10CFR50.4(a)
10CFR50.54(f)
GL 92-04

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Generic Letter 92-04: Resolution of
the Issues Related to Reactor Vessel Water
Level Instrumentation in BWRs Pursuant to 10CFR50.54(f)

The attachment to this letter provides the Boston Edison Company response to Generic Letter 92-04 (GL 92-04) dated August 19, 1992. GL 92-04 addresses various issues related to and corrective actions for Boiling Water Reactor water level instrumentation under transient conditions of reactor pressure reduction.

This response reflects Pilgrim-specific analysis and actions as well as information developed by our participation in the BWR Owner's Group effort to address this topic.

E T Boulette for
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PMK/clc/gl9204

Attachment

cc: See Page 2

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NRC Request #1

In light of potential errors resulting from the effects of noncondensible gas, each licensee should determine:

a. The impact of potential level indication errors on automatic safety system response during all licensing basis transients and accidents;

b. The impact of potential level indication errors on operator's short and long term actions during and after all licensing basis accidents and transients;

c. The impact of potential level indication errors on operator actions prescribed in emergency operating procedures or other affected procedures not covered in (b).

Response

This evaluation for Pilgrim focuses on the ability of the reactor vessel water level instrumentation to initiate required automatic protective actions in response to a postulated water level decrease coincident with the onset of reference leg voiding. Each specific accident/transient crediting low or high reactor water level with a protective action initiating signal is discussed. Operator responses to such conditions are also addressed.

- Safety Assessment

Conservative estimates of water level effects induced by noncondensibles indicate that only small level errors are postulated above 600 psig. Although GE estimates that potential errors above 470 psig would be small, 600 psig is used in this assessment for conservatism. During slow depressurizations, Pilgrim has experienced no level fluctuations above 470 psig. Events for which automatic safety system actuations occur above 600 psig are, therefore, considered unaffected by this postulated concern.

- Postulated Abnormal Operational Transients for Pilgrim

Low Level Scram and Isolation Functions (+9")

The only transient event where a low level (+9") scram is credited for performing the scram function is the total loss of feedwater flow. Other transients either do not result in a low level or receive scrams from other initiators. For a loss of feedwater flow event at full power, reactor pressure is well above 600 psig and the low level scram function is unaffected (Pilgrim FSAR Appendix R.2.4.3). For other reduced power operation conditions where a loss of feedwater flow occurs, the reactor will scram when and if reactor pressure drops below approximately 880 psig (in RUN mode) due to Main Steam Isolation Valve (MSIV) closure. Otherwise, since the event is above 600 psig, no level errors will occur.

If the event occurs with the reactor in the STARTUP mode, feedwater flow will initially be low and a loss of feedwater will not cause a significant reactor pressure drop (i.e., below 600 psig) before scram level is reached. Water level scram response would, therefore, be unaffected.

For plant Startup operation below 600 psig reactor power and feedwater flow will be very small or zero. A loss of feedwater in this condition would be a very mild transient and does not lead to reactor water low level prior to operator intervention.

Core Standby Cooling Systems (CSCS), Containment Isolation, Reactor Core Isolation Cooling (RCIC), Diesel Generator Functions at Low-Low Level (-49")

Loss of feedwater, loss of offsite power, and pressure regulator failure events may result in a low-low level with initial full reactor power. For the loss of feedwater event in RUN mode, reactor pressure will remain above 880 psig due to MSIV closure. For the loss of offsite power event, MSIVs close on loss of power to Primary Containment Isolation System (PCIS) logic. Diesel generators start when the unit auxiliary and startup transformer breakers trip. For the pressure regulator failure event, MSIVs close due to low reactor pressure (less than 880 psig in RUN mode). These events then become reactor isolation events at high pressure. If reactor level reaches the low-low level CSCS initiation point (-49"), the reactor will be above 600 psig and level error would not exist.

If the reactor is operating at low power in STARTUP mode during a pressure regulator failure, a high water level (+48") MSIV closure would occur returning the reactor to high pressure (Pilgrim FSAR 7.3). This level swell will occur rapidly due to coolant voiding before significant vessel depressurization has occurred. Any false high water level indications would only cause the isolation to occur sooner. Therefore, a pressure regulator failure in STARTUP mode will not result in coincident low reactor pressure and low-low level. If the reactor is operating at low power in STARTUP mode during a loss of feedwater event, the transient will be mild. At low power and low feed flows, steaming rates will be low. Reactor level would drop slowly and the pressure drop would be small (Pilgrim FSAR Appendix R.2.4.3). Level errors would not be expected and plant response would be unaffected.

With plant conditions initially below 600 psig, these transients would be mild because reactor power and feedwater flow to the reactor would be very small or zero. Considering the low steaming rates when shutdown below 600 psig, substantial level exists above the top of active fuel. These events would not represent rapid depressurizations and level errors and are, therefore, not a concern.

High Level Isolation Functions (+48")

Transient events that could lead to high water level conditions are not affected since the level errors are in the high direction and will only cause the required functions to occur sooner. Premature tripping of the HPCI/RCIC systems is not a concern because these are high pressure systems only required for events that do not involve rapid, sustained depressurizations.

Pipe Breaks Inside Containment (PBIC) At Full Power

Pipe breaks inside containment at full power conditions (other than the large steam line breaks) result in reactor pressure exceeding 600 psig when low and low-low water levels are reached. As discussed previously, no water level effects will occur above 600 psig.

Large steam line breaks can depressurize to below 600 psig. However, large steam line breaks initially lead to level swells due to extensive voiding. Since response of level sensors to this event cannot provide necessary timeliness, other design features exist to mitigate these events (i.e., high drywell pressure). Containment analysis does not take credit for a low water level scram; it uses high drywell pressure (Pilgrim FSAR 14.5.3.1). Under this condition the MSIVs close on high steam line flow or low steam line pressure. Primary and secondary containment isolations and CSCS and EDG initiation also occur on high drywell pressure. The only actuations that do not occur on high drywell pressure or other designed accident response signals that would otherwise occur on low or low-low level signals are:

- Reactor Water Cleanup (RWCU) isolation
- ADS actuation
- RCIC actuation

The Reactor Water Cleanup (RWCU) System receives isolation signals in response to reactor vessel low water level (+9"). This isolation is provided to ensure the containment is isolated prior to core uncovery. Loss of Coolant Accident (LOCA) analysis of the main steam line break indicates that potential core uncovery does not occur until approximately 95 seconds after the break. Conservatisms associated with this analysis indicate that any core uncovery is unlikely. Original analysis of this event indicated that core uncovery would not occur. (Pilgrim FSAR 5.2.8.3). Delays in this isolation response are of little safety consequence because there is no expected core uncovery, and because the RWCU system represents a significant barrier to gross leakage (i.e., it is a high pressure system not expected to randomly or passively fail as the reactor depressurizes and, it refills with water as the CSCS systems reflood the reactor vessel). Also, RWCU isolation valves remain available to be closed by operator action.

For large steamline breaks, reactor pressure drops so rapidly that the ADS and RCIC functions are unnecessary.

LOCA analyses are not affected because water level responses are not credited in the analyses.

In STARTUP Mode

For breaks that occur in STARTUP mode, the above discussion is applicable. MSIVs close on high steam flow. Also, since startup procedures require level reference leg back flushing, the likelihood for significant noncondensibles buildup is small.

Less than 600 PSIG

For breaks that occur with the plant initially below 600 psig, reactor power will be negligible, vessel blowdown rates will be reduced and the breaks are bounded by the above evaluations. CSCS initiation on high drywell pressure provides substantial coolant makeup for these conditions. Since this is a zero power event, CSCS will reflood the reactor and core uncovery does not occur. If MSIVs do not close on high steam flow or low-low level operators can close the MSIVs manually.

Based on the above discussions and considering the Pilgrim LOCA analyses assume an initial power of 102%, initial pressure of 1050 psig, and reduced Low Pressure Coolant Injection (LPCI) and core spray flow rates (5% and 10%, respectively), considerable margin exists for PBICs to conclude that water level errors will not affect required safety functions.

2/3 Core Coverage

A 2/3 core coverage condition is only expected for a recirculation pipe break. These are large break events with rapid level decreases and core level recovery to 2/3 core height using core spray and/or LPCI pumps. Diversion of LPCI flow to the containment cooling mode requires operator action. LPCI flow will not be diverted for containment cooling unless operators are directed to by EOPs. Operators are trained to recognize potential water level errors by level indicator comparisons and other indications. With indeterminate level indications, EOPs direct operators to flood the reactor vessel, thereby assuring adequate core cooling. EOPs also direct operators to maintain reactor level above +9". This level is more than 11 feet above the active fuel and more than 15 feet above 2/3 core coverage (required to assure adequate core cooling). The likelihood that all level indicators will consistently be in error in excess of these parameters is very remote.

Regulatory Guide 1.97, Post Accident Indication

Reactor water level indication is an important parameter for post-accident operator responses. Operator training regularly emphasizes that multiple water level indications be checked to confirm accurate water level conditions. Discrepancies between water level channel readings would indicate to operators that water level is indeterminate. With different reference leg geometries and gas concentrations, it is extremely unlikely that all level indicators will provide consistent yet inaccurate data in an accident situation. Sufficient core cooling would be assured via the appropriate EOPs.

Pipe Breaks Outside Containment (PBOC)

Analysis of PBOCs indicates reactor pressure remains high because PBOCs are reactor isolation events. These events have peak clad temperatures substantially below PBICs because inventory loss is limited. If pressure is reduced later in the event (i.e., due to ADS actuation), it would be the result of a low-low level condition that existed at high pressure (i.e., above 600 psig). PBOC isolation capability is unaffected by level errors because most PBOCs isolate on high flow or high area temperature. The only PBOC that isolates due to a level signal only is the shutdown cooling line break. This event occurs at initial low pressures and does not represent a rapid depressurization scenario. Once the break is isolated, the need for CSCS initiation is bounded by full power LOCA analyses (i.e., in shutdown cooling only decay heat is being generated, a break no longer exists due to isolation, and the reactor is at low pressure).

Stuck Open Relief Valve (SORV) Events

Inadvertent opening of a relief valve during normal operations has been analyzed in Pilgrim FSAR Appendix R.2.4.2. This event is not a rapid depressurization since the pressure regulator maintains reactor pressure by throttling steam flow to the turbine.

Control Rod Drop Accident

This event leads to MSIV closure on high main steam line radiation. At that point, the event becomes an isolation event similar to loss-of-offsite power and level errors are not expected.

Anticipated Transient Without Scram (ATWS) Events

The recirculation pump trip function supports ATWS by reducing core power and protects the recirculation pump from cavitation due to low water levels. ATWS events involve high reactor pressures because the reactor continues to generate some power. EOPs direct operators to stabilize reactor pressure. In these cases, water level errors are not expected at the resulting pressures.

Fire Events

Fire events are also isolation events where reactor pressure remains high. No water level errors will therefore occur. Automatic safety functions are assumed to be lost during these events. Operators manually perform required actions. Multiple water level indications are available during the ensuing slow cooldown and depressurization.

- Summary

Based on conservative estimates of the effects of the noncondensible gas phenomenon, sufficient margin exists in transient and LOCA analyses for conditions when water level errors are predicted to occur. Water level errors have no effect on limiting Pilgrim FSAR transient and accident analyses because the fluctuations would be minor above 600 psig. In the unlikely event low water level containment isolation actuations have not occurred before reactor pressure drops below 600 psig, core uncovery is not expected and operators will be able to close isolation valves. In the event that operators are unable to determine reactor water level, EOPs provide clear direction to flood the reactor vessel.

NRC Request #2

Based upon the results of (1) above, each licensee should notify the NRC of short term actions taken, such as:

a. Periodic monitoring of level instrumentation system leakage; and,

b. Implementation of procedures and operator training to assure that potential level errors will not result in improper operator actions.

Response

BECo was investigating water level inconsistencies and spiking issues prior to issuance of GL 92-04. Our efforts include:

- We commissioned a consultant to analyze Pilgrim's design and operational data to determine the root cause, assess the significance, and recommend possible corrective actions. The preliminary report was received September 21, 1992.

- We continue to monitor for mismatches between water level instruments by having operators compare readings from different instruments on a routine basis.

- Quantifying/identifying leakage in water level instrument racks.

- We confirmed appropriate operations crew response to various water level scenarios that could result from this phenomenon. Appropriate crew response was demonstrated on the Pilgrim simulator and was observed by NRC personnel.

- We are preparing procedures to backfill the reference legs if water level signatures are experienced during plant shutdown.

- We have prepared a procedure to monitor and collect data during the next plant shutdown presently scheduled for October 24, 1992.

- We may backfill the reference leg if "spiking" occurs in the future.

NRC Request #3

Each licensee should provide its plans and schedule for corrective actions, including any proposed hardware modifications necessary to ensure the level instrumentation system design is of high functional reliability for long term operation. Since this instrumentation plays an important role in plant safety and is required for both normal and accident conditions, the staff recommends that each utility implement its longer term actions to assure a level instrumentation system of high functional reliability at the first opportunity but prior to starting up after the next refueling outage commencing 3 months after the date of this letter.

Response

BECo is participating in the BWROG effort on water level and endorses the plan provided in the BWROG letter to the NRC dated August 12, 1992. We also support the BWROG plan provided in its September 24, 1992 letter to the NRC. Pilgrim is scheduled to start its next refueling outage, RFO #9, in April 1993.

If our assessment of the BWROG program indicates that modifications are necessary to ensure level instrumentation is of high functional reliability, such modification will be scheduled for future implementation. We will keep the NRC advised of our progress on this topic in our submittals every six months of the Long Term Plan.