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

March 9, 2004

TVA-BFN-TS-434

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of) Docket No. 50-259
Tennessee Valley Authority)

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1 - TECHNICAL SPECIFICATION 434 - LOWERING THE ALLOWABLE VALUE FOR REACTOR VESSEL WATER LEVEL - LOW LEVEL 3

Pursuant to 10 CFR 50.90, Tennessee Valley Authority (TVA) is submitting a request for an amendment to license DPR-33 for BFN Unit 1. The proposed amendment will reduce the Allowable Value used for Reactor Vessel Water Level - Low, Level 3 for several instrument functions.

The primary purpose of this proposed Technical Specification change is to reduce the likelihood of unnecessary reactor scrams and the resultant engineered safety feature actuations by increasing the operating range between the normal reactor vessel water level and Level 3 trip functions. The increased range will provide additional time for operators or automatic features to respond to recoverable transients and, thus, may avert unnecessary reactor scrams.

Industry studies have identified low water level scrams as being initiators of a significant number of plant trips. The Boiling Water Reactor Operating Group, Scram Frequency Reduction Committee identified some of these scrams as unnecessary, since the reactor water level would have stabilized above the top of active fuel and recovered to normal level even without the scram. To provide relief from unnecessary scrams, a possible solution is to lower the

instrument Allowable Value at which the scram will occur. The safety analysis in Enclosure 1 shows that the Allowable Value may be lowered without adversely affecting the plant response to postulated transients and accidents. As discussed in Section 3.3 of Enclosure 1, the proposed changes to the Unit 1 Technical Specifications are the same changes as that approved for Units 2 and 3 in Reference 1.

The proposed amendment is necessary to support the restart of Unit 1 and improves the fidelity with Units 2 and 3. In addition, TVA relies on the approval of this proposed change as part of the design assumptions used to justify an extended power uprate for Unit 1 prior to restart. Therefore, TVA requests that the amendment be approved by March 11, 2005.

TVA has determined that there are no significant hazards considerations associated with the proposed amendment and that the amendment 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 attachments to the Alabama State Department of Public Health.

Enclosure 1 provides TVA's evaluation of the proposed amendment. Enclosure 2 provide mark-ups of the proposed change to the Technical Specifications. Enclosure 3 provide draft Technical Specification pages that have been updated to reflect the proposed change.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 9, 2004.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

cc: See page 3

Enclosures:
1. TVA Evaluation of Proposed Amendment
2. Proposed changes to the Technical Specifications (mark-ups)

References:

1. NRC letter, Long to Scalice, dated August 16, 1999, "Browns Ferry Nuclear Amendments Regarding Allowable Value for Reactor Vessel Water Level (TAC Nos. MA5697 AND MA5698)."

Enclosure
cc (Enclosures):
State Health Officer
Alabama State Department of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 303017
Montgomery, Alabama 36130-3017

ENCLOSURE 1

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

TECHNICAL SPECIFICATION CHANGE (TS-434) - LOWERING THE ALLOWABLE VALUE FOR REACTOR VESSEL WATER LEVEL - LOW LEVEL 3

TVA EVALUATION OF PROPOSED AMENDMENT

INDEX

1.0 DESCRIPTION

This letter requests an amendment to license DPR-33 for BFN Unit 1. The proposed amendment will reduce the Allowable Value used for Reactor Vessel Water Level - Low, Level 3 for several instrument functions.

The primary purpose of this proposed Technical Specification change is to reduce the likelihood of unnecessary reactor scrams and the resultant engineered safety feature actuations by increasing the operating range between the normal reactor vessel water level and Level 3 trip functions. The increased range will provide additional time for operators or automatic features to respond to recoverable transients and, thus, may avert unnecessary reactor scrams.

The proposed amendment is necessary to support the restart of Unit 1 and improves the fidelity with Units 2 and 3. In addition, TVA relies on the approval of this proposed change as part of the design assumptions used to justify an extended power uprate for Unit 1 prior to restart. Therefore, TVA requests that the amendment be approved by March 11, 2005.

2.0 PROPOSED AMENDMENT

The proposed change will lower the current Reactor Vessel Water Level - Low, Level 3 Allowable Value in the Unit 1 Technical Specification for several instrument functions. The following specific Technical Specification functions are affected by this proposed change:

- Reactor Protection System (RPS) Actuation (SCRAM)
- Emergency Core Cooling System (ECCS) (including Automatic Depressurization System (ADS) Reactor Vessel Water Level Confirmatory Signal)
- Primary Containment Isolation (including Reactor Water Cleanup [RWCU] System and Shutdown Cooling System Isolation)
- Secondary Containment Isolation
- Control Room Emergency Ventilation (CREV) System Initiation

The proposed changes to the Technical Specifications are listed below. Enclosure 2 contains copies of the appropriate marked-up Technical Specification pages for Unit 1 showing the changes. No changes to the Technical Specification Bases are required.

1. Table 3.3.1.1-1, Reactor Protection System Instrumentation

Function	Allowable Value Current	Allowable Value Proposed
4. Reactor Vessel Water Level - Low, Level 3	≥ 538 inches above vessel zero	≥ 528 inches above vessel zero

2. Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation

Function	Allowable Value Current	Allowable Value Proposed
4. ADS Trip System A		
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)	≥ 544 inches above vessel zero	≥ 528 inches above vessel zero
5. ADS Trip System B		
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)	≥ 544 inches above vessel zero	≥ 528 inches above vessel zero

3. Table 3.3.6.1-1, Primary Containment Isolation Instrumentation

Function	Allowable Value Current	Allowable Value Proposed
2. Primary Containment Isolation		
a. Reactor Vessel Water Level - Low, Level 3	≥ 538 inches above vessel zero	≥ 528 inches above vessel zero
5. Reactor Water Cleanup (RWCU) System Isolation		
h. Reactor Vessel Water Level - Low, Level 3	≥ 538 inches above vessel zero	≥ 528 inches above vessel zero
6. Shutdown Cooling System Isolation		
b. Reactor Vessel Water Level - Low, Level 3	≥ 538 inches above vessel zero	≥ 528 inches above vessel zero

4. Table 3.3.6.2-1, Secondary Containment Isolation Instrumentation

Function	Allowable Value Current	Allowable Value Proposed
1. Reactor Vessel Water Level - Low, Level 3	≥ 538 inches above vessel zero	≥ 528 inches above vessel zero

5. Table 3.3.7.1-1, Control Room Emergency Ventilation System Instrumentation

Function	Allowable Value Current	Allowable Value Proposed
1. Reactor Vessel Water Level - Low, Level 3	≥ 538 inches above vessel zero	≥ 528 inches above vessel zero

3.0 BACKGROUND

Provided in this section is the reason for this proposed change and a description of the modifications required to implement the proposed change. Also included at the end of this section is a comparison of the proposed change, reason for change and technical analysis submitted in support of this proposed amendment with the information provided by TVA and approved by NRC for the Units 2 and 3 license amendments (References 1 and 2).

3.1 Reason for the Proposed Change

During reactor operation, there is approximately 23 inches between the normal reactor water level and the reactor scram initiation point. Plant systems are designed such that the reactor can usually automatically recover from many transients such as a trip of a feedwater system pump. However, in some cases, with this tight water level range, reactor scrams may result that would have been avoidable if plant control systems or operators had slightly more time to take control. In addition, since Boiling Water Reactors operate with a high steam void fraction, water level is sensitive to mild pressure perturbations. Often, the prompt water level drop due to rapid void collapse caused by a manual or automatic scram is large enough to cause a Level 3 trip. This initiates primary and secondary containment isolation, and SGT and CREV system initiations. These system isolations and initiations are an unneeded distraction for the operators responding to scrams.

This proposed Technical Specification change increases the operating range between the normal reactor vessel water level (561 inches above vessel zero) and the Reactor Vessel Level - Low, Level 3 actuation Allowable Value by 10 inches (current value of 538 inches, proposed value of 528 inches). The increased range will provide additional time for operators or plant systems to automatically respond to recoverable transients such as feedwater system malfunctions. With the small increase in water level range, over the course of the reactor operating life, it is expected that several unnecessary scrams will be avoided. This also has a positive effect in that unnecessary challenges to other Engineered Safety Features (ESFs) will likewise be avoided.

In addition to reducing the low reactor water level scram initiation point, several other instrument functions that occur at Level 3 are being lowered to maintain consistency with the low level scram trip setting as well as to provide a similar margin to unnecessary initiation of ESFs. This reduction in the Allowable Value can be achieved without increasing the consequences of events that rely on these instrument functions and without having an adverse effect on plant safety analyses.

3.2 Description of the Proposed Modifications

The proposed change will lower the current Reactor Vessel Water Level - Low, Level 3 Allowable Value in the Unit 1 Technical Specification for several instrument functions. The setpoints for the affected instruments will be adjusted and associated procedures revised. The safety related systems and components that are initiated by a Reactor Vessel Water Level - Low, Level 3 signal will still operate in the same manner as they currently do. There are no changes to component maintenance or testing associated with the proposed Technical Specification change.

3.3 Comparison with previous Technical Specification changes for Unit 2 and 3

TVA has compared the proposed change, reason for change, background information, and technical analysis submitted in support of this proposed amendment with the information provided by TVA and approved by NRC in TS 397 (References 1 and 2) for lowering the Units 2 and 3 Allowable Value for the Reactor Vessel Water Level - Low Level 3 signal. The comparison for each of these areas is provided below:

- The proposed changes to the Unit 1 Technical Specifications are the same changes as that proposed and approved for Units 2 and 3.

- The reason for the Unit 1 Technical Specification change is the same as that which was previously submitted for the Units 2 and 3 Technical Specification change (i.e., reduce the likelihood of unnecessary reactor scrams and the resultant engineered safety feature actuations by increasing the operating range between the normal reactor vessel water level and Level 3 trip functions).

- The background information provided in support of the Unit 1 Technical Specification change incorporates the same elements previously submitted in support of the Units 2 and 3 Technical Specification change.

- The technical analysis submitted for this Unit 1 Technical Specification change incorporates the majority of the same elements which were previously submitted for the Units 2 and 3 Technical Specification change. The Units 2 and 3 submittal contained a qualitative evaluation of the effect of lowering the Level 3 Allowable Value on the Probabilistic Safety Analysis (PSA). A Unit 1 PSA is not currently available. Therefore, the PSA evaluation was based on design similarities between the units.

4.0 TECHNICAL ANALYSIS

4.1 Analytical Limit / Allowable Value Determination

The instrument function Analytical Limit is the value used in the safety analyses to demonstrate acceptable nuclear safety system performance is maintained. The Allowable Value and trip setpoints are then chosen / calculated such that the instrument will function before reaching the Analytical Limit under the worst case environmental / event conditions. Instrument setpoints account for measurable instrument characteristics (e.g., drift, accuracy, repeatability).

The Allowable Value / Setpoint instrument calculations for this proposed change were performed in accordance with the methodology in TVA procedure EEB-TI-28 (Reference 3). This methodology is consistent with NRC Regulatory Guide 1.105 (Reference 4) and has been previously reviewed by the NRC (Reference 5). The same methodology was also used for Technical Specification Change TS-390 to extend the instrument function surveillance frequencies for 24-month fuel cycle operation (Reference 6). The NRC approved TS-390 on November 30, 1998 (Reference 7).

The attached figure illustrates the relationship between the setpoint, the minimum and maximum acceptable Allowable Values [Allowable Value (min) and Allowable Value (max)], and the Analytical Limit for a process that decreases toward the setpoint. To provide operational reliability and to ensure that the instrument will perform its design basis function, the Technical Specification Allowable Value is established within the "Allowable Value Band."

The current Technical Specification Allowable Value is based on an Analytical Limit of 530 inches above vessel zero. In the safety evaluation for this proposed change, a conservatively low Analytical Limit value of 512 inches above vessel zero was used. This 512 inches value is actually below the lower instrument tap located at 517 inches. Since the water level instruments cannot physically measure levels below the instrument tap, the proposed Technical Specification Allowable Values and setpoint calculations are based on an assumed Analytical Limit of 518 inches. This is a conservative approach and provides additional margin in the safety evaluation.

4.2. Safety Analysis

A safety analysis was performed to support lowering the Reactor Vessel Water Level - Low, Level 3 Analytical Limit by 18 inches from the present 530 inches to 512 inches above vessel zero. As discussed above, 512 inches is conservatively lower than the minimum measurable value for this instrumentation. As also discussed in Section 2.0, several specific Technical Specification functions are affected by this proposed change. These functions and references to the associated Updated Final Safety Analysis Report (UFSAR) descriptions of these design functions is provided below:

- RPS Actuation - UFSAR Section 7.2;
- ECCS - UFSAR Sections 6.4 and 6.5;
- Primary Containment Isolation - UFSAR Section 7.3;
- Secondary Containment Isolation - UFSAR Section 5.3; and
- CREVS - UFSAR Section 10.12.

For the RPS actuation function (SCRAM), the following events were evaluated: abnormal operational occurrences, loss-of-coolant accident (LOCA), anticipated transient without scram (ATWS), Appendix R fire event, radiological release, and containment loading and heating. The effects of lowering the corresponding Analytical Limit for the remaining Level 3 instrument functions were also evaluated. The results of the evaluations are summarized below.

1. Method of Analysis

The analysis for LOCA events were performed with the SAFER/GESTR-LOCA model, which is the current licensing basis methodology used for BFN (Reference 8). For ATWS events, abnormal operating occurrences, and Appendix R fire events, radiological release, and containment loading and heating, and the other instrument functions, the engineering analysis reviewed previous analyses to determine any potential impact of a reduced Level 3 Allowable Value.

2. Purpose of Analysis

The analysis was conducted to demonstrate that lowering of the Reactor Vessel Level Low, Level 3 Analytical Limit by 18 inches from the present 530 inches to 512 inches did not affect the licensing safety limits and did not affect the ability of the plant to operate safely and mitigate the consequences of a design basis accident and abnormal operational occurrences.

3. Analysis for Reduced Level 3 RPS and ECCS Actuations

A low water level in the reactor vessel indicates that reactor coolant is being lost through a breach in the nuclear system process barrier or that the supply of reactor feedwater is less than required to maintain normal level due to a system malfunction. Should the water level decrease too far, fuel damage could ultimately occur if the reactor core is uncovered. The purpose of the reactor low scram is to reduce the rate of water inventory loss by shutting down the reactor. Scramming the reactor drastically reduces the steaming rate and allows time for feedwater systems or emergency injection systems to operate to prevent core damage. The setting of the water level scram signal is chosen far enough below normal operating level to avoid spurious scrams, but high enough above the top of active fuel to assure that adequate cooling will be available following the most severe abnormal operating transient including a level decrease.

The following evaluates the effects of the Reactor Vessel Water Level - Low, Level 3 scram function for events in the safety analyses for the plant.

- Abnormal Operational Occurrences

  The abnormal operational occurrences evaluated in the UFSAR for BFN were reviewed with respect to the proposed change. The scenario for each event was examined to determine if a RPS actuation was assumed to occur on low vessel water level. A reduced Level 3 Allowable Value has no effect on the events for which a reactor scram does not occur on low water level.

  The only analyzed abnormal operational occurrence for which a Level 3 water level scram occurs is the total Loss-of-Feedwater (LOFW) event. In a LOFW event, the reactor water level decreases due to loss of feed flow resulting in a low water level scram at Level 3. Reactor level continues to drop until it reaches Level 2 (470 inches above vessel zero), at which point the Reactor Core Isolation Cooling (RCIC) system and High Pressure Cooling Injection (HPCI) system auto-initiate to restore the reactor water level.

  The safety evaluation shows that the RCIC system alone continues to be able to maintain the reactor water level above Level 1 and refill the vessel (as is the case with the existing Allowable Value for the LOFW event). Level 1 is at 398 inches above vessel zero and is above the top of the core.

  Therefore, no unacceptable safety consequences will result for abnormal operational occurrences for the reduced Level 3 Allowable Value and there is no significant impact on the plant response to abnormal operational occurrences.

- Loss-of-Coolant Accident

  Current pipe break analyses (Reference 9) indicate that the limiting LOCA event is a design basis accident (DBA) recirculation suction line break with a battery failure. The DBA LOCA bounds the limiting small break LOCA which is a 0.08 ft^2 reactor recirculation system discharge line break with a battery failure.

For the DBA LOCA, the initial reactor water level is assumed to be the normal reactor water level and the reactor scrams on high drywell pressure at the same time the break occurs. Therefore, there is no impact on the DBA LOCA analysis associated with the reduced Level 3 RPS actuation Allowable Value.

For the limiting (0.08 ft^2) small break LOCA, initial water level is assumed to be at the scram water level Analytical Limit and the reactor has already scrammed due to high drywell pressure at the time the break occurs. Therefore, reducing the Level 3 Analytical Limit only lowers the assumed initial water level for the small break analysis (530 inches versus 512 inches). With this reduced Analytical Limit, the calculated peak clad temperature (PCT) for the small break is also reduced. This reduction in the PCT is directly related to the earlier initiation of ADS on the Reactor Vessel Water Level - Low Low Low, Level 1 signal due to the lower assumed initial water level.

The proposed Technical Specification change also lowers ADS confirmatory signal Level 3 Allowable Value from 544 inches to 528 inches to maintain consistency with the other Level 3 trip functions. This Level 3 signal is a confirmatory low water level signal for ADS initiation, which serves to prevent unnecessary ADS initiation resulting from spurious Level 1 (398 inches) water level actuations or as a result of a break in the Level 1 instrument line. The intended function of this confirmatory signal will still be successfully accomplished even if the Level 3 signal is reduced since the Level 3 signal will occur well prior to Level 1. Therefore, reducing the Level 3 Allowable Value will not affect the ability of ADS to perform its intended function.

Therefore, lowering the Level 3 RPS Allowable Value will not have an adverse affect on reactor performance for postulated LOCA events and no changes in the plant licensing limits are required.

- Anticipated Transient Without Scram

  The four limiting ATWS events for BFN are:

  1) Closure of all Main Steam Line Isolation Valves,
  2) Pressure Regulator Failure to Maximum Steam Demand Flow,
  3) Loss of Normal Feedwater, and
  4) Inadvertent Opening of a Relief Valve.

  These events assume the failure of the reactor scram and instead utilizes the alternate rod insertion, recirculation pump trip, and the standby liquid control system equipment to reduce core thermal power. Therefore, reducing the Level 3 RPS Allowable Value does not affect the ATWS evaluations.

- Appendix R Fire Event Analysis

  The Appendix R fire event analysis for BFN assumes that the reactor is manually scrammed with reactor water level assumed to be at normal operating level. Therefore, reducing the Level 3 RPS actuation Allowable Value does not affect the Appendix R analysis.

- Radiological Release

  The limiting pipe break for radiological releases inside the containment is the DBA LOCA. The DBA LOCA assumes that the reactor scram occurs at time zero due to high drywell pressure with a normal reactor water level. Therefore, reducing the Level 3 RPS Allowable Value has no impact on the radiological release analyses inside the containment for the DBA LOCA analyses.

  The limiting pipe break for radiological releases outside containment is the design basis main steam line break outside the containment. The main steam line break outside the containment assumes a normal initial reactor vessel water level and that the reactor scrams when the main steam isolation valves close on high main steam line flow. Therefore, reducing the Level 3 RPS Allowable Value has no effect on the calculated radiological

releases for the main steam line break outside containment event.

- Containment Loads and Heating

  Containment dynamic loads and main safety relief valve loads associated with the DBA LOCA were also reviewed. These analyses assume the reactor scrams on high drywell pressure. Therefore, the DBA LOCA short-term and long-term containment loads, and drywell/wetwell temperature response for the DBA LOCA are not affected by a reduced Level 3 RPS Allowable Value.

4. Review of Other Level 3 Functions

As listed previously, several other system functions are initiated by a Level 3 water level trip signal. The Allowable Values for these functions are also proposed to be changed to the Level 3 RPS actuation Allowable Value to maintain consistency with current Technical Specification. Impacts on these functions are addressed below.

- Primary Containment Isolation Systems (PCIS) (Including Shutdown Cooling System and RWCU System Isolation)

  A low reactor vessel water level indicates that the capability to cool the fuel may be threatened if level continues to drop. Therefore, valves whose penetrations communicate with the primary containment or the reactor coolant system automatically isolate at Level 3 to limit the potential for loss of reactor coolant and to limit the potential release of fission products. The isolation of primary containment valves at Level 3 supports actions to ensure that onsite and offsite dose limits of 10 CFR 20 and 100 are not exceeded. The Reactor Vessel Water Level - Low, Level 3 isolation function is assumed in the Chapter 14 UFSAR pipe break analyses since these leakage paths are considered isolated post-LOCA.

The Level 3 low water level setting for primary containment isolation was selected to initiate isolation at the earliest indication of a possible breach in the nuclear system process barrier, yet far enough below normal operational levels to avoid spurious isolation. Historically, the containment isolation low level trip and the RPS actuation trip setpoints are set at the same value.

Isolation of the following is initiated on Reactor Vessel Low Water, Level 3.

- Residual Heat Removal (RHR) reactor shutdown cooling supply
- RWCU
- Drywell equipment drain discharge
- Drywell floor drain discharge
- Drywell purge inlet
- Drywell main exhaust
- Suppression chamber exhaust valve bypass
- Suppression chamber purge inlet
- Suppression chamber main exhaust,
- Drywell exhaust valve bypass
- Suppression chamber drain
- RHR- Low Pressure Coolant Injection (LPCI) to reactor (in shutdown mode)
- Drywell make-up
- Suppression chamber make-up
- Exhaust to Standby Gas Treatment
- Drywell radiation monitor
- Drywell control air compressor
- Containment atmosphere monitor
- Drywell differential pressure air compressor
- Traversing incore probes

During postulated accidents, significant radiation releases cannot occur until after the core is uncovered. Since the reduced Level 3 actuation is still approximately 12 feet above the top of the core, the Level 3 PCIS actuation will still occur well before core uncovery. Therefore, a small delay of this isolation signal due to the reduction in Allowable Value will not affect the ability of the containment isolation valves to perform their intended functions. For LOCA events inside containment, a high drywell pressure signal will also initiate primary containment isolation for all the above systems (except RWCU) very early in the event (prior to a Level 3 water level trip).

The shutdown cooling mode of the RHR system is also isolated by the Level 3 water level trip for a malfunction of the RHR which results in a reactor coolant inventory loss. Shutdown cooling is in service only when the reactor is shutdown. Isolation of system will also cause any operating RHR pumps to trip on loss of suction path. These automatic actions prevent further coolant loss through the RHR shutdown cooling loop if the water level decrease is being caused by the an RHR system malfunction. The reduction of the Level 3 Allowable Value will not affect the intended function of these isolation valves since the system still isolates at a water level far above the top of core. Also, the emergency mode of the RHR system (LPCI) is not required to function until vessel level has dropped to Level 1. Therefore, reducing the Level 3 Allowable Value has no impact on the ability of the shutdown cooling mode isolation to perform its intended functions.

The RWCU system also isolates on Level 3 water level trip in the event that reactor coolant is being lost though a RWCU system line break. The Level 3 RWCU isolation is not directly analyzed in the UFSAR because the RWCU system line break is bounded by breaks of larger systems (DBA LOCA and main steam line break outside the containment). Therefore, reducing the Level 3 actuation has no impact on the ability of the RWCU isolation valves to perform their intended functions.
Additionally, from an operations perspective, in order to maintain reactor water quality it is

beneficial not to isolate the RWCU system unnecessarily.

The remaining systems which are isolated by the primary containment isolation signal are not required immediately following a loss of water inventory event since they do not directly contribute to the replenishment of the vessel water inventory. Therefore, lowering the water Level 3 Allowable Value for automatic isolation will not impact the ability to replenish inventory. As previously discussed, the release of fission products will not occur until after the core is uncovered. Since the Level 3 actuation will always occur well before core uncovery, the delay of this isolation signal will not affect the ability of the containment isolation valves to perform their intended functions. Also, as noted previously, these valves, except for RWCU, also automatically isolate on high drywell pressure for LOCA events prior to the water level trip. In summary, the primary containment isolation function is not adversely affected by reducing the Level 3 actuation.

- Secondary Containment Isolation

The isolation of the secondary containment and initiation of the SGT system support actions to ensure that any radiological releases to secondary containment do not result in exceeding offsite release limits. The LOCA provides the most severe radiological release and, thus, serves as the bounding design basis accident in determining the post-accident offsite dose. For LOCA events, secondary containment and SGT will actuate on high drywell pressure prior to reaching the Level 3 water level trip; therefore, a reduced Level 3 Allowable Value has no effects on the LOCA event analysis. For other loss of inventory events, as described previously, the Level 3 actuations will always occur well before any core uncovery which could result in potential radiological release. Therefore, the small delay introduced by a change in the Level 3 Allowable Value will not affect the ability of the secondary containment or SGT to perform their intended function.

- Control Room Emergency Ventilation System Initiation

  The CREV system is designed to provide a radiologically controlled environment to ensure the habitability of the control room for all plant conditions. In the event of a Level 3 signal, the CREV system is automatically initiated to pressurize the control room to minimize the consequences of radiological releases to the control room environment. The LOCA provides the most severe radiological release to the primary and secondary containment and, thus, serves as the bounding DBA in determining the control room dose. For LOCA events, the CREV system will actuate on high drywell pressure prior to reaching the Level 3 water level trip. Therefore, a reduced Level 3 Allowable Value has no effects on the LOCA event analysis. For other loss of inventory events, as described previously, the Level 3 actuations will occur well before any core uncovery, which could result in potential radiological release. Therefore, the small delay introduced by a change in the Level 3 actuation will not affect the ability of the CREV system to perform its intended function.

5. Operational Concerns on Reduced Level

The proposed Level 3 Allowable Value is slightly below the level of the steam dryer seal skirt. Long-term reactor operation with water level below the dryer seal could affect the steam separator-dryer performance since additional moisture might be carried over into turbine side equipment. However, plant operators continuously monitor reactor water level and take actions promptly to ensure normal level is maintained. Also, there is a water level alarm at 555 inches (about 6 inches below normal level) which would prompt operators to restore normal level if automatic controllers were not operating properly. Therefore, the potential to operate with water level below the steam dryer seal skirt is not considered a practical concern. This condition would also not be a safety concern since the main and reactor feed pump turbines are not required for safe shutdown of the plant.

6. Effect of Lowering the Level 3 Allowable Value on Probabilistic Risk

There are two minor effects on probabilistic risk, which will be addressed qualitatively. The first and more substantial effect is the reduction (i.e., improvement) in the initiating event frequencies due to the lowering of the Level 3 setpoint. This results from the reduction in number of inadvertent scrams from minor operational transients that are avoided by the lower level Allowable Value. The improvement in initiating event frequency will result in a slight improvement in the core damage frequency and large early release frequency.

The other potential effect on the PSA is a small effect on the timing of operator actions after the scram and isolation functions of the Level 3 set point are completed. The reduction of the Level 3 Allowable Value by 10 inches will result in a small reduction in time between the scram and isolation function, and other follow on actions. This effect is considered insignificant and overshadowed by the risk reduction due to the initiating event frequency changes discussed above.

7. Conclusion

Safety analysis to support lowering the Reactor Water Level 3 Allowable Value were performed for BFN Unit 1. Based on the analysis, it is concluded that lowering the Level 3 Allowable Value to 528 inches above vessel zero is acceptable and has no significant impact on abnormal operational occurrences, LOCA, ATWS, Appendix R fire events, radiological releases, or containment loads and heating. Furthermore, lowering Level 3 will provide additional operating range to the Level 3 RPS actuation during plant operational transients which reduces the probability of undesired reactor scrams and other ESF actuations on low reactor water level. Therefore, it concluded that the proposed change has a beneficial effect on plant operations and safety.

5.0 REGULATORY SAFETY ANALYSIS

The Tennessee Valley Authority (TVA) is submitting an amendment request to license DPR-33 for the Browns Ferry Nuclear Plant (BFN) Unit 1.

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 Reactor Vessel Water Level - Low, Level 3 functions are in response to water level transients and are not involved in the initiation of accidents or transients. Therefore, reducing the BFN Unit 1 Level 3 Allowable Value does not increase the probability of an accident previously evaluated.

Additionally, the results of the safety evaluation associated with the lowering of the Level 3 Allowable Value concludes that the previously evaluated transient and accident consequences are not significantly affected by the change. Therefore, the proposed amendment 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?

Response: No

The proposed amendment to lower the BFN Unit 1 Reactor Vessel Water Level - Low, Level 3 Allowable Value does not involve a hardware change and the purpose of the Level 3 function is not affected. The Level 3 functions will continue to fulfill their design objective. The proposed changes do not create the possibility of any new failure mechanisms. No new external threats or release pathways are created. Therefore, reduction of the Allowable Value does not result in the possibility of a new or different kind of accident.

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

Response: No

The results of the safety evaluation associated with the reducing the BFN Unit 1 Reactor Vessel Water Level - Low, Level 3 Allowable Value concluded that transient and accident consequences remain within the required acceptance criteria. Therefore, the margin of safety is not reduced for any event evaluated.

Based on the above, TVA concludes that the proposed amendments present 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

The safety analysis provided above evaluated the reduction in the BFN Unit 1 Reactor Vessel Water Level - Low, Level 3 Allowable Value in the mitigation of (a) abnormal operational occurrences, (b) loss of coolant accidents, (c) anticipated operational occurrences, (d) anticipated transients without scram, (e) Appendix R events (fires) and (f) other events involving a potential radiological release. Compliance with the following requirements is not changed:

- 10 CFR 50.46 (Acceptance Criteria For Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors) and Appendix K (ECCS Evaluation Models);
- General Design Criterion 19 (Control Room);
- 10 CFR 50.62 (Requirements for reduction of risk from anticipated transients without scram [ATWS] events for light-water-cooled nuclear power plants);
- 10 CFR 50.48 (Fire Protection);
- 10 CFR Part 20 (Standards for protection against radiation);
- 10 CFR Part 100 (Reactor site criteria)

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 50.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. TVA letter, T.E. Abney to NRC, dated June 3, 1999, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specifications Change (TS) No. 397 - Request for License Amendment to Lower the Allowable Value for Reactor Vessel Water Level - Low, Level 3."

2. NRC letter, Long to Scalice, dated August 16, 1999, "Browns Ferry Nuclear Amendments Regarding Allowable Value for Reactor Vessel Water Level (TAC Nos. MA5697 AND MA5698)."

3. EEB-TI-28, "Setpoint Calculations," Branch Technical Instruction, Revision 2, Tennessee Valley Authority, October 6, 1992.

4. NRC Regulatory Guide 1.105, "Instrument Setpoints for Safety-Related Systems," Revision 2, February 1986.

5. NRC letter to TVA, dated May 8, 1989, "Notice of Violation (NRC Inspection Report Nos. 50-259/89-06, 50-260/89-06, and 50-296/89-06)."

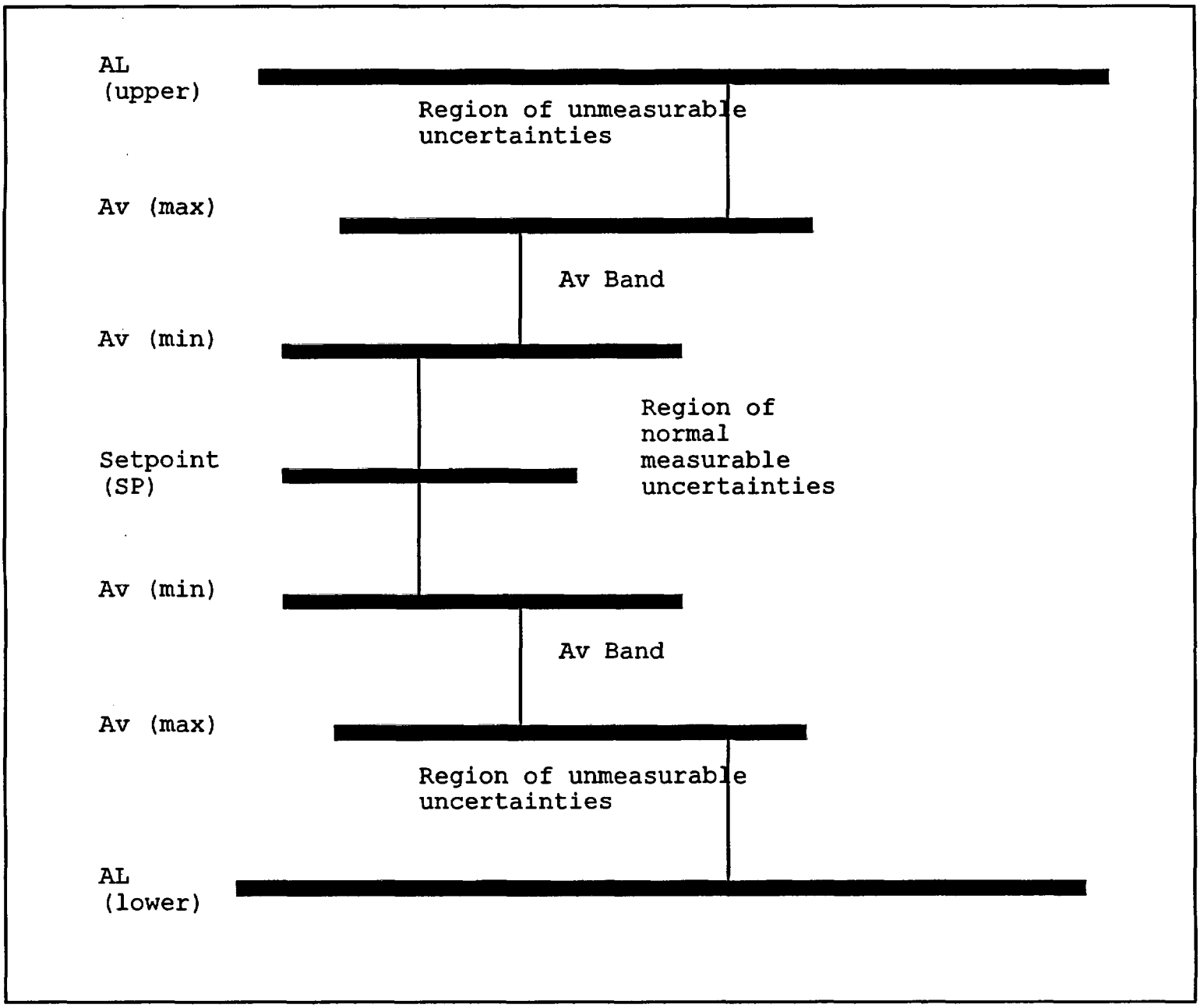
6. TVA letter to NRC, dated August 14, 1998, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - TS Change TS-390 Supplement 1 - Request for License Amendment to Support 24-Month Fuel Cycles."

7. NRC letter to TVA, dated November 30, 1998, "Issuance of Amendments - Browns Ferry Nuclear Plants Units 1, 2, and 3 (TAC Nos. MA2081, MA2082, and MA2083)."

8. General Electric SAFER/GESTR-LOCA, Loss of Coolant Analysis, Browns Ferry Units 1, 2, and 3, NEDC-32484P, Rev. 5, January 2002.

9. General Electric, "Browns Ferry Nuclear Plant Units 1, 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDC-32484P. Revision 2, December 1997.

Figure: Instrument Value Relationships



ENCLOSURE 2

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

TECHNICAL SPECIFICATION CHANGE (TS-434) -
LOWERING THE ALLOWABLE VALUE FOR REACTOR VESSEL WATER
LEVEL - LOW LEVEL 3

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

AFFECTED PAGE LIST

3.3-7
3.3-46
3.3-58
3.3-60
3.3-64
3.3-69

MARKED PAGES

e attached.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Downscale	1	2	F	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	≥ 3% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

Deleted: 538

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 528 inches above vessel zero
e. Core Spray Pump Discharge Pressure - High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure - High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds
5. ADS Trip System B					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure - High	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 528 inches above vessel zero

Deleted: 544

Deleted: 544

(continued)

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 398 inches above vessel zero
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 825 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140% rated steam flow
d. Main Steam Tunnel Temperature - High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ ~~828~~ inches above vessel zero
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure - Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

Deleted: 538

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Main Steam Valve Vault Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 201°F
b. Pipe Trench Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 135°F
c. Pump Room A Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
d. Pump Room B Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
e. Heat Exchanger Room Area (West Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 143°F
f. Heat Exchanger Room Area (East Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
g. SLC System Initiation	1,2	1[a]	H	SR 3.3.6.1.6	NA
h. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 115 psig
b. Reactor Vessel Water Level - Low, Level 3	3,4,5	2[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
c. Drywell Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig

Deleted: 538

Deleted: 538

(a) One SLC System Initiation signal provides logic input to close both RWCU valves.

(b) Only one channel per trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ 538 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3, (a)(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr

Deleted: 538

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3,(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ ~~528~~ inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3 (a),(b)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a),(b)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
5. Control Room Air Supply Duct Radiation - High	1,2,3, (a),(b)	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 270 cpm above background

Deleted: 538

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.

ENCLOSURE 3

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1

TECHNICAL SPECIFICATION CHANGE (TS-434) -
LOWERING THE ALLOWABLE VALUE FOR REACTOR VESSEL WATER
LEVEL - LOW LEVEL 3

PROPOSED TECHNICAL SPECIFICATION CHANGES
(REVISED PAGES)

I. AFFECTED PAGE LIST

3.3-7
3.3-46
3.3-58
3.3-60
3.3-64
3.3-69

II. UPDATED PAGES

See attached.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Downscale	1	2	F	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	≥ 3% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 528 inches above vessel zero
e. Core Spray Pump Discharge Pressure - High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure - High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds
5. ADS Trip System B					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure - High	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds
d. Reactor Vessel Water Level - Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 528 inches above vessel zero

(continued)

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 398 inches above vessel zero
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 825 psig
c. Main Steam Line Flow - High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140% rated steam flow
d. Main Steam Tunnel Temperature - High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 528 inches above vessel zero
b. Drywell Pressure - High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure - Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Main Steam Valve Vault Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 201°F
b. Pipe Trench Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 135°F
c. Pump Room A Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
d. Pump Room B Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
e. Heat Exchanger Room Area (West Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 143°F
f. Heat Exchanger Room Area (East Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
g. SLC System Initiation	1,2	1[a]	H	SR 3.3.6.1.6	NA
h. Reactor Vessel Water Level - Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 528 inches above vessel zero
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure - High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 115 psig
b. Reactor Vessel Water Level - Low, Level 3	3,4,5	2[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 528 inches above vessel zero
c. Drywell Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig

(a) One SLC System Initiation signal provides logic input to close both RWCU valves.

(b) Only one channel per trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3, (a)(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)(b)	1	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 100 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3,(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3 (a),(b)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a),(b)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
5. Control Room Air Supply Duct Radiation - High	1,2,3, (a),(b)	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 270 cpm above background

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.