BOSTON EDISON
Pilgrim Nuclear Power Station
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April 5, 1992
BECo Ltr. 92-37

Roy A. Anderson
Senior Vice President — Nuclear

Mr. Thomas T. Martin
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
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Docket 50-293
License DPR-35

Subject: **Request for Temporary Waiver of Compliance**

Dear Mr. Martin:

Boston Edison requests a temporary waiver of compliance for 14 days from applicable portions of the Pilgrim Nuclear Power Station (PNPS) Technical Specifications (T.S.) associated with Reactor Vessel Water Level Instrumentation. Applicable T.S. sections include Section 3.1, "Reactor Protection System", and 3.2, "Protective Instrumentation", as it affects the automatic initiation of certain systems. The requested waiver would permit plant restart and low power operation to verify the effectiveness of corrective actions to resolve a Reactor Vessel Water Level spiking phenomenon occurring under certain low pressure plant conditions. Power will be maintained at 25 percent or less. Reactor operation would not create a significant hazard to public health and safety. The basis for this conclusion is discussed in Attachment #1 which provides a detailed evaluation of the operational impact of the reactor water level spiking phenomenon. The evaluation contained in Attachment #1 and this waiver request were reviewed by the plant Operations Review Committee and approved by the Plant Manager.

Discussion

False high reactor water level spiking has been noted previously during plant shutdowns. LER 50-293/91-08-01 discusses three Group 1 main steam line isolations during plant shutdown on April 30, 1991. Root cause determination identified the head equalizing line connecting the condensing pot to the reactor vessel as being undersized. A plant modification completed during Refueling Outage Number 8 increased the size of this line from a 1 inch line to a 2 inch line. Post modification testing was satisfactorily completed by pressurizing the reactor to approximately 400 psig and then depressurizing to demonstrate that the equalizing line provided sufficient capacity to allow flow of condensate return to the reactor vessel. This testing did not identify similar false high reactor water level spiking.

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A shutdown performed on March 26, 1992, resulted in two Group 1 isolations due to false high reactor water level indications. A multidisciplined root cause team investigated possible causes.

Testing and evaluations were conducted and we have determined the most probable cause for the false level indications. We believe the cause to be improper thermal performance of the reference leg condensing pot. We have also identified three other phenomena that hinder our ability to be as definitive as we would like to be on root cause determination. These phenomena include:

- possible trapped air in the sensing line
- Barton reactor water level and/or Narrow range pressure indicators causing perturbations on instrument racks
- and possible non-condensibles in condensing pot.

Trapped air in the "B" instrument rack lines is being evaluated by off-line testing. Prior to this startup, we will remove the insulation from the head equalizing line to correct the improper performance of the reference leg condensing pot. We must start and operate the reactor at low power to establish thermal equilibrium at rated temperature and pressure. This will more closely approximate conditions where spiking occurred on March 26, 1992. Installed temporary instrumentation will allow monitoring of the condensing pot as we perform a controlled shutdown and depressurization. The possible complications associated with the Barton level instrument and the narrow range pressure instrument will be determined by isolating them at a selected time during depressurization.

The collection of non-condensible gases in the condensing pot is also being evaluated although this is the least likely of the possible causes. Temperature measurements at the top and bottom of the condensing pot taken prior to the present shutdown suggest the potential of non-condensible gases slowly collecting in the condensing pot during power operations. A typical measurement of condensing pot temperatures was 350 degrees Fahrenheit at the top and 325 degrees Fahrenheit at the bottom. These temperatures are higher than the ambient drywell temperature demonstrating normal operation of the condensing pot (i.e., steam is flowing into the pot and condensing on the pot walls keeping the pot temperature above ambient). Since the temperatures are much lower than reactor coolant temperature of approximately 440 degrees Fahrenheit, we believe non-condensible gases coexist in the condensing pot with reactor steam. These gases partially insulate the condensing pot walls and may retard the condensing rate of the steam and lower the pot temperature. However, we do not believe these non-condensible gases are a significant contributor to the phenomenon we have experienced with reactor water level spiking.

Justification for Operation

A detailed evaluation of the operational impact of this condition has been performed. The results of the evaluation are provided in Attachment 1.

Significant Hazards Consideration and Environmental Consequences

Boston Edison evaluated operation of the facility with the conditions described in Attachment 1 and concluded operation under these conditions will not involve a significant hazards consideration.

Operating in this condition does not significantly increase the probability or consequences of an accident previously evaluated because delays in initiation of Core Standby Cooling Systems (CSCS) or containment isolation equipment will not affect the ability of these systems to perform their safety functions when false high reactor water level spiking is predicted to occur.

This condition does not create the possibility of a new or different kind of accident from any accident previously evaluated because the plant responds as designed to an indicated high reactor level indication.

This condition does not significantly reduce the margin of safety since sufficient margin exists in transient and LOCA analyses for conditions when false high reactor water level spiking is predicted to occur. The level fluctuations have not occurred above 600 psig and therefore do not effect limiting FSAR transient and accident analyses.

Additionally, this request does not affect the in-plant controls on the release of radioactive materials; therefore, operating under these conditions does not involve irreversible environmental consequences.

Conclusion

Based on the engineering evaluations of the plant design described in Attachment 1, we believe plant safety is maintained and an acceptable basis for this waiver is established.

This information is provided for your review and concurrence. Please do not hesitate to contact me if you have any questions regarding this matter.

R. A. Anderson

Attachment 1

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT 1
BOSTON EDISON COMPANY
ENGINEERING EVALUATION

1. Initiating Documents

F&MRs 92-78, 79, 80

2. Affected items (System, Subsystem, Train, Component, or Device)

LT79, Indication
LT1001-650A(B), Indication
LT73A(B), Containment Cooling Permissive at 2/3 Core Height
LT646A(B), Feedwater Control Signal
LT120A(B,C,D), ATWS - Trip Recirc Pumps, ARI (-49")
LI59A(B), Indication
LT57A(B) and LT58A(B),
Scram and Containment Isolation (+9")
Group 1 Isolation - High Level (+48")
Group 1 Isolation, Recirc Pump Trip, Cont. Isol. (-49")
LT72A(B,C,D),
Initiate CSCS and RCIC, Start EDGs (-49")
Trip HPCI/RCIC (+48")
Close Main Turbine Stop Valves (+48")

3. Specified Functions of Affected Items

Devices that provide input to the Feedwater Control System do not perform an active safety function and are not evaluated below.

LT79, LT1001-650A(B), LI59A(B)

These devices provide local or control room indication of reactor water level.

LT73A(B), Containment Spray Permissive at 2/3 Core Height

During a pipe break inside containment, drywell or suppression pool pressure or temperature may become high enough that operators choose to initiate containment cooling. This permissive allows operators to initiate containment cooling only after the core has been reflooded and 2/3 core coverage has been achieved. After reflood, core cooling flow requirements are reduced and diversion of LPCI flow is permitted (FSAR 7.4.3.5.4 and 3.3.6.5.2). The core can be cooled sufficiently should the water level be reduced to 2/3 core height (Tech. Spec. Bases 2.1.3).

LT120A(B,C,D), ATWS - Trip Recirculation Pumps, Alternate Rod Insertion

These devices provide a signal to trip the recirculation pumps and actuate vent valves in the scram air header to initiate a reactor scram. These actions provide a backup means of introducing negative reactivity to the reactor in the unlikely event of RPS failure. (FSAR 3.9).

LT57A(B) and LT58A(B)

Reactor Scram (+9")

A reactor scram provides timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier (by excessive temperature) and the nuclear system process barrier (by excessive pressure). The reactor shall scram to prevent fuel damage for abnormal operational transients. A low level in the reactor vessel indicates that the reactor has the potential of being inadequately cooled. The effect of a decreasing water level is to decrease the reactor coolant inlet subcooling. The effect is the same as raising feedwater temperature. Should level decrease too far, fuel damage could result as steam forms around fuel rods. The level setting is selected high enough above the top of active fuel to assure that enough water is available to account for evaporation losses and displacements of coolant following the most severe level decrease transients. The selected setting is used in the development of thermal hydraulic operational limits (FSAR 7.2).

Primary and Secondary Containment Isolation (+9")

For pipe breaks inside containment (PBICs), the low water level isolation function provides timely protection against the onset and consequences of gross release of radioactive materials from the fuel and nuclear system process barrier by closing off release routes through primary containment. For pipe breaks outside containment (PBOCs), the low water level isolation provides a barrier between the reactor and the breach, thus stopping the release of radioactive materials and conserving reactor coolant. For PBOCs, the isolation valves for the break shall be closed prior to core uncovery. A low level in the reactor vessel could indicate either a pipe break or level reducing transient such as a loss of feedwater. (FSAR 7.3 and 5.2). For pipe breaks inside containment, the standby gas treatment and reactor building isolation systems create another barrier to the release of radioactive materials that could lead to offsite doses in excess of 10 CFR100 guidelines (FSAR 5.3).

High Level Group 1 Isolation (+48")

The high level Group 1 isolation protects against rapid depressurization due to a pressure regulator system malfunction during startup (FSAR 7.3).

Containment Isolation and Recirc. Pump Trip (-49")

The containment isolation signal at (-49") serves the same function as the (+9") setting but was selected to allow the removal of heat from the vessel for a predetermined time after reaching the (+9") scram level setting, and high enough relative to the TAF to assure CSCS performance in the event of a large break in the nuclear process barrier. The (-49") containment isolation setting completes the isolation of containment and the reactor vessel by closing main steam isolation valves and other minor process lines (FSAR Section 7.3).

The (-49") recirculation pump trip protects the recirculation pump from damage due to low NPSH. This function is not a safety function.

LT72A (B,C,D)

Initiate CSCS and RCIC, Start EDGs (-49")

Reactor vessel low-low water level is indicative of a loss of coolant event and the potential to overheat the fuel clad. These devices initiate high pressure coolant injection and reactor core isolation cooling immediately. Core Spray and RHR-LPCI are initiated if these devices trip coincident with reactor vessel low pressure, or if low-low water level is sustained beyond a preset time delay. The integrated response of the CSCS systems assures the fuel is adequately cooled under abnormal and accident conditions.

Starting of the diesel generators immediately after reaching low-low water level is an anticipatory measure that assures the availability of AC power for CSCS and mitigative systems without any diesel start time delay if offsite power is subsequently lost. If offsite power is available, the diesel generators will start and run without closing onto safety buses.

The HPCI/RCIC trip at (+48") terminates the addition of water to the reactor vessel because water level is near the top of the steam separators and the trip prevents gross moisture carryover to the HPCI and RCIC turbines.

The turbine stop valves trip at (+48") to protect the main turbine from moisture carryover when the MSIVs are open. This protective feature is not safety-related.

4. References

1. Drawing M253, SH 1
2. FSAR Sections 7.2, 7.3, 7.4, 7.8, 3.9, 7.10
3. Technical Specifications 2.1, 3.1, 3.2, 3.5
4. GE-NE-187-38-1091, November 1991, Safety Evaluation of the Water Level Spiking Phenomenon Observed at Pilgrim Nuclear Power Station
5. GE-NE-187-69-1291, December 1991, New Analytical Limit for Low-Low Water Level (SUDDS/RF 91-178)
6. GE Letter LLC-52-91, November 15, 1991, L.L. Chi to J. Gosnell, Analytical Limit For Scram Water Level and HPCI/RCIC High Water Level Trip
7. NEDC-31852P, SAFER/GESTR-LOCA Analysis
8. FSAR Section 14 and Appendix R
9. NEDO-24708A, Additional Information Required for NRC Staff Generic Report on BWRs

5. Safety Concern

During shutdowns of PNPS, water level fluctuations have been observed at zero power, low pressure (roughly 470 psig or lower) conditions. The peaks of these fluctuations are small (approximately 5 inches) at 470 psig and progressively increase to an observed maximum of 22 inches at 10 psig. The maximum duration of observed peaks has been approximately 40 to 60 seconds. Since protection systems are receiving erroneously high level indications, there is a concern that actual low water level conditions may not initiate required safety actions in the time required to fulfill their safety functions. Also, under actual normal water level conditions, inadvertent false high level actuations may affect safety functions.

This evaluation focuses on the ability of the reactor vessel water level (RVWL) instrumentation to initiate automatic protective actions in response to an actual RVWL decrease or increase coincident with onset of spiking (a highly improbable event). Each specific accident/transient crediting low or high RVWL with a protective action initiating signal is discussed. It is recognized that this spiking could also cause a conservative (but necessary) Group 1 isolation during a normal cooldown/depressurization sequence. This eventuality is addressed since it results in temporary loss of access to the preferred heat sink.

This evaluation assesses safety impact based on observed empirical data. As such, the following general assumptions are implicit in the evaluation:

- Spiking does not occur above 600 psig.
- The spiking phenomenon occurs similarly during transient and accident conditions as has been actually observed.

Specific assumptions are discussed in the body of the evaluation.

6. Safety Assessment

The water level spiking phenomenon has occurred on at least four separate occasions. The phenomenon has been random in frequency but is generally repeatable in magnitude and duration at various low reactor pressures. At accelerated cooldown/depressurization, spiking tends to occur more frequently. It has not been observed above roughly 470 psig and has occurred only during depressurizations during shutdown operations. The 'B' Train instruments experienced spiking beginning at roughly 470 psig (approximately 5 inch spikes), whereas the 'A' Train instruments began spiking at 70 psig (approximately 2 inch spikes). 'A' Train responses have generally been bounded by 'B' Train responses in amplitude, duration, and reactor pressure when spiking began to occur.

Although the magnitude, duration, and frequency of level spiking is substantially less on the 'A' train instrument rack, the 'A' train will be assumed to respond similar to 'B' Train in this evaluation for conservatism. Typically, spike amplitudes have increased as reactor pressure decreased with largest observed spikes occurring at 10 psig (approximately 22 inches). Above approximately 100 psig, the largest observed spikes have been 6" in the 'B' Train and no spiking has occurred in 'A' Train. Durations are typically 20 to 30 seconds and have been as long as 60 seconds. Spikes have always indicated higher than actual water level. Smaller spikes at higher pressures tend to be shaped like a plateau (i.e., square wave). The height of these spikes is often sustained. The larger spikes at lower pressures typically ramp up and down, with the peak values existing only momentarily. Although apparently random in occurrence, spiking does not generally occur simultaneously in both trains.

Variations in water level indications are not an unknown phenomenon in BWRs. Level indications are sensitive to changes in pressure, temperature, static head and flow across the sensing nozzle. All of these variables change rapidly during transient and accident conditions in the reactor. During a LOCA, water will be flashing and boiling throughout the vessel. Actual level will exist more as a range of levels as opposed to a specific value with indicated level tending to oscillate about the actual level (see Reference 9). Level oscillations as small as two to four inches would tend to mask most observed level spiking above 100 psig. At trip setpoint levels, little or no trip delay is therefore expected. Below 100 psig, spiking may not be completely masked by oscillations but the durations of the spikes would be shortened.

Various possible mechanisms may be the cause for this phenomenon. Although the exact cause has not been identified, the empirical data over the last four shutdowns demonstrates that the spikes, although random in frequency, are generally predictable and repeatable in the associated reactor conditions.

Postulated Abnormal Operational Transients

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. (References 4 and 8). 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 (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 MSIV closure. Otherwise, since the event is above 600 psig, no level spiking will occur.

If the event occurs with the reactor not in RUN mode but at power (i.e., 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. 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 low level prior to operator intervention. For plant Shutdown, all control rods are inserted prior to going below 600 psig.

CSCS, Containment Isolation, RCIC, Diesel Generator Functions (-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 (see References 4 and 8). 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 PCIS logic. Diesel generators start on emergency bus low voltage. 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 no level spiking concerns 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 (FSAR 7.3). 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. Level would drop relatively slowly and the pressure drop would be small (FSAR Appendix 6.2.4.3). Spiking 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. Reference 5 indicates that reactor level drops no lower after RCIC initiation, assuming only RCIC is available, a high initial reactor power, and using a level initiation point of -57". Considering the timeframes associated with the observed spikes and the low steaming rates when shutdown below 600 psig, substantial level exists above top of active fuel.In the unlikely event that water level spiking occurs for events initially below 600 psig, considerable margin exists for core cooling.

High Level Isolation Functions (+48")

Transient events that could lead to high water level conditions are not affected since the level spikes are in the high direction and will only cause the required functions to occur sooner. Premature trip of the HPCI/RCIC systems is not a concern because the water level is still substantially above top of active fuel.

<u>Pipe Breaks Inside Containment (PBIC)</u>

At Rated Power

Reference 4 indicates that for pipe breaks inside containment at full power conditions (other than the large steam line breaks), reactor pressure will exceed 600 psig when low and low-low water levels are reached, thereby demonstrating that no water level spike effects will occur.

Large steam line breaks may cause depressurization 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 credit low water level scram; high drywell pressure is used (FSAR 14.5.3.1). The MSIVs close promptly on high steam line flow or low steam line pressure. Primary and secondary containment isolations and CSCS and EDG initiation occur promptly 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 isolation
- ADS actuation
- RCIC actuation

The reactor water cleanup system receives isolation signals in response to reactor vessel low water level (+9"), rupture of associated piping, standby liquid control injection, or high system temperature that could effect resin performance. Provided the RWCU isolation valves are closed prior to reactor level reaching the top of active fuel, no containment isolation concerns exist. LOCA analysis of the main steam line break (Reference 7) 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. (FSAR 5.2.8.3). With a RWCU isolation valve closure time of 25 seconds, a delay of greater than 70 seconds would be required to potentially have the valves open with the core uncovered. Such delays concurrently associated with large spiking are not expected based on empirical data. Margins indicate this is not a concern.

For large steamline breaks, reactor pressure drops so rapidly (Reference 7) that the ADS and RCIC functions are unnecessary.

LOCA analyses are not affected because level responses are not credited in the analyses (Reference 4).

In STARTUP Mode

For breaks that occur in STARTUP mode, the above discussion is applicable. MSIVs close on high steam flow.

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. Unlikely spiking that may slightly delay MSIV closure on low-low level (if not isolated on high steam flow) are not a concern.

Based on the above discussions and considering that PNPS LOCA analyses (Reference 7) assume an initial power of 102%, initial pressure of 1050 psig, and reduced LPCI and core spray flowrates (5% and 10% respectively), considerable margin exists for PBICs to conclude that water level spikes will not prevent 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 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 directed by EOPs. Therefore, if vessel level spiking were to occur coincident with initiation of containment cooling, the procedural controls provided by EOPs (during design basis events) with respect to vessel level ensure that actual water level is not below the 2/3 core coverage containment cooling permissive. The height difference between 2/3 core height and TAF is approximately 4 feet, well above the worst observed momentary peak.

High water level conditions would only be expected during a PBIC for a small break where HPCI/RCIC flows exceed break losses. However, for breaks this small, reactor pressure will remain high and premature high level HPCI/RCIC trips or slightly delayed re-initiations on low-low level are not considered a concern.

Pipe Breaks Outside Containment (PBOC)

Analysis of PBOCs indicates that reactor pressure remains high because PBOCs are reactor isolation events. These events have PCTs 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 spikes 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.

An analysis was performed of potential level spiking delays during a shutdown cooling line break. The safety concern is the requirement to have isolation valves closed before core uncovery. The analysis assumed shutdown cooling isolation pressure (110 psig) to maximize level reduction. Piping friction loss was not considered nor was contribution to level recovery from any CSCS. The analysis concluded that a 29 inch spike of continuous duration (i.e., isolation does not begin until actual level is 29 inches lower than the prescribed setpoint) would be required to uncover the core. A spike of 22 inch magnitude has been observed once in the 'B' train at 50 psig. Also, 22 inches represents the spike peak and is not a steady state condition. Based on empirical data, it is concluded that adequate margin exists to prevent core uncovery. Once the break is isolated, delays in CSCS initiation are bounded by full power LOCA analyses (i.e., in shutdown cooling, only decay heat is being generated, a break no longer exists, reactor is at low pressure, etc.).

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 spiking is not expected.

Inadvertent Group 1 Isolation During Shutdown Operations

High water level spiking leading to Group 1 isolations (+48") during low pressure shutdown operations are bounded by a loss-of-offsite power or inadvertent MSIV closure event at full power. As such, these events are of minor FSAR safety consequence. However, these events represent a loss of preferred heat sink which is considered an abnormal transient and a challenge to safety equipment.

Since the observed spiking is transient in nature, loss of the preferred heat sink (main condenser) is a temporary condition. Operations personnel have demonstrated the ability to restore the preferred heat sink in a timely manner during actual isolations. Also, numerous standby systems are available to support decay heat removal (e.g. HPCI, RCIC, ADS, etc.). Each of these systems can promptly be operated from the control room. Plant personnel are aware of the potential for spiking and will take precautions within practical operational limits to prevent isolations.

ATWS Events

The recirculation pump trip function supports ATWS by reducing core power and otherwise protects the pump from cavitation on low levels. ATWS events generally involve high reactor pressures because the reactor continues to generate some power. EOPs direct operators to primarily control pressure. In these cases, water level spiking is not expected (see Reference 4).

Fire Events

Fire events are also isolation events where reactor pressure remains high. No spiking will therefore occur. Automatic safety functions are assumed to be lost during these events. Operators manually perform required actions.

Summary

Based on the observed spiking phenomena, adequate margins exist in transient and LOCA analyses for conditions when spiking is predicted to occur. Delays in initiation of some CSCS or containment isolation equipment by water level instruments will not affect the ability of the combined systems to perform their safety functions assuming a single active failure.

Inadvertent high level isolations while shutting down represent an operational difficulty. However, the condition does not prevent performance of any safety function and is bounded by FSAR analysis of MSIV closure at full power.

Finally, level fluctuations have no effect on limiting FSAR transient and accident analyses because the fluctuations do not occur above 600 psig.