U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NO: 50-289/94-25

LICENSEE: GPU Nuclear Corporation

FACILITY: Three Mile Island, Unit 1
Middletown, Pennsylvania

DATES: December 5, 1994 through January 20, 1995

LEAD INSPECTOR: Harold I. Gregg 2/17/95
Harold I. Gregg, Sr. Reactor Engineer Date
Systems Section
Division of Reactor Safety

APPROVED BY: Eugene M Kelly 2/17/95
Eugene M. Kelly, Chief Date
Systems Section
Division of Reactor Safety

SUMMARY: Management oversight and involvement in engineering activities were evident at the site, and between the site and corporate engineering. The design modifications reviewed contained good modification descriptions, 10 CFR 50.59 reviews, and safety evaluations. Plant experience reports on the pressurizer spray valve body-to-bonnet leak, control rod drop times, and the leaking pressurizer safety valve were comprehensive. The licensee's design basis document program has recently shifted to preparation by licensee staff at the site. Self-assessment-directed training of plant-experienced problems and their causes was objective. The self-assessment training given to all engineers presented good topics that included a recent water-solid pressurizer event (at another facility), and the TMI pressurizer spray valve body-to-bonnet leak problem. The presentations were objective and, in the case of the pressurizer spray valve, self-critical and useful.

DETAILS

1.0 ENGINEERING AND TECHNICAL SUPPORT (IP37550)

The objectives of this inspection were to review and evaluate engineering activities at the TMI-1 Nuclear Station. The inspection scope included review of design modifications, a review of site engineering organizations and their performance in support of the plant, and management oversight. Activities developed and performed by engineering, such as plant experience reports, material nonconformance reports, resolution of licensing actions and licensee event reports, and self-assessment directed training were also selected for review.

2.0 DESIGN MODIFICATIONS

The inspector reviewed several modifications developed by TMI-1 plant engineering that were in process, or recently completed and implemented, to assess the effectiveness of the design packages, the modification implementation, and adherence to procedural requirements. Discussions were held with licensee's personnel, and walkdown observations were made by the inspector.

The inspector concluded the following modifications were appropriately described with clear and detailed instructions in each package reviewed. The 10 CFR 50.59 reviews and safety evaluations were comprehensive and consistent with established procedures and regulatory requirements. No discrepancies were identified relating to either the modification packages or quality of workmanship of the installed modifications.

- CMR No. 94-019, dated March 8, 1994. The scope of this configuration change pertained to installing a second latching relay and wiring the relay coils in parallel with the existing coils and connecting the data acquisition system input wiring to dry contacts from the new relays. This modification provides status on whether loop "A," loop "B," or loop "A/B" average reactor coolant temperature signals are selected for integrated control system control.

- CMR-90-045, dated March 22, 1990. Revising the security diesel generator (EG-Y-0002) logic circuits that provide the "trouble" signal that initiates the PRF7-4 annunciator in the control room. Both changes were made within the boundaries of the diesel generator cabinet and controller electronic tray that were located within the security building.

- CMR 90-181, dated November 13, 1990. Replacement of obsolete L&N and TIGRAPH recorders, required by Regulatory Guide 1.97, installed in the radiation monitoring system.

- CMR 92-143, dated November 19, 1992. Replacement of two existing time delay relays in the EG-Y-4 station blackout diesel generator governor control circuit. The relays were working intermittently and determined to be unreliable.

- CMR 94-004, dated January 20, 1994. Replacement of obsolete Lambda Power Supply M/N LXS-CC-28R with new Lambda Power Supply M/N LNS-Y-28 in each of the four reactor protection system cabinets.

3.0 PLANT EXPERIENCE REPORTS

The inspector reviewed three plant experience reports (PERs) of high visibility safety-related component problems that were developed by plant engineering. The PERs reviewed were: PER 94-002, "Control Rod Drop Times," PER 94-001, "RC-VI, Body to Bonnet Leak and Degraded Bonnet Studs," and PER 93-004, "Leaking Pressurizer Safety Valve (Serial No. BL-08898)." The inspector found these PERs to be comprehensive plant engineering products that are sponsored by plant management. The PERs included detailed technical descriptions of the problem, the problem cause, safety implication, recommended actions to prevent recurrence, and discussions of the NRC concerns. The PER on the pressurizer spray valve body-to-bonnet leak and degraded studs was especially objective and comprehensive. Several root causes were discussed, involving: 1) inadequate evaluation of a large increase in the motor operator torque setting that resulted in loss of body-to-bonnet gasket preload, and 2) failure to remove insulation to fully determine the leak source (thought to be a packing leak).

4.0 MATERIAL NONCONFORMANCE REPORTS ON NUCLEAR SERVICES RIVER WATER SYSTEM

The inspector reviewed the licensee's material nonconformance reports (MNCRs) for two recent small through-hole leaks in the nuclear services river water system. The inspector considered the technical approaches and resolution for these MNCRs to be comprehensive. One was the through-hole leak in a 45° downslope section of the backwash supply line dead leg (MNCR 94-0019 of 7/12/94), and the second was a similar through-hole leak (less than one liter/day) in a vertical section of a backwash discharge line (MNCR 94-0026 of 11/21/94).

Plant engineering had strong involvement with the MNCR and developed the engineering evaluation and with utilization of the NDE inspection data developed an effective repair resolution. Repair for MNCR 94-0019 was made by welding a 2" diameter half-coupling over the leak hole and installing a threaded plug in the coupling. The repair of MNCR 94-0026 was similar. Management has committed resources and TF has the lead in developing or expanding the erosion-corrosion program to address this corrosion. Water samples have shown a biologically-aggressive environment, and radiographs of the pipe near the leaks show numerous areas of pitting and nodule buildup indicative of microbiological influence corrosion (MIC). The licensee contracted a consultant who was presenting MIC training on December 6-7, 1994, during this inspection.

A meeting of the MIC task group was held at TMI on January 19, 1995, to provide a status update on the problem and to develop plans for further actions.

5.0 LICENSING OPERATING EXPERIENCE PROGRAM TRACKING

The inspector reviewed the licensee's tracking system that addresses NRC commitments, information notices (INs), licensee event reports (LERs), TS change requests, generic letter activities, inspection report responses, and other licensing action items (LAIs). The licensee had recently implemented an upgraded computer program and data base. The system enables searches for both NRC required activities and the licensee's internal action activities.

Several licensee responses to NRC INs and a LAI tracking activity were selected by the inspector for review of engineering resolution and are described below.

- IN 94-44, Main steam isolation valve failure to close on demand because of inadequate maintenance and testing. Plant engineering reviewed the IN and responded with a written memorandum that provided appropriate technical information that enabled closure of this item.

- IN 94-45, Potential common-mode failure mechanism for large vertical pumps. Plant engineering had review responsibility for this IN and documented by memorandum the technical details of differences in the pumps on site that provided the appropriate closeout conclusion.

- IN 94-48, Snubber lubricant degradation in high temperature environments. Plant engineering reviewed this IN and provided technical details as to why this had no applicability to TMI.

- LAI 94-002, Pressurizer code safety valve setpoint analysis. Corporate engineering performed the requested evaluation and provided justification based on a BWOG analysis and a plant specific analysis for increasing the pressurizer safety valve setpoint to $\pm 3\%$ due to concerns of setpoint drift beyond the $\pm 1\%$ TS requirement. The evaluation also provided technical reasons for not pursuing the TS change. The inspector determined the LAI 94-002 engineering evaluation and calculation C-1101-223-5400-009 to be comprehensive and well performed, and the item was appropriately closed by the licensee.

The inspector concluded that the licensing action tracking system was effective, and that engineering provides the technical basis or corrective action to enable closure of the item.

6.0 DESIGN BASIS DOCUMENT ACTIVITIES

The inspector evaluated the current matrix of design basis documents (DBDs). The licensee has recently implemented a major DBD program change, that of developing DBDs under the direction of the corporate technical function's mechanical engineering director at the site. A group of six technical personnel was authorized, positions were filled, and the activity has started. The inspector met with the corporate TF engineering director and attended the meeting held with the site DBD staff to review the assigned DBDs and their status. The licensee's objectives are to produce improved DBDs that have additional GPU documentation references, build and maintain in-house knowledge, and provide awareness and visibility of DBDs throughout the plant.

DBDs for emergency Power 1E, electrical distribution 4160/480V, electrical distribution vital ac, electrical distribution dc, and core flooding system are in the initial reference data gathering phase. DBDs for secondary river water and closed cooling water, decay heat river water and closed cooling water, and nuclear service river water and closed cooling water and intermediate closed cooling water are in the 10-30% initial draft phase. The licensee plans to do a service water inspection in the spring of 1995 and is expediting completion of the three service water DBDs.

7.0 PREVIOUSLY-IDENTIFIED NRC ITEMS

(Closed) Unresolved Item 50-289/90-81-02 pertaining to 120 Vac voltage drop calculations.

This item pertains to the verification and acceptability of voltage levels at end devices/component terminals supplied from 120 Vac distribution panels VBA, VBB, VBC, and VBD. During a July 1993 NRC inspection (Report No. 93-15), the inspector reviewed all four voltage drop studies for the incoming and outgoing circuits for each of the four vital power ac distribution panels. Based on these calculations, the voltage levels at the end devices/components, including the inverters, were evaluated to verify that the available voltage was greater than the minimum voltage needed for operation. Calculation Nos. C1101-735-5350-003 and -004, except for Circuits EA-6872 and EA-6873, contained the voltage drop calculations for vital power ac distribution panels VBA and VBB. Calculation Nos. C1101-735-5350-005 and -006 contained the voltage calculation for vital power ac distribution panels VBC and VBD. A base voltage for inverter output of 115.64V and voltage drop consideration for feeder and branch circuits was used to determine the voltage at the end device terminals. Assumptions made within all four calculations were appropriate and conservative. For circuits EA-6872/EA6873, GPUN recommended replacement of existing cables with larger cables to reduce the voltage drop across them and improve end device terminal voltage. However, GPUN also indicated that current/voltage measurements would be performed to verify the necessity of increasing cable size.

During this inspection, the inspector reviewed the supporting documentation of current and voltage measurements made prior to and during the monthly channel test (SP1303.4.23) of the Reactor Building Post-LOCA Hydrogen Monitor. The test data indicates that the voltage available at the hydrogen monitor was 103.75 Vac during post-LOCA when the system was fully loaded. In addition, vendor test data indicates that the system equipment operated effectively at 80% of the rated 120 Vac voltage (96 Vac). The inspector concluded that circuits EA-6872/EA-6873 to the hydrogen analyzer, as presently designed and installed, would perform their function at 103.75 volts and above during post-LOCA when the inverter output voltage could be at its design minimum of 115.64 volts.

(Closed) Unresolved Item No. 289/90-81-03 regarding the ac system fault analysis.

The EDSFI team's review of the adequacy of the interrupting capability of the Class 1E distribution system equipment identified discrepancies between the short-circuit study and actual plant equipment settings. During a follow-up NRC inspection (Report No. 93-15) and in response to this item, GPUN initiated Licensing Action Item (LAI) 91-9060 to perform a short-circuit study (Calculation No. C1101-700-5350-006) with proper considerations for auxiliary transformer impedances based on actual tap settings of -2.5% and a maximum grid voltage of 242 kV.

The inspector reviewed Calculation No. C1101-700-5350-006, Revision 0, dated January 25, 1994. The calculation determined the short-circuit currents at each bus in the TMI-1 electrical distribution system, during worst-case conditions. The tabulated results of Calculation C1101-700-5350-006 indicate all values for medium voltage interrupting and momentary fault currents were within the breaker ratings, and that all short-circuit currents were within equipment ratings for the worst-case condition. No revisions to the coordination data were necessary.

(Closed) Unresolved Item 50-289/92-04-01, pertaining to the inspection concern that the licensee's response to the industry initiated check valve program was not in a consolidated program document.

A "Program Description" for the industry initiated check valve inspection and testing program, dated April 28, 1994, was developed by TF and was concurred on by plant engineering and approved by Plant Operations and Maintenance. The documented record of the program was placed in the licensee's computer-assisted record information retrieval system (CARIRS) as Document No. 990-2243, Revision 0.

The inspector reviewed the industry check valve program description and found it was comprehensive and described the program objectives to provide testing, inspection, and maintenance of check valves within safety-related systems, or within systems important to plant availability. The program defined the valves in the program, the departmental responsibilities for its implementation, and detailed descriptions of each program activity. Additionally, the program description provided justifications for exclusion of specific check valves from the industry check valve inspection and test program.

The inspector determined that document revisions would be implemented through CARIRS, and would require the same level of signature review and distribution of the original document. The licensee's check valve program appropriately addressed the IR 92-04-01 concern.

<u>(Closed) Unresolved item 50-289/92-04-03</u>. An inspection finding identified there was no documented justification for three valves (DH-V50 and 59A/B) not being in the inservice test program (IST).

The inspector reviewed the plant engineering memorandum, dated July 7, 1994. Due to the NRC inspection finding and the licensee's low pressure injection safety system functional inspection concern that leakage through these type valves could lead to offsite radiological release, plant engineering reviewed and reevaluated the IST boundary drawings. The plant engineering memorandum also noted that NRC Inspection and Enforcement Notice IEN 88-70 also addressed concerns with the testing of check valves in the closed position. The plant engineering evaluation of each valve (NRC and licensee-identified) and the action to be implemented by plant engineering, technical functions, operations and licensing were described in the memorandum. The inspector determined the licensee was also in the process of the completion of new IST testing procedures (1300-32.1 and 32.2) for the emergency core cooling (ECCS) bypass valves. Because the task of leak testing of these valves is complex, these procedures required several drafts and comment resolutions.

The inspector reviewed the specific licensee's actions planned for DH-V50 and 59A/B. These valves are to be included in the IST program, and there is a need to state they are under constant test, since observation of the borated water storage tank level verifies valve closure. There is also need for a relief request because leakage through pairs of valves is involved and it is not known which valve is holding. These actions were considered appropriate by the inspector. This item is closed.

<u>(Closed) Violation 50-289/92-80-03</u>. Cracked actuator housing of MOV MS-V-2A.

The licensee replaced the cracked actuator with an actuator from the warehouse. Disassembly at the GPUN system's laboratory with Limitorque participation identified a damaged torque switch dowel pin. Laboratory analyses of the housing crack surfaces concluded that the cause of the crack was independent of the torque switch dowel pin failure, but was due to an overload that occurred in three distinct steps. The licensee's root cause conclusions were: the crack was due to overload caused by use of a handwheel helper; a shock load was caused by stem/disc rebound on unseating; or there was a flaw in the initial fabrication. The GPUN measures to prevent recurrence included adding maintenance procedure guidance on overthrusted valves and inspecting torque switch dowel (or roll) pins during MOV preventive maintenance.

This item was reviewed and updated in NRC Inspection Report (IR) 94-12. Because post-test evaluations fell short of NRC expectations, three specific corrective action concerns were identified in the IR 94-12 update. The inspector reviewed the licensee's corrective measures that were implemented in response to the three specific NRC concerns.

For concern (1), plant engineering has developed and implemented a "GL 89-10 Motor-Operated Valve Operability Checklist" that contains parameters for operability determination. The checklist contains the acceptance criteria for each of the parameters, the actual test result data and the field engineers initials, and a comment section for exceptions and other observations. Plant engineering has reviewed each GL 89-10 valve test result, and has upgraded the test records with a completed operability checklist for each valve. Incorporation of the operability checklist in procedures was assigned to plant electrical engineering to be completed by March 3, 1995.

For concern (2), in addition to the trending of failures process defined in Administrative Procedure (AP) 1073, the plant electrical engineering memorandum of December 20, 1994, has defined the parameters to be trended for the GL 89-10 valves. The parameters to be trended will be included in a new plant engineering or maintenance procedure, not yet determined, that has been tasked for completion. The licensee had not specified the date for completion of this action.

For concern (3), the technician performs the test, makes necessary adjustments, records the test data, and signs the operability data sheet. The test data is independently reviewed and signed off by the plant engineer. The test reviews verify acceptance criteria are met and are signed and dated prior to the shift supervisor returning the MOV to service.

The inspector's review of each of the concern resolutions determined that they appropriately addressed the inspection (IR 94-12) concerns. The inspector reviewed selected valve test certification packages (valves MS-V-2A, DH-V-5A/B, RB-V-2A), and verified the operability checklist was included in each package. The inspector also reviewed Plant Procedures 1420-LTQ-5 and 1420-LTQ-7 and verified that the licensee's letter, dated August 24, 1992, commitment to provide overthrust guidance in the MOV maintenance procedures was implemented. These procedures contain a precaution that the actuator thrust rating should not be exceeded (the actuator thrust ratings are included).

<u>(Closed) Unresolved Item (UNR 50-289/94-05-01)</u>, relating to leaking pressurizer safety valve (SV) and manual lever exercising.

Concerns were identified with the initial planned actions to attempt reseating/stop leakage of the pressurizer safety valve RC-RV-1B at power with the use of a lifting lever extension. The attempts to reset the valve were performed with the plant in hot shutdown condition (not at power), and with a gag in place at ⅛ turn to a maximum of ¼ turn open to permit disc lift of 0.014" to 0.028". Attempts on November 14, 1993, to stop the leakage were unsuccessful and the plant was shutdown, a replacement valve was installed and the leaking SV was shipped to the test laboratory for evaluation and repair. The licensee's plant experience report (PER 93-004) described the events and prescribed recommended actions. The NRC IR 94-05 defined actions were to review the licensee's final corrective actions, changes to the specifications and procedures, safety evaluations, and the laboratory and manufacturer reports (Wyle and Dresser).

The inspector verified that procedure changes, as recommended in the licensee's PER, were implemented. The inspector determined that: Specifications SP 1101-12-020, Revision 5, and SP 1101-12-102, Revision 2, contained the revised seat leakage acceptance criteria "zero leakage, no fog";

Maintenance Procedure 1401-2.1 contained guidance in handling PSVs to prevent seat leakage; Surveillance Procedure 1303-8.1 incorporated actions to verify PSV seat tightness prior to taking the reactor critical, and to implement Operating Procedure OP 1103-5; and Operating Procedure OP 1103-5 was revised to include lever extension attempts to stop leakage within 24 hours at hot shutdown conditions and with the PSV gagged ⅛ to ¼ turn (a gag with a ⅞"-9TPI allows a disc lift 0.014"-0.028"). The licensee was in the process of issuing a PCR to add a note under "Section 3.0, Plant Status" procedure that reactor is not critical.

The Dresser and Wyle reports of the as-found testing and disassembly of the leaking PSV did not detail or identify the cause of leakage. The Wyle report reviewed by the inspector, reported as-found seat leakage of 1645 ml/10 min. at 2,250 psig; as-found setpoint was 2,487 psig, and within acceptance criteria of 2,500 psig ±1%; and post as-found test identified visual leakage at 1,050 psig. The report identified that the nozzle, disc, and disc holder were steam cut. It was noted by the inspector that the plant engineer who had responsibility for PSV's witnessed the testing and valve disassembly.

The Dresser report also did not identify a cause of seat leakage. It identified the bellows assembly nose was stretched beyond acceptance. The report also identified the nozzle, disc, and disc holder to be steam cut. The licensee's PER action to use accelerometers on the valves or shipping containers was not implemented. This was due to inability to size the accelerometer, and because an acceleration acceptance criteria to prevent leakage couldn't be determined. Therefore, the use of accelerometers was determined not to be necessary.

8.0 QUALITY ASSURANCE AUDIT

The inspector reviewed Audit No. S-TMI-94-12, conducted between August 8 and October 13, 1994, that assessed fire protection equipment and program implementation. Two audit findings were issued as a result of the TMI audit. Finding No. 1 involved TMI-1 failure to perform safety evaluations for specific modifications and subsequent procedural changes to reflect the modification. Finding No. 2 found that several preventive maintenance procedures were not scheduled or were scheduled at a less conservative frequency than specified.

NRC inspection of selected plant areas found that fire protection features specified by the safety analysis review (SAR) were installed. The inspector's review of one hour fire watches in areas containing TSI Thermolag also found that the fire watches were being appropriately conducted on an hourly basis. The inspector's review of the fire brigade performance (observed by GPUM during a previous drill) identified no deficiencies. Review of surveillance results and corrective maintenance found that the fire protection equipment was well maintained.

9.0 SELF-ASSESSMENT-DIRECTED TRAINING

The licensee incorporates lessons learned into engineering support personnel requalification training that is given semiannually. This is required training for all engineering personnel, including operations and maintenance engineers. The current GPUN training module included: a recent event at a pressurized water reactor (PWR); the problem causes of the TMI pressurizer spray valve body-to-bonnet leak and degraded studs; the DBD program and its usefulness; and, the new configuration change procedure for plant engineering-directed small modifications. The inspector attended the licensee's training sessions on the PWR event and the TMI pressurizer spray valve problem. Each of these issues was presented well and was unbiased (i.e., explained what was done inappropriately), and there was good discussion and attendee questions during each presentation.

10.0 MANAGEMENT OVERSIGHT

The inspector reviewed the safety issue assessment program (SIAP), which includes plant engineering, plant operations, technical functions engineering and design, and technical function system engineers. The inspector noted that SIAP was used by the independent on-site safety review group (ISORG) to maintain a list of safety issues applicable to GPUN nuclear power plants and to provide relative ranking of the safety issues. SIAP meets semiannually to review, revise, and bring new issues to management attention. Another management tool is the monthly monitoring report that summarizes the significant activities of the TMI-1 Nuclear Safety Assessment Group.

The president of GPUN made a notable presentation titled, "Nuclear Safety Culture," to the EPRI-sponsored Operational Reactor Safety Engineering and Review Group in September 1994, in Dallas, Texas. The presentation message was that safety is of uppermost importance, and management must be responsible for developing and promoting a nuclear safety culture.

11.0 EXIT MEETING

During the course of the inspection, the inspector's findings were discussed with licensee representatives. Exit meetings were conducted on December 8, 1994, and January 20, 1995, at which time the preliminary findings were presented. Lead licensee personnel in attendance at the exit meeting are listed below. The licensee acknowledged the inspector's findings and conclusions and had no additional comments regarding the inspection results. The bases for the preliminary conclusions did not involve proprietary information.

T. Basso	Manager, Plant Engineering
T. Broughton	Vice President/Director TMI
R. Knight	Licensing Engineer
M. Nelson	Manager, Nuclear Safety
G. Skillman	Technical Functions Site Director, TMI
C. Smyth	Manager, Nuclear Safety Assessment
P. Walsh	Plant Engineering Director
J. Wetmore	Manager, TMI Licensing