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October 1, 2008

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Additional Information Supporting Request to Revise Containment Tendon Surveillance Program

References:

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "License Amendment Request to Revise Containment Tendon Surveillance Program," dated April 9, 2008

2. Letter from Marshall J. David (U.S. NRC) to Charles G. Pardee, "Braidwood Station, Units 1 and 2, and Byron Station, Unit Nos. 1 and 2 Request for Additional Information Related to Containment Tendon Surveillance Program," dated September 9, 2008

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2. The proposed change revises Technical Specifications (TS) Section 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program," and Section 5.6.8, "Tendon Surveillance Report," for consistency with the requirements of 10 CFR 50.55a, "Codes and standards," paragraph (g)(4) for components classified as Code Class CC. Specifically, the proposed changes replace or delete the reference to the specific ASME Code year for the tendon surveillance program with a requirement to use the applicable ASME Code and addenda as required by 10 CFR 50.55a(g)(4).

In Reference 2, the NRC provided questions to EGC to support NRC review of Reference 1. The attachments of this letter provide the requested responses.

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 Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the additional information provided in this submittal 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 submittal. Should you have any questions concerning this letter, please contact Ms. Tricia Mattson at (630) 657-2813.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the first day of October 2008.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Response to Request for Additional Information
2. Markup of Proposed Technical Specifications Pages for Braidwood Station
3. Markup of Proposed Technical Specifications Pages for Byron Station
4. Markup of Proposed Technical Specifications Bases Pages for Braidwood Station
5. Markup of Proposed Technical Specifications Bases Pages for Byron Station

NRC Request 1

The NRC staff notes an inconsistency when comparing the wording used in the second sentence of revised Technical Specification (TS) Bases Section SR 3.6.1.2 and in revised TS Section 5.5.6. Revised TS Bases Section SR 3.6.1.2 states that "Testing and Frequency are consistent with the requirements of 10 CFR 50.55a, "Codes and standards" (Ref. 4)." Revised TS Section 5.5.6 states that "The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC."

The NRC staff also notes an inconsistency when comparing the wording used in the second sentence of revised TS Bases Section SR 3.6.1.2 and in Bases Section SR 3.6.1.2 of NUREG-1431, Revision 3.1. Bases Section SR 3.6.1.2 of NUREG-1431 states that "Testing and Frequency are in accordance with the ASME Code, Section XI, Subsection IWL (Ref. X), and applicable addenda as required by 10 CFR 50.55a."

Please justify or resolve these inconsistencies.

Response

To be consistent, the proposed wording describing acceptance criteria for testing and frequency will be "The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC," for both the second sentence of revised TS Bases Section SR 3.6.1.2, and TS Section 5.5.6. The revised markup TS Bases are attached and reflect this change. Attachments 2 through 5 provide revised TS and TS Bases mark-ups reflecting these changes.

NRC Request 2

The NRC staff notes that the reference to Regulatory Guide (RG) 1.35.1 has been removed from revised TS Section 5.5.6 and the revised TS Bases Section SR 3.6.1.2. The staff notes that the last sentence of SR 3.6.1.2 was revised to read: "Predicted tendon lift off forces will be determined in accordance with the ASME Code, Section XI, Subsection IWL (Ref. 5), and applicable addenda as required by 10 CFR 50.55a." The NRC staff finds that the ASME Code, Section XI, Subsection IWL does not provide guidance with regard to the determination of predicted tendon forces. Please resolve this inconsistency. Also, please provide information regarding the method that will be used for determining the predicted prestressing tendon forces for the Tendon Surveillance Programs at Braidwood and Byron.

Response

Predicted tendon lift off forces have been and will continue to be determined in accordance with NRC Regulatory Guide 1.35.1 "Determining Prestressing Forces For Inspection Of Prestressed Concrete Containments," July 1990 edition. Reference to Regulatory Guide 1.35.1 will be retained in TS Section 5.5.6 and the References of the Bases Section 3.6.1.

Predicted prestressing tendon forces are calculated using the guidance contained in Regulatory Guide 1.35.1. The calculation for determining prestressing tendon forces utilizes the following inputs when determining prestressed losses: Initial Seating Force and date, Tendon Properties, Tendon Length, Elastic Shortening Losses, Minimum Design Stresses, Concrete Properties, Initial Stressing Date, Shrinkage Creep and Losses, Creep Function, Predicted Prestressed Losses, and Upper Bound Stresses. The revised markup TS Bases are attached and reflect this change.

ATTACHMENT 2
Markup of Proposed Technical Specifications Pages for Braidwood Station

Braidwood Station, Units 1 and 2

Facility Operating License Nos. NPF-72 and NPF-77

REVISED TECHNICAL SPECIFICATIONS PAGES

5.5 - 5

5.5 Programs and Manuals

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in ~~conformance~~ accordance with ~~requirements of 10 CFR 50.55a(b)(2)(vi), 10 CFR 50.55a(b)(2)(ix), ASME Boiler and Pressure Vessel Code Subsection IWL, 1992 Edition with the 1992 Addenda and Regulatory Guide 1.35.1, July 1990.~~ Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC. Determining pre-stressing forces for inspections shall be consistent with the recommendations of Regulatory Guide 1.35.1, July 1990.

The provisions of ~~SR 3.0.2 and~~ SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel in general conformance with the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Regulatory Position c.4.b(1) and c.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheel may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI.

ATTACHMENT 3
Markup of Proposed Technical Specifications Pages for Byron Station

Byron Station, Units 1 and 2

Facility Operating License Nos. NPF-37 and NPF-66

<u>REVISED TECHNICAL SPECIFICATIONS PAGES</u>

5.5 - 5

5.5 Programs and Manuals

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ATTACHMENT 4
Markup of Technical Specifications Bases Pages for Braidwood Station

Braidwood Station, Units 1 and 2

Facility Operating License Nos. NPF-72 and NPF-77

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

B 3.6.1 - 3

B 3.6.1 - 6

B 3.6.1 - 7

APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA, secondary system pipe break, or fuel handling accident (Ref. 3). In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment leakage rate, used to evaluate doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.20% per day in the safety analysis at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 (Ref. 3).

The radiological dose assessments performed for the design basis LOCA assume a maximum allowable containment leakage rate of 0.20% per day. In this case, the dose limits of 10 CFR 50.67 (Ref. 7) are not exceeded.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required leakage test is required to be < 0.6 L_a for combined Type B and C leakage and < 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. ~~Testing and Frequency are consistent with the requirements of 10 CFR50.55a(b)(2)(vi) "Effective Edition and Addenda of Subsection IWE and Subsection IWL, SECTION XI" (Ref. 4), and Section 10 CFR50.55a(b)(2)(ix), "Examination of Concrete Containments" (Ref. 5). Predicted tendon lift off forces will be determined consistent with the recommendations of Regulatory Guide 1.35.1, (Ref. 6).~~

The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption, or relief has been authorized by the NRC. Determining pre-stressing forces for inspections shall be consistent with the recommendations of Regulatory Guide 1.35.1, July 1990 (Ref. 6).

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Chapter 15.
3. UFSAR, Section 6.2.
4. 10 CFR50.55a, ~~(b)(2)(vi) "Effective Edition and Addenda of Subsection IWE and Subsection IWL, SECTION XI."~~

"Codes and standards."

5. ~~10 CFR50.55a(b)(2)(ix), "Examination of Concrete Containments."~~
6. Regulatory Guide 1.35.1, July 1990.
7. 10 CFR 50.67.

ASME Code, Section XI, Subsection IWL

ATTACHMENT 5
Markup of Technical Specifications Bases Pages for Byron Station

Byron Station, Units 1 and 2

Facility Operating License Nos. NPF-37 and NPF-66

REVISED TECHNICAL SPECIFICATIONS BASES PAGES

B 2.6.1 – 3

B 3.6.1 - 5

B 3.6.1 - 6

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA, secondary system pipe break, or fuel handling accident (Ref. 3). In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment leakage rate, used to evaluate doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.20% per day in the safety analysis at P_a = 42.8 psig for Unit 1 and P_a = 38.4 psig for Unit 2 (Ref. 3).

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