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RS-16-077 10 CFR 50.90

April 14, 2016

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

References: (1) 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 (ADAMS Accession No. ML15231A097)

(2) Letter from U.S. NRC to Bryan Hanson (Exelon Generation Company, LLC), "Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2 – Request for Additional Information Regarding License Amendment Request to Revise the Reactor Steam Dome Pressure in Technical Specification 2.1.1, "Reactor Core SLs" (CAC Nos. MF6640–MF6644)," dated March 15, 2016

In Reference 1, Exelon Generation Company, LLC (EGC) requested 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 proposed change 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. 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 CPS, DNPS, and QCNPS.

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. In Reference 2, the NRC requested that EGC provide additional information to support their review of the subject amendment request (i.e., Reference 1). The requested information is provided in the attachment to this letter.

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 14th day of April 2016.

Respectfully,

Patrick R. Simpson
Manager – Licensing
Exelon Generation Company, LLC

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

By application dated August 18, 2015, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 15231A097), Exelon Generation Company, LLC (the licensee) requested a change to the facility operating licenses for Clinton Power Station (CPS), Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2. The proposed change revises the reactor steam dome pressure specified in Technical Specification (TS) 2.1.1, "Reactor Core SLs [Safety Limits]," for each of these facilities. Specifically, the licensee proposes to reduce the reactor steam dome pressure from 785 pounds per square inch gauge (psig) to 685 psig.

The U.S. Nuclear Regulatory Commission (NRC) has reviewed the application and determined that the information below is needed to complete its review.

RAI-1

The licensee proposes to reduce the reactor steam dome pressure specified in TS 2.1.1.1 and TS 2.1.1.2 from 785 psig to 685 psig based on the lower-bound pressure for the critical power correlation for the fuel currently used in the reactor cores. The application states that CPS has a mixed core of GE14 and GNF2 fuel which are produced by Global Nuclear Fuel - Americas, LLC (GNF). For the GNF2 fuel, the application references GNF reports NEDC-33292P, NEDC-33270P, and NEDE-24011-P-A as the basis supporting the proposed change. For the GE14 fuel, the application references GNF reports NEDC-32851P-A as the basis supporting the proposed change.

Section 3.8.3 of GNF report NEDC-33270P discusses the critical power correlation for GNF2 fuel (i.e., GEXL17 correlation). This section includes the pressure range over which the GEXL17 correlation is valid for GNF2 fuel consistent with the information provided in Table 5-4 of GNF2 report NEDC-33292P. In Section 3.0 of Attachment 1 to the application, the licensee states that the lower bound pressure limit for the GEXL17 correlation is 700 pounds per square inch atmospheric (psia).

GNF report NEDC-32851P-A discusses the critical power correlation for GE14 fuel (i.e., GEXL14 correlation). Similar to the GEXL17 correlation, Section 5.2 of the report states that the lower bound pressure limit for the GEXL14 correlation is 700 psia.

Converting 700 psia to psig, the lower bound pressure for the GEXL 17 and GEXL14 correlations is approximately 685.3 psig. As such, the 685 psig value specified in the proposed TS change at CPS is slightly outside the pressure range in which the GEXL17 and GEXL14 correlations are valid for GNF2 and GE14 fuel. Provide further justification for the proposed 685 psig value at CPS or propose a revised pressure value for this TS change that is supported by the GEXL17 and GEXL14 correlations (e.g., 700 psia).

Response:

As described in the original Exelon Generation Company, LLC (EGC) license amendment request (ADAMS Accession No. ML15231A097), Dresden Nuclear Power Station (DNPS) Units 2 and 3 and Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2 currently have full cores of Westinghouse SVEA-96 Optima2 fuel. Since DNPS and QCNPS do not have any GE14 or GNF2 fuel loaded in their cores, the above request is not applicable to these two stations. The following response is provided to the above request regarding the fuel loaded in the CPS Unit 1 core.

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 GE14 fuel currently used in the CPS Unit 1 core. This will ensure the proposed Technical Specification (TS) is consistent with the lower bound limit for the GEXL17 and GEXL14 correlations.

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.

ATTACHMENT 2

Revised Markup of Proposed CPS 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 ≤ 21.6% RTP.

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

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ATTACHMENT 3

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

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

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.

APPLICABLE SAFETY ANALYSES

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

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

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ ~~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 28×10^3 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 psig~~700 psia 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.

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BASES

APPLICABLE SAFETY ANALYSES (continued)	<u>2.1.1.3</u> <u>Reactor Vessel Water Level</u> During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active 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 less than two-thirds 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 also to provide adequate margin for effective action.
SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to 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 perforation.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	<u>2.2.1</u> If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3). <u>2.2.2</u> Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.2 (continued)

with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal. In the event reactor vessel water level is below the top of active irradiated fuel, water level would normally be restored by manually initiating Emergency Core Cooling Systems.

2.2.3

If any SL is violated, the CPS Plant Manager and the CPS Site Vice President shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 5). A copy of the report shall also be submitted to the CPS Plant Manager and the CPS Site Vice President.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR-II," (latest approved revision).
3. 10 CFR 50.72.
4. 10 CFR 100.
5. 10 CFR 50.73.