspector performed an independent calculation by using a more accurate method without simplification. The plant parameters taken at 2305 on November 29, 1984 were used as inputs for comparison.

Test Date	Plant Computer	Procedure 1001.6 Hand Calculation Method	Inspector's Calculation
2305	(MWT)	(MWT)	(MWT)
11/29/84	1008	1004	998

The inspector determined that the method used by the licensee to calculate the core thermal power is conservative. The deviation is expected to be minimized when the core reaches the full power, since the lumped fixed term is determined based on the full power conditions. However, accurate thermal power calculation is important for tracking nuclear fuel performance. The licensee Core Engineering group is currently assessing the impact of this method on the plant operation.

The APRM calibration was performed by adjusting the output using heat balance result. TS requires the APRM calibration be performed once per three days during power operation. The inspector noted that the calibration results were only logged in the control room operator log. To enhance the tracking of the APRM channel behavior and have a better documentation control, a licensee representative agreed to include the APRM calibration in an appropriate test procedure.

The inspector had no further questions.

4.4 Thermal Hydraulic Limits and Power Distribution

The inspector reviewed test procedure 1001.33, Core Daily Checks Using PSMS/MODE-B, Revision 8 and surveillance results of November 26 through 28, 1984. The inspector verified, by review of Computer program PSMS results, that the thermal limits, LHGR, APLHGR, and MCPR were all within the TS limits during this period.

The procedure and method used by the licensee to verify that the plant is operating within the power distribution limits defined in TS were reviewed and discussed with cognizant licensee personnel. The power distribution and associated thermal limits were monitored by the Power Shape Monitoring System (PSMS). The software package is built around EPRI's NODE-B/THERM-B, a three-dimensional simulator code. Comparisons of the PSMS predicted (File: TIPS, PRES. 112884.1) and TIP measured (File: TIPS. CHECK. 112884.1) power distributions were in good agreement with a maximum node error of 7.8%. The PSMS predicted values in this case are prior to TIP adjusted values. The inspector also reviewed the document GPU memo A6584B3002, "Current Version PSMS Readiness for Cycle 10 Operation", dated October 26, 1984. From the information provided, the inspector determined that the calculations from the backup computer program (IBM NODE B/THERM-B and thermal Limits Program - TLP) and PSMS are either identical or fall within the acceptance criteria of ± 0.5%.

No unacceptable conditions were identified.

4.5 Local Power Range Monitor (LPRM) System Calibration

The inspector reviewed test procedure 1001.39, LPRM Adjustment Using PSMS, Revision 0, for technical adequacy. The actual LPRM console meter reading is compared with the TIP reading. The gain of each LPRM amplifier is then adjusted to produce the desired reading. The inspector witnessed portions of the LPRM calibration performed on November 29, 1984, and noted that the required procedure was in use and the calibration was performed by qualified personnel. A subsequent PSMS run was made to verify the adequacy of LPRM's calibration. The comparisons of the calibrated LPRM's readings and PSMS predicted values were in good agreement with overall deviation (PMS) of 8.8%.

No unacceptable conditions were identified.

5. QA Role in Cycle 8 Startup Testing

The inspector discussed the subject of QA's role in Cycle 10 startup testing with cognizant licensee QA personnel. The inspector was told that QA independently verified the core loading and plans to audit refueling activities which include core physics, surveillance and startup tests, six weeks after the plant startup. The inspector reviewed QA monitoring report 84-12011, Black & White Core Tape Review. An unsatisifed finding was generated as a result of this audit. A prompt and complete inspection (84-12017) was initiated by QA and successfully resolved the previous findings.

6. Control Room Observations and Facility Tours

The inspector observed control room operations for control room manning and facility operation in accordance with administrative procedures and Technical Specification requirements. Inspection tours of the Turbine/Building and Reactor Building were conducted.

No unacceptable conditions were identified.

7. Exit Interview

Licensee management was informed of the purpose and scope of the inspection at the entrance interview. The findings of the inspection were periodically discussed and were summarized at the conclusion of the inspection November 30, 1984. Attendees at the exit interview are denoted in paragraph 1.

No written material was provided to the licensee by the inspector at any time during this inspection.