ATTACHMENT 1
Evaluation of Proposed Change

1.0 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 Renewed Facility Operating License No. DPR-30 for Quad Cities Nuclear Power Station, Unit 2. The proposed change revises the values of the safety limit minimum critical power ratio (SLMCPR) in Technical Specification (TS) Section 2.1.1, "Reactor Core SLs." Specifically, the proposed change would require that for Unit 2, the minimum critical power ratio (MCPR) for Global Nuclear Fuel (GNF) fuel shall be ≥ 1.09 for two recirculation loop operation, or ≥ 1.10 for single recirculation loop operation. Additionally, the proposed change would require that MCPR for Westinghouse fuel shall be ≥ 1.11 for two recirculation loop operation, or ≥ 1.13 for single recirculation loop operation. The proposed change is described below.

EGC evaluated submittals currently under review by the NRC to determine the impact of the proposed change. In Reference 1, EGC requested NRC approval of a license amendment that would, in part, revise TS Section 5.6.5, "Core Operating Limits Report (COLR)," to allow Westinghouse methodologies, which have been generically approved by the NRC, to be used for core reload evaluations. The methodology used for SLMCPR evaluations was part of the Reference 1 submittal. Therefore, approval of the proposed change herein is predicated on approval of the Reference 1 request. Reference 1 is currently under NRC review, and NRC approval has been requested prior to QCNPS Unit 2 startup for Cycle 19.

2.0 PROPOSED CHANGE

TS Section 2.1.1.2 specifies the value for the SLMCPR. For QCNPS, Unit 2, the values specified are as follows.

> For Unit 2, MCPR shall be ≥ 1.09 for two recirculation loop operation, or ≥ 1.10 for single recirculation loop operation.

The proposed change will revise TS Section 2.1.1.2 for Unit 2 to read as follows.

> For Unit 2, MCPR for GNF fuel shall be ≥ 1.09 for two recirculation loop operation, or ≥ 1.10 for single recirculation loop operation. MCPR for Westinghouse fuel shall be ≥ 1.11 for two recirculation loop operation, or ≥ 1.13 for single recirculation loop operation.

Attachment 2 provides the marked-up TS page indicating the proposed change. Attachment 3 provides the retyped TS page incorporating the proposed change. Attachment 4 provides the marked-up TS Bases pages for informational purposes.

3.0 BACKGROUND

The fuel cladding integrity SLMCPR is established to assure that at least 99.9% of the fuel rods in the core do not experience boiling transition during an anticipated operational occurrence (AOO). To determine the explicit value for the cycle specific safety limit, a full core statistical analysis is performed. The core model incorporates the uncertainty in the measurement of core operating parameters, critical power ratio (CPR) calculation uncertainties, and the statistical

uncertainty associated with the fuel vendor's correlation. The number of rods that might experience boiling transition as a function of the nominal MCPR is calculated.

The Global Nuclear Fuel (GNF) NRC-approved methodology (i.e., References 2 and 3) was used previously to determine the appropriate SLMCPR values for the current QCNPS Unit 2 fuel cycle (i.e., Cycle 18). The Cycle 18 core is a mixed core containing both GNF GE14 fuel and Framatome-ANP (FANP) ATRIUM-9B fuel assemblies. Consistent with the GNF methodology, the resulting SLMCPR values for Cycle 18 apply to all fuel types in the core, such that the same SLMCPR values are applied to both the GE14 and ATRIUM-9B fuel.

For Cycle 19, EGC will load Westinghouse SVEA-96 Optima2 fuel assemblies in QCNPS Unit 2. Therefore, the Westinghouse NRC-approved methodology (i.e., Reference 4) was used to determine the SLMCPR values for Cycle 19. Unlike the GNF methodology, the Westinghouse methodology generates a unique SLMCPR value for each fuel product line present in the core. Since Cycle 19 will be a mixed core containing both GE14 and SVEA-96 Optima2 fuel assemblies, the proposed change specifies unique SLMCPR values for the two fuel types. There will be no ATRIUM-9B fuel assemblies in the core for Cycle 19.

4.0 TECHNICAL ANALYSIS

Attachment 5 provides technical information to support the proposed change. A description of the SLMCPR evaluation for QCNPS Unit 2 Cycle 19, as well as a summary of the Westinghouse establishment of a CPR correlation for GNF GE14 fuel is provided in Attachment 5.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License No. DPR-30 for Quad Cities Nuclear Power Station, Unit 2. The proposed change revises the values of the safety limit minimum critical power ratio (SLMCPR) in TS Section 2.1.1, "Reactor Core SLs." Specifically, the proposed change would require that for Unit 2, the minimum critical power ratio (MCPR) for Global Nuclear Fuel (GNF) fuel shall be $\geq$ 1.09 for two recirculation loop operation, or $\geq$ 1.10 for single recirculation loop operation. Additionally, the proposed change would require that MCPR for Westinghouse fuel shall be $\geq$ 1.11 for two recirculation loop operation, or $\geq$ 1.13 for single recirculation loop operation.

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 QCNPS, Unit 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 probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC-approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change conservatively establishes the SLMCPR for QCNPS, Unit 2, Cycle 19 such that the fuel is protected during normal operation and during plant transients or anticipated operational occurrences (AOOs).

   Changing the SLMCPR does not increase the probability of an evaluated accident. The change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

   The proposed change revises the SLMCPR to protect the fuel during normal operation as well as during plant transients or AOOs. Operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criterion (i.e., that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs) is met. Since the proposed change does not affect operability of plant systems designed to mitigate any consequences of accidents, the consequences of an accident previously evaluated are not expected to increase.

   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

   Creation of the possibility of a new or different kind of accident would require creating one or more new accident precursors. New accident precursors may be

created by modifications of plant configuration, including changes in allowable modes of operation. The proposed change does not involve any plant configuration modifications or changes to allowable modes of operation. The proposed change to the SLMCPR assures that safety criteria are maintained for QCNPS, Unit 2, Cycle 19.

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 SLMCPR provides a margin of safety by ensuring that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. The proposed change will ensure the appropriate level of fuel protection by continuing to ensure that at least 99.9% of the fuel rods do not experience transition boiling during normal operation and AOOs if the MCPR limit is not violated. Additionally, operational limits will be established based on the proposed SLMCPR to ensure that the SLMCPR is not violated. This will ensure that the fuel design safety criteria (i.e., that no more than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated) are met.

Therefore, the proposed change does 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.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, "Technical specifications," paragraph (c)(1), requires that power reactor facility TS include safety limits for process variables that protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. The fuel cladding integrity SLMCPR is established to assure that at least 99.9% of the fuel rods in the core do not experience boiling transition during normal operation and AOOs. Thus, SLMCPR is required to be contained in TS.

10 CFR 50, Appendix A, General Design Criterion (GDC) 10 requires that the reactor core and associated coolant, control, and protection systems 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 AOOs. To ensure compliance with GDC 10, EGC has performed the plant-specific SLMCPR analyses using NRC-approved methodologies as prescribed in NUREG-0800, Standard Review Plan Section 4.4. The SLMCPR ensures that sufficient conservatism exists in the operating MCPR limit such that, in the event of an AOO, there is a reasonable expectation that at least 99.9% of the fuel rods in the core will avoid boiling transition for the power distribution within the core including all uncertainties.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 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.

7.0 REFERENCES

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Transition to Westinghouse Fuel," dated June 15, 2005

2. Letter from F. Akstulewicz (NRC) to G. A. Watford (GE), "Acceptance for Referencing of Licensing Topical Reports NEDC-32601P, "Methodology and Uncertainties for Safety Limit MCPR Evaluations; NEDC-32694P, "Power Distribution Uncertainties for Safety Limit MCPR Evaluation; and Amendment 25 to NEDE-24011-P-A on Cycle Specific Safety Limit MCPR (TAC Nos. M97490, M99069, and M97491)," dated March 11, 1999

3. NEDO-10958-A, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," dated January 1977

4. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," dated July 1996

ATTACHMENT 2
Markup of Proposed Technical Specifications Page

QUAQ CITIES NUCLEAR POWER STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-30

<u>REVISED TECHNICAL SPECIFICATIONS PAGE</u>

2.0-1

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 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 and core flow ≥ 10% rated core flow:

For Unit 1, MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

~~For Unit 2, MCPR shall be ≥ 1.09 for two recirculation loop operation, or ≥ 1.10 for single recirculation loop operation.~~

For Unit 2, MCPR for GNF fuel shall be ≥ 1.09 for two recirculation loop operation, or ≥ 1.10 for single recirculation loop operation. MCPR for Westinghouse fuel shall be ≥ 1.11 for two recirculation loop operation, or ≥ 1.13 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 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 3
Retyped Technical Specifications Page for Proposed Change

QUAQ CITIES NUCLEAR POWER STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-30

REVISED TECHNICAL SPECIFICATIONS PAGE

2.0-1

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 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 and core flow ≥ 10% rated core flow:

For Unit 1, MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

For Unit 2, MCPR for GNF fuel shall be ≥ 1.09 for two recirculation loop operation, or ≥ 1.10 for single recirculation loop operation. MCPR for Westinghouse fuel shall be ≥ 1.11 for two recirculation loop operation, or ≥ 1.13 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 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 Technical Specifications Bases Pages

QUAQ CITIES NUCLEAR POWER STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-30

<u>REVISED TECHNICAL SPECIFICATIONS BASES PAGES</u>

B 2.1.1-3
B 2.1.1-4
B 2.1.1-6

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). 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:

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).

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 psig is conservative. Although the ANFB correlation is valid at reactor steam dome pressures > 600 psia, application of the fuel cladding integrity SL at reactor steam dome pressure < 785 psig is conservative.

and the Westinghouse D4.1.1 correlation is valid at reactor steam dome pressures > 362 psia,

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 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

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BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

in the fuel vendor's critical power correlation. ~~References 2, 3, 4, 5 and 6~~ describe the methodology used in determining the MCPR SL.

References 2, 3, 4, 5, 6, and 9

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 reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that

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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 100.

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).