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10 CFR 50.90

RS-08-036

April 9, 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: License Amendment Request to Revise Containment Tendon Surveillance Program

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests 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).

This request is subdivided as follows.

- Attachment 1 provides a description and evaluation of the proposed change.
- Attachment 2 provides a markup of the affected TS pages for Braidwood Station.
- Attachment 3 provides a markup of the affected TS pages for Byron Station.
- Attachments 4 and 5 provide a markup of the affected TS Bases pages for Braidwood Station and Byron Station, respectively. The TS Bases pages are provided for information only and do not require NRC approval.

The proposed change has been reviewed by the Plant Operations Review Committees at each of the stations 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 change by April 9, 2009. Once approved, the amendment will be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. If 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 9th day of April 2008.

Respectfully,

Patrick R. Simpson
Manager - Licensing

Attachments:

1. Evaluation of Proposed Change
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

ATTACHMENT 1
Evaluation of Proposed Change

1.0 DESCRIPTION

2.0 PROPOSED CHANGE

3.0 BACKGROUND

4.0 TECHNICAL ANALYSIS

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

5.2 Applicable Regulatory Requirements/Criteria

6.0 ENVIRONMENTAL CONSIDERATION

7.0 REFERENCES

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 the Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF 37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively.

The proposed change revises 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. This regulation requires licensees to update their 10-year containment inservice inspection interval to comply with the latest edition and addenda of the Code incorporated by reference in paragraph (b) of 10 CFR 50.55a.

This proposed change is a conforming administrative amendment in that it revises TS Section 5.5.6 and 5.5.8 to remove the reference to the specific ASME Code edition and addenda, since 10 CFR 50.55a(g)(4) provides the regulatory requirement to update this for each 10-year interval.

2.0 PROPOSED CHANGE

The proposed change revises TS Section 5.5.6 to read:

"...The Containment 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."

Additionally, the note stating the provisions of Surveillance Requirement (SR) 3.0.2 is applicable to the frequency of the SR is being deleted from these specifications.

TS Section 5.6.8 is revised to delete the reference to the dated edition and addenda.

3.0 BACKGROUND

On January 7, 1994, the NRC published a proposed amendment to the regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code (the Code). The final rule, 10 CFR 50.55a(g)(6)(ii)(B), became effective on September 9, 1996, and requires licensees to implement Subsections IWE and IWL, with specified modifications and limitations, by September 9, 2001.

For Braidwood, the next 10-year containment inservice inspection interval for Unit 1 and Unit 2 is scheduled to begin on July 29, 2008 and October 17, 2008, respectively. For Byron, the current 10-year containment inservice inspection interval for Unit 1 and Unit 2 began on January 16, 2006, as outlined in Reference 1.

The containment consists of a pre-stressed, reinforced concrete, cylindrical structure with a hemispherical dome. The post-tensioning system used for the shell and dome of the containment employs tendons. Each tendon consists of high strength steel wires and anchoring components. The pre-stressing load is transferred, by cold formed button heads on the ends of the individual wires through stressing washers, to steel bearing plates embedded in the structure. The unbonded tendons are installed in tendon ducts and tensioned in a predetermined sequence.

4.0 TECHNICAL ANALYSIS

TS Section 5.5.6, states in part, "The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in conformance 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." TS Section 5.6.8, states in part, "Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI, 1992 Edition with the 1992 Addenda." As identified above, 10 CFR 50.55a(g)(4) requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR 50.55a(b)(2)(ix). The requirements in 10 CFR 50.55a(g)(4) and ASME Code Section XI, Subsection IWL, do not reference the 1992 Edition with the 1992 Addenda or Regulatory Guide 1.35.1. Rather, 10 CFR 50.55a(g)(4) requires the latest edition and addenda of the ASME Code incorporated by reference in paragraph (b) of 10 CFR 50.55a, to be used for containment inservice inspections. As such, the TS are inconsistent with the requirements of 10 CFR 50.55a and should be updated to refer to the most updated edition of the ASME Code as specified in 10 CFR 50.55a and to remove the reference to Regulatory Guide 1.35.1.

Additionally, since the tendon inspection frequencies will be in accordance with ASME Section XI, Subsection IWL, the provisions of SR 3.0.2 are no longer applicable and are deleted from Technical Specification 5.5.6. As discussed in the TS Bases for SR 3.0.2, the regulations take precedence over the TS. As such, 10 CFR 50.55a requires the implementation of ASME Section XI, Subsection IWL, and specifies the requirements for extending inspection frequencies.

The proposed change adopts wording that is identical to NUREG-1431 (i.e., Reference 2) for the containment tendon surveillance program.

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 Technical Specifications (TS) Section 5.5.6 and Section 5.6.8 of Facility Operating License Nos. NPF-72, NPF-77, NPF 37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively.

ATTACHMENT 1
Evaluation of Proposed Change

The proposed amendment revises the 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. Additionally, the note stating the provisions of Surveillance Requirement (SR) 3.0.2 is applicable to the frequency of the SR is being deleted from these specifications.

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.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change revises the TS administrative controls program for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC.

The revised requirements do not affect the function of the containment post-tensioning system components. The post-tensioning systems are passive components whose failure modes could not act as accident initiators or precursors. The proposed change does not impact the physical configuration or function of plant structures, systems, or components (SSCs) or the manner in which SSCs are operated, maintained, modified, tested, or inspected. The proposed change does not impact the initiators or assumptions of analyzed events, nor does it impact the mitigation of accidents or transient events.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. It does not involve the addition or removal of any equipment, or any design changes to the facility.

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 change revises the TS administrative controls program for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC.

   The function of the containment post-tensioning system components is not altered by this change. The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does the proposed change involve a significant reduction in a margin of safety?**

   Response: No

   The proposed change revises the TS administrative controls program requirements for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC.

   The function of the containment post-tensioning system components are not altered by this change. The change is conforming and administrative in nature in that it will allow the TS to be updated to refer to the most recently approved edition of the ASME Boiler and Pressure Vessel Code Subsection IWL. The safety function of the containment as a fission product barrier will be maintained.

   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 50.55a(g)(4) requires that for each successive 10-year inservice inspection interval, licensees adopt the latest edition and addenda of the ASME Code Section XI, incorporated by

reference in 10 CFR 50.55a(b). The proposed amendment revises TS Section 5.5.6 and Section 5.6.8 for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC.

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 evaluated this proposed license amendment and determined that the proposed change would change a requirement with respect to installation or use of a facility component located within the restricted areas, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change 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(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

7.0 REFERENCES

1. Letter from D. Hoots (EGC) to U. S. NRC, "Byron Station, Units 1 and 2, Transmittal of Inservice Inspection Program Plan for Third Ten year Inspection Interval," dated February 14, 2006

2. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Volume 1, Revision 3, dated March 2004

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

5.6 - 6

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~~ 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.~~ accordance

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.

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.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI~~, 1992 Edition with the 1992 Addenda~~.

5.6.9 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged or repaired to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing,

h. The effective plugging percentage for all plugging and tube repairs in each SG, and

i. Repair method utilized and the number of tubes repaired by each repair method.

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

REVISED TECHNICAL SPECIFICATIONS PAGES

5.5 - 5

5.6 - 6

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~~ 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.~~

accordance

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

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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.

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.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI~~, 1992 Edition with the 1992 Addenda~~.

5.6.9 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged or repaired to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing,

h. The effective plugging percentage for all plugging and tube repairs in each SG, and

i. Repair method utilized and the number of tubes repaired by each repair method.

j. For Unit 2, following completion of an inspection performed in Refueling Outage 13 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, and whether initiated on primary or secondary side for each indication detected in the upper 17-inches of the tubesheet thickness.

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

REVISED TECHNICAL SPECIFICATIONS BASES PAGES

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.

6 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).~~

"Codes and standards"

in accordance with the ASME Code, Section XI, Subsection IWL (Ref. 5), and applicable addenda as required by 10 CFR 50.55a.

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

"Codes and standards."

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).

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.

6 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 ≤ 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.

BASES

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).~~

"Codes and standards"

in accordance with the ASME Code, Section XI, Subsection IWL (Ref. 5), and applicable addenda as required by 10 CFR 50.55a.

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

"Codes and standards."

ASME Code, Section XI, Subsection IWL