Exelon Generation.

200 Exelon Way
Kennett Square, PA 19348

www.exeloncorp.com

10 CFR 50.90

NMP2L2584

December 13, 2016

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

Nine Mile Point Nuclear Station, Unit 2
Renewed Facility Operating License No. NPF-69
NRC Docket No. 50-410

Subject: License Amendment Request - Proposed Revision to Technical Specification 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) requests an amendment to the Technical Specifications, Appendix A, of Renewed Facility Operating License No. NPF-69 for Nine Mile Point Nuclear Station, Unit 2 (NMP2).

The proposed amendment would revise the NMP2 Technical Specification (TS) Safety Limit (SL) for TS SL 2.1.1.1, TS SL 2.1.1.2 from 785 psig to 700 psia, and TS Table 3.3.6.1-1, function 1b, Main Steam Line Pressure - Low, from ≥746 psig to ≥814 psig. In addition, the associated TS Bases will be revised to reflect the above changes.

The implementation of this amendment will result in an increased Low Pressure Isolation Setpoint Allowable Value that, in turn, will result in earlier main steam line isolation. The revised main steam line low pressure isolation capability and the revised SL addresses the GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," issued on March 29, 2005. This change ensures the plant remains within the TS SLs in the event of a Pressure Regulator Failure Maximum Demand (Open) transient.

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

Attachment 1 provides the Evaluation of Proposed Changes. Attachment 2 provides the Proposed Technical Specification Marked-Up Pages. Attachment 3 provides the Proposed Technical Specifications Bases Marked-Up Pages (for information only). Attachment 4 provides the setpoint calculation.

The proposed changes have been reviewed by the NMP Plant Operations Review Committee in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon requests approval of the proposed amendment by February 28, 2018. Once approved, the amendment shall be implemented within 120 days.

There are no regulatory commitments contained in this request.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is transmitting a copy of this application and its attachments to the designated State Officials.

Should you have any questions concerning this submittal, please contact Ron Reynolds at (610) 765-5247.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 13th day of December 2016.

Respectfully,

David T. Gudger for

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

Attachments:
1) Evaluation of Proposed Change
2) Proposed Technical Specification Marked-Up Pages
3) Proposed Technical Specification Bases Marked-Up Pages
4) Evaluation of the Effect of Changes in Technical, Specification Surveillance Intervals to, Accommodate a 24-Month Fuel Cycle, Nine Mile Point 2 Appendix C

cc:		
	USNRC Region I, Regional Administrator	w/attachments
	USNRC Senior Resident Inspector, NMP	w/attachments
	USNRC Project Manager, NMP	w/attachments
	A. L. Peterson, NYSERDA	w/attachments

ATTACHMENT 1

License Amendment Request

Nine Mile Point Nuclear Station Unit 2

Docket No. 50-410

<u>EVALUATION OF PROPOSED CHANGES</u>

CONTENTS

SUBJECT: **Proposed Revision to Technical Specification 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) requests an amendment to the Technical Specifications (TS), Appendix A, of Renewed Facility Operating License No. NPF-69 for Nine Mile Point Nuclear Station, Unit 2 (NMP2).

The proposed changes would reduce the reactor steam dome pressure specified in TS 2.1.1.1 from < 785 psig to < 700 psia and the reactor steam dome pressure listed in TS 2.1.1.2 from ≥ 785 psig to ≥ 700 psia. Also, the Allowable Value (AV) on TS Table 3.3.6.1-1, Function 1b, Main Steam Line Pressure–Low, is proposed to change from ≥ 746 psig to ≥ 814 psig. Finally, the nominal trip setpoint identified in the Bases will be revised from ≥766 psig to ≥821 psig.

The proposed changes were identified as a result of GE Energy-Nuclear (GE) 10 CFR 21 Safety Communication SC05-03, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," issued on March 29, 2005 (Reference 1), on an event that can, in the limiting case, result in a TS Safety Limit (SL) non-compliance on the Exelon BWR plants. The event is a Pressure Regulator Failure - Maximum Demand (Open) (PRFO) from rated power conditions. In this event, the reactor depressurizes to below the Main Steam Isolation Valve (MSIV) low pressure isolation setpoint prior to initiation of a reactor scram on MSIV position resulting in power above 23% for a few seconds while dome pressure may be below the TS SL.

2.0 DETAILED DESCRIPTION

On March 29, 2005, GE issued a Safety Communication (SC05-03) (Reference 1) 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. GE informed the affected licensees that their recent code calculations confirmed that during the PRFO transient, reactor pressure could fall below the TS SL. Depending upon the Low Pressure Isolation Setpoint (LPIS), the margin to the low pressure TS SL may not be adequate. This condition does not challenge the fuel cladding integrity or constitute a safety hazard as determined by GE in Reference 1. However, Exelon is proposing to revise the reactor vessel steam dome pressure specified in TS SLs 2.1.1.1 and 2.1.1.2 to 700 psia supported by the expanded GEXL14 correlation applicability range for GE14 fuel (Reference 3) and the GEXL17 correlation for GNF2 (Reference 4). Both fuel types are currently in the NMP2 reactor.

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 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 Reference 2 methodology considers the most limiting plant configuration and operating conditions for evaluating the effect of SC05-03 using a scaling approach. This scaling methodology is utilized to assess the adequacy of the NMP2 current LPIS setting.

Based on the results of the studies documented in Reference 2, Exelon has determined that changing the pressure limit in TS SL 2.1.1.1 to <700 psia and TS SL 2.1.1.2 to ≥700 psia as permitted by References 3 and 4, and increase the AV for the Main Steam Line Pressure–Low to ≥ 814 psig provides adequate margin for the PRFO transient, such that the reactor dome pressure will remain above the proposed revision to the TS SL.

Proposed Technical Specification Changes:

- Reduce the reactor vessel steam dome pressure limit specified in TS SL 2.1.1.1 from < 785 psig to < 700 psia,
- Reduce the reactor vessel steam dome pressure limit specified in TS SL 2.1.1.2 from ≥ 785 psig to ≥ 700 psia,
- Increase the Allowable Value for TS Table 3.3.6.1-1, Function 1.b, Main Steam Line Pressure - Low, from ≥ 746 psig to ≥ 814 psig.

The specific changes to Technical Specifications and associated Bases pages are shown on Attachments 2 and 3, respectively. The Bases pages are being provided for information only. Attachment 4 contains the Main Steam Line (MSL) low pressure isolation nominal trip setpoint and AV calculation.

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. 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. When the MSL low pressure 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 safety limit of 700 psia while core thermal power is still above 23% of rated thermal power. With an initial condition that is restricted by the Minimum Critical Power Ratio (MCPR) Operating Limit 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 exist.

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 the plant specific value of rated thermal power specified in TS SL 2.1.1.1. 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 NMP2 TS SLs 2.1.1.1 and 2.1.1.2 be changed from <785 psig to <700 psia. In addition, MSIV LPIS allowable value on TS Table 3.3.6.1-1, Function 1b, Main Steam Line Pressure–Low, will be increased from ≥ 746 psig to ≥ 814 psig.

Safety Limit 2.1.1.1 and 2.1.1.2 Changes

TS SLs are specified to ensure that acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, or 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 NMP2 TS SL 2.1.1.1 requires that thermal power shall be ≤ 23% Rated Thermal Power (RTP) when reactor steam dome pressure is < 785 psig or core flow is < 10% of rated core flow. NMP2 TS SL 2.1.1.2 states with the reactor steam dome pressure ≥785 psig and core flow ≥ 10% rated core flow, the MCPR shall be ≥1.15 for two recirculation loop operation or ≥1.15 for single recirculation loop operation. These SLs were introduced to ensure the validity of MCPR calculations when power is >23% and the reactor pressure is within the validity range of the GEXL correlations.

GE has updated the validity range of GEXL14 and GEXL17 Correlations (References 3 and 4), which allows the pressure to be reduced to 700 psia from 785 psig. In addition, the proposed change to the LPIS AV from 746 psig to 814 psig is calculated based on a revised LPIS Analytical Limit (AL) determined utilizing the BWROG approach documented in Reference 2. The combination of the lower SL and the higher AV provides a wider pressure range for transients to demonstrate compliance with MCPR limits. Therefore, the proposed change offers a greater pressure range for a PRFO transient than what is currently available.

NMP2 currently has GE14 and GNF2 fuel in the reactor core. GNF2 fuel was introduced to the core during the 2016 refueling outage. The lower bound limit of 700 psia for the GEXL14 correlation is documented and justified in GE Topical Report NEDC-32851P-A for GE14 fuel (Reference 3). This topical report has been reviewed and approved by the NRC. The GEXL17 correlation is documented and justified in NEDC-33292P (Reference 4) for GNF2 Fuel. This lower bound limit is discussed in NEDC-33292P and is referenced in NEDC-33270P (Reference 5). 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 6). Therefore, the use of 700 psia as a lower bound limit for GNF2 fuel has been approved by the NRC for use per GE Topical Report NEDE-24011-P-A by reference.

Use of GEXL 17 does not change the thermal power limit (23%) corresponding to 10% rated core flow. The 23% rate power limit is conservative value which provides significant margin between fuel assembly operating power and critical power. The basic GEXL correlation is supported by ATLAS and Stern test data with GEXL17 coefficients determined from Stern testing of the GNF2 fuel design. TS Bases includes the statement "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." Therefore, since 700 psia is within the range 14.7 to 800 psia, the TS Bases statement remains unchanged.

The proposed change in NMP2 TS 2.1.1.2 which specifies the SL on the MCPR expands the range of applicability of the SL on the MCPR to a low pressure established by the 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 pressure SL. The low pressure SL protects transition boiling at the reactor fuel cladding. The condition 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, operating margin is increased due to the proposed change to ensure NMP2 cannot enter into an unanalyzed condition during a PRFO event such as is potentially possible with the current low pressure SL.

The revised AV calculated at 814 psig based on the new AL of 805 psig is higher than the current AV of 746 psig and will result in earlier MSL isolation to terminate a rapid depressurization event. Following the BWROG methodology in Reference 2, the results most applicable to the NMP2 plant configuration was used to determine the new AL. This required scaling up the results from Table 5 in Reference 2 for increased LPIS AL of 805 psig to meet the acceptance criterion. The increased LPIS AL of 805 psig was used as input to revise the setpoint calculation for NMP2. Based on this new AL, the associated changes to the AV and actual trip setpoint were established as part of the setpoint calculation update (see Attachment 4) and established a new AV, which was incorporated in the license amendment request for the NRC review and approval.

This event while the reactor is near full power could result in undesirable effects such as differential pressures of sufficient magnitude across the channels around some fuel bundles to cause mechanical deformation of channel walls. The steam pressure at the turbine inlet is monitored to forestall theses effects. The proposed MSIV LPIS trip setting, calculated at 821 psig based on the new AL of 805 psig, is far enough below normal turbine throttle pressure to prevent spurious isolation, yet high enough to provide timely detection of a pressure controller malfunction. In addition, this isolation function is not required to satisfy any of the safety design bases for this system.

The PRFO event remains non-limiting for thermal limit impact with respect to the LPIS change. Also, the change in LPIS AV will not affect the outcome of the limiting PRFO Anticipated Transient Without Scram (ATWS) analysis. For the NMP2 ATWS analysis, the increased LPIS setpoint will result in an earlier MSL isolation and Recirculation Pump Trip (RPT). As a result, the analysis with its margins to the ATWS acceptance criteria remains applicable with respect to the sepoint change.

Conclusion

Exelon has determined that reducing the reactor vessel steam dome pressure limit specified in TS SLs 2.1.1.1 and 2.1.1.2, in conjunction with increasing the AV and trip setpoint specified in TS Table 3.3.6-1, Function 1.b for the main steam line low pressure isolation, adequately mitigate the PRFO transient event, such that the reactor vessel steam dome pressure will remain above the proposed revision to the TS SLs. The combination of the lower TS SLs and the higher LPIS trip setpoint and AV provides a wider pressure range for transients while maintaining compliance with MCPR limits. Therefore, the proposed change offers a greater pressure range 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 lower-bound pressure for the GE14 fuel in GEXL14 and for the GNF2 fuel in GEXL17 comprising the NMP2 reactor core.

Therefore, the proposed changes resolve the 10 CFR Part 21 condition concerning the potential to exceed reactor core SL 2.1.1.1 during a PRFO transient reported in Reference 1.

The use of 700 psia as the steam dome pressure limit for TS 2.1.1.1 is supported by the CPR correlations in use for NMP2. The minimum steam dome pressure resulting from a PRFO event is demonstrated to be above 700 psia using Reference 2 information. Revising the Reactor Core Safety Limits 2.1.1.1 reactor steam dome pressure from 785 psig to 700 psia in conjunction with the change to LPIS AL resolves the 10 CFR Part 21 condition concerning the potential to exceed Reactor Core Safety Limit 2.1.1.1 during a PRFO transient reported in Reference 1.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, provides criteria for Emergency Core Cooling System performance and 10 CFR 50.36, Technical Specifications, requires safety system settings to ensure the integrity of the reactor pressure boundary during normal and abnormal operations and to mitigate transient and accident conditions. The proposed change in the reactor dome pressure limit in TS SL 2.1.1.1 and TS SL 2.1.1.2 and the proposed change in the main steam line low pressure AV follows the requirements cited above and ensures the fuel cladding integrity.

Regulatory Guide 1.105, Revision 2, "Instrument Setpoints for Safety-Related Systems," describes a method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

4.2 Precedent

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 NMP2 as documented in the following 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 27, 2016, the NRC completed their review and issued amendments 306 and 310 for Peach Bottom Atomic Power Station, Units 2 and 3 (Reference 15).

On May 11, 2016, the NRC completed their review and issued amendment 209 for Clinton Nuclear Power Station Unit 1, amendments 250 and 243 for Dresden Nuclear Power Station, Units 2 and 3, and amendments 262 and 257 for Quad Cities Nuclear Power Station, Units 1 and 2 (Reference 16).

On January 15, 2016, as supplemented by letters dated April 19, 2016, May 9, 2016, and June 21, 2016, Exelon Generation Company submitted an amendment request to revise the Limerick Generating Station (LGS) reactor steam dome pressure associated with the Reactor Core Safety Limit in the TS. The NRC completed their review and issued amendments 222 and 183 on November 21, 2016 (Reference 17).

4.3 <u>No Significant Hazards Consideration</u>

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. **Does the proposed amendment 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 dome pressure in TS SL 2.1.1.1 and TS SL 2.1.1.2 for reactor RTP ranges and increasing the AV for the Main Steam Line Pressure-Low on TS Table 3.3.6.1-1, Function b, effectively expands the range of applicability for GEXL correlation and the calculation of MCPR. The CPR rises during the pressure reduction following the scram that terminates the PRFO transient. The reduction in the reactor dome pressure value in the SL from 785 psig to 700 psia and the increase in the AV from ≥746 psig to ≥814 psig adequately accommodate the pressure reduction during the PRFO transient within the revised TS limit without compromising fuel integrity.

   The expanded GEXL correlation range supports NMP2 revised low pressure safety limit of 700 psia. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident or transient operating conditions.

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

2. **Does the proposed amendment 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 dome pressure value in the SL from 785 psig to 700 psia reflects a wider range of applicability for the GEXL correlation which is approved by the NRC for both GE14 currently in NMP2 and GNF2 fuels proposed for NMP2. The proposed changes do not involve physical changes to the plant or its operating characteristics. In addition, the increase in the AV for the MSL low pressure from ≥746 psig to ≥814 psig will result in the MSIV closure signal initiation at a higher temperature. As a result, no new failure modes are being introduced.

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

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

Response: No.

The proposed changes do not involve a significant reduction in a margin of safety because 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. The proposed change in reactor dome pressure SLs and the AV for the MSL low pressure ensures the safety margin is maintained, which protects the fuel cladding integrity during steady state operation, normal operational transients, or AOOs such as a depressurization transient, but does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. The proposed changes do not involve physical changes to the plant or its operating characteristics. The reduction in the reactor dome pressure value in the SL from 785 psig to 700 psia and the increase to the AV for the MSL low pressure provides added margin to accommodate the pressure reduction during the PRFO transient within the revised TS limit without compromising fuel integrity.

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 does not involve a 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 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit", March 29, 2005. (ADAMS Accession No. ML050950428).

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

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

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

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

6. NEDC-24011-P-A,, "General Electric Standard Application for Reactor Fuel," Global Nuclear Fuel.

7. Letter from John C. Grubb (Northern States Power Company - Minnesota) to US 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 (US 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 US 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 (US 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 US NRC, "License Amendment Request Changes to Technical Specification 2.1.1, 'Reactor Core SLs'," dated May 28, 2013.

12. Letter from Alan Wang (US 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 US NRC, "Application to Revise Technical Specifications for Technical Specification Low Pressure Safety Limit," dated October 8, 2013.

14. Letter from Douglas V. Pickett (US 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 R. E. Ennis (US NRC) to President and Chief Nuclear Officer (Exelon Nuclear), "Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendments Re: Reduce Steam Dome Pressure Specified in Reactor Core Safety Limits (CAC NOS. MF7184 AND MF7185)," dated April 27, 2016 (ADAMS Accession No.: ML 16064A150).

16. Letter from Blake Purnell, (US NRC) to President and Chief Nuclear Officer (Exelon Nuclear), "Clinton Power Station Unit1; Dresden Nuclear Power Station Units 2 and 3; and Quad Cities Nuclear Power Station Units 1 and 2 - Issuance of Amendments to Revise the Reactor Steam Dome Pressure in Technical Specifications 2.1.1, "Reactor Core SLs (CAC NOS. MF6640-MF6644)," dated May 11, 2016. (ADAMS Accession No.: ML 15231A097 and ML 16105A421).

17. Letter from R. Ennis (US NRC) to B. Hanson (Exelon), "Limerick Generating Station, Units 1 and 2 - Issuance of Amendments to Reduce Steam Dome Pressure Specified in Reactor Core Safety Limits (CAC NOS. MF7263 and MF7264)," dated November 21, 2016.

ATTACHMENT 2

License Amendment Request

Nine Mile Point Nuclear Station Unit 2
Docket No. 50-410

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

Proposed Technical Specification Marked-Up Pages

TS Pages

2.0-1

3.3.6.1-6

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

700 psia

2.1.1.1 With the reactor steam dome pressure < ~~785 psig~~ or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

700 psia

2.1.1.2 With the reactor steam dome pressure ≥ ~~785 psig~~ and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.8 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ ~~746~~ 814 psig
c. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 184.4 psid
d. Condenser Vacuum – Low	1,2(a), 3(a)	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 7.6 inches Hg vacuum
e. Main Steam Line Tunnel Temperature – High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 170.6°F
f. Main Steam Line Tunnel Differential Temperature – High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 71.7°F
g. Main Steam Line Tunnel Lead Enclosure Temperature – High	1,2,3	2 per area	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 175.6°F(b)
h. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA

(continued)

(a) With any turbine stop valve not closed.

ATTACHMENT 3

License Amendment Request

Nine Mile Point Nuclear Station Unit 2
Docket No. 50-410

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

Proposed Technical Specification Bases Marked-Up Pages
(for information only)

Bases Pages

B 2.0-2

B 2.0-3

B 2.0-5

B 3.3.6.1-9

BASES

BACKGROUND (continued)

reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR Safety Limit.

≥700 psia

<u>2.1.1.1 Fuel Cladding Integrity</u>

GE critical power correlations are applicable for all critical power calculations at pressures ≥ ~~785 psig~~ and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> 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 > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28×10^3 lb/hr, 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 > 28×10^3 lb/hr. Full scale ATLAS test

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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 > 50% RTP. Thus, a THERMAL POWER limit of 23% RTP for reactor pressure ~~≤ 785 psig~~ <700 psia is conservative. Additional information on low flow conditions is available in Reference 7.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. 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 damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio 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.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in References 3, 4 and 6. Reference 3 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and Reference 4 also provides the nominal values of the parameters used in the MCPR SL statistical analysis.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. GE Service Information Letter No. 516, Supplement 2, "Core Flow Indication in the Low-Flow Region," January 19, 1996.

3. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).

4. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2 (revision specified in the COLR).

5. 10 CFR 50.67, "Accident Source Term."

6. NEDC-33173-P-A, "Applicability of GE Methods to Expanded Operating Domains."

7. SIL No. 516 Supplement 2, January 19, 1996.

8. Clarification of SIL 516 S2 Recommendations Related to Technical Specifications for Low Pressure Conditions-003N8314 Revision 1

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure – Low (continued)

Function closes the MSIVs ~~prior to pressure decreasing below 766 psig, which~~ results in a scram ~~due to MSIV closure~~, thus reducing reactor power to < 23% RTP.)

during depressurization transient to maintain reactor steam dome pressure > 700 psia. The MSIV closure at normal trip setpoint of 821 psig

The MSL low pressure signals are initiated from four pressure transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure – Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value ~~was selected to be~~ is also high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure – Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 4).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow – High

Main Steam Line Flow – High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow – High Function is directly assumed in the analysis of the main steam line break (MSLB) accident (Ref. 6). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 50.67 limits.

There is a plant specific program that verifies that this instrument channel functions as required by verifying the As-Found and As-Left settings are consistent with those established by the setpoint methodology.

The MSL flow signals are initiated from 16 differential pressure transmitters that are connected to the four MSLs (the differential pressure transmitters sense differential

(continued)

ATTACHMENT 4

License Amendment Request

Nine Mile Point Nuclear Station Unit 2
Docket No. 50-410

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

Evaluation of the Effect of Changes in Technical,

Specification Surveillance intervals to,

Accommodate a 24-Month Fuel Cycle,

Nine Mile Point 2 Appendix C

Design Analysis | **Last Page No.:** [6] 5

Analysis No.: [1] NSSS168805000 **Revision:** [2] 01.00 Major ☐ Minor ☒

Title: [3] EVALUATION OF THE EFFECT OF CHANGES IN TECHNICAL, SPECIFICATION SURVEILLANCE INTERVALS TO, ACCOMMODATE A 24-MONTH FUEL CYCLE, NINE MILE POINT 2 APPENDIX C

EC/ECR No.: [4] ECP-15-000016 **Revision:** [5] 0000

Station(s): [7]	NMP	**Component(s):** [14]	
Unit No.: [8]	2	2MSS*PT20A	2MSS*PIS1020A
Discipline: [9]	I: Instrumentation & Control	2MSS*PT20B	2MSS*PIS1020B
Descrip. Code/Keyword: [10]	N/A	2MSS*PT20C	2MSS*PIS1020C
Safety/QA Class: [11]	SR	2MSS*PT20D	2MSS*PIS1020D
System Code: [12]	MSS		
Structure: [13]	N/A		

CONTROLLED DOCUMENT REFERENCES [15]

Document No.:	From/To	Document No.:	From/To
ECP-14-000910	N/A		

Is this Design Analysis Safeguards Information? [16] Yes ☐ No ☒ If yes, see SY-AA-101-106

Does this Design Analysis contain Unverified Assumptions? [17] Yes ☐ No ☒ If yes, ATI/AR#:

This Design Analysis SUPERCEDES: [18] N/A

Description of Revision (list changed pages when all pages of original analysis were not changed): [19]
Revise the Main Steam Line (MSL) Low Pressure Isolation Nominal Trip Setpoint (NTSP) and Allowable Value (AV) based on a new Analytic Limit (AL) of 805 psig.

Affected Page Nos.: Pages 10-1, 10-2, 10-7, 10-9F

	Print Name	Sign Name	Date
Preparer: [20]	J. R. Gilbert (S&L)/A. Emanuele (S&L)	[signature]	7/21/16

Method of Review: [21] Detailed Review ☒ Alternate Calculations (attached) ☐ Testing ☐

	Print Name	Sign Name	Date
Reviewer: [22]	L. W. Stahl (S&L)	LW Stahl	07-21-2016

Review Notes: [23] Independent review ☒ Peer review ☐

(For External Analyses Only)

	Print Name	Sign Name	Date
External Approver: [24]	A. A. Emanuele (S&L)	[signature]	8-2-16
Exelon Reviewer: [25]	Jeremy Rossman	[signature]	11/22/16

Independent 3rd Party Review Reqd? [26] Yes ☐ No ☒

	Print Name	Sign Name	Date
Exelon Approver: [27]			

On page 10-1, add the following references:

2.12 ECP-14-000910, Rev. 0000, ESR-14-000567 NMP-ESR (0000) - SC05-03 Assessment for Nine Mile Point Unit 2 (NMP2)

2.13 ECP-15-000016, Rev. 0000, NMP2 Design Change - Increase MSL LPIS Setpoint

On page 10-2, Revise Section 7.6 as follows:

FROM:

7.6 Analytical and Operational Limits
per Reference 2.4, AL = 720 psig
OL = 866 psig

TO:

7.6 Analytical and Operational Limits
per Reference 2.12, AL = 805 psig
OL = 866 psig

SETPOINT SUMMARY SHEET

TRANSMITTER I.D.:	NMB22-N076A-D
LAST INSTR I.D.:	NMB22-N676A-D
SETPOINT FUNCTION:	MSL PRESSURE LOW (NS4)
TYPE OF LOGIC:	One out of Two, Twice

INPUT DATA

ANALYTICAL LIMIT:	$\boxed{\cancel{720}}$	805
OPERATIONAL LIMIT:	866	REVISE
PROCESS MEASUREMENT ACCURACY:	0	
PRIMARY ELEMENT ACCURACY:	0	
CHANNEL CALIBRATION ACCURACY:	6	
CHANNEL INSTRUMENT ACCURACY (NORMAL):	5.230489	
CHANNEL INSTRUMENT ACCURACY (TRIP):	9.143742	
CHANNEL INSTRUMENT DRIFT:	13.280155	
(FOR SURVEILLANCE INTERVAL (MONTHS)):	30	
LER AVOIDANCE BASIS:	Multiple	

REVISE

CALCULATION RESULTS (IN UNITS OF PSIG)		
ALLOWABLE VALUE:	$\boxed{\cancel{728.995308}}$	813.995308
NOMINAL TRIP SETPOINT (LICENSING):	$\boxed{\cancel{715.365887}}$	820.265887
NOMINAL TRIP SETPOINT (TRIP AVOIDANCE):	856.865083	

INSTRUMENT LOOP CALCULATION SHEET

UNITS (PROCESS): PSIG	/------ACCURACY------\ NORMAL	TIME OF TRIP	DRIFT (30 MO.)
TRANS: NMB22-N076A-D	3.332867	8.207192	12.14753
LAST: NMB22-N676A-D	4.031130	4.031130	5.366566
TOTAL LOOP:	5.230489	9.143742	13.280155

SETPOINT CALCULATION, COMPARISON TO EXISTING TECH SPEC):
NTSP = AL +/- (1.645 / 2) * SQRT(SQ(AT) + SQ(CL) + SQ(DL) + SQ(PMA) + SQ(PEA))
+/- BIAS

NTSP = ~~734.150121~~ (WITHOUT LER AVOIDANCE CONSIDERATIONS)
819.1501121 REVISE

ALLOWABLE VALUE CALCULATION:
AV = AL +/- (1.645 / 2) * SQRT(SQ(AT) + SQ(CL) + SQ(PMA) + SQ(PEA))
+/- BIAS

AV = ~~728.995308~~ 813.995308 REVISE

CHECK OF LER AVOIDANCE CONSIDERATIONS:
FOR LER AVOIDANCE --
NTSP = AV +/- (Z / 2) * SQRT(SQ(AN) + SQ(CL) + SQ(DL))

Z = 0.81

820.265887 REVISE
NTSP = ~~735.265887~~ (VALUE NEEDED FOR ADEQUATE LER AVOIDANCE MARGIN)

**USE NTSP = ~~735.265887~~ 820.265887 REVISE

EVALUATION OF TRIP AVOIDANCE CONDITIONS:
FOR TRIP AVOIDANCE --:
NTSPTA = OL +/- (Z / 2) * SQRT(SQ(AN) + SQ(CL) + SQ(DL) + SQ(PMA) + SQ(PEA))
Z = 1.18

**NTSPTA = 856.865083 (LIMITING NTSP FOR TRIP AVOIDANCE)