U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-289/88-21

Docket No. 50-289

License No. DPR-50

Licensee: GPU Nuclear
P.O. Box 480
Middletown, Pennsylvania 17057

Facility Name: Three Mile Island Nuclear Station, Unit 1

Inspection At: Middletown, Pennsylvania

Inspection Conducted: August 1-5, 1988

Inspector: Joseph A. Golla
Joseph A. Golla, Reactor Engineer — 8-30-88 date

Approved by: P. K. Eapen
Dr. P.K. Eapen, Chief, Special Test Programs Section, EB, DRS — 8/30/88 date

Inspection Summary: Inspection on August 1-5, 1988 (Inspection Report No. 50-289/88-21)

Areas Inspected: Routine unannounced inspection of test witnessing and administrative control of Local Leak Rate Testing (LLRT) and followup of previously identified open items.

Results: No violations or deviations were identified. LLRT was found to be implemented adequately. Three open items were closed.

DETAILS

1.0 Persons Contacted

* R. Barley, TMI-1, Manager of Plant Engineering
  J. Bashista, Plant Engineering
  T. Graham, QC Supervisor
* G. Gurigan, GPUN Licensing
  D. Hosking, Operations QA Manager
* C. Incorvati, TMI Audit Manager
  M. R. Knight, Licensing Engineer
  R. Stoehr, Plant Engineering Department LLRT Cognizent Engineer
  R. Summers, Supervisor Plant Engineering
  D. Washko, Plant Engineering
  P. Webber, I&C Foreman

Nuclear Regulatory Commission

* D. Johnson, Resident Inspector
* T. Moslak, Acting Senior Resident Inspector
* A. Sidpara, Resident Inspector

* Indicates those present at the exit meeting held on August 5, 1988.

2.0 Local Leak Rate Testing (LLRT)

2.1 Inspection Purpose and Scope

The purpose of this inspection was to ascertain that local leak rate testing is being administered adequately and conducted in compliance with the requirements and commitments referenced in the following section. The LLRT procedures were reviewed for technical adequacy to perform the intended activity. Other record keeping and LLRT related documentation were reviewed to determine adequacy of overall administrative control of the local leak rate test program.

At the time of the inspection approximately 81 of 89 local leak rate tests scheduled for the current outage had been performed. The total maximum pathway leakage at that point was not yet calculated because testing was still ongoing. The licensee must show that the combined maximum pathway leakage for all penetrations testable under the requirements of 10 CFR 50, Appendix J is less than .6La before starting up from this (7R) outage.

2.2 References

• 10 CFR 50, Appendix J, Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors.

- TMI Unit 1 Technical Specifications, Section 4.4.1.2.
- Final Safety Analysis Report (FSAR)
- ANSI/ANS 56.8-1981, Containment Systems Leakage Testing Requirements.
- USNRC IE Information Notice Number 85-71, Containment Integrated Leak Rate Tests.

2.3 Documents Reviewed

- Current and previous outage LLRT results.
- QC Plant Inspection Reports and related QA auditing documentation.
- LLRT instrument calibration documentation.
- Standardization Procedure of Local Leak Rate Testing Rotameters No. 1430-Y-22, Revision 5.
- Reactor Building Local Leak Rate Testing Procedure; No. 1303-11.18, Revision 42.

2.4 LLRT Procedure Review and Administrative Control

The inspector reviewed the LLRT procedures to determine their technical adequacy. The local leak rate test procedure is structured such that each penetration is treated by a subsection which covers plant status required for the test, test limitations and precautions, diagram of the penetration showing the valve lineup to be used for the test, and test equipment required. Local leak rate test methods include mass flow-in for penetrations involving containment isolation valves and pressure decay for airlocks. These methods are acceptable per 10 CFR 50, Appendix J and current industry practice. The inspector verified that the licensee is calculating penetration leakage utilizing maximum pathway methodology per the industry standard ANSI/ANS 56.8-1981, Containment Systems Leakage Testing Requirements. It is the NRC's position that penetration leakage be calculated in this manner. The inspector verified by an audit of records that containment isolation valves and other penetrations are being local leak rate tested at their required frequency. The inspector also verified that the licensee is performing as-found LLRT's on all penetrations and subsequent as-left LLRT's for those penetrations which received maintenance on some component or seal which affects the pressure boundary. No unacceptable conditions were identified.

2.5 Test Witnessing

The inspector witnessed the performance of test activities to verify that qualified test equipment and tools were used and that the test technicians were competent to perform their duties. The following mass flow-in tests were witnessed:

2.5.1 Instrument Air Valve No. 6 and 20 at containment penetration No. 109 on August 2, 1988.

2.5.2 Once Through Steam Generator (OTSG) Chemical Cleaning penetration Nos. 105 and 106 on August 3, 1988.

The inspector reviewed the system lineup for the tests witnessed and determined that they were in an effective test configuration. The tests passed their specific target criteria and the penetrations did not require adjustments or maintenance to pressure retaining components. The test technicians followed approved procedures, utilized qualified test equipment, and were knowledgeable and competent in local leak rate testing. No unacceptable conditions were identified.

3.0 Personnel Training and Qualifications

The qualification and training of selected test personnel were discussed with a licensee representative. In addition, the inspector evaluated the performance of test technicians during the test witnessing. It was determined that LLRT technicians attended a training class given by experienced LLRT personnel. Attendance records were reviewed by the inspector.

The inspector determined that the test technician's qualifications met the requirements specified in ANSI N 18.1-1971 "Selection and training of nuclear power plant personnel". They were knowledgrable of their responsibilities and technical aspects of leak testing. No unacceptable conditions were identified.

4.0 Test Instrument Calibration

The inspector reviewed calibration records for the LLRT instruments (pressure gages and rotameters) being used this outage. All instruments used were found to be in current calibration. The standards used to calibrate the instruments were in current calibration and certified to be traceable to the National Bureau of Standards (NBS). No unacceptable condition was identified.

5.0 Quality Assurance and Quality Control

The inspector discussed coverage of local leak rate testing with representatives from the QA and QC organizations. Several quality control plant inspection reports and a quality assurance monitoring report covering local leak rate test activity monitoring this outage were reviewed by the inspector. It was determined from the discussions that QC covers surveillances from a list provided by QA which is generated from the weekly maintenance plan. QA and QC monitoring/inspection report findings were well documented. The QA and QC representatives interviewed were knowledgeable of local leak rate testing and their duties as QA/QC inspector/auditors. It was noted that a QA monitor was present at the LLRT witnessed on August 3, 1988, Section 2.5 of this report. The attention that the QA and QC departments are giving to local leak rate testing appears to be adequate. No deficiencies were identified within the scope of this review.

6.0 Plant Tour

The inspector made several tours of the plant facilities including the intermediate building, turbine building, control room, and plant exterior to monitor outage activities and housekeeping. All areas inspected were generally clean and free from debris. No unacceptable conditions were identified.

7.0 Follow-Up of Previously Identified Open Items

(Closed) Unresolved Item 50-289/87-09-19: Plant Modifications/Procedure Changes to Allow Inservice Testing of Components Prior to Startup From 7R

The licensee was required by NRR to make system modifications to allow pump flow measurements to be taken per Section XI of the ASME Boiler and Pressure Vessel Code. The pumps involved are the fuel oil transfer pumps DF-P1A, P1B, P1C, P1D, the control building chilled water pumps AH-P-3A/B, and the screenhouse ventilation equipment pumps SW-P-2A/B. Additionally, the testing of six check valves in the fuel oil transfer system, DF-V-23A/B, DF-V-7A/B, B/A, A/B, and B/B needed to be proceduralized. The inspector observed the newly installed flow meters in the system piping of the associated pumps identified above. He also reviewed newly revised IST procedures which require and provide for flow measurements. The revised IST procedures which implement the changes are:

1. "Emergency Power System" No. 1303-4.16, Revision 41. This procedure provides for the flow measurement of the fuel oil transfer pumps and IST of the six fuel oil transfer system check valves identified above.

2. "IST of AH-P-3A/B and Valves" No. 1300-3N, Revision 16. This procedure provides for the flow measurement of these control building chilled water pumps.

3. "IST of SW-P-2A/B and Valves" No. 1300-3M A/B, Revision 22. This procedure provides for the flow measurement of these screenhouse ventilation equipment pumps.

The licensee indicated that the pump flow reference values required by Section XI of the ASME Code would be established after the first performance of each test. The inspector reviewed the modification packages which administered the installation of the flow devices in the above systems. All provisions made for the flow measurement of pumps and the IST of fuel transfer system check valves to address this issue were found adequate. This item is closed.

(Closed) Unresolved Item 50-289/86-21-01: Penetration Pressurization System Check Valves at Containment Purge Valves Interspace to Receive Corrective Maintenance After Failing the November 1986 Unit 1 Containment Integrated Leak Rate Test (CILRT)

During the initial stages of the November 1986, "as-found" CILRT performance the licensee identified and quantified several containment boundary leaks. Of these, the containment purge line manifold "J" was shown to be a major contributor to the boundary leakage which failed the as-found portion of the CILRT. Check valves PP-V-101 and 102 were identified as the source of leakage at this manifold. These valves were subsequently isolated from the containment boundary and a CILRT was conducted for the purpose of establishing an as-left overall containment leak rate. Upon completion of the CILRT the licensee was evaluating corrective and preventive measures for the penetration pressurization check valves. This unresolved item was written to track completion of the above corrective/preventive measures.

The leaking check valves are associated with the automatic actuation of the penetration pressurization system into the interspace between reactor building purge valves AH-V-1A/1B and 1C/1D. A modification has been performed by the licensee whereby check valves PP-V-101/102/133/134 were converted to normally closed manual glove valves and previous automatic initiation valves were failed open. The check valve internals were removed from the valve bodies and renewable threaded globe valve seat rings and bonnet assemblies were installed. Check valves PP-V-101/102/133/134 were then renumbered PP-V-210/212/213/211 respectively. The licensee is maintaining the new globe valves in the normally closed position.

The inspector while on site observed the new globe valves in place. Local leak rate testing of these valves was ongoing. The inspector also reviewed the modification package associated with the above changes to the penetration pressurization system. He verified that a safety evaluation was written by the licensee per the requirements of 10 CFR 50.59, "Changes, tests and experiments" addressing the modification. The inspector reviewed the safety evaluation and found it acceptable. No credit is taken in the plant Technical Specifications or FSAR for the penetration pressurization system performing an active, automatic function, therefore, there will be no increase in the probability or consequences of an accident previously evaluated. The licensee indicated that leak testing history has demonstrated that the reliability of globe valves is much better than check valves and that containment integrity will be made more reliable by this modification. The inspector found the licensee's assessment concerning improvement to the containment boundary acceptable. The new globe valves will be added to the plant Technical Specification 4.4.1.2.1 which identifies isolation valves testable as defined in 10 CFR 50, Appendix J. This item is closed.

(Closed) Unresolved Item 50-289/85-26-06: Nuclear Service Valve Penetration Design Adequacy

A question arose during a previous inspection concerning the design adequacy of three containment penetrations. The penetrations in question are Nos. 404, 405, and 406 of the Nuclear Services Closed Cooling Water (NSCCW) System. These penetrations are Nuclear Services (NS) Closed Cooling water return lines (seismic class I) from the Reactor Building Fan Motor Coolers. Each of these return lines has a relief valve (NS-B-36A, B & C) located between the outside containment wall and a control valve. The supply lines have a single isolation valve at the outside containment wall. The coolers are located inside containment.

The purpose of the relief valves is to provide overpressure protection for the emergency fan motor coolers due to thermal expansion of trapped fluid. The thermal expansion would result from heating of the fluid in the cooler motor with both the inlet and outlet isolation valves closed. This system is closed loop. The question which arose concerning the design adequacy of these penetrations addressed the implications of having a stuck-open relief valve outside of containment concurrent with a break of the NSCCW piping inside containment. The relief valves are nominally set at 175 psig and the lines are needed for motor cooling. Containment accident pressure is 53 psig. This piping system is seismically designed and needed for containment post-accident mitigation.

This issue was addressed previously by a region based specialist inspector, who concluded that there was no safety concern. For an accident scenario in which the cooler pressure integrity is breeched, the maximum pressure applied to the underseat of the relief valve would be the peak containment accident pressure (53 psig) which is insufficient to lift the relief valve (set to lift at 175 psig). If the valve did lift, with inlet and outlet isolation valves closed, to provide thermal relief it would be passing uncontaminated closed cooling water. This would imply that the cooler's pressure boundary was still intact, thus precluding any escapement of building atmosphere to the outside. A postulated NS pipe break with containment pressure at the relief setting (175 psig) is beyond the design basis event.

Another question addressed the penetration design with regard to current general design criteria (GDC) No. 57 of 10 CFR 50, Appendix A. This criterion states that containment isolation valves for closed-cycle systems that penetrate containment either be automatic or locked closed or capable of remote isolation. A relief valve is none of the above. However, criterion 50 of the GDC states that penetrations, and heat removal systems shall be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the calculated pressure and temperature conditions, resulting from any loss-of-coolant accident. The reactor building fans and the NSCCW to these fan motors are needed for post accident mitigation. The normal and post accident position of the isolation valves for each loop (there are three loops) is open. To require that these penetrations conform to a requirement for valves that isolate during an accident would be incongruous. Furthermore, the design is in accordance with FSAR commitments on which the operating license was issued in 1974. Accordingly, this aspect of the unresolved item is closed.

Finally, another question was raised regarding the inaccurate depiction of these NSCCW penetrations in the FSAR. The updated FSAR Figure 5.3-1, Valve Arrangement No. 23 did not show the subject relief valves on the return lines between the outside isolation valve and the containment wall. A licensee representative indicated at the time that the system drawing in other sections of the FSAR was accurate and that Figure 5.3-1 drawings reflected general arrangements only. Since that time the licensee has revised Valve Arrangement No. 23 of Figure 5.3-1 to show the relief valve in the return lines. The inspector reviewed the revised Figure 5.3-1 and found it acceptable. This item is now closed.

8.0 Exit Meeting

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 exit meeting on August 5, 1988. Attendees at the exit meeting are listed in Section 1.0 of this report. At no time during the inspection was written material provided to the licensee by the inspectors. The licensee did not indicate that the inspection involved any proprietary information.