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10 CFR 50.90

April 4, 2016

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

SUBJECT: License Amendment Request - Proposed Changes to the High Pressure Coolant Injection System and Reactor Core Isolation Cooling System Actuation Instrumentation Technical Specifications

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon) is requesting approval for proposed changes to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, respectively.

The proposed changes modify the High Pressure Coolant Injection (HPCI) system and Reactor Core Isolation Cooling (RCIC) system actuation instrumentation TS by adding a footnote indicating that the injection functions of Drywell Pressure – High (HPCI only) and Manual Initiation (HPCI and RCIC) are not required to be operable under low reactor pressure conditions.

Exelon has concluded that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92, "Issuance of amendments."

The proposed changes have been reviewed by the LGS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the Exelon Quality Assurance Program.

This amendment request contains no regulatory commitments.

Attachment 1 provides the evaluation of the proposed changes. Attachment 2 provides a copy of the marked up TS pages that reflect the proposed changes.

Exelon requests approval of the proposed amendments by April 4, 2017. Upon NRC approval, the amendments shall be implemented within 60 days of issuance.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is notifying the Commonwealth of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the $4^{th}$ day of April 2016.

Respectfully,

James Barstow
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1. Evaluation of Proposed Changes
2. Markup of Proposed Technical Specifications Pages

| | | |
|---|---|---|
| cc: | Regional Administrator - NRC Region I | w/ attachments |
| | NRC Senior Resident Inspector - Limerick Generating Station | " |
| | NRC Project Manager, NRR - Limerick Generating Station | " |
| | Director, Bureau of Radiation Protection - Pennsylvania Department of Environmental Protection | " |

# ATTACHMENT 1

## License Amendment Request

## Limerick Generating Station, Units 1 and 2

## Docket Nos. 50-352 and 50-353

## EVALUATION OF PROPOSED CHANGES

**Subject:** **Proposed Changes to the High Pressure Coolant Injection System and Reactor Core Isolation Cooling System Actuation Instrumentation Technical Specifications**

## 1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon) is requesting approval for proposed changes to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, respectively.

The proposed changes modify the High Pressure Coolant Injection (HPCI) system and Reactor Core Isolation Cooling (RCIC) system actuation instrumentation TS by adding a footnote indicating that the injection functions of Drywell Pressure – High (HPCI only) and Manual Initiation (HPCI and RCIC) are not required to be operable under low reactor pressure conditions.

## 2.0 DETAILED DESCRIPTION

The HPCI system provides coolant to the reactor vessel following a small break Loss of Coolant Accident (LOCA) until reactor pressure is below the pressure at which the low pressure coolant injection systems, i.e., the Core Spray (CS) system or the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) system, maintain core cooling. The HPCI system is also capable of providing sufficient coolant to the reactor vessel to prevent actuation of the Automatic Depressurization System (ADS) and ensure that the reactor core remains covered in the event of a small pipe break with a break size of one-inch diameter or less. The RCIC system is designed to ensure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling in the event of a loss of normal feedwater flow.

The HPCI and RCIC systems consist of a steam driven turbine, pump assembly and associated system piping, valves, controls and instrumentation. HPCI and RCIC system controls automatically start the systems from the receipt of a reactor vessel low water level signal (Level 2). In addition, the HPCI system is designed to automatically start on primary containment (drywell) high pressure. Primary containment high pressure is an indication that a breach of the nuclear process barrier has occurred inside the drywell. The systems can also be initiated manually. In all actuation modes, the systems are prevented from operating above high reactor vessel water level (Level 8) using one-out-of-two twice trip logic that originates from wide range reactor vessel level instrumentation. The HPCI and RCIC system controls function to provide design makeup water flow to the reactor vessel until the amount of water delivered to the reactor vessel is adequate (Level 8), at which time the HPCI and RCIC systems automatically shut down. The HPCI and RCIC systems are designed to automatically cycle between the low (Level 2) and high (Level 8) reactor vessel water levels.

Reactor vessel low water level is monitored by four level sensors that sense the difference between the pressure of the water column in a constant reference leg (which is independent of reactor water level and density) and the pressure of the water column in the variable leg which varies linearly with the reactor vessel water level but is also dependent on the reactor water density. Each level sensor provides input to a trip unit

and the four trip units are connected in a one-out-of-two twice logic to provide an automatic HPCI and RCIC actuation signal. Reactor vessel low water level is an indication that reactor coolant is being lost and that the fuel is in danger of being overheated. The reactor vessel low water level setting for HPCI and RCIC system actuation (-38 inches of water level measured from instrument zero or Level 2) is selected high enough above the active fuel to start the HPCI and RCIC systems in time both to prevent excessive fuel cladding temperatures and to prevent more than a small fraction of the core from reaching the temperature at which gross fuel failure occurs.

The same four wide range reactor vessel water level sensors that provide the HPCI and RCIC low water level actuation signals also provide the HPCI and RCIC high reactor vessel water level trip signals. Each sensor is connected to a trip unit and the four trip units are connected in a one-out-of-two twice logic to automatically shut down the HPCI and RCIC systems. High water level in the reactor vessel indicates that the HPCI and RCIC systems have performed satisfactorily in providing makeup water to the reactor vessel and core cooling requirements are satisfied. The reactor vessel high water level setting that shuts down HPCI and RCIC (+54 inches of water level or Level 8) is near the top of the steam separators and is sufficient to prevent gross moisture carry over to the HPCI and RCIC turbines.

Due to the undesirable effects of flooding the reactor vessel above the main steam lines, the HPCI and RCIC systems are prevented from operating above the high reactor vessel water level (Level 8) setting in all actuation modes. Once actuated, the HPCI high reactor vessel water level trip is sealed in and will inhibit automatic (or manual) system actuation until indicated water level drops below the Level 8 setting and the high reactor vessel water level trip is manually reset, or the trip signal is automatically reset when indicated reactor vessel water level reaches the Level 2 actuation setting. For RCIC, there is no seal-in circuit for the Level 8 trip, so once reactor vessel water level drops below the Level 8 setting, the RCIC system can be manually initiated, or the system is automatically initiated when indicated reactor vessel water level reaches the Level 2 actuation setting.

The LGS wide range reactor vessel level instruments are differential pressure type instruments that are reactor coolant density sensitive and are calibrated to be most accurate at normal reactor operating conditions. As a result, at low reactor coolant temperatures and pressures, because the reactor vessel water density is higher than at calibration conditions, these instruments read higher than actual water level. This level indication condition at low reactor pressures is acknowledged via an existing footnote in TS Table 3.3.3-2, which references a discussion of this condition in TS Bases Figure B 3/4.3-1. As a result of this level instrumentation condition, a high reactor vessel water level (Level 8) trip signal is present for the HPCI and RCIC systems at low reactor pressures (up to 550 psig), but above the pressure at which the systems are required to be operable (150 psig for RCIC; 200 psig for HPCI).

TS Table 3.3.3-1 requires HPCI Drywell Pressure – High and Manual Initiation actuation instrumentation to be operable in various operational conditions. Any challenge to the reactor coolant pressure boundary (RCPB) that results in a high drywell pressure condition will result in an automatic actuation of the HPCI system actuation logic; however, the system will not start (either automatically or manually) with the Level 8 trip present, as

designed. The system would automatically start, without operator intervention, when a demand for inventory is sensed at low reactor vessel water low level (Level 2). This operation is the same for a high reactor vessel water level occurring at rated pressure and temperature, or at low reactor vessel pressures and temperatures. An evaluation has shown that high drywell pressure or manual actuation of the HPCI system while at low reactor pressure and temperature conditions is not necessary for HPCI to perform its intended safety function.

Similar to HPCI, TS Table 3.3.5-1 requires RCIC Manual Initiation actuation instrumentation to be operable in various operational conditions. However, as a result of the level instrumentation condition, a high reactor vessel water level trip signal is present for the RCIC system at low reactor pressures (up to 550 psig), but above the pressure at which the RCIC system is required to be operable (150 psig), and the system will not manually start with the Level 8 trip present, as designed. The system would automatically start, without operator intervention, when a demand for inventory is sensed at low reactor vessel water low level (Level 2). This operation is the same for a high reactor vessel water level occurring at rated pressure and temperature, or at low reactor vessel pressures and temperatures. An evaluation has shown that manual actuation of the RCIC system while at low reactor pressure and temperature conditions is not necessary for RCIC to perform its intended safety function.

Therefore, Exelon proposes to change the HPCI system actuation instrumentation Limiting Condition for Operation requirements specified in TS Table 3.3.3-1, "Emergency Core Cooling System Actuation Instrumentation," to add a footnote indicating that the injection functions of Drywell Pressure – High and Manual Initiation are not required to be operable under low reactor pressure conditions. In addition, Exelon proposes to change the RCIC system actuation instrumentation Limiting Condition for Operation requirements specified in TS Table 3.3.5-1, "Reactor Core Isolation Cooling System Actuation Instrumentation," to add a footnote indicating that the injection function of Manual Initiation is not required to be operable under low reactor pressure conditions.

In particular, the following changes are proposed for LGS, Units 1 and 2:

1. TS Table 3.3.3-1 will be revised on TS page 3/4 3-33 to add "###" to the end of Item b., "Drywell Pressure – High," and Item f., "Manual Initiation," listed under Item 3, "High Pressure Coolant Injection System," on the table.

2. The corresponding notation "### The injection functions of Drywell Pressure – High and Manual Initiation are not required to be OPERABLE with reactor steam dome pressure less than 550 psig," will be added to the table notation section on TS page 3/4 3-35.

3. TS Table 3.3.5-1 will be revised on TS page 3/4 3-53 to add "##" to the end of Item d., "Manual Initiation."

4. The corresponding notation "## The injection function of Manual Initiation is not required to be OPERABLE with reactor steam dome pressure less than 550 psig," will be added to the bottom of TS page 3/4 3-53.

See Attachment 2 for a copy of the marked up TS pages that reflect the proposed changes described above.

## 3.0 TECHNICAL EVALUATION

### Effect of HPCI and RCIC Unavailability on Plant Operation and Safety

Although HPCI and RCIC are technically not available when the indicated reactor vessel water level is above Level 8, the HPCI and RCIC systems are not needed under normal operating conditions when the reactor vessel water level is at some steady high level, and are only needed during a loss of level transient (e.g., loss of normal feedwater flow) or loss of coolant accident (LOCA) when the level has decreased to Level 2. For Limerick, the Level 2 setpoint is -38 inches from instrument zero, which is below the minimum level above which HPCI and RCIC are technically unavailable due to the wide range off-calibration condition at the lowest applicable reactor steam dome pressure of 150 psig. Thus, even the highest wide range off-calibration condition (corresponding to the lowest applicable dome pressure of 150 psig), is not large enough to prevent the Level 2 actuation. In addition, both the Level 8 trip and the Level 2 actuation for the HPCI and RCIC systems come from the same wide range instrumentation, so the Level 8 trip clears before the level reaches the Level 2 actuation, regardless of the off-calibration conditions. Thus, if a loss-of-level event (e.g., LOCA or loss of feedwater flow) occurred at 150 psig, the wide range off-calibration condition does not prevent HPCI and RCIC from performing their water injection function when the indicated reactor vessel water level reaches Level 2. Although the indicated level is higher than the calibrated level at Level 2, the water mass inventory is the same when the indicated Level 2 setpoint is reached; therefore, the HPCI and RCIC system response with respect to the core cooling function remains the same.

The unavailability of HPCI and RCIC due to wide range off-calibration condition is not an operational or safety problem for the following reasons described below.

1. The system response for LOCA and loss of feedwater flow events is governed by the liquid mass inventory in the reactor vessel. The difference between the indicated level in the vessel and the actual level in the vessel does not reflect a change in the liquid mass inventory in the downcomer region of the vessel because the level indication is a function of the total mass above the variable leg level tap elevation (approximately liquid density times height). This makes the level indication effectively self-compensating with respect to tracking the liquid mass inventory. Therefore, there is approximately the same liquid mass in the downcomer region at the same indicated level, regardless of the vessel pressure conditions. At the very low reactor powers considered in the off-calibrated conditions, the liquid mass inventory in the vessel is higher than the vessel inventory in the full power analyses of record because there is substantially less steam voiding in the core and upper plenum regions of the vessel. Because the HPCI and RCIC Level 2 system actuations occur on indicated level, the system response for LOCA and loss of feedwater flow events at the low reactor pressure off-calibration events is bounded by the full power analyses of record.

2. From the safety point of view, HPCI and RCIC are systems designed to mitigate events such as LOCA or loss of feedwater flow, and assure that the reactor vessel water level stays high enough to provide adequate core cooling. According to the boiling water reactor design base, HPCI and RCIC are only required to inject water into the reactor when the reactor vessel water level decreases to Level 2. For Limerick, the Level 2 setpoint is -38 inches from instrument zero. Because the HPCI and RCIC level trips/actuations come from the same wide range level instrumentation, the Level 8 trip clears and the HPCI and RCIC systems are available when the Level 2 actuation occurs, regardless of the off-calibration condition. When the calibration condition (due to low reactor pressure) is present, then for the same level loss scenario, the Level 2 actuation is reached at a later time than it would under calibrated conditions; however, the actual level is still well above the top of active fuel. Since the Level 2 actuation occurs, and HPCI and RCIC systems are available at that time, adequate core cooling is assured.

3. Once the reactor vessel water level reaches Level 2, the HPCI and RCIC actuation logic is designed to automatically trigger HPCI and RCIC injection. Depending upon the accident scenario and size of break, this injection could start to raise the water level inside the reactor, so the HPCI and RCIC trip logic is designed to automatically trip HPCI and RCIC if the reactor vessel water level reaches Level 8 in order to protect the turbine. The systems stay in the tripped configuration when the indicated level is above Level 8. If this level decreases below Level 8, the Level 8 trip signal is lifted, and when the level reaches Level 2, the HPCI and RCIC trip logic automatically clears the Level 8 trip and the actuation logic starts HPCI and RCIC injection. Thus, if the break is such that the HPCI and RCIC systems can provide enough water to raise the reactor vessel water level above Level 2, the HPCI and RCIC trip/actuation logic is designed to allow the system to automatically maintain level between Level 2 and Level 8 with no operator intervention. If the break is such that HPCI and RCIC cannot provide enough water to maintain level, then the low pressure Emergency Core Cooling Systems (ECCS) (ADS, Core Spray, and LPCI) are initiated when the indicated reactor vessel water level drops to Level 1 (-129 inches from instrument zero). Once the low pressure systems are initiated, HPCI and RCIC are not needed, although they would continue to inject (assuming the vessel pressure was high enough to drive the turbines) unless the reactor vessel water level recovered sufficiently to initiate the Level 8 trip.

4. When the reactor vessel water level is above Level 8, the HPCI and RCIC systems stay in a tripped configuration and cannot be manually initiated as long as the Level 8 trip signal is present. However, once the level drops below Level 8 and the Level 8 trip signal lifts, the systems can be manually initiated by the operator before auto-initiation at Level 2. Once the level decreases below Level 8, the high reactor vessel water level HPCI Level 8 seal-in can be cleared by pushing the manual reset button, and the HPCI system can then be manually initiated by pushing the manual initiate button, assuming the HPCI turbine is in the standby mode. There is no seal-in circuit for the RCIC Level 8 trip, so once the level decreases below Level 8 and the Level 8 trip signal lifts, the RCIC system can be manually initiated by pushing the manual initiate button, assuming the RCIC turbine is in the standby mode. For normal system

operation, manual action is not required, and the systems automatically cycle and maintain the reactor vessel water level between Level 2 and Level 8. However, the systems can be manually initiated between Level 2 and Level 8 if required, as described above.

5. When the HPCI and RCIC systems are injecting between Level 2 and Level 8, the systems can be manually stopped regardless of what level trip signals are present. HPCI can be manually stopped by pushing the HPCI trip push button, and RCIC can be manually stopped by pushing the RCIC trip push button.

6. The HPCI system is also initiated by the high drywell pressure LOCA actuation signal, and the Technical Specifications state that HPCI should be available for dome pressures as low as 200 psig. So the unavailability of HPCI due to the wide range off-calibration Level 8 trip also potentially affects the Technical Specification related to high drywell pressure HPCI initiation. Functionally, HPCI is only required to maintain sufficient water inventory in the reactor, and HPCI initiation is not required until the reactor vessel water level reaches Level 2. Therefore, initiation of HPCI on high drywell pressure is not required until the reactor vessel water level reaches Level 2, and because the wide range off-calibration condition does not affect reaching Level 2, the unavailability of HPCI on the high drywell pressure is not a safety concern. If a LOCA occurs, HPCI is automatically initiated on either low reactor vessel water level or high drywell pressure, but once initiated, the HPCI cycles on and off between Level 2 and Level 8 automatically regardless of whether the high drywell pressure trip is on or off. Manual initiation can be performed if required as described above. Automatic HPCI initiation on high drywell pressure would not be available when the Level 8 trip is sealed-in due to off-calibration low pressure operation. However, this is not a safety concern because the LOCA mitigation analysis of record at full power and high dome pressure, which bounds the low dome pressure LOCA, does not take credit for HPCI initiation on high drywell pressure.

**HPCI and RCIC Performance at Low Pressure**

The function of the HPCI and RCIC systems is to provide make-up coolant at high reactor pressure conditions to prevent core uncovery when the reactor vessel water level is low. The functional pressure range for the HPCI and RCIC overlaps the operational range of the low pressure systems (i.e., Core Spray and LPCI) as well as the safety and relief valve overpressure setpoints. While these systems are primarily designed to operate effectively at near rated operating conditions, they are also effective near the low end of their operating pressure range at mitigating coolant losses from decay heat (RCIC) and small leaks or vessel line breaks (HPCI). The following discussion addresses three scenarios that demonstrate the adequacy of the HPCI and RCIC performance considering the wide range level instrumentation behavior. These three scenarios are: 1) licensing basis LOCA analysis, 2) licensing basis loss of feedwater flow analysis, and 3) loss of inventory under low pressure start-up and shut-down operations.

1. For the licensing basis LOCA analysis, the HPCI system, which is most effective for small breaks, is disabled by the limiting assumed single failure (the RCIC system is not credited in the LOCA analysis). So for this case, the off-calibration of the wide range

level instrumentation and the delay of HPCI injection at low pressure are of no consequence. For the case where the HPCI system may be credited in the LOCA analysis, the consequences of off-calibration of the wide range level instrumentation at low pressure is not significant because the mass of water that provides the core cooling is unaffected by the density differences and so the core cooling analysis results would not be significantly affected. For larger break sizes, where the vessel depressurizes faster, the mitigation capability of HPCI is minimal. For these cases, the consequences of off-calibration of the wide range level instrumentation at low pressure are not significant and would not affect the core cooling analysis results.

2. For the licensing basis loss of feedwater flow analysis at rated conditions following a reactor scram, the mitigation by the RCIC and HPCI systems is demonstrated to be effective in preventing core uncovery and actuation of the ADS. But because there is no wide range off-calibration condition at rated pressure, the low pressure wide range level off-calibration behavior is not of concern for the short-term. However, for the long-term mitigation following the event, as the vessel is depressurized, the RCIC and HPCI systems maintain the level consistent with the wide range level indication as described previously, and there would be no core cooling or core uncovery concerns.

3. When the plant is in start-up or shut-down low pressure conditions, and an assumed loss of normal make-up flow or water inventory loss causes a reactor vessel water level decrease, the HPCI and RCIC systems would be initiated on low reactor vessel water level (Level 2) in a similar way as in the case of the licensing basis LOCA or loss of feedwater flow events described above. However, because the initial power conditions at low pressure would be in the 0 to 2% level, and more energy is required to boil the water at low pressure, the HPCI and RCIC systems provide more than enough water to mitigate the event and are more effective at providing core cooling than either the licensing basis LOCA or loss of feedwater flow events initiated at rated power and high pressure.

Therefore, the consequences of wide range reactor vessel water level off-calibration do not lead to a more severe reactor condition when low pressure conditions are considered. The effect of pressure on water density results in comparable mass inventory above the lower instrument tap; therefore, the wide range off-calibration condition has a minimal effect on the reactor water inventory available for core cooling.

**HPCI Discussion Specific to the High Drywell Pressure Initiation Function**

Automatic actuation of the HPCI system on the high drywell pressure signal is not required at low (200 to 550 psig) reactor pressures when HPCI may not be available because of a Level 8 trip, in order to satisfy HPCI system safety functions. The ECCS, including the HPCI system, are actuated on either of two separate and diverse indications of a pipe break LOCA: high drywell pressure or low reactor vessel water level. The high drywell pressure is the leading LOCA indicator, i.e., the high drywell pressure actuation occurs before the low-low reactor vessel water level (Level 2) actuation. Phenomenologically, there is no need for any of the ECCS, including HPCI, to be actuated on the earlier of the LOCA indications. The ECCS-LOCA analysis of record does not take credit for the earlier high drywell actuation of the ECCS in the determination of the licensing basis Peak

Cladding Temperature (PCT) for compliance with the 10 Code of Federal Regulations (CFR) 50.46 acceptance criteria. For a LOCA event initiating at low reactor pressures, HPCI automatically actuates and injects into the vessel when the wide range indicated level reaches the Level 2 low-low level initiation setpoint, with or without the presence of the high drywell pressure actuation. In this sense, for a LOCA event initiating at low reactor pressures, the HPCI system performs in a manner that is consistent with the HPCI actuation assumptions in the ECCS-LOCA analysis of record. Therefore, automatic actuation of the HPCI system on the high drywell pressure signal is not required in order to satisfy HPCI system safety functions.

The ECCS-LOCA analysis of record was performed assuming the normal operating conditions of full core thermal power and normal reactor vessel pressure. This analysis is bounding for LOCAs that initiate at low core power levels and reactor pressures because the low initial core power results in a much lower core heat-up rate compared to the heat-up rate for the full power condition, and the low initial vessel pressures result in a much lower break flow rate for a given break size and a much shorter time for the vessel to depressurize to the point when the low pressure ECCS can inject. The lower heat-up rates and shorter depressurization times result in lower PCTs. For these reasons, the ECCS-LOCA analysis of record is bounding for LOCAs that initiate at low reactor pressure conditions.

The ECCS low-level initiation signals come from the wide range reactor level instrumentation which, because of the off-calibration conditions, indicate a higher water level than is actually in the reactor vessel when the reactor pressure is low. At a reactor vessel pressure of 200 psig, the actual reactor vessel water level is approximately -55 inches (17 inches below the Level 2 setpoint) when the wide range indicated level reaches the Level 2 setpoint. However, the actual liquid mass inventory in the vessel is more than the liquid mass inventory in the vessel at Level 2 for the LOCA event starting at normal full power operating conditions. The increase in liquid mass inventory at low reactor pressure is due to the fact that the power level in the core is very low and there is very little steam being produced. At full power, steam makes up approximately 15% of the mass flow through the core, upper plenum, and separator standpipe regions below the reactor vessel water level. At very low powers, this steam volume is filled with liquid mass. The difference between the indicated level in the vessel and the actual level in the vessel does not reflect a change in the liquid mass inventory in the downcomer region of the vessel because the level indication is a function of the total mass above the variable leg level tap elevation (approximately liquid density times height). This makes the level indication effectively self-compensating with respect to tracking the liquid mass inventory. Therefore, there is approximately the same liquid mass in the downcomer region at the same indicated level, regardless of the vessel pressure conditions.

As the LOCA event proceeds, the low initial reactor vessel pressure results in a much lower break flow rate for a given break size and a much shorter time required for the vessel to depressurize to the point when the low pressure ECCS can inject. The HPCI system was sized to effectively quench the steam in the vessel and rapidly depressurize the vessel to allow the low pressure ECCS to inject without the aid of the ADS. For a LOCA event initiating at low reactor pressure, HPCI operates for only a very brief period before the vessel depressurizes below the lower end of the HPCI operating pressure

range (i.e., 200 psig), which limits the contribution of the HPCI in mitigating the event. Finally, in the ECCS-LOCA analysis of record, the limiting single failure for the full pipe break spectrum is the failure of the Division 2 direct current power source, which prevents the HPCI system from operating at all during the event. In addition, the very low initial core power level and decay heat at the time of the event (less than 3% of rated core power) results in a much lower core heat-up rate if the core uncovers compared to the heat-up rates in the full power analysis of record. Actuation of HPCI during a LOCA event initiated at low reactor pressure may provide a beneficial contribution in reducing the vessel pressure; however, operation of HPCI for this event is not necessary to assure adequate core cooling. For these reasons, the ECCS-LOCA analysis of record at full power and normal operating pressure is bounding for a LOCA in the 200-550 psig range, and automatic actuation of the HPCI system on the high drywell pressure signal is not required for the HPCI to perform its system safety functions in mitigating the consequences of a LOCA initiating at low reactor pressure.

**Conclusion**

The proposed changes make the LGS TS consistent with the plant design and licensing basis. Further, they clarify that the required initiation functions of the HPCI (high drywell pressure and manual) and RCIC (manual) actuation instrumentation are not required to be operable below 550 psig, and that the false indications of high reactor vessel water level due to instrumentation design and calibration requirements do not affect the safe operation of the plant.

## 4.0 REGULATORY EVALUATION

### 4.1 Applicable Regulatory Requirements/Criteria

The following regulatory requirements have been considered:

- Title 10 of the Code of Federal Regulations (10 CFR), Section 50.36, "Technical specifications," in which the Commission established its regulatory requirements related to the contents of the TS. Specifically, 10 CFR 50.36(c)(2) states, in part, "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility."

The proposed changes to the HPCI and RCIC actuation instrumentation requirements do not affect compliance with these regulations.

The applicable 10 CFR Part 50, Appendix A, General Design Criteria for Nuclear Power Plants, was considered as follows:

- Criterion 13 - Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure

boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

The proposed changes would retain the necessary safety function on low reactor vessel water level at reactor coolant system pressures below 550 psig because the measurement condition in reactor vessel water level decreases as the reactor vessel water level decreases and actuation of HPCI and RCIC would still occur.

## 4.2 Precedence

A similar change, to add a footnote to the High Pressure Core Spray (HPCS) system actuation instrumentation TS indicating that the injection functions of Drywell Pressure – High and Manual Initiation are not required to be operable when the indicated reactor vessel water level on the wide range instrument is greater than Level 8 coincident with low reactor pressure conditions, was requested by Mississippi Power & Light Company for Grand Gulf Nuclear Station, Unit 1 and was approved by the NRC in 1983 (References 1 and 2), and then again in 1986 (References 3 and 4). The 1983 change was initially a one-time change which was made permanent in 1986.

## 4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (Exelon), proposes changes to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, respectively.

The proposed changes modify the High Pressure Coolant Injection (HPCI) system and Reactor Core Isolation Cooling (RCIC) system actuation instrumentation TS by adding a footnote indicating that the injection functions of Drywell Pressure – High (HPCI only) and Manual Initiation (HPCI and RCIC) are not required to be operable under low reactor pressure conditions.

Exelon has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed changes involve the addition of clarifying footnotes to the HPCI and RCIC actuation instrumentation TS to reflect the as-built plant design and operability requirements of HPCI and RCIC instrumentation as described in the LGS Updated Final Safety Analysis Report (UFSAR).

   HPCI and RCIC are not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not increased. In

addition, the automatic start of HPCI on high drywell pressure, and the manual initiation of HPCI and RCIC, are not credited to mitigate the consequences of design basis accidents, transients or special events within the current LGS design and licensing basis.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No.

The proposed changes do not alter the protection system design, create new failure modes, or change any modes of operation. The proposed changes do not involve a physical alteration of the plant, and no new or different kind of equipment will be installed. Consequently, there are no new initiators that could result in a new or different kind of accident.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

Response: No.

The proposed changes have no adverse effect on plant operation. The plant response to the design basis accidents does not change. The proposed changes do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analyses. There is no change being made to safety analysis assumptions, safety limits or limiting safety system settings that would adversely affect plant safety as a result of the proposed changes.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Exelon concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

## 4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

## 6.0 REFERENCES

1. Letter from L. F. Dale, Mississippi Power & Light Company, to H. R. Denton, U.S. Nuclear Regulatory Commission, "Transmittal of Proposed Changes to Grand Gulf Technical Specifications," dated August 1, 1983.

2. Letter from A. Schwencer, U.S. Nuclear Regulatory Commission, to J. P. McGaughy, Mississippi Power & Light Company, "Amendment No. 10 to Facility Operating License No. NPF-13 – Grand Gulf Nuclear Station, Unit 1," dated September 23, 1983.

3. Letter from O. D. Kingsley, Jr., Mississippi Power & Light Company, to H. R. Denton, U.S. Nuclear Regulatory Commission, "Proposed amendment to the Operating License," dated January 29, 1986.

4. Letter from L. L. Kintner, U.S. Nuclear Regulatory Commission, to O. D. Kingsley, Jr., Mississippi Power & Light Company, "Change to Technical Specifications and Operating License Condition," dated October 17, 1986.

**ATTACHMENT 2**

**License Amendment Request**

**Limerick Generating Station, Units 1 and 2**
**Docket Nos. 50-352 and 50-353**

**Proposed Changes to the High Pressure Coolant Injection System and Reactor Core Isolation Cooling System Actuation Instrumentation Technical Specifications**

**Markup of Proposed Technical Specifications Pages**

**Unit 1 TS Pages**
3/4 3-33
3/4 3-35
3/4 3-53

**Unit 2 TS Pages**
3/4 3-33
3/4 3-35
3/4 3-53

## TABLE 3.3.3-1

## EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

| TRIP FUNCTION | MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a] | APPLICABLE OPERATIONAL CONDITIONS | ACTION |
|---|---|---|---|
| 1. CORE SPRAY SYSTEM*** | | | |
| a. Reactor Vessel Water Level - Low Low Low, Level 1 | 2/pump[b] | 1, 2, 3, 4*, 5* | 30 |
| b. Drywell Pressure - High | 2/pump[b] | 1, 2, 3 | 30 |
| c. Reactor Vessel Pressure - Low (Permissive) | 6[b] | 1, 2, 3<br>4*, 5* | 31<br>32 |
| d. Manual Initiation | 2[e] | 1, 2, 3, 4*, 5 | 33 |
| 2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM*** | | | |
| a. Reactor Vessel Water Level - Low Low Low, Level 1 | 2 | 1, 2, 3, 4*, 5* | 30 |
| b. Drywell Pressure - High | 2 | 1, 2, 3 | 30 |
| c. Reactor Vessel Pressure - Low (Permissive) | 2 | 1, 2, 3 | 31 |
| d. Injection Valve Differential Pressure-Low (Permissive) | 1/valve | 1, 2, 3, 4*, 5* | 31 |
| e. Manual Initiation | 1 | 1, 2, 3, 4*, 5* | 33 |
| 3. HIGH PRESSURE COOLANT INJECTION SYSTEM## | | | |
| a. Reactor Vessel Water Level - Low Low Level 2 | 4 | 1, 2, 3 | 34 |
| b. Drywell Pressure - High### | 4 | 1, 2, 3 | 34 |
| c. condensate Storage Tank Level - Low | 2[c] | 1, 2, 3 | 35 |
| d. Suppression Pool Water Level - High | 2 | 1, 2, 3 | 35 |
| e. Reactor Vessel Water Level - High, Level 8 | 4[d] | 1, 2, 3 | 31 |
| f. Manual Initiation### | 1/system | 1, 2, 3 | 33 |

INSERT

TABLE 3.3.3-1 (Continued)
EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also provides input to actuation logic for the associated emergency diesel generators.

(c) One trip system. Provides signal to HPCI pump suction valves only.

(d) On 1 out of 2 taken twice logic, provides a signal to trip the HPCI pump turbine only.

(e) The manual initiation push buttons start the respective core spray pump and diesel generator. The "A" and "B" logic manual push buttons also actuate an initiation permissive in the injection valve opening logic.

(f) A channel as used here is defined as the 127 bus relay for Item 1 and the 127, 127Y, and 127Z feeder relays with their associated time delay relays taken together for Item 2.

* When the system is required to be OPERABLE per Specification 3.5.2.

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

** Required when ESF equipment is required to be OPERABLE.

\## Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

\### The injection functions of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with reactor steam dome pressure less than 550 psig.

INSERT

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

| FUNCTIONAL UNITS | MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION* | ACTION |
|---|---|---|
| a. Reactor Vessel Water Level - Low Low, Level 2 | 4# | 50 |
| b. Reactor Vessel Water Level - High, Level 8 | 4# | 51 |
| c. Condensate Storage Tank Water Level - Low | 2** | 52 |
| d. Manual Initiation## | 1/system*** | 53 |

INSERT

*A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided all other channels monitoring that parameter are OPERABLE.

**One trip system with one-out-of-two logic.

***One trip system with one channel.

INSERT

#One trip system with one-out-of-two twice logic.

##The injection function of Manual Initiation is not required to be OPERABLE with reactor steam dome pressure less than 550 psig.

# TABLE 3.3.3-1

## EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

| TRIP FUNCTION | MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[(a)] | APPLICABLE OPERATIONAL CONDITIONS | ACTION |
|---|---|---|---|
| 1. CORE SPRAY SYSTEM*** | | | |
| a. Reactor Vessel Water Level - Low Low Low, Level 1 | 2/pump[(b)] | 1, 2, 3, 4*, 5* | 30 |
| b. Drywell Pressure - High | 2/pump[(b)] | 1, 2, 3, | 30 |
| c. Reactor Vessel Pressure - Low (Permissive) | 6[(b)] | 1, 2, 3<br>4*, 5* | 31<br>32 |
| d. Manual Initiation | 2[(e)] | 1, 2, 3, 4*, 5* | 33 |
| 2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM*** | | | |
| a. Reactor Vessel Water Level - Low Low Low, Level 1 | 2 | 1, 2, 3, 4*, 5* | 30 |
| b. Drywell Pressure - High | 2 | 1, 2, 3 | 30 |
| c. Reactor Vessel Pressure - Low (Permissive) | 2 | 1, 2, 3 | 31 |
| d. Injection Valve Differential Pressure-Low (Permissive) | 1/valve | 1, 2, 3, 4*, 5* | 31 |
| e. Manual Initiation | 1 | 1, 2, 3, 4*, 5* | 33 |
| 3. HIGH PRESSURE COOLANT INJECTION SYSTEM## | | | |
| a. Reactor Vessel Water Level - Low Low, Level 2 | 4 | 1, 2, 3 | 34 |
| b. Drywell Pressure - High#### | 4 | 1, 2, 3 | 34 |
| c. Condensate Storage Tank Level - Low | 2[(c)] | 1, 2, 3 | 35 |
| d. Suppression Pool Water Level - High | 2 | 1, 2, 3 | 35 |
| e. Reactor Vessel Water Level - High, Level 8 | 4[(d)] | 1, 2, 3 | 31 |
| f. Manual Initiation#### | 1/system | 1, 2, 3 | 33 |

INSERT

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also provides input to actuation logic for the associated emergency diesel generators.

(c) One trip system. Provides signal to HPCI pump suction valves only.

(d) On 1 out of 2 taken twice logic, provides a signal to trip the HPCI pump turbine only.

(e) The manual initiation push buttons start the respective core spray pump and diesel generator. The "A" and "B" logic manual push buttons also actuate an initiation permissive in the injection valve opening logic.

(f) A channel as used here is defined as the 127 bus relay for Item 1 and the 127, 127Y, and 127Z feeder relays with their associated time delay relays taken together for Item 2.

* When the system is required to be OPERABLE per Specification 3.5.2.

# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

** Required when ESF equipment is required to be OPERABLE.

## Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

### The injection functions of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with reactor steam dome pressure less than 550 psig.

INSERT

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

| FUNCTIONAL UNITS | | MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION* | ACTION |
|---|---|---|---|
| a. | Reactor Vessel Water Level - Low Low, Level 2 | 4# | 50 |
| b. | Reactor Vessel Water Level - High, Level 8 | 4# | 51 |
| c. | Condensate Storage Tank Water Level - Low | 2** | 52 |
| d. | Manual Initiation## | 1/system*** | 53 |

INSERT

*A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided all other channels monitoring that parameter are OPERABLE.

**One trip system with one-out-of-two logic.

***One trip system with one channel.

#One trip system with one-out-of-two twice logic.

INSERT

##The injection function of Manual Initiation is not required to be OPERABLE with reactor steam dome pressure less than 550 psig.