Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

July 27, 2009

10 CFR 50.90

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

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **Technical Specifications Change 465 - Revision of Technical Specifications to Eliminate Unnecessary Water Local Leak Rate Tests**

Pursuant to 10 CFR 50.90, the Tennessee Valley Authority (TVA) is submitting a request for a Technical Specifications (TS) change (TS-465) to licenses DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant, Units 1, 2, and 3, respectively.

The proposed amendment would eliminate the Surveillance Requirement (SR) related to the performance of water local leak rate tests for identified containment isolation valves.

The Enclosure provides a description of the proposed change. Attachment 1 identifies Regulatory Commitments. Attachment 2 provides the existing Units 1, 2 and 3 TS pages marked-up to show the proposed change for valves which terminate below the suppression pool minimum water level and remain water sealed throughout the post accident period. Attachment 3 provides a list of affected valves. This proposed change will eliminate the need to perform certain water local leak rate tests.

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 Enclosure to the Alabama State Department of Public Health.

TVA is asking that this TS change request be approved by January 31, 2010, and that the implementation of the revised TS be made within 30 days of NRC approval.

As indicated in Attachment 1, there are no commitments contained within this submittal.

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Should you have any questions concerning this submittal, please contact Mr. Bob Birchell at (423) 751-4498.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 27, 2009.

Respectfully,

R. M. Krich
Vice President
Nuclear Licensing

Enclosure: Technical Specifications (TS) Change 465

Attachments: Attachment 1 Regulatory Commitments
Attachment 2 Proposed TS Changes (mark-up)
Attachment 3 Affected Valve List

Enclosure
cc: (Enclosure):

Regional Administrator – NRC Region II

NRC Senior Resident Inspector – Browns Ferry Nuclear Plant

State Health Officer – Alabama State Department of Public Health

Enclosure

Browns Ferry Nuclear Plant (BFN)
Units 1, 2, and 3

Technical Specifications (TS) Change 465

Revision of Technical Specifications to Eliminate Unnecessary Water Local Leak Rate Tests for Units 1, 2, and 3.

1.0 SUMMARY DESCRIPTION

BFN TS Surveillance Requirement (SR) 3.6.1.3.11 requires local leak rate testing to be performed on Primary Containment Isolation Valves (CIVs) that serve lines which terminate below the suppression pool minimum water level and CIVs which are water sealed throughout the post accident period. These local leak rate tests are performed with water. SR 3.6.1.3.11 states, "Verify combined leakage through water tested lines that penetrate primary containment are within the limits specified in the Primary Containment Leakage Rate Testing Program."

TS 3.6.1.3.11 Bases states, "Surveillance of water tested lines ensures that sufficient inventory will be available to provide a sealing function for at least 30 days at a pressure of 1.1 Pa. Sufficient inventory ensures there is not a path for leakage of primary containment atmosphere to the environment following a DBA. Leakage from containment isolation valves that terminate below the suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available as described in 10 CFR 50, Appendix J, Option B. Leakage through valves in closed loop seismic class I lines that are considered as extensions of primary containment present no potential for leakage to the environment. Leakage from these valves will be measured, but will be excluded when computing the total leakage. This leakage will be reported as required by the Primary Containment Leakage Rate Testing Program."

The proposed change eliminates local leak rate testing of the components listed in Attachment 3. Dose savings can be realized by eliminating testing that is not required by the applicable codes. This change eliminates water local leak rate testing of valves in the Containment Leak Rate Program (0-TI-360) which were being tested to verify the combined leakage rate (when tested at 1.1 Pa) is within the limit which ensures suppression pool level is sufficient to keep lines that terminate below the water level for at least 30 days without additional make-up. Water local leak rate testing described in TS bases for SR 3.6.1.3.11 for closed system CIVs is also being eliminated. The closed system CIVs which have a dual function as pressure isolation valves (PIVs) will continue to be leak rate tested in accordance with TS 5.5.6, Inservice Testing (IST) Program, requirements. A historical perspective of BFN's Appendix J Program indicates that components currently being water leakage rate tested were tested to the requirements of a qualified seal system. ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements," defines a qualified seal system as "a system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 Pa for at least 30 days following a DBA." At BFN, the suppression pool level is assured for 30 days during all design basis, post accident modes of operations. For those lines which terminate below the minimum suppression pool water level, the maintained pool inventory serves as a

continuous passive barrier to containment atmospheric leakage. TVA has determined that water leak rate testing of the subject valves is unnecessary to ensure that post accident radiological releases from the containment are minimized consistent with the existing accident analysis. Adequate testing, to ensure valve operability and position indication, is performed in accordance with the IST Program (TS 5.5.6).

2.0 DETAILED DESCRIPTION

A Primary Containment Leakage Test Program is established to provide the leakage rate testing requirements for CIVs as set forth in BFN TS 5.5.12. TS 5.5.12 implements Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," which endorses NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J," and ANSI/ANS 56.8-1994. Both NEI 94-01 and ANSI/ANS 56.8-1994 provide the following definition, "An LLRT is a test performed on Type B and Type C components. An LLRT is not required for the following cases: Primary containment boundaries that do not constitute potential primary containment atmospheric pathways during and following a Design Basis Accident (DBA)." Additional cases where an LLRT is not required are also provided in NEI 94-01 and ANSI/ANS 56.8-1994 but are not applicable to this evaluation. The DBA is further defined in ANSI/ANS 56.8-1994 as "the accident initiated by a single component failure or operator error, as described in the safety analysis of the plant, which results in the maximum primary containment internal peak pressure and in fission product release to the containment atmosphere."

SR 3.6.1.3.11 currently requires the plant to perform water local leak rate testing of CIVs, not associated with closed systems, listed in Attachment 3. The leakage of the group of valves that terminate below the suppression pool water level is combined and compared to the limit contained in 0-TI-360, Containment Leak Rate Program. 0-TI-360, Appendix A, establishes a combined leakage rate of less than or equal to 72.79 cubic feet per hour for water local leakage rate tested lines. This leakage is applicable to the group of valves previously designated as potentially impacting suppression pool inventory. The limit is based on the liquid volume of the suppression pool and the design requirement to maintain a 30 day supply of water in the pool without exposing the uppermost valve line to primary containment atmosphere. The evaluation presented below demonstrates that for these valves, the water local leakage rate testing and leakage limit is unnecessary to maintain primary containment leakage within required limits.

The proposed change is to eliminate SR 3.6.1.3.11 and the requirement to perform water leak rate testing on CIVs listed in Attachment 3. Water leak rate testing has been determined to be unnecessary for CIVs on lines that penetrate the suppression pool and terminate below the minimum water level, lines that take suction from the suppression pool water, and for components sealed by water during the post accident period.

The affected valves will continue to be tested in accordance with applicable ASME Code requirements. The CIVs, except those CIVs that are also PIVs, will be reclassified and tested as Category B or Category C valves in accordance with ASME OM Code. PIVs will remain classified as Category A valves. Category B valves are those for which seat leakage in the closed position is inconsequential for fulfillment of the required function(s). Category C valves are those that are self-actuating in response to some

system characteristic, such as pressure (relief valves) or flow direction (check valves) for fulfillment of the required function(s). Currently, the CIVs in question are classified a Category A or A/C valves indicating that seat leakage is limited to a specific maximum amount.

3.0 TECHNICAL EVALUATION

3.1 Background

BFN TS SR 3.6.1.3.11 requires local leak rate testing to be performed on CIVs that serve lines which terminate below the suppression pool minimum water level and CIVs which are water sealed throughout the post accident period. These local leak rate tests are performed with water. SR 3.6.1.3.11 states, "Verify combined leakage through water tested lines that penetrate primary containment are within the limits specified in the Primary Containment Leakage Rate Testing Program."

SR 3.6.1.3.11 Bases states, "Surveillance of water tested lines ensures that sufficient inventory will be available to provide a sealing function for at least 30 days at a pressure of 1.1 Pa. Sufficient inventory ensures there is not a path for leakage of primary containment atmosphere to the environment following a DBA. Leakage from containment isolation valves that terminate below the suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available as described in 10 CFR 50, Appendix J, Option B. Leakage through valves in closed loop seismic class I lines that are considered as extensions of primary containment present no potential for leakage to the environment. Leakage from these valves will be measured, but will be excluded when computing the total leakage. This leakage will be reported as required by the Primary Containment Leakage Rate Testing Program."

The proposed change eliminates local leak rate testing of the components listed in Attachment 3. This change eliminates local leak rate testing of CIVs that terminate below the suppression pool water level that are identified as being "combined" and maintained within the "limit" established in the Primary Containment Leakage Rate Testing Program (TS SR 3.6.1.3.11). Water local leak rate testing described in TS Bases for SR 3.6.1.3.11 for closed system CIVs will also be eliminated as part of implementation of the TS change pending approval. Required leakage rate testing for these valves will be performed per TS 5.5.6 (IST Program) requirements. A historical perspective of BFN's Appendix J Program indicates that components currently being water leakage rate tested in accordance with TS SR 3.6.1.3.11 were originally classified as a qualified seal system. ANSI/ANS 56.8-1994 provides a definition of a qualified seal system as "a system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 Pa for at least 30 days following a DBA." At BFN, the suppression pool level is assured for 30 days during all design basis, post accident modes of operations. For those lines which terminate below the minimum suppression pool water level, the maintained pool inventory serves as a continuous passive barrier to containment atmospheric leakage. Water leak rate testing of the subject valves has been determined to be unnecessary to ensure that post accident radiological releases from the containment are

minimized consistent with the existing accident analysis. Adequate testing, to ensure valve operability and position indication, is performed in accordance with the IST Program (TS 5.5.6). For the closed system CIVs tested in accordance with the Bases for SR 3.6.1.3.11, testing will also be in accordance with TS 5.5.2, Primary Coolant Sources Outside Containment, Program. For CIVs which are also PIVs, leak rate testing will continue to be performed in accordance with TS 5.5.6.

3.2 ASSESSMENT

The Code of Federal Regulations, 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," establishes requirements for containment leakage tests for all operating licenses for water-cooled power reactors. Three tests are specified in the regulation: Type A (integrated leakage rate), Type B (penetration local leakage rate), and Type C (CIV local leakage rate), the last of which is the focus of the proposed change. The regulations further specify types of CIVs which are required to be Type C tested. BFN maintains programs and procedures to perform Type C tests and will continue to perform these tests on CIVs; however, not all CIVs currently included in the testing program will continue to be water leak rate tested. BFN is seeking to eliminate specific CIVs from the current testing requirement. The following analysis provides basis for the proposed elimination. With the implementation of this local leakage rate testing elimination at BFN, the design basis for primary containment isolation will not be affected by the proposed change. The change does not require an exemption from the existing applicable codes, but rather allows implementation in accordance with the codes.

BFN TS 5.5.12 implements RG 1.163, which allows for the performance based testing Option B of 10 CFR 50 Appendix J. RG 1.163 endorses NEI 94-01 and ANSI/ANS 56.8-1994 with a few exceptions. NEI 94-01 and ANSI/ANS 56.8-1994 provide the following: "Primary containment boundaries not requiring Type B or Type C testing include (1) boundaries that do not constitute potential primary containment atmospheric pathways during and following a DBA." The DBA is defined in ANSI/ANS 56.8-1994, which states, "The accident initiated by a single component failure or operator error, as described in the safety analyses of the plant, which results in the maximum primary containment internal peak pressure and in fission product release to the containment atmosphere." Valves considered for removal from water leak rate testing are components that do not constitute potential primary containment atmospheric pathways during and following a DBA. A survey of the nuclear industry indicates that most licensees have excluded these components from their Appendix J Type C local leak rate testing programs.

Based on the above discussion, if a component can be shown to not constitute a potential primary containment atmospheric pathway during and following a DBA (as defined above), then Appendix J does not require local leak rate testing of the component.

TS 5.5.6 requires an IST Program to be established in accordance with applicable ASME Codes. This program is implemented at BFN by procedure 0-TI-362. 0-TI-362 references the ASME OM Code, 1995 Edition through 1996

Addenda, which provides the following guidance regarding containment isolation valves: "Containment isolation valves with a leakage rate requirement based on an Appendix J Program commitment shall be tested in accordance with the 10 CFR 50 Appendix J Program. Containment isolation valves with a leakage requirement based on other functions shall be tested in accordance with paragraph ISTC 4.3.3. Examples of these other functions are reactor coolant system pressure isolation valves and certain Owner-defined system functions such as inventory preservation, system protection, or flooding protection." Based on the above statement, local leak rate testing is required for components if : 1) 10 CFR 50 Appendix J Program requires the testing; 2) the component is a Pressure Isolation Valve (PIVs); or 3) the owner has specified an individual leak rate for the component.

The components being considered for water local leak rate testing elimination are components that are not required to be local leak rate tested per Appendix J because they do not constitute a potential atmospheric leakage pathway during and following a DBA. The plant will continue to leak test identified PIVs of closed loop systems in accordance with TS 5.5.6 (IST Program). There are no components in Attachment 3 with owner specified leak rates.

NUREG 0800 Section 6.2.6, "Containment Leakage Testing," was also reviewed. This document references guidance contained in NEI 94-01 and ANSI/ANS 56.8-1994. Case 1 is Containment boundaries that do not constitute potential containment atmospheric leakage pathways during and following a design-basis loss-of-coolant accident (DB LOCA). Additionally, page 6.2.6-6 contains the following guidance, "Examples of Case 1 are lines that terminate below the minimum post-accident water level of the suppression pool in a BWR or the recirculation sump in a PWR." Case 2 is Containment boundaries sealed with a qualified seal system. Additional guidance is provided for Case 2 qualified seal systems which states, "A qualified seal system as defined in ANSI/ANS 56.8-1994 is a system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 Pa, for at least 30 days following the DB LOCA. The staff's position is that the analysis of the sealing capability includes the assumption of the most limiting single failure of any active component. Also, unless there is a virtually unlimited supply of sealing liquid (such as from a suppression pool or recirculation sump), limits for liquid leakage should be assigned to these valves based on analysis and included in the plant technical specifications...." This guidance indicates that, when the suppression pool provides the supply of seal liquid, limits for liquid leakage for the associated CIVs need not be assigned. At BFN, the suppression pool level is assured for 30 days during all design basis, post accident modes of operations. For those lines which terminate below the minimum suppression pool water level, the maintained pool inventory serves as a continuous passive barrier to containment atmospheric leakage. In addition, a review of the BFN TSs, Final Safety Analysis Report, and other licensing documents did not identify any valve specific leakage limits for those CIVs in lines connected to the suppression pool.

The acceptability of the proposed changes for CIVs on lines that terminate below the minimum suppression pool water level is based on maintaining the existing barriers to primary containment leakage and ensuring that the suppression pool level is assured for 30 days during all design basis, post-accident modes of

operation. These two objectives are related, in that, the suppression pool inventory creates a passive barrier to primary containment atmospheric leakage for penetrations which are located below the minimum water level of the pool. The suppression pool is designed and operated so that it is filled with water in accordance with TS 3.6.2.2, "Suppression Pool Water Level," and the associated bases. As such, the supply of water in the suppression pool is assured for 30 days during all design basis, post-accident modes of operation. As an additional consideration, the capability does exist to make-up water to the suppression pool as necessary. Emergency Operating Instruction (EOI) Appendix 18 provides the necessary instructions for suppression pool make-up if needed. Thus, the level of the suppression pool is ensured, independent of the current CIV water local leak rate testing requirement.

Water local leak rate testing has historically been performed on valves associated with lines that connect to the suppression pool. The acceptance criterion for combined leakage from these penetrations is 72.79 cfh. The limit is based on the liquid volume of the suppression pool to maintain a 30 day supply of water in the pool without exposing the uppermost valve line to primary containment atmosphere. This acceptance criteria is conservative as it does not take into account additional water inputs to the suppression pool resulting from the design basis accidents or suppression pool makeup capability. Historically, the test leakage has been within the acceptance criteria. The technical basis for not water local leak rate testing the CIVs is the fact that existing barriers to primary containment leakage will remain unchanged and the suppression pool level is assured for 30 days during all design basis, post accident modes of operations. For these lines which terminate below the minimum suppression pool water level, the maintained pool inventory serves as a continuous passive barrier to containment atmospheric leakage.

The TS change does not alter the configuration of the subject CIVs and their associated systems. The valves will continue to be classified as CIVs and will continue to be tested and maintained to ensure their operability. The subject valves are all isolation valves associated with lines that penetrate the primary containment.

For the closed system valves, the redundant isolation boundary for each of the affected valves is the closed system associated with the valve. The closed system piping meets TVA BFN's design requirements, i.e., seismic class I.

The integrity of the closed system piping is verified via 10 CFR 50 Appendix J Type A test and is monitored and controlled via TS 5.5.2. TS 5.5.2 establishes a program to monitor and control leakage from systems located outside containment that could contain highly radioactive fluids during a serious transient or accident. This program is applied to the ECCS and the RCIC systems affected by the proposed changes and ensures that leakage into secondary containment via packing, flanges, seals, etc. is controlled. Leakage from these systems has been found to be very low, and well below the 20 gpm maximum limit established for these systems. The proposed change is not expected to contribute to higher levels of system leakage. Normal operational monitoring of suppression pool level, operator rounds, housekeeping inspections, and system

pressure testing further ensure external leakage is identified and minimized while suppression pool level is being maintained.

The subject valves may be open, or change state, post-accident to support the design function of their associated ECCS systems (High Pressure Coolant Injection (HPCI), Core Spray (CS), Residual Heat Removal (RHR)), or Reactor Core Isolation Cooling (RCIC) System or RHR Sampling using the Post Accident Sampling System (PASS). The subject valves function as system valves during the periods when they are open or in an intermediate state, not as containment isolation valves. Reliance is placed on the suppression pool seal and the closed system piping to maintain the barrier between primary and secondary containment atmospheres. Therefore, with the valve configuration and closed systems configuration unaffected by the proposed change, the existing barriers to primary containment atmospheric leakage are maintained, so long as the suppression pool level is ensured.

A review of the postulated effect of design basis events on primary containment isolation was performed. While several design basis events are postulated to result in conditions which could cause primary containment isolation, the design basis for primary containment is the LOCA (up to and including the DBA). The containment structure, including access opening, penetrations, and the containment heat removal system, is designed so that the containment structure and its internal compartments can withstand, without exceeding the design leakage rate (2.0% per day), the peak accident pressure and temperature that could occur during any postulated LOCA. Appendix J testing is performed to ensure that the primary containment structure maintains its design basis leakage characteristics, which in turn ensures compliance with the radiological dose guidance levels identified in 10 CFR 50.67. The proposed change was evaluated in light of the design basis of the primary containment under LOCA conditions. The proposed change does not impact offsite dose since the penetrations do not represent containment atmospheric pathways during the DBA that produces peak containment pressure.

For the purpose of primary containment isolation, a single active failure of the CIV or a passive failure of the closed system were considered within the limits of the existing licensing basis. A pipe rupture of seismically qualified ECCS piping does not have to be assumed concurrent with LOCA, unless it is a consequence of the LOCA (reference FSAR Section 14.6.3.1.e) which states, "A seismic event is neither postulated to occur concurrently with the LOCA nor as an initiator of the pipe break."

Consideration of consequential failures can be eliminated, since LOCAs inside containment are separated from the ECCS piping by the containment structure. Consequential failures of the ECCS piping from LOCAs outside containment are outside Appendix J design considerations, although they are adequately addressed through the redundancy and separation of the ECCS design. A single active failure of the CIV, under the LOCA condition, can be accommodated since the closed and filled system piping and the suppression pool water inventory remain as the leakage barriers. The ECCS passive failure criterion requires consideration of system leak, but not pipe breaks, beyond the initiating LOCA. Pipe leakage, equivalent to the leakage from a valve or pump

seal failure, should be considered at 24 hours or greater post LOCA. The capability to make up inventory to the suppression pool is adequate to ensure that postulated seat leakage and pipe leakage does not result in a condition that jeopardizes pool level. Make up capability exists to the suppression pool as previously indicated in the EOIs.

Therefore, postulated design basis events do not compromise the ability to maintain primary containment isolation under the proposed change.

4.0 REGULATORY EVALUATION

4.1 PRECEDENT

A search of NRC actions on TS changes revealed that NRC has previously approved similar changes for the following plants:

1. Susquehanna SER for Amendment Nos. 149 and 119 dated August 15, 1995
2. Hatch SER, dated October 30, 1986
3. Monticello SER dated June 3, 1984
4. Fitzpatrick SER dated November 14, 1989 indicates similar valves are not local leak rate tested
5. Duane Arnold Exemption, dated April 8, 1987
6. Peach Bottom Exemption, dated October 23, 1991
7. Brunswick Exemption, dated May 12, 1987

Note, in some cases above the utility submitted an exemption similar to this proposed change being requested and NRC stated that no exemption was necessary.

4.2 SIGNIFICANT HAZARDS CONSIDERATIONS

This analysis addresses the proposed change to the BFN TS SR 3.6.1.3.11 that will implement a change to the scope of water leak rate testing of CIVs. The proposed change eliminates the requirement to water leak rate test specific CIVs for systems that penetrate the suppression pool and terminate below the minimum water level and those that are associated with closed systems.

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

   This proposal does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change to the scope of water leak rate testing for the subject valves does not affect the probability of the design basis accidents. The valves will continue to be maintained in an operable state, and in their current design configuration. There is no correlation between the scope of the water leak rate testing and accident probability.

TVA reviewed the postulated consequences of design basis events on primary containment isolation under the proposed change. The primary containment structure, including access openings, penetrations and the containment heat removal system, is designed so that the containment structure and its internal compartments can withstand, without exceeding the design leakage rate (2.0% per day), the peak accident pressure and temperature that could occur during any postulated LOCA.

For the purposes of considering the consequences of LOCAs under the proposed change, a single active failure of a CIV or a passive failure of the closed system were reviewed, within the limits of the existing licensing basis. Under the existing licensing basis, a pipe rupture of seismically qualified ECCS piping does not have to be assumed concurrent with the LOCA, except if it is a consequence of the LOCA. Consequential failures can be eliminated, since a LOCA inside containment is separated from the ECCS piping by the containment structure. Consequential failures of the ECCS piping from LOCA's outside containment are outside the Appendix J design considerations, although they are adequately addressed through the redundancy and separation of the ECCS design. A single active failure of the CIV, under the LOCA condition, can be accommodated since the closed and filled system piping and the suppression pool water inventory remain as the leakage barriers. The ECCS passive failure criterion does require consideration of system leaks, but not pipe breaks, beyond the initiating LOCA. Pipe leakage, equivalent to the leakage from a valve or pump seal failure, should be considered at 24 hours or greater post-LOCA. The capability to make-up inventory to the suppression pool is adequate to ensure that postulated seat leakage and pipe leakage does not result in a condition that jeopardizes pool level. Make-up capability exists to the suppression pool. Actions to make-up to the suppression pool are delineated in Emergency Operating Instructions.

Therefore, the proposal to eliminate the subject water leak rate tests does not involve a significant 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?

   This proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The acceptability of the proposed change to the scope of water leak rate testing for the subject valves is based on maintaining the existing barriers to primary containment leakage, and ensuring that the suppression pool level is assured for 30 days during all design basis, post-accident modes of operation. By meeting these dual objectives, the plant response to the design basis events will be unchanged, and no new accident scenarios will be encountered. These two objectives are related, in that, the suppression pool inventory creates a passive barrier to primary containment

atmospheric leakage for valves associated with penetrations which are located below the minimum water level of the pool.

The proposed Technical Specification change does not alter the configuration of the subject containment isolation valves or their associated systems. The valves will continue to be tested and maintained to ensure their operability. The subject valves are all isolation valves associated with lines that penetrate the primary containment. For closed system valves, the redundant isolation boundary for each of the affected valves is the closed system associated with the valve. The closed system piping is verified via a 10CFR50 Appendix J Type A test. The integrity of the closed systems is also monitored and controlled via Technical Specification 5.5.2, "Primary Coolant Sources Outside Containment."

The subject valves may be open, or change state, post-accident to support the design function of their associated ECCS systems (HPCI, Core Spray, RHR), RCIC or RHR Sampling using the Post Accident Sampling System. The subject valves function as system valves during the periods when they are open or in an intermediate state, not as containment isolation valves. Reliance is placed on the suppression pool seal and the closed system piping to maintain the barrier between primary and secondary containment atmospheres.

Therefore, with the valve configuration and closed systems configuration unaffected by the proposed change, the existing barriers to primary containment atmospheric leakage are maintained, so long as the suppression pool level is ensured.

The suppression pool is designed and operated so that it is filled with water in accordance with Technical Specifications 3.6.2.2, "Suppression Pool Water Level," and the associated Bases. As such, the supply of water in the suppression pool is assured for 30 days during all design basis, post-accident modes of operation. Water leak rate testing has historically been performed on valves associated with lines that connect to the suppression pool. The acceptance criteria for combined leakage from these penetrations is 72.79 cfh. This leakage rate is at a level which ensures the 30 day post-accident suppression pool level.

As mentioned above, the integrity of the closed system piping is verified via a 10CFR50 Appendix J Type A test and is monitored and controlled via Technical Specification 5.5.2. TS 5.5.2 establishes a program to monitor and control leakage from systems located outside containment that could contain highly radioactive fluids during a serious transient or accident. This program applies to the ECCS and RCIC systems affected by the proposed change and ensures that leakage into secondary containment via packing, flanges, seals, etc., is controlled. Leakage from these systems has been found to be very low, and well below the 20 gpm limit established for these systems. The proposed change is not expected to contribute to higher levels of system leakage. Normal operational monitoring of suppression pool level, operator rounds, housekeeping inspections, and system

pressure testing further ensure external leakage is identified and minimized while suppression pool level is being maintained.

A review of water leak rate test data for the subject CIVs showed that the valves have had leakage rates within the acceptance criteria. Testing of the valves in accordance with ASME Code requirements ensure valve operability.

Therefore, leakage past the CIVs is expected to be low and in keeping with the design basis for the suppression pool. However, the capability does exist to make-up water to the suppression pool if necessary. Existing Emergency Operating Instructions require actions if suppression pool level is less than the required level. Thus, the level of the suppression pool is ensured, independent of the current CIV water leak rate testing requirement.

The proposed change to the scope of water leak rate testing for the subject valves maintains the existing barriers to primary containment leakage, and ensures that the suppression pool level is assured for 30 days during all design basis, post-accident modes of operation. Therefore, the plant response to the design basis events is unchanged, and the proposal does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

This change does not involve a significant reduction in a margin of safety.

As discussed in the responses to questions 1 and 2, the proposed change does not alter the plant response to existing accident scenarios, and does not introduce new or different scenarios. So the margin of safety from a design basis accident standpoint is maintained.

Historically, the leakage rate through the subject valves has been determined in accordance with TS SR 3.6.1.3.11. This leakage rate has always been within the acceptance criteria. Quantifying leakage past the CIVs has been used to ensure that the suppression pool level is assured for 30 days post-accident. Under the proposed change, this leakage rate will not be quantified. In addition, closed system leakage is monitored and controlled by an existing Technical Specification program. Closed system leakage has been found to be very low on each of the units, and is currently well below the 20 gpm allowable. Therefore, leakage past the CIVs is expected to be low and in keeping with the design basis for the suppression pool. However, the capability does exist, and is proceduralized, to make-up water to the suppression pool if necessary. Thus the current capability to maintain adequate suppression pool level for 30 days post-accident is assured under the proposed change.

Therefore the proposed change to the scope of water leak rate testing for the subject valves does not involve a significant reduction in a margin of safety.

4.3 CONCLUSION

The proposed elimination of TS 3.6.1.3.11 is acceptable based on the following:

1. The lines associated with the valves penetrate the suppression pool below the minimum water level of the suppression pool, thus the suppression pool inventory creates a passive barrier to primary containment atmosphere leakage through the subject penetrations.
2. The closed and filled system piping provides a barrier to prevent potential CIV seat leakage from entering secondary containment. BFN has committed to testing the valves (which will no longer be water local leakage rate tested) in accordance with the applicable inservice testing requirements.
3. The current code requirements of ANSI/ANS 56.8-1994 and NEI 94-01 specifically allow the elimination of local leak rate testing of these components.
4. System level leakage integrity for the closed systems is maintained by an existing TS required program which monitors and controls leakage into the Secondary Containment.
5. Design basis postulated failures do not compromise the ability to maintain suppression pool level for 30 days post-accident.

5.0 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 effluents 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)(10). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

ATTACHMENT 1

Browns Ferry Nuclear Plant (BFN)
Units 1, 2, and 3

Technical Specifications (TS) Change 465

Revision of Technical Specifications to Eliminate Unnecessary Water Local Leak Rate Tests for Units 1, 2, and 3.

Regulatory Commitments

None

ATTACHMENT 2

Browns Ferry Nuclear Plant (BFN)
Units 1, 2 AND 3

Technical Specifications (TS) Change 465

Revision of Technical Specifications to Eliminate Unnecessary Water Local Leak Rate Tests for Units 1, 2, and 3.

Proposed Technical Specifications Changes (mark-up)

The following pages have been revised.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on a simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is ≤ 100 scfh and that the combined leakage rate for all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
~~SR 3.6.1.3.11~~	~~Verify combined leakage through water tested lines that penetrate primary containment are within the limits specified in the Primary Containment Leakage Rate Testing Program.~~	~~In accordance with the Primary Containment Leakage Rate Testing Program~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on a simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is ≤ 100 scfh and that the combined leakage rate for all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
~~SR 3.6.1.3.11~~	~~Verify combined leakage through water tested lines that penetrate primary containment are within the limits specified in the Primary Containment Leakage Rate Testing Program.~~	~~In accordance with the Primary Containment Leakage Rate Testing Program~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on a simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is ≤ 100 scfh and that the combined leakage rate for all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
~~SR 3.6.1.3.11~~	~~Verify combined leakage through water tested lines that penetrate primary containment are within the limits specified in the Primary Containment Leakage Rate Testing Program.~~	~~In accordance with the Primary Containment Leakage Rate Testing Program~~

ATTACHMENT 3

Browns Ferry Nuclear Plant (BFN)
Units 1, 2 AND 3

Technical Specifications (TS) Change 465

Revision of Technical Specifications to Eliminate Unnecessary Water Local Leak Rate Tests for Units 1, 2, and 3.

AFFECTED VALVE LIST

SR 3.6.1.3.11 Applicable Components

Valve Number	Valve Function	Comment
System 43 - Post Accident Sampling System (PASS) (RHR Sample)		
FSV-43-50	PASS Liquid Sample	Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
FSV-43-56	PASS Liquid Sample	Water sealed by RHR closed loop outside containment/ not a potential containment atmospheric boundary during and following a DBA
System 71 - Reactor Core Isolation Cooling (RCIC)		
SHV-71-32	RCIC Vacuum Pump Discharge	Enters suppression pool above water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
CKV-71-592	RCIC Vacuum Pump Discharge	Enters suppression pool above water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
CKV-71-547	RCIC Pump Minimum Flow Bypass	Enters suppression pool above water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
FCV-71-34	RCIC Pump Minimum Flow Bypass	Enters suppression pool above water line and terminates below the minimum suppression pool level /Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
FCV-71-17	RCIC Pump Suction	Enters suppression pool below water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
FCV-71-18	RCIC Pump Suction	Enters suppression pool below water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA

SR 3.6.1.3.11 Applicable Components

Valve Number	Valve Function	Comment
System 73 - High Pressure Coolant Injection (HPCI)		
ISV-73-24	HPCI Turbine Exhaust Drain	Enters suppression pool above water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
CKV-73-609	HPCI Turbine Exhaust Drain	Enters suppression pool above water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
FCV-73-30	HPCI Miniflow Bypass	Enters suppression pool above water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
CKV-73-559	HPCI Miniflow Bypass	Enters suppression pool above water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
FCV-73-26	HPCI Pump Suction	Enters suppression pool below water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
FCV-73-27	HPCI Pump Suction	Enters suppression pool below water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
System 74 - Residual Heat Removal (RHR)		
FCV-74-53	RHR-LPCI To Reactor	Reactor Coolant System (RCS) CS to RHR Pressure Isolation Valve (PIV). Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA. To be tested in accordance with TS 5.5.6; IST requires leak rate testing of PIVs
CKV-74-54	RHR-LPCI To Reactor	RCS to RHR Pressure Isolation Valve (PIV). Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA. To be tested in accordance with TS 5.5.6; IST requires leak rate testing of PIVs
FCV-74-67	RHR-LPCI To Reactor	RCS to RHR Pressure Isolation Valve (PIV). Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA. To be tested in accordance with TS 5.5.6; IST requires leak rate testing of PIVs
CKV-74-68	RHR-LPCI To Reactor	RCS to RHR Pressure Isolation Valve (PIV). Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA. To be tested in accordance with TS 5.5.6; IST requires leak rate testing of PIVs

SR 3.6.1.3.11. Applicable Components

Valve Number	Valve Function	Comment
System 74 - Residual Heat Removal (RHR) (Continued)		
FCV-74-57/58/59	RHR Containment Cooling	Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
FCV-74-71/72/73	RHR Containment Cooling	Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
FCV-74-60/61	RHR Drywell Spray	Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
FCV-74-74/75	RHR Drywell Spray	Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
74-722	Torus Drain	Enters suppression pool below water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
74-722 Flange	Torus Drain	Enters suppression pool below water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
CKV-74-792	Pressure Suppression Chamber (PSC) Head Tank To RHR	Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
CKV-74-804	PSC Head Tank To RHR	Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
CKV-74-803	PSC Head Tank To RHR	Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
CKV-74-802	PSC Head Tank To RHR	Water sealed by RHR closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA

SR 3.6.1.3.11 Applicable Components

System 75 - Core Spray (CS)		
FCV-75-25	Core Spray To Reactor	RCS to CS Pressure Isolation Valve (PIV). Water sealed by CS closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA. To be tested in accordance with TS 5.5.6; IST requires leak rate testing of PIVs
CKV-75-26	Core Spray To Reactor	RCS to CS Pressure Isolation Valve (PIV). Water sealed by CS closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA. To be tested in accordance with TS 5.5.6; IST requires leak rate testing of PIVs
FCV-75-53	Core Spray To Reactor	RCS to CS Pressure Isolation Valve (PIV). Water sealed by CS closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA. To be tested in accordance with TS 5.5.6; IST requires leak rate testing of PIVs
CKV-75-54	Core Spray To Reactor	RCS to CS Pressure Isolation Valve (PIV). Water sealed by CS closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA. To be tested in accordance with TS 5.5.6; IST requires leak rate testing of PIVs
FCV-75-57	Pressure Suppression Chamber Drain	Enters suppression pool below water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
FCV-75-58	Pressure Suppression Chamber Drain	Enters suppression pool below water line and terminates below the minimum suppression pool level / Sealed by torus water / not a potential containment atmospheric boundary during and following a DBA
CKV-75-606	PSC Head Tank To Core Spray	Water sealed by CS closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
CKV-75-607	PSC Head Tank To Core Spray	Water sealed by CS closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
CKV-75-609	PSC Head Tank To Core Spray	Water sealed by CS closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA
CKV-75-610	PSC Head Tank To Core Spray	Water sealed by CS closed loop outside containment / not a potential containment atmospheric boundary during and following a DBA