UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 27, 2018

Mr. Joseph W. Shea
Vice President, Nuclear Regulatory
  Affairs and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – ISSUANCE OF AMENDMENT NOS. 305, 328, AND 288 TO REVISE TECHNICAL SPECIFICATION 5.5.12, "PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (CAC NOS. MG0113, MG0114, AND MG0115; EPID L-2017-LLA-0292)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 305, 328, and 288 to Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68, for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, respectively.

These amendments are in response to Tennessee Valley Authority's license amendment request dated August 15, 2017, as supplemented by letters dated February 5, March 27, and July 27, 2018. The amendments revise Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program," at Browns Ferry Nuclear Plant, Units 1, 2, and 3, by adopting Nuclear Energy Institute 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," as the implementation document for the performance-based Option B of 10 CFR Part 50, Appendix J. The proposed changes allow the licensee to extend the Type A containment integrated leak rate testing interval from 10 years to 15 years, and the Type C local leakage rate testing intervals from 60 months to 75 months.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Sincerely,

Farideh E. Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, and 50-296

Enclosures:
1. Amendment No. 305 to DPR-33
2. Amendment No. 328 to DPR-52
3. Amendment No. 288 to DPR-68
4. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 305
Renewed License No. DPR-33

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 15, 2017, as supplemented by letters dated February 5, March 27, and July 27, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the renewed license is amended by changes to the Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment.

3. The first paragraph of paragraph 2.C.(2) of Renewed Facility Operating License No. DP-33 is hereby amended to state as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 305, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. Further, a new paragraph (21) is added to paragraph 2.C. as follows:

    (21) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

5. The license amendment is effective as of its date of issuance and shall be implemented prior to Browns Ferry Nuclear Plant, Unit 2 startup following the spring 2019 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Boome

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and Technical
Specifications

Date of Issuance: September 27, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 305

BROWNS FERRY NUCLEAR PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace the following pages of Renewed Facility Operating License No. DPR-33 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of changes.

REMOVE	INSERT
3	3
6e	6e

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contain marginal lines indicating the areas of changes.

REMOVE	INSERT
5.0-20	5.0-20

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3952 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 305, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 234 to Facility Operating License DPR-33, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 234. For SRs that existed prior to Amendment 234, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 234.

(h) The results of the visual inspections of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall be submitted to the NRC within 90 days following startup from each of the first two respective refueling outages.

(i) Within 6 months following completion of the second refueling outage, after the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results.

The license condition described above shall expire: (1) upon satisfaction of the requirements in items (g) and (h), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and (2) upon satisfaction of the requirements specified in item (i).

(19) Neutron Absorber Monitoring Program

The licensee shall, at least once every ten years, withdraw a neutron absorber coupon from the spent fuel pool and perform Boron-10 (B-10) areal density measurement on the coupon. Based on the results of the B-10 areal density measurement, the licensee shall perform any technical evaluations that may be necessary and take appropriate actions using relevant regulatory and licensing processes.

(20) Radiological Consequences Analyses Using Alternative Source Terms

TVA shall perform facility and licensing basis modifications to resolve the non-conforming/degraded condition associated with the Alternate Leakage Treatment pathway such that the current licensing basis dose calculations (approved in License Amendment Nos. 251/282 (Unit 1), 290/308 (Unit 2) and 249/267 (Unit 3)) would remain valid. These facility and licensing basis modifications shall be completed prior to initial power ascension above 3458 MWt.

(21) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than December 20, 2013, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation report in NEI 94-01, Revision 2-A, dated October 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.1 psig. The maximum allowable primary containment leakage rate, L_a, shall be 2% of primary containment air weight per day at P_a.

(continued)

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 328
Renewed License No. DPR-52

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 15, 2017, as supplemented by letters dated February 5, March 27, and July 27, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the renewed license is amended by changes to the Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment.

3. The first paragraph of paragraph 2.C.(2) of Renewed Facility Operating License No. DP-52 is hereby amended to state as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 328, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. Further, a new paragraph (21) is added to paragraph 2.C. as follows:

   (21) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

5. The license amendment is effective as of its date of issuance and shall be implemented prior to Browns Ferry Nuclear Plant, Unit 2 startup following the spring 2019 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

V Boome

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: September 27, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 328

BROWNS FERRY NUCLEAR PLANT, UNIT 2

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace the following pages of Renewed Facility Operating License No. DPR-52 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of changes.

REMOVE	INSERT
3	3
6e	6e

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contain marginal lines indicating the areas of changes.

REMOVE	INSERT
5.0-20	5.0-20

sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3952 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 328, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 253 to Facility Operating License DPR-52, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 253. For SRs that existed prior to Amendment 253, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 253.

3) The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's

(h) The results of the visual inspections of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall be submitted to the NRC within 90 days following startup from each of the first two respective refueling outages.

(i) Within 6 months following completion of the second refueling outage, after the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results.

The license condition described above shall expire: (1) upon satisfaction of the requirements in items (g) and (h), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and' (2) upon satisfaction of the requirements specified in item (i).

(19) Neutron Absorber Monitoring Program

The licensee shall, at least once every ten years, withdraw a neutron absorber coupon from the spent fuel pool and perform Boron-10 (B-10) areal density measurement on the coupon. Based on the results of the B-10 areal density measurement, the licensee shall perform any technical evaluations that may be necessary and take appropriate actions using relevant regulatory and licensing processes.

(20) Radiological Consequences Analyses Using Alternative Source Terms

TVA shall perform facility and licensing basis modifications to resolve the non-conforming/degraded condition associated with the Alternate Leakage Treatment pathway such that the current licensing basis dose calculations (approved in License Amendment Nos. 251/282 (Unit 1), 290/308 (Unit 2) and 249/267 (Unit 3)) would remain valid. These facility and licensing basis modifications shall be completed prior to initial power ascension above 3458 MWt.

(21) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than June 28, 2014, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012, and Section 4.1, "Limitations of Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 288
Renewed License No. DPR-68

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 15, 2017, as supplemented by letters dated February 5, March 27, and July 27, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the renewed license is amended by changes to the Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment.

3. The first paragraph of paragraph 2.C.(2) of Renewed Facility Operating License No. DP-52 is hereby amended to state as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 288, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. Further, a new paragraph (17) is added to paragraph 2.C. as follows:

    (17) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

5. The license amendment is effective as of its date of issuance and shall be implemented prior to Browns Ferry Nuclear Plant, Unit 2 startup following the spring 2019 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

VBooma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and Technical
Specifications

Date of Issuance: September 27, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 288

BROWNS FERRY NUCLEAR PLANT, UNIT 3

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Replace the following pages of Renewed Facility Operating License No. DPR-68 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of changes.

REMOVE	INSERT
3	3
6f	6f

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contain marginal lines indicating the areas of changes.

REMOVE	INSERT
5.0-20	5.0-20

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3952 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 288, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 212 to Facility Operating License DPR-68, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 212. For SRs that existed prior to Amendment 212, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 212.

(16) Radiological Consequences Analyses Using Alternative Source Terms

TVA shall perform facility and licensing basis modifications to resolve the non-conforming/degraded condition associated with the Alternate Leakage Treatment pathway such that the current licensing basis dose calculations (approved in License Amendment Nos. 251/282 (Unit 1), 290/308 (Unit 2) and 249/267 (Unit 3)) would remain valid. These facility and licensing basis modifications shall be complete prior to initial power ascension above 3458 MWt.

(17) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than July 2, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008.

(continued)

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 305

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 328 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-52

AND

AMENDMENT NO. 288 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

By letter dated August 15, 2017 (Reference 1), as supplemented by letters dated February 5, 2018 (Reference 2); March 27, 2018 (Reference 3); and July 27, 2018 (Reference 4), Tennessee Valley Authority (TVA or the licensee) submitted a license amendment request (LAR) to revise the Browns Ferry Nuclear Power Plants (Browns Ferry or BFN), Units 1, 2, and 3, Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program." Specifically, the amendments would revise the BFN, Units 1, 2, and 3, TSs to allow extension of the 10-year frequency of the Type A integrated leak rate test (ILRT) as required by TS 5.5.12 to 15 years on a permanent basis and to allow the extension of the containment isolation valves (CIVs) leakage test intervals (i.e., Type C tests) from the current 60-month frequency to 75 months. In particular, TVA proposed changes to BFN TS 5.5.12 by replacing the reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Rate Testing Program," dated September 1995 (Reference 5), with a reference to Nuclear Energy Institute (NEI) Topical Report (TR) NEI 94-01, Revision 3-A (Reference 6), and the conditions and limitations specified in NEI 94-01, Revision 2-A (Reference 7), as the implementation documents used by the licensee to implement the BFN performance-based leakage testing program in accordance with Option B of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J. The NRC previously approved License Amendment Nos. 228, 243, and 203, for BFN, Units 1, 2, and 3, on February 22, 1996 (Reference 8), authorizing the implementation of 10 CFR Part 50, Appendix J, Option B, for Types A, B, and C tests.

The licensee also proposes an administrative change for both BFN, Units 2 and 3. This change consists of deleting no longer relevant exceptions to TS 5.5.12 regarding timelines for the performance of previously completed Units 2 and 3 Type A tests. More specifically, the exception to Unit 2 TS 5.5.12 states that "NEI 94-01-1995, Section 9.2.3: The first Unit 2 Type A test performed after the November 6,1994, Type A test shall be performed no later than

November 6, 2009." Similarly, the exception to Unit 3 TS 5.5.12 states that "NEI 94-01-1995, Section 9.2.3: The first Unit 3 Type A test performed after the October 10, 1998, Type A test shall be performed no later than October 10, 2013." As stated, these two TS exceptions were cited as exceptions to the limitations of Section 9.2.3 of NEI 94-01, Revision 0 (Reference 9). These one-time Type A test extensions for both Units 2 and 3 were approved by the NRC on March 9, 2005, as Amendment Nos. 293 and 252 to Facility Operating License Nos. DPR 52 and DPR-68, respectively (Reference 10). These exceptions allowed a one-time Type A test interval extension to 15 years for both units. With staff approval of the subject LAR of August 15, 2017, these two one-time Type A test extensions no longer need to be reflected in the Units 2 and 3 TSs.

The licensee's supplemental letters dated February 5, March 27, and July 31, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on November 21, 2017 (82 FR 55415).

The licensee requested approval of the proposed license amendments on or before October 31, 2018, in order to incorporate these changes into the BFN schedule. Based on the currently required frequency of 10 years for all three units, these amendments would allow deferral of the next ILRT Type A tests currently scheduled for:

- Unit 1 in the fall of 2020
- Unit 2 in the spring of 2019
- Unit 3 in the spring of 2022

The licensee plans to implement the amendments prior to the BFN, Unit 2 startup following the spring 2019 refueling outage.

2.0 REGULATORY EVALUATION

2.1 Description of the BFN Containment

All three BFN containments house a General Electric Type 4 Reactor. LAR (Reference 1), Enclosure 1, Section 4.1, "Description of Containment," provides an abbreviated description of the three Mark 1 containments as follows:

> The Browns Ferry design employs a pressure suppression primary containment that houses the reactor vessel, the reactor coolant recirculating loops, and other branch connections of the Reactor Primary System. The pressure suppression system consists of a drywell, a pressure suppression chamber that stores a large volume of water, connecting vents between the drywell and the pressure suppression chamber, isolation valves, containment cooling systems, and other service equipment. In the event of a process system piping failure within the drywell, reactor water and steam would be released into the drywell air space. The resulting increased drywell pressure would then force a mixture of air, drywell atmosphere, steam, and water through the vents into the pool of water in the pressure suppression chamber. The steam would condense in the pressure suppression pool, resulting in a rapid pressure reduction in the drywell. Air that was transferred to the pressure suppression chamber pressurizes the pressure suppression chamber, and is subsequently vented back to the drywell to equalize

the pressure between the two vessels. Cooling systems are provided to remove heat from the reactor core, the drywell, and from the water in the pressure suppression chamber, and thus provide continuous cooling of the primary containment under accident conditions. Appropriate isolation valves are actuated during this period to ensure containment of radioactive material, which might otherwise be released from the reactor containment during the course of the accident.

2.2 Licensee's Proposed Changes

The current BFN, Unit 1, TS 5.5.12 states, in part:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide [RG] 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The licensee proposed to extend the BFN, Unit 1, interval for the containment ILRT to 15 years from the last ILRT. The last Unit 1 ILRT was completed during November 2010. Accordingly, the next Unit 1 ILRT is due during the fall of 2020. Using the proposed interval of 15 years, the next Unit 1 ILRT would need to be completed before the end of November 2025.

The current BFN, Unit 2, TS 5.5.12, states, in part:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception:
>
> - NEI 94-01-1995, Section 9.2.3: The first Unit 2 Type A test performed after the November 6, 1994, Type A test shall be performed no later than November 6, 2009.

The licensee proposed to extend the BFN, Unit 2, interval for the containment ILRT to 15 years from the last ILRT. The last Unit 2 ILRT was completed during June 2009. Accordingly, the next Unit 2 ILRT is due during the spring of 2019. Using the proposed interval of 15 years, the next Unit 2 ILRT would need to be completed before the end of June 2024.

The current BFN, Unit 3, TS 5.5.12, starts, in part:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception:

- NEI 94-01-1995, Section 9.2.3: The first Unit 3 Type A test performed after the October 10, 1998, Type A test shall be performed no later than October 10, 2013.

Similarly, the licensee proposed to extend the BFN, Unit 3, interval for the containment ILRT to 15 years from the last ILRT. The last Unit 3 ILRT was completed during May 2012. Therefore, the next Unit 3 ILRT is due during the spring of 2022. Using the proposed interval of 15 years, the next Unit 3 ILRT would need to be completed before the end of May 2027.

The licensee's proposed amendment to BFN, Units 1, 2, and 3, TS 5.5.12, would replace the reference to RG 1.163 (Reference 5) with a reference to both NEI 94-01: Revision 2-A (Reference 7), and Revision 3-A (Reference 6). In addition, the proposed amendment would revise BFN TS 5.5.12 by deleting each exception associated with Units 2 and 3. Therefore, the first paragraph of TS 5.5.12 would be identical for all three units after approval of the proposed change and would state:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008.

2.3 Regulatory Requirements

The licensee proposed changes to the Renewed Facility Operating Licenses and TSs for BFN, Units 1, 2, and 3, in accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit."

The regulation in 10 CFR 50.36(c)(5), "Administrative controls," requires, in part, the inclusion of administrative controls in TSs that are necessary to assure operation of the facility in a safe manner. The LAR seeks changes to the "Administrative Controls" section of the BFN TSs.

The regulations in 10 CFR 50.54, "Conditions of licenses," paragraph (o), requires that the primary reactor containments for water cooled power reactors be subject to the requirements set forth in Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J includes two options: Option A, "Prescriptive Requirements," and Option B, "Performance-Based Requirements," either of which may be chosen for meeting the requirements of Appendix J.

The testing requirements in 10 CFR Part 50, Appendix J, ensure that (a) leakage through containments or systems and components penetrating containments does not exceed allowable leakage rates specified in the TSs, and (b) integrity of the containment structure is maintained during the service life of the containment. BFN has voluntarily adopted and has implemented Option B for meeting the requirements of 10 CFR Part 50, Appendix J.

Option B of 10 CFR Part 50, Appendix J, specifies performance-based requirements and criteria for preoperational and subsequent leakage-rate testing. These requirements are met by performance of Type A tests to measure the containment system overall integrated leakage rate, Type B pneumatic tests to detect and measure local leakage rates across pressure-retaining leakage-limiting boundaries such as penetrations, and Type C pneumatic tests to measure CIVs leakage rates. After the preoperational tests, these tests are required to be conducted at periodic intervals based on the historical performance of the overall containment system (for Type A tests), and based on the safety significance and historical performance of each boundary and isolation valve (for Type B and Type C tests) to ensure integrity of the overall containment system as a barrier to fission product release.

The leakage rate test results must not exceed the allowable leakage rate (L_a) at design-basis loss-of-coolant accident (DBLOCA) pressure, (P_a), with margin, as specified in the TSs. Option B also requires that a general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration that may affect the containment leaktight integrity must be conducted prior to each Type A test and at a periodic interval between tests.

Appendix J, Option B, Section V.B.3 of 10 CFR requires that RG 1.163 (Reference 5) or another implementation document used by a licensee to develop a performance-based leakage-testing program be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in RG 1.163.

Currently, BFN TS 5.5.12 requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. For Unit 1, the program is in accordance with the guidelines contained in RG 1.163. For Units 2 and 3, the programs are also in accordance with the guidelines contained in RG 1.163, dated September 1995, but as modified by two approved exceptions (i.e., one exception for each unit) pertaining to Section 9.2.3 of NEI 94-01, Revision 0 (Reference 9).

NEI 94-01, Revisions 2 and 3, were reviewed by the NRC staff and approved for use. The final safety evaluation (SE) for Revision 2, issued by letter dated June 25, 2008 (Reference 11), documents the NRC's evaluation and acceptance of NEI 94-01, Revision 2, subject to six specific limitations and conditions listed in Section 4.1. The final SE for Revision 3, issued by letter dated June 8, 2012 (Reference 12), includes two specific limitations and conditions listed in Section 4.0 of that SE for the Type C tests. NEI 94-01, Revisions 2-A and 3-A, include the limitations and conditions from the corresponding SEs.

The regulations in 10 CFR 50.55a, "Codes and standards," contain the containment inservice inspection (CISI) requirements that, in conjunction with the requirements of Appendix J, ensure the continued leaktight and structural integrity of the containment during its service life.

The regulation in 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," Section (a)(1), states, in part, that the licensees:

> ... shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and, where practical, take into account industrywide operating experience.

A Type A test is an overall ILRT of the containment structure. NEI 94-01, Revision 0 (Reference 9), specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months, but this "... should be used only in cases where refueling schedules have been changed to accommodate other factors." Amendment Nos. 293 and 252 (Reference 10) to BFN, Units 2 and 3, respectively, allowed a one-time extension of the ILRT interval to 15 years. However, subsequent to these one-time extensions for both Units 2 and Unit 3, the long-term ILRT test interval requirements in TS 5.5.12 remained at 10 years.

In accordance with BFN TS 5.5.12, the peak calculated containment internal pressure for the DBLOCA, P_a, equals 49.1 pounds per square inch gauge (psig), and the maximum allowable containment leakage rate, L_a at P_a, equals 2 percent of primary containment air weight per day.

The NRC issued Amendment Nos. 299, 323, and 283 for BFN, Units 1, 2, and 3, respectively, related to extended power uprate (EPU) on August 14, 2017 (Reference 13). Amendment No. 299 (Unit 1) resulted in an increase in the value of P_a from 48.5 psig to 49.1 psig. Amendment No. 323 (Unit 2) resulted in a decrease in the value of P_a from 50.6 psig to 49.1 psig. Amendment No. 283 (Unit 3) resulted in a decrease in the value of P_a from 50.6 psig to 49.1 psig.

The NRC staff issued Amendment Nos. 251, 290, and 249 for BFN, Units 1, 2, and 3, respectively, on September 27, 2004 (Reference 14). All three of these amendments were related to adoption of alternative source term (AST). These amendments did not result in a change in the value of L_a of 2.0 percent of primary containment air weight per day.

Guidance for extending Type A ILRT surveillance intervals beyond 10 years is provided in TR NEI 94-01, Revision 2-A (Reference 7).

Guidance for extending Type C test local leak rate test (LLRT) surveillance intervals beyond 60 months is provided in TR NEI 94-01, Revision 3-A (Reference 6).

RG 1.200, Revision 2, "An Approach tor Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009 (Reference 15), describes an acceptable approach tor determining whether the quality of the probable risk assessment (PRA) in total, or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decisionmaking for light-water reactors. This RG provides guidance for assessing the technical adequacy of a PRA. Revision 2 of RG 1.200, endorses, with clarifications and qualifications, the use of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard, RA-Sa-2009 (Reference 16), "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment tor Nuclear Power Plant Applications" (i.e., the PRA Standard).

RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2011 (Reference 17), describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

3.0 TECHNICAL EVALUATION

3.1 NRC Staff's Deterministic Evaluation

The proposed changes in the LAR will revise portions of each BFN TS 5.5.12 by replacing the reference to RG 1.163 with a reference to NEI TR NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, as the documents used by BFN to implement the performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. By invoking these NEI 94-01 TRs as the reference documents for TS 5.5.12, the NRC staff notes that the licensee would be:

- adopting the use of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002, Containment System Leakage Testing Requirements (Reference 18); and

- adopting a more conservative grace interval of 9 months, for Type A, Type B, and Type C leakage tests in accordance with NEI 94-01, Revision 3-A.

With respect to the deletion of exceptions currently contained in TS 5.5.12 of Units 2 and 3, the NRC staff agrees with the licensee that the deletion of each is appropriate. As indicated in the table contained in LAR Section 4.2, the first Unit 2 Type A ILRT after November 1994 was completed during June 2009. Similarly, the first Unit 3 Type A ILRT after October 1998 was completed during May 2012. Accordingly, the NRC staff approves the deletion of the longer, relevant exceptions contained in TS 5.5.12 of Units 2 and 3.

Consistent with the guidance contained in both NEI 94-01, Revision 2-A, and NEI 94-01, Revision 3-A, the licensee justified the proposed changes by demonstrating adequate performance of the BFN containments based on (a) the historical plant-specific containment leakage testing program results, (b) the CISI program results, and (c) a BFN plant-specific risk assessment.

The NRC staff reviewed the BFN LAR and supplements from the perspective of deterministic considerations with regard to containment leaktight integrity and CISI program results, and a supporting plant-specific risk assessment. The NRC staff's analysis of these changes is conveyed in the following subsections of Sections 3.1.1 and 3.1.2

3.1.1 Type A Integrated Leak Rate Test History

BFN, Unit 1

Per TS 5.5.12, the Unit 1 primary containment was designed for a maximum allowable leakage rate L_a of 2.0 percent by weight of the containment atmosphere air mass per day at the DBLOCA maximum peak containment pressure, P_a. The second paragraph of TS 5.5.12 indicates that the current calculated peak containment internal pressure for the DBLOCA, P_a, is 49.1 psig.

Since 1981, a total of three ILRTs have been performed on the Unit 1 primary containment. All three ILRTs had satisfactory leakage rate results. These three ILRT test results were documented in the table contained in LAR Section 4.2. These test results are summarized in Table 3.1.1-1 below:

TABLE 3.1.1-1

BFN, Unit 1, Type A ILRT History

Test Date	Final Test Pressure (psig)	Design-Basis Peak Accident Pressure, P_a (psig)	Upper 95% Confidence Level (with penalties) (fraction of L_a)	ILRT Leakage Rate (wt%/day)	Acceptance Criteria, L_a (wt%/day)	Test Method/ Data Analysis Technique
May 1981	(4)		0.086(5)	0.172 (5)	2.0(3)	Mass Point
March 2007	49.82(6)	48.5(6)	0.04 (5)	0.080 (5)	2.0(2)	Mass Point(1)
November 2010	51.16(6)	48.5(6)	0.10 (5)	0.200 (5)	2.0(2)	Mass Point(1)

Notes:
(1) Mass point analysis per Section 5.5.3 of ANSI-56.8-1994
(2) Per TS 5.5.12
(3) Per Limiting Condition for Operation (LCO) 3.7.A (before BFN adopted 10 CFR 50, Appendix J, Option B, in February 1996)
(4) Not available as not provided in the LAR
(5) Data source – table contained in LAR Section 4.2
(6) Reference TVA response to SCPB RAI 1 in the letter dated February 5, 2018 (Reference 2)

The NRC staff notes that the last sentence of Section 9.2.3, "Extended Test Intervals," of NEI 94-01, Revision 3-A, states, "In the event where previous Type A tests were performed at reduced pressure (as described in 10 CFR 50, Appendix J, Option A), at least one of the two consecutive periodic Type A tests shall be performed at peak accident pressure (P_a)." Section 9.1.2 of the same NEI TR states, in part, "The elapsed time between the first and the last tests in a series of consecutive passing tests used to determine performance shall be at least 24 months."

The NRC staff confirmed that the P_a requirement of Section 9.2.3 has been satisfied as the last two Unit 1 historical ILRTs were performed at or above P_a. As shown in Table 3.1.1-1, the last two Unit 1 Type A tests were performed in March 2007 and November 2010. Both Type A tests were performed at a pressure higher than the peak calculated design-basis internal accident pressure for the DBLOCA, P_a, which per TS 5.5.12 for Unit 1, was equal to 48.5 psig at the time of each ILRT. On August 14, 2017 (Reference 13), the NRC issued Amendment No. 299 associated with the Unit 1 EPU, where P_a was increased from 48.5 psig to its current TS value of 49.1 psig. As shown in Table 3.1.1-1, the "Final Test Pressure" recorded in each of the two most recent ILRTs exceeds the revised Amendment No. 299 P_a value of 49.1 psig. On September 27, 2004 (Reference 14), the NRC issued Amendment No. 251 associated with the Unit 1 adoption of the AST methodology where L_a remained the same at 2.0 percent by weight of the primary containment atmosphere air mass per day. Therefore, the NRC staff finds that the above requirements of both NEI TR Sections 9.1.2 and 9.2.3 have been satisfied.

The Unit 1 primary containment was designed for a leakage rate L_a not to exceed 2.0 percent by weight of the primary containment atmosphere air mass per day at the calculated peak pressure, P_a. As displayed in Table 3.1.1-1, there has been adequate margin (≥ 90 percent) to the Type A test performance limit as described in TS 5.5.12 of L_a for the three historical ILRTs spanning a time period of nearly 30 years.

The past three Unit 1 ILRT results dating back to 1981 have confirmed that the primary containment leakage rates are acceptable with respect to the design criterion leakage of primary containment air weight (L_a) per day at the DBLOCA pressure, (P_a). Since the last two Type A tests for BFN, Unit 1, had ILRT test results of less than 1.0L_a, a test frequency of 15 years in accordance with NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A, would be acceptable for Unit 1.

Therefore, based on the last two Type A Unit 1 ILRT test results, the NRC staff concludes that the requirements of Sections 9.1.2 and 9.2.3 of NEI 94-01, Revision 3-A, have been satisfied.

BFN, Unit 2

Per TS 5.5.12, the Unit 2 primary containment was designed for a maximum allowable leakage rate L_a of 2.0 percent by weight of the containment atmosphere air mass per day at the DBLOCA maximum peak containment pressure P_a. The second paragraph of TS 5.5.12 indicates that the current calculated peak containment internal pressure for the DBLOCA, P_a, is 49.1 psig.

Since 1991, a total of three ILRTs have been performed on the Unit 2 primary containment. All three ILRTs had satisfactory leakage rate results. These three ILRT test results were documented in the table contained in LAR Section 4.2. These test results are summarized in Table 3.1.1-2 below:

TABLE 3.1.1-2

BFN, Unit 2, Type A ILRT History

Test Date	Final Test Pressure (psig)	Design-Basis Peak Accident Pressure, P_a (psig)	Upper 95% Confidence Level (with penalties) (fraction of L_a)	ILRT Leakage Rate (wt%/day)	Acceptance Criteria, L_a (wt%/day)	Test Method/ Data Analysis Technique
March 1991	[4]		0.125[5]	0.250[5]	2.0[3]	Mass Point
November 1994	50.28[6]	49.6[6]	0.175[5]	0.350[5]	2.0[2]	Mass Point[1]
June 2009	51.05[6]	50.6[6]	0.31[5]	0.620[5]	2.0[2]	Mass Point[1]

Notes:
[1] Mass point analysis per Section 5.5.3 of ANSI-56.8-1994
[2] Per TS 5.5.12
[3] Per LCO 3.7.A (before BFN adopted 10 CFR 50, Appendix J, Option B, in February 1996)
[4] Not available as not provided in the LAR
[5] Data source – table contained in LAR Section 4.2
[6] Reference TVA response to SCPB RAI 1 in the letter dated February 5, 2018 (Reference 2)

The NRC staff confirmed that the P_a requirement Section 9.2.3 has been satisfied, as the last two BFN, Unit 2, historical ILRTs were performed at or above P_a. As can be seen in Table 3.1.1-2, the last two Unit 2 Type A tests were performed in November 1994 and June 2009. Both Type A tests were performed at a pressure higher than the peak calculated design-basis internal accident pressure for the DBLOCA, P_a. On August 14, 2017, the NRC staff issued Amendment No. 323 associated with the Unit 2 EPU, where P_a was decreased from 50.6 psig to its current TS value of 49.1 psig. As shown in Table 3.1.1-2, the "Final Test Pressure" recorded in each of the two most recent ILRTs exceeds the revised Amendment No. 323 P_a value of 49.1 psig. On September 27, 2004, the NRC staff issued Amendment No. 290 associated with the Unit 2 Adoption of the AST Methodology where L_a remained the same at 2.0 percent by weight of the primary containment atmosphere air mass per day. Therefore, the NRC staff finds that the above requirements of both NEI TR Sections 9.1.2 and 9.2.3 have been satisfied.

The Unit 2 primary containment was designed for a leakage rate L_a not to exceed 2.0 percent by weight of the primary containment atmosphere air mass per day at the calculated peak pressure, P_a. As displayed in Table 3.1.1-2, there has been adequate margin (greater than or equal (≥) 69 percent) to the Type A test performance limit as described in TS 5.5.12 of L_a for the three historical ILRTs spanning a time period of 18 years.

The past three Unit 2 ILRT results dating back to 1991 have confirmed that the primary containment leakage rates are acceptable with respect to the design criterion leakage of primary containment air weight (L_a) per day at the DBLOCA pressure, (P_a). Since the last two Type A tests for BFN, Unit 2, had ILRT test results of less than 1.0 L_a, a test frequency of 15 years in accordance with NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A, would be acceptable for Unit 2.

Therefore, based on the last two Type A BFN, Unit 2, ILRT test results, the NRC staff concludes that the requirements of Sections 9.1.2 and 9.2.3 of NEI 94-01, Revision 3-A, have been satisfied.

BFN, Unit 3

Per TS 5.5.12, the Unit 3 primary containment was designed for a maximum allowable leakage rate L_a of 2.0 percent by weight of the containment atmosphere air mass per day at the DBLOCA maximum peak containment pressure, P_a. The second paragraph of TS 5.5.12 indicates that the current calculated peak containment internal pressure for the DBLOCA, P_a, is 49.1 psig.

Since 1982, a total of three ILRTs have been performed on the Unit 3 primary containment. All three ILRTs had satisfactory leakage rate results. These three ILRT test results were documented in the table contained in LAR Section 4.2. These test results are summarized in Table 3.1.1-3 below:

TABLE 3.1.1-3

BFN, Unit 3, Type A ILRT History

Test Date	Final Test Pressure (psig)	Design-Basis Peak Accident Pressure, P_a (psig)	Upper 95% Confidence Level (with penalties) (fraction of L_a)	ILRT Leakage Rate (wt%/day)	Acceptance Criteria, L_a (wt%/day)	Test Method/ Data Analysis Technique
March 1982	(4)		0.164 (5)	0.328 (5)	2.0(3)	Mass Point
Oct. 1998	51.2(6)	50.6(6)	0.15(5)	0.300(5)	2.0(2)	Mass Point(1)
May 2012	50.83(6)	50.6(6)	0.29(5)	0.580 (5)	2.0(2)	Mass Point(1)

Notes:
(1) Mass point analysis per Section 5.5.3 of ANSI-56.8-1994
(2) Per TS 5.5.12
(3) Per LCO 3.7.A (before BFN adopted 10 CFR 50, Appendix J, Option B in February 1996)
(4) Not available as not provided in the LAR
(5) Data source – table contained in LAR Section 4.2
(6) Reference TVA response to SCPB RAI 1 in the letter dated February 5, 2018 (Reference 2)

The NRC staff confirmed that the P_a requirement Section 9.2.3 has been satisfied, as the last two BFN, Unit 3, historical ILRTs were performed at or above P_a. As can be seen in Table 3.1.1-3, the last two Unit 3 Type A tests were performed in October 1998 and May 2012. Both Type A tests were performed at a pressure higher than the peak calculated design-basis internal accident pressure for the DBLOCA, P_a. On August 14, 2017, the NRC staff issued Amendment No. 283 associated with the Unit 3 EPU where P_a was decreased from 50.6 psig to its current TS value of 49.1 psig. As can be seen in Table 3.1.1-2, the "Final Test Pressure" recorded in each of the two most recent ILRTs exceeds the revised Amendment No. 283 P_a value of 49.1 psig. On September 27, 2004, the NRC staff issued Amendment No. 249 associated with the Unit 3 adoption of the AST methodology where L_a remained the same at 2.0 percent by weight of the primary containment atmosphere air mass per day. Therefore, the NRC staff finds that the above requirements of both NEI TR Sections 9.1.2 and 9.2.3 have been satisfied.

The Unit 3 primary containment was designed for a leakage rate L_a not to exceed 2.0 percent by weight of the primary containment atmosphere air mass per day at the calculated peak pressure, P_a. As displayed in Table 3.1.1-3, there has been adequate margin (≥ 71 percent) to the Type A test performance limit as described in TS 5.5.12 of L_a for the three historical ILRTs spanning a time period of 30 years.

The past three Unit 3 ILRT results dating back to 1982 have confirmed that the primary containment leakage rates are acceptable with respect to the design criterion leakage of primary containment air weight (L_a) per day at the DBLOCA pressure. (P_a). Since the last two Type A tests for BFN, Unit 3, had ILRT test results of less than 1.0 L_a, a test frequency of 15 years in accordance with NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A, would be acceptable for Unit 3.

Therefore, based on the last two Type A BFN, Unit 3, ILRT test results, the NRC staff concludes that the requirements of Sections 9.1.2 and 9.2.3 of NEI 94-01, Revision 3-A, have been satisfied.

Overall Type A Test Program Assessment of BFN, Units 1, 2, and 3

In summary, the NRC staff concludes that:

- Any lack of information or ambiguities contained in the LAR [Reference 5.1] were adequately explained by the licensee in its response to the staff's RAI (Reference 2).
- The licensee has been compliant with the guidance contained in its current license basis as defined by RG 1.163 and NEI 94-01, Revision 0.
- The historical Type A test results were below the acceptance limit of BFN TS 5.5.12.
- The permissives, as contained in Sections 9.1.2 and 9.2.3 of NEI 94 01, Revision 3-A, for extending the 10-year frequency of the Type A integrated leak rate test (ILRT) to 15 years on a permanent basis, have been satisfied.

Therefore, the NRC staff finds that the licensee is effectively implementing the BFN Type A leakage rate test program as required by Option B of 10 CFR Part 50, Appendix J.

3.1.2 Type B and Type C Leak Rate Test History

In a request for additional information (RAI), the NRC staff noted that for an established 10 CFR, Appendix J, Option B LLRT program with a sufficient historical base, the percentage of Type B or Type C components on repetitive frequencies can indicate the quality of the maintenance program and corrective action process. The licensee responded that the total domain of all BFN, Units 1, 2, and 3, Type B components tested equals 393 penetrations. Of this total domain, 18 components are not eligible for extended test intervals either due to NEI 94-01, Revision 0, limitations, or due to the lack of post-maintenance test (PMT) histories as a result of recent modifications. Therefore, the total domain of BFN components eligible for extended test intervals equals 375. There are four Type B components currently not on extended test intervals due to recent Type B test performance failures. The percent of the total domain of eligible BFN Type B penetrations tested on a 120-month extended performance-based interval is approximately 99 percent.

Also, as in the licensee's response to the staff's RAI, the total domain of all BFN, Units 1, 2, and 3, Type C components tested equals 291 penetrations. Of this total domain:

- 81 components are on a 30-month test interval as required by RG 1.163.
- 15 components are tested every refuel cycle to satisfy inservice testing program pressure isolation valve leakage criteria.
- 9 components are tested every refuel cycle due to only having a single isolation.
- 15 components are tested at 30 months to satisfy inservice remote position indication verification.

Therefore, the total domain of BFN Type C components eligible for extended test intervals equals 171. There are 20 Type C components currently not on extended test intervals due to recent Type C test performance failures. The percent of the total domain of eligible BFN Type C components tested on a 60-month extended performance-based interval is approximately 88 percent.

BFN, Unit 1

Unit 1 TS 5.5.12 states, in part:

> Leakage Rate acceptance criteria are:
>
> a. The primary containment leakage rate acceptance criteria is ≤ 1.0 L_a. During the first unit startup following the testing performed in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and Type C tests, and ≤ 0.75 L_a for the Type A test; and ...

The NRC staff notes that subsections 10.2.1.2, "Extended Test Intervals (Except Containment Airlocks)," and 10.2.3.2, "Extended Test Interval," of NEI 94-01, Revision 0, provide concise requirements (i.e., thresholds) for the respective extension of Type B and Type C test intervals beyond the minimum allowed Appendix J, Option B interval. These thresholds pertain to the elapsed time between the satisfactory completion of two consecutive Type B and Type C tests.

The NRC staff reviewed the local leak rate summary table entitled "Browns Ferry, Unit 1" contained in the enclosure to the LAR (Reference 1) Section 4.3, "Type B and Type C Testing Programs." Section 4.3 indicates that 0.6 L_a equals 646.3 standard cubic feet per hour (scfh) which implies that L_a equals 1077.2 scfh.

With the use of this numeric L_a value and the data contained in the table, the staff derived "Fraction of L_a" values from the information contained in the table and concludes that:

- The Unit 1 "As-Found" minimum pathway leakage rates for the last three refueling outages since 2012 have an average of 10.8 percent of L_a with a high of 13.9 percent L_a.

- The Unit 1 "As-Left" maximum pathway leakage rates for the last three refueling outages since 2012 have an average of 17.2 percent of L_a with a high of 18.1 percent L_a.

The staff inquired in an RAI whether there had been repetitive failures of "Administrative Limits" for any Unit 1 LLRTs associated with Type B or Type C penetration since the last ILRT of November 2010. TVA responded in its letter dated February 5, 2018, that there had been two repeat failures of Unit 1 Type C tests since the last ILRT in 2010.

(1) During refueling outage U1R9 in 2012, valve 1-FCV-1-56 failed its LLRT 6 standard cubic feet per hour (scfh) administrative limit with an as-found test result of 44.696 scfh. Before the completion of U1R9, this valve was refurbished with no apparent cause of failure identified in the RAI response. The valve passed its PMT with a LLRT result of 0.197 scfh.

    During refueling outage U1R10 in 2014, 1-FCV-1-56 again failed its LLRT administrative limit with an as-found test result of 8.80 scfh. The cause of this failure was attributed to seat wear in a specific disc location. This problem was alleviated by limit switch adjustment, which allowed the valve's double disc to seat in a new location. The valve passed its post-maintenance LLRT with a test result of 0.0 scfh.

(2) During refueling outage U1R10 in 2014, valves 1-SHV-71-0014 and 1-CKV-71-0580, which is simultaneously tested, failed their LLRT 30 standard cubic feet per hour (scfh) administrative limit with an as-found test result of 124.07 scfh. The failure was attributed to 1-SHV-71-0014. Inspection of this valve revealed a scored seat. The valve seat was replaced. The valve passed its PMT with a LLRT result of 29.601 scfh. During U1R11 in 2016, as-found LLRT was 81.361 scfh. Following this failure, 1-SHV-71-0014 was refurbished and the seat was replaced. The valve passed its PMT with a LLRT result of 9.885 scfh.

TVA stated, in part, that, "A review of test results for the same components of the same design in BFN, Units 2 and 3, have not shown similar results. The upcoming RF12 test result in the fall of 2018 will determine if the action to repair the seat after the previous failure is sufficient to prevent reoccurrence."

NEI 94-01, Revision 3-A, Section 11.3.1, "Performance Factors," states that past component performance is one of the factors to be considered when establishing test intervals for Type B and Type C components. In the conglomerate for BFN, Units 1, 2, and 3, 98 percent of the eligible Type B components and 88 percent of the eligible Type C components meet the extended test interval performance criteria. These percentages demonstrate that the current maintenance strategy on the Type B and Type C components is sufficient to maintain long-term Unit 1 primary containment component isolation integrity.

In summary, the licensee provided a comprehensive response about the cause of each consecutive LLRT failure and explained the corrective actions performed to prevent repetitive and/or common cause failures (CCFs). From the review of the information contained in the LAR and the licensee's responses to the RAIs, the NRC staff concludes that the licensee has complied with the guidance of both Sections 10.2.1.2 and 10.2.3.2 of NEI 94-01, Revision 0. Also the aggregate test results at the end of each operating cycle were all significantly below (i.e., > 76 percent margin) the Type B and Type C test TS leakage rate acceptance criteria of ≤ 0.60 L_a contained in TS 5.5.12a. Therefore, the NRC staff concludes that the aggregate results of the "As-Found Minimum Pathway Leakage" for all Unit 1 Type B and Type C tests from fall 2012 through fall 2016 demonstrate a history of adequate maintenance

Based on review of the information in the LAR, and in particular, based on the licensee's RAI responses, the NRC staff concludes that the licensee's percentage of Unit 1 Type B and Type C components on extended frequencies demonstrates compliance with 10 CFR Part 50, Appendix J, Option B. This conclusion supports allowing the licensee an extended test interval of up to 75 months for the BFN, Unit 1, Type C tested CIVs in accordance with the guidance of NEI 94-01, Revision 3-A.

<u>BFN, Unit 2</u>

Unit 2 TS 5.5.12 states, in part:

> Leakage Rate acceptance criteria are:
>
> a. The primary containment leakage rate acceptance criteria is ≤ 1.0 L_a. During the first unit startup following the testing performed in accordance with this

> program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and Type C tests, and ≤ 0.75 L_a for the Type A test; and ...

The NRC staff reviewed the local leak rate summary table entitled "Browns Ferry, Unit 2" contained in the enclosure to LAR Section 4.3, "Type B and Type C Testing Programs." Section 4.3 indicates that 0.6L_a equals 655.9 scfh, which implies that L_a equals 1093.17 scfh.

With the use of this numeric L_a value and the data contained in the table, the staff derived "Fraction of L_a" values from the information contained in the table and concludes that:

- The Unit 2 "As-Found" minimum pathway leakage rates for the last three refueling outages since 2013 have an average of 14.5 percent of L_a with a high of 18.2 percent L_a.

- The Unit 2 "As-Left" maximum pathway leakage rates for the last three refueling outages since 2013 have an average of 20.6 percent of L_a with a high of 22.6 percent L_a.

The NRC staff inquired in an RAI whether there had been repetitive failures of "Administrative Limits" for any Unit 2 LLRTs associated with any Type C tests since the last ILRT of June 2009. TVA responded in its letter dated February 5, 2018 (Reference 2) that:

> Based on a review of all Browns Ferry Unit 2 administrative limit failures since the last ILRT in 2009, there have been no repetitive as-found failures (i.e., failed consecutive tests).

NEI 94-01, Revision 3-A, Section 11.3.1, "Performance Factors," states that past component performance is one of the factors to be considered when establishing test intervals for Type B and Type C components. In the conglomerate for BFN, Units 1, 2 and 3, 98 percent of the eligible Type B components and 88 percent of the eligible Type C components meet the extended test interval performance criteria. These percentages demonstrate that the current maintenance strategy on the Type B components and Type C components is sufficient to maintain long-term Unit 2 primary containment component isolation integrity.

From the review of the information contained in the LAR and the licensee's responses to the RAIs, the staff concludes that the licensee has been compliant with the guidance of both Sections 10.2.1.2 and 10.2.3.2 of NEI 94-01, Revision 0. Based on this review, the NRC staff concludes that the aggregate results of the "As-Found Minimum Pathway Leakage" for all Unit 2 Type B and Type C tests from spring 2013 through spring 2017 demonstrate a history of adequate maintenance since the aggregate test results at the end of each operating cycle were all significantly below (i.e., > 69 percent margin) the Type B and Type C test TS leakage rate acceptance criteria of ≤ 0.60 L_a contained in TS 5.5.12a.

Based on the review of the information in the LAR, and in particular, based on the licensee's RAI responses, the NRC staff concludes that the licensee's percentage of Unit 2 Type B and Type C components on extended frequencies demonstrates compliance with 10 CFR Part 50, Appendix J, Option B. This conclusion supports allowing the licensee an extended test interval of up to 75 months for the BFN, Unit 2, Type C tested CIVs in accordance with the guidance of NEI 94-01, Revision 3-A.

BFN, Unit 3

Unit 3 TS 5.5.12 states, in part:

> Leakage Rate acceptance criteria are:
>
> a. The primary containment leakage rate acceptance criteria is ≤ 1.0 L_a. During the first unit startup following the testing performed in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and Type C tests, and ≤ 0.75 L_a for the Type A test; and ...

The NRC staff reviewed the local leak rate summary table entitled "Browns Ferry, Unit 3" contained in the enclosure to the LAR, Section 4.3, "Type B and Type C Testing Programs." Section 4.3 indicates that 0.6L_a equals 655.9 scfh, which implies that L_a equals 1093.17 scfh. With the use of this numeric L_a value and the data contained in the table, the staff derived "Fraction of L_a" values from the information contained in the table and concludes that:

- The Unit 3 "As-Found" minimum pathway leakage rates for the last three refueling outages since 2012 have an average of 14.1 percent of L_a with a high of 16.3 percent L_a.
- The Unit 3 "As-Left" maximum pathway leakage rates for the last three refueling outages since 2012 have an average of 19.2 percent of L_a with a high of 20.6 percent L_a.

The NRC staff inquired in an RAI whether there had been repetitive failures of "Administrative Limits" for any Unit 3 LLRTs associated with any Type C tests since the last ILRT of May 2012. TVA responded that there had been two repeat failures of Unit 3 Type C tests since the last ILRT in 2012:

(1) In 2014, during refueling outage U3R16, 3-FCV-77-2B failed its administrative limit of 6 scfh with an as-found test result of 10.85 scfh. The corrective action performed consisted of valve was disassembly, cleaning, and inspection. The U3R16 PMT as-left result was 1.91 scfh. During U3R17 in 2016, as-found LLRT result was 6.69 scfh. Troubleshooting and evaluation attributed the U3R17 failure to LLRT boundary valve leakage. The boundary leakage was quantified and as-found leak rate adjusted for 3-FCV-77-2B to 2.549 scfh. To prevent reoccurrence, maintenance is scheduled in U3R17 to repair 3-SHV-77-601 (i.e., leaking boundary valve).

(2) 3-FCV-64-31/34/139/140 and 3-FCV-84-20 are tested simultaneously with an administrative limit of 6 scfh. During U3R16, the as-found test result was 13.34 scfh. Troubleshooting performed during U3R16 determined the major leakage to be from 3-FCV-64-140. The corrective action performed during U3R16 consisted of 3-FCV-64-140 refurbishment. The PMT as-left test result was 2.53 scfh. During U3R17 in 2016, the as-found test result was 10.72 scfh. Troubleshooting performed during U3R17 determined the major leakage to be from 3-FCV-64-139. Air operated valve diagnostics were performed, and backstop adjustments were made. The PMT as-left test result was 2.62 scfh.

TVA stated, in part, that, "Although this was a repeat failure of a composite group of valves tested, the failures were due to different valves with different failure mechanisms. No single valve failure was repeated. Because there was no repeat failure at the component level, no cause analysis with preventative recurrence controls were performed for this configuration."

NEI 94-01, Revision 3-A, Section 11.3.1, "Performance Factors," states that past component performance is one of the factors to be considered when establishing test intervals for Type B and Type C components. In the conglomerate for BFN, Units 1, 2, and 3, 98 percent of the eligible Type B components and 88 percent of the eligible Type C components meet the extended test interval performance criteria. These percentages demonstrate that the current maintenance strategy on the Type B and Type C components is sufficient to maintain long-term Unit 3 primary containment component isolation health.

In summary, the licensee provided a comprehensive response about the cause of the consecutive LLRT failures and explained the corrective actions performed to prevent repetitive and/or CCFs. From the review of the information contained in the LAR and the licensee's responses to the RAIs, the staff concludes that the licensee has been compliant with the guidance of both Sections 10.2.1.2 and 10.2.3.2 of NEI 94-01, Revision 0. Based on this review, the NRC staff concludes that the aggregate results of the "As-Found Minimum Pathway Leakage" for all Unit 3 Type B and Type C tests from spring 2012 through spring 2016 demonstrates a history of adequate maintenance since the aggregate test results at the end of each operating cycle were all significantly below (i.e., > 72 percent margin) the Type B and Type C test TS leakage rate acceptance criteria of ≤ 0.60 L_a contained in TS 5.5.12a.

Based on the review of the information in the LAR, and in particular, based on the licensee's RAI responses, the NRC staff concludes that the licensee's percentage of Unit 3 Type B and Type C components on extended frequencies demonstrates compliance with 10 CFR Part 50, Appendix J, Option B. This conclusion supports allowing the licensee an extended test interval of up to 75 months for the BFN, Unit 3, Type C tested CIVs in accordance with the guidance of NEI 94-01, Revision 3-A.

Type B and Type C Test Program Assessment – BFN

In summary, the NRC staff concludes that:

- Any lack of information or ambiguities contained in the LAR [Reference 5.1] were adequately explained by the licensee in its responses to the staff's RAI (Reference 2).
- The licensee has been compliant with the guidance of RG 1.163 and Sections 10.2.1.2, 10.2.3.2, and 11.3.1 of NEI 94-01, Revision 0.
- The cumulative Type B and Type C test results were below the acceptance limit of BFN TS 5.5.12a.
- The licensee has a corrective action program that appropriately addresses poor performing valves and penetrations.

Therefore, the NRC staff finds that the licensee is effectively implementing the BFN Type B and Type C leakage rate test program as required by Option B of 10 CFR 50, Appendix J.

3.1.3 Containment Inspection Program

The licensee stated in its LAR (Reference 1) that the inservice inspection (ISI) program plan details the requirements for the examination and testing of ASME Class 1, 2, 3, and metal containment pressure retaining components, supports, and containment structures at BFN. The ISI program plan also includes CISI.

ISI Program for Containment Metal Liner - IWE

In the LAR, the licensee stated that, currently, the BFN, Units 1, 2, and 3, Code of Record is the 2001 Edition, including 2003 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, in accordance with 10 CFR 50.55a(g)(4)(ii), and the additional requirements specified in 10 CFR 50.55a(b)(2)(ix)(A), (B), and (F) through (I). The licensee also stated the following:

> The examinations performed in accordance with the Browns Ferry IWE program satisfy the general visual examination requirements specified in 10 CFR 50, Appendix J, Option B. Identification and evaluation of inaccessible areas are addressed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix)(A). Examination of pressure-retaining bolted connections and evaluation of containment bolting flaws or degradation are performed in accordance with the requirements of 10 CFR 50.55a(b)(ix)(G) and 10 CFR 50.55a(b)(ix)(H). Each ten-year IWE interval is divided into three approximately equal duration inspection periods. A minimum of one inspection during each inspection period of the IWE interval is required by the IWE program.
>
> Subsection IWE ensures that at least three general visual examinations of metallic components would be conducted before the next Type A test, if the Type A test interval is extended to 15 years. This meets the requirements of Section 9.2.3.2 of NEI 94-01, Revision 3-A and Condition 2 in Section 4.1 of the NRC SE for NEI 94-01, Revision 2.

Table 3.1.3-1 below provides dates of completed and scheduled ILRTs, completed containment surface examinations, and an approximate schedule for future containment surface examinations, assuming the Type A test frequency is extended to 15 years.

TABLE 3.1.3-1

Completed and Scheduled ILRTs and Containment Surface Examinations

Unit 1		
Calendar Year	Type A Test (ILRT)	General Visual Examination of Accessible Interior and Exterior
2010	11/19/2010	11/17/2010
2011		
2012		
2013		
2014		10/27/2014
2015		
2016		10/27/2016
2017		
2018		
2019		
2020		10/2020

2021		
2022		
2023		
2024	11/2024	11/2024
Unit 2		
Calendar Year	Type A Test (ILRT)	General Visual Examination of Accessible Interior and Exterior
2009	06/03/2009	05/26/2009
2010		
2011		
2012		
2013		04/30/2013
2014		
2015		
2016		
2017		03/28/2017
2018		
2019		
2020		
2021		03/2021
2022		
2023		
2024	06/2024	06/2024
Unit 3		
Calendar Year	Type A Test (ILRT)	General Visual Examination of Accessible Interior and Exterior
2012	05/12/2012	05/08/2012
2013		
2014		
2015		
2016		03/24/2016
2017		
2018		
2019		
2020		03/2020
2021		
2022		
2023		
2024		03/2024
2025		
2026		
2027	05/2027	05/2027

<u>Protective Coating System</u>

In the LAR, the licensee stated that the interior carbon steel surface and all other exposed carbon steel surfaces within the pressure suppression chamber were originally coated for corrosion protection with Valspar Hi-Build Epoxy 78:00. As outlined in ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," this coating has passed test criteria for a design-basis accident (DBA) for carbon steel substrate. In addition, the licensee stated that any repairs or replacement of this protective coating system are performed using other coating system(s) that are DBA-qualified to ANSI N101.2-1972.

The NRC staff reviewed the coating inspection results and determined that the condition of the safety-related coatings program in the BFN drywell and torus are in adequate condition. There are multiple areas with coatings damaged by mechanical means; however, the surrounding substrate is in an acceptable condition, with no coatings peeling or delaminating from the point of damage. The primary cause of the degraded drywell coating is attributed to age-related degradation. Paint flakes have been found during the inspections for the last several outages and have been determined to be occurring during normal operating cycles, rather than being caused by abnormal conditions. Based on the NRC staff's evaluation of the information provided in the LAR, the staff finds that the licensee has adequate ISI programs in place to monitor and manage age-related degradation of the BFN, Units 1, 2, and 3, containment structures.

Previous Type A ILRT Results

In the LAR (Reference 1), the licensee stated that based on previous ILRT testing, it was confirmed that BFN, Units 1, 2, and 3, containment structures leakage is acceptable. As discussed in Section 3.1.1 of this SE, the NRC staff finds that a test frequency of at least once per 15 years would be in accordance with NEI 94-01, Revision 3-A, because the last three BFN, Units 1, 2, and 3, Type A results (as shown in Table 3.1.3-2) meet performance leakage rate criteria from NEI 94-01, Revision 3-A.

TABLE 3.1.3-2

Type A - Integrated Leakage Rate Testing History

Test Date	As Found Leakage		Acceptance Limit*
Unit 1			
05/06/81	Mass Point Upper Confidence Limit (UCL) leakage with penalties	0.086 of L_a	1.0 L_a
03/14/07	Mass Point UCL leakage with penalties	0.04 of L_a	1.0 L_a
11/19/10	Mass Point UCL leakage with penalties	0.10 of L_a	1.0 L_a
Unit 2			
03/18/91	Mass Point UCL leakage with penalties	0.125 of L_a	1.0 L_a
11/07/94	Mass Point UCL leakage with penalties	0.175 of L_a	1.0 L_a
06/03/09	Mass Point UCL leakage with penalties	0.31 of L_a	1.0 L_a
Unit 3			
03/22/82	Mass Point UCL leakage with penalties	0.164 of L_a	1.0 L_a
10/11/98	Mass Point UCL leakage with penalties	0.15 of L_a	1.0 L_a
05/12/12	Mass Point UCL leakage with penalties	0.29 of L_a	1.0 L_a
* The total allowable "as-left" leakage is 0.75 L_a, (L_a, 2% of primary containment air by weight per day, is the leakage assumed in DBA radiological analyses) with 0.6 L_a, the maximum leakage from Type B and Type C components.			

Type B and Type C LLRT Program

In the LAR (Reference 1), the licensee stated that BFN, Types B and C, leakage rate test program requires testing of electrical penetrations, airlocks, hatches, bellows, flanges, and valves within the scope of the program as required by Option B of 10 CFR 50, Appendix J, and TS 5.5.12. The Type B and Type C leakage rate test program consists of LLRT of penetrations with a resilient seal, hatches, bellows, flanges, and CIVs that serve as a barrier to the release of the post-accident containment atmosphere.

The licensee stated that it performed a review of the most recent Type B and Type C test results and a comparison with the allowable leakage rate and noted that the combined Type B and Type C leakage has remained below 0.6 L_a (i.e., approximately 646.3 scfh) for BFN, Unit 1, and 655.9 scfh for BFN, Units 2 and 3. Table 3.2.3 below shows the maximum and minimum pathway leak rate summary totals for the last three refueling outages.

TABLE 3.2.3

Maximum and Minimum Pathway Leak Rates

Unit 1		
U1R9 Fall 2012	As-Found Minimum Pathway Leakage	107.8 scfh
	As-Left Maximum Pathway Leakage	194.7 scfh
U1R10 Fall 2014	As-Found Minimum Pathway Leakage	149.8 scfh
	As-Left Maximum Pathway Leakage	177.8 scfh
U1R11 Fall 2016	As-Found Minimum Pathway Leakage	90.65 scfh
	As-Left Maximum Pathway Leakage	184.3 scfh
Unit 2		
U2R17 Spring 2013	As-Found Minimum Pathway Leakage	199.3 scfh
	As-Left Maximum Pathway Leakage	220.9 scfh
U2R18 Spring 2015	As-Found Minimum Pathway Leakage	152.9 scfh
	As-Left Maximum Pathway Leakage	247.6 scfh
U2R19 Spring 2017	As-Found Minimum Pathway Leakage	122.17 scfh
	As-Left Maximum Pathway Leakage	207.01 scfh
Unit 3		
U3R15 Spring 2012	As-Found Minimum Pathway Leakage	162.1 scfh
	As-Left Maximum Pathway Leakage	225.1 scfh
U3R16 Spring 2014	As-Found Minimum Pathway Leakage	123.3 scfh
	As-Left Maximum Pathway Leakage	199.5 scfh
U3R17 Spring 2016	As-Found Minimum Pathway Leakage	177.8 scfh
	As-Left Maximum Pathway Leakage	204.8 scfh

Based on the information provided above in Tables 3.2.2 and 3.2.3, the NRC staff finds that the combined leakage rate from Type A, Type B, and Type C tests remained below 1.0 L_a (the acceptable limit in Table 3.2.2). Therefore, the staff determines that the licensee's containment leakage rate testing programs and containment examination programs adequately examine, monitor, and manage age-related and environmental degradation of the BFN containment. The staff concludes that there is reasonable assurance that leaktightness of the BFN, Units 1, 2, and 3, containments is maintained.

Containment Inspection Program Summary

Based on the information provided in Tables 3.2.2 and 3.2.3, the NRC staff determined that the licensee's containment inspection programs support extension of the ILRT frequency, as requested in the LAR dated August 15, 2017 (Reference 1). Based on its evaluation as described above, the NRC staff finds that there is reasonable assurance that the structural and leaktight integrity of the BFN primary containment will continue to be monitored and maintained with the performance-based Type A test interval extended up to one test in 15 years without undue risk to public health and safety. Therefore, the staff concludes that the licensee's containment inspection programs support the proposed license amendment to: (1) change TS 5.5.12 to extend ILRT frequency to 15 years for Type A on a permanent basis; and (2) adopt NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, as the documents used by BFN to implement the performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J, as requested in the LAR.

3.1.4 NEI 94-01, Revision 2, Conditions

As required by 10 CFR 50.54(o), the three BFN primary containments are subject to the requirements set forth in 10 CFR Part 50, Appendix J. Option B of Appendix J requires that test intervals for Type A, Type B, and Type C testing be determined by using a performance-based approach. Currently, the BFN 10 CFR Part 50, Appendix J, Primary Containment Leakage Rate Testing Program invokes RG 1.163 as the plan implementation document. The licensee's LAR and supplements propose to revise the BFN 10 CFR Part 50, Appendix J, Primary Containment Leakage Rate Testing Program by replacing this implementation document with the guidance contained in NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A.

By letter dated June 25, 2008 (Reference 11), the NRC issued an SE with limitations and conditions for NEI 94-01, Revision 2. In the SE, the NRC concluded that NEI 94-01, Revision 2, describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR Part 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions, noted in Section 4.0 of the SE. Section 4.1 of the SE establishes limitations and conditions pertaining to deterministic requirements; while Section 4.2 establishes limitations and conditions pertaining to the plant's PRA analysis. More explicitly, the SE included provisions for extending the integrated leak rate testing Type A interval to a maximum of 15 years subject to the six limitations and conditions provided in the SE. The NRC noted in the SE that NEI 94-01, Revision 2, incorporates the regulatory positions stated in RG 1.163. The accepted version of NEI 94-01, Revision 2, was subsequently issued as Revision 2-A. NEI issued Revision 2-A to NEI TR 94-01 on November 19, 2008. With Revision 2-A, the TR was revised to incorporate the NRC Final Safety Evaluation Report (SER), dated June 25, 2008.

TVA provided responses to each of the six limitations and conditions provided in the SE dated June 25, 2008, in the first table in LAR Section 4.0, "Technical Evaluation." The NRC staff's review of this table indicates that BFN intends to satisfy the limitations and conditions of NEI 94-01 Revision 2, Section 4.1. Accordingly, as previously noted, BFN intends to adopt the testing methodology of ANSI/ANS 56.8–2002 in place of the methodology of ANSI/ANS 56.8-1994 (Reference 19).

The leakage rate testing requirements of 10 CFR Part 50, Appendix J, Option B (Types A, B, and C Tests), and the CISI requirements mandated by 10 CFR 50.55a, together, ensure the continued leaktight and structural integrity of the BFN primary containments during their service lives.

Type B testing ensures that the leakage rate of individual containment penetration components is acceptable. Type C testing ensures that individual CIVs are essentially leaktight. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of both primary containments by minimizing potential leakage paths.

In the LAR, the licensee proposes that BFN invoke NEI 94-01, Revision 3-A, along with the conditions and limitations of NEI 94-01, Revision 2-A, as the reference documents for the BFN "Primary Containment Leakage Rate Testing Program" in TS 5.5.12. Therefore, the licensee is also requesting to extend the frequencies of the Type C performance-based test intervals beyond 60 months.

The NRC staff has found that the use of NEI TR 94-01, Revision 2-A, is acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT surveillance interval to 15 years, provided the following applicable six limitations and conditions are satisfied:

Condition 1

The first limitation/condition of Enclosure 1 of the LAR is derived from Sections 3.1.1.1 and 4.1 (i.e., Attachment 1) of the NRC SE dated June 25, 2008, and stipulates that for calculating the Type A leakage rate, the licensee should use the definition in NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS-56.8-2002.

TVA Response Limitation/Condition 1

In the first table contained in Section 4.0 of Enclosure 1 to the LAR, the licensee state:

> Following NRC approval of this LAR, Browns Ferry, Units 1, 2, and 3, will use the definition in Section 5.0 of NEI 94-01, Revision 3-A, for calculating the Type A leakage rate when future Browns Ferry, Units 1, 2, and 3 Type A tests are performed. The definitions in Revision 2-A and 3-A are identical.

Staff Assessment

Section 3.2.9, "Type A test performance criterion," of ANSI/ANS-56.8-2002 defines the "performance leakage rate" and states, in part:

> The performance criterion for a Type A test is met if the performance leakage rate is less than L_a. The performance leakage rate is equal to the sum of the measured Type A test UCL and the total as-left MNPLR of all Type B or Type C pathways isolated during performance of the Type A test.

NRC staff SE Section 3.1.1.1, Enclosure page 6 for NEI 94-01, Revision 2, states, in part:

> Section 5.0 of NEI TR 94-01, Revision 2, uses a definition of "performance leakage rate" for Type A tests that is different from that of ANSI/ANS-56.8-2002. The definition contained in NEI TR 94-01, Revision 2, is more inclusive because it considers excessive leakage in the performance determination. In defining the

> minimum pathway leakage rate, NEI TR 94-01, Revision 2, includes the leakage rate for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position prior to the performance of the Type A test. Additionally, the NEI TR 94-01, Revision 2, definition of performance leakage rate requires consideration of the leakage pathways that were isolated during performance of the test because of excessive leakage in the performance determination. The NRC staff finds this modification of the definition of "performance leakage rate" used for Type A tests to be acceptable.

Section 5.0, "Definitions," of NEI 94-01, Revision 2-A, states, in part:

> The performance leakage rate is calculated as the sum of the Type A upper confidence limit (UCL) and as-left minimum pathway leakage rate (MNPLR) leakage rate for all Type B and Type C pathways that were inservice, isolated, or not lined up in their test position (i.e., drained and vented to containment atmosphere) prior to performing the Type A test. In addition, leakage pathways that were isolated during performance of the test because of excessive leakage must be factored into the performance determination. The performance criterion for Type A tests is a performance leak rate of less than $1.0L_a$.

The NRC staff reviewed the definitions of "performance leakage rate" contained in NEI 94-01, Revisions 2 and 2-A. The NRC staff concluded that the definitions contained in both documents are identical.

Therefore, the NRC staff concludes that BFN will use the definition found in Section 5.0 of NEI 94-01, Revision 2 (Reference 20), for calculating the Type A leakage rate in the BFN "Primary Containment Leakage Rate Testing Program."

Based on the above review, the NRC staff finds that the licensee has adequately addressed "Condition 1."

<u>Condition 2</u>

The second limitation/condition of Enclosure 1 to the LAR is derived from Sections 3.1.1.3 and 4.1 of the NRC SE dated June 25, 2008 (Reference 11), and stipulates that the licensee submits a schedule of containment inspections to be performed prior to and between Type A tests.

<u>BFN Response to Limitation/Condition 2</u>

In the first table contained in Section 4.0 of Enclosure 1 to the LAR, the licensee stated:

> The schedule of containment inspections is provided in Section 4.4 below.

<u>Staff Assessment</u>

NRC staff SE Section 3.1.1.3 for NEI 94-01, Revision 2, states, in part:

> NEI TR 94-01, Revision 2, Section 9.2.3.2, states that: "To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment

> for structural deterioration that may affect the containment leaktight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test, if the interval for the Type A test has been extended to 15 years." NEI TR 94-01, Revision 2, recommends that these inspections be performed in conjunction or coordinated with the examinations required by ASME Code, Section XI, Subsections IWE and IWL. The NRC staff finds that these visual examination provisions, which are consistent with the provisions of regulatory position C.3 of RG 1.163, are acceptable considering the longer 15 year interval. Regulatory Position C.3 of RG 1.163 recommends that such examination be performed at least two more times in the period of 10 years. The NRC staff agrees that as the Type A test interval is changed to 15 years, the schedule of visual inspections should also be revised. Section 9.2.3.2 in NEI TR 94-01, Revision 2, addresses the supplemental inspection requirements that are acceptable to the NRC staff.

NEI 94-01, Revision 3-A, Section 9.2.1, "Pretest Inspection and Test Methodology," states, in part:

> Prior to initiating a Type A test, a visual examination shall be conducted of accessible interior and exterior surfaces of the containment system for structural problems that may affect either the containment structure leakage integrity or the performance of the Type A test. This inspection should be a general visual inspection of accessible interior and exterior surfaces of the primary containment and components. It is recommended that these inspections be performed in conjunction or coordinated with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE/IWL required examinations.

NEI 94-01, Revision 3-A, Section 9.2.3.2, "Supplemental Inspection Requirements," states:

> To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leaktight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years. It is recommended that these inspections be performed in conjunction or coordinated with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE/IWL required examinations.

BFN, Units 1, 2, and 3, have no ISI Class CC components that meet the criteria of Subarticle IWL-1100; therefore, no requirements to perform examinations in accordance with Subsection IWL are incorporated into the BFN containment ISI plan.

Prior to initiating a Type A test, a general visual examination of accessible interior and exterior surfaces of the BFN containment systems is performed to identify any potential structural problems that could affect either the containment structure leakage integrity or the performance of the Type A test. These inspections are typically conducted in accordance with the BFN CISI program, which implements the requirements of ASME, Section XI, Subsection IWE. The examinations performed in accordance with the BFN IWE program satisfy the general visual examination requirements specified in 10 CFR Part 50, Appendix J, Option B.

As noted above in this SE Table 3.1.1-1, and again in the "Unit 1" table of LAR Section 4.4, "Supplemental Inspection Requirements," the last Unit 1 Type A test was completed on November 19, 2010. Accordingly, with the LAR proposed extension of Unit 1 Type A test interval to a maximum of 15 years, the next Unit 1 Type A test would be due for completion prior to the end of November 2025. Based on the review of the LAR Section 4.4 Unit 1 table, the staff concludes that the requirements of NEI 94-01, Revision 3-A, Section 9.2.3.2, can be satisfied by the licensee. As displayed in the above SE Table 3.1.1-2 and in the "Unit 2" table of LAR Section 4.4, the last Unit 2 Type A test was completed on June 3, 2009. It follows that with the licensee's proposed extension of the Unit 2 Type A test interval to a maximum of 15 years, the next Unit 2 Type A test would be due for completion prior to the end of June 2024. Based on the review of the LAR Section 4.4 Unit 2 table, the staff concludes that the requirements of NEI 94-01, Revision 3-A, Section 9.2.3.2, can be satisfied by the licensee. Similarly, as displayed in both SE Table 3.1.1-3 and the "Unit 3" table of LAR Section 4.4, the last Unit 3 Type A test was completed on May 12, 2012. It follows that with the licensee's proposed extension of the Unit 3 Type A test interval to a maximum of 15 years, the next Unit 3 Type A test would be due for completion prior to the end of May 2027. Based on the review of the LAR Section 4.4 Unit 3 table, the staff concludes that the requirements of NEI 94-01, Revision 3-A, Section 9.2.3.2, can be satisfied by the licensee.

Subsection IWE ensures that at least three general visual examinations of BFN metallic components would be conducted before the next Type A test, if the Type A test interval is extended to 15 years. The schedules to satisfy this requirement for extending the Type A test interval to 15 years for all BFN, Units 1, 2, and 3, were documented in the three tables entitled "Unit 1," "Unit 2," and "Unit 3" of LAR (Reference 1), Section 4.4.

Based on the foregoing discussion, the NRC staff concludes that the licensee intends to comply with the guidance contained in NEI 94-01, Revision 3-A, Sections 9.2.1 and 9.2.3.2, and that BFN intends to satisfy the provisions contained in NRC staff SE Section 3.1.1.3. Accordingly, the NRC staff finds that the licensee has adequately addressed "Condition 2."

Condition 3

The third limitation/condition of Enclosure 1 to the LAR is derived from Sections 3.1.3 and 4.1 of the NRC SE dated June 25, 2008, and stipulates that the licensee addresses the areas of the containment structure potentially subjected to degradation.

BFN Response to Limitation/Condition 3

In the first table contained in LAR Section 4.0 of Enclosure 1 to the LAR, the licensee stated:

> General visual examination of accessible interior and exterior surfaces of the containment system for structural problems is conducted in accordance with the Browns Ferry, Units 1, 2, and 3, Containment Inservice Inspection Plans that implement the requirements of the ASME, Section XI, Subsection IWE, as required by 10 CFR 50.55a. The provisions of 10CFR50.55a relative to examination of Class CC [concrete containment] components (ASME Section XI Subsection IWL) are not applicable to Browns Ferry, since Browns Ferry does not have any ASME Code Class CC equivalent components.

Staff Assessment

The NRC staff reviewed the information contained in LAR Section 4.4.1, "IWE Examinations."

NRC staff SE (Reference 11) Section 3.1.3 of the enclosure for NEI 94-01, Revision 2, states, in part:

> In approving for Type A tests the one-time extension from 10 years to 15 years, the NRC staff has identified areas that need to be specifically addressed during the IWE and IWL inspections including a number of containment pressure-retaining boundary components (e.g., seals and gaskets of mechanical and electrical penetrations, bolting, penetration bellows) and a number of the accessible and inaccessible areas of the containment structures (e.g., moisture barriers, steel shells, and liners backed by concrete, inaccessible areas of ice condenser containments that are potentially subject to corrosion).

BFN has no ISI Class CC components that meet the criteria of Subarticle IWL-1100; therefore, no requirements to perform examinations in accordance with Subsection IWL are incorporated into this containment ISI plan.

General visual examinations of the accessible surfaces of containment are performed to assess the general condition of each of the accessible BFN interior and exterior primary containment vessels prior to each Type A test. In addition, Subsection IWE ensures that at least three general visual examinations of metallic components would be conducted before the next Type A test, if the Type A test interval is extended to 15 years. The BFN schedules for satisfying this requirement are contained in Section 4.1 of the NRC SE for NEI 94-01, Revision 2, and are reflected in the three tables labeled as "Unit 1," "Unit 2," and "Unit 3" in LAR Section 4.4, "Supplemental Inspection Requirements."

Section 4.4 of Enclosure 1 to the LAR states, in part:

> The examinations performed in accordance with the Browns Ferry IWE program satisfy the general visual examination requirements specified in 10 CFR 50, Appendix J, Option B. Identification and evaluation of inaccessible areas are addressed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix)(A). Examination of pressure-retaining bolted connections and evaluation of containment bolting flaws or degradation are performed in accordance with the requirements of 10 CFR 50.55a(b)(ix)(G) and 10 CFR 50.55a(b)(ix)(H).

For each unit at BFN, each 10-year IWE interval is divided into three approximately equal duration inspection periods. A minimum of one inspection during each inspection period of the IWE interval is required by the IWE program. LAR Section 4.4.1, "IWE Examinations," details several historical examples of documented nonconforming conditions from IWE "First ten-year interval" and "Second ten-year interval" examination results. These examples provided inspection results pertaining to moisture seal barriers and bellow assemblies.

One area of both Units 2 and 3 that was determined to be susceptible to accelerated degradation and aging was the Suppression Pool waterline region. These areas in both units experienced repeated loss of coatings and were designated for augmented examination.

In summary, the NRC staff concludes that based on the information presented in LAR Section 4.4.1, the licensee has established its intent to satisfy the issues of SE Section 3.1.3. Accordingly, the NRC staff finds that the licensee has adequately addressed "Condition 3."

Condition 4

The fourth limitation/condition of Enclosure 1 to the LAR is derived from Sections 3.1.4 and 4.1 of the NRC SE dated June 25, 2008, and stipulates that the licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable.

BFN Response to Limitation/Condition 4

In the first table contained in LAR Section 4.0 of Enclosure 1 to the LAR, the licensee stated:

> Any future containment modifications will be addressed by the site design change process including any containment post-modification testing as required by Section 3.1.4 of the NRC staff SE for NEI 94-01, Revision 2.

Staff Assessment

NRC staff SE Section 3.1.4 for NEI 94-01, Revision 2, states, in part:

> Section 9.2.4 of NEI TR 94-01, Revision 2, states that: "Repairs and modifications that affect the containment leakage integrity require LLRT or short duration structural tests as appropriate to provide assurance of containment integrity following the modification or repair. This testing shall be performed prior to returning the containment to operation." Article IWE-5000 of the ASME Code, Section XI, Subsection IWE (up to the 2001 Edition and the 2003 Addenda), would require a Type A test after major repair or modifications to the containment. In general, the NRC staff considers the cutting of a large hole in the containment for replacement of steam generators or reactor vessel heads, replacement of large penetrations, as major repair or modifications to the containment structure.

The NRC staff notes that this condition is intended to verify any past major modification or maintenance repair of the primary, since the last ILRT has been appropriately accompanied by either a structural integrity test or ILRT and that any plans for future major modification also include appropriate pressure testing. The last Type A test for the primary containments was performed during November 2010 for Unit 1, June 2009 for Unit 2, and May 2012 for Unit 3.

LAR Section 4.2, "Integrated Leak Rate Test History," states, in part:

> There are no known modifications that will require a Type A test to be performed prior to U1R15 (Fall 2024), U2R22 (Spring 2023), and U3R22 (Spring 2026), when the next Type A tests will be performed in accordance with this proposed change. Any unplanned modifications to the containment prior to the next scheduled Type A test would be subject to the testing requirements of NEI 94-01, Revision 0, Section 9.2.4 or NEI 94-01, Revision 3-A, Section 9.2.4, as applicable.
>
> There have been no pressure or temperature excursions in either Browns Ferry, Units 1, 2, or 3 containments which could have adversely affected containment

> integrity since the performance of the last Type A tests. There is no other anticipated addition or removal of plant hardware within either Browns Ferry, Units 1, 2, or 3 containments that could affect leaktightness.

The NRC staff notes that the above excerpt from LAR Section 4.2 and the TVA response to Condition 4 is forward looking with respect to plans for any future BFN containment modifications. However, all three BFN containments have been commissioned for at least 25 years. In an RAI, the staff requested that the licensee provide historical information (i.e., a synopsis) about any modifications and the subsequent associated post-modification testing to the BFN containments since the most recent ILRTs for each unit.

TVA responded by letter dated February 5, 2018 (Reference 2), that:

> There have been no major containment structure modifications made to the Browns Ferry Unit 1 containment since last ILRT performed on 11/8/2010.
>
> There have been no major containment structure modifications made to the Browns Ferry Unit 2 containment since last ILRT performed on 5/27/2009.
>
> There have been no major containment structure modifications made to the Browns Ferry Unit 3 containment since last ILRT performed on 5/9/2012.

Therefore, the staff concludes that there have been no major containment structural modifications for any of the BFN containments since the most recent ILRTs. Furthermore, there are no major modifications planned that could affect its leaktightness and subsequently require either a structural integrity test or ILRT. Therefore, the NRC staff concludes that the licensee has adequately addressed the issues of SE Section 3.1.4 and "Condition 4."

Condition 5

The fifth limitation/condition of Enclosure 1 to the LAR is derived from Sections 3.1.1.2 and 4.1 of the NRC SE dated June 25, 2008, and stipulates that the normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition.

BFN Response to Limitation/Condition 5

In the first table contained in LAR Section 4.0 of Enclosure 1 to the LAR, the licensee stated, in part:

> TVA acknowledges and accepts this NRC staff position, as communicated to the nuclear industry in Regulatory Issue Summary 2008-27 dated December 8, 2008 (Reference 21).

Staff Assessment

NRC staff SE (Reference 11) Section 3.1.1.2 for NEI 94-01, Revision 2, states:

> As noted above, Section 9.2.3, NEI TR 94-01, Revision 2, states, "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 15 years based on acceptable performance history." However, Section 9.1 states that the "required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions but should not be used for routine scheduling and planning purposes." The NRC staff believes that extensions of the performance-based Type A test interval beyond the required 15 years should be infrequent and used only for compelling reasons. Therefore, if a licensee wants to use the provisions of Section 9.1 in TR NEI 94-01, Revision 2, the licensee will have to demonstrate to the NRC staff that an unforeseen emergent condition exists.

The licensee stated in its response that TVA acknowledges and accepts the NRC staff position, as communicated to the nuclear industry in Regulatory Issue Summary 2008-27. The above passage from SE Section 3.1.1.2 accurately reflects the regulatory issue summary NRC staff position. Therefore, the licensee has demonstrated its understanding that any extension of the Type A test interval beyond the upper-bound performance-based limit of 15 years should be infrequent and that any requested permission (i.e., for such an extension) will demonstrate to the NRC staff that an unforeseen emergent condition exists.

Based on the above review, the NRC staff finds that the licensee has adequately addressed "Condition 5."

Condition 6

The sixth limitation/condition of Enclosure 1 to the LAR is derived from Section 4.1 of the NRC SE dated June 25, 2008, and stipulates that for plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2 (Reference 22), including the use of past containment ILRT data.

BFN Response to Limitation/Condition 6

In the first table contained in Section 4.0 of Enclosure 1 to the LAR, the licensee stated:

> Not applicable. Browns Ferry Units 1, 2 and 3 are not licensed pursuant to 10 CFR Part 52.

Staff Assessment

The NRC staff finds that "Condition 6" does not apply, since BFN, Units 1, 2, and 3, are not licensed under 10 CFR Part 52.

Summary for NEI 94-01, Revision 2 Conditions

Based on the above evaluation of each condition, the NRC staff determined that the licensee has adequately addressed the six conditions identified in Section 4.1 of the NRC SE for TR NEI 94-01, Revision 2-A. Therefore, the NRC staff finds it acceptable for BFN to adopt the "conditions and limitations" of NEI 94-01, Revision 2-A, as part of the implementation documents in BFN TS 5.5.12, "Primary Containment Leakage Rate Testing Program," for all three units.

3.1.5 NEI 94-01, Revision 3, Conditions Satisfied

As required by 10 CFR 50.54(o), the BFN primary containments are subject to the requirements set forth in 10 CFR Part 50, Appendix J. Option B of Appendix J allows that test intervals for Types A, B, and C testing be determined by using a performance-based approach. Currently, BFN TS 5.5.12 is implemented in accordance with the guidelines contained in RG 1.163, dated September 1995 (Reference 5), as modified by approved exemptions. As noted in SE Section 2.2, both Units 2 and 3 TS 5.5.12 contain an outdated exception to Section 9.2.3 of NEI 94-01−1995. The licensee's LAR and supplements propose to revise the BFN TS 5.5.12 by replacing Option B implementation document RG 1.163 with NEI 94-01, Revision 3-A, along with the conditions and limitations of NEI 94-01, Revision 2-A to govern the test frequencies and the grace periods for Types A, B, and C tests. In addition, the LAR proposes to delete the two obsolete exceptions associated with TS 5.5.12 for both Units 2 and 3.

By letter dated June 8, 2012 (Reference 23), the NRC published an SE with limitations and conditions for NEI 94-01, Revision 3. In the SE, the NRC concluded that NEI 94-01, Revision 3, describes an acceptable approach for implementing the optional performance-based requirements of 10 CFR Part 50, Appendix J, and is acceptable for referencing by licensees proposing to amend their TSs regarding containment leakage rate testing subject to the limitations and conditions identified in SE Section 4.0 and summarized in SE Section 5.0. The accepted version of NEI 94-01, Revision 3, was subsequently issued as Revision 3-A. NEI issued Revision 3-A to NEI 94-01 on July 31, 2012 (Reference 6). With Revision 3-A, the report was revised to incorporate the June 8, 2012, NRC Final SER.

The licensee indicated in the LAR that BFN will meet the limitations and conditions of NEI 94-01, Revision 3-A, Section 4.0. Accordingly, all three units will be adopting, in part, the testing criteria ANSI/ANS 56.8–2002 as part of its licensing basis. As stated in Section 2.0, "Purpose and Scope," of NEI 94-01, Revision 3-A, where technical guidance overlaps between NEI 94-01, Revision 3-A, and ANSI/ANS 56.8–2002, the guidance of NEI 94-01, Revision 3-A, takes precedence.

In the LAR, the licensee proposes to invoke NEI 94-01, Revision 3-A, as the implementation document for BFN TS 5.5.12 to govern its Type B and Type C LLRT programs.

The NRC staff has found that the use of NEI TR 94-01, Revision 3, is an acceptable reference for use in licensee TSs to extend Option B to 10 CFR Part 50, Appendix J, Type B test and Type C test intervals beyond 60 months, provided the following two conditions are satisfied:

Condition 1

Section 4.0 of Enclosure of the NRC letter dated June 8, 2012 (Reference 23), stipulates that:

> NEI TR 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs be increased to 75 months with the requirement that a licensee's post-outage report include the margin between the Type B and

> Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84-months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g., BWR [boiling water reactor] MSIVs [main steam isolation valves]), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months.

Condition 1 presents three separate issues that are required to be addressed:

(1) The allowance of an extended interval for Type C LLRTs of 75 months carries the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit;

(2) A corrective action plan shall be developed to restore the margin to an acceptable level; and

(3) Use of the allowed 9-month extension for eligible Type C valves is only authorized for nonroutine emergent conditions with exceptions as detailed in NEI 94-01, Revision 3-A, Section 10.1.

In the second table contained in Section 4.0 of Enclosure 1 to the LAR (Reference 1), the licensee responded to Condition 1 by stating:

> Following NRC approval of this LAR, Browns Ferry, Units 1, 2 and 3, will follow the guidance of NEI 94-01, Revision 3-A to assess and monitor margin between the Type B and Type C leakage rate summation and the regulatory limit. This will include corrective actions to restore margin to an acceptable level.

Staff Assessment

The NRC staff notes that with approval of the LAR, BFN TS 5.12 will identify NEI 94-01, Revision 3-A, as the implementation document for the 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B program. The staff also notes that the licensee stated, "Following NRC approval of this LAR, BFN, Units 1, 2 and 3, will follow the guidance of NEI 94-01, Revision 3-A." The staff notes that the requirement of all three issues is specifically addressed in the following sections of NEI 94-01, Revision 3-A.

- Issue (1) – Section 12.1, "Report Requirements"
- Issue (2) – Section 12.1, "Report Requirements"
- Issue (3) – Section 10.1, "Introduction"

The NRC staff has reviewed the requirements of NEI TR 94-01, Revision 3, against the licensee's response for Condition 1. Based on this review, the NRC staff concludes that the licensee acknowledges all the requirements of Condition 1 and that the licensee has established its intent for BFN to comply with these requirements.

Condition 2

Section 4.0 of Enclosure 1 to the NRC letter dated June 8, 2012 (Reference 23), stipulates that:

> The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust containment inspection program and the local leakage rate testing of penetrations. Most of the containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time. The containment leakage condition monitoring regime involves a portion of the penetrations being tested each refueling outage, nearly all LLRTs being performed during plant outages. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as-found minimum pathway leakage rates for the just tested penetrations are summed with the as-left minimum pathway leakage rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Type C tests involve valves, which in the aggregate, will show increasing leakage potential due to normal wear and tear, some predictable and some not so predictable. Routine and appropriate maintenance may extend this increasing leakage potential. Allowing for longer intervals between LLRTs means that more leakage rate test results from farther back in time are summed with fewer just tested penetrations and that total is used to assess the current containment leakage potential. This leads to the possibility that the LLRT totals calculated understate the actual leakage potential of the penetrations. Given the required margin included with the performance criterion and the considerable extra margin most plants consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI TR 94-01, Revision 3, Section 12.1.
>
> When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Types B and C total leakage, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Condition 2 presents two separate issues that are required to be addressed:

(1) Extending the Type C, LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative, provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI TR 94-01, Revision 3, Section 12.1.

(2) When routinely scheduling any LLRT valve interval beyond 60 months and up to 75 months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B and Type C total, and must be included in a licensee's post-outage report. The report must include

the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

In the second table contained in Section 4.0 of Enclosure 1 to the LAR, the licensee responded to Condition 2 by stating:

> Following NRC approval of this LAR, Browns Ferry, Units 1, 2, and 3, will estimate the amount of understatement in the Types B and C total and include determination of the acceptability in a post-outage report, consistent with the guidance of Section 11.3.2 of NEI 94-01, Revision 3-A.

Staff Assessment

The NRC staff notes that with approval of the LAR, BFN TS 5.12 will identify NEI 9-01, Revision 3-A, as the implementation document for the 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B program. The staff notes that the licensee indicated that the 10 CFR Part 50, Appendix J, Option B program will be "consistent with the guidance of Section 11.3.2 of NEI 94-01, Revision 3-A."

Section 11.3.2 states, in part:

> Type B and C leakage - When routinely scheduling any LLRT interval beyond 60-months, the primary containment leakage rate testing program trending or monitoring (Section 12.1) shall include an estimate of the amount of understatement in the minimum pathway Type B & C summation. The estimate must be included in the post-outage report of Section 12.1 and include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

The NRC staff has reviewed the requirements of NEI TR 94-01, Revision 3, against the licensee's response for Condition 2 and against the above excerpt from Section 11.3.2 of NEI 94-01, Revision 3-A. Based on this review, the NRC staff concludes that the licensee acknowledges all the requirements of Condition 2 and that the licensee has established its intent for BFN to comply with these requirements.

Summary

Based on the above evaluation of each condition, the NRC staff determined that the licensee has adequately addressed both conditions in Section 4.0 of the NRC SE for TR NEI 94-01, Revision 3-A. Therefore, the NRC staff finds it acceptable for BFN to adopt TR NEI 94-01, Revision 3-A, as the implementation document in TS 5.5.12, "Primary Containment Leakage Rate Testing Program" for BFN.

3.1.6 NRC Overall Evaluation of the Proposed Extension of ILRT and LLRT Test Intervals

The NRC staff reviewed the Types A, B, and C leakage test results related to the licensee's proposal to extend 10 CFR Part 50, Appendix J test intervals.

The ILRT results provided in the table contained in LAR Section 4.2, "Integrated Leak Rate Test History," for "Browns Ferry, Unit 1" indicate that the previous two consecutive Type A tests for

Unit 1 were successful, with primary containment performance leakage rates less than the maximum allowable (i.e., 1.0 L_a at P_a) contained in the leakage rate acceptance criteria of TS 5.5.12. Therefore, the NRC staff finds that the performance history of the Unit 1 Type A tests supports extending the current ILRT interval on a permanent basis to 15 years, as permitted by NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A.

Also, the ILRT results provided in the table contained in LAR Section 4.2, "Integrated Leak Rate Test History," for "Browns Ferry, Unit 2" indicate that the previous two consecutive Type A tests for Unit 2 were successful, with primary containment performance leakage rates less than the maximum allowable (i.e., 1.0 L_a at P_a) contained in the leakage rate acceptance criteria of TS 5.5.12. Therefore, the NRC staff finds that the performance history of the Unit 2 Type A tests supports extending the current ILRT interval on a permanent basis to 15 years, as permitted by NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A.

Similarly, the ILRT results provided in the table contained in LAR Section 4.2, "Integrated Leak Rate Test History," for "Browns Ferry, Unit 3" indicate that the previous two consecutive Type A tests for Unit 3 were successful, with primary containment performance leakage rates less than the maximum allowable (i.e., 1.0 L_a at P_a) contained in the leakage rate acceptance criteria of TS 5.5.12. Therefore, the NRC staff finds that the performance history of the Unit 3 Type A tests supports extending the current ILRT interval on a permanent basis to 15 years, as permitted by NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A.

The NRC staff reviewed the Unit 1 "As-Found Minimum Pathway Leakage" local leak rates contained in the table contained in LAR Section 4.3, "Type B and Type C Testing Programs," and notes that the results of the "As-Found Minimum Pathway Leakage" for all the recent (i.e., since refueling outage U1R9 in 2012) Type B and Type C tests are significantly less (i.e., > 76 percent margin) than the Type B and Type C test TS limit of ≤ 0.60 L_a contained in TS 5.5.12a. The NRC staff reviewed the information contained in the licensee's response to SCPB RAI 2 with details about consecutive repetitive failures of "Administrative Limits" and subsequent corrective actions for LLRTs associated with the Unit 1 Type C tests since the last ILRT of November 2010. Based on the responses to SCPB RAI 2 and SCPB RAI 3, the staff concludes that the licensee takes adequate corrective action for failed Unit 1 LLRTs.

Also, the NRC staff reviewed the Unit 2 "As-Found Minimum Pathway Leakage" local leak rates contained in the table contained in LAR Section 4.3, "Type B and Type C Testing Programs," and notes that the results of the "As-Found Minimum Pathway Leakage" for all the recent (i.e., since refueling outage U2R17 in 2013) Type B and Type C tests are significantly less (i.e., > 69 percent margin) than the Type B and Type C test TS limit of ≤ 0.60 L_a contained in TS 5.5.12a. The NRC staff reviewed the information contained in the licensee's response to SCPB RAI 2. In its response, TVA indicated that "there have been no repetitive as-found failures (i.e., failed consecutive tests)" since the last ILRT of June 2009. Furthermore, the high percentages of Type B and Type C components currently on extended test intervals demonstrate that the current maintenance strategy for the Type B and Type C components is sufficient to maintain long-term Unit 2 primary containment component isolation integrity. Based on its responses to SCPB RAI 2 and SCPB RAI 3, the staff concludes that the licensee takes adequate corrective action for the failed Unit 2 LLRTs.

Similarly, the NRC staff reviewed the Unit 3 "As-Found Minimum Pathway Leakage" local leak rates contained in the table contained in LAR Section 4.3, "Type B and Type C Testing Programs," and notes that the results of the "As-Found Minimum Pathway Leakage" for all the recent (i.e., since refueling outage U3R15 in 2012) Type B and Type C tests are significantly less (i.e., > 72 percent margin) than the Type B and Type C test TS limit of ≤ 0.60 L_a contained in TS 5.5.12a.

The NRC staff reviewed the information contained in the licensee's response to SCPB RAI 2 with details about repetitive failures of "Administrative Limits" and subsequent corrective actions for LLRTs associated with the Unit 3 Type C tests since the last ILRT of May 2012. Based on its responses to SCPB RAI 2 and SCPB RAI 3, the staff concludes that the licensee takes adequate corrective action for the failed Unit 3 LLRTs.
Accordingly, the NRC staff finds that the licensee is effectively implementing the BFN, Units 1, 2, and 3, Type B and Type C leakage rate test programs, as required by 10 CFR Part 50, Appendix J, Option B. Accordingly, the NRC staff finds that the performance history of Type B tests and Type C tests supports extending the current Type C test interval to 75 months, as permitted by NEI 94-01, Revision 3-A, for BFN.

3.1.6 Conclusion of Deterministic Evaluation

In the LAR and supplements, the licensee proposed to extend the BFN current performance-based Type A test interval to no longer than 15 years by adopting NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A, as the implementation documents in TS 5.5.12. This change would allow Unit 1 to conduct the next Type A test no later than November 2025, in lieu of the current requirement of no later than plant restart from refueling outage U1R13 (i.e., fall 2020). This change would also allow Unit 2 to conduct the next Type A test no later than June 2024, in lieu of the current requirement of no later than plant restart from refueling outage U2R20 (i.e., spring 2019). Similarly, this change would allow Unit 3 to conduct the next Type A test no later than May 2027, in lieu of the current requirement of no later than plant restart from refueling outage U3R20 (i.e., spring 2022).

Consistent with the guidance in NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A, the licensee justified the proposed change by demonstrating adequate performance of the BFN primary containments based on: (a) plant-specific containment leakage testing program results, (b) CISI results, and (c) a plant-specific risk assessment.

Based on the NRC staff review of the LAR dated August 15, 2017 (Reference 1); supplemental information provided in the RAI response letter of February 5, 2018 (Reference 2), and the regulatory and technical evaluations above, NRC staff finds there is reasonable assurance that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, as the 10 CFR Part 50, Appendix J, Option B implementation documents for BFN, Units 1, 2, and 3.

The NRC staff also finds that the licensee adequately implemented its Primary Containment Leakage Rate Testing Program (i.e., Types A, B, and C leakage tests) for the BFN primary containments. The results of past ILRTs and recent LLRTs demonstrate acceptable performance of the BFN primary containments and demonstrate that the structural and leaktight integrity of the containment structures are being adequately maintained. The NRC staff also finds that the structural and leaktight integrity of the BFN primary containments will continue to be monitored and maintained if TVA adopts NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, as the 10 CFR Part 50, Appendix J, Option B implementation documents for Units 1, 2, and 3. Accordingly, the NRC staff determined that there is reasonable assurance that the structural and leaktight integrity for the BFN primary containments will continue to be maintained, without undue risk to public health and safety, if the current Type A test intervals are extended to 15 years and if the current Type C test intervals for qualifying CIVs are extended to 75-months.

The NRC staff concludes that it is acceptable for BFN to:

- Revise TS 5.5.12, "Primary Containment Leakage Rate Testing Program," to adopt NEI 94-01 3-A and the conditions and limitations specified in NEI 94-01, Revision 2-A, as the 10 CFR Part 50, Appendix J, Option B implementation documents.
- Delete the existing the historical exception in TS 5.5.12 for Units 2 and 3.
- Extend on a permanent basis the Type A test interval up to 15 years.
- Extend the Type C test intervals for qualifying CIVs up to 75-months.

3.2 The NRC Staff's Risk-Informed Evaluation

The key information used in the NRC staff's review of the licensee's risk evaluation is contained in the LAR Section 4.6, "Confirmatory Analysis," dated August 15, 2017 (Reference 1), as supplemented by letters dated February 5, 2018 (Reference 2); March 27, 2018 [(Reference 3); and July 27, 2018 (Reference 4). Additionally, the NRC staff reviewed the SE in support of the amendment issued for "BFN Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c)," dated October 28, 2015 (Reference 24), to support the assessment of the licensee's PRA acceptability for use in the ILRT risk evaluation.

3.2.1 Method of Staff Review

An acceptable approach for making risk-informed decisions about proposed TS changes, including both permanent and temporary changes, is to show that the proposed changes meet the five key principles stated in RG 1.174, Section 2. These key principles are:

Principle (1) The proposed change meets the current regulations unless it is explicitly related to a requested exemption.

Principle (2) The proposed change is consistent with the defense-in-depth philosophy.

Principle (3) The proposed change maintains sufficient safety margins.

Principle (4) When the proposed change results in an increase in core damage frequency (CDF) and/or large early release frequency (LERF), the increase should be small and consistent with the intent of the Commission's Safety Goal Policy statement.

Principle (5) The impact of the proposed change should be monitored by using performance measures strategies.

The deterministic evaluations presented in Section 3.1 of this SE address the first three key principles of the staff's standards for risk-informed decisionmaking, which concern compliance with current regulations, evaluation of defense-in-depth, and evaluation of safety margins.

The NRC staff's evaluations of the licensee's LAR and supplements that are associated with Principles 4 and 5 of risk-informed evaluation are presented in the following subsections.

3.2.2 Key Principle 4: Change in Risk is Consistent with the Commission's Policy Statement on Safety Goals

The NRC's evaluation of Key Principle 4, which is centered on risk considerations, is presented in this section. For changes resulting in an increase in risk, the increase should be small and consistent with the intent of the Commission's policy statement (Reference 25) on safety goals for the operations of nuclear power plants. The licensee stated in Section 4.6.1 of Enclosure 1 of the LAR that the BFN Model of Record, Version 7, which represents the current as-built, as-operated plant and associated risk profile for internal events (with internal flooding) was utilized to perform the ILRT interval extension analysis. TVA also noted that the BFN fire probabilistic risk analysis (FPRA) that represents the plant, once all NFPA 805 modifications (including operator actions) are installed, was used to quantify and assess the impact of this application for internal fire risk. Key Principle 4 was evaluated using the risk-informed decisionmaking framework for TSs described in the Standard Review Plan (SRP) Chapter 16.1 (Reference 26), RG 1.200 (Reference 15), and RG 1.174 (Reference 17).

Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval to once in 15 years. The risk assessment was provided in Attachment 4 to the LAR (Reference 1). Additional information was provided by the licensee in responses to the NRC RAIs, dated February 5, 2018 (Reference 2); March 27, 2018 (Reference 3); and July 27, 2018 (Reference 4).

In Section 4.6.1, "Methodology," of the LAR, the licensee stated that the plant-specific risk assessment followed the guidance of NEI 94-01, Revision 3-A; the methodology described in EPRI Report 1018243; and the NRC regulatory guidance outlined in RG 1.174 on the use of PRA and risk insights in support of an LAR for changes to a plant's licensing basis. In addition, the methodology used for the Calvert Cliffs Nuclear Power Plant, which is approved by the NRC in the SE dated May 1, 2002 (Reference 27), to estimate the likelihood and risk implication of undetected corrosion-induced leakage of steel containment liners for the additional window of vulnerability from extending the ILRT interval was used by the licensee for the BFN units to estimate the conditional containment failure probability (CCFP) and its effect on the LERF and the estimated population dose.

Revision 2-A of NEI 94-01 (Reference 7) describes an approach for implementing the optional performance-based requirements of Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A test intervals to up to 15 years. In the NRC final SE for Revision 2 of NEI 94-01, dated June 25, 2008 (Reference 11), the NRC staff concluded that Revision 2-A of NEI 94-01 describes an acceptable approach for implementing the optional performance-based requirements of Option B and is acceptable for referencing by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to the specific limitations and conditions listed in Section 4.1 of the final SE for Revision 2 of NEI 94-01.

Revision 3-A of NEI 94-01 (Reference 6) provides the BFN operating license and guidance for extending Type C LLRT intervals beyond 60 months. This amendment also requests to extend the Type C test interval up to 75 months.

Revisions 2 and 3 of NEI 94-01 have been reviewed by the NRC staff and approved for use. The final SE for Revision 2 of NEI 94-01 documents the NRC staff's evaluation and acceptance of Revision 2 of NEI 94-01, subject to six specific limitations and conditions listed in Section 4.1 of the final SE dated June 25, 2008 (Reference 11). The final SE for Revision 3 of NEI 94-01, dated June 8, 2012 (Reference 23), includes two specific limitations and conditions listed in Section 4.0 of that SE for the Type C test. Revision 3-A of NEI 94-01 and Revision 2-A of NEI 94-01 include their corresponding SEs.

Section 9.2.3.1 of Revision 3-A (Reference 6) of NEI 94-01 states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4 of Revision 3-A of NEI 94-01 states that the assessment should be performed using the approach and methodology described in EPRI TR 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated August 2007 (Reference 28). The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.

In the final SE for Revision 2 of NEI 94-01 (Reference 11), the NRC staff found the methodologies in Revision 2-A of NEI 94-01 (Reference 7) and EPRI Final TR 1009325, Revision 2 (Reference 28), are acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied. The NRC staff set forth the following conditions related to referencing the methodology in EPRI Final TR 1009325, Revision 2:

1. The licensee submits documentation indicating that the technical adequacy of its PRA is consistent with the requirement of RG 1.200 relevant to the ILRT extension application.

2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small and consistent with the clarification provided in Section 3.2.4.6 of the final SE for Revision 2 of NEI 94-01.

3. The methodology in EPRI TR 1009325, Revision 2, is acceptable, provided the average leak rate for the preexisting containment large leak accident case (i.e., accident case 3b) used by licensees is assigned a value of 100 times the maximum L_a, instead of 35 L_a.

4. An LAR is required in instances where containment overpressure is relied upon for emergency core cooling system (ECCS) performance.

The licensee addressed in its LAR each of the applied conditions listed above. A summary of how each condition has been met is provided in the sections below.

Evaluation of PRA Acceptability

The first condition discussed above stipulates that the licensee submits documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

RG 1.200 describes one acceptable approach for determining whether the technical acceptability of a PRA is sufficient for use in regulatory decisionmaking for light-water reactors (Reference 15). The purpose of RG 1.200 is (a) to provide guidance to licensees for use in determining the technical acceptability of the base PRA used in a risk-informed regulatory activity and (b) to endorse industry standards and peer-review guidance.

In Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation" (Reference 21), the NRC clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200 (Reference 29) to assess technical adequacy of the PRA used to support risk-informed applications. Revision 2 of RG 1.200 will be used for all risk-informed applications received after March 2010. In Section 3.2.4.1 of the SE for EPRI TR 1009325, Revision 2, the NRC staff stated, in part, that:

> [l]icensee requests for a permanent extension of the ILRT surveillance interval to 15 years pursuant to NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, will be treated by NRC staff as risk-informed license amendment requests. Consistent with information provided to industry in Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," the NRC staff will expect the licensee's supporting Level 1 /Large Early Release Frequency (LERF) PRA to address the technical adequacy requirements of RG 1.200, Revision 1. Any identified deficiencies in addressing this standard shall be assessed further in order to determine any impacts on any proposed decreases to surveillance frequencies. If further revisions to RG 1.200 are issued which endorse additional standards, the NRC staff will evaluate any application referencing NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, to examine if it meets the PRA quality guidance per the RG 1.200 implementation schedule identified by the NRC staff.

In the same section of the NRC SE, the NRC staff stated that Capability Category I of the ASME/ANS PRA Standard shall be applied as the standard for assessing PRA quality for ILRT extension applications as approximate values of CDF and LERF, and their distribution among release categories is sufficient to support the evaluation of changes to ILRT frequencies.

The BFN internal events probabilistic risk analysis (IEPRA) model is an at-power, Level I and LERF PRA that includes internal events and internal floods. For seismic and high winds, the licensee provided results from a bounding analysis performed using insights gained from the Internal Plant Examination of the External Events (IPEEE) (Reference 30) and the IEPRA model to generate a scaled approach for approximate contribution to CDF and LERF used to assess the change in risk.

Internal Events and Internal Flooding

The PRA technical adequacy for BFN is also discussed in Section 4.6.2.3, "Consistency with Applicable PRA Standards," of Enclosure 1 of the LAR (Reference 24). The licensee's evaluation of the technical adequacy of its IEPRA model consisted of a full scope peer review performed in May 2009, an internal flooding focused-scope peer review performed in September 2009, and a focused-scope peer review of the IEPRA 2009 facts and observations (F&O) that was performed in July 2015. The peer reviews were performed using the NEI 05-04 process (Reference 31) and the ASME/ANS PRA Standard (Reference 16), as clarified by RG 1.200, Revision 2 (Reference 15).

In Section A-4.0 in Enclosure 2 of the LAR (Reference 1), the licensee stated that since the last peer review performed in July 2015, there have been no changes to the IEPRA model that would constitute an upgrade. Section A-5.0, "PRA Model History," in Enclosure 2 of the LAR provides a list of the corresponding revision history from Revision 0 through Revision 6 for all the BFN PRA models.

In Table 50 in Enclosure 2 of the LAR (Reference 1), the licensee reconciled each F&O by either providing a description of how the F&O was resolved or providing an assessment of the impact of resolution of the F&O on the results for the ILRT extension. The NRC staff evaluated each F&O and the licensee's disposition in Table 50 to determine whether the F&O had any significant impact for the application. In addition, the NRC staff reviewed information provided in the BFN RAI responses to a separate licensing action (Reference 32), the NRC's Record of Review (Reference 33), and the approved SER for the BFN transition to NFPA 805 (Reference 13) to support the NRC staff review of the FPRA acceptability for use in this LAR.

In letter dated, January 25, 2018 (Reference 34), the NRC staff requested additional information to assess the adequacy of some of the dispositions for the review. A summary of issues identified during the NRC staff's review of the F&Os and methods used in the IEPRA is provided below along with the associated resolution.

For Supporting Requirement (SR) Initiating Event (IE)-B4 associated with F&O 6-5, the peer review team identified that high pressure coolant injection (HPCI) steamline breaks are excluded as an initiator from the PRA. In addition, the pipe break frequencies used noncurrent EPRI and WASH-1400 data. Under previous staff review for a separate licensing action, the staff requested BFN provide additional information to justify the exclusion of HPCI steamline breaks or update the IEPRA model to include this initiator. In response to staff RAI 10.c to support a separate licensing action in a letter dated April 29, 2016 (Reference 32), the licensee stated the BFN PRA uses the frequency of 1.0E-04/reactor (reactor)-year for medium loss-of-coolant accident inside containment and that HPCI steamline breaks outside containment have been excluded as an initiator from the PRA based on analysis showing that they are insignificant contributors. The BFN PRA initiating events analysis was revised in March 2016 to incorporate new pipe rupture frequencies provided in EPRI TR 1021086, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments," dated April 2013 (Reference 35), and this revision also included initiating events from industry experience through September 2015. The licensee provided the calculated probability values for HPCI, reactor core isolation cooling, and reactor water cleanup line breaks outside containment to demonstrate that the individual cut sets are less than 1 percent, and the summation of all the line break contributions was less than 95 percent of the internal events hazard group results. The NRC staff finds that the results of the licensee's analysis for the HPCI, reactor core isolation cooling, and reactor water cleanup breaks outside containment are considered to be insignificant contributors to CDF and LERF as defined in the ASME/ANS RA-Sa-2009 PRA Standard, and therefore, exclusion of the initiators in the IEPRA is acceptable for the ILRT extension request.

For F&O 2-31 in Table 50, the licensee's resolution states, in part, the failure of two MOVs to close is less than 2 orders of magnitude lower than another failure that would fail the system in a similar manner. In response to staff RAI 10.a to support a separate licensing action, in a letter dated April 29, 2016 (Reference 32), the staff requested the licensee provide additional information to confirm that the plant-specific bayesian update of the generic motor-operated valve (MOV) failure rate could reduce the failure probabilities of the MOVs to less than 1.0E-05. In response to the RAI, the licensee provided discussion pertaining to the updated alpha factors used in the BFN model for CCF probability for a BWR residual heat removal motor-operated valve (MOV) to close. The licensee determined that as a result of using a bayesian updated failure probability of 1.7E-03 for MOV failure to close, the CCF estimate (1.36E-05) is slightly greater than 1.0E-05/demand for generic CCF MOV failure, but less than two orders of magnitude below the updated MOV failure to close probability. The licensee further stated the MOVs have been considered in the BFN interfacing systems loss-of-coolant accident analysis. The licensee determined that the interfacing systems loss-of-coolant accident scenarios involving residual heat removal paths were within the spectrum of internal event challenges discussed in SR IE-A2 and have been explicitly modeled in the PRA. The NRC staff finds that the licensee's incorporation of the two MOV failures into the IEPRA model is appropriate for the resolution of F&O 2-31, and therefore, acceptable for the ILRT extension request.

In NRC staff APLA RAI 03.a, dated January 25, 2018 (Reference 34), associated with F&O 1-17 related to SR Data Analysis (DA)-C6, the NRC staff requested that the licensee provide the results and clarification as they pertain to the analysis performed to assess not excluding PMT from the count of plant-specific demands on standby components. In response to the RAI (Reference 3), the licensee performed a sensitivity to assess the impact of including PMT demands on the CDF results. The sensitivity concluded that including the demands due to preventative maintenance results in an under-estimation of the failure probability. The licensee adjusted the number of demands to correct the under-estimation by removing 10 percent of the demands due to PMTs from each type code. The removal of 10 percent was assumed to be conservative in review of other industry plant data where recent analysis concluded that for the type codes evaluated, the fraction of total demands that were PMTs was less than 2.5 percent of the total demands for the type code. The licensee provided an updated Table 31 in its response for the CDF and LERF values for each unit. The NRC staff finds that the assumption of 10 percent removal demands due to PMTs is acceptable because it is greater than the industry fraction of the total demands for the associated type code, and therefore, appropriate for the ILRT application.

In APLA RAI 03.b (Reference 34) associated with F&O 4-18 related to SR Human Reliability (HR)-G2, the NRC staff requested that the licensee provide additional information pertaining to the data considered and how the execution error was excluded from some human failure events (HFE). In response to this RAI, the licensee provided discussion of the review performed of the five human error probabilities (HEP) identified in F&O 4-18. The licensee's review also included the human reliability analysis calculator to identify any HFEs that excluded execution errors in the calculated estimate for the HEP values used in the PRA model. In review of the human reliability analysis calculator, the licensee determined that three of the HEPs identified in F&O 4-18 were no longer applicable to the finding as a result of inclusion of the execution error in the HEP value computed, or the HEP was no longer modeled in the updated PRA. For the two HEPs associated with HFA_0024RCWINTAKE (failure to clear debris at the intake before reactor scam) and HFA_0_ADSINHIBIT (failure to inhibit automatic depressurization system (ADS) during anticipated transient without scram), the licensee provided a qualitative assessment of some human performance shaping factors that included consideration of location, event time and response, redundancy, etc. The licensee identified

HFA_0_ADSINHIBIT as a time critical action, and therefore, performed a sensitivity to confirm the time available for recovery. The sensitivity analysis used an execution error of commission with a probability value of 4.3E-04, then applied a scaled ration of the new HEP value to the baseline HEP value. The results of the licensee's sensitivity analysis conclude when including execution error into the HEP values, the contribution to ΔCDF and ΔLERF remains acceptable for the ILRT application. The NRC staff finds the licensee's assessment for determination of the exclusion of error into the HEP values resolves F&O 4-18 appropriately for this ILRT extension request.

In NRC staff APLA RAI 03.c, dated January 25, 2018 (Reference 34), associated with the F&O Internal Flood Scenario Development (IFSN)-A9-01, the staff requested that the licensee confirm that the use of a 0.1 screening factor did not screen out any scenarios that could potentially be non-negligible risk contributors for IEs. In response to this RAI (Reference 2), the licensee stated that the BFN PRA Computer Aided Fault Tree Analysis Model Revision 7 was resolved using a probability of 1.0 for the basic event, and the change in CDF and LERF was calculated. In comparison of the 0.1 screening value and the 1.0 failure probability used, the percent change for CDF was 0.03 percent to 0.17 percent respectively, and for LERF, it was 0.15 percent to 0.77 percent, respectively. In review of the licensee's response to RAI 03.c, the NRC staff finds that the change in contribution for the CDF and LERF is very small, and application of the screening value of 0.1 for the basic event IF-CB593-DOOR is reasonable for this ILRT application.

In APLA RAI 03.d (Reference 34) associated with F&O 6-8, the NRC staff requested that the licensee provide additional information as it pertains to the reactor coolant water (RCW) initiating event frequency and the reduction factor used. In response to this RAI (Reference 2), the licensee confirmed that the calculated factor identified in the 2015 peer review was incorrect and has been removed from the model of record. Subsequent to removal of the factor, TVA updated the model of record to explicitly include all potential combinations of a single RCW pump failure-to-run basic event combined with the conditional failure of at least two additional RCW running pumps. The failure-to-run basic events probabilities consider a 1 year exposure time. The NRC staff finds that upon update of the logic model for the failure-to-run of the RCW pumps in response to the staff RAI as described above, removal of the correction facture is appropriate for this ILRT request.

For F&O 2-50, the peer review team identified that for instances where cues for human actions involve multiple individual instrumentation devices, they are modeled in the PRA as multiple inputs to a single AND gate. The peer review team further concluded the licensee did not consider or confirm the development of operator guidance that would allow operators to discern which instrument is not impacted by the fire. In an NRC staff RAI (Reference 34), the staff requested the licensee provide additional information to confirm that the assumption and the risk metrics for the ILRT application continue to be met. The licensee performed a sensitivity analysis that included the probability associated with the human failure to choose the functional instrument. The licensee notes that fire safe shutdown procedures denote which instrumentation trains are impacted by a fire, and failure to choose instrument is included in the cognitive portion of the HRA action. For the specific proceduralized action, the licensee stated this was identified as an assumption that remains consistent with the fire safe shutdown procedures. The licensee provided further excerpts from the fire procedures that direct operators to use credited instrumentation in the event of a fire. The licensee stated that the sensitivity analysis results demonstrated that the probability of a human failure to choose the functioning instrument causes a negligible impact to the human failure probability. The NRC staff finds the incorporation of the operator action into the HEP appropriate for the ILRT application because the licensee describe its detailed evaluation of the safe shutdown procedure.

In an NRC staff RAI (Reference 34) associated with the F&O for SR IE-C8, the NRC staff requested the licensee provide a quantitative basis to support the conclusion that modeling of the CCF battery charges beyond the 24-hour mission time is over-conservative. In response to the RAI (Reference 2), the licensee performed an evaluation and included the intra-system CCF for the battery charges in the fault tree logic and resolved the model. Additionally, the licensee identified errors in the probability of battery charger A for the loss of both direct current busses initiator. The probability for the failure of battery charger A was updated with a 1-year mission time, thus increasing the probability from 1.22E-04 to 4.35E-02. Additionally, the licensee reviewed the CCF variables and updated throughout the fault tree logic to confirm the correct CCF probabilities were used. Upon inclusion of the CCF for the battery chargers and update of applicable CCF variables, the additional contribution to LERF was added to the total LERF values for all three units in response to APLA RAI 03.g (Reference 15). The NRC staff finds the resolution for the F&O associated with SR IE-C8 resolved because the CCF modelling of the battery charges has been reevaluated and updated in the IEPRA model.

In review of the LAR dated August 15, 2017, and responses to the RAIs, the NRC staff concludes that the issues identified in the peer review have been addressed, and therefore, the licensee's responses are acceptable. Accordingly, the NRC staff concludes that the licensee has demonstrated that the internal events PRA (IEPRA) meets the guidance in RG 1.200, Revision 2, and it is acceptable to support the requested ILRT extension. A more detailed review of the ILRT risk metrics is provided in Section 3.5 of this SE.

Internal Fire Hazards

The licensee evaluated the technical adequacy of the FPRA model by conducting peer reviews of the FPRA model using the NEI 07-12, Revision 1 process (Reference 36), and FPRA Part 4 of the ASME/ANS RA-Sa-2009 PRA Standard (Reference 16), as clarified by RG 1.200, Revision 2 (Reference 15). (These included a January 2012 full-scope peer review, June 2012 focused scope peer review, June 2015 focused-scope peer review, and the disposition of open F&Os as it pertains to the ILRT extension.)

In Table 51 in Enclosure 2 of the LAR, the licensee reconciled each fire F&O by either providing a description of how the F&O was resolved or providing an assessment of the impact of resolution by the F&O on the results for the ILRT extension. The NRC staff evaluated each F&O and the licensee's disposition in Table 51 to determine whether the F&O had any significant impact for the application. In addition, the NRC staff reviewed information provided in the BFN RAI responses to a separate licensing action (Reference 32), the NRC's Record of Review (Reference 33), and the approved SE for BFN transition to NFPA 805 (Reference 10) to evaluate the FPRA model for acceptability when determined to be applicable.

The NRC staff requested additional information to assess the adequacy of some of the dispositions of the FPRA F&Os. A summary of issues identified during the NRC staff's review of the F&Os and methods used in the FPRA is provided below, along with the associated resolution.

For FPRA F&Os 4-12, 4-21, 9-2, 2-38, 2-39, and 2-50 listed in Table 51 of the LAR, the licensee specified that the evaluation used the FPRA model that will represent BFN at the time this ILRT application is applied. Therefore, the HFEs that will be in place will no longer be a strategy employed by BFN for fire hazards. In an RAI (Reference 37), the NRC staff requested the licensee confirm that the HFEs that have been proposed to be modeled in the FPRA will be representative of the strategy employed (i.e., as built, as operated) upon completion of all NFPA 805 milestones. The licensee provided clarification in its APLA-RAI 03.f (Reference 3) response that states the FPRA credited actions have been developed to the extent possible to make the HRAs representative of those proposed actions due to incomplete fire procedures. Completion of the fire procedures and recovery actions modeled in the FRPA are included as implementation item 33 in the updated version of Table S-3 provided in Enclosure 1 of the letter dated October 23, 2017 (Reference 38), to the NRC.

The NRC staff approved the BFN NFPA 805 in the SE dated October 28, 2015 (Reference 24). The BFN NFPA 805 SE states, in part, "The licensee indicated that it updated the FPRA to apply a floor value of 1.0E-05 to all HEP combinations that do not include long-term decay heat removal HFEs for FPRA CDF or those HFEs cued and guided by severe accident mitigation guideline procedures for FPRA LERF." For the remaining combinations, the licensee stated that the FPRA applied a floor value of 1.0E-06, given a low dependency exists between long-term decay heat removal and severe accident mitigation guideline actions and other earlier actions. In its response, the licensee indicated that the revised floor values were incorporated in the integrated analysis.

In review of the total LERF and delta (Δ) LERF values provided in Table 34 of the BFN ILRT submittal, the NRC staff requested the licensee confirm that the justification provided above for resolution in the FPRA for NFPA 805 approval remains the same and unchanged for the ILRT application. In an NRC staff RAI (Reference 34) associated with SR HR-G7, the NRC staff requested the licensee identify any of the joint human error probabilities (JHEP) that use a value less than the floor value of 1.0E-06, perform a sensitivity applying a floor value of 1.0E-06 to those JHEPs, and provide justification for JHEP values where a lower value is determined to be acceptable. In Table 1 of its response to this RAI (Reference 3), the licensee provided the results of a sensitivity analysis performed in response to NFPA 805 PRA RAI 01.v and PRA RAI 24 that compared the FPRA CDF and LERF metrics from the baseline case that used multiple HRA floor values of 1.0E-05 and 1.0E-06 with a sensitivity analysis that applied only the floor value of 1.0E-05.

The licensee stated, "The increased contribution to CDF and LERF is primarily driven by JHEP combinations that have already been identified and included in the baseline results that use a 1.0E-07 value for the JHEP minimum floor value," and describes the contribution to the risk metrics as a function of the selected floor value. The NRC staff reviewed the results of the sensitivity that confirm application of a higher value for JHEPs does increase contribution to the risk metrics. NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines (Reference 39)," states, in part, "While it might be reasonable to adopt some sort of limit, it needs to be done carefully, so that the results of PRAs are not distorted by arbitrary assignments of probabilities." As discussed in further detail, any limiting values should be consistent within the context of the scenarios in which they are applied." The NRC staff finds the licensee's use of JHEP values, as approved in the NFPA 805 SE (Reference 24), and discussed in the licensee's response to an NRC staff RAI (Reference 3), remains consistent with NUREG-1921 and appropriate for use in the requested ILRT extension.

NRC RAI (Reference 34) pertains to the incorporation of recently approved NRC fire methodologies. The methodologies described in NUREG/CR-7150, Volume 3 (Reference 39), and NUREG/CR-2169 (Reference 40) are considered enhancements and/or improvements to existing FPRA methodologies. The BFN FPRA Revision 7 model incorporating these methodology changes has not been peer reviewed; however, the staff requested the licensee provide additional information to evaluate the significance of its use on the risk metrics for this application, consistent with the use of state-of-the-art methods and data as described in the 1995 Commissioner's PRA Policy statement, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Reference 25). In response to an RAI (Reference 3), the licensee confirmed that the insights provided in the letter to NEI dated June 21, 2012 (Reference 41), documenting the series of recent FPRA Methods Review Panel decisions on four methods used for analyzing fire risk contributions had been incorporated into Revision 7 of the BFN FPRA model, in addition to the guidance provided in NUREG/CR-7150 (Reference 39) and NUREG/CR-2169 (Reference 40).

The interim staff guidance provided in the letter dated November 23, 2009, for closure of NFPA 805 FAQ 08-0046, "Incipient Fire Detection Systems," has been rescinded, and NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities" (DELORES-VEWFIRE), provides the state-of-the-art methods, tools, and data for consideration of very early warning detection systems to be modeled in an FPRA. The licensee further stated in response to APLA RAI 02 (Reference 3) that TVA had not incorporated NUREG-2180 guidance into the FPRA model. Upon an NRC staff RAI request, the licensee performed a sensitivity to assess the impact of NUREG-2180 on the CDF and LERF values. The licensee identified three fire compartments that take credit for incipient detection in the BFN FPRA model for which the incorporation of NUREG-2180 would impact. In Tables 1, 2, and 3 of TVA response to the above RAI, the licensee provided the results of the sensitivity performed for each of the three fire compartment areas for Units 1, 2, and 3.

The sensitivity results (including both internal and external hazards) provided by TVA for total LERF for each unit, in its response dated March 27, 2018 (Reference 3), to an NRC staff RAI, slightly exceeded the RG 1.174 LERF risk metric (i.e., 1.0E-05 per reactor/year). The response discussed several PRA conservatisms and plant features that, if addressed in the PRA, were expected to reduce the risk estimates provided. In RAI 02-01.a (Reference 37), the NRC staff requested the licensee provide a summary and results (i.e., ΔLERF, CCFP, population dose rate, total LERF) that address the potential FPRA modeling refinements described in its response to APLA RAI 02. In response to this RAI dated July 27, 2018 (Reference 4), the licensee revised the FPRA model to incorporate the approved NRC methodology in NUREG-2180 and considered all other applicable changes discussed in responses to RAIs provided in its previous response. The NRC staff finds that the risk results met all the applicable risk metrics, and the changes to the PRA are consistent with the use of realistic results based on the as-built, as operated plant, contingent upon completion of the modifications included in the license condition provided in Section 7.0 of this SE.

In APLA RAI 02-02 (Reference 37), the NRC staff requested the licensee provide additional information to explain the asymmetrical results and/or potential anomalies across the three fire compartment areas for each unit. In response to this RAI dated July 27, 2018 (Reference 4), the licensee stated that the effect of fire on CDF is expected to vary from unit to unit, depending on the exact physical locations of equipment, instrumentation, power supplies, and connective cabling. The same is true for LERF and for the ratio of LERF to CDF, when comparing the effects between units. As it pertains to inter-unit differences, the licensee provided discussion to confirm the percentage changes for CDF and LERF when incorporating NUREG-2180 into the FPRA were a result of the impacts on specific components and systems in the scenario results. In review of the licensee's response to RAI 02-02, the NRC staff finds the licensee's justification of the results for the inter-unit differences is reasonable.

The NRC staff reviewed the disposition of F&Os, combined with the BFN-approved SE for transition to NFPA 805, and the supplemental information received in letters dated February 5, 2018 (Reference 2); March 27, 2018 (Reference 3); and July 27, 2018 (Reference 4). Based on the information provided and discussed above, the NRC staff finds the BFN FPRA is consistent with RG 1.200, Revision 2, and the BFN FPRA model is acceptable for use in performing the risk assessment for the ILRT extension requested because fire risk is clearly dependent on the spatial configuration of the structures, systems, and components. A more detailed review of the ILRT risk metrics is provided in Section 3.5 of this SE.

Other External Events

In Section 3.2.4.2 of the final SE (Reference 11) for Revision 2 of NEI 94-01, the NRC staff states that:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: '[w]here possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., seismic) in the risk impact assessment for extended ILRT intervals." This section also states that: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [i.e., EPRI Report No. 1009325, Revision 2], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed." This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

In Section 10.0 in Enclosure 2 of the LAR (Reference 1), the licensee stated it performed an analysis of the external events contribution to risk and assessed the impact on the ILRT extension application. The licensee stated that the BFN IPEEE considered seismic events, external flood, high winds (including tornadoes), transportation and nearby facility hazards, and other plant-unique hazards.

Seismic Events

Regarding seismic risk estimates, the licensee stated the IPEEE seismic evaluation does not estimate seismic CDF and LERF results. The NRC staff considered results of the NRC study published in "Results of Safety/Risk Assessment of Generic Issue (GI) 199, Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants" (Reference 42). The GI 199 analysis estimated a CDF of 3.70E-06/reactor-year, 5.4E-06/reactor-year, and 5.4E-06/reactor-year for Units 1, 2, and 3, respectively. In review of Table D-1 of GI 199 NRC study (Reference 43), the staff finds that the seismic values used are acceptable for use in this risk evaluation.

In response to APLA RAI 04.d (Reference 2), the licensee confirmed that the fire and seismic rows in Table 31 of the LAR had been transposed incorrectly and confirmed the above values reported in the LAR are used in the ILRT risk estimates. In its response to RAI 01.b (Reference 3), the licensee further provided updated seismic LERF values of 1.10E-06/year, 1.70E-06/year, and 1.66E-06/year for BFN, Units 1, 2, and, 3, respectively. As supplemented in response to RAI 02-01 (Reference 4), the licensee stated that the final seismic LERF values are 1.07E-06/reactor-year, 1.70E-06/reactor-year, and 1.65E-06/reactor-year for BFN, Units 1, 2, and 3, respectively.

The NRC staff noted in APLA RAI 01.b (Reference 34) to the licensee that, since external events tend to subject plants to CCFs of multiple structures, systems, and components, it is possible that the LERF to CDF ratio for external events to be higher than for internal events. In response to APLA RAI 01.b (Reference 3), the licensee agreed that CCFs could result in a greater LERF to CDF ratio for seismic. The licensee provided two reference plants that are BWR-4 units of similarity to BFN that have a seismic PRA model to assess the effect of increasing the LERF to CDF ratio. The licensee identified that the seismic PRA LERF to CDF ratios for the reference plants was 40.6 percent and 63.6 percent. The larger of the two values was used for the BFN seismic risk assessment.

The NRC staff finds that using the most limiting reference plant, and therefore, increasing the LERF to CDF ratio for seismic contribution by 63.6 percent, is acceptable for this ILRT application. With this limiting value, the total LERF values for all three BFN units remained below the acceptance criteria for total LERF less than 1.0E-05/year. The NRC staff confirmed the licensee propagated the new seismic values into the other ILRT risk metrics (i.e., population dose rate, ΔLERF, and CCFP), and the licensee provided updated tables in response to APLA RAI 02-02 (Reference 4). The impact of the updated seismic contribution to LERF estimates on the risk results used to support the application for extending the Type A test interval is provided in Section 3.5 of this SE. The staff finds the licensee's evaluation acceptable because the licensee has used the most conservative available information to include seismic risk.

High Winds Events

In response to GL 88-20 for IPEEE (Reference 44), BFN performed a bounding analysis for high winds. The NRC staff concluded in its technical evaluation report (Reference 45) that the estimate provided in the licensee's IPEEE submittal was less than the NUREG/CR-4461, Revision 1 estimate; however, the estimate for high winds contribution remains below the NUREG-1407 screening criteria of 1.0E-06/reactor-year. For the BFN ILRT risk evaluation, the licensee assumed that the high winds contribution for all three units is 1.0E-06/reactor-year. In response to APLA RAI 01.b (Reference 3), the licensee stated that the seismic LERF to CDF ratio determined from the limiting reference plant was applied conservatively to assess the contribution for high winds. Furthermore, in response to APLA RAI 01.b (Reference 3), the licensee provided updated conservative LERF values of 2.98E-07/reactor-year, 3.14E-07/reactor-year, and 3.08E-07/reactor-year for Units 1, 2, and 3, respectively, for high winds. In RAI 02-01 (Reference 37), the NRC staff requested the licensee incorporate recently approved fire methodology, which caused the total LERF values to exceed the RG 1.174 risk metrics criteria. In response to RAI 02-01 (Reference 4), the licensee subsequently removed the conservatism with respect to the [h]igh [w]inds LERF contribution. The licensee stated that, "In the response to APLA RAI 02, the [s]eismic CDF [to] LERF ratio was used, which is over conservative, because [s]eismic events can reasonably be assumed to cause CCFs that would not be expected from a [h]igh [w]inds initiator. The weather related-LOOP [loss of offsite power] event, compared to other LOOP events (e.g., plant centered, switchyard), includes relatively long recovery times due to the expected extensive repair work that would be required to restore power to the plant." The licensee used the values from the quantified internal events weather-related LOOP event.

The NRC staff finds that removal of the seismic CDF to LERF ratio and use of the quantified internal events weather-related LOOP event for accessing the high winds contribution to LERF is acceptable to use for the ILRT risk application because it more appropriately reflects the expected accident scenarios.

The licensee provided updated risk values for the high winds contribution to LERF using the quantified internal events weather-related LOOP event. The contribution to LERF values provided in response to RAI 02-01 (Reference 4) is 9.46E-08/reactor-year, 7.82E-08/reactor-year, and 1.49E-07/reactor-year for BFN, Units 1, 2, and 3, respectively. The impact of the updated high wind contribution to LERF estimates on the risk results used to support the application for extending the Type A test interval is provided in Section 3.5 of this SE.

Summary of PRA Acceptability

In summary, the licensee has evaluated its IEPRA and FPRA against the ASME/ANS RA-Sa-2009 PRA Standard and Revision 2 of RG 1.200 (Reference 15), addressed or evaluated the impact of the findings developed during the peer reviews for the PRAs for applicability to the ILRT interval extension, and included a quantitative assessment of the contribution for other external events. The staff finds the licensee's risk estimates included risk from all hazards and is, therefore, acceptable for the ILRT application.

Estimated Risk Increase

The second condition identified in Section 3.2.2 of this SE states that the licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small and is consistent with the guidance in RG 1.174 (Reference 17) and the clarification provided in Section 3.2.4.5 of the final SE for Revision 2 of NEI 94-01 (Reference 11). Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-roentgen equivalent man (rem) per year, or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in CCFP should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points. Additionally, for plants that rely on containment overpressure for net positive suction for ECCS injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174, Revision 3. The associated risk metrics include LERF, population dose, and CCFP, but not CDF because BFN does not rely on containment overpressure for ECCS performance.

The licensee provided a plant-specific risk assessment in Enclosure 2 to the LAR (Reference 1). The metrics of concern for the ILRT extension specifically include ΔLERF, ΔDose, and CCFP, as discussed above. In letters dated February 5, 2018 (Reference 2); March 27, 2018 (Reference 3); and July 27, 2018 (Reference 4), the licensee provided revisions to the ILRT risk metric values reported.

In Section 8.2.3 of the licensee's LAR for containment allowable leakage factor, the licensee states that Peach Bottom Atomic Power Station (Peach Bottom) has an allowable leakage rate of 0.5 percent/day and BFN has a 2.0 percent/day allowable leakage rate. Therefore, consistent with the EPRI Final TR 1009325 (Reference 28), BFN has a factor of 4.0 greater than the reference plant. As noted in the Updated Final Safety Analysis Report for Peach Bottom, and the BFN Updated Final Safety Analysis Report for the dry well free volume and pressure suppression free volume for the two plants, the NRC staff confirmed the containments were of similar and comparative specification, and therefore, the EPRI approach for ratio was determined to be valid. The NRC staff finds that the licensee's use of its plant-specific CDF distribution across the EPRI accident classes is consistent with the EPRI TR 1009325 report for performing the ILRT risk analysis, and therefore, acceptable for the ILRT extension request.

Throughout the submittal, inconsistencies in values were identified by the NRC staff across the tables, and some editorial context was incomplete. In APLA RAI 04 (Reference 34), the NRC staff requested the licensee address (correct) each of the identified inconsistencies in APLA RAI 04.a – h. In response to APLA RAI 04.a-h (Reference 2), the licensee agreed that the values provided were inconsistent, verified the corrected values, and corrected the miscellaneous editorial context identified. The NRC staff performed review of the updated tables provided in the licensee's response to APLA RAI 02 (Reference 3) to confirm all the inconsistencies had been corrected and the values were appropriately updated. In further response to RAI 02-01.a (Reference 4), the licensee provided updated tables and values that superseded the information corrected in the licensee's responses to APLA RAI 04.a-h. A more detailed review of the sensitivity analyses performed by the licensee that updated the ILRT risk values (i.e., increase in LERF, total LERF, population dose rate, and CCFP) is discussed in Section 3.2.3.3 of this SE.

The reported risk impacts are based on a change in test frequency from three tests in 10 years (the test frequency under 10 CFR Part 50, Appendix J, Option A) to one test in 15 years. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A ILRT frequency:

1. The reported increase in LERF values for internal events as updated in Table 33 RAI 02-01 (Reference 4) is 3.31E-08/year, 3.23E-08/year, and 2.99E-08/year for Units 1, 2, and 3, respectively. Including the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval increases these values to 4.64E-08/year, 4.53E-08/year, and 4.20E-08/year for Units 1, 2, and 3, respectively. As updated in response to APLA RAI 02-02 in Table 34 (2180) (Reference 4), the total LERF for combined internal and external events is 9.60E-06/year, 9.75E-06/year, and 9.13E-06/year for Units 1, 2, and 3, respectively. The risk contribution from external events includes the effects of seismic events, high winds, floods, and other external hazards, as discussed in Section 3.2.3.4 of this SE. These changes in risk are considered to be "small" per the acceptance guidelines in RG 1.174, Revision 3 (Reference 17).

2. The reported change in Type A ILRT frequency from three in 10 years to once in 15 years, as updated in Table 33-RAI 02-01 (Reference 4), results in a reported increase in the total population dose of 3.47E-01 person-rem/year, 3.76E-01 person-rem/year, and 2.62E-01 person-rem/year for Units 1, 2, and 3, respectively, which included the increases in Classes 3a and 3b. The reported increase in total population dose is below the values acceptance guidelines in Section 3.2.4.6 of the final SE for Revision 2 of NEI 94-01. Thus, this increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

3. The increase in CCFP due to change in test frequency from three in 10 years to once in 15 years, as updated in Table 33-RAI 02-01, is approximately 1.08 percent, 1.16 percent, and 0.87 percent for Units 1, 2, and 3, respectively. These values are

below the acceptance guidelines in Section 3.2.4.6 of the final SE for Revision 2 of NEI 94-01.

Based on the risk assessment results discussed above, the NRC staff concludes that the estimated risk increase associated with permanently extending the Type A ILRT interval to once in 15 years is small and is consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.6 of the final SE for Revision 2 of NEI 94-01. The defense-in-depth philosophy is maintained, as the independence of barriers will not be degraded as a result of the requested change, and the use of the three quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, the second condition, as discussed in Section 2.1 above and Key Principle 4 in RG 1.174, Revision 3, are met.

Leak Rate for the Large Preexisting Containment Leak Rate Case

The third condition, as discussed in Section 2.1 above, stipulates that in order to make the methodology in EPRI TR 1009325, Revision 2-A, acceptable, the average leak rate for the preexisting containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a. As noted by the licensee in the table in Section 4.6.1 of Enclosure 1 to the LAR, the methodology in EPRI TR 1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the preexisting containment large leakage rate accident case (accident case 3b), and this value has been used in the BFN specific risk assessment. Accordingly, the NRC staff finds that the third condition is met.

Applicability if Containment Overpressure is Credited for ECCS Performance

The fourth condition stipulates that in instances where containment overpressure is relied upon for ECCS performance, an LAR is required to be submitted. In the table in Section 4.6.1 of Enclosure 1 to the LAR, the licensee stated that containment overpressure is not relied upon for ECCS performance for BFN. Accordingly, as discussed in Section 2.1 above, NRC staff finds that the fourth condition is met.

3.2.3 Key Principle 5: Monitor the Impact of the Proposed Change

The NRC staff SEs incorporated into Revision 3-A of NEI 94-01 (Reference 6) and Revision 2-A of NEI 94-01 (Reference 7) contain a section entitled, "The Impact of the Proposed Change Should be Monitored Using Performance Measurements Strategies," which states:

> In addition to maintaining the defense-in-depth philosophy as described in Section 3.2.2 of this SE, the applicants for TS amendments will continue to perform containment inspections during the Type A test interval as discussed in Sections 3.1.3 and 3.1.4 of this SE.
>
> As documented in NUREG-1493, industry experience has shown that most ILRT failures result from leakage that is detectable by local leakage rate testing (Type B and Type C testing). Specific testing frequencies for the local leak rate tests are reviewed prior to every refueling outage (18-month cycle). An outage scope document is issued to document the local leak rate test periodically and to ensure that all pre-maintenance and post-maintenance testing is complete. The post-outage report provides a written record of the extended testing interval changes and the reasons for the changes based on testing results and

maintenance history. Based on the above measures, the LLRT program will provide continuing assurance that the most likely sources of leakage will be identified and repaired.

ANSI/ANS-56.8-2002, Section 6.4.4, also specifies surveillance acceptance criteria for Type Band Type C tests and states that: "The combined [as-found] leakage rate of all Type B and Type C tests shall be less than 0.6 La when evaluated on a minimum pathway leakage rate basis, at all times when containment operability is required." It states, moreover, that: "The combined leakage rate for all penetrations subject to Type B and Type C test shall be less than or equal to 0.6 La as determined on an maximum pathway leakage rate basis from the as-left LLRT results." These combined leakage rate determinations shall be done with the latest leakage rate test data available, and shall be kept as a running summation of the leakage rates.

The containment components' monitoring and maintenance activities will be conducted according to the requirements of 10 CFR Part 50, Appendix J, and 10 CFR 50.55a.

The above provisions are considered to be acceptable performance monitoring strategies for assuring that the risk of the proposed change will remain small. Accordingly, as discussed in Section 3.2.1 above, the NRC staff finds that Key Principle 5 is met.

3.2.4 Conclusion of Risk Evaluation

Consistent with the guidance in NEI 94 01, Revision 3 A, and the conditions and limitations of NEI 94 01, Revision 2 A, the licensee performed a risk evaluation to support the proposed extension of the BFN current performance-based Type A test interval to no longer than 15 years. The licensee provided the results of the risk evaluation in the LAR dated August 15, 2017, and provided supplemental response to staff RAIs in letters dated February 5, 2018; March 27, 2018; and July 27, 2018 (References 1, 2, 3, and 4, respectively).

Based on the NRC staff review of the LAR dated August 15, 2017 (Reference 1), supplemental information provided in the RAI responses (References 2, 3, and 4), and the regulatory and technical evaluations above, the NRC staff finds there is reasonable assurance that the licensee's PRA is acceptable. The NRC staff also finds that the licensee adequately performed the risk evaluation consistent with the guidelines provided in NEI 94-01, Revision 3-A (Reference 6); the corresponding methodology provided in EPRI Report 1009325, Revision 2-A (Reference 28); and the guidance of RG 1.174, Revision 3 (Reference 17). Therefore, the NRC staff finds that the results of the risk evaluation provided by the licensee demonstrate the risk metrics delineated in EPRI Report 1009325 for ILRTs and RG 1.174 are met.

The NRC staff concludes the risk results provided by the licensee for the proposed extension of the Type A ILRT interval beyond 10 years, considering completion of the following license condition, is acceptable for BFN to:

- Revise TS 5.5.12, "Primary Containment Leakage Rate Testing Program," to adopt NEI 94 01, Revision 3-A, and the conditions and limitations specified in NEI 94 01, Revision 2 A, as the 10 CFR Part 50, Appendix J, Option B, implementation documents.

- Extend on a permanent basis the Type A test interval up to 15 years.

4.0 LICENSE CONDITION

In the LAR, TVA specified that the risk assessment includes external event analyses, which assume all modifications are implemented in support of transition to NFPA 805. Furthermore, the licensee acknowledged that the license amendments have been requested such that implementation will occur prior to the BFN, Unit 2 startup following the spring 2019 refueling outage, thus allowing the completion of all required NFPA 805 modifications that coincide with the next scheduled ILRT for Unit 2. In APLA RAI 06.b.2 (Reference 34), the NRC staff requested TVA propose a license condition assuring that the requested ILRT extensions for all modifications that are credited in the as-built, as-operated FPRA model listed in Table S-2 of the TVA letters dated September 8, 2015, and October 20, 2015, will be completed immediately following the first outage of occurrence for the next scheduled ILRT for each unit. Completion of the modifications and the list of implementation items confirm the FPRA model to represent the as-built, as-operated configuration of the plant that the NRC staff reviewed in Section 3.2 of this SE for issuance of the license amendments to extend the Type A test intervals for up to 15 years.

In response to APLA RAI 06.b (Reference 2), the licensee proposed the following license condition:

> Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

The NRC staff found this license condition acceptable since it provides assurance that modifications associated with complete transition to NFPA 805 will be implemented prior to extending the frequency of ILRT at BFN, Units 1, 2, and 3.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendments on September 12, 2018. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on November 21, 2017 (82 FR 55415). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1 Letter from TVA to U.S. NRC, "Application to Revise Technical Specification 5.5.12 Primary Containment Leakage Rate Testing Program (BFN TS-497)," dated August 15, 2017 (ADAMS Accession No. ML17228A490).

2 Letter from TVA to U.S. NRC, "TVA Response to NRC Request for Additional Information Related to BFN Application to Revise Technical Specification 5.5.12 'Primary Containment Leakage Rate Testing Program'," dated February 5, 2018 (ADAMS Accession No. ML18036A901).

3 Letter from TVA to U.S. NRC, "TVA Response to NRC Request for Additional Information (Set 2) Related to BFN Application to Revise Technical Specification 5.5.12 'Primary Containment Leakage Rate Testing Program' (BFN-TS-497)," dated March 27, 2018 (ADAMS Accession No. ML18087A426).

4 Letter from TVA to U.S. NRC, "TVA Response to NRC Request for Additional Information (Round 2) Related to BFN Application to Revise Technical Specification 5.5.12 'Primary Containment Leakage Rate Testing Program' (BFN-TS-497)," dated July 27, 2018 (ADAMS Accession No. ML18208A587).

5 NRC Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995 (ADAMS Accession No. ML003740058).

6 NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July, 2012 (ADAMS Accession No. ML12221A202).

7 NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated November 19, 2008 (ADAMS Accession No. ML100620847).

8 Letter from NRC to TVA, "Issuance of Technical Specification Amendment for the Browns Ferry Nuclear Plant, Units 1, 2, and 3," dated February 22, 1996 (ADAMS Accession No. ML020040080).

9 NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 21, 1995 (ADAMS Accession No. ML11327A025).

10 Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 2 And 3 — Issuance of Amendments Regarding One-Time Frequency Extension For Containment Integrated Leakage Rate Test Interval," dated March 9, 2005 (ADAMS Accession No. ML050340624).

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12 NRC SE to NEI, "Final Safety Evaluation of Nuclear Energy Institute (NEI) Report, 94-01, Revision 3, Industry Guideline for Implementing Performance-Base Option of 10 CFR Part 50, Appendix J," dated June 8, 2012 (ADAMS Accession No. ML121030286).

13 Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Issuance of Amendments Regarding Extended Power Uprate," dated August 14, 2017 (ADAMS Accession No. ML17032A120).

14 Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 — Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term," dated September 27, 2004 (ADAMS Accession No. ML042730028).

15 NRC Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated March 2009 (ADAMS Accession No. ML090410014).

16 ASME/ANS RA-S-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated February 2009.

17 NRC Regulatory Guide 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated January 2018 (ADAMS Accession No. ML17317A256).

18 ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements," Reaffirmed August 9, 2011.

19 ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements," dated August 4, 1994.

20 NEI Topical Report 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J and EPRI Report No. 1009325, Revision 2, August 2007, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Interval," dated August 31 2007 (ADAMS Accession No. ML072970206).

21 NRC Regulatory Issue Summary 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50," dated December 8, 2008 (ADAMS Accession No. ML080020394).

22 EPRI Report 1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Interval, Revision 2-A of EPRI 1009325," dated October 2008.

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24 Letter from NRC to TVA, "Browns Ferry Nuclear Power Plant, Units 1, 2, and 3 – Issuance of Amendments Regarding Transition to a Risk-Informed Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c)," dated October 28, 2015 (ADAMS Accession No. ML15212A796).

25 Commission Policy Statement, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," dated August 16, 1995 (60 FR 42622).

26 NUREG 0800, Revision 1 - Standard Review Plan, Section 16.1, "Risk-informed Decision Making: Technical Specifications," dated March 2007 (ADAMS Accession No. ML070380228).

27 Letter from NRC to CCNP, "Calvert Cliffs Nuclear Power Plant, Unit No. 1: One-Time Amendment Extension of Appendix J, Type A, Integrated Leak Rate Test Interval and Exception from Performing a Post-Modification Type A Test," dated May 1, 2002 (ADAMS Accession No. ML021080753).

28 EPRI Report 1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 2, dated August 2007 (ADAMS Accession No. ML072970208).

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31 NEI 05-04, Revision 2 , "Process for Performing Internal Events PRA Peer Reviews Using ASME/ANS PRA Standard," dated November 2008 (ADAMS Accession No. ML083430462).

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33 NRC Dispositions to TVA, "Record of Review Dispositions to Browns Ferry Nuclear Plant, Units 1, 2, and 3 Internal Events PRA and Fire PRA Facts and Observations," dated July 9, 2015 (ADAMS Accession No. ML15190A2342).

34 Letter from NRC to TVA, "Browns Ferry Nuclear Power Plant, Units 1, 2, and 3 – Request for Additional Information Related to License Amendment Request to Revise Technical Specification 5.5.12 "Primary Containment Leakage Rate Testing Program," dated January 25, 2018 (ADAMS Accession No. ML18010B055).

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37 Email from NRC to TVA, "Browns Ferry Nuclear Power Plant, Units 1, 2, and 3 – Second Round RAI Related to LAR to Revise Technical Specification 5.5.12 "Primary Containment Leakage Rate Testing Program," dated June 15, 2018 (ADAMS Accession No. ML18169A012).

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Date: September 27, 2018

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – ISSUANCE OF AMENDMENT NOS. 305, 328, AND 288 TO REVISE TECHNICAL SPECIFICATION 5.5.12, "PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (CAC NOS. MG0113, MG0114, AND MG0115; EPID L-2017-LLA-0292) DATED SEPTEMBER 27, 2018

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