Exelon Generation.

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10 CFR 50.90

NMP1L3095
RA-16-060

August 1, 2016

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

Nine Mile Point Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-63
NRC Docket No. 50-220

Oyster Creek Nuclear Generating Station
Renewed Facility Operating License No. DPR-16
NRC Docket No. 50-219

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), proposes a change to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-16 for Oyster Creek Nuclear Generating Station (OCNGS) and Renewed Facility Operating License No. DPR-63 for Nine Mile Point Nuclear Station, Unit 1 (NMP1).

The proposed change revises OCNGS TS Section 2.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.A and 2.1.B. The proposed change also revises NMP1 TS Section 2.1.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.1.a and 2.1.1.b. In addition, the associated TS Bases for each plant will be revised to reflect the above changes.

The proposed change was identified as a result of 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, and is being submitted based on the results of subsequent GE analyses that were sponsored by the Boiling Water Reactor Owners Group. This change is valid for the NRC approved pressure range pertinent to the critical power correlations applied to the fuel types in use at OCNGS and NMP1 when the License Amendment Request (LAR) is implemented.

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

The proposed change has been reviewed by the OCNGS and NMP1 Plant Operations Review Committees 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 change. Attachment 2 provides a copy of the marked up TS pages for OCNGS and NMP1 that reflect the proposed change. Attachment 3 provides a copy of the marked up TS Bases pages for OCNGS and NMP1 that reflect the proposed change (for information only).

Exelon requests approval of the proposed amendments by August 1, 2017. Upon NRC approval, the amendment shall be implemented within 60 days of issuance.

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

If you have any questions or require additional information, please contact Stephanie J. Hanson at 610-765-5143 or Ronnie Reynolds at 610-765-5247.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 1st day of August 2016.

Respectfully,

James Barstow

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

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

cc:
USNRC Region I, Regional Administrator
USNRC Senior Resident Inspector, OCNGS
USNRC Project Manager, OCNGS
USNRC Senior Resident Inspector, NMP
USNRC Project Manager, NMP
Manager, Bureau of Nuclear Engineering, New Jersey Department of Environmental Protection
Mayor of Lacey Township, Forked River, NJ
A. L. Peterson, NYSERDA

ATTACHMENT 1

EVALUATION OF PROPOSED CHANGE

License Amendment Request

Oyster Creek Nuclear Generating Station
Nine Mile Point Nuclear Station, Unit 1

Docket No. 50-219
Docket No. 50-220

Subject: **License Amendment Request to Reduce Steam Dome Pressure Specified in the Reactor Core Safety Limits**

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), proposes a change to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License No. DPR-16 for Oyster Creek Nuclear Generating Station (OCNGS) and Renewed Facility Operating License No. DPR-63 for Nine Mile Point Nuclear Station, Unit 1 (NMP1).

The proposed change will revise OCNGS TS Section 2.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.A and 2.1.B. Furthermore, the proposed change revises NMP1 TS Section 2.1.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.1.a and 2.1.1.b. This change to TS Section 2.1 for OCNGS and TS Section 2.1.1 for NMP1 became necessary as a result of GE Energy - Nuclear (GE) Part 21 report SC05-03, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure TS Safety Limit." This change is consistent with the Nuclear Regulatory Commission (NRC) approved pressure range for the critical power correlations applied to the fuel types in use at OCNGS and NMP1 when the License Amendment Request (LAR) is implemented.

2.0 DETAILED DESCRIPTION

On March 29, 2005, GE submitted a 10 CFR Part 21 notification (Reference 1) identifying that, as a result of applying improved methodologies for licensing basis transient analyses, the anticipated operational occurrence (AOO) Pressure Regulator Failure-Maximum Demand (Open) (PRFO) had been identified as an event in which Reactor Core Safety Limit 2.1.B for OCNGS and 2.1.1.b for NMP1 could potentially be violated. GE has determined that this does not challenge the fuel cladding integrity. However, there is a potential vulnerability for the PRFO transient event to result in a condition in which TS SL 2.1.B for OCNGS and 2.1.1.b for NMP1 may be exceeded. The Minimum Critical Power Ratio (MCPR) Safety Limit specified in Reactor Core Safety Limit 2.1.A for OCNGS and 2.1.1.a for NMP1 is established to protect fuel cladding integrity. This change supports operation of OCNGS and NMP1 with GNF2 fuel.

GE indicated that the approved methodology for modeling had evolved from REDY, to ODYN, to TRACG. Reactor depressurization transients, such as PRFO, are non-limiting for fuel cladding integrity because critical power ratio (CPR) increases during the PRFO event, and are not typically included in the scope of cycle-specific reload evaluations. GE determined that REDY, ODYN, and TRACG all show the CPR increasing during the PRFO transient, and hence fuel cladding integrity not being challenged, and that the difference in reactor level swell predicted by REDY, versus ODYN and TRACG, can impact the predicted plant response to the PRFO.

GE indicated within the 10 CFR Part 21 notification letter that no clear compensatory action can be defined to appropriately mitigate this vulnerability, and since the condition does not challenge the physical barrier that the Safety Limit intends to protect (i.e., the fuel cladding integrity), there is no safety basis for a compensatory action. While this condition had been determined by GE to not involve an actual safety hazard, the potential for violation of a Reactor Core Safety Limit had been identified, and restoration to comply with the safety limit is required for the PRFO event. As a consequence, OCNGS and NMP1 are revising the reactor steam dome pressure TS Safety Limit consistent with the NRC approved pressure range of critical power correlations for the fuel types in use at OCNGS and NMP1 when the LAR is implemented.

The proposed changes to the OCNGS TS are summarized below:

- The proposed change would revise the reactor steam dome pressure value in TS 2.1.A and TS 2.1.B from 800 psia to 700 psia.

The proposed changes to the NMP1 TS are summarized below:

- The proposed change would revise the reactor steam dome pressure value in TS 2.1.1.a and TS 2.1.1.b from 800 psia to 700 psia.

The marked up pages that reflect the proposed changes are provided in Attachment 2 (TS pages) and Attachment 3 (TS Bases pages - information only).

3.0 TECHNICAL EVALUATION

Excessive thermal overheating of the fuel rod cladding can result in cladding damage and the release of fission products. In order to protect the cladding against thermal overheating due to boiling transition, the Safety Limits (SL) in OCNGS TS 2.1 and in NMP1 TS 2.1.1 were established. 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 violated.

The fuel cladding integrity in the OCNGS and NMP1 reactors is maintained through application of the safety limits in TS 2.1.A and TS 2.1.B (OCNGS) and TS 2.1.1.a and TS 2.1.1.b (NMP1). When reactor pressure and core flow are greater than the specified values, Reactor Core Safety Limit 2.1.A in the OCNGS TS and 2.1.1.a in the NMP1 TS prohibits operation with a MCPR less than the Safety Limit MCPR to assure the fuel cladding integrity.

Currently, the OCNGS TS SL 2.1.B states that when the reactor steam dome pressure is less than 800 psia or when core flow is less than 10% of rated core flow, the reactor core thermal power shall be less than or equal to 25% rated thermal power (RTP). Furthermore, the NMP1 TS SL 2.1.1.b states that when the reactor steam dome pressure is less than or equal to 800 psia or when core flow is less than 10% of rated core flow, the reactor core thermal power shall be less than or equal to 25% RTP. This SL was introduced to preclude the need for CPR calculations when reactor steam dome pressure and core flow are less than the specified values while ensuring that reactor power would remain well below the fuel assembly critical power for the conditions in which CPR calculations are not performed.

Reactor depressurization transients, such as PRFO, are non-limiting for fuel cladding integrity because CPR increases during the event, and they are not typically included in the scope of reload evaluations. Previous evaluations by GE using the REDY model predicted that reactor water level would swell during a PRFO transient; the depressurization would be terminated by a high level turbine trip. However, level swell is difficult to predict and the level swell portion of transient models have larger uncertainties than other portions of the transient models. Recent evaluations by GE with the improved transient models have determined that the reactor level swell may not be sufficient to reach the high level trip, in which case the depressurization could be terminated by Main Steam Isolation Valve (MSIV) closure at the low-pressure isolation setpoint (LPIS). Depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor steam dome pressure could decrease to below 800 psia for a few seconds while thermal power exceeds 25% RTP, which would exceed the conditions in TS SL 2.1.B for OCNGS and TS SL 2.1.1.b for NMP1. This issue was identified in Reference 1. OCNGS LPIS analytical limit is 825 psig and NMP1 LPIS analytical limit is 834.2 psig.

In response to Reference 1, the BWR Owners' Group commissioned General Electric Hitachi (GEH) SC05-03 analysis documented in NEDC-33743P, "BWR Owners' Group Reload Analysis and Core Management Committee SC05-03 Analysis Report," Revision 0 (Reference 2). Exelon has previously provided a copy of this proprietary document to the NRC via the Reference 12 transmittal letter. The scaling method in the Reference 2 report, using the results for a maximum MSIV closure time of 10 seconds was utilized to assess the adequacy of the OCNGS and NMP1 current LPIS setting for the SC05-03 issue. This assessment was further confirmed via a separate evaluation by GEH. Results of this assessment show that the current LPIS setting at OCNGS and NMP1 is adequate to prevent reactor dome pressure from falling below 700 psia when above 25% RTP during a PRFO event. The CPR correlation for GNF2 fuel supports a lower bound pressure limit of 700 psia and a LAR is required to update OCNGS TS 2.1.A and TS 2.1.B and NMP1 TS 2.1.1.a and 2.1.1.b to reflect this pressure limit.

The CPR correlation for GNF2 fuel in OCNGS and NMP1 reactors supports a lower bound pressure of 700 psia. GEXL17 CPR correlation with the lower bound limit of 700 psia is applicable to GNF2 fuel. 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 700 psia as lower bound limit for GNF2 fuel has been approved by the NRC for use per NEDE-24011-P-A by reference. Use of GEXL17 does not change the thermal power limit (25%) corresponding to 10% rated core flow. The 25% RTP limit is a 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.

Use of 700 psia as steam dome pressure limit for OCNGS TS 2.1.A and TS 2.1.B and NMP1 TS 2.1.1.a and 2.1.1.b is supported by the CPR correlations for fuel designs at OCNGS and NMP1, respectively. 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.A and 2.1.B for OCNGS and 2.1.1.a and 2.1.1.b for NMP1 reactor steam dome pressure from 800 psia to 700 psia resolves the 10 CFR Part 21 condition concerning the potential to violate Reactor Core Safety Limit 2.1.B (OCNGS) and TS 2.1.1.b (NMP1) 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 OCNGS and NMP1 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 specified TS limit, then a LAR will be submitted for staff 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.

The proposed change in OCNGS TS 2.1 and NMP1 TS 2.1.1 which specifies the SL on the MCPR expands the range of applicability of the SL on the MCPR to a lower 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 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, operating margin is increased due to the proposed change to ensure OCNGS and NMP1 will not enter into an unanalyzed condition during a pressure regulator failed open event such as is potentially possible with the current low pressure safety limit. For OCNGS and NMP1, PRFO is not a limiting event for establishing a thermal limit and is not analyzed on a reload basis.

Results of the above Exelon and GEH evaluations show that the LPIS setting at OCNGS and NMP1 is adequate to prevent reactor pressure from falling below 700 psia. The CPR correlation for GNF2 fuel in OCNGS and NMP1 reactors supports a lower bound pressure of 700 psia. A LAR is required to update the OCNGS TS 2.1.A and 2.1.B and the NMP1 TS 2.1.1.a and TS 2.1.1.b and corresponding TS Bases to reflect lower bound pressure limit of 700 psia.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, provides criteria for Emergency Core Cooling System (ECCS) performance and 10 CFR 50.36 (TS) 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 TS change for OCNGS revises the reactor steam dome pressure stated in the Reactor Core Safety Limits 2.1.A and 2.1.B to remove the potential to violate Reactor Core Safety Limit 2.1.B during a PRFO transient. Similarly, the proposed TS change for NMP1 revises the reactor steam dome pressure stated in the Reactor Core Safety Limits 2.1.1.a and 2.1.1.b to remove the potential to violate Reactor Core Safety Limit 2.1.1.b during a PRFO transient.

Exelon has evaluated the proposed change against the applicable regulatory requirements and acceptance criteria. As long as the 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 Safety Limits 2.1.A and 2.1.B for OCNGS and 2.1.1.a and 2.1.1.b for NMP1 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 General Design Criterion 10 regarding acceptable fuel design limits and continues to assure that the underlying criteria of the safety limit is met. Based on this, there is reasonable assurance that the health and safety of the public is unaffected.

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 OCNGS and NMP1 as documented in the following approved amendments:

1. The NRC approved amendment 185 for the Monticello Nuclear Generating Plant on November 25, 2014 (Reference 6).

2. The NRC issued amendments 269 and 213 for the Edwin I. Hatch Plant Units 1 and 2 on October 20, 2014 (Reference 7).

3. The NRC issued amendment 182 for the River Bend Station on December 11, 2014 (Reference 8).

4. The NRC issued amendment 309 for the James A. FitzPatrick Nuclear Power Plant on February 9, 2015 (Reference 9).

5. The NRC issued amendments 306 and 310 for Peach Bottom Atomic Power Station, Units 2 and 3 on April 27, 2016 (Reference 10).

6. The NRC 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 on May 11, 2016 (Reference 11).

4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (Exelon), is proposing a change to the TS, Appendix A of Renewed Facility Operating License No. DPR-16 for OCNGS and Renewed Facility Operating License No. DPR-63 for NMP1.

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 change to the OCNGS TS for the reactor steam dome pressure in Reactor Core Safety Limits 2.1.A and 2.1.B does not alter the use of the analytical methods used to determine the safety limits that have been previously reviewed and approved by the NRC. Additionally, the proposed change to NMP1 for the reactor steam dome pressure in Reactor Core Safety Limits 2.1.1.a and 2.1.1.b does 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 change is in accordance with an NRC approved critical power correlation methodology, and as such, maintains required safety margins. The proposed change does not adversely affect accident initiators or precursors, nor does it alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained.

The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not require any physical change to any plant SSCs nor does it require any change in systems or plant operations. The proposed change is consistent with the safety analysis assumptions and resultant consequences.

Lowering the value of reactor steam dome pressure in the TS has no physical effect on plant equipment and therefore, no impact on the course of plant transients. The change is an analytical exercise to demonstrate the applicability of correlations and methodologies. There are no known operational or safety benefits.

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 reduction in the reactor dome pressure safety limit from 800 psia to 700 psia is a change based upon previously approved documents and does not involve changes to the plant hardware or its operating characteristics. As a result, no new failure modes are being introduced. There are no hardware changes nor are there any 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 change.

   The proposed change does not introduce any new accident precursors, nor does it involve any physical plant alterations or changes in the methods governing normal plant operation. Also, the change does not impose any new or different requirements or eliminate any existing requirements. The change does not alter assumptions made in 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 GE determined that since the MCPR improves during the PRFO transient, there is no decrease in the safety margin and therefore there is not a threat to fuel cladding integrity. The proposed change in reactor dome pressure supports the current safety margin, which protects the fuel cladding integrity during 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 change does not alter the behavior of plant equipment, which remains unchanged.

The proposed change to Reactor Core Safety Limits 2.1.A and 2.1.B is consistent with and within the capabilities of the applicable NRC approved critical power correlation for the fuel designs in use at OCNGS. Additionally, the proposed change to Reactor Core Safety Limits 2.1.1.a and 2.1.1.b is consistent with and within the capabilities of the applicable NRC approved critical power correlation for the fuel designs in use at NMP1. No setpoints at which protective actions are initiated are altered by the proposed change. The proposed change does not alter the manner in which the safety limits are determined. This change is consistent with plant design and does not change the TS operability requirements; thus, previously evaluated accidents are not affected by this proposed change.

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

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

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

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

6.0 REFERENCES

1. Letter from Jason Post (GE Energy – Nuclear) to U.S. NRC, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," dated 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," GE Hitachi Nuclear Energy, 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, "General Electric Standard Application for Reactor Fuel," Global Nuclear Fuel.

6. 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).

7. 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).

8. Letter from Alan 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.

9. 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.

10. Letter from R. E. Ennis (U.S. NRC) to President and Chief Nuclear Officer (Exelon Generation Company, LLC), "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. ML16064A150).

11. Letter from Blake Purnell, (U.S. NRC) to President and Chief Nuclear Officer (Exelon Generation Company, LLC), "Clinton Power Station Unit No.1; 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 Nos. ML15231A097 and ML16105A421)

12. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request - Supplement, Proposed Revision to Technical Specifications in Response to GE Energy - Nuclear 10 CFR Part 21 Safety Communication SC05-03," dated May 9, 2016 (ADAMS Accession No. ML16131A698).

ATTACHMENT 2

Markup of Proposed Technical Specifications Pages

Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219

Nine Mile Point Nuclear Station, Unit 1 (NMP1)
Docket No. 50-220

License Amendment Request to Reduce Steam Dome Pressure Specified in the Reactor Core Safety Limits

TS Pages for OCNGS

2.1-1

TS Pages for NMP1

9

SECTION 2

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT - FUEL CLADDING INTEGRITY

Applicability: Applies to the interrelated variables associated with fuel thermal behavior.

Objective: To establish limits on the important thermal hydraulic variables to assure the integrity of the fuel cladding.

Specifications:

700

A. When the reactor pressure is greater than or equal to ~~800~~ psia and the core flow is greater than or equal to 10% of rated, the existence of a minimum CRITICAL POWER RATIO (MCPR) less than 1.10 for both four or five loop operation and 1.12 for three loop operation shall constitute violation of the fuel cladding integrity safety limit.

700

B. When the reactor pressure is less than ~~800~~ psia or the core flow is less than 10% of rated, the core THERMAL POWER shall not exceed 25% of RATED THERMAL POWER.

C. In the event that reactor parameters exceed the limiting safety system settings in Specification 2.3 and a reactor scram is not initiated by the associated protective instrumentation, the reactor shall be brought to, and remain in, the COLD SHUTDOWN CONDITION until an analysis is performed to determine whether the safety limit established in Specification 2.1.A and 2.1.B was exceeded.

D. During all modes of reactor operation with irradiated fuel in the reactor vessel, the water level shall not be less than 4'8" above the TOP OF ACTIVE FUEL.

SAFETY LIMIT

2.1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits on the important thermal-hydraulic variables to assure the integrity of the fuel cladding.

Specification:

a. When the reactor pressure is greater than ~~800~~ 700 psia and the core flow is greater than 10%, the existence of a Minimum Critical Power Ratio (MCPR) less than the Safety Limit Critical Power Ratio (SLCPR) (Reference 12) shall constitute violation of the fuel cladding integrity safety limit.

b. When the reactor pressure is less than or equal to ~~800~~ 700 psia or core flow is less than 10% of rated, the core power shall not exceed 25% of rated thermal power.

LIMITING SAFETY SYSTEM SETTING

2.1.2 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings on automatic protective devices related to variables on which the fuel loading safety limits have been placed.

Objective:

To provide automatic corrective action to prevent exceeding the fuel cladding safety limits.

Specification:

Fuel cladding limiting safety system settings shall be as follows:

a. The flow-biased APRM scram and rod block trip settings shall be established according to the following relationships:

The minimum of:

For $W \geq 0\%$:

$S \leq (0.55W + 67\%)T$ with a maximum value of 122%

$S_{RB} \leq (0.55W + 62\%)T$ with a maximum value of 117%

ATTACHMENT 3

Markup of Proposed Technical Specifications Bases Pages
(For Information Only)

License Amendment Request

Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219

Nine Mile Point Nuclear Station, Unit 1 (NMP1)
Docket No. 50-220

License Amendment Request to Reduce Steam Dome Pressure Specified in the Reactor Core Safety Limits

TS Bases Pages for OCNGS

2.1-2
2.1-3
2.3-5

TS Bases Pages for NMP1

13
14
22

For operation at low pressure or low flows, another basis is used as follows:

Bases:

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily 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 procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity safety limit 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 Safety Limit MCPR[1] is determined using the General Electric Thermal Analysis Basis, GETAB[2], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The power distribution uncertainty is treated in accordance with a NRC approved method[3][4]. The revised analysis results in lower SLMCPR values primarily due to an improved treatment of the power distribution uncertainty that reduces the conservatism of the GETAB method of power allocation. All other uncertainties are consistent with the GETAB basis. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The use of the GEXL correlation is not valid for the critical power calculations at pressures below ~~800~~ 700 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is protected by limiting the core THERMAL POWER.

~~At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core.~~ Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. 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 core THERMAL POWER of more than 50%. Thus, a core THERMAL POWER limit of 25% for reactor pressures below ~~800 psi~~ 700 psia or core flow less than 10% of rated is conservative. Additional information on low flow conditions is available in Reference 5.

Plant safety analyses have shown that the scrams ca[...]y safety setting will assure that the Safety Limit of Specification 2.1.A or 2[...]ded. Scram times are checked periodically to assure the insertion times are adequate. The THERMAL POWER transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. Specification 2.1.C requires that appropriate analysis be performed to verify that backup protective instrumentation has prevented exceeding the fuel cladding integrity safety limit prior to resumption of POWER OPERATION. The concept of not approaching a Safety Limit provided scram signals are OPERABLE is supported by the extensive plant safety analysis.

If reactor water level should drop below the TOP OF ACTIVE FUEL, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. With a water level above the TOP OF ACTIVE FUEL, adequate cooling is maintained and the decay heat can easily be accommodated. It should be noted that during power generation there is no clearly defined water level inside the shroud and what actually exists is a mixture level. This mixture begins within the active fuel region and extends up through the moisture separators. For the purpose of this specification water level is defined to include mixture level during power operations.

The lowest point at which the water level can presently be monitored is 4'8" above the TOP OF ACTIVE FUEL. Although the lowest reactor water level limit which ensures adequate core cooling is the TOP OF ACTIVE FUEL, the safety limit has been conservatively established at 4'8" above the TOP OF ACTIVE FUEL.

REFERENCES

(1) NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II) (latest approved version as specified in the COLR)

(2) General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO-10958-A, January 1977.

(3) NEDC-32694P-A, Power Distribution Uncertainties for Safety Limit MCPR Evaluations.

(4) NEDC-32601P-A, Methodology and Uncertainties for Safety Limit MCPR Evaluations.

(5) SIL No. 516 Supplement 2, January 19, 1996.

The low pressure isolation of the main steam line at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cool-down of the vessel. The low-pressure isolation protection is enabled with entry into IRM range 10 or the RUN mode. In addition, a scram on 10% main steam isolation valve (MSIV) closure anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. Bypass of the MSIV closure scram function below 600 psig is permitted to provide sealing steam and allow the establishment of condenser vacuum. Advantage is taken of the MSIV scram feature to provide protection for the low-pressure portion of the fuel cladding integrity safety limit. To continue operation beyond 12% of rated power, the IRM's must be transferred into range 10. Reactor pressure must be above 825 psig to successfully transfer the IRM's into range 10. Entry into range 10 at less than 825 psig will result in main steam line isolation valve closure and MSIV closure scram. This provides automatic scram protection for the fuel cladding integrity safety limit which allows a maximum power of 25% of rated at pressures below ~~800~~ 700 psia. Below 600 psig, when the MSIV closure scram is bypassed, scram protection is provided by the IRMs.

Operation of the reactor at pressure lower than 825 psig requires that the mode switch be in the STARTUP position and the IRMs be in range 9 or lower. The protection for the fuel clad integrity safety limit is provided by the IRM high neutron flux scram in each IRM range. The IRM range 9 high flux scram setting at 12% of rated power provides adequate thermal margin to the safety limit of 25% of rated power. There are few possible significant sources of rapid reactivity input to the system through IRM range 9:effects of increasing pressure at zero and low void content are minor; reactivity excursions from colder makeup water, will cause an IRM high flux trip; and the control rod sequences are constrained by operating procedures backed up by the rod worth minimizer. In the unlikely event of a rapid or uncontrolled increase in reactivity, the IRM system would be more than adequate to ensure a scram before power could exceed the safety limit. Furthermore, a mechanical stop on the IRM range switch requires an operator to pull up on the switch handle to pass through the stop and enter range 10. This provides protection against an inadvertent entry into range 10 at low pressures. The IRM scram remains active until the mode switch is placed in the RUN position at which time the trip becomes a coincident IRM upscale, APRM downscale scram.

The adequacy of the IRM scram was determined by comparing the scram level on the IRM range 10 to the scram level on the APRMs at 30% of rated flow. The IRM scram is at 38.4% of rated power while the APRM scram is at 59.3% of rated power. The minimum flow for Oyster Creek is at 30% of rated flow and this would be the lowest APRM scram point. The increased recirculation flow to 65% of flow will provide additional margin to CPR Limits. The APRM scram at 65% of rated flow is 100.8% of rated power, while the IRM range 10 scram remains at 38.4% of rated power. Therefore, transients requiring a scram based on flux excursion will be terminated sooner with a IRM range 10 scram than with an APRM scram. The transients requiring a scram by nuclear instrumentation are the loss of feedwater heating and the improper startup of an idle recirculation loop. The loss of feedwater heating transient is not affected by the range 10 IRM since the feedwater heaters will not be put into service until after the LPRM downscales have cleared, thus insuring the operability of the APRM system. This will be administratively controlled. The improper startup of an idle recirculation loop becomes less severe at lower power level and the IRM scram would be adequate to terminate the flux excursion.

BASES FOR 2.1.1 FUEL CLADDING - SAFETY LIMIT

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a safety limit such that the Minimum Critical Power Ratio (MCPR) is no less than the Safety Limit Critical Power Ratio (SLCPR) (Reference 12). The SLCPR 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 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 protection system safety 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 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.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, at reactor pressure >~~800~~ 700 psia and core flow >10% of rated the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the Critical Power Ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the Minimum Critical Power Ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective set points via the instrumented variables, by the nominal expected flow control line. The SLCPR has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the SLCPR is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in References 1 and 12.

Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the condition of the SLCPR would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit SLCPR, operation is constrained to ensure that actual fuel operation is maintained within the assumptions of the fuel rod thermal-mechanical design and the safety analysis basis. At full power, this limit is the linear heat generation rate limit with overpower transients constrained by the unadjusted APRM scram and rod block. During steady-state operation at lower power levels, where the fraction of rated thermal power is less than the core maximum fraction of limiting power density, the APRM flow biased scram and rod block settings are adjusted by the equations in Specification 2.1.2a.

~~At pressure equal to or below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all core flows, this pressure differential is maintained in the bypass region of the core.~~ Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi.

Analyses 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. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than 28×10^3 lb/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28×10^3 lb/hr is approximately 3.35 MWt. With the design peaking factor, this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below ~~800~~ 700 psia or core flow less than 10% of rated is conservative. Additional information on low flow conditions is available in Reference 19.

The use of GEXL correlation is not valid for the critical power calculations at pressures below 700 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is protected by limiting the core thermal power. For operation at low pressures or low flows, another basis is used as follows:

REFERENCES FOR BASES 2.1.1 AND 2.1.2 FUEL CLADDING

(1) General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO-10958 and NEDE-10958.

(2) Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10801, February 1973.

(3) UFSAR Section XV-A and B.

(4) UFSAR Section XV-A and B.

(5) UFSAR Section XV-A and B.

(6) UFSAR Section XV-A and B.

(7) Letters, Peter A. Morris, Director of Reactor Licensing, USAEC, to John E. Logan, Vice-President, Jersey Central Power and Light Company, dated November 22, 1967 and January 9, 1968.

(8) UFSAR Section XV-A and B.

(9) Letter, T. J. Brosnan, Niagara Mohawk Power Corporation, to Peter A. Morris, Division of Reactor Licensing, USAEC, dated February 28, 1972.

(10) Letter, Philip D. Raymond, Niagara Mohawk Power Corporation, to A. Giambusso, USAEC, dated October 15, 1973.

(11) Nine Mile Point Nuclear Power Station Unit 1 Load Line Limit Analysis, NEDO 24012, May, 1977.

(12) Licensing Topical Report "General Electric Standard Application for Reactor Fuel," (GESTAR II) NEDE-24011-P-A, latest approved revision.

(13) Nine Mile Point Nuclear Power Station Unit 1, Extended Load Line Limit Analysis, License Amendment Submittal (Cycle 6), NEDO-24185, April 1979.

(14) General Electric SIL 299 "High Drywell Temperature Effect on Reactor Vessel Water Level Instrumentation."

(15) Letter (and attachments) from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-B, Amendment 10."

(16) GENE-909-16-0393, "IRM/APRM Overlap Analysis for Nine Mile Point Nuclear Station Unit One," Revision 1, dated April 14, 1993.

(17) GENE-909-39-1093, "IRM/APRM Overlap Improvement for Nine Mile Point Nuclear Station Unit One," dated March 8, 1994.

(18) GENE-C5100196-04, "APRM Flow-Biased Trip Setpoints Stability Long-Term Solution Option II," dated June 1997.

(19) SIL No. 516 Supplement 2, January 19, 1996.