Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

July 08, 2004

TVA-BFN-TS-448

10 CFR 50.90

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-260
Tennessee Valley Authority	)	50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 2 AND 3 - TECHNICAL SPECIFICATIONS (TS) CHANGE 448 - ONE-TIME FREQUENCY EXTENSION FOR CONTAINMENT INTEGRATED LEAKAGE RATE TEST (ILRT) INTERVAL

Pursuant to 10 CFR 50.90, the Tennessee Valley Authority (TVA) is submitting a request for a TS change (TS-448) to licenses DPR-52 and DPR-68 for BFN Units 2 and 3, respectively. The proposed amendment revises TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A (ILRT) test to no later than November 6, 2009, for Unit 2, and no later than October 10, 2013, for Unit 3. TS Section 5.5.12 provides the requirements for the Primary Containment Leakage Rate Testing Program. TS Section 5.5.12 requires that this program establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. Additionally, the testing is required to conform to the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

This proposed TS revision takes a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests required by NEI 94-01, Revision 0, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J." The one-time exception is to NEI 94-01 guidelines to perform an ILRT at a frequency of up to 10 years, with allowance for a 15 month extension. The requested exception is to allow ILRT testing within 15 years from the performance of the last ILRT. This application represents a cost-beneficial licensing change. The ILRT imposes significant expense on the station, while the safety benefit of performing the test within 10 years, versus 15 years, is minimal.

A plant-specific, risk-based assessment has been performed in support of the one-time deferral of the Type A test frequency from once in 10 years to once in 15 years. The assessment demonstrates that a change in the Type A test interval from 10 years to 15 years represents a very small impact on risk, as defined by NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, dated November 2002.

Enclosure 1 provides a description and assessment of the proposed TS change. Enclosure 2 provides the existing Unit 2 and 3 TS pages marked-up to show the proposed changes. There are no associated TS Bases changes. Enclosure 3 provides the risk assessment supporting the proposed change.

TVA is asking that this TS change be approved by February 8, 2005, and that the implementation of the revised TS be made within 30 days of NRC approval to support the Spring 2005 Unit 2 refueling outage during which the next ILRT would be required without approval of the TS change.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS change qualifies for a categorical exclusion from environmental review pursuant to the provisions of

10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and Enclosures to the Alabama State Department of Public Health.

There are no regulatory commitments associated with this submittal. If you have any questions about this TS change, please contact me at (256)729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 08, 2004.

Sincerely,

T. E. Abney
Manager of Licensing
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Enclosures:
1. Description and Assessment
2. Proposed Technical Specifications Changes (mark-up)
3. Risk Assessment

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Enclosure 1

Browns Ferry Nuclear Plant (BFN)
Units 2 and 3

Technical Specifications (TS) Change 448

One-Time Frequency Extension for Containment Integrated Leakage Rate Test (ILRT) Interval

Description and Assessment

1.0 DESCRIPTION

This letter is a request to amend Operating Licenses DPR-52 and DPR-68 for Browns Ferry Nuclear Plant (BFN) Units 2 and 3, respectively. The proposed amendment revises TS Section 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A (ILRT) test to no later than November 6, 2009, for Unit 2, and no later than October 10, 2013, for Unit 3. TS Section 5.5.12 provides the requirements for the Primary Containment Leakage Rate Testing Program.

2.0 PROPOSED CHANGE

The proposed change involves a one-time exception to the 10 year frequency of the performance-based leakage rate testing program for Type A (ILRT) tests as prescribed by Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0 (Reference 1). The proposed change revises TS Section 5.5.12 of the BFN Units 2 and 3 TS to add the following statements, respectively:

, 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.

* 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.

Mark-up copies of the proposed TS change are in Enclosure 2.

3.0 BACKGROUND

BFN Units 2 and 3 are General Electric BWR/4 plants with Mark I primary containments. The Mark I type primary containment consists of a drywell, which encloses the reactor vessel; reactor coolant recirculation system and branch lines of the reactor coolant system; a toroidal shaped pressure suppression chamber (torus) containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber. The primary containment is penetrated by personnel and equipment access hatches, and piping and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests, and the overall leak-tight integrity of the primary containment is verified periodically by a Type A test (ILRT) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure.

Revisions to 10 CFR 50, Appendix J (i.e., Option B), allow individual plants to extend the Type A ILRT surveillance testing frequency from three-in-ten years to at least once per 10 years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage is less than the maximum allowable primary containment leakage rate. The basis for the current 10 year test interval is provided in Section 11.0 of NEI 94-01. This document was established in 1995 during development of the performance-based Option B program. Section 11.0 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 2), provides the technical basis that was used to support rulemaking to revise leakage rate testing requirements contained in Option B of 10 CFR 50, Appendix J.

The basis consisted of qualitative and quantitative assessments of the risk impact, in terms of increased public dose, associated with a range of extended leakage rate test intervals. To supplement NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electrical Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals" (Reference 3).

Option B of 10 CFR 50, Appendix J, requires that a Type A test be conducted at a periodic interval based on historical performance of the overall primary containment system. BFN TS Section 5.5.12 provides the requirements for the Primary Containment Leakage Rate Testing Program. TS Section 5.5.12 requires that this program establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. Additionally, the testing is required to conform to the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (Reference 4). Reference 4 endorses, with certain exceptions, NEI 94-01.

NEI 94-01 specifies for Type A tests, an initial test interval of 48 months, and allows an extension of the interval to 10 years based on two consecutive successful tests. BFN Units 2 and 3 are both currently on 10 year testing intervals. This proposed TS change adds an exception to TS Section 5.5.12 to allow a one-time deferral from the guidelines contained in RG 1.163 and NEI 94-01 regarding the Type A test interval, which will extend the next Type A test for Units 2 and 3 to a 15-year interval. This change is justified based on a combination of a successful containment leak rate test history, the BFN containment inspection program, containment operating performance, and a risk-informed assessment of the extended test interval.

TVA is asking that this TS change be approved by February 8, 2005, and that the implementation of the revised TS be made within 30 days of NRC approval to support the Spring 2005 Unit 2 refueling outage during which the next ILRT would be required without approval of the TS change.

4.0 TECHNICAL ANALYSIS

4.1 10 CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the primary containment, including systems and components that penetrate the primary containment, does not exceed allowable leakage rate values specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumptions in the safety analyses are not exceeded. The limitation of primary containment leakage provides assurance that the primary containment would perform its design function following an accident, up to and including the design basis accident.

Adoption of the Option B performance-based primary containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed or its acceptance criteria; however, it did alter the frequency at which Type A, B, and C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, test frequency is based upon an evaluation that reviews as-found and as-left leakage history to determine the frequency for leakage testing, which provides assurance that leakage limits will be maintained.

The extended frequency interval for testing allowed by NEI 94-01 is based upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program." NUREG-1493 made the following observations with regard to extending the test frequency:

- Reducing the Type A (ILRT) testing frequency to one per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above the existing requirements. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing has minimal impact on public risk.

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

Exceptions to the requirements of RG 1.163, are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, implementation, which states, "The regulatory guide or other implementing document used by a licensee, or applicant for an operating license, to develop a performance based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide." Since exceptions meeting the stated requirements are permitted, TS amendment applications satisfying these requirements do not require an exemption to Option B.

4.2 BFN Units 2 and 3 ILRT History

NEI 94-01 requires that Type A testing be performed at least once per 10 years based upon an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart or refueling cycles where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3.

Type A testing is performed to verify the integrity of the containment structure in its loss-of-coolant accident configuration. Industry test experience has demonstrated that Type B and C testing detects a large percentage of containment leakages and that the percentage of containment leakages detected only by integrated containment leakage testing is very small. Results of the previous Type A tests for each unit, presented below, demonstrate the BFN Units 2 and 3 containment structures remain essentially leak-tight barriers and represent minimal risk to increased leakage. These plant specific results support the conclusions of NUREG-1493.

Unit	Test Date	Total Leakage*	Fraction of La
2	11/06/1994	0.35001	0.1750
2	03/17/1991	0.25074	0.1254
3	10/10/1998	0.29640	0.1482
3	11/06/1995	0.92275	0.4614

*Leakage rates are expressed in units of percent containment air weight per day. The maximum allowable primary containment leakage rate, La, is 2% of primary containment air weight per day. TS leakage rate acceptance criteria for a Type A test for unit startup is $0.75L_a$ (i.e., 1.5% containment air weight per day), as discussed in TS Section 5.5.12. Calculated results are expressed at a 95% confidence level.

4.3 BFN Type B and C Testing

Type B and C testing assures containment penetrations such as flanges, sealing mechanisms, and containment isolation valves are essentially leak-tight. Type B and C tests identify the vast majority of all potential leakage paths. The Type B and C testing requirements will not be changed as a result of the proposed extended ILRT interval.

4.4 BFN Containment Inspections

Appendix J Visual Inspections

The Appendix J program requires visual inspections to be performed of accessible interior and exterior surfaces of the containment system for structural problems that may affect either the containment structural leakage integrity or performance of the Type A test. These examinations are conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test, based on a 10-year Type A test frequency in order to allow for early uncovering of evidence of structural deterioration. The examination of the primary containment structure shall be conducted in accordance with the schedule requirements of RG 1.163. Additionally, the plant instruction which implements this examination reflects a required frequency in accordance with Subsection IWE of American Society of Mechanical Engineers (ASME) Section XI, and

10 CFR 50.55a(b)(2)(ix)(E), such that one examination is scheduled during each inspection period. Acceptance criteria shall be in accordance with 10 CFR 50, Appendix J, and Article IWE 3000 of ASME Section XI. These requirements will not be changed as a result of the proposed extended ILRT interval.

Containment Inservice Inspection (CISI) Program

Effective September 1996, the NRC endorsed Subsections IWE and IWL of the ASME Section XI, 1992 Edition including 1992 Addenda. These subsections contain inservice inspection and repair/replacement rules for Class MC and Class CC components. The BFN reactor containment is a free standing steel containment to which the requirements of Subsection IWE apply.

For BFN, these inspection requirements are included in the inservice inspection program described in BFN Technical Instruction (TI) 0-TI-376, "ASME Section XI Containment Inservice Inspection Program." The first ten-year interval for IWE containment inspections for BFN started September 9, 1998, and is effective through September 8, 2008. As noted, the program is contained in 0-TI-376, which details inservice inspection requirements for Class MC components in accordance with the requirements of 10 CFR 50.55a(b)(2) and the 1992 Edition of ASME Boiler and Pressure Vessel Code Section XI including 1992 Addenda, Inspection Program B. There are five relief requests in effect for BFN Units 2 and Unit 3, which were approved by NRC on August 6, 2001 (Reference: TAC Nos. MB1634 and MB1635).

For the CISI inspections performed, various indications were observed, documented, evaluated, and determined to be acceptable. One area of the containment liner surface on both Unit 2 and Unit 3 has been scheduled for augmented examination. This area is the torus waterline region, elevation 536' to elevation 538'. No loss of structural integrity of primary containment has been observed and no significant degradation of containment has been identified since the implementation of the CISI Program visual inspections in 1998.

IWE Program Inspection Activities

The BFN IWE Program includes examination of containment surfaces, Examination Category E-A; containment surfaces requiring augmented examination, Examination Category E-C; seals, gaskets, and moisture barriers, Examination Category E-D; pressure retaining bolting, Examination Category E-G; and all pressure retaining components, Examination Category E-P. These requirements will not be changed as a result of the proposed extended ILRT interval.

Category E-A Examinations (Containment Surfaces)

BFN has performed two ASME Section XI IWE Examination Category E-A General Visual examinations on each unit. The last examinations conducted were on Unit 2 Cycle 12 (2003) and Unit 3 Cycle 10 (2002). These examinations were performed to detect evidence of degradation that may affect either leak-tightness or structural integrity of the Primary Containment. Included in this examination are all accessible interior and exterior pressure retaining surfaces, and their integral attachments. The General Visual examinations performed identified flaws such as coating cracking, coating peeling, coating flaking, coating blistering, rusting, discoloration, and some mechanical damage. The mechanical damage identified included pitting, gouges, dents, rust, and arc strikes. These observations were recorded, evaluated, and found to be acceptable.

Category E-C Examinations (Containment Surfaces Requiring Augmented Examination)

Class MC components have been evaluated for determination of augmented examination requirements. These evaluations are included in 0-TI-376. Areas specific to BFN plant design and operating characteristics which were determined to be susceptible to accelerated degradation and aging have been scheduled for examination in accordance with Table IWE-2500-1, Examination Category E-C. The areas determined to be susceptible to accelerated degradation and aging are the Unit 2 and Unit 3 torus waterline region, elevation 536' to elevation 538'.

This area is subject to corrosion due to moisture, and repeated wetting and drying in the waterline region. Accessible portions of the torus inside surface are inspected each refueling outage as required by Surveillance Instruction (SI), 0-SI-4.7.A.2.K, "Primary Containment Drywell Surface Visual Inspection." Additionally, underwater inspections are performed as part of the coatings maintenance program. Underwater coatings processes allow divers to inspect, document, and perform minor coatings repairs as one sequenced activity. These programs have been demonstrated to be effective in early detection of degradation of the torus surfaces. VT-1 visual examinations of the interior surfaces of the Unit 2 and Unit 3 torus waterline region, elevation 536' to elevation 538', have identified areas of coating failure leading to conditions ranging from discoloration to minor localized corrosion and pitting. These observations were recorded, evaluated, and found to be acceptable. There is no evidence of accelerated degradation in these areas. The routine surveillance conducted in accordance with 0-SI-4.7.A.2.K complements the periodic visual inspections of the waterline region and serves to provide added assurance against structural or material degradation.

Category E-D Examinations (Seals, Gaskets, and Moisture Barriers)

Table IWE-2500-1, Examination Category E-D, Item Numbers E5.10 and E5.20, requires seals and gaskets on airlocks, hatches, and other devices to be VT-3 visually examined once each inspection interval to assure containment leak-tight integrity. BFN request for relief CISI-1 was approved to allow containment leak-tight integrity to be demonstrated by testing in accordance with 10 CFR 50, Appendix J, in lieu of performing VT-3 examinations for containment penetration seals and gaskets.

Table IWE-2500-1, Examination Category E-D, Item Number E5.30 requires caulking, flashing, and sealants used to prevent moisture intrusion against the pressure retaining metal containment shell or liner at concrete-to-metal interfaces, both internal and external to containment, to be VT-3 visually examined once each inspection interval. In addition, the Moisture Seal Barrier (MSB) is inspected each operating cycle in accordance with 0-SI-4.7.A.2.K and defective portions replaced. This surveillance complements the periodic visual inspections of the MSB and serves to provide added assurance to prevent moisture intrusion against the pressure retaining metal containment liner at the concrete-to-metal interface.

Category E-G Examinations (Pressure Retaining Bolting)

Table IWE-2500-1, Examination Category E-G, Item Number E8.10 requires a VT-1 visual examination of pressure retaining bolting including bolts, studs, nuts, bushings, washers, threads in base material, and flange ligaments between threaded stud holes. Non-conforming bolting has been found and has been addressed by minor rework or replacement of the bolting material.

Table IWE-2500-1, Examination Category E-G, Item Number E8.20 requires a bolt torque or tension test. BFN request for relief CISI-4 was approved to allow 10 CFR 50, Appendix J testing to be performed in lieu of a bolt torque or tension test for bolted connections that have not been disassembled and reassembled during the inspection interval.

Category E-P Examinations (All Pressure Retaining Components)

Table IWE-2500-1, Examination Category E-P, Item Number E9.10 requires system leakage test in accordance with 10 CFR 50, Appendix J following repair, modification, or replacement of a pressure retaining component. BFN request for relief CISI-2 was approved to allow performance of a VT-1 visual examination during or following the 10 CFR 50, Appendix J leak rate test in those cases where BFN elects not to perform a VT-2 visual examination of repaired or replaced areas during the 10 CFR 50, Appendix J leak rate test.

Table IWE-2500-1, Examination Category E-P, Item Numbers E9.20, E9.30, and E9.40 require containment penetration bellows, airlocks, and seals and gaskets to be tested in accordance with 10 CFR 50, Appendix J. These tests are included in the Containment Leak Rate Programs governed by Engineering instruction, NEDP-14, "Containment Leak Rate Programs," and 0-TI-360, "BFN Containment Leak Rate Programs." These requirements will not be changed as a result of the proposed extended ILRT interval.

Generic Letter 87-05 Inspections

In response to Generic Letter 87-05, Request for Additional Information Assessment of Licensee Measures to Mitigate and/or Detect Degradation of Mark I Drywells (Reference 5), BFN performed ultrasonic thickness (UT) measurements of the drywell shell plates adjacent to the sand cushion. The results of these examinations indicated that significant corrosion had not occurred in the sand bed region, and that water had not been observed leaking from the sand bed drains.

There has been evidence of water leaking from the sand bed drains on both Units 2 and 3 since the 1987 inspections. The sand bed extends from elevation 548.79′ to elevation 550.29′. The drywell floor is at elevation 549.92′, which means there is 0.37 feet of the sand bed extending above the floor. This area is accessible for examination from the inside surface. The horizontal weld connecting the first and second course of drywell liner plates is approximately 8 inches above the floor. UT thickness measurements from the drywell floor up to this weld, around the drywell circumference, would conservatively bound this area. This area is accessible for examination from the inside surface. UT thickness measurements of this area were obtained during the Unit 3 Cycle 8 (September 1998) and Unit 2 Cycle 10 (April 1999) refueling outages. These inspections verified the integrity of the liner. In addition, VT-3 examinations of the steel liner in the area below the MSB are conducted whenever portions of the moisture seal are excavated for repair and UT thickness measurements are taken in any suspect areas.

4.5 BFN Containment Operating Performance

The BFN containment consists of two primary interconnected structures: the drywell, housing the reactor and related components, and a toroidal suppression chamber (torus). The drywell, which includes the major primary containment volume, is inerted with nitrogen and maintained in the range of 1.1 to 1.35 psid positive pressure with respect to the torus. This pressure differential (minimum 1.1 psid) is required to be maintained by TS 3.6.2.6, Drywell-to-Suppression Chamber Differential Pressure. Typically the torus air space pressure is also slightly positive with respect to atmosphere (about .1 psig). The drywell-to-suppression chamber differential pressure is verified every 12 hours in accordance with TS Surveillance Requirement 3.6.2.6.1.

The major portion of the BFN containment is thus normally pressurized. Although the pressure is not as significant as that resulting from a Design Basis Accident, the fact that the containment is normally pressurized provides a high degree of assurance of containment structural integrity (i.e., no large leak paths in the containment structure).

BFN Technical Requirements Manual, Section 3.6.5, Nitrogen Makeup to Containment, monitors the containment for gross leakage by monitoring the average daily nitrogen consumption used by the containment inerting system. Makeup nitrogen use is determined daily by the performance of Surveillance Instruction, SI-4.7.A.2.a, "Primary Containment Nitrogen Consumption and Leakage." Significant containment leakage would be identified using plant instrumentation or through increased nitrogen usage needed to maintain the required differential pressure, and would be investigated promptly and addressed within the scope of the plant Corrective Action Program. This is a complement to the periodic visual inspections of the interior and exterior of the containment structure, and serves to provide added assurance of structural integrity for those areas that may be inaccessible for visual examination.

4.6 NRC Information Notice 92-20

NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," (Reference 6) was issued to alert licensees of problems with local leak rate testing of two-ply stainless steel bellows. The information notice discusses an event at Quad Cities Nuclear Power Station Unit 1 where a Type B test on the containment penetration bellows could identify leakage, but could not accurately quantify the extent of the leakage.

The event at Quad Cities revealed that the local leak rate testing performed between the two plies could not be relied upon to accurately measure the leakage rate that would occur through the bellows under accident conditions. The two plies of the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem. The two-ply expansion bellows installed at BFN incorporate a stainless steel mesh between the plies to eliminate binding as described in the bellows event at Quad Cities and allows full pressure to be transmitted to all portions of the bellows during Appendix J testing.

4.7 Plant Specific Risk Assessment

A plant-specific, risk-based assessment has been performed in support of the one-time deferral of the Type A test frequency from once in 10 years to once in 15 years. This risk analysis is provided in Enclosure 3. The risk assessment approach is based on EPRI TR-104285 and NEI Interim Guidance (References 7 and 8), and is consistent with previous ILRT risk assessments prepared for other Boiling Water Reactors.

The BFN Extended Power Uprate (EPU) full power internal events Level 1 and 2 probabilistic safety assessment (PSA) model for Unit 3 was used as input for this assessment in addition to the Severe Accident Mitigation Alternatives Analysis (SAMA) performed for license renewal. While not currently at EPU conditions, using the EPU model is conservative relative to the current full power PSA model. EPU is planned to be implemented prior to the end of the requested ILRT test interval extension. The Unit 3 model is slightly more conservative than the Unit 2 model in terms of Core Damage Frequency (CDF), Large Early Release Frequency (LERF), and off site dose consequences. It, therefore, bounds the Unit 2 model and the results of the Unit 3 assessments are valid for Unit 2. No separate PSA quantifications were performed using the Unit 2 EPU PSA model. Since the PSA is judged applicable to both Units 2 and 3, the ILRT interval extension risk assessment is applicable to both units.

The risk analysis determined that the proposed TS change results in the following.

- Increasing the current 10-year ILRT interval to 15 years results in a negligible increase in total population dose rate of 0.001 person-rem/year.

- The increase in the LERF risk measure is also insignificant; an approximate 1.47E-08/year increase. This LERF increase is categorized as a "very small" increase per NRC RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002 (Reference 9).

- Likewise, the conditional containment failure probability (CCFP) increases insignificantly by 0.5%.

RG 1.174 provides guidance for determining the risk impact of plant specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of CDF below 1E-06/year and increases in LERF below 1E-07/year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once per 10 years to once per 15 years using the change in the EPRI Category 3b frequency per the NEI Interim Guidance is approximately 1.47E-08/year. As noted, RG 1.174 defines very small changes in LERF as below 1E-07/year. Therefore, increasing the BFN ILRT interval from 10 years to 15 years results in a very small change in risk and is an acceptable plant change from a risk perspective.

The change in CCFP is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The change in CCFP is found to be very small (i.e., 0.5% increase) and represents a negligible change in BFN defense-in-depth.

The impacts on LERF resulting from a change in the Type A ILRT test interval from once per 10 years to once per 15 years due to external events was estimated to be less than 1.13E-08/year, which is also categorized as a very small change in risk. Combining the internal and bounding external events increase in LERF would give a value of 2.60E-08/year, which is still characterized as a very small change in risk, and is an acceptable plant change from a risk perspective. Additional detail is provided in Enclosure 3.

Finally, an assessment of the risk due to non-detectible corrosion induced containment leakage was performed. With the assumed corrosion induced, non-detectable containment leakage probabilities increased by a factor of ten above the "best estimate" values, the increase in LERF due to this proposed TS change was 4.19E-08/year. Combining this internal events bounding value with the previously discussed external events bounding value gives an increase in LERF due to the proposed change of only 5.32E-08/year. This again is classified as a very small change in risk, and is an acceptable plant change from a risk perspective. Additional detail is provided in Enclosure 3.

4.8 Conclusions

Based on the above, the proposed change to TS Section 5.5.12 will continue to provide assurance that leakage through the BFN primary containment will not exceed allowable leakage rate values specified in the TS and Bases, and that the containment features will perform their design function following an accident, up to and including the design basis accident.

5.0 REGULATORY SAFETY ANALYSIS

The Tennessee Valley Authority (TVA) is submitting an amendment request to licenses DPR-52 and DPR-68 for Browns Ferry Nuclear Plant (BFN) Units 2 and 3 Technical Specifications (TS). The proposed amendment revises Technical Specification (TS) Section 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A Integrated Leakage Rate Test (ILRT) for Unit 2 and 3.

5.1 No Significant Hazards Consideration

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

   Response: No

The proposed revision to TS adds a one-time extension to the current interval for Type A testing. The current test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot increase the probability of an accident previously evaluated since the containment Type A testing extension is not a modification and the test extension is not of a type that could lead to equipment failure or accident initiation.

The proposed extension to Type A testing does not involve a significant increase in the consequences of an accident since research documented in NUREG-1493 has found that, generically, very few potential containment leakage paths are not identified by Type B and C tests. The NUREG concluded that reducing the Type A (ILRT) testing frequency to once per 20 years was found to lead to an imperceptible increase in risk. These generic conclusions were confirmed by a plant specific risk assessment.

Testing and the containment inspection programs in place at BFN provide a high degree of assurance that the containment will not degrade in a manner detectable only by Type A testing. The last four Type A tests show leakage to be below acceptance criteria, indicating a very leak tight containment. Type B and C testing required by TS will identify any containment opening such as valves that would otherwise be detected by the Type A tests. Inspections, including those required by the American Society of Mechanical Engineers code are also performed in order to identify indications of containment degradation that could affect that leak tightness.

Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The change does not create the possibility of a new or different kind of accident from any accident previously analyzed. The proposed revision to TS adds a one-time extension to the current interval for Type A testing. The current test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test. The proposed extension to Type A testing cannot create the possibility of a new or different type of accident since there are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident or affecting the mitigation of an accident.

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

Response: No

BFN Units 2 and 3 are General Electric BWR/4 plants with Mark I primary containments. The Mark I primary containment consists of a drywell, which encloses the reactor vessel; reactor coolant recirculation system and branch lines of the Reactor Coolant System; a toroidal-shaped pressure suppression chamber containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber. The primary containment is penetrated by personnel access hatches, piping, and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests and the overall leak-tight integrity of the primary containment is verified by a Type A integrated leak rate test as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure. The proposed change for a one-time extension of the Type A tests does not affect the method for Type A, B, or C testing, or the test acceptance criteria. In addition, based on previous Type A testing results, TVA does not expect additional degradation during the extended period between Type A tests, which would result in a significant reduction in a margin of safety.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36 provides the regulatory requirements for the content required in a licensee's TS. 10 CFR 50.36(c)(5),

"Administrative Controls," requires provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner be included in a licensee's TS.

Additionally, 10 CFR 50, Appendix J, Section V.B, specifies that the regulatory guide or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TS. Deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant's TS.

The proposed change will revise TS Section 5.5.12 to reflect a one-time deferral from the program requirements for the Type A test for BFN Units 2 and 3. The deferral represents an exception to the guidelines contained in RG 1.163 and NEI 94-01. Thus, the proposed change is consistent with the requirements of 10 CFR 50.36(c)(5) and 10 CFR 50, Appendix J, Section V.B. Additionally, in accordance with 10 CFR 50, Appendix J, Section V.B, the proposed changes to BFN TS do not require a supporting request for an exemption to Option B of Appendix J in accordance with 10 CFR 50.12, "Specific exemptions."

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

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENT

The proposed amendment incorporates into the BFN TS changes that are similar to changes approved by the NRC for Fermi on March 27, 2003, Hope Creek on April 16, 2003, LaSalle on November 19, 2003, and Quad Cities on March 8, 2004.

8.0 REFERENCES

1. Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995.

2. NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995.

3. Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," August 1994.

4. NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

5. Generic Letter 87-05, "Request for Additional Information Assessment of Licensee Measures to Mitigate and/or Detect Degradation of Mark I Drywells," March 1987.

6. NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," March 3, 1992.

7. Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals," November 13, 2001.

8. Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, "One-Time Extensions for Containment Integrated Leak Rate Test Intervals - Additional Information," November 30, 2001.

9. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.

Enclosure 2

Browns Ferry Nuclear Plant (BFN)
Units 2 and 3

Technical Specifications (TS) Change 448

One-Time Frequency Extension for Containment
Integrated Leakage Rate Test (ILRT) Interval

Proposed Technical Specifications Changes (mark-up)

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 Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

Unit 2 insert

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 50.6 psig. The maximum allowable primary containment leakage rate, L_a, shall be 2% of primary containment air weight per day at P_a.

(continued)

Unit 2 insert

, 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.

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 Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

Unit 3 insert

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 50.6 psig. The maximum allowable primary containment leakage rate, L_a, shall be 2% of primary containment air weight per day at P_a.

(continued)

Unit 3 insert

, 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.

Enclosure 3

Browns Ferry Nuclear Plant (BFN)
Units 2 and 3

Technical Specifications (TS) Change 448

One-Time Frequency Extension for Containment Integrated Leakage Rate Test (ILRT) Interval

Risk Assessment

TVAN CALCULATION COVERSHEET/CCRIS UPDATE

REV 0 EDMS/RIMS NO.	EDMS TYPE: calculations(nuclear)	EDMS ACCESSION NO (N/A for REV. 0)

Calc Title: **Risk Assessment for Integrated Leak Rate Test (ILRT) Extension**

CALC ID	TYPE	ORG	PLANT	BRANCH	NUMBER	CUR REV	NEW REV	REVISION APPLICABILITY
CURRENT	CN	NUC						Entire calc ☒
NEW	CN	NUC	BFN	NTB	NDN0-064-2004-0005		0	Selected pages ☐

ACTION	NEW ☒ REVISION ☐	DELETE ☐ RENAME ☐	SUPERSEDE ☐ DUPLICATE ☐	CCRIS UPDATE ONLY ☐ (Verifier Approval Signatures Not Required)	No CCRIS Changes ☐ (For calc revision, CCRIS been reviewed and no CCRIS changes required)

UNITS	SYSTEMS	UNIDS
0	064	N/A

DCN.EDC.N/A	APPLICABLE DESIGN DOCUMENT(S)	CLASSIFICATION
N/A	N/A	D

QUALITY RELATED?	SAFETY RELATED? (If yes, QR = yes)	UNVERIFIED ASSUMPTION	SPECIAL REQUIREMENTS AND/OR LIMITING CONDITIONS?	DESIGN OUTPUT ATTACHMENT?	SAR/TS AFFECTED
Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒

PREPARER ID	PREPARER PHONE NO	PREPARING ORG (BRANCH)	VERIFICATION METHOD	NEW METHOD OF ANALYSIS
JDMCCAMY	729-2474	NTB	N/A	☐ Yes ☒ No

PREPARER SIGNATURE	DATE	CHECKER SIGNATURE	DATE
John D. McCamy		R. Abbas	
VERIFIER SIGNATURE	**DATE**	**APPROVAL SIGNATURE**	**DATE**
N/A	N/A	Tim S. Taylor	

STATEMENT OF PROBLEM/ABSTRACT

This analysis supports a one-time extension request for Integrated Leak Rate Testing (ILRT) from 10 years to 15 years for Units 2 and 3. The calculation determines the change in population dose rate (50 miles), the change in Large Early Release Frequency (LERF), and the change in conditional containment failure probability (CCFP) as a result of the ILRT extension and compares the changes to guidance from Regulatory Guide 1.174.

MICROFICHE/EFICHE Yes ☐ No ☒ FICHE NUMBER(S) N/A

☐ LOAD INTO EDMS AND DESTROY
☒ LOAD INTO EDMS AND RETURN CALCULATION TO CALCULATION LIBRARY. ADDRESS:POB TIC -BFN
☐ LOAD INTO EDMS AND RETURN CALCULATION TO:

TVAN CALCULATION COVERSHEET/CCRIS UPDATE

CALC ID	TYPE	ORG	PLANT	BRANCH	NUMBER	REV
	CN	NUC	BFN	NTB	NDN0-064-2004-0005	R00

ALTERNATE CALCULATION IDENTIFICATION

BLDG	ROOM	ELEV	COORD/AZIM	FIRM	Print Report Yes ☐
00					

CATEGORIES NA

KEY NOUNS (A-add, D-delete)

ACTION (A/D)	KEY NOUN	A/D	KEY NOUN
A	ILRT	A	RISK
A	PROBABILISTIC		
A	PROBABILITY		
A	PRA		
A	PSA		

CROSS-REFERENCES (A-add, C-change, D-delete)

ACTION (A/C/D)	XREF CODE	XREF TYPE	XREF PLANT	XREF BRANCH	XREF NUMBER	XREF REV
A	P	IO	BFN	NTB	EPRI TR-104285	
A	P	IO	BFN	NTB	A Petrangelo (NEI) letter dated 11/13/2001	
A	P	IO	BFN	NTB	A Petrangelo (NEI) letter dated 11/30/2001	
A	P	CN	BFN	MEB	MDN0-999-2001-0011	0
A	P	CN	BFN	MEB	MDN0-999-2001-0018	1
A	P	IO	BFN	NTB	Quad Cities letter to NRC dated 2/27/2003	
A	P	RG	BFN	NTB	RG 1.174	1
A	P	CN	BFN	NTB	NTN2-999-2002-0012	0
A	P	CN	BFN	NTB	NTN3-999-2003-0010	0
A	P	VD	BFN	NTB	R06040616012	
A	P	SD	BFN	NTB	PCIS PSA System Notebook	
A	P	IO	BFN	NTB	R06040619021	
A	S	TN	BFN	NTB	Unit 2 TS 5.5.12	
A	S	TN	BFN	NTB	Unit 3 TS 5.5.12	

CCRIS ONLY UPDATES:
Following are required only when making keyword/cross reference CCRIS updates and page 1 of form NEDP-2-1 is not included:

PREPARER SIGNATURE	DATE	CHECKER SIGNATURE	DATE
PREPARER PHONE NO. (256) 729-2474	EDMS ACCESSION NO.		

TVAN CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER NDN0-064-2004-0005	
Title	**Risk Assessment for Integrated Leak Rate Test (ILRT) Extension**
Revision No.	DESCRIPTION OF REVISION
Initial Issue	SAR section 5.2 has been reviewed by Don McCamy and this calculation does not affect the SAR. Technical Specifications have been reviewed and determined not to be affected. Total pages: 27

TVAN COMPUTER INPUT FILE STORAGE INFORMATION SHEET			
Document NDN0-064-2004-0005	Rev. 0	Plant: BFN	

Subject:

Risk Assessment for Integrated Leak Rate Test (ILRT) Extension

Electronic storage of the input files for this calculation is not required. Comments:

X Input files for this calculation have been stored electronically and sufficient identifying information is provided below for each input file. (Any retrieved file requires re-verification of its contents before use.)

The RISKMAN model used in this calculation (U3ILRT) has been placed in filekeeper. The file number is 306621 and the file name is U3ILRT.zip.

Microfiche/eFiche

TVAN CALCULATION TABLE OF CONTENTS

Calculation Identifier: NDN0-064-2004-0005	Revision: 0

TABLE OF CONTENTS

1.0 PURPOSE:

This calculation supports a one-time extension request to extend the required Integrated Leak Rate Test (ILRT) from 10 years to 15 years for both Units 2 and 3. The risk evaluation approach is based on EPRI TR-104285 (ref. 1), NEI Interim Guidance (ref. 2, 3) and previous ILRT risk assessment submittals for other plants.

2.0 REFERENCES:

1. EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals", August 1994.
2. Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals", November 13, 2001.
3. Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, "One-Time Extensions for Containment Integrated Leak Rate Test Intervals - Additional Information", November 30, 2001.
4. MDN0-999-2001-0011, "Level 3 Consequence Assessment for Browns Ferry Nuclear (SAMA)", Rev. 0
5. MDN0-999-2001-0018, "Severe Accident Mitigation Alternatives Analysis (SAMA) for Browns Ferry Nuclear Plant", Rev. 1
6. Quad Cities Nuclear Power Station, Docket Nos. 50-254 and 50-265, Request for Amendment to Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program", dated February 27, 2003.
7. United States Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", Regulatory Guide 1.174, Revision 1, November 2002.
8. Browns Ferry Technical Specifications, Unit 2.
9. Browns Ferry Technical Specifications, Unit 3.
10. ABS Consulting, "Tennessee Valley Authority Browns Ferry Unit 3 Nuclear Power Plant Containment Integrated Leak Rate Test (ILRT) Extension Analysis Support", dated June 15, 2004 (R06040616012).
11. BFN PSA System Notebook, Primary Containment Isolation System, Revision 0.
12. U3EPUP Model Changes to Support ILRT Extension Request, (R06040619021).
13. Dresden Nuclear Power Station, Docket Nos. DPR-19 and DPR-25, Request for Amendment to Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program", dated January 15, 2004.
14. James A. Fitzpatrick Nuclear Power Plant, Docket No 50-333, "Proposed License Amendment to Provide a One-time Integrated Leak Rate Test (ILRT) Interval Extension", dated July 28, 2003.
15. Browns Ferry Nuclear Plant, Docket Nos. 50-259/260/296, "Browns Ferry Nuclear Plant (BFN) - Generic letter (GL) 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Partial Submittal of Report, dated July 24, 1995 (R08950724976).
16. Browns Ferry Nuclear Plant, Docket Nos. 50-260/296, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Generic letter (GL) 87-02, Supplement 1, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46 and Generic letter (GL) 88-20, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Submittal of Seismic Evaluation Reports (TAC Nos. M69431, M69432, M83596, and M83597), dated June 28, 1996 (R08960628859).
17. Browns Ferry Nuclear Plant, Docket No. 50-296, "Browns Ferry Nuclear Plant (BFN) - Unit 3 - Generic letter (GL) 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities (TAC No. M83597), dated July 11, 1997 (R08970711841).
18. NDN2-999-2002-0012, "Unit 2 IPEEE Fire Induced Vulnerability Evaluation".
19. NDN3-999-2003-0010, "Unit 3 IPEEE Fire Induced Vulnerability Evaluation".
20. Kennedy, R. P., "Overview of Methods for Seismic PSA and Margin Analysis Including Recent Innovations", Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan, August, 1999.
21. Electric Power Research Institute, "Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of Charleston Earthquake Issue:, NP-6395-D, April 1989.

22. Electric Power Research Institute, "Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™", EPRI TR-105189, Final Report, May 1995.

3.0 DESIGN INPUT DATA

1. PSA models at Extended Power Uprate (EPU) conditions:
   a. U2EPUB (output in LERF/No LERF)
   b. U2EPUP (output in Plant Damage States)
   c. U3EPUB (output in LERF/No LERF)
   d. U3EPUP (output in Plant Damage States)
   e. U3ILRT (output modified to add Containment Intact bin in addition to Plant Damage States)
2. MDN0-999-2001-0011, "Level 3 Consequence Assessment for Browns Ferry Nuclear (SAMA)", Rev. 0.
3. MDN0-999-2001-0018, "Severe Accident Mitigation Alternatives Analysis (SAMA) for Browns Ferry Nuclear Plant", Rev. 1.
4. ABS Consulting, "Tennessee Valley Authority Browns Ferry Unit 3 Nuclear Power Plant Containment Integrated Leak Rate Test (ILRT) Extension Analysis Support", dated June 15, 2004 (R06040616012).
5. U3EPUP Model Changes to Support ILRT Extension Request, (R06040619021).

4.0 ASSUMPTIONS

1. The Unit 3 EPU models (U3EPUB/U3EPUP) results bound the Unit 2 EPU model (U2EPUB/U2EPUP) results and no specific evaluation for Unit 2 is required. Both the Unit 3 CDF (3.36E-06/yr) and LERF (4.53E-07/yr) are greater than the Unit 2 CDF (2.62E-06/yr) and LERF (3.93E-07/yr). The off site dose consequences based on the above models are greater for Unit 3 (1.95 person-rem/yr) than for Unit 2 (1.64 person-rem/yr). Since the CDF, LERF, and dose consequences for Unit 3 are greater than for Unit 2, it is clear that the Unit 3 results would bound a Unit 2 specific analysis. See reference 5 for additional comparison information.
2. The representative containment leakage for EPRI Category 1 sequences is 1 L_a. This is in accordance with the recommendation of the NEI interim guidance (reference 2).
3. The representative containment leakage for EPRI Category 3a sequences is 10 L_a. This is in accordance with the recommendation of the NEI interim guidance (reference 2).
4. The representative leakage probability for EPRI Category 3a is 0.027. This is in accordance with the recommendation of the NEI interim guidance (reference 2).
5. The representative containment leakage for EPRI Category 3b sequences is 35 L_a. This is in accordance with the recommendation of the NEI interim guidance (reference 2).
6. The representative leakage probability for EPRI Category 3b is 0.0027. This is in accordance with the recommendation of the NEI interim guidance (reference 2).

5.0 SPECIAL REQUIREMENTS/LIMITING CONDITIONS

None.

6.0 COMPUTATIONS AND ANALYSIS

This calculation uses the approach outlined in references 2 and 3. This methodology calculates the change in dose, LERF, and conditional containment failure probability (CCFP). There is no change in CDF as a result of the proposed change. The change in LERF will be compared to the guidelines of Regulatory Guide 1.174. The CCFP will be used as the basis for showing the proposed change is consistent with the defense in depth requirement of

Regulatory Guide 1.174 and completes the requirements for risk informed decision making of Regulatory Guide 1.174. Below is a step-by-step summary of the NEI interim methodology.

1. Quantify the baseline (nominal ten year ILRT interval) frequency per reactor year for the EPRI accident categories of interest. Note that EPRI categories 4, 5, and 6 are not affected by changes in ILRT frequency.
2. Determine the containment leakage rates for EPRI categories 3a and 3b.
3. Develop the baseline population dose (person-rem) for the applicable EPRI categories.
4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in step (3) by the associated frequency calculated in step (1).
5. Determine the change in probability of leakage detectible only by ILRT and the associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however, the probability of leakage detectible only by ILRT does increase.
6. Determine the new population dose rate for the new surveillance intervals of interest.
7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension changes.
8. Evaluate the risk impact in terms of LERF.
9. Evaluate the change in conditional containment failure probability (CCFP).

The EPRI accident classes used in the above methodology are defined as follows:

1. Containment remains intact includes accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant. Changes to leak rate testing frequency do not affect this classification.
2. Containment isolation failures (as reported in the IPEs) include those accidents in which the pre-existing leakage is due to failure to isolate the containment. These include those that are dependent on the core damage accident in progress (e.g., initiated by common cause failure or support system failure at power) and random failures to close a containment path. Changes in Appendix J testing requirements do not impact these accidents.
3. Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak tight containment) is not dependent on the sequence in progress. This accident class is applicable to sequences involving ILRTs (Type A tests) and potential failures not detectable by LLRTs (for example, a hole in the containment liner). The impact on risk from changes in Type A test frequency and leakage rates above L_a has been evaluated in this study (EPRI, ref. 1).
4. Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure is not dependent on the sequence in progress. This accident class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures.
5. Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6. Containment isolation failures include those leak paths not identified by LLRTs. The type of penetration failures considered under this class includes those covered in the plant test and maintenance requirements or verified per in-service inspection and testing (ISI/IST) program. Changes in Appendix J LLRT test intervals do not impact this class of accidents.
7. Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8. Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena). Changes in Appendix J testing do not typically impact these accidents.

6.1 Quantify the baseline risk in terms of frequency per reactor year for the EPRI accident classes of interest (Step 1)

EPRI category 4, 5, and 6 will not be considered in accordance with references 2 and 3. They represent type B/C tests and multiple failures of redundant isolation valves to stroke closed, which will not be considered since they are not impacted by changes in the ILRT frequency and are of low impact. The U3 EPU PSA model (U3EPUP) is bounding for both U2 and U3 (see assumption 1). Since it was designed to produce only CDF and LERF, certain minor modifications are required to further break down the classifications into the required EPRI categories. These modifications are discussed in reference 12. The U3EPUP, modified as described in reference 12, was named U3ILRT and provides Key Plant Damage States (KPDS) for this analysis. It should be noted that the CDF associated with U3ILRT is 3.28E-06/yr. This differs slightly from the U3EPUB model value of 3.36E-06/yr. This difference is negligible for this evaluation and is due to the minor model changes required to determine the "Containment Intact" value. Below these KPDS are described (at the time of core uncovery) and placed in the appropriate EPRI classification. The EPRI category frequencies are tabulated in Section 6.4.

KPDS	DESCRIPTION (AT CORE UNCOVERY)	EPRI CLASS
MIA	High Rx pressure, water on DW floor, containment intact, water to core debris, DW sprays & Suppression Pool Cooling (SPC) available	7
MKC	High Rx pressure, water on DW floor, containment failed early, water to core debris, DW sprays available	7
NIH	High Rx pressure, DW floor dry, containment intact, no water to core debris, DW sprays, SPC & torus vent not available	7
OIA	Low Rx pressure, water on DW floor, containment intact, water to core debris, DW sprays & SPC available	7
PID	Low Rx pressure, DW floor dry, containment intact, water to core debris, DW sprays not available, SPC available	7
PIH	Low Rx pressure, DW floor dry, containment intact, no water to core debris, DW sprays, SPC & torus vent not available	7
PJH	Low Rx pressure, DW floor dry, containment bypassed, no water to core debris	8
PLF	Low Rx pressure, DW floor dry, containment intact, water to core debris, DW sprays available	7

Frequency of EPRI Category 1

This group consists of core damage sequences in which the containment remains intact throughout the accident sequences. The core damage sequences in which the containment remains intact is 1.18E-6/yr. Per NEI interim guidance (references 2 and 3), this value is the Category 1 frequency. In order to maintain the sum of the accident frequencies equal to the CDF, this Category 1 frequency is adjusted by subtracting Category 3a and 3b frequencies. As determined in the subsequent sections, the Category 3a frequency is 8.85E-08/yr and the Category 3b frequency is 8.85E-09/yr. Therefore, the frequency of Category 1 sequences is calculated to be:

EPRI Category 1 frequency = 1.18E-06/yr – 8.85E-08/yr – 8.85E-09/yr = 1.09E-06/yr

Frequency of EPRI Category 2

This group consists of core damage sequences in which the containment fails to isolate. This would be a subset of KPDS PJH and could be determined by multiplying the frequency of KPDS PJH by the split fraction value of a large containment isolation path failure to isolate (top event CIL). The split fraction value for containment isolation with all support available is 1.18E-02/demand (reference 11). Multiplying this value by the KPDS PJH give a frequency of:

EPRI Category 2 frequency = 1.18E-02 * 4.64E-08/yr = 5.47E-10/yr

Frequency of EPRI Category 3a
This group consists of core damage sequences in which the containment is failed from a pre-existing "small" containment leak that would only be identifiable from an ILRT. The CDF for the U3ILRT model is 3.28E-06/yr. Consistent with the NEI interim guidance this is calculated as:

EPRI Category 3a frequency = 3.28E-06/yr * 0.027 = **8.85E-08/yr**

The 3a failure probability (0.027) is obtained from references 2 and 3 and is based on data collected by NEI from 91 nuclear plants. This value assumes an ILRT testing interval of 3 tests every 10 years. Previous ILRT extension requests from other utilities have been questioned regarding the exclusion of station blackout scenarios from the CDF in the calculation of EPRI Category 3a and 3b frequencies as scenarios that cannot cause LERF. For conservatism and simplicity, the full BFN CDF will be used to calculated the 3a and 3b frequencies and will not be adjusted for sequences that independently cause LERF or will never cause LERF.

Frequency of EPRI Category 3b
This group consists of core damage sequences in which the containment is failed from a pre-existing "large" containment leak that would only be identifiable from an ILRT. Consistent with the NEI interim guidance this is calculated as:

EPRI Category 3b frequency = 3.28E-06/yr * 0.0027 = **8.85E-09/yr**

The 3b failure probability (0.0027) is obtained from references 2 and 3 and is based on data collected by NEI from 91 nuclear plants. This value assumes an ILRT testing interval of 3 tests every 10 years. Previous ILRT extension requests from other utilities have been questioned regarding the exclusion of station blackout scenarios from the CDF in the calculation of EPRI Category 3a and 3b frequencies as scenarios that cannot cause LERF. For conservatism and simplicity, the full BFN CDF will be used to calculated the 3a and 3b frequencies and will not be adjusted for sequences that independently cause LERF or will never cause LERF.

Frequency of EPRI Category 7
This group consists of core damage sequences involving containment failure induced by severe accident phenomena. In the U3ILRT model used in this evaluation, all KPDS are binned after it is determined if containment failed. Therefore, KPDS MKC, MIA, NIH, OIA, PID, PIH, PLF would be categorized as accidents involving containment failure induced by severe accident phenomena (Class 7).

EPRI Category 7 frequency (from section 6.3) = **2.05E-06/yr**

Frequency of EPRI Category 8
This group consists of core damage sequences in which the containment is bypassed. KPDS PJH would be categorized as accidents in which the containment is bypassed. The Category 2 value of 5.47E-10/yr must be subtracted from value of KPDS PJH the so that the Category 2 class is not doubly counted. The frequency for Category 8 is then:

EPRI Category 8 frequency = 4.64E-08/yr - 5.47E-10/yr = **4.59E-08/yr**

6.2 Determine the containment leakage rates for EPRI categories 3a and 3b (Step 2)
EPRI categories 3a ("small") and 3b ("large") are accidents with pre-existing containment leakage paths in which the pre-existing isolation failure to seal (i.e., provide a leak tight containment) is not dependent on the sequence in progress. This accident class is applicable to sequences involving ILRTs (Type A tests) and potential failures not detectable by LLRTs (for example, a hole in the containment liner). As recommended by reference 2 and documented in assumption 3, the representative containment leakage for EPRI Category 3a sequences is 10 L_a. The representative containment leakage for EPRI Category 3b sequences is 35 L_a. L_a is the Technical Specification

(references 8 and 9) maximum allowable containment leak rate and is 2% of primary containment air weight per day at P_a (50.6 psig). The containment leakage rate for Category 1 accidents is 1.0 times L_a.

6.3 Develop the baseline population dose for the applicable accident classes (Step 3)

Below is an output from the U3ILRT model listing all plant damage states (PDS), categorized into the correct EPRI accident categories:

Plant Damage States	Frequency (per yr)			
	Category 7	Category 8	Category 1	Category 2
CONTOK			1.1829E-06	
NID	6.7095E-07			
MIB	6.6889E-07			
MLC	2.6669E-07			
MKC	1.1478E-07			
MIA	1.0702E-07			
NIG	7.3369E-08			
NLH	5.4802E-08			
PJA		4.6342E-08		
OLC	2.2379E-08			
OIA	1.9330E-08			
NIE	1.6220E-08			
OIB	1.0039E-08			
MIC	8.3136E-09			
NLF	5.5163E-09			
NKF	3.7561E-09			
OKH	3.4155E-09			
OLF	3.4172E-10			
NIH	2.7919E-10			
PID	2.4814E-10			
PIG	1.8705E-10			
NKH	1.7721E-10			
OIF	1.6718E-10			
OID	1.0228E-10			
OIH	9.8211E-11			
NIF	9.4331E-11			
OIC	8.5639E-11			
PIE	8.4981E-11			
PLH	6.3692E-11			
PJH		5.8000E-11		
OKF	5.7717E-11			
PLF	6.7401E-12			
PIH	0.0000E+00			
				5.4752E-10
	2.0475E-06	**4.6400E-08**	**1.1829E-06**	**5.4752E-10**
(Total) CDF[1]	**3.2768E-06**			

NOTE: (1) Does not include Category 2, which is a subset of PDS PJH. Category 8 value will be adjusted by subtracting Category 2 value to avoid double counting.

These PDS are then binned into the corresponding KPDS as described below as determined by reference 5.

PDS MIA, MIB, MIC, and NID are binned into KPDS MIA.
PDS OIA, OIB, OIC, and OID are binned into KPDS OIA.
PDS NIE, NIF, NIG, NIH, OIF, and OIH are binned into KPDS NIH.
PDS PID is binned into KPDS PID.
PDS PIE, PIG, and PIH, are binned into KPDS PIH.
PDS PJA and PJH, are binned into KPDS PJH.
PDS MKC, NKF, NKH, OKF, and OKH are binned into KPDS MKC.
PDS MLC, NLF, NLH, OLC, OLF, PLF, and PLH are binned into KPDS PLF.

Reference 4 provides the population dose for all KPDS involving containment failure type releases. Reference 10 provides the population dose for Category 1 releases (L_a). Reference 10 ran two additional cases. These additional cases were run using the same assumptions as described in references 4 and 5 for the SAMA results in support of License Renewal using MACCS2. The cross-sectional area of a junction connecting the drywell to the reactor building was calculated to achieve the Technical Specification 5.5.12 primary containment leakage rate of 2% per day at the design-basis LOCA pressure of 50.6 psig and a total containment vapor space volume of 283,000 ft^3. The MAAP equations were used to calculate the junction area. A MAAP test case was performed to verify that the flow rate matched the criterion. The MIA case, selected as the base for the analysis effort, was revised to include an additional junction from the drywell to the reactor building. To maximize leakage a series of cases was performed to evaluate the effect of and determine the minimum drywell spray flow rate that would prevent drywell failure on overpressure, but maintain the drywell at a pressure that was just below the failure pressure. The case with normal drywell spray flow and the case with reduced drywell spray flow were used to perform a MACCS analysis. The former case is considered best estimate (5.70E+01 person-rem) and the latter a conservative release case (6.60E+02 person-rem). For purposes of this evaluation the conservative release value of 6.60E+02 person-rem will be used.

These doses (person-rem) represent the population (out to 50 miles) and other site data extrapolated to the year 2036 and is conservative for present conditions. These dose results were calculated using the MACCS2 code specifically for Browns Ferry. The following table describes the doses associated with each KPDS (and for an intact containment), the EPRI accident category, and the frequency associated with each KPDS.

KPDS	50 MILE RADIUS DOSE (Person-rem)	EPRI CLASS	FREQUENCY (per year)
Containment Intact	6.60E+02	1	1.18E-06
MIA	3.56E+05	7	1.46E-06
MKC	5.56E+06	7	1.22E-07
NIH	7.57E+05	7	9.02E-08
OIA	2.88E+06	7	2.96E-08
PID	6.96E+04	7	2.48E-10
PIH	3.59E+06	7	2.72E-10
PJH	2.02E+05	8	4.64E-08
PLF	3.69E+05	7	3.75E-07
TOTAL	N/A	N/A	**3.28E-06**

The dose for EPRI Category 7 is determined by a weighted average of KPDS comprising Category 7. This approach is acceptable because the total frequency and dose associated with EPRI Category 7 does not change as a result of the ILRT extension. The below table summarizes the results of the weighted averaging process:

KPDS	RELEASE FREQUENCY	50 MILE POPULATION DOSE (Person-rem)	50 MILE POPULATION DOSE RISK (Person-rem/yr)
MIA	1.46E-06	3.56E+05	5.20E-01
MKC	1.22E-07	5.56E+06	6.78E-01
NIH	9.02E-08	7.57E+05	6.83E-02
OIA	2.96E-08	2.88E+06	8.52E-02
PID	2.48E-10	6.96E+04	1.73E-05
PIH	2.72E-10	3.59E+06	9.76E-04
PLF	3.75E-07	3.69E+05	1.38E-01
TOTAL	2.05E-06	7.16E+05(1)	1.47E+00

NOTE: (1) Obtained by dividing total population dose risk (column 4) by the total release frequency (column 2).

The population dose for the "Containment Intact" case (EPRI Category 1) was determined in reference 10 to be <u>6.60E+02 person-rem</u>.

In accordance with the guidance contained in reference 2, the dose for EPRI Category 3a is obtained by multiplying the EPRI Category 1 dose by 10:

EPRI Category 3a dose = 6.60E+02 * 10 = <u>6.60E+03 person-rem</u>

Similarly, the dose for EPRI Category 3b is obtained by multiplying the EPRI Category 1 dose by 35:

EPRI Category 3b dose = 6.60E+02 * 35 = <u>2.31E+04 person-rem</u>

Combining the information from the previous tables, the population doses for the EPRI Categories are:

EPRI CATEGORY	CATEGORY DESCRIPTION	50 MILE POPULATION DOSE (person-rem)
1	Containment Intact	6.60E+02
2	Containment Isolation Failures	2.02E+05
3a	Small Pre-Existing Failures	6.60E+03
3b	Large Pre-Existing Failures	2.31E+04
4	Type B Failures (LLRT)	N/A (1)
5	Type C Failures (LLRT)	N/A (1)
6	Other Containment Isolation Failures	N/A (1)
7	Containment Failures Due to Severe Accidents	7.16E+05
8	Containment Bypass Accidents	2.02E+05

NOTE: (1) EPRI Category 4, 5, and 6 were not be considered in accordance with references 2 and 3. They represent type B/C tests and multiple failures of redundant isolation valves to stroke closed which were not be considered since they are not impacted by changes in the ILRT frequency and are of low impact.

6.4 Determine the baseline population dose rate (Step 4)

The baseline population dose rate is determined by multiplying the EPRI Category population dose by the corresponding frequency (per year). This dose rate is based on an ILRT testing frequency of 3 tests every ten years and is tabulated below:

EPRI CATEGORY	CATEGORY DESCRIPTION	50 MILE DOSE (person-rem)	FREQUENCY (per year)	DOSE RATE (person-rem/yr)
1	Containment Intact	6.60E+02	1.09E-06	7.16E-04
2	Containment Isolation Failures	2.02E+05	5.47E-10	1.10E-04
3a	Small Pre-Existing Failures	6.60E+03	8.85E-08	5.84E-04
3b	Large Pre-Existing Failures	2.31E+04	8.85E-09	2.04E-04
4	Type B Failures (LLRT)	N/A (1)	N/A (1)	N/A (1)
5	Type C Failures (LLRT)	N/A (1)	N/A (1)	N/A (1)
6	Other Containment Isolation Failures	N/A (1)	N/A (1)	N/A (1)
7	Containment Failures Due to Severe Accidents	7.16E+05	2.05E-06	1.47E+00
8	Containment Bypass Accidents	2.02E+05	4.59E-08	9.26E-03
TOTAL			3.28E-06	1.477E+00

NOTE: (1) EPRI Category 4, 5, and 6 were not be considered in accordance with references 2 and 3. They represent type B/C tests and multiple failures of redundant isolation valves to stroke closed which were not be considered since they are not impacted by changes in the ILRT frequency and are of low impact

6.5 Determine the change in probability of leakage only detectible by ILRT (Step 5)

The calculated pre-existing ILRT detectable probabilities, based on three tests every ten years, were discussed previously in section 6.1. They are:

- EPRI Category 3a (small pre-existing leakage) = **2.70E-2**
- EPRI Category 3b (large pre-existing leakage) = **2.70E-3**

Using the standby failure rate model, as discussed in reference 2, the average time that a leak would go undetected is one-half the surveillance interval. Browns Ferry currently performs an ILRT once every ten years. This represents the current leakage probabilities to which the proposed once every fifteen year test will be compared. The once every 10 year pre-existing ILRT detectible leakage probabilities, as calculated per reference 2 are:

- EPRI Category 3a (small pre-existing leakage) = (2.70E-02*120/2)/18 = **9.00E-02**
- EPRI Category 3b (large pre-existing leakage) = (2.70E-03*120/2)/18 = **9.00E-03**

Similarly, the once every 15 year pre-existing ILRT detectible leakage probabilities as calculated per reference 2 are:

- EPRI Category 3a (small pre-existing leakage) = (2.70E-02*180/2)/18 = **1.35E-01**
- EPRI Category 3b (large pre-existing leakage) = (2.70E-03*180/2)/18 = **1.35E-02**

Based on the above adjusted leakage probabilities, the EPRI category frequencies as a function of ILRT test interval are calculated below. Only EPRI Categories 1, 3a, and 3b will be affected by the proposed change in the ILRT test interval. The frequencies for the 10 and 15 year intervals are calculated as explained in Section 6.1 for the 3 in 10 year testing interval.

- EPRI Category 3a frequency (one in ten years) = 3.28E-06/yr * 9.00E-02 = **2.95E-07/yr**

- EPRI Category 3b frequency (one in ten year) = 3.28E-06/yr * 9.00E-03 = 2.95E-08/yr
- EPRI Category 1 frequency (one in ten year) = 1.18E-06/yr - 2.95E-07/yr - 2.95E-09/yr = 8.56E-07/yr
- EPRI Category 3a frequency (one in fifteen year) = 3.28E-06/yr * 1.35E-01 = 4.42E-07/yr
- EPRI Category 3b frequency (one in fifteen year) = 3.28E-06/yr * 1.35E-02 = 4.42E-08/yr
- EPRI Category 1 frequency (one in fifteen year) = 1.18E-06/yr - 4.43E-07/yr - 4.43E-08/yr = 6.93-07/yr

EPRI Category Frequency as a Function of ILRT Test Interval

EPRI CATEGORY	BASELINE 3 ILRTs every 10 years	CURRENT 1 ILRT every 10 years	PROPOSED 1 ILRT every 15 years
1	1.09E-06/yr	8.56E-07/yr	6.93E-07/yr
3a	8.85E-08/yr	2.95E-07/yr	4.42E-07/yr
3b	8.85E-09/yr	2.95E-08/yr	4.42E-08/yr

6.6 Determine the population dose rate for the new surveillance interval (Step 6)
The population dose rate for the new surveillance interval is determined by multiplying the EPRI Category 1, 3a, and 3b frequencies established in Section 6.5 for the various surveillance intervals above by the doses for the respective EPRI categories. Only the dose rates for categories 1, 3a, and 3b will be affected by the change in ILRT surveillance test frequency. The dose rates for all EPRI categories for the various ILRT surveillance intervals are summarized in the table below:

Dose Rate Estimates (person-rem/yr) as a Function of ILRT Test Interval

EPRI CATEGORY	CATEGORY DESCRIPTION	BASELINE (3/10 year ILRT)	CURRENT (1/10 year ILRT)	PROPOSED (1/15 year ILRT)
1	Containment Intact	7.16E-04	5.65E-04	4.58E-04
2	Containment Isolation Failures	1.10E-04	1.10E-04	1.10E-04
3a	Small Pre-Existing Failures	5.84E-04	1.95E-03	2.92E-03
3b	Large Pre-Existing Failures	2.04E-04	6.81E-04	1.02E-03
4	Type B Failures (LLRT)	N/A (1)	N/A (1)	N/A (1)
5	Type C Failures (LLRT)	N/A (1)	N/A (1)	N/A (1)
6	Other Containment Isolation Failures	N/A (1)	N/A (1)	N/A (1)
7	Containment Failures Due to Severe Accidents	1.47E+00	1.47E+00	1.47E+00
8	Containment Bypass Accidents	9.26E-03	9.26E-03	9.26E-03
TOTAL		1.477E+00	1.479E+00	1.480E+00

NOTE: (1) EPRI Category 4, 5, and 6 were not be considered in accordance with references 2 and 3. They represent type B/C tests and multiple failures of redundant isolation valves to stroke closed, which were not be considered, since they are not impacted by changes in the ILRT frequency and are of low impact

6.7 Determine the change in dose rate for the new ILRT interval (Step 7)
From the dose rates established in the above table, the dose rate increases only 0.001 person-rem/yr from the current ILRT test frequency for the once every ten years value of 1.479 person-rem/yr to the once every 15 years value of 1.480 person-rem/yr.

The increase of total dose rate/yr from the proposed increase in ILRT test frequency from once every ten years to once every fifteen years is therefore:

$$[(1.480 - 1.479)/1.479] * 100 = \underline{0.07\%}$$

For the 10 year testing interval the percentage dose rate contribution of the Category 3a and 3b with respect to the total dose rate is:

$$[(1.95E\text{-}03 + 6.81E\text{-}04)/1.479] * 100 = \underline{0.18\%}$$

For the 15 year testing interval the percentage dose rate contribution of the Category 3a and 3b with respect to the total dose rate is:

$$[(2.92E\text{-}03 + 1.02E\text{-}03)/1.480] * 100 = \underline{0.27\%}$$

The percentage contributions of dose rate from Categories 3a and 3b due to the proposed ILRT testing interval extension from once every ten years to once every fifteen years remains very minor and the increase is less than 0.1%. This insignificant increase in dose rate from the proposed increased ILRT testing frequency provides additional assurance that defense in depth is maintained.

6.8 Determine the change in LERF for the new ILRT interval (Step 8)

The risk associated with extending the ILRT test interval involves a potential that a core damage event that normally would result in only a small radioactive release from containment could result in a large release due to an undetected release path existing during the extended interval. As discussed in reference 2, only class 3b sequences have the potential to result in large early releases if a pre existing leak were present. The frequency of class 3b sequences are used as a measure of LERF and the change in LERF is determined by the change in class 3b frequency.

$$\Delta LERF = [\text{Class 3b frequency (15 year ILRT interval)}] - [\text{Class 3b frequency (10 year ILRT interval)}]$$
$$= 4.42E\text{-}08/yr - 2.95E\text{-}08/yr = \underline{1.47E\text{-}08/yr}$$

This change in LERF falls deeply into Region III, "Very Small Change in Risk", of the acceptance guidelines in NRC Regulatory Guide 1.174. Therefore, increasing the ILRT testing frequency from the current once every ten years to a once every fifteen year interval is a very small change in risk and is acceptable from a risk perspective.

6.9 Determine the change in CCFP for the new ILRT interval (Step 9)

Another risk measure which is potentially affected by this proposed change is the Conditional Containment Failure Probability (CCFP). The CCFP would be indicative of the effect of the proposed change on all radioactive releases, not only LERF. In accordance with reference 2 guidance, CCFP is defined as those sequences other than Category 1 and 3a.

$$CCFP\% = [1 - (\text{Class 1 frequency} + \text{Class 3a frequency})/CDF] * 100\%$$

For the 10 year ILRT test interval:

$$CCFP_{10\ year} = 1 - [(8.56E\text{-}07 + 2.95E\text{-}07)/3.28E\text{-}06] * 100\% = \underline{64.9\%}$$

For the 15 year ILRT test interval:

$$CCFP_{15\ year} = 1 - [(6.93E\text{-}07 + 4.42E\text{-}07)/3.28E\text{-}06] * 100\% = \underline{65.4\%}$$

This results in a ΔCCFP of 0.5%. This change in risk is insignificant in terms of risk, therefore, the change is acceptable in terms of ΔCCFP. This extremely insignificant increase in CCFP from the proposed increased ILRT testing frequency provides additional assurance that defense in depth is maintained.

6.10 External Events Evaluation

6.10.1 High Winds, Flooding, Transportation, and Nearby Industrial Facility Accidents

Reference 15 determined that the risk due to high winds, external flooding, transportation, and nearby industrial facility accidents which might lead to core damage were below the screening criteria frequency of less than 10^{-6}/yr. Therefore, these external events are not evaluated further in this calculation and are expected to have an insignificant impact on the results of this calculation.

6.10.2 Fire

Reference 15 (for Unit 2) and reference 17 (for Unit 3) utilized a combination of the EPRI Fire Induced Vulnerability Evaluation (FIVE) and PSA methodology. All plant areas evaluated for both Unit 2 and Unit 3 were screened from further evaluation since the frequency of those fire induced accidents, which might lead to core damage, were below the screening criteria frequency of less than 10^{-6}/yr. Fire related LERF was also below the screening criteria frequency of 10^{-7}/yr. Therefore, fires do not result in or cause containment breach concerns beyond those already addressed in the BFN risk model. While the screening of all fire areas may be unusual, BFN has a robust electrical distribution system with substantial cross connect capabilities between the individual units and boards. For example, Browns Ferry has seven 500 KV and two 161 KV off site power lines, eight diesel generators, eight 4 KV shutdown boards, six 480 V shutdown boards, and 3 main battery boards; many of which have cross-connect capabilities. In addition, there exists a cross connect capability between adjacent units in which Residual Heat Removal (RHR) from an adjacent unit can supply accident unit requirements. These FIVE evaluations have been updated in references 18 and 19 to reflect results from the latest PRA updates. The conclusion of these updated calculations did not change the conclusions of references 15 and 17. Therefore, fire events are not evaluated further in this calculation and are expected to have an insignificant impact on the results of this calculation.

6.10.3 Seismic

Reference 16 documented completion of a Seismic Margins Assessment (SMA) following the guidance of NUREG-1407 and EPRI NP-6041 as a focused scope plant assessment. The SMA is a deterministic process which does not calculate risk values.

Reference 20 provides a simplified methodology (Simple Hybrid Method) for estimating the seismic risk based on a SMA analysis. It has shown that only the individual plant High Confidence Low Probability of Failure (HCLPF) seismic capacity is required in order to estimate the seismic CDF within an approximate factor of two. This approach has been used in previous NRC submittals, such as references 13 and 14, and will be used here. The approach is:

Step 1: Determine the BFN HCLPF seismic capacity (C_{HCLPF}) from the SMA analysis

Step 2: Estimate the 10% conditional failure probability capacity ($C_{10\%}$)from:

$$C_{10\%} = F_\beta * C_{HCLPF}$$

$$F_\beta = e^{1.044\beta}$$

Where 1.044 is the difference between the 10% NEP standard normal variable (-1.282) and the 1% NEP standard normal variable (-2.326).

Experience gained from previous high quality seismic PSA's indicates the plant damage state fragility determined by rigorous convolution will tend to have β_c values in the range of 0.30 to 0.35 (the plant damage state β_c value is less than the β_c values for the fragilities of the individual components that dominate the seismic risk). Therefore, the Simplified Hybrid Model recommends:

$$C_{10\%} = 1.4 * C_{HCLPF}$$

Step 3: Determine the hazard exceedance frequency ($H_{10\%}$) that corresponds to $C_{10\%}$ from the hazard curves.

Step 4: Determine the seismic risk PF (i.e., seismic related CDF) from:

$$PF = 0.5 * H_{10\%}$$

Using the above steps the Simplified Hybrid Model can be applied to BFN to estimate the seismic risk below:

Step 1: The plant HCLPF is determined in reference 16 to be at least 0.26g peak ground acceleration (PGA).

Step 2: Using the relationship described above:

$$C_{10\%} = 1.4 * 0.26\text{g PGA} = \underline{0.364\text{g PGA}} \text{ or } \underline{357\text{cm/sec}^2}$$

Step 3: The seismic hazard curve for BFN is obtained from reference 21 (Table 3-9). It is replicated below with the BFN HCLPF of 0.364g PGA estimated from the available data points and added to the table:

BFN SEISMIC HAZARD CURVE
(FROM EPRI NP-6395-D)

Peak Ground Acceleration		Exceedance Frequency
Cm/sec²	g	(mean annual)
5	0.01	2.2E-02
50	0.05	5.9E-04
100	0.10	1.4E-04
250	0.25	1.5E-05
357 (1)	0.364 (1)	5.0E-06 (1)
500	0.51	1.7E-06
700	0.71	4.9E-07
1000	1.02	1.1E-07

NOTE (1) The value of <u>5.0E-06/yr</u> for 0.364g was obtained from Attachment 1. The above points were plotted on Attachment 1 and a best fit curve was plotted. The value for 0.364g was taken directly from the plotted curve.

Step 4: Using the recommended relationship of:

$$PF \text{ (i.e., seismic related CDF)} = 0.5 * H_{10\%} = 0.5 * 5.0\text{E-06/yr} = \underline{2.5\text{E-06/yr}}$$

The Simplified Hybrid Model provides an overall estimate of seismic risk, but does not provide information as to the specific accident sequences. Conservatively, as in the calculation of Category 3b frequency in Section 6.1, the full estimated seismic CDF will be used to calculated the 3b frequency and will not be adjusted for sequences that independently cause LERF or will never cause LERF.

In order to assess the impact of external events on the proposed ILRT extension request, the impact on LERF will be evaluated in accordance with the NEI Interim Guidance , references 2 and 3. Per the Interim Guidance, the impact of the proposed change in LERF is:

$$\Delta\text{LERF} = [\text{Class 3b frequency (15 year ILRT interval)}] - [\text{Class 3b frequency (10 year ILRT interval)}]$$

The frequency for EPRI Category 3b is:

$$\text{3b frequency} = [\text{3b conditional failure probability}] * \text{CDF}$$

Therefore, the baseline 3b frequency due to external events is:

$$\text{3b frequency} = 2.70\text{E-}03 * 2.5\text{E-}06/\text{yr} = \underline{6.75\text{E-}09/\text{yr}}$$

Performing the 3b calculation for the one in ten year and one in fifteen year testing frequency, as performed in Section 6.5, previously:

$$\text{EPRI Category 3b frequency (one in ten year)} = 9.00\text{E-}03 * 2.5\text{E-}06/\text{yr} = \underline{2.25\text{E-}08/\text{yr}}$$

$$\text{EPRI Category 3b frequency (one in fifteen year)} = 1.35\text{E-}02 * 2.5\text{E-}06/\text{yr} = \underline{3.38\text{E-}08/\text{yr}}$$

Therefore, ΔLERF for external events is:

$$\Delta\text{LERF} = 3.38\text{E-}08 - 2.25\text{E-}08 = \underline{1.13\text{E-}08/\text{yr}}$$

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL SEISMIC EVENT

EPRI Category	Quantitative Results as a Function of ILRT Interval		
	Baseline (3 per 10 year ILRT)	Current (1 per 10 year ILRT)	Proposed (1 per 15 year ILRT)
	Accident Frequency per year	Accident Frequency per year	Accident Frequency per year
3b	6.75E-09/yr	2.25E-08/yr	3.38E-08/yr
Increase in LERF		**1.57E-08/yr**	**1.13E-08/yr**

The external events ΔLERF (1.13E-08/yr) is slightly less but approximately equal to the internal events ΔLERF (1.47E-08/yr). The total of the internal and external ΔLERF is:

$$\Delta\text{LERF}_{\text{TOTAL}} = 1.47\text{E-}08/\text{yr} + 1.13\text{E-}08/\text{yr} = \underline{2.60\text{E-}08/\text{yr}}$$

Since the ΔLERF value for external events is approximately equal to the internal events ΔLERF, it is expected the other figures of merit (dose rate and CCFP changes) are likewise insignificant and will not be replicated for external events.

6.11 Containment Degradation Evaluation

Inspection of some reinforced and steel containments (e.g., North Anna, Brunswick, D.C. Cook, and Oyster Creek) have indicated degradation from uninspectable (embedded) side of the steel shell and liner of primary containment. The NRC has consistently requested licensees to perform a quantitative evaluation of the impact on LERF due to age-related degradation of non-inspectable areas of the containment. In response to these requests, a quantitative assessment using the same approach used by other industry plants (e.g., Dresden) will be performed in this section.

6.11.1 Corrosion Analysis

Reference 13 (Dresden ILRT submittal) was reviewed for applicability to BFN. Dresden is a BWR Mark I containment, as is BFN. The analysis and results are judged applicable to BFN. The derivation of the corrosion induced, non-detectable containment leakage probabilities will not be reproduced here. They are well described in reference 13. The values are:

- At 3 years - 1.07E-04
- At 10 years - 6.21E-04
- At 15 years - 1.45E-03

6.11.2 Change in EPRI Category 1 and 3b due to Non-Detectible Corrosion

Conservatively assuming that the impact of non-detectible corrosion induced containment leakage sequences results in EPRI Category 3b type leakage, the category 3b frequency and resulting category 1 frequency changes were calculated as in Section 6.5 previously. The change in category 3b frequency due to non-detectable corrosion leakage will be added to the previously determined category 3b leakage as determined in Section 6.5. Likewise, the change in category 3b frequency due to non-detectable corrosion leakage will be subtracted from the category 1 leakage value from Section 3.5 to maintain the total CDF constant.

EPRI Category 3b frequency (three in ten year) = 8.85E-09/yr + (3.28E-06/yr * 1.07E-04) = **9.20E-09/yr**

EPRI Category 1 frequency (three in ten year) = 1.09E-06/yr - (3.28E-06/yr * 1.07E-04) = **1.09E-06/yr**

EPRI Category 3b frequency (one in ten year) = 2.95E-08/yr + (3.28E-06/yr * 6.21E-04) = **3.15E-08/yr**

EPRI Category 1 frequency (one in ten year) = 8.56E-07/yr - (3.28E-06/yr * 6.21E-04) = **8.54E-07/yr**

EPRI Category 3b frequency (one in fifteen year) = 4.42E-08/yr + (3.28E-06/yr * 1.45E-03) = **4.89E-08/yr**

EPRI Category 1 (one in fifteen year) = 6.93E-7/yr - (3.28E-06/yr * 1.45E-03) = **6.88E-07/yr**

EPRI Category Frequency as a Function of ILRT Test Interval With Non-Detectible Corrosion Effects Included

EPRI CATEGORY	BASELINE 3 ILRTs every 10 years	CURRENT 1 ILRT every 10 years	PROPOSED 1 ILRT every 15 years
1	1.09E-06/yr	8.54E-07/yr	6.88E-07/yr
3a	8.85E-08/yr	2.95E-07/yr	4.42E-07/yr
3b	9.20E-09/yr	3.15E-08/yr	4.89E-08/yr

The same methodology described in Sections 6.6 through 6.9 can be used to determine the population dose rate, change in dose rates, change in LERF and change in CCFP for the above frequencies which include the effects of non-detectible corrosion on the proposed ILRT testing extension. These results are summarized in the table on Attachment 2.

As can be seen from the below table, there is either no increase or an insignificant increase in the risk metrics if non-detectible corrosion effects are considered:

Comparison of ILRT Extension With and Without the Effects of Non-Detectable Corrosion

	Baseline (Non-Detectible Corrosion Effects not Considered)	Non-Detectible Corrosion Effects Considered
% Increase in Total Dose	0.07%	0.07%
% of Category 3a and 3b to Total Dose	0.27%	0.27%
Δ LERF	1.47E-08/yr	1.74E-08/yr
Δ CCFP %	0.5%	0.5%

6.11.3 <u>Sensitivity of Results to Corrosion Induced, Non-Detectable Containment Leakage Probabilities</u>
In order to determine the sensitivity of the analysis to the assumed corrosion induced, non-detectable containment leakage probabilities, the leakage probabilities are increased by a factor of ten:

- At 3 years - 1.07E-03
- At 10 years - 6.21E-03
- At 15 years - 1.45E-02

As in Section 6.11.2 above, the following revised leakage probabilities are obtained:

EPRI Category 3b frequency (three in ten year) = 8.85E-09/yr + (3.28E-06/yr * 1.07E-03) = <u>1.24E-08/yr</u>

EPRI Category 1 frequency (three in ten year) = 1.09E-06/yr - (3.28E-06/yr * 1.07E-03) = <u>1.09E-06/yr</u>

EPRI Category 3b frequency (one in ten year) = 2.95E-08/yr + (3.28E-06/yr * 6.21E-03) = <u>4.99E-08/yr</u>

EPRI Category 1 frequency (one in ten year) = 8.56E-07/yr - (3.28E-06/yr * 6.21E-03) = <u>8.36E-07/yr</u>

EPRI Category 3b frequency (one in fifteen year) = 4.42E-08/yr + (3.28E-06/yr * 1.45E-02) = <u>9.18E-08/yr</u>

EPRI Category 1 (one in fifteen year) = 6.93E-07 – (3.28E-06/yr * 1.45E-02) = <u>6.45-E07/yr</u>

EPRI Category Frequency as a Function of ILRT Test Interval With Non-Detectible Corrosion Effects Included Sensitivity Case

EPRI CATEGORY	BASELINE 3 ILRTs every 10 years	CURRENT 1 ILRT every 10 years	PROPOSED 1 ILRT every 15 years
1	1.09E-06/yr	8.36E-07/yr	6.45E-07/yr
3a	8.85E-08/yr	2.95E-07/yr	4.42E-07/yr
3b	1.24E-08/yr	4.99E-08/yr	9.18E-08/yr

Using the same process as in Section 6.11.2 above, the population dose rate, change in dose rates, change in LERF and change in CCFP for the above frequencies which include the effects of non-detectible corrosion on the proposed

ILRT testing extension for the above sensitivity case are calculated. These results are summarized in the table on Attachment 3.

Comparison of ILRT Extension With and Without the Effects of Non-Detectable Corrosion (Sensitivity Case)

	Baseline (Non-Detectible Corrosion Effects not Considered)	Non-Detectible Corrosion Effects Considered	Non-Detectible Corrosion Effects Considered (Sensitivity Case)
% Increase in Total Dose	0.07%	0.07%	0.14%
% of Category 3a and 3b to Total Dose	0.27%	0.27%	0.34%
Δ LERF	1.47E-08/yr	1.74E-08/yr	4.19E-08/yr
Δ CCFP %	0.5%	0.5%	1.3%

From the above table it can be seen that the proposed ILRT extension has negligible impact on the above risk metrics, even if the corrosion induced, non-detectable containment leakage probabilities are a factor of ten greater that the best estimate values.

7.0 SUMMARY OF RESULTS

This evaluation is based on NEI Guidance (references 2 and 3), EPRI-TR-104285 (reference 1) and previous NRC submittals concerning ILRT extension requests (references 6, 13 and 14). These results indicate a very small risk impact associated with a one-time extension of the ILRT test interval from ten years to fifteen years.

This calculation reviewed specific Browns Ferry Plant Damage States (PDS) and binned them into the correct Key Plant Damage States (KPDS). These KPDS were then placed into the correct EPRI accident categories in accordance with references 1, 2, and 3. The change in dose rate, LERF, and CCFP were determined for the increase of ILRT testing frequency from a ten year to a fifteen year interval. The results are summarized in the following table:

EPRI Category	Dose (Person-Rem within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Baseline (3 per 10 year ILRT)		Current (1 per 10 year ILRT)		Proposed (1 per 15 year ILRT)	
		Accident Frequency per year	Population Dose rate (Person-Rem/Year within 50 miles)	Accident Frequency per year	Population Dose rate (Person-Rem/Year within 50 miles)	Accident Frequency per year	Population Dose rate (Person-Rem/Year within 50 miles)
1	6.60E+02	1.09E-06	7.16E-04	8.56E-07	5.65E-04	6.93E-07	4.58E-04
2	2.02E+05	5.47E-10	1.10E-04	5.47E-10	1.10E-04	5.47E-10	1.10E-04
3a	6.60E+03	8.85E-08	5.84E-04	2.95E-07	1.95E-03	4.42E-07	2.92E-03
3b	2.31E+04	8.85E-09	2.04E-04	2.95E-08	6.81E-04	4.42E-08	1.02E-03
4	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6	N/A	N/A	N/A	N/A	N/A	N/A	N/A
7	7.16E+05	2.05E-06	1.47E+00	2.05E-06	1.47E+00	2.05E-06	1.47E+00
8	2.02E+05	4.59E-08	9.26E-03	4.59E-08	9.26E-03	4.59E-08	9.26E-03
Totals		3.28E-06	1.477E+00	3.28E-06	1.479E+00	3.28E-06	1.480E+00
Increase in Dose rate					2.00E-03		1.00E-03
ILRT Dose rate % of Total Dose					0.11%		0.07%
Percentage of Total Dose Attributable to Cat. 3a and 3b					0.18%		0.27%
Increase in LERF				2.06E-08		1.47E-08	
CCFP %		64.1%		64.9%		65.4%	
Increase in CCFP %				0.8%		0.5%	

1. Total dose rate and the dose rate due to Categories 3a and 3b increased by less than 0.1% and is negligable.
2. LERF increased by 1.47E-08/yr. This is well below the NRC Regulatory Guide (RG) 1.174 value of 1E-07, which can be classified a "very small" risk increase.
3. CCFP increased by 0.5% and is negligible.

The increase in LERF due to external events (1.13E-08/yr) is approximately equal to the internal events ΔLERF (1.47E-08/yr). The total LERF increase from both internal and external events is 2.60E-08/yr, well below the RG 1.174 value of 1E-07, which can be classified a "very small" risk increase.

The "best estimate" results when non-detectible corrosion effects was negligible. The total dose rate, the dose rate percentage from Categories 3a and 3b and the CCFP did not change when compared to the case without non-detectible corrosion effects and LERF increased by only 2.70E-09/yr. Even with the non-detectable containment leakage probabilities increased by a factor of ten, the risk metrics were insignificant. LERF increased by only 4.19E-08/yr, well below the RG 1.174 value of 1E-07, which can be classified a "very small" risk increase.

8.0 CONCLUSION

The changes in the three evaluated metrics: dose rate, LERF, and CCFP are insignificant and well below the level of regulatory concern. The proposed change in ILRT test intervals from ten to fifteen years is acceptable from a risk perspective. This calculation has evaluated the change in risk due to extension of ILRT testing from once every ten years to once every fifteen years and the changes in the three metric discussed in Section 7.0 above were negligible. It should be pointed out that the performance of an ILRT in itself places the plant in an unusual configuration with negative effects on overall plant risk. EPRI has documented in reference 22 that there are real risk impacts associated with the setup and performance of ILRTs. In view of the positive risk aspects of avoidance of unnecessary ILRTs and the negligible increase in risk due to the proposed ILRT extension, it is judged that the proposed extension is risk neutral.

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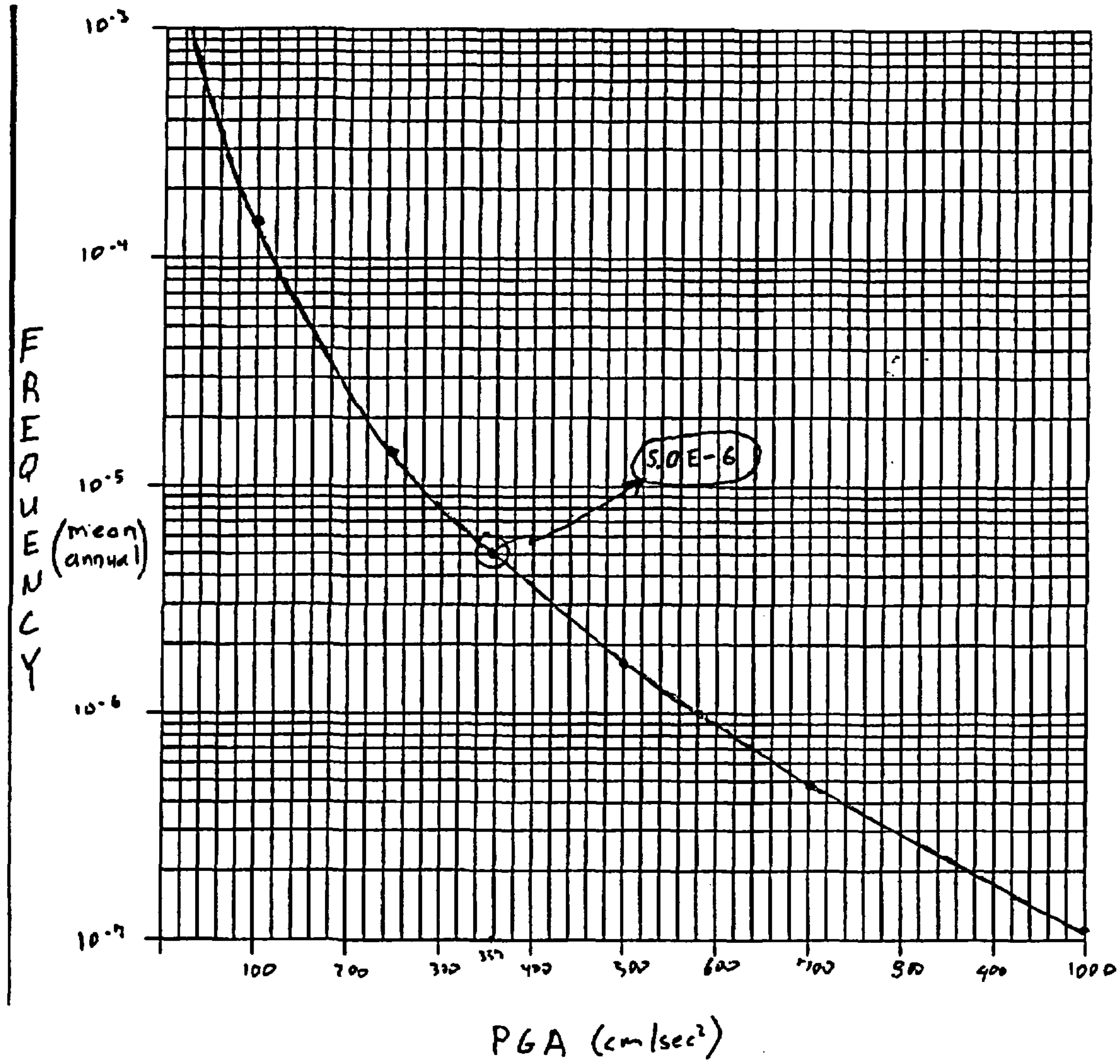
		Checked: RA	Date: 6/21/04

ATTACHMENT 1
SEISMIC HAZARD CURVE



ATTACHMENT 2
ILRT Results Considering Non-Detectible Corrosion Effects

EPRI Category	Dose (Person-Rem within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Baseline (3 per 10 year ILRT)		Current (1 per 10 year ILRT)		Proposed (1 per 15 year ILRT)	
		Accident Frequency per year	Population Dose rate (Person-Rem/Year within 50 miles)	Accident Frequency per year	Population Dose rate (Person-Rem/Year within 50 miles)	Accident Frequency per year	Population Dose rate (Person-Rem/Year within 50 miles)
1	6.60E+02	1.09E-06	7.19E-04	8.54E-07	5.63E-04	6.88E-07	4.54E-04
2	2.02E+05	5.47E-10	1.10E-04	5.47E-10	1.10E-04	5.47E-10	1.10E-04
3a	6.60E+03	8.85E-08	5.84E-04	2.95E-07	1.95E-03	4.42E-07	2.92E-03
3b	2.31E+04	9.20E-09	2.13E-04	3.15E-08	7.28E-04	4.89E-08	1.13E-03
4	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6	N/A	N/A	N/A	N/A	N/A	N/A	N/A
7	7.16E+05	2.05E-06	1.47E+00	2.05E-06	1.47E+00	2.05E-06	1.47E+00
8	2.02E+05	4.59E-08	9.26E-03	4.59E-08	9.26E-03	4.59E-08	9.26E-03
Totals		3.28E-06	1.477E+00	3.28E-06	1.479E+00	3.28E-06	1.480E+00
Increase in Dose rate					**2.000E-03**		**1.000E-03**
ILRT Dose rate % of Total Dose					**0.14%**		**0.07%**
Percentage of Total Dose Attributable to Cat. 3a and 3b					**0.18%**		**0.27%**
Increase in LERF				**2.23E-08**		**1.74E-08**	
CCFP %		**64.0%**		**65.0%**		**65.5%**	
Increase in CCFP %				**1.0%**		**0.5%**	

ATTACHMENT 3
ILRT Results Considering Non-Detectible Corrosion Effects (Sensitivity Case)

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

		Quantitative Results as a Function of ILRT Interval					
		Baseline (3 per 10 year ILRT)		Current (1 per 10 year ILRT)		Proposed (1 per 15 year ILRT)	
EPRI Category	Dose (Person-Rem within 50 miles)	Accident Frequency per year	Population Dose rate (Person-Rem/Year within 50 miles)	Accident Frequency per year	Population Dose rate (Person-Rem/Year within 50 miles)	Accident Frequency per year	Population Dose rate (Person-Rem/Year within 50 miles)
1	6.60E+02	1.09E-06	7.19E-04	8.36E-07	5.52E-04	6.45E-07	4.26E-04
2	2.02E+05	5.47E-10	1.10E-04	5.47E-10	1.10E-04	5.47E-10	1.10E-04
3a	6.60E+03	8.85E-08	5.84E-04	2.95E-07	1.95E-03	4.42E-07	2.92E-03
3b	2.31E+04	1.24E-08	2.86E-04	4.99E-08	1.15E-03	9.18E-08	2.12E-03
4	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6	N/A	N/A	N/A	N/A	N/A	N/A	N/A
7	7.16E+05	2.05E-06	1.47E+00	2.05E-06	1.47E+00	2.05E-06	1.47E+00
8	2.02E+05	4.59E-08	9.26E-03	4.59E-08	9.26E-03	4.59E-08	9.26E-03
Totals		3.28E-06	1.477E+00	3.28E-06	1.479E+00	3.28E-06	1.481E+00
Increase in Dose rate					**2.000E-03**		**2.000E-03**
ILRT Dose rate % of Total Dose					**0.14%**		**0.14%**
Percentage of Total Dose Attributable to Cat. 3a and 3b					**0.21%**		**0.34%**
Increase in LERF				**3.75E-08**		**4.19E-08**	
CCFP %		**64.0%**		**65.5%**		**66.8%**	
Increase in CCFP %				**1.5%**		**1.3%**	