Exelon Generation.

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10 CFR 50.90

RS-15-108

August 18, 2015

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Request for License Amendment to Reduce the Reactor Steam Dome Pressure Specified in the Technical Specification 2.1.1, "Reactor Core SLs"

Reference: GE Nuclear Energy 10 CFR Part 21 Communication SC05-03, "10CFR21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," dated March 29, 2005

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License No. NPF-62 for Clinton Power Station (CPS) Unit 1, Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3, and Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2.

The amendment will revise the CPS, DNPS, and QCNPS Technical Specifications (TS) Section 2.1.1, "Reactor Core SLs," to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits (SLs) 2.1.1.1 and 2.1.1.2. Specifically, the proposed amendment will reduce the reactor steam dome pressure in TS Section 2.1.1.1 and 2.1.1.2 from 785 psig to 685 psig. This change to TS Section 2.1.1 was identified as a result of General Electric (GE) Part 21 report SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," (see referenced document). This change is valid for the NRC approved pressure range pertinent to the critical power correlations applied to the fuel types in use at CPS, DNPS, and QCNPS.

This request is subdivided as follows.

- Attachment 1 provides an evaluation supporting the proposed change.
- Attachments 2, 3, and 4 contain the marked-up Technical Specifications (TS) pages for CPS, DNPS, and QCNPS, respectively, with the proposed changes indicated.
- Attachments 5, 6, and 7 provide the marked-up TS Bases pages for CPS, DNPS, and QCNPS, respectively, with the proposed changes indicated. These attachments are provided for information only.

The proposed change has been reviewed by the Plant Operations Review Committee and approved by the Nuclear Safety Review Board for the respective facilities in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed changes by August 31, 2016. The proposed changes will be implemented within 60 days of issuance of the amendment. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91(b), EGC is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions related to this letter, please contact Mr. Timothy A Byam at (630) 657-2818.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of August 2015.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachments:

Attachment 1:	Evaluation of Proposed Change
Attachment 2:	Markup of Proposed Technical Specifications Pages for CPS
Attachment 3:	Markup of Proposed Technical Specifications Pages for DNPS
Attachment 4:	Markup of Proposed Technical Specification Pages for QCNPS
Attachment 5:	Markup of Proposed Technical Specifications Bases Pages for CPS (For Information Only)

Attachment 6:	Markup of Proposed Technical Specifications Bases Pages for DNPS (For Information Only)
Attachment 7:	Markup of Proposed Technical Specifications Bases Pages for QCNPS (For Information Only)

cc: Regional Administrator – Region III
NRC Senior Resident Inspector – Clinton Power Station
NRC Senior Resident Inspector – Dresden Nuclear Power Station
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
EVALUATION OF PROPOSED CHANGE

Subject: Request for License Amendment to Reduce the Reactor Steam Dome Pressure Specified in the Technical Specification 2.1.1, "Reactor Core SLs"

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License No. NPF-62 for Clinton Power Station (CPS) Unit 1, Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3, and Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2.

The amendment will revise the CPS, DNPS, and QCNPS Technical Specifications (TS) Section 2.1.1, "Reactor Core SLs," to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits (SLs) 2.1.1.1 and 2.1.1.2. Specifically, the proposed amendment will reduce the reactor steam dome pressure in TS Section 2.1.1.1 and 2.1.1.2 from 785 psig to 685 psig. This change to TS Section 2.1.1 was identified as a result of General Electric (GE) Part 21 report SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," (Reference 1). This change is valid for the NRC approved pressure range pertinent to the critical power correlations applied to the fuel types in use at CPS, DNPS, and QCNPS.

2.0 DETAILED DESCRIPTION

In 2005, GE issued 10 CFR Part 21 report SC05-03 identifying the potential vulnerability for the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient event to result in a condition in which TS SL 2.1.1.1 may be exceeded. This does not challenge the fuel cladding integrity or constitute a safety hazard as determined by GE. However, there exists a potential for violation of a reactor core safety limit for the PRFO event. As such, EGC is revising the reactor steam dome pressure TS safety limit consistent with the NRC approved pressure range of critical power correlations for the current fuel designs in CPS, DNPS, and QCNPS reactor cores.

In response to Reference 1, the BWR Owners' Group commissioned development of a methodology for plants to assess the adequacy of their current Main Steam Isolation valve (MSIV) closure at the low-pressure isolation setpoint (LPIS) setting and to provide a set of recommendations for what actions should be taken based on the outcome of their assessment. The methodology 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. The methodology is documented in Reference 2. Based on the results of the studies documented in Reference 2, the low pressure CPS, DNPS and QCNPS TS SL 2.1.1.1 and 2.1.1.2 is proposed to be changed from 785 to 685 psig. The current LPIS setting at these stations is adequate to prevent the reactor pressure from falling below 685 psig in a PRFO event.

The proposed change revises CPS TS SLs 2.1.1.1 and 2.1.1.2 to read as follows.

> "2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 21.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation."

The proposed change revises DNPS Units 2 and 3 TS SLs 2.1.1.1 and 2.1.1.2 to read as follows.

"2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

For two recirculation loop operation, MCPR shall be ≥ 1.12, or for single recirculation loop operation, MCPR shall be ≥ 1.14."

The proposed change revises QCNPS Units 1 and 2 TS SLs 2.1.1.1 and 2.1.1.2 to read as follows.

"2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

For Unit 1, two recirculation loop operation, MCPR shall be ≥ 1.11, or for single recirculation loop operation, MCPR shall be ≥ 1.14.

For Unit 2, two recirculation loop operation, MCPR shall be ≥ 1.12, or for single recirculation loop operation, MCPR shall be ≥ 1.14."

Mark-ups of the above proposed TS changes are provided in Attachments 2, 3, and 4 for CPS, DNPS, and QCNPS, respectively. In addition, mark-ups of the associated TS Bases pages are provided in Attachments 5, 6, and 7 for CPS, DNPS, and QCNPS, respectively. The Bases mark-ups are provided for information only, and do not require NRC approval.

3.0 TECHNICAL EVALUATION

The CPS, DNPS, and QCNPS TS SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). Reactor Core SLs are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur if the SLs are not exceeded. The CPS, DNPS, and QCNPS TS specify SL 2.1.1.1 to require that thermal power shall be less than or equal to a specified rated thermal power (RTP) (i.e., 21.6% RTP for CPS and 25% RTP for DNPS and QCNPS) when reactor steam dome pressure is less than 785 psig (i.e., 800 psia) or core flow is less than 10% of rated core flow. This SL was introduced to preclude the need for Critical Power Ratio (CPR) calculations when reactor steam dome pressure is less than 785 psig (i.e., 800 psia). The thermal power value in CPS, DNPS, and QCNPS TS SL 2.1.1.1 is selected to ensure that thermal power remains well below the fuel assembly critical power for the conditions in which CPR calculations are not performed.

Reactor depressurization transients, such as the 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 predicted that reactor water level would swell during a PRFO transient and 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 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 LPIS. Depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor steam dome pressure could decrease to below 785 psig (i.e., 800 psia) for a few seconds while thermal power exceeds 21.6% RTP for CPS and 25% of rated power for DNPS and QCNPS, which would exceed the conditions in TS SL 2.1.1.1 (Reference 1). The methodology developed to assess the adequacy of the current LPIS setting and to provide a set of recommendations for the actions to be taken is documented in Reference 2. Based on the results of the studies documented in Reference 2, it is proposed that the low pressure CPS, DNPS, and QCNPS TS SL 2.1.1.1 and 2.1.1.2 be changed from 785 to 685 psig. The current LPIS at these stations is adequate to prevent the reactor pressure from falling below 685 psig (i.e., 700 psia) while above the TS SL specified rated thermal power in a PRFO event.

<u>CPS Evaluation</u>
EGC has completed an evaluation that demonstrates the current LPIS setting at CPS is sufficient to preclude steam dome pressure from falling below 685 psig (i.e., 700 psia) while above 21.6% RTP during a PRFO event. This evaluation was performed utilizing the methodology described in Reference 2.

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

DNPS and QCNPS Evaluation

EGC performed an evaluation, based on the Reference 2 methodology, which demonstrates the current LPIS settings at DNPS and QCNPS are sufficient to preclude steam dome pressure from falling below 685 psig (i.e., 700 psia) while above 25% RTP. DNPS and QCNPS currently have full cores of Westinghouse SVEA-96 Optima2 fuel. The corresponding Westinghouse CPR correlation is approved by the NRC for the lower bound pressure limit of 362 psia as documented in WCAP-16081-P-A (Reference 7).

The DNPS and QCNPS units are currently in the process of transitioning to AREVA ATRIUM10XM fuel starting November 2016 (Reference 8). The lower bound pressure of 290.8 psia for the ACE critical power correlation is documented and justified in ANP-10298PA for AREVA ATRIUM10XM fuel (Reference 9). This topical report has been reviewed and approved by the NRC. The AREVA ATRIUM10XM fuel CPR correlation lower bound pressure is less than the proposed 685 psig (i.e., 700 psia).

AREVA also supported in Reference 8, the use of the SPCB critical power correlation for the Optima2 fuel. Justification for applying SPCB-9 branch of the SPCB correlation to Westinghouse SVEA-96 Optima2 fuel by AREVA is provided in FS1-0015517 (Reference 10). A lower bound pressure of 600 psia for applying SPCB-9 correlation to SVEA Optima2 fuel is specified. The use of SPCB-9 correlation for SVEA-96 Optima2 fuel was submitted as part of ATRIUM10XM fuel transition amendment request for NRC approval (Reference 8). The CPR correlation lower bound pressure is less than the proposed 685 psig (i.e., 700 psia).

Summary

Use of 685 psig (i.e., 700 psia) as steam dome pressure limit for TS 2.1.1.1 and TS 2.1.1.2 is supported by the CPR correlations in use for CPS, DNPS, and QCNPS. The minimum steam dome pressure resulting from a PRFO event is demonstrated to be above 685 psig (i.e., 700 psia) using Reference 2 methodology. Revising the Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 reactor steam dome pressure from 785 to 685 psig resolves the potential to violate Reactor Core Safety Limit 2.1.1.1 during a PRFO transient reported in Reference 1. If EGC decides to switch to a different fuel design from those currently in use in the CPS, DNPS, and QCNPS reactor cores, the CPR correlation will be reviewed as part of the normal fuel design change and reload licensing processes. If the CPR correlation for the new fuel design has a lower bound pressure which is higher than the limit specified in the TS, then a LAR will be submitted for NRC review and approval. If the CPR correlation has a lower bound pressure which is lower than the TS limit, then no LAR will be required since the TS would set a conservative lower bound.

Results of the above EGC evaluations show that the current LPIS settings at CPS, DNPS, and QCNPS are adequate to prevent reactor pressure from falling below 685 psig while thermal power is above 21.6% RTP for CPS and 25% RTP for DNPS and QCNPS. CPR correlations currently in use at CPS, DNPS and QCNPS and CPR correlations projected for use at DNPS and QCNPS support a lower bound pressure 685 psig.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, Technical Specifications, provides the regulatory requirements for the content required in the TSs. As stated in 10 CFR 50.36, 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. If any safety limit is exceeded, the reactor must be shut down."

The following General Design Criterion (GDC) is applicable to this amendment request. It should be noted that, although DNPS and QCNPS are not formally committed to the GDC due to the vintage of the stations, an evaluation was performed addressing the DNPS and QCNPS conformance with the GDC. This evaluation is documented in the DNPS and QCNPS Updated Final Safety Analysis Report (UFSAR) Section 3.1, "Conformance with NRC General Design Criteria." This evaluation concluded that DNPS and QCNPS fully satisfies the intent of the (then draft) GDC. CPS is licensed to the 10 CFR 50 Appendix A criteria.

10 CFR 50 Appendix A, GDC 10, "Reactor design," states that 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 reactor core components consist of fuel assemblies, control rods, incore ion chambers and related items. The fuel is designed to provide high integrity over a complete range of power levels including transient conditions. As described above, the CPS, DNPS, and QCNPS TS SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and AOOs. Reactor Core SLs are set such that fuel cladding integrity is maintained and no significant fuel damage is calculated to occur if the SLs are not exceeded.

EGC has evaluated the proposed changes 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 2.1.1.1 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 this, there is reasonable assurance that the health and safety of the public, following approval of this TS change, 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 CPS, DNPS, and QCNPS as documented in the following approved amendments.

On March 11, 2013, Northern States Power Company – Minnesota, submitted a License Amendment request proposing to reduce the reactor steam dome pressure specified in Reactor Core Safety Limit Specification 2.1.1 (Reference 11). The NRC approved amendment 185 for the Monticello Nuclear Generating Plant on November 25, 2014 (Reference 12).

On March 24, 2014, Southern Nuclear Operating Company, submitted an amendment request to revise the Edwin I. Hatch Plant Units 1 and 2 TS Section 2.1.1 to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 (Reference 13). The NRC completed their review and issued amendments 269 and 213 on October 20, 2014 (Reference 14)

On May 28, 2013, Entergy Operations, Inc., submitted an amendment request to revise the River Bend Station TS Section 2.1.1 to reflect a lower reactor steam dome pressure specified in Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2 (Reference 15). The NRC completed their review and issued amendment 182 on December 11, 2014 (Reference 16).

On October 8, 2013, Entergy Nuclear Operations, Inc., proposed an amendment to modify the James A FitzPatrick Nuclear Power Plant TS to reduce the reactor pressure associated with the Reactor Core Safety Limit in TS 2.1.1.1 and TS 2.1.1.2 (Reference 17). The NRC completed their review and issued amendment 309 on February 9, 2015 (Reference 18).

4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License No. NPF-62 for Clinton Power Station (CPS) Unit 1, Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3, and Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2. The amendment will revise the CPS, DNPS, and QCNPS Technical Specifications (TS) Section 2.1.1, "Reactor Core SLs," to reflect a lower reactor steam dome pressure stated for Reactor Core Safety Limits (SLs) 2.1.1.1 and 2.1.1.2. Specifically, the proposed amendment will reduce the reactor steam dome pressure in TS Section 2.1.1.1 and 2.1.1.2 from 785 psig to 685 psig. This change to TS Section 2.1.1 was identified as a result of General Electric (GE) Part 21 report SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit." This change is consistent with the NRC approved pressure range for the critical power correlations applied to the fuel types in use at CPS, DNPS, and QCNPS.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change to the TS for CPS, DNPS Units 2 and 3, and QCNPS Units 1 and 2, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change to the reactor steam dome pressure in the CPS, DNPS, and QCNPS Reactor Core Safety Limits TS 2.1.1.1 and 2.1.1.2 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.

   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 dome pressure safety limit from 785 psig to 685 psig 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 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 General Electric determined that since the Minimum Critical Power Ratio 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 2.1.1.1 and 2.1.1.2 is consistent with and within the capabilities of the applicable NRC approved critical power correlation for the fuel designs in use at CPS, DNPS, and QCNPS. 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 upon the above, EGC 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.

5.0 ENVIRONMENTAL CONSIDERATION

EGC 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, "Standards for Protection Against Radiation." 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, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment needs be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Letter from Jason Post (GE Energy – Nuclear) to U. S. NRC, "10CFR21 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," dated April 2012

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

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

5. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel"

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

7. WCAP-16081-P-A, Revision 0, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2," dated March 2005

8. Letter from Patrick Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Transition to AREVA Fuel," dated February 6, 2015

9. ANP-10298PA Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," dated March 2014

10. FS1-0015517, Revision 2, "Critical Power Correlation and Associated Additive Constants for OPTIMA2 Fuel at Quad Cities Units 1 & 2 and Dresden Units 2 & 3," dated August 2014

11. 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)

12. 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)

13. 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)

14. 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)

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

16. 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

17. 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

18. 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

ATTACHMENT 2

Markup of Proposed Technical Specifications Pages for CPS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785~~ 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 21.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ 685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.12 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:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant manager and the corporate executive responsible for overall plant nuclear safety.

(continued)

ATTACHMENT 3

Markup of Proposed Technical Specifications Pages for DNPS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785~~ 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ 685 psig and core flow ≥ 10% rated core flow:

For two recirculation loop operation, MCPR shall be ≥ 1.12, or for single recirculation loop operation, MCPR shall be ≥ 1.14.

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 ≤ 1345 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 4

Markup of Proposed Technical Specifications Pages for QCNPS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785~~ 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ 685 psig and core flow ≥ 10% rated core flow:

For Unit 1, two recirculation loop operation, MCPR shall be ≥ 1.11, or for single recirculation loop operation, MCPR shall be ≥ 1.14.

For Unit 2, two recirculation loop operation, MCPR shall be ≥ 1.12, or for single recirculation loop operation, MCPR shall be ≥ 1.14.

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 ≤ 1345 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 5

Markup of Proposed Technical Specifications Bases Pages for CPS
(For 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.

<u>2.1.1.1 Fuel Cladding Integrity</u>

GE critical power correlations are applicable for all critical power calculations at pressures ≥ ~~785~~ 685 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)

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 > 41.7% RTP. Thus, a THERMAL POWER limit of 21.6% RTP for reactor pressure < ~~785~~ **685** 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)

ATTACHMENT 6

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

~~The use of the Siemens Power Corporation correlation (ANFB) is valid for critical power calculations at pressures > 600 psia and bundle mass fluxes > 0.1 x 10⁶ lb/hr-ft² (Refs. 2 and 3). The use of the General Electric (GE) Critical Power correlation (GEXL) is valid for critical power calculations at pressures > 785 psig and core flows > 10% (Ref. 4).~~ The use of the Westinghouse (WEC) Critical Power correlation (D4.1.1) is valid for critical power calculations at pressures > 362 psia and bundle mass fluxes > 0.23 x 10^6 lb/hr-ft^2 (Ref. 7). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/hr (approximately a mass velocity of 0.25 X 10^6 lb/hr-ft^2), 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 x 10^3 lb/hr. Full scale critical power test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < ~~785~~ 685 psig is conservative. Although ~~the ANFB correlation is valid at reactor steam dome pressures > 600 psia, and~~ the Westinghouse correlation is valid at reactor steam dome pressures > 362 psia, application of the fuel cladding integrity SL at reactor steam dome pressure < ~~785~~ 685 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the fuel vendor's critical power correlation. References ~~2, 3, 4, 5~~, 7, and 8 describe the methodology used in determining the MCPR SL.

The fuel vendor's critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the fuel vendor's correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.2.2.1.

~~2. ANF-524(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors, (as specified in Technical Specification 5.6.5).~~

~~3. ANF-1125(P)(A) and Supplements 1 and 2, ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, (as specified in Technical Specification 5.6.5).~~

~~4. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR) (as specified in Technical Specification 5.6.5)~~

~~5. ANF-1125(P)(A), Supplement 1, Appendix E, ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).~~

6. 10 CFR 50.67.

7. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2" (as specified in Technical Specification 5.6.5).

8. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (as specified in Technical Specification 5.6.5).

ATTACHMENT 7

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

~~The use of the Siemens Power Corporation correlation (ANFB) is valid for critical power calculations at pressures > 600 psia and bundle mass fluxes > 0.1 x 10^6 lb/hr-ft^2 (Refs. 2 and 3). The use of the General Electric (GE) Critical Power correlation (GEXL) is valid for critical power calculations at pressures > 785 psig and core flows > 10% (Ref. 4).~~ The use of the Westinghouse critical power correlation (D4.1.1) is valid for critical power calculations at pressures > 362 psia and bundle mass fluxes > 0.23 x 10^6 lb/hr-ft^2 (Ref. 8). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/hr (approximately a mass velocity of 0.25 X 10^6 lb/hr-ft^2), 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 x 10^3 lb/hr. Full scale critical power test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < ~~785~~ 685 psig is conservative. Although ~~the ANFB correlation is valid at reactor steam dome pressures > 600 psia, and~~ the Westinghouse D4.1.1 correlation is valid at reactor steam dome pressures > 362 psia, application of the fuel cladding integrity SL at reactor steam dome pressure < ~~785~~ 685 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the fuel vendor's critical power correlation. References ~~2, 3, 4, 5, 6,~~ 8 and 9 describe the methodology used in determining the MCPR SL.

The fuel vendor's critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the fuel vendor's correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this

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BASES (continued)

REFERENCES

1. UFSAR, Section 3.1.2.1.

2. ~~ANF-524(P)(A), Revision 2, Supplement 1, Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, (as specified in Technical Specification 5.6.5).~~

3. ~~ANF-1125(P)(A) and Supplements 1 and 2, ANFB Critical Power Correlation, Advanced Nuclear Fuels Corporation, (as specified in Technical Specification 5.6.5).~~

4. ~~NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR)" (as specified in Technical Specification 5.6.5).~~

5. ~~ANF-1125(P)(A), Supplement 1, Appendix E, ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).~~

6. ~~EMF-1125(P)(A), Supplement 1, Appendix C, ANFB Critical Power Correlation Application for Coresident Fuel, Siemens Power Corporation, (as specified in Technical Specification 5.6.5).~~

7. 10 CFR 50.67.

8. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2" (as specified in Technical Specification 5.6.5).

9. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (as specified in Technical Specification 5.6.5).