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REGION I

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LICENSEE:	PECO Energy Nuclear Group Headquarters Correspondence Control Desk P. O. Box 195 Wayne, PA 19087-0195
FACILITY:	Limerick Units 1 & 2
INSPECTION DATES:	January 3-10, 1994

INSPECTOR: Larry L. Scholl 2/3/94

Larry L. Scholl, Reactor Engineer
Electrical Section, EB, DRS
Date

APPROVED BY: William H. Ruland 2/24/94

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Date

Areas Inspected: The hardware modification installed to ensure the reactor vessel level instrumentation design is of high functional reliability for long term operation, as requested in NRC Bulletin 93-03, was reviewed. The modification provides a continuous flow of water from the control rod drive (CRD) system to the reactor vessel water level instrumentation reference legs. This flow is designed to prevent the buildup of noncondensible gases in the reference leg water that could reduce the reliability of the vessel level instrumentation during a plant depressurization. The modification design, installation, testing and associated procedures were reviewed. The vulnerability of the Limerick plant design to problems identified at other facilities, with similar modifications, was also reviewed. These problems are discussed in NRC Information Notice 93-89, "Potential Problems With BWR Level Instrumentation Backfill Modifications."

An unresolved item concerning oscillations of reactor vessel water level instruments during certain plant transients, and resultant spurious initiations of the high pressure coolant injection (HPCI) system, was also reviewed.

Results: The modification installed in Unit 1, and planned for Unit 2, was well designed and incorporated lessons learned during a similar modification installed at PECO's Peach Bottom facility. The modification was adequately tested and system operating procedures are in place. The inspector found that allowable outage times for the system have not been defined. PECO intends to evaluate each system outage on a case-by-case basis. At the conclusion of the inspection PECO agreed to establish an appropriate time in which an evaluation will be performed such that continued water level instrumentation reliability is assured. PECo has performed high cycle fatigue calculations and determined the thermal stress on the affected piping and reactor vessel nozzles are acceptable. Low cycle fatigue calculations have not yet been performed. The inspector also found that due to differences in design, the plant transients that could result from inadvertent valve operations would not be as severe at Limerick as for those plants described in NRC Information Notice 93-89.

PECO plans to include dampening circuits in the water level instrumentation to prevent the spurious actuations of the HPCI system during transients. The inspector found these actions to be appropriate and this item was closed.

DETAILS

1.0 BACKGROUND

On May 28, 1993, NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," was issued to address a concern that noncondensible gases may become dissolved in the reference legs of boiling water reactor (BWR) vessel water level instrumentation. These gases could result in false high level indication after a plant depressurization event. In the bulletin, the NRC requested the licensees implement short term compensatory actions to ensure that potential level errors, caused by reference leg de-gassing, would not result in improper system response or improper operator actions during transients and accident scenarios initiated from reduced pressure conditions. These short term measures augmented previous short term compensatory measures, requested in NRC Generic Letter 92-04, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," to mitigate the consequences of potential level indication errors after a rapid depressurization event during power operation. This generic letter also requested that licensees provide the NRC staff with plans for long term corrective actions, including any proposed hardware modifications. The licensees, through the BWR Owners Group, requested a delay in the implementation of long term corrective actions until a de-gas test program could be completed. The NRC staff agreed to extending the time for submission of licensee plans for long term corrective actions, and subsequently, in NRC Bulletin 93-03, requested that licensees implement hardware modifications at the next cold shutdown beginning after July 30, 1993. PECO responded to NRC Bulletin 93-03 by letter dated, July 30, 1993, and agreed to implement hardware modifications during the next cold shutdown on each unit.

2.0 MODIFICATION DESCRIPTION

Four reference legs provide a constant reference pressure to the reactor vessel level instrumentation. The piping for each reference leg is maintained full by condensed steam provided from a condensing pot located at the top of the reference leg. This design has been found to be potentially susceptible to the accumulation of noncondensible gases that may result in the elimination of steam flow to the condensing pot, thus resulting in the loss of makeup water to the reference leg. Another potential effect is that the water in the reference legs may absorb significant amounts of dissolved gases. The dissolved gases may then come out of solution during a depressurization of the reactor vessel and introduce gas voids in the piping and, during rapid depressurizations, may expel significant amounts of the reference leg water inventory. The loss of inventory or the introduction of voids can result in false high level indications.

Limerick Design Modification P-00132 provides continuous makeup water to the reference legs, ensuring they remain full, and also prevents the accumulation of significant amounts of dissolved gases in the reference leg water. The source of the backfill water is the control rod drive (CRD) system that is maintained at approximately 1500 psig during normal plant operations. The water from the CRD is filtered to prevent the introduction of particulate in the piping that could interfere with the operation of system components. Fine control needle valves are used to maintain the flow rate to each reference leg to between 2 and 6 lbm/hr - approximately 0.25 to 0.7 gallons per hour (gph). A flow indicator is installed in each of the four individual supplies to the reference legs. A total system flow indication is also available. The backfill water enters the reference leg at the connection point where the instrumentation tubing connects to the containment isolation valve. Just prior to the connection point at the containment isolation valve, two check valves have been added to provide a boundary between the nonsafety CRD system piping and the safety-related reactor vessel instrumentation system.

3.0 MODIFICATION REVIEW

Modification Design

The design of the modification is based on testing sponsored by the BWR Owners Group (BWROG) that was performed to determine the effects of noncondensible gases dissolved in reference leg piping, and to evaluate the effectiveness of continuous backfill systems. The modification has been installed and is in operation on Unit 1. The Unit 2 system has been installed with the exception of the final connection to the CRD system and the reference leg piping. These connections will be made the next time the plant is in a cold shutdown condition. The system will then be tested and placed in service prior to the next Unit 2 startup.

Backfill Flow Rate

The BWR owners group testing determined that a continuous backfill flow of approximately 0.25 gph is sufficient to maintain the noncondensible gases in the reference leg water at a low enough concentration to prevent any significant level instrumentation errors during plant depressurization events. PECO utilized this information to specify the desired backfill flow rates for Limerick Units 1 and 2.

Backfill Water Source

The control rod drive system was chosen as a source for the backfill system for the following reasons:

- it is a source of clean, reactor grade water that is low in dissolved gases;

- the discharge pressure of the CRD pumps is sufficiently high to permit injection flow into the reactor vessel at normal and transient condition reactor vessel pressures; and

- the CRD system operation is essential for keeping the hydraulic control unit scram accumulators pressurized during plant operation and as a result it provides a highly reliable source of water for the backfill system. Also, if the system is lost a plant shutdown would be required by plant procedures, due to decreasing scram accumulator pressure, before noncondensible gas buildup in the reference leg water would be significant.

System Flow Path

Though the CRD system water is relatively clean, the backfill system modification includes two micron filters immediately downstream of the point where the backfill system connects to the CRD system. These filters reduce the possibility that particulates will affect the operation of the system components that have close tolerances, such as the flow control needle valves and check valve seating surfaces. After passing through the filters, the backfill flow goes through a total system flow indicator and then divides into four branches, one for each reference leg. Each branch has two throttle control valves and a flow indicator to permit accurate adjustment and monitoring of individual reference leg flow rates. Bypass piping around the throttle control valves includes two manual valves. By having two valves in series, the failure of, or inadvertent opening of one valve would not affect system operation. The branch leg piping then connects to its respective reference leg piping immediately outside of the primary containment at the excess flow check valve. This connection point eliminates any impact on the existing containment isolation boundary. Two in-series safety grade check valves have been added to establish a boundary between the safety grade (instrumentation) piping and the non-safety grade (backfill system) piping. These valves are intended to prevent a failure in the backfill system piping from affecting the operability of the reactor vessel level instrumentation.

Thermal Stress Effects

Since the temperature of the backfill system water is significantly lower that of the reactor vessel, the effects of thermal stresses on the reactor vessel nozzles, condensing chamber inlet steam piping and condensing chamber were required to be analyzed. PECO reviewed their plant configuration against a thermal stress resolution plan developed by General Electric (GE).

The GE analysis determined that high cycle fatigue of inlet steam piping and condensing chamber are not a concern for flow rates less than approximately 45 lb/hr and that high cycle fatigue of the nozzles is not a concern for low flow rates as long as the differential temperature between the backfill water and the nozzle is less than 50°F. GE also determined that low cycle fatigue of the inlet steam piping and the condensing chambers is possible and, therefore, plant specific calculations are necessary to confirm that the thermal stresses associated with low flow rates will not be of concern for the life of the plant. Failure due to low cycle fatigue would take several operating cycles and, therefore, interim operation was determined to be acceptable during the period when plant specific calculations are being performed.

PECO calculations determined that the backfill water temperature would be approximately 29°F less than the nozzle temperature. Based on this calculation and the fact that the normal maximum flow rate is approximately 6 lb/hr, PECO concluded that high cycle fatigue of the piping, condensing chambers and nozzles is not a concern. Limerick plant specific calculations to evaluate low cycle fatigue have not yet been performed. The need to complete these calculations is being tracked by PECO in their action item tracking system. NRC review of the completed calculations may be included within the scope of future inspections.

System Testing

The inspector reviewed testing performed on the system in accordance with Modification Acceptance Test MAT P-00132, "Reactor Vessel Level Reference Leg Backfill Modification Acceptance Test."

Pre-installation calibration of the system flow indicators and check valve leak tests were performed. The system was then filled, flushed and hydrostatically tested. An additional leak test was performed on the system check valves after installation in the system. While monitoring the response of the reactor vessel level instruments, the system was placed in service and the individual reference leg flow rates adjusted to approximately 2 gal/hr. The following testing was then performed, with the plant shut down and depressurized, while continuing to monitor the water level instrumentation response:

- Each train was valved out of service and then returned to service in accordance with the System Operating Procedure IC-11-00505, "RPV Instrumentation Reference Leg Backfill System Operation - Units 1 and 2";

- The operating CRD pump was swapped with the standby pump in accordance with Procedure S46.6.A, "Placing Alternate Control Rod Drive Pump in Service"; and

- The operating CRD pump was secured in accordance with Procedure S46.2.A, "Shutdown of Control Rod Drive Hydraulic System."

During the system startup and testing, no significant effects were noted on the water level instrumentation. When restarting a CRD pump, a level transient of approximately one inch was noted on one of the narrow range level instruments. The other system operations and tests resulted in less noticeable effects on the indicated level.

The effects of a reactor scram on the instrumentation, with the backfill system in operation, was not tested. After a scram, reference leg backfill flow would be decreased, or stopped, during the time the CRD system is recharging the scram accumulators. PECO determined that the testing that involved securing the CRD pump would bound the transient resulting from a reactor scram.

During the plant startup and power ascension, the level instrumentation was again monitored during control rod motion, both single notch and continuous motion, and when valving the system in and out of service in accordance with Procedure IC-11-00505.

The effect of vessel pressurization on the backfill flow rates was monitored and upon achieving rated reactor vessel pressure the flow rates were adjusted to 0.4 to 0.6 lb/hr. At normal reactor plant pressure the backfill system was inspected to ensure there were no system piping leaks. These operations and tests performed during plant startup did not result in any significant effects on the level indication.

The inspector also reviewed periodic testing of the 'A' train check valves performed in accordance with Procedure ST-2-036-490-1, "Division 1 RPV Instrumentation 'A' Reference Leg Backfill System Check Valves (42-1046A, 42-1044A) Test." This test performs a periodic leak test and exercise test of the valves. Since the leak tightness of the check valve may be affected by the differential pressure across the seat, the test is written to perform a seat tightness check at a low pressure (250 to 305 inches of water) and at high pressure (1220 to 1280 psig). The acceptance criteria for the maximum allowable seat leakage is 2 ml/hr which is intended to ensure essentially zero leakage. Test Procedures ST-2-036-491, -492, -493-1 perform the identical testing on the valves in the other three trains.

The inspector concluded that the system testing effectively demonstrated system operability.

System Operation

The plant startup and shutdown procedures specify when the system is to be placed in and removed from service. All valve manipulations are specified in Procedure IC-11-00505 and are performed by the instrumentation and controls department technicians. The inspector reviewed this procedure and found that it provided detailed instructions and precautions for operation of the system. When returning the system to service, the flow rate to each reference leg is verified to be correct and the as-left flows are recorded in the procedures. This procedure step provides a positive action that ensures the system has been properly aligned. When in operation, the only required action is to monitor the system flow rates. The inspector reviewed the auxiliary plant operator rounds sheet for the reactor building and confirmed that the monitoring of the backfill system flows was included as an item to be verified at least once per day.

The inspector noted that the plant procedures did not specify an allowable out of service time for the backfill system. Without this guidance, plant operators cannot determine how long the system may be out of service before the reliability of the reactor vessel level instrumentation may be affected. PECO engineers stated that the system is highly reliable and out-of-service periods should be minimal, and at the present time, the effects of the loss of a portion or entire system will be evaluated on a case-by-case basis. PECO is continuing to work with the BWROG on resolution of this issue and agreed that additional procedural guidance will be provided so that, as a minimum, the time allowed to perform an evaluation of a system outage will be specified.

Limerick Procedure T-210, "Injection From the Standby Liquid Control Storage Tank with the CRD System," provides a backup method of injecting borated water into the reactor vessel utilizing the CRD system. Section 4.1.3 of the procedure has been revised to isolate the backfill system prior to initiating boron addition with the CRD system to prevent the introduction of boron into the instrumentation piping.

Conclusion

The water level instrumentation reference leg backfill modification has been well designed, installed and tested on Unit 1. Unit 2 installation is complete and ready for final tie in and testing during the next time the plant is in a cold shutdown condition. The system operating procedures are in place and contain adequate directions to support system operation. Final thermal stress calculations and the establishment of guidance on allowable out of service time for the system remain to be performed.

4.0 NRC INFORMATION NOTICE 93-89: POTENTIAL PROBLEMS WITH BWR LEVEL INSTRUMENTATION BACKFILL MODIFICATIONS

Background

At the Susquehanna nuclear power plant, the licensee determined that severe plant transients could result if a plant operator inadvertently shut an isolation valve in the reference leg piping. Closing a manual isolation valve could cause the reference leg to become pressurized to CRD system pressure via the backfill system. The transient resulting from pressurizing the reference leg would result in erroneous indication of low reactor water level and high reactor pressure. This could result in a reactor scram and the opening of all safety relief valves (SRVs) depending on which leg was pressurized. Loss of reactor vessel inventory and decreasing pressure, caused by the open SRVs, could eventually result in the initiation of the emergency core cooling systems. Further, if a single failure in the ECCS low pressure permissive logic is postulated to occur coincident with this transient, all low pressure ECCS could be disabled. This event was also reviewed for the LaSalle and Quad Cities nuclear power plants where similar conclusions were reached regarding the potential severe plant transients.

Applicability to Limerick

At the time of the inspection PECO was in the process of reviewing this information notice to determine its applicability to the Limerick plant design. During interviews with PECO engineers and reviews of plant instrumentation and control diagrams, the inspector found the following:

- The only valves that could result in the pressurization of a reference leg are normally locked open in accordance with plant procedures and are not operated by any plant procedures during power operation;

- The SRVs at Limerick do not receive an open signal from the plant pressure instruments and would not open upon pressurization of the reference leg. (The valves open on an actual high plant pressure condition when the valve inlet pressure is sufficiently high to operate the pilot valve.);

- The SRVs could open as the result of the completion of the automatic depressurization system (ADS) logic; however, system time delays would allow the operators several minutes to diagnose the event and then inhibit the initiation of the ADS. Also, in addition to being locked, the valves associated with the reference legs whose instruments could initiate ADS are located in radiation areas that are not normally readily accessible;

- The ECCS permissive logic for the low pressure coolant injection system does not receive an input from the pressure instruments connected to the reference legs. The injection valve permissive logic is based on the differential pressure across the injection valve, and utilizes instruments that are independent of the reactor vessel pressure instrumentation; and

- A single failure in the core spray system logic, coincident with a reference leg pressurization, could result in the loss of both loops of the core spray system. However, the four low pressure coolant injection (LPCI) trains of ECCS would remain available to provide sufficient low pressure injection. Also, in addition to the valve being locked, the specific valve that would have to be inadvertently closed for this event is located in a locked high radiation area.

While these findings indicate that Limerick is not susceptible to all of the potential operational events discussed in IN 93-89, PECo is reviewing the information to ensure that any related condition does not constitute an unreviewed safety question. NRC review of the PECo evaluation may be included in future inspections.

5.0 UNRESOLVED ITEM 50-353/93-07-01 - HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) INITIATIONS

On March 26, 1993, a reactor scram and main turbine trip occurred due to an inadvertent closure of the main turbine stop valves. The associated pressure transient resulted in momentary spiking of reactor vessel level instruments that caused spurious actuations of the high pressure coolant injection system. The level spikes were of sufficient duration to cause the HPCI turbine to start. They were not of sufficient duration to allow the injection valve opening logic to go to completion and, therefore, no injection to the vessel occurred. Instrument spiking has been observed previously on both Limerick Units 1 and 2 during testing and following operational transients. GE Safety Information Letter (SIL) No. 463 also discusses the potential operational problems that may result from process instrumentation noise. These problems have been experienced at plants that utilize fast response transmitters, such as the Rosemount transmitters installed at Limerick, that do not have adjustable electronic filtering installed.

PECO reviewed this event and determined that, although the HPCI system remained operable during the transient, dampening circuits would be utilized on the reactor vessel level transmitters to minimize the challenges to the HPCI system. The dampening circuit boards are already installed in the Unit 1 transmitters and will be adjusted to provide filtering of the input signal during the next refueling outage. Dampening circuits will be installed in Unit 2 during the next outage. Time response testing of the instrument channels will be performed following the implementation of the dampening circuits. The inspector found these actions to be consistent with the recommendations contained in SIL-463. This item is closed.

6.0 EXIT MEETING

At the conclusion of the inspection on January 10, 1994, the inspector met with PECO representatives denoted in Attachment 1. The inspector summarized the scope of the inspection findings at that time. The facility licensee representatives acknowledged the NRC inspector findings.

ATTACHMENT 1

Persons Contacted

PECO Energy Company

* T. Bell, Plant Engineering
  P. Driehaus, Operations Support
* J. Muntz, Plant Engineering
* G. Stewart, PECO Experience Assessment
* R. Weingard, Plant Engineering
  R. Braun, Shift Manager

U. S. Nuclear Regulatory Commission

* T. Easlick, Resident Inspector
* N. Perry, Senior Resident Inspector
* A. Wilford, NRR Intern Program

* Denotes those present at exit meeting.