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10 CFR 50.90

January 15, 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 Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03

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 would reduce the reactor vessel steam dome pressure associated with the LGS TS Safety Limits (SLs) specified in TS 2.1.1 and TS 2.1.2 from 785 psig to 685 psig; increase the Trip Setpoint for TS Table 3.3.2-2, Function 1.c, Main Steam Line Pressure – Low, from ≥ 756 psig to ≥ 840 psig; and increase the Allowable Value for TS Table 3.3.2-2, Function 1.c, Main Steam Line Pressure – Low, from ≥ 736 psig to ≥ 821 psig. In addition, the associated TS Bases will be revised to reflect the above changes.

The proposed changes were identified as a result of GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," issued on March 29, 2005, and are being submitted based on the results of subsequent GE analyses that were sponsored by the Boiling Water Reactor Owners Group. These changes are valid for the NRC approved pressure range pertinent to the critical power correlations applied to the fuel types in use at LGS.

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

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. Attachment 3 provides a copy of the marked up TS Bases pages that reflect the proposed changes (information only).

Exelon requests approval of the proposed amendment by January 15, 2017. Upon NRC approval, the amendment shall be implemented within 120 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 15th day of January 2016.

Respectfully,

James Barstow

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

Attachments:
1. Evaluation of Proposed Changes
2. Markup of Proposed Technical Specifications Pages
3. Markup of Proposed Technical Specifications Bases Pages (Information Only)

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	"

bcc:	Senior Vice President - Mid-Atlantic Operations	w/o attachments
	Senior Vice President, Engineering and Technical Services	"
	Site Vice President - Limerick Generating Station	"
	Plant Manager - Limerick Generating Station	"
	Director, Operations - Limerick Generating Station	"
	Director, Site Engineering - Limerick Generating Station	"
	Director, Site Training - Limerick Generating Station	"
	Manager, Regulatory Assurance - Limerick Generating Station	w/ attachments
	Manager, Licensing, KSA	"
	G. Stewart, KSA	"
	PA DEP BRP Inspector - LGS, SSB2-4	"
	Commitment Coordinator - KSA	"
	Correspondence Control Desk - KSA	"

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 Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03**

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 would reduce the reactor vessel steam dome pressure associated with the LGS TS Safety Limits (SLs) specified in TS 2.1.1 and TS 2.1.2 from 785 psig to 685 psig; increase the Trip Setpoint for TS Table 3.3.2-2, Function 1.c, Main Steam Line Pressure – Low, from ≥ 756 psig to ≥ 840 psig; and increase the Allowable Value for TS Table 3.3.2-2, Function 1.c, Main Steam Line Pressure – Low, from ≥ 736 psig to ≥ 821 psig. In addition, the associated TS Bases will be revised to reflect the above changes.

The proposed changes were identified as a result of GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," issued on March 29, 2005 (Reference 1). These changes are valid for the NRC approved pressure range pertinent to the critical power correlations applied to the fuel types in use at LGS.

2.0 DETAILED DESCRIPTION

In 2005, GE Energy – Nuclear issued 10 CFR Part 21 Safety Communication SC05-03 identifying the potential vulnerability for the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient event to result in a condition in which TS SL 2.1.1 may be exceeded. This does not challenge the fuel cladding integrity or constitute a safety hazard as determined by GE. However, there exists a potential for violation of a TS SL for the PRFO event. As such, Exelon is proposing to revise the reactor vessel steam dome pressure specified in TS SLs 2.1.1 and 2.1.2 to 685 psig consistent with the NRC approved pressure range of critical power correlations for the current fuel designs in the LGS, Unit 1 and Unit 2, reactor cores.

In addition, in response to Reference 1, the BWR Owners Group commissioned development of a methodology for plants to assess the adequacy of their current Main Steam Isolation Valve (MSIV) closure at the low pressure isolation setpoint (LPIS) setting and to provide a set of recommendations for what actions should be taken based on the outcome of their assessment. The methodology and recommendations are documented in a BWR Owners Group report (Reference 2). The methodology is developed by analyzing a limiting plant, assessing uncertainties, and determining a method to conservatively scale the limiting plant's results to other plant configurations and operating flexibility options through sensitivity studies. The scaling methodology is applied to an example plant to demonstrate its adequacy. Additionally, a parametric study using a 720 psig LPIS setting with various plant configurations is provided in the Reference 2 report.

Based on the results of the studies documented in Reference 2, it has been determined that the current MSIV LPIS analytical limit of 720 psig at LGS is not sufficient to preclude reactor vessel steam dome pressure from falling below the proposed 685 psig while above 25% power for current operation during a PRFO event. As a result, a change to the MSIV LPIS analytical limit from 720 psig to 805 psig is required. Based on this new LPIS analytical limit, associated changes to MSIV LPIS allowable value and trip setpoint specified in the LGS TS are required.

Therefore, changes to the LGS, Unit 1 and Unit 2, TS are proposed as follows:

1. Reduce the reactor vessel steam dome pressure limit specified in TS SL 2.1.1 from less than 785 psig to less than 685 psig. The proposed SL would read:

   "THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 685 psig or core flow less than 10% of rated flow."

   In addition, the associated Action requirement would read:

   "With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 685 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1."

2. Reduce the reactor vessel steam dome pressure limit specified in TS SL 2.1.2 from greater than 785 psig to greater than 685 psig. The proposed SL would read:

   "The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 for two recirculation loop operation and shall not be less than 1.12 for single recirculation loop operation with the reactor vessel steam dome pressure greater than 685 psig and core flow greater than 10% of rated flow."

   In addition, the associated Action requirement would read:

   "With MCPR less than 1.09 for two recirculation loop operation or less than 1.12 for single recirculation loop operation and the reactor vessel steam dome pressure greater than 685 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1."

3. Increase the Trip Setpoint for TS Table 3.3.2-2, Function 1.c, Main Steam Line Pressure – Low, from ≥ 756 psig to ≥ 840 psig.

4. Increase the Allowable Value for TS Table 3.3.2-2, Function 1.c, Main Steam Line Pressure – Low, from ≥ 736 psig to ≥ 821 psig.

Markups of the above proposed TS changes are provided in Attachment 2. In addition, markups of the associated TS Bases pages are provided in Attachment 3. The Bases markups are provided for information only, and do not require NRC approval.

3.0 TECHNICAL EVALUATION

Reactor depressurization transients, such as PRFO, are non-limiting for fuel cladding integrity because the critical power ratio (CPR) increases during the event, and they are not typically included in the scope of reload evaluations. Previous evaluations by GE predicted that reactor vessel water level would swell during a PRFO transient; the depressurization would be terminated by a high level turbine trip. However, reactor vessel water level swell is difficult to predict and the reactor vessel water level swell portion of transient models have larger uncertainties than other portions of the transient models.

Recent evaluations by GE with improved transient models have determined that the reactor vessel water level swell may not be sufficient to reach the high level trip, in which case the depressurization could be terminated by MSIV closure at the LPIS. Depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor vessel steam dome pressure could decrease to below 785 psig for a few seconds while thermal power exceeds 25% of rated, which would exceed the conditions specified in TS SL 2.1.1. The methodology developed to assess the adequacy of the current LPIS setting and to provide a set of recommendations for the actions to be taken is documented in Reference 2. Based on the results of the studies documented in Reference 2, it is proposed that the low reactor vessel steam dome pressure specified in LGS TS SLs 2.1.1 and 2.1.2 be changed from 785 to 685 psig. In addition, the MSIV LPIS analytical limit associated with TS Table 3.3.2-2, Function 1.c is proposed to be increased from 720 psig to 805 psig.

<u>Safety Limit 2.1.1 and 2.1.2 Changes</u>

TS SLs are specified to ensure that acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). Reactor core SLs are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur if the SLs are not exceeded.

The current LGS TS SL 2.1.1 requires that thermal power shall be ≤ 25% rated thermal power when reactor vessel steam dome pressure is < 785 psig or core flow is < 10% of rated. This SL was introduced to preclude the need for CPR calculations when reactor vessel steam dome pressure is less than 785 psig. The reactor power value in TS SL 2.1.1 is selected to ensure that reactor power remains well below the fuel assembly critical power for the conditions in which CPR calculations are not performed.

The current LGS TS 2.1.2 requires that the minimum critical power ratio (MCPR) shall be ≥1.09 for two recirculation loop operation or ≥1.12 for single recirculation loop operation with the reactor vessel steam dome pressure at ≥785 psig and core flow at ≥ 10% of rated thermal power. This SL is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties. This fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated.

These TS SLs ensure the validity of MCPR calculations when reactor power is >25% and the reactor vessel steam dome pressure is within the validity range of the GEXL correlation. GE Energy-Nuclear has updated the validity range of GEXL14 and GEXL17 Correlations as noted below, which allows the pressure to be reduced to 685 psig from 785 psig.

Limerick currently has a mixed core of GE14 and GNF2 fuel types in Unit 1, and GNF2 fuel type in Unit 2. GE utilizes the GEXL correlation to perform CPR calculations for all of the fuel types in use at LGS. The lower bound limit of 685 psig (i.e., 700 psia) for the GEXL17 correlation is documented and justified in NEDC-33292P for GNF2 Fuel (Reference 3). This lower bound limit is discussed and NEDC-33292P is referenced in NEDC-33270P (Reference 4). NEDC-33270P was submitted to the NRC as part of Amendment 33 to NEDE-24011-P. NEDE-24011-P Amendment 33 was approved by the NRC and incorporated into Revision 17 of NEDE-24011-P-A (Reference 5). Therefore, the use of 685 psig as lower bound limit for GNF2 fuel has been approved by the NRC for use per NEDE-24011-P-A by reference. Furthermore, the lower bound limit of 685 psig (i.e., 700 psia) for the GEXL14 correlation is documented and justified in NEDC-32851P-A for GE14 Fuel (Reference 6). This topical report has been reviewed and approved by the NRC.

The proposed change in LGS TS 2.1.2, which specifies the SL on the MCPR, expands the range of applicability of the SL on the MCPR to a low pressure established by GEXL correlation. There is no reduction in margin of safety as a result of expanding the range of applicability of the GEXL correlation, which allows decreasing the low reactor vessel steam dome pressure SL. The low pressure SL protects transition boiling at the reactor fuel cladding. The conditions under which this occurs are determined by the physical configuration of the fuel and reactor thermal-hydraulics, neither of which are affected by the proposed change in the SL. The margins are enhanced by the proposed change since the applicability of the GEXL correlation has been expanded through increased testing demonstrating adequate performance of the correlation over an expanded range. Furthermore, safety margin is increased due to the proposed change to ensure LGS will not enter into an unanalyzed condition during a PRFO event such as is potentially possible with the current low pressure SL.

The PRFO event involves the failure of the pressure regulator in the open direction causing the turbine control valves to fully open. This causes the reactor to depressurize rapidly. When the main steam line low pressure isolation setpoint is reached, a closure signal for the MSIVs is initiated and a reactor scram occurs. As the MSIVs approach full closure, reactor depressurization terminates and pressure commences to rise to the safety-relief valve setpoint, thus preventing reactor pressure from decreasing below the proposed SL of 685 psig while core thermal power is still above 25% of rated thermal power. This event, which is non-limiting for fuel cladding integrity and is not typically included in the scope of reload evaluations, also causes the CPR to increase. With an initial condition that is restricted by the Operating Limit Minimum Critical Power Ratio (OLMCPR) and an event that causes the CPR to increase, the margin to the Safety Limit MCPR increases during the event, and therefore, no threat to fuel cladding integrity exists.

Main Steam Isolation Valve (MSIV) Low Pressure Isolation Setpoint (LPIS) Changes

Low steam pressure at the turbine inlet while the reactor is operating could indicate a malfunction of the steam pressure controller in which the turbine control valves or the turbine bypass valves become fully open and cause rapid depressurization of the reactor vessel. From partial-load operating conditions, the rate of decrease of saturation temperature could exceed the allowable rate of change of reactor vessel temperature. A rapid depressurization of the reactor vessel while the reactor is near full power could result in undesirable differential pressures across the channels around some fuel bundles of sufficient magnitude to cause mechanical deformation of channel walls. The steam pressure at the turbine inlet is monitored to forestall the effects due to undesirable pressure differential.

The proposed MSIV LPIS setting, calculated at 840 psig based on the new analytical limit of 805 psig, is far enough below normal turbine inlet pressure to prevent spurious isolation, yet high enough to provide timely detection of a pressure controller malfunction.

No new failure modes are created by changing the MSIV LPIS. In addition, the reactor fuel system is unaffected by this setpoint change. The PRFO is the only AOO that could potentially be affected by this setpoint change. However, AOOs that cause reactor depressurization are not analyzed for major plant modifications (e.g., power uprate) because thermal margins increase during these events, and therefore, they are non-limiting.

The revised analytical limit of 805 psig is higher than the current value of 720 psig and will result in earlier main steam line isolation to terminate the depressurization event. The conclusion that PRFO is non-limiting for thermal limit impact is not affected by this change. Also, the change in LPIS analytical limit will not affect the outcome of the limiting PRFO AOO transient.

In addition, for the LGS anticipated transient without scram (ATWS) analysis, the increased LPIS analytical limit will result in an increased allowable value that results in earlier steam line isolation and recirculation pump trip. As a result, the analysis with its margins to the ATWS acceptance criteria remains applicable with respect to this change.

Conclusion

Exelon has determined that reducing the reactor vessel steam dome pressure limit specified in TS SLs 2.1.1 and 2.1.2, in conjunction with increasing the allowable value and trip setpoint specified in TS Table 3.3.2-2, Function 1.c for the main steam line low pressure isolation, provides adequate margin for the PRFO transient, such that the reactor vessel steam dome pressure will remain above the proposed revision to the TS SLs. The combination of the lower TS SL and the higher LPIS trip setpoint and allowable value provides a wider pressure range for transients to demonstrate compliance with MCPR limits. Therefore, the proposed change offers a greater pressure margin for a PRFO transient than what is currently available.

In addition, the proposed reduction of the reactor vessel steam dome pressure in the TS SLs is consistent with the NRC-approved GEXL14 and GEXL17 correlations lower-bound pressure for the GE14 fuel type and the GNF2 fuel type, respectively, in the LGS, Unit 1 reactor core, and GEXL17 correlation lower-bound pressure for GNF2 fuel type in the LGS, Unit 2 reactor core.

Therefore, the proposed changes resolve the 10 CFR Part 21 condition concerning the potential to violate reactor core SL 2.1.1 during a PRFO transient reported in Reference 1.

If Exelon decides to switch to a different fuel design from those currently in use in the LGS reactor cores, the CPR correlation will be reviewed as part of the normal fuel design change and reload licensing processes. If the CPR correlation for the new fuel design has a lower bound pressure which is higher than the limit specified in the TS, then a license amendment request (LAR) will be submitted for NRC review and approval. If the CPR correlation has a lower bound pressure which is lower than the TS limit, then no LAR will be required since the TS would set a conservative lower bound.

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. 10 CFR 50.36(c) requires that the TS include, among other things, items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. In addition, 10 CFR 50.36 states that the TS will include Safety Limits for nuclear reactors which are stated to be "limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

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

- Criterion 10 - Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The purpose of the safety limit is to ensure that specified acceptable fuel design limits are not exceeded during steady state operation and analyzed transients. The fuel cladding is one of the physical barriers that separate the radioactive materials from the environment.

The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Fuel cladding perforations can result from thermal stresses, which can occur from reactor operation significantly above design conditions. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel cladding damage could occur. The reactor core safety limits are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur due to onset of transition boiling if the safety limits are not exceeded.

In addition, the reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The reactor core components consist of fuel assemblies, control rods, in-core ion chambers and related items. The fuel is designed to provide high integrity over a complete range of power levels including transient conditions. As described above, the LGS TS SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and AOOs. Reactor core SLs are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur if the SLs are not exceeded.

As long as the reactor core pressure and flow are within the range of validity of the specified critical power correlation, the proposed reactor steam dome pressure change to reactor core SLs 2.1.1 and 2.1.2, in conjunction with increasing the allowable value and trip setpoint specified in TS Table 3.3.2-2, Function 1.c for the main steam line low pressure isolation, will continue to ensure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition. This satisfies the requirements of GDC 10 regarding acceptable fuel design limits and continues to assure that the underlying criteria of the safety limit is met.

Based on the above, the proposed changes satisfy the regulatory requirements cited above and ensure that the safety limits are not exceeded, and therefore, fuel cladding integrity is maintained during any condition of normal operation and analyzed transients. As a result, there is reasonable assurance that the health and safety of the public is unaffected.

4.2 Precedence

The NRC has previously reviewed requests for TS changes in support of resolving the GE Part 21 concern similar to this proposed amendment request for LGS as documented in the following submittals and approved amendments.

On March 11, 2013, Northern States Power Company – Minnesota submitted a License Amendment request proposing to reduce the reactor steam dome pressure specified in Reactor Core Safety Limit Specification 2.1.1 (Reference 7). The NRC approved

Amendment 185 for the Monticello Nuclear Generating Plant on November 25, 2014 (Reference 8).

On March 24, 2014, Southern Nuclear Operating Company submitted an amendment request to revise the Edwin I. Hatch Plant Units 1 and 2 TS Section 2.1.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 (Reference 9). The NRC completed their review and issued Amendments 269 and 213 on October 20, 2014 (Reference 10).

On May 28, 2013, Entergy Operations, Inc. submitted an amendment request to revise the River Bend Station TS Section 2.1.1 to reflect a lower reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 (Reference 11). The NRC completed their review and issued Amendment 182 on December 11, 2014 (Reference 12).

On October 8, 2013, Entergy Nuclear Operations, Inc. proposed an amendment to modify the James A. FitzPatrick Nuclear Power Plant TS to reduce the reactor pressure associated with the Reactor Core Safety Limit in TS 2.1.1.1 and TS 2.1.1.2 (Reference 13). The NRC completed their review and issued Amendment 309 on February 9, 2015 (Reference 14).

On April 5, 2013, Entergy Nuclear Operations, Inc. proposed an amendment to modify the Pilgrim Nuclear Power Station TS to reduce the reactor pressure associated with the Reactor Core Safety Limit in TS 2.1.1 and TS 2.1.2 (Reference 15). The NRC completed their review and issued Amendment 242 on March 12, 2015 (Reference 16)

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 would reduce the reactor vessel steam dome pressure associated with the LGS TS Safety Limits (SLs) specified in TS 2.1.1 and TS 2.1.2 from 785 psig to 685 psig; increase the Trip Setpoint for TS Table 3.3.2-2, Function 1.c, Main Steam Line Pressure – Low, from ≥ 756 psig to ≥ 840 psig; and increase the Allowable Value for TS Table 3.3.2-2, Function 1.c, Main Steam Line Pressure – Low, from ≥ 736 psig to ≥ 821 psig. In addition, the associated TS Bases will be revised to reflect the above changes.

The proposed changes were identified as a result of GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03, “Potential to Exceed Low Pressure Technical Specification Safety Limit,” issued on March 29, 2005, and are being submitted based on the results of subsequent GE analyses that were sponsored by the Boiling Water Reactor Owners Group. These changes are valid for the NRC approved pressure range pertinent to the critical power correlations applied to the fuel types in use at LGS.

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 do not involve a significant increase in the probability or consequences of an accident previously evaluated because decreasing the reactor vessel steam dome pressure in TS Safety Limits 2.1.1 and 2.1.2 for reactor thermal power ranges and increasing the trip setpoint and allowable value for the main steam line low pressure isolation effectively expands the validity range for GEXL critical power correlation and the calculation of the minimum critical power ratio. The critical power ratio rises during the pressure reduction following the scram that terminates the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient. The reduction in the reactor vessel steam dome pressure value in the SL and the increase in the trip setpoint and the allowable value for the main steam line low pressure isolation provides adequate margin to accommodate the pressure reduction during the PRFO transient within the revised TS limit.

   The proposed changes do not alter the use of the analytical methods used to determine the safety limits that have been previously reviewed and approved by the NRC. The proposed changes are in accordance with an NRC approved critical power correlation methodology and do not adversely affect accident initiators or precursors.

   The proposed changes do not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the applicable acceptance limits. The proposed changes are consistent with the safety analysis and resultant consequences.

   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 create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed reduction in the reactor vessel steam dome pressure value in the safety limit in conjunction with the increase in the trip setpoint and the allowable value for the main steam line low pressure isolation reflects a wider range of applicability for the GEXL critical power correlation which is approved by the NRC for both GE14 and GNF2 fuel types in LGS reactor cores.

In addition, no new failure modes are being introduced. There are no changes in the method by which any plant systems perform a safety function. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

The proposed changes do not introduce any new accident precursors, nor do they involve any changes in the methods governing normal plant operation. The proposed changes do not alter the outcome of the safety analysis.

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 margin of safety is established through the design of the plant structures, systems, and components, and through the parameters for safe operation and setpoints for the actuation of equipment relied upon to respond to transients and design basis accidents. Evaluation of the 10 CFR Part 21 condition by General Electric determined that, since the critical power ratio improves during the PRFO transient, there is no impact on the fuel safety margin, and therefore, there is no challenge to fuel cladding integrity. The proposed changes do not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

The proposed changes are consistent with the applicable NRC approved critical power correlation for the fuel designs in use at LGS. The proposed changes do not alter the manner in which the safety limits are determined.

The reduction in value of the reactor vessel steam dome pressure safety limit and the increase in the trip setpoint and allowable value for the main steam line low pressure isolation provides adequate margin to accommodate the pressure reduction during the PRFO transient within the revised TS limit.

Therefore, the proposed changes do not involve a significant reduction in any 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. GE Energy - Nuclear, 10 CFR Part 21 Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit", March 29, 2005.

2. NEDC-33743P, Revision 0, "BWR Owners' Group Reload Analysis and Core Management Committee SC05-03 Analysis Report," dated April 2012.

3. NEDC-33292P, Revision 3, "GEXL17 Correlation for GNF2 Fuel," Global Nuclear Fuel, June 2009.

4. NEDC-33270P, Revision 3, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," Global Nuclear Fuel, March 2010.

5. NEDE-24011-P-A, Revision 17, "General Electric Standard Application for Reactor Fuel (GESTAR II)," Global Nuclear Fuel, dated September 2010.

6. NEDC-32851P-A, Revision 5, "GEXL14 Correlation for GE14 Fuel," Global Nuclear Fuel, April 2011.

7. Letter from John C. Grubb (Northern States Power Company – Minnesota) to U.S. NRC, "License Amendment Request: Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits," dated March 11, 2013. (ADAMS Accession No. ML13074A811)

8. Letter from Terry A. Beltz (U.S. NRC) to Karen D. Fili (Northern States Power Company – Minnesota), "Monticello Nuclear Generating Plant – Issuance of Amendment to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits (TAC No. MF1054)," dated November 25, 2014. (ADAMS Accession No. ML14281A318)

9. Letter from C. R. Pierce (Southern Nuclear Operating Company) to U.S. NRC, "License Amendment Request to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits," dated March 24, 2014. (ADAMS Accession No. ML14084A201)

10. Letter from Robert Martin (U.S. NRC) to C. R. Pierce (Southern Nuclear Operating Company), "Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Issuance of Amendments Reducing the Reactor Steam Dome Pressure in the Reactor Core Safety Limits (TAC Nos. MF3722 and MF3723)," dated October 20, 2014. (ADAMS Accession No. ML14276A634)

11. Letter from Eric W. Olson (Entergy Operations, Inc.) to U.S. NRC, "License Amendment Request Changes to Technical Specification 2.1.1, 'Reactor Core SLs'," dated May 28, 2013.

12. Letter from Alan B. Wang (U.S. NRC) to Vice President, Operations (Entergy Operations, Inc.), "River Bend Station, Unit 1 – Issuance of Amendment Re: Technical Specification 2.1.1, 'Reactor Core SLs' (TAC No. MF1948)," dated December 11, 2014.

13. Letter from Lawrence M. Coyle (Entergy Nuclear Operations, Inc.) to U.S. NRC, "Application to Revise Technical Specifications for Technical Specification Low Pressure Safety Limit," dated October 8, 2013.

14. Letter from Douglas V. Pickett (U.S. NRC) to Vice-President, Operations (Entergy Nuclear Operations, Inc.), "James A FitzPatrick Nuclear Power Plant – Issuance of Amendment Re: Application to Revise Technical Specifications for Technical Specification Low Pressure Safety Limit (TAC No. MF2897)," dated February 9, 2015.

15. Letter from Robert G. Smith (Entergy Nuclear Operations, Inc.) to U.S. NRC, "Proposed License Amendment: Revision to Technical Specifications (TS) 2.1, Safety Limits to Resolve Pressure Regulatory Fail-Open (PRFO) Transient Reported by General Electric Nuclear Energy in Accordance with 10 CFR 21.21(d)," dated April 5, 2013.

16. Letter from Nadiyah S. Morgan (U.S. NRC) to John A. Dent, Jr. (Entergy Nuclear Operations, Inc.), "Pilgrim Nuclear Power Station - Issuance of Amendment Regarding Safety Limits to Resolve Pressure Regulator Fail-Open Transient License Amendment Request (TAC NO. MF1382), dated March 12, 2015.

ATTACHMENT 2

License Amendment Request

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

Proposed Revision to Technical Specifications in Response to
GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03

Markup of Proposed Technical Specifications Pages

Unit 1 TS Pages
2-1
3/4 3-18

Unit 2 TS Pages
2-1
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2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than ~~785~~ 685 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than ~~785~~ 685 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 for two recirculation loop operation and shall not be less than 1.12 for single recirculation loop operation with the reactor vessel steam dome pressure greater than ~~785~~ 685 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.09 for two recirculation loop operation or less than 1.12 for single recirculation loop operation and the reactor vessel steam dome pressure greater than ~~785~~ 685 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with the reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level 1) Low, Low - Level 2 2) Low, Low, Low - Level 1	 ≥ - 38 inches* ≥ - 129 inches*	 ≥ - 45 inches ≥ - 136 inches
b. DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	840 ≥ ~~756~~ psig	821 ≥ ~~736~~ psig
d. Main Steam Line Flow - High	≤ 122.1 psid	≤ 123 psid
e. Condenser Vacuum - Low	10.5 psia	≥10.1 psia/≤ 10.9 psia
f. Outboard MSIV Room Temperature - High	≤ 192°F	≤ 200°F
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	≤ 165°F	≤ 175°F
h. Manual Initiation	N.A.	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 75 psig	≤ 95 psig
c. Manual Initiation	N.A.	N.A.

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

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2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than ~~785~~ psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

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ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than ~~785~~ psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 for two recirculation loop operation and shall not be less than 1.12 for single recirculation loop operation with the reactor vessel steam dome pressure greater than ~~785~~ psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

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ACTION:

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With MCPR less than 1.09 for two recirculation loop operation or less than 1.12 for single recirculation loop operation and the reactor vessel steam dome pressure greater than ~~785~~ psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATION CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Low - Level 2	≥ - 38 inches*	≥ - 45 inches
2) Low, Low, Low - Level 1	≥ - 129 inches*	≥ - 136 inches
b. DELETED	DELETED	DELETED
c. Main Steam Line Pressure - Low	≥ ~~756~~ $\boxed{840}$ psig	≥ ~~736~~ $\boxed{821}$ psig
d. Main Steam Line Flow - High	≤ 122.1 psid	≤ 123 psid
e. Condenser Vacuum - Low	10.5 psia	≥10.1 psia/≤ 10.9 psia
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a. Reactor Vessel Water Level Low - Level 3	≥ 12.5 inches*	≥ 11.0 inches
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c. Manual Initiation	N.A.	N.A.

ATTACHMENT 3

License Amendment Request

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

Proposed Revision to Technical Specifications in Response to
GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03

Markup of Proposed Technical Specifications Bases Pages
(Information Only)

Unit 1 TS Bases Page
B 2-1

Unit 2 TS Bases Page
B 2-1

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that more than 99.9% of the fuel rods avoid transition boiling. Meeting the Safety Limit can be demonstrated by analysis that confirms less than 0.1% of fuel rods in the core are susceptible to transition boiling or by demonstrating that the MCPR is not less than the values specified in Specification 2.1.2 for two recirculation loop operation and for single recirculation loop operation. Less than 0.1% of fuel rods in transition boiling and MCPR greater than the values specified for two recirculation loop operation and for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

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The use of the (GEXL) correlation is not valid for all critical power calculations at pressures below ~~785~~ psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/h, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lb/h. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below ~~785~~ psig is conservative.

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2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principle barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that more than 99.9% of the fuel rods avoid transition boiling. Meeting the Safety Limit can be demonstrated by analysis that confirms less than 0.1% of fuel rods in the core are susceptible to transition boiling or by demonstrating that the MCPR is not less than the values specified in Specification 2.1.2 for two recirculation loop operation and for single recirculation loop operation. Less than 0.1% of fuel rods in transition boiling and MCPR greater than the values specified for two recirculation loop operation and for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

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