Exelon Generation

4300 Winfield Road
Warrenville, IL 60555
630 657 2000 Office

10 CFR 50.90

RS-16-076

April 11, 2016

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

References:

(1) Letter from David M. Gullott (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Request for License Amendment to Reduce the Reactor Steam Dome Pressure Specified in the Technical Specification 2.1.1, 'Reactor Core SLs'," dated November 19, 2015

(2) Email from Bhalchandra Vaidya (U. S. Nuclear Regulatory Commission) to Lisa A. Simpson and David M. Gullott (Exelon Generation Company, LLC), "LaSalle, RAIs on LAR Re: TS 2.1.1.1 & TS 2.1.1.2 Change, CAC Nos. MF7109 & MF7110," dated February 22, 2016

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change 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. 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.

The NRC reviewed the license amendment request and identified the need for additional information in order to complete its evaluation of the subject amendment request. The request for additional information (RAI) was sent from the NRC to EGC by electronic mail message on February 22, 2016 (Reference 2).

The EGC response to the RAI is provided in Attachment 1. Attachment 2 provides revised TS markups. For information, Attachment 3 provides revised TS Bases markups.

EGC has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration that were previously provided to the NRC in Attachment 1 of the Reference 1 letter. EGC has concluded that the information provided in this supplement does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92. In addition, EGC has concluded that the information in this supplement does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments contained in this letter. Should you have any questions concerning 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 11th day of April 2016.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachments:
1. Response to Request for Additional Information
2. Revised Markup of Proposed Technical Specifications Pages
3. Revised Markup of Proposed Technical Specifications Bases Pages (For Information Only)

cc: Regional Administrator – Region III
NRC Senior Resident Inspector – LaSalle County Station
Illinois Emergency Management Agency – Division of Nuclear Safety

**NRC RAI 1:**

In the License Amendment Request (LAR) submitted on November 19, 2015, it was stated that the GEXL17 correlation is conservatively applied in establishing MCPR operating limits for GNF3 Lead Use Assemblies (LUAs). Please provide the following additional clarification:

a) Has GEXL17 correlation been approved for application to GNF3 fuel? If not, then identify the NRC approved Critical Power Ratio (CPR) correlation that is applicable to GNF3 fuel; including the value for the lower pressure limit of the correlation.

b) If there is not an approved CPR correlation available for GNF3, then justify how GEXL17 correlation is conservatively applied in establishing MCPR operating limits for GNF3 LUAs, as stated in the LAR.

**Response:**

a) The GEXL17 correlation has not been approved for application to GNF3 fuel. The GNF3 fuel bundle is a new fuel design that was loaded as part of the LaSalle County Station (LSCS) Unit 2 Reload 15 Cycle 16 during the 2015 refueling outage. These bundles, also referred to as GNF3 Lead Use Assemblies (LUAs), are intended to be in operation as part of a joint program with Global Nuclear Fuel – Americas, LLC (GNF). The GNF3 fuel was designed for mechanical, nuclear, and thermal-hydraulic compatibility with previous GE/GNF fuel designs. On January 20, 2015, Exelon Generation Company, LLC (EGC) transmitted a letter to the NRC (Reference 1) notifying the NRC that EGC intended to load four GNF3 LUAs as part of the LSCS Unit 2 Reload 15 (Cycle 16). Attachment 1 to Reference 1 contains the proprietary GNF report, NEDC-33862P, "GNF3 Lead Use Assembly for LaSalle County Station, Unit 2," that contains a discussion of the GNF3 licensing analyses. Specifically, Section 3.1 of this report contains the technical discussion and provides proprietary information in support of the GEXL17 applicability to the GNF3 fuel. The GEXL17 correlation is applicable to GNF3 fuel and is conservatively applied in establishing its MCPR operating limits. The GEXL17 correlation, which is documented in NEDC-33292P, "GEXL17 Correlation for GNF2 Fuel," Revision 3 (Reference 2), has a lower bound pressure limit of 700 psia applicable to GNF3 fuel.

b) As stated in the November 19, 2015 LAR (Reference 3), the GEXL17 correlation is conservatively applied in establishing MCPR operating limits for GNF3 LUAs. There are four GNF3 LUAs in the LSCS Unit 2 core that were inserted into core locations projected to be non-limiting. Cycle-specific analyses were performed for LSCS Unit 2 Cycle 16 to establish fuel operating limits for the LUAs. As stated in Section 3.0 of Attachment 1 to Reference 3, the analyses ensured that the core loading had been designed such that the GNF3 LUAs would not be the most limiting fuel assemblies at any time during Cycle 16 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. Additionally, licensing analysis will be performed for GNF3 LUAs for each cycle of their operation, wherein the effect of the GNF3 LUAs is considered for each of the appropriate licensing events and anticipated operational occurrences to establish appropriate reactor core operating limits. The LUAs have been analyzed using the GNF NRC approved methods documented in GESTAR II (Reference 4). These methods are capable of analyzing all the GNF3 LUA features.

As noted above in the response to RAI 1.a, Section 3.1 of Attachment 1 to Reference 1 provides the basis for use of GEXL17 for GNF3 fuel. Technical discussions in Attachment 1 to Reference 1 substantiate the applicability of GEXL17 for use with the GNF3 LUAs. All evaluations performed in support of the introduction of the GNF3 LUAs were performed in accordance with the approved licensing methods in GESTAR II (Reference 4).

**NRC RAI 2:**

In page 4, 2nd paragraph, of the LAR, it was stated,

> The GEXL17 correlation, with the lower bound limit of 700 psia (i.e., 685 psig), is applicable to GNF2 fuel.

In converting to psig, this lower bound pressure is approximately 685.3 psig. As such, the 685 psig value specified in the proposed TS change is slightly outside the pressure range in which the GEXL17 correlation is valid for GNF2 fuel. Please provide further justification for the proposed 685 psig value; or propose a revised pressure value for this TS change that is supported by the GEXL17 correlation (e.g., 686 psig).

**Response:**

In response to this request, EGC has decided to reference the lower bound pressure limit for the critical power correlation in absolute pressure (i.e., 700 psia) for the GNF2 and GNF3 LUAs fuel currently used in the LSCS, Units 1 and 2 cores. This will ensure the proposed Technical Specifications (TS) are consistent with the lower bound limit for the GEXL17 correlation.

Based on the above, EGC proposes to revise the lower bound pressure limit for the reactor core safety limits specified in TS 2.1.1.1 and TS 2.1.1.2 to reference the absolute pressure value of 700 psia.

Attachment 2 provides a copy of the revised marked up TS pages that reflect the proposed change. For information only, a copy of the revised marked up TS Bases pages that reflect the proposed change are provided in Attachment 3.

**References:**

1. Letter from David M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "LaSalle County Station Introduction of Lead Use Assemblies," dated January 20, 2015 (ADAMS Accession No. ML15020A660)

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

3. Letter from David M. Gullott (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Request for License Amendment to Reduce the Reactor Steam Dome Pressure Specified in the Technical Specification 2.1.1, 'Reactor Core SLs'," dated November 19, 2015 (ADAMS Accession No. ML15324A309)

4. Global Nuclear Fuel, "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A-20, dated December 2013

ATTACHMENT 2

Revised Markup of Proposed Technical Specifications Pages

# 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 psig~~700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785 psig~~700 psia 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 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

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

# B 2.0 SAFETY LIMITS (SLs)

## B 2.1.1 Reactor Core SLs

BASES

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BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

(continued)

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.

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

APPLICABLE SAFETY ANALYSES

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

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

Cores with fuel that is all from one vendor utilize that vendor's critical power correlation for determination of MCPR. For cores with fuel from more than one vendor, the MCPR is calculated for all fuel in the core using the licensed critical power correlations. This may be accomplished by using each vendor's correlation for the vendor's respective fuel. Alternatively, a single correlation can be used for all fuel in the core. For fuel that has not been manufactured by the vendor supplying the critical power correlation, the input parameters to the reload vendor's correlation are adjusted using benchmarking data to yield conservative results compared with the critical power correlation results from the co-resident fuel.

(continued)

BASES

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 psig~~700 psia 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 psig~~700 psia 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)

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

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.

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 reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

(continued)

BASES (continued)

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| SAFETY LIMITS | The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations. |
| APPLICABILITY | SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES. |
| SAFETY LIMIT VIOLATIONS | <u>2.2</u><br><br>Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal. |
| REFERENCES | 1. 10 CFR 50, Appendix A, GDC 10.<br><br>2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).<br><br>3. NEDC-33106P-A, "GEXL97 Correlation Applicable to ATRIUM-10 Fuel," Revision 2, June 2004.<br><br>4. 10 CFR 50.67. |