FENOC™
FirstEnergy Nuclear Operating Company

Perry Nuclear Power Plant
PO Box 97
10 Center Road
Perry, Ohio 44081

David B. Hamilton
Vice President

440-280-5382

November 1, 2016
L-16-179

10 CFR 50.90

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

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
License Amendment Request to Reduce the Reactor Steam Dome Pressure Specified in Technical Specification 2.1.1, "Reactor Core Safety Limits"

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) hereby requests an amendment to the facility operating license for the Perry Nuclear Power Plant (PNPP). The proposed amendment would revise Technical Specification (TS) 2.1.1, "Reactor Core Safety Limits," to reduce the reactor steam dome pressure value specified in TS 2.1.1.1 and TS 2.1.1.2 from 785 pounds per square inch gauge (psig) to 686 psig. This change would align the pressure value with the Nuclear Regulatory Commission (NRC) approved pressure ranges associated with the Global Nuclear Fuel (GNF) critical power correlations that are applicable to the fuel types in use at PNPP.

An evaluation of the proposed license amendment is enclosed. FENOC requests NRC staff approval of the proposed license amendment by October 31, 2017. The proposed changes will be implemented within 60 days of the approval of the amendment.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 315-6810.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 1, 2016.

Sincerely,

David B. Hamilton

Enclosure:
Evaluation of Proposed License Amendment

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager
Executive Director, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

Enclosure
L-16-179

Evaluation of Proposed License Amendment
(23 pages, excluding this page)

Subject: Request for Licensing Action to Revise the Reactor Steam Dome Pressure Value Identified in Perry Technical Specification (TS) Reactor Core Safety Limits, TS 2.1.1.1 and TS 2.1.1.2

Attachments:

1. Proposed Technical Specification Changes (Mark-up)
2. Proposed Technical Specification Change (Retyped)
3. Proposed Technical Specification Bases Changes (Mark-up)

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License NPF-58 for the Perry Nuclear Power Plant (PNPP).

The proposed amendment would revise PNPP Technical Specification (TS) 2.1.1 to reduce the reactor steam dome pressure value associated with the TS Reactor Core Safety Limits specified in TS 2.1.1.1 and TS 2.1.1.2 from 785 pounds per square inch gauge (psig) to 686 psig. This change would align the PNPP TS Reactor Core Safety Limit (SL) pressure value with the Nuclear Regulatory Commission (NRC) approved pressure ranges associated with the Global Nuclear Fuel (GNF) critical power correlations that are applicable to the fuel types in use at PNPP.

The proposed changes were initiated as a result of General Electric (GE) Energy - Nuclear 10 CFR Part 21 Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit" (Reference 1). Implementation of the proposed changes will eliminate the potential to exceed TS Reactor Core SL 2.1.1 during a postulated Pressure Regulator Failure-Maximum Demand (Open) (PRFO) transient event.

2.0 DETAILED DESCRIPTION

2.1 Present PNPP TS Reactor Core Safety Limits

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). GE considers a PRFO event to be an AOO. 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 PNPP TS SL 2.1.1.1 requires that thermal power shall be $\leq$ 23.8% rated thermal power when reactor vessel steam dome pressure is < 785 psig or core flow is < 10% of rated. This SL was introduced to preclude the need for critical power ratio (CPR) calculations when reactor vessel steam dome pressure is less than 785 psig. The reactor power value identified in TS SL 2.1.1.1 was selected to ensure that reactor power remains well below the fuel assembly critical power for the conditions where CPR calculations are not performed.

The current PNPP TS 2.1.1.2 requires that the minimum critical power ratio (MCPR) shall be $\geq$ 1.10 for two recirculation loop operation or $\geq$ 1.13 for single recirculation loop operation with the reactor vessel steam dome pressure $\geq$ 785 psig and core flow $\geq$ 10% rated core flow. This SL is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties. This fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not exceeded.

These TS SLs ensure the appropriate use of MCPR calculations to conditions when the reactor vessel steam dome pressure is within the applicability range of the GEXL correlations.

2.2 Reason for Proposed Change

On March 29, 2005, GE Energy - Nuclear issued 10 CFR Part 21 Safety Communication SC05-03 (Reference 1) that identified the potential vulnerability for the PRFO transient event to result in a condition in which TS SL 2.1.1.1 may be exceeded. From the GE report, "The standard Improved Technical Specifications (ITS) specify SL 2.1.1.1 to require that thermal power shall be $\leq$ [25]% rated (this value is a plant-specific number), when reactor steam dome pressure is < 785 psig or core flow is < 10% of rated." GE identified that certain plants have the potential to experience a PRFO event that results in reactor dome pressure dropping below the TS SL of 785 psig while reactor thermal power exceeds the plant-specific thermal limit identified in TS 2.1.1.1. As determined by GE, this does not challenge the fuel cladding integrity or constitute a safety hazard, since the CPR increases as pressure decreases below the bottom of the approved range. Nonetheless, the potential to exceed the TS SL during the PRFO event exists. As such, FirstEnergy Nuclear Operating Company (FENOC) is proposing to revise the reactor vessel steam dome pressure specified in TS SLs 2.1.1.1 and 2.1.1.2 from 785 psig to 686 psig. This value is consistent with the NRC approved applicable pressure range of critical power correlations for the current fuel designs in the PNPP Unit 1 reactor core.

In response to Reference 1, the BWR Owners' Group (BWROG) commissioned the development of a methodology for plants to assess the adequacy of their current main steam isolation valve (MSIV) closure at the low pressure isolation setpoint (LPIS) setting and to provide a set of recommendations for what actions should be taken based on the outcome of their assessment. The methodology and recommendations are documented in a BWROG report (Reference 18). The methodology is developed by analyzing a limiting plant, assessing uncertainties, and determining a method to conservatively scale the limiting plant's results to other plant configurations and operating flexibility options through sensitivity studies. The scaling methodology is applied to an example plant to demonstrate its adequacy. Additionally, a parametric study using a 720 psig LPIS setting with various plant configurations is provided in the BWROG report.

2.3 Proposed TS Changes

The proposed TS changes are as follows:

1. TS 2.1.1.1: Reduce the reactor steam dome pressure value of 785 psig to 686 psig. The proposed TS SL would read:

   With the reactor steam dome pressure < 686 psig or core flow < 10% rated core flow:

   THERMAL POWER shall be ≤ 23.8% RTP.

2. TS 2.1.1.2: Reduce the reactor steam dome pressure value of 785 psig to 686 psig. The proposed TS SL would read:

   With the reactor steam dome pressure ≥ 686 psig and core flow ≥ 10% rated core flow:

   The Minimum Critical Power Ratio (MCPR) shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.13 for single recirculation loop operation.

The proposed TS changes are shown on the annotated page provided in Attachment 1. The retyped TS page is provided in Attachment 2. The associated changes to the TS Bases are provided for information only in Attachment 3 and will be controlled by TS 5.5.1.1 "Technical Specification (TS) Bases Control Program."

3.0 TECHNICAL EVALUATION

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

Evaluations performed 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 thermal limit, which would exceed the conditions specified in TS SL 2.1.1.1.

As stated previously, the methodology developed by the BWROG 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 18. Based on the results of the studies documented in Reference 18, it is proposed that the reactor vessel steam dome pressure specified in PNPP TS SLs 2.1.1.1 and 2.1.1.2 be changed from 785 to 686 psig. In addition, calculations done by FENOC have determined that the current PNPP MSIV LPIS analytical limit of 782.3 psig (with corresponding allowable limit of 795.2 psig and setpoint of 807.0 psig) is sufficient to preclude reactor vessel steam dome pressure from falling below the proposed 686 psig while above 23.8% power (the PNPP site-specific thermal power rating) for current operation during a PRFO event.

GNF advanced fuel designs have NRC approved critical power correlations with lower-bound pressures below the 785 psig reactor steam dome pressure currently specified in TS Reactor Core SLs 2.1.1.1 and 2.1.1.2. FENOC proposes to utilize this information and reduce the reactor steam dome pressure SL consistent with the approved lower-bound pressure for the GNF fuel comprising the PNPP reactor core, as described below.

PNPP currently has a mixed core of GE14 and GNF2 fuel types in Unit 1. The GEXL correlations that apply to these fuel types are GEXL14 and GEXL17 respectively. The GEXL correlations are used to perform CPR calculations for all of the fuel types in use at PNPP. Both of these GEXL correlations have NRC-approved pressure applicability ranges of 700 to 1400 pounds per square inch absolute (psia). This corresponds to approximately 685 to 1385 psig.

GEXL14 (GE14 fuel)
The lower bound limit of 700 psia for the GEXL14 correlation is documented and justified in NEDC-32851P-A for GE14 Fuel (Reference 2). This topical report has been reviewed and approved by the NRC.

GEXL17 (GNF2 fuel)
The lower bound limit of 700 psia for the GEXL17 correlation is documented and justified in NEDC-33292P "GEXL17 Correlation for GNF2 Fuel" (Reference 3). This lower bound limit is discussed and NEDC-33292P is referenced in NEDC-33270P "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)" (Reference 4). The summary and conclusion section of NEDC-33270P indicates that all of the criteria defined in GESTAR II have been met for the GNF2 fuel design. NEDC-33270P was submitted to the NRC as part of Amendment 33 to NEDE-24011-P "General Electric Standard Application for Reactor Fuel (GESTAR II)". NEDE-24011-P Amendment 33 was approved by the NRC and incorporated into Revision 17 of NEDE-24011-P-A (Reference 17).

As such, the GEXL17 correlation for GNF2 fuel is approved for use per NEDE-24011-P-A by reference. NEDE-24011-P-A also specifically states:

> Fuel design compliance with the fuel licensing acceptance criteria constitutes USNRC acceptance and approval of the fuel design without specific USNRC review. The fuel licensing acceptance criteria are presented in the [NEDE-24011-P-A] subsections that follow.

The 700 psia lower bound application range pressure limit applies to both GE14 and GNF2 fuels. This low end pressure range limit of 700 psia corresponds to approximately 685.3 psig. As such, the proposed revision of the reactor steam dome pressure to 686 psig within TS SL 2.1.1.1 and TS SL 2.1.1.2 is established as follows:

(700 psia - 14.7 psia = 685.3 psig, rounded up to next whole number = 686 psig).

Establishment of the reactor steam dome pressure SL value in this manner ensures that the identified TS SL value resides within the applicable pressure range of the GEXL correlations.

The proposed change in PNPP TS 2.1.1.2, which specifies the SL on the MCPR, expands the range of applicability of the SL on the MCPR to the low pressure end of the GEXL correlation applicability range. This low pressure SL protects against 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 applicability of the GEXL correlation has been expanded through increased testing demonstrating adequate performance of the correlation over the expanded range. As a result, there is no change in safety margin due to the proposed TS change. The proposed change will also ensure that PNPP will not exceed the TS SL during a postulated PRFO event.

The PRFO event involves the failure of the pressure regulator in the open direction causing the turbine control valves and the turbine bypass valves to fully open. This causes the reactor to depressurize rapidly. When the main steam line low pressure isolation setpoint is reached, a closure signal for the MSIVs is initiated and a reactor scram occurs. As the MSIVs approach full closure, reactor depressurization terminates, thus preventing reactor pressure from decreasing below the proposed SL of 686 psig while core thermal power is still above 23.8% of rated thermal power. Reactor depressurization transients, such as a 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. This means that the CPR at the start of the event is the limiting CPR condition during the entire transient. With an initial condition that is restricted by the Operating Limit Minimum Critical Power Ratio (OLMCPR) and an event that causes the CPR to increase, the margin to the Safety Limit MCPR increases during the event, and therefore, no threat to fuel cladding integrity exists.

Conclusion

The proposed reduction in TS SL reactor steam dome pressure provides a wider acceptable pressure range for transients to comply with MCPR limits and provides a greater allowable pressure range for a PRFO transient. FENOC has determined that the reactor vessel steam dome pressure will remain above the proposed lower TS SL pressure (686 psig) during the PRFO transient.

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

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

When evaluating a different fuel design from those currently in use in the PNPP reactor core, the CPR correlation is 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, that is higher than the TS pressure limit proposed herein (686 psig), then a license amendment request (LAR) would need to be submitted for NRC review and approval. If the CPR correlation has a lower bound pressure, that is lower than the TS pressure limit proposed herein (686 psig), then a LAR would not be required since the proposed TS pressure value would have a conservative lower bound.

4.0 REGULATORY EVALUATION

4.1 Significant Hazards Consideration

FirstEnergy Nuclear Operating Company (FENOC), proposes an amendment to Operating License NPF-58 for the Perry Nuclear Power Plant (PNPP). The proposed amendment would revise PNPP Technical Specification (TS) 2.1.1 to reduce the reactor steam dome pressure value associated with the TS Reactor Core Safety Limits specified in TS 2.1.1.1 and TS 2.1.1.2 from 785 pounds per square inch gauge (psig) to 686 psig. This change would align the PNPP TS Reactor Core Safety Limit (SL) pressure value with the NRC approved pressure ranges associated with the Global Nuclear Fuel (GNF) critical power correlations that are applicable to the fuel types currently in use at PNPP.

The proposed changes were initiated as a result of General Electric (GE) Energy - Nuclear 10 CFR Part 21 Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," issued March 29, 2005, and are being submitted based on the results of subsequent GE analyses that were sponsored by the Boiling Water Reactor Owners Group. Implementation of the proposed changes will resolve the 10 CFR Part 21 condition concerning the potential to exceed TS Reactor Core SL 2.1.1.1 during a postulated Pressure Regulator Failure-Maximum Demand (Open) (PRFO) transient event.

FENOC 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 change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Decreasing the reactor steam dome pressure limit in Technical Specification Safety Limits 2.1.1 expands the range of use of the GEXL14 and GEXL17 correlations (applicable to GE14 and GNF2 fuel respectively) and the calculation of the minimum critical power ratio (CPR). The CPR increases during the pressure reduction that occurs during the PRFO event, so that the initial CPR is the limiting CPR condition during the entire transient. CPR increases during the event relative to the initial CPR value, so fuel cladding integrity is not threatened. Since the change does not involve a modification of any plant hardware, the probability and consequence of the PRFO transient are essentially unchanged.

   The proposed change will continue to support the application range of the GEXL correlations applied at PNPP and the calculation of the minimum CPR. The proposed TS revision involves no significant changes to the operation of any systems or components in normal, accident or transient operating conditions.

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

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed reduction in the reactor steam dome pressure limit in Technical Specification Safety Limits 2.1.1 from 785 psig to 686 psig is a change based on NRC approved documents that permit a wider range of applicability for the GEXL critical power correlations for both GE14 and GNF2 fuel types in the reactor core. This change does not involve changes to the plant hardware or its operating characteristics. There are no changes in the method by which any plant systems perform a safety function, nor are there any changes in the methods governing normal plant operation. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. As a result, no new failure modes are being introduced.

Therefore, the change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change 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 critical power ratio improves during the PRFO transient, there is no impact on the fuel safety margin, and there is no challenge to fuel cladding integrity. The proposed changes do not change the requirements governing operation or the availability of safety equipment assumed to operate to preserve the margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FENOC 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.

4.2 Applicable Regulatory Guidance/Criteria

The following regulatory requirements are applicable.

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in the TSs. As stated in 10 CFR 50.36(c)(1), the TSs will include safety limits for nuclear reactors, which are stated to be "limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

The proposed TS change revises reactor steam dome pressure stated in TS 2.1.1.1 and TS 2.1.1.2, and will remove the potential to exceed the reactor core safety limit 2.1.1.1 during a PRFO transient. Compliance with 10 CFR Part 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants, GDC 10, "Reactor Design," is achieved by preventing exceedance of fuel design limits. GDC 10 states:

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

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

As long as the reactor core pressure and flow are within the applicability range of the specified critical power correlation (in this case the GEXL14 and GEXL17 critical power correlations), the proposed reactor steam dome pressure change to reactor core SL 2.1.1.1 and 2.1.1.2 will continue to ensure that 99.9 percent of the fuel rods in the core are not expected to experience boiling transition. This satisfies the requirements of GDC 10 regarding acceptable fuel design limits and continues to assure that the underlying criteria of the safety limit is met.

Based on the above, the proposed changes satisfy the regulatory requirements cited above.

4.3 Precedent

With the issuance of Safety Communication SC05-03, GE identified plants that have the potential for reactor dome pressure to drop below 785 psig during a PRFO event. Depending upon the plant-specific response to a PRFO, reactor steam dome pressure could drop below 785 psig while thermal power exceeds the plant-specific thermal limit identified in TS 2.1.1.1. The NRC previously reviewed a number of industry requests for TS changes that support the resolution of this GE Part 21 concern. Those requests are similar to this PNPP amendment request, as documented in the following submittals and the associated approved amendments.

On September 8, 2010, Entergy Operations, Inc., on behalf of the Grand Gulf Nuclear Station (GGNS), submitted an amendment request for extended power uprate that included proposing the reduction of reactor steam dome pressure specified in Reactor Core Safety Limit Specification 2.1.1 (Reference 5). The NRC approved Amendment 191 for the GGNS on July 18, 2012 (Reference 6).

On March 11, 2013, Northern States Power Company – Minnesota, on behalf of the Monticello Nuclear Generating Plant (MNGP), submitted an 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 MNGP on November 25, 2014 (Reference 8).

On March 24, 2014, Southern Nuclear Operating Company, on behalf of the Edwin I. Hatch Nuclear Plant (HNP), submitted an amendment request to revise HNP 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., on behalf of the River Bend Station (RBS), submitted an amendment request to revise RBS 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 (Reference12).

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

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

4.4 Conclusions

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. 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 effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. GE Energy - Nuclear, 10 CFR Part 21 Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," March 29, 2005.

2. NEDC-32851P-A, Rev. 5, "GEXL14 Correlation for GE 14 Fuel," dated April 2011.

3. NEDC-33292P, Rev. 3, "GEXL17 Correlation for GNF2 Fuel," dated June 2009.

4. NEDC-33270P, Rev. 6, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," dated March 2016.

5. Later from Michael A. Krupa (Entergy Operations, Inc.) to U.S. NRC, "License Amendment Request, Extended Power Uprate, Grand Gulf Nuclear Station, Unit 1," dated September 8, 2010. (ADAMS Accession No. ML102660403)

6. Later from Alan B. Wang (U.S. NRC) to Vice President, Operations (Entergy Operations, Inc.), "Grand Gulf Nuclear Station, Unit 1 - Issuance of Amendment Re: Extended Power Uprate (TAC No. ME4679)," dated July 18, 2012. (ADAMS Accession No. ML121210020)

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

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

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

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

11. Letter from Eric Olson (Entergy Operations, Inc.) to U.S. NRC, "License Amendment Request, Changes to Technical Specification 2.1.1, 'Reactor Core SLs'," dated May 28, 2013. (ADAMS Accession No. ML13155A138)

12. Letter from Alan B. Wang (U.S. NRC) to Vice President, Operations (Entergy Operations, Inc.), "River Bend Station, Unit 1 - Issuance of Amendment Re: Technical Specification 2.1.1, 'Reactor Core SLs' (TAC No. MF1948)," dated December 11, 2014. (ADAMS Accession No. ML14192A831)

13. Letter from Lawrence M. Coyle (Entergy Nuclear Operations, Inc.) to U.S. NRC, "Application to Revise Technical Specifications for Technical Specification Low Pressure Safety Limit," dated October 8, 2013. (ADAMS Accession No. ML13282A559)

14. Letter from Douglas V. Pickett (U.S. NRC) to Vice-President, Operations (Entergy Nuclear Operations, Inc.), "James A FitzPatrick Nuclear Power Plant-Issuance of Amendment Re: Application to Revise Technical Specifications for Technical Specification Low Pressure Safety Limit (TAC No. MF2897)," dated February 9, 2015. (ADAMS Accession No. ML15014A277)

15. Letter from Robert G. Smith (Entergy Nuclear Operations, Inc.) to U.S. NRC, "Proposed License Amendment: Revision to Technical Specifications (TS) 2.1, Safety Limits to Resolve Pressure Regulator Fail-Open (PRFO) Transient Reported by General Electric Nuclear Energy in Accordance with 10 CFR 21.21(d)," dated April 5, 2013. (ADAMS Accession No. ML13108A217)

16. Letter from Nadiyah S. Morgan (U.S. NRC) to John A. Dent, Jr. (Entergy Nuclear Operations, Inc.), "Pilgrim Nuclear Power Station - Issuance of Amendment Regarding Safety Limits to Resolve Pressure Regulator Fail-Open Transient License Amendment Request (TAC NO. MF1382)," dated March 12, 2015. (ADAMS Accession No. ML14272A070)

17. NEDE-24011-P-A, Rev. 22, "General Electric Standard Application for Reactor Fuel (GESTAR II)," Global Nuclear Fuel, dated November 2015.

18. NEDC-33743P, Rev. 0, BWROG-TP-12-001, Rev. 0, "BWR Owners Group Reload Analysis and Core Management Committee SC05-03 Analysis Report," dated April 2012.

Attachment 1

Proposed Technical Specification Changes (Mark-Up)
(1 page follows)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 686~~785~~ psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23.8% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 686~~785~~ psig and core flow ≥ 10% rate core flow:

The Minimum Critical Power Ratio (MCPR) shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.13 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.

Attachment 2

Proposed Technical Specification Changes (Retyped)
For Information Only
(1 page follows)

INFORMATION ONLY

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 686 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23.8% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 686 psig and core flow ≥ 10% rate core flow:

The Minimum Critical Power Ratio (MCPR) shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.13 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.

Attachment 3

Proposed Technical Specification Bases Changes (Mark-Up)
For Information Only
(4 pages follow)

INFORMATION ONLY

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp 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.

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 an MCPR SL 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 SL.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 686~~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 data taken at pressures from 14.7 psia to 800 psia

(continued)

INFORMATION ONLY

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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 > 47.6% RTP. Thus, a THERMAL POWER limit of 23.8% RTP for reactor pressure < 686~~785~~ psig is conservative.

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 Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

(continued)

INFORMATION ONLY

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

6. Main Steam Isolation Valve-Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the Nuclear Steam Supply System and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve – Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analysis of Reference 2, the Average Power Range Monitor Fixed Neutron Flux – High Function, along with the S/RVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 4 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below limits of 10 CFR 50.46. The reactor scram resulting from an MSIV closure due to a Low Main Steam Line Pressure Isolation also ensures reactor power is less than 23.8% RTP before reactor pressure decreases below the Safety Limit 2.1.1 Low Pressure Limit of 686 psig.

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each MSIV has two position switches: one inputs to RPS trip system A while the other inputs to RPS trip system B. Thus, each RPS trip system receives an input from eight Main Steam Isolation Valve – Closure Function channels, each consisting of one position switch. The logic for the Main Steam Isolation Valve – Closure Function is arranged such that either the inboard or outboard valve on three or more of the main steam lines (MSLs) must close in order for a scram to occur.

The Main Steam Isolation Valve – Closure Allowable Value is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

Sixteen channels of the Main Steam Isolation Valve – Closure Function with eight channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude the scram from this Function on a valid signal. This Function is only required in MODE 1 since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs

(continued)

INFORMATION ONLY

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b Main Steam Line Pressure – Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level conditions and the RPV cooling down more than 100°F/hour if the pressure loss is allowed to continue. The Main Steam Line Pressure-Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs during the depressurization transient in order to maintain reactor steam dome pressure > 686 psig. The MSIV closure ~~prior to pressure decreasing below 785 psig, which~~ results in a scram~~due to MSIV closure~~, thus reducing reactor power to < 23.8% RTP.)

The MSL low pressure signals are initiated from four 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 required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be 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. 2).

This Function isolates the Group 6 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. 1). The isolation action, along with the scram function of the RPS,

(continued)