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10 CFR 50.90

RS-15-279

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

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

Reference: General Electric Nuclear Energy 10 CFR Part 21 Communication SC05-03, "10 CFR 21 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, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The amendment will revise the LSCS 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 SLs 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 LSCS.

This request is subdivided as follows.

- Attachment 1 provides an evaluation supporting the proposed change.
- Attachment 2 contains the marked-up TS pages for LSCS, with the proposed changes indicated.
- Attachment 3 provides the marked-up TS Bases pages for LSCS, with the proposed changes indicated. This attachment is provided for information only.

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

EGC requests approval of the proposed changes by November 30, 2016. Because the evaluation provided in Attachment 1 is based on complete cores of GNF fuel, implementation of the approved amendment will be staggered between LSCS Units 1 and 2, following the discharge of AREVA fuel. The proposed changes will be implemented for Unit 1 within 30 days of issuance of the amendment. The proposed changes will be implemented for Unit 2 prior to startup following refueling outage L2R16 in February 2017.

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 19th day of November 2015.

Respectfully,

David M. Gullott
Manager – Licensing

Attachments:

Attachment 1:	Evaluation of Proposed Change
Attachment 2:	Markup of Proposed Technical Specifications Page for LaSalle County Station, Units 1 and 2
Attachment 3:	Markup of Proposed Technical Specifications Bases Pages for LaSalle County Station, Units 1 and 2 (For Information Only)

cc: Regional Administrator – Region III
NRC Senior Resident Inspector – LaSalle County 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, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The amendment will revise the LSCS 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 SLs 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 LSCS.

2.0 DETAILED DESCRIPTION

In 2005, GE issued 10 CFR Part 21 report SC05-03 (Reference 1) identifying the potential vulnerability for the Pressure Regulator Failure Maximum Demand (Open) (PRFO) transient event to result in a condition in which TS SL 2.1.1.1 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 full core of GNF fuel designs in the LSCS Unit 1 and 2 reactors.

The proposed change revises LSCS 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, MCPR shall be ≥ 1.13 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

For Unit 2, MCPR shall be ≥ 1.14 for two recirculation loop operation or ≥ 1.17 for single recirculation loop operation."

Mark-ups of the above proposed TS changes are provided in Attachment 2 for LSCS, Units 1 and 2. In addition, mark-ups of the associated TS Bases pages are provided in Attachment 3 for LSCS, Units 1 and 2. The Bases mark-ups are provided for information only, and do not require NRC approval.

3.0 TECHNICAL EVALUATION

The LSCS, Units 1 and 2 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 LSCS TS specify SL 2.1.1.1 to require that thermal power shall be less than or equal to 25% rated thermal power (RTP) when the 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 LSCS 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 low-pressure isolation setpoint (LPIS). Depending upon the plant-specific response to a PRFO, including the value of the LPIS, reactor steam dome pressure could decrease to below 785 psig (i.e., 800 psia) for a few seconds while thermal power exceeds 25% of rated power, which would exceed the conditions in LSCS TS SL 2.1.1.1. This issue was identified in SC05-03 (Reference 1).

GE Hitachi Nuclear Energy (GEH) performed an evaluation for the PRFO event to assess the adequacy of the LPIS setting as documented in Reference 2. In this evaluation, GEH concludes that the approved GEXL application range correlation lower boundary of 700 psia is not violated. The results of the LSCS PRFO analysis for all cases show significant margin to 700 psia (i.e., 685 psig). The minimum margin is greater than 15 psi for all cases analyzed. The Reference 2 PRFO analysis is cycle-independent and applicable to all future reloads assuming no plant modifications are made that would adversely impact the analysis basis. A change to the LPIS analytical limit (AL) is not required to address SC05-03.

LSCS Units 1 and 2 currently have a mixed core of AREVA ATRIUM-10 and Global Nuclear Fuel (GNF) GNF2 fuel. In addition, four GNF3 Lead Use Assemblies (LUA) and two AREVA ATRIUM 10XM LUAs are in the LSCS Unit 2 core. On February 27, 2013, the U. S. Nuclear Regulatory Commission (NRC) issued Amendment No. 192 to LSCS, Unit 2 (Reference 3). This amendment changed the safety limit minimum critical power ratio (SLMCPR) in TS Section 2.1.1 to support the LSCS Unit 2 transition to GNF2 fuel from the resident AREVA ATRIUM-10 fuel. In support of this amendment, EGC proposed to use a version of the critical heat flux (CHF) correlation (i.e., GEXL97) for the SLMCPR calculation not approved for generic use. The NRC determined that EGC adequately justified the use of the version of the CHF correlation, except for the expanded range of applicability. Therefore, in order to find the EGC approach acceptable, the NRC imposed a license condition to limit the SLMCPR calculation to the range of applicability for the CHF correlation to that which was previously approved by the NRC staff in a letter dated January 14, 2004 (Reference 4).

In Reference 3, the NRC imposed license condition 2.C.(34) on LSCS Unit 2 which restricts use of the GEXL97 correlation for AREVA fuel for the SLMCPR calculation. This restriction limits applicability of the GEXL97 correlation low pressure range to 800 psia for AREVA fuel (Reference 4). However, LSCS Unit 1 is currently operating with two batches of GNF2 and one batch of ATRIUM-10 fuel which is going to be discharged in the March 2016 refueling outage (i.e., LSCS Unit 1 will have an all GNF2 core following the March of 2016 refueling outage). LSCS Unit 2 is currently operating with one batch of ATRIUM-10 fuel and two batches of GNF2 fuel as well as the four GNF3 and two ATRIUM 10XM LUAs. By February 2017 (i.e., the next Unit 2 refueling outage), the core will contain all GNF2 fuel and GNF3 LUAs. By the time this amendment is implemented, both reactor cores at LSCS will contain only GNF fuel and, therefore, only GNF fuel is addressed in this evaluation.

GEH uses the GEXL correlation to perform CPR calculations for all the GNF fuel types in use at LSCS. The GEXL17 correlation, with the lower bound limit of 700 psia (i.e., 685 psig), is applicable to GNF2 fuel. The GEXL17 correlation is documented and justified in NEDC-33292P for GNF2 Fuel (Reference 5). This lower bound limit is discussed and NEDC-33292P is referenced in NEDC-33270P (Reference 6). 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 7). Therefore, the use of 700 psia (i.e., 685 psig) as lower bound limit for GNF2 fuel has been approved by the NRC for use per NEDE-24011-P-A by reference.

GNF3 LUAs in the LSCS Unit 2 reactor core are inserted into core locations projected to be non-limiting with respect to compliance with Linear Heat Generation Rate (LHGR), Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), and MCPR limits with planned, steady state control rod patterns. The GEXL17 correlation is conservatively applied in establishing MCPR operating limits for GNF3 LUAs (Reference 8).

Use of 685 psig (i.e., 700 psia) as steam dome pressure limit for TS SLs 2.1.1.1 and TS 2.1.1.2 is supported by the CPR correlations in use for LSCS. The minimum steam dome pressure resulting from a PRFO event will remain above 685 psig as demonstrated by the evaluation in Reference 2. Revising the Reactor Core SLs 2.1.1.1 and 2.1.1.2 reactor steam dome pressure from 785 to 685 psig resolves the 10 CFR Part 21 condition concerning the potential to violate Reactor Core SL 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 LSCS, Units 1 and 2 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 an amendment request 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 amendment request 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 LSCS are adequate to prevent reactor pressure from falling below 685 psig while thermal power is above 25% RTP. CPR correlations currently in use at LSCS support a lower bound pressure of 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."

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 LSCS 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 SLs 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 LSCS 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 9). The NRC approved amendment 185 for the Monticello Nuclear Generating Plant on November 25, 2014 (Reference 10).

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 11). The NRC completed their review and issued amendments 269 and 213 on October 20, 2014 (Reference 12).

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 13). The NRC completed their review and issued amendment 182 on December 11, 2014 (Reference 14).

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 15). The NRC completed their review and issued amendment 309 on February 9, 2015 (Reference 16).

On August 18, 2015, Exelon Generation Company, LLC, submitted a license amendment request to modify the Clinton Power Station, Dresden Nuclear Power Station, and the Quad Cities Nuclear Power Station TS to reduce the reactor pressure associated with the Reactor Core Safety Limit in TS 2.1.1.1 and TS 2.2.1.2 (Reference 17).

4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The amendment will revise the LSCS, Units 1 and 2 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 SLs 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 valid for the NRC approved pressure range pertinent to the critical power correlations applied to the fuel types in use at LSCS.

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 LSCS, 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 LSCS 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 LSCS, Units 1 and 2. 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. 002N7147-R0, Revision 0, "LaSalle MSL Pressure – Low Setpoint Evaluation," GE Hitachi Nuclear Energy, September 2015

3. Letter from U. S. NRC to Michael J. Pacilio (Exelon Nuclear), "LaSalle County Station, Unit 2 – Issuance of Amendment No. 192 Regarding Technical Specification Change for Safety Limit Minimum Critical Power Ratio (TAC No. ME9769)," dated February 27, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 13050A637)

4. Letter from U. S. NRC to John L. Skolds (Exelon Generation Company, LLC), "LaSalle County Station, Units 1 and 2 – Correction to Issuance of Amendments (TAC Nos. MB9888 and MB9889)," dated January 14, 2004 (ADAMS Accession No. ML040130278)

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

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

7. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (GESTAR II) Revision 17

8. 002N3086.1, Revision 0, "Technical Evaluation to Support Introduction of GNF3 Lead Use Assemblies (LUAs) in LaSalle County Station, Unit 2," Global Nuclear Fuel, December 2014

9. 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)

10. 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)

11. 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)

12. 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)

13. 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

14. 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

15. 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

16. 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

17. Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment to Reduce the Reactor Steam Dome Pressure Specified in the Technical Specification 2.1.1, 'Reactor Core SLs'," dated August 18, 2015

ATTACHMENT 2

Markup of Proposed Technical Specifications Page for
LaSalle County Station, Units 1 and 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 <u>Reactor Core SLs</u>

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, MCPR shall be ≥ 1.13 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

For Unit 2, MCPR shall be ≥ 1.14 for two recirculation loop operation or ≥ 1.17 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 <u>Reactor Coolant System Pressure SL</u>

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

Markup of Proposed Technical Specifications Bases Pages for
LaSalle County Station, Units 1 and 2
(For Information Only)

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

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 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. Compatible ATRIUM-10 information is documented in Reference 3.

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.

(continued)