U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 89-07
89-06

Docket No. 50-272
50-311

License No. DPR-70
DPR-75

Licensee: Public Service Electric and Gas Company
P. O. Box 236
Hancocks Bridge, New Jersey 08038

Facility Name: Salem Units No. 1 & 2

Inspection At: Hancocks Bridge, NJ

Inspection Conducted: April 17-21, 1989

Inspectors: P. K. Eapen for J. M. Trapp, Reactor Engineer, DRS, EB — 5/12/89 date

Astrid E. Lopez — A. E. Lopez, Reactor Engineer, DRS, EB — 5/11/89 date

Approved by: P. K. Eapen — P. K. Eapen, Chief, Special Test Programs Section, DRS, EB — 5/12/89 date

Inspection Summary: Routine Unannounced Inspection on April 17-21, 1989 (Inspection Report Nos. 50-272/89-07 (Unit 1), 50-311/89-06 (Unit 2)

Areas Inspected: Review of licensee actions in response to the "expeditious enhancements" described in Generic Letter No 88-17, "Loss of Decay Heat Removal." The inspection reviewed supporting instrumentation, training, procedures and staff awareness as related to mid-loop operation.

Results: The inspectors found that all "expeditious enhancements" described in Generic Letter 88-17 were implemented at Salem prior to drain down to mid-loop operation. The inspectors found the management involvement, training, and staff awareness to problems related to mid-loop operation to be highly effective. Procedures, instrumentation, and systems required to support mid-loop operation were consistent with the licensees response to Generic Letter 88-17. No unresolved items or violations were identified during this inspection.

DETAILS

1.0 Persons Contacted

1.1 Public Service Electric and Gas Company

* R. Dulee, Quality Assurance Principal Engineer
J. Gueller, Operations Manager
* C. Lashkari, Senior Staff Engineer
J. Lloyd, Principal Training Supervisor
* L. Miller, Station Operation Manager
M. Reese, Nuclear Training Coordinator
* G. Roggio, Station Licensing Engineer
T. Worrell, Quality Assurance Lead Engineer

1.2 U. S. Nuclear Regulatory Commission

* P. K. Eapen, Section Chief, Special Programs Section
K. Gibson, Sr. Resident, Salem
* A. Lopez, Reactor Engineer
* J. Trapp, Reactor Engineer

*Denotes presence at exit meeting held on April 21, 1989.

2.0 Review of Licensee Actions in Response to Generic Letter (GL) No. 88-17, Loss of Decay Heat Removal (2515/101)

Loss of decay heat removal (DHR) during non-power operation and the consequences of such a loss have been of increasing concern to the NRC. Many events of loss of DHR have occurred while the reactor coolant system has been drained down for mid-loop activities such as steam generator inspection or repair of reactor coolant pumps. The possibility exists that two fission product barriers could be breached while these activities are in progress, since the reactor coolant system and containment will both be open.

GL 87-12, "Loss of Residual Heat Removal (RHR) while the Reactor Coolant System (RCS) is partially filled" was issued to all licensees of operating PWR's and holders of construction permits on July 9, 1987. Responses indicated that the licensees did not understand the identified problems, and the problem continued as evidenced by events at Waterford on May 12, 1988 and Sequoyah on May 23, 1988.

The seriousness and continuation of this problem has resulted in the issuance of GL 88-17. In addition, the Director of NRR has written to the CEO of each licensee operating a PWR, in which he said, "We consider this issue to be of high priority and request that you assure that your organization addresses it accordingly." He also wrote to each licensed operator at all PWR plants on "Operator Diligence while in Shutdown Conditions," and enclosed a copy of Generic Letter 88-17.

GL 88-17 requires the recipients to respond with two plans of actions:

a. A short-term program entitled "Expeditious Actions," and
b. A long-term program entitled "Programmed Enhancements."

This inspection addressed the short-term licensee actions as outlined under "Expeditious Actions," of GL 88-17.

The inspectors reviewed the licensee response dated January 6, 1989 to Generic Letter 88-17. The licensee response provided a detailed description of action taken to address the eight recommended expeditious actions identified in the Generic Letter. The inspectors verified that the licensee actions are consistent with the NRC guidance provided in Generic Letter 88-17.

The NRC reviewed the licensee's mid-loop operations in 1987 as detailed in the NRC inspection reports 50-272/87-28 and 50-311/87-30. For the details of the NRC review of the licensee's preparations and conduct of mid-loop operation during the current Unit 1 refueling outage (No. 8) see NRC inspection report 50-272/89-03.

2.1 Temperature Indication

The inspectors verified that for mid-loop conditions, the licensee has taken adequate administrative and procedural steps to provide at least two independent, continuous coolant temperature indicators that are representative of the core exit conditions. The licensee monitors the core exit temperature using fifty-eight bottom mounted thermocouples. Step 2.11.2 of Procedure II-1.3.6 "Draining of the Reactor Coolant System," requires as an initial condition that at least two bottom mounted temperature indicators are providing RCS temperature indication in the control room or control room racks. Steps 5.1.1(f)&(g) of Procedure II-1.3.6 requires the thermocouples be displayed in the control room on the plant process computer, or if local indication is used constant surveillance and communication with the control room be established. The procedure requires that a high temperature alarm be operational with a set point of 200°F. This alarm is initiated by the plant process computer and provides an audible alarm in the control room, in addition to a print out on the computer alarm printer. The inspectors found the temperature indication system to be consistent with the expeditious actions of Generic Letter 88-17. The inspectors had no further questions concerning the core exit temperature monitoring system.

2.2 RCS Water Level Indication

The inspectors verified that the licensee has procedures and administrative controls to provide at least two independent, continuous RCS water level indications whenever the RCS is in a reduced inventory condition. The licensee uses two RCS flow transmitters (F0441A, F0400A), recalibrated to indicate RCS level, to provide indication of RCS level in the control room. Each transmitter provides indication in the control room with an alarm set at 97'-6". The alarm is audible and lights an overhead annunciator in the control room. In addition to the two level indicators in the control room, a tygon tube is connected to the No. 13 intermediate leg and provides local indication in the containment. The licensee requires level indications with an operable low level alarm in the control room in Procedure II-1.3.6 Steps 2.11.3 and 2.11.5. Appendix 1 of Procedure II-1.3.6 provides detailed guidance on how to install the tygon tube so that it will provide an accurate level indication. The inspectors verified that the tygon tube level indication was installed in accordance with this procedure. The inspectors noticed that the licensee had provided accurate elevation marks so that the temporary level indicating scale for the tygon tubing could be properly placed. The inspectors verified that the scales for the RCS flow indications and alarm description on the overhead annunciators had been changed. Procedures were reviewed to assure that precautions were provided for possible variations in indicated RCS level.

The level indicating system was found to be consistent with the expeditious actions described in Generic Letter 88-17. The inspectors had no further questions concerning RCS level indication during mid-loop operation.

2.3 RCS Inventory Control

The inspectors verified that the licensee has procedures and administrative controls to provide at least two available or operable means of adding inventory to the RCS, in addition to pumps that are a part of the normal DHR systems. One source of inventory makeup is from the charging pumps. Technical Specification 3/4.1.2 requires one charging pump to be operable in modes 5 and 6. The second independent source of makeup comes from either of the two safety injection pumps. Step 2.11.7 of Procedure II-1.3.6 requires one charging pump be operable and one safety injection pump to be available. A hot leg injection path for the safety injection pump is also required to be available by this step. The inspectors found this to be consistent with expeditious action described in Generic Letter and had no further questions concerning this issue.

2.4 RCS Perturbations

The inspectors verified that the licensee has implemented procedures and administrative controls to avoid operations that could perturb the RCS. Step 5.1.1(h) of Procedure II-1.3.6 requires work activities which could effect RCS inventory be minimized during mid-loop operation. During each shift the Containment Coordinator and the Senior/Nuclear Shift Supervisor meet and discuss work activities which could perturb RCS inventory. The content of these meetings is documented.

In addition, during each shift the Nuclear Shift Supervisor will review shift work activities for impact on RCS inventory. The licensee issued a supervisory letter SL-37 "Salem Primary Systems Loss of Decay Heat Removal" to all supervisory personnel. Attached to this letter were posters which describe "Do" and "Do Not" activities which should be performed to prevent perturbations during mid-loop operation. These were displayed thoughout the plant. The inspectors found that the licensee had taken positive actions to avoid perturbations of the RCS during mid-loop operation. The procedures and controls were found to be consistent with the applicable requirements described in Generic Letter 88-17. The inspectors had no further questions concerning this issue.

2.5 Hot Leg Flow Paths

The inspectors verified that the licensee has implemented procedures and administrative controls to assure that all hot legs are not blocked simultaneously by nozzle dams unless a vent path is provided to prevent pressurization of the upper plenum of the reactor vessel. The licensee uses steam generator nozzle dams during mid-loop operation. The installation procedures require the cold leg nozzle dams to be installed prior to the hot leg dams and the hot leg dams to be removed prior to the cold leg dams. This method of nozzle dam installation reduces the effects of upper reactor vessel plenum pressurization.

Prior to draining the reactor vessel to mid-loop operation, Procedure II-1.3.6 Step 5.1.1(b) requires the removal of all three pressurizer safety valves. This provides a vent area of approximately 0.5 square foot. The licensee has calculated that the back pressure in the reactor vessel with three safety valves removed, at 72 hours after shutdown, would be approximately 3.1 psig. The licensee stated that the 3.1 psig back pressure was acceptable as the existing cold leg openings were sufficiently high to avoid spilling of the reactor coolant under this back pressure. The licensee stated that cold leg openings for maintenance during mid-loop operation would be considered on a case by case basis, and the required vent area for each opening of the cold legs would be recalculated. The reactor vessel back pressure is important to determine the amount of water which could spill out of a cold leg opening. This would also affect the time to core uncovery. The licensee stated at the exit meeting that they would reassess the back pressure issue as part of the long-term corrective action plan. The inspectors determined that this was acceptable based on the fact that the licensee is aware of the importance of the upper plenum back pressure and does considered cold leg openings on a case by case basis. The inspectors had no further questions concerning this issue.

2.6 Loop Stop Valves

Loop stop valves are not part of the Salem Unit No. 1 or 2 system design.

2.7 Containment Closure

The inspectors verified that the licensee has prepared procedures and administrative controls to assure containment closure prior to core uncovery during a loss of DHR event. Procedure II-1.3.6 Step 2.11.1, requires that the equipment hatch be installed prior to decreasing reactor vessel level more than three feet below the reactor vessel flange. Other penetrations in the containment are isolated using Abnormal Operating Procedure AOP-CONT-2 "Containment Closure on Loss of RHR." This procedure is entered when RHR is lost and the reactor vessel level can not be restored. In most situations, the containment could be isolated from the control room using the manual phase-A initiation. The inspectors concluded that the containment building could be isolated prior to core uncovery. The inspectors had no further questions concerning containment closure.

2.8 Training

The inspectors verified that training conducted by the licensee, made the licensee personnel aware of the risks associated with mid-loop operation. The inspectors verified the effectiveness of this training by interviewing operating personnel and reviewing the training material provided to the trainees. The operators interviewed were found to have an indepth knowledge of the issues discussed in Generic Letter 88-17. All operating personnel were trained just prior to draining the reactor vessel to mid-loop.

The training material and lesson plans reviewed by the inspectors did not contain all of the material committed to be part of training in the licensee response to Generic Letter 88-17. The inspectors discussed the topics missing from the lesson plans with the licensed operators. It became apparent to the inspectors that all the topics committed to be taught in the response to the Generic Letter had in fact been presented during the training sessions. The inspectors discussed the weakness in the lesson plan and training material with the Plant Operations Manager who stated at the exit meeting that the lesson plans would be improved prior to the next use of this course material. The inspectors had no further questions concerning this issue.

2.9 QA/QC Involvement

The QA organization was found to have extensive involvement in the mid-loop operation issue. The inspectors reviewed licensee surveillance report 88-0633 which was a surveillance conducted during the last drain down operation at Salem Unit 2. The surveillance results were satisfactory. QA had also performed indepth root cause analysis of industry events relating to loss of RHR and studied these events for applicability at Salem. The inspectors interviewed QA personnel involved in mid-loop operation concerns and found that they had full knowledge of industry events, and plant specific details relative to mid-loop concerns.

2.10 Summary

The inspectors found the licensee management and staff to be aware of the types of problems which may occur during mid-loop operation. The actions taken by the licensee in installation of instrumentation, training, staff awareness and management involvement were found to be highly effective. All the expeditious actions addressed in the licensee response to Generic Letter 88-17, dated January 6, 1989, were implemented during mid-loop operation. The licensee was found to be pursuing additional improvement, for mid-loop operation, as part of a long-term corrective action program.

3.0 Exit Meeting

At the conclusion of the site inspection, on April 21, 1989, an exit interview was conducted with the licensee's senior site representatives (denoted in Section 1) to discuss the results and conclusions of this inspection.

At no time during this inspection was written material provided to the licensee by the inspector. Based on the NRC Region I review of this report and discussions held with licensee representatives during this inspection, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.