November 15, 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 SPECIFICATIONS (TS) CHANGE 434 – RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LOWERING THE ALLOWABLE VALUE FOR REACTOR VESSEL WATER LEVEL - LOW LEVEL 3

This letter provides TVA's responses to the NRC request for additional information (Reference 1) regarding proposed Technical Specification 434.

On March 9, 2004 (Reference 2), TVA requested a TS change (TS 434) to reduce the Allowable Value used for Reactor Vessel Water Level - Low, Level 3 for several instrument functions. NRC requested additional information to support the review of the submittal. The NRC requests and TVA's responses are enclosed.

TVA has determined that the provided information does not affect the no significant hazards considerations associated with the proposed amendments and Technical Specification changes. The proposed amendments and Technical Specification changes still qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

If you have any questions about this submittal, please contact me at (256) 729-2636.

Sincerely,

ORIGINAL SIGNED BY:

T. E. Abney
Manager of Licensing
and Industry Affairs

References:

1. NRC letter, K.N. Jabbour to Karl W. Singer, dated September 13, 2004, "Browns Ferry Nuclear Plant, Unit 1 – Request for Additional Information Regarding the Allowable Value for Reactor Vessel Water Level (TAC No. MC2305)."

2. TVA letter, T.E. Abney to NRC, dated March 9, 2004, "Browns Ferry Nuclear Plant (BFN) Unit 1 – Technical Specification 434 – Lowering the Allowable Value for Reactor Vessel Water Level - Low Level 3."

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TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT, UNIT 1
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
PROPOSED TECHNICAL SPECIFICATION CHANGE 434
ALLOWABLE VALUE FOR REACTOR VESSEL WATER LEVEL SETPOINTS

NRC REQUEST

1. Please provide a copy of the document that calculates the new instrument allowable value for Reactor Vessel Water Level - Low (Level 3) function. Discuss the instrument setpoint methodology used to calculate the allowable values.

TVA RESPONSE

As discussed in TVA's May 6, 2004 response[1] to a request for additional information on another proposed Technical Specification change, the primary instrument setpoint methodology used at TVA is based on Method 3 of ISA S67.04.02. As evidenced below, TVA's method for performing setpoint calculations has been reviewed and approved by NRC.

- Prior to Unit 2 restart, NRC (including NRR personnel) performed an inspection[2] to assess the adequacy of the testing, calibration, maintenance and configuration control of safety-related instrumentation. Section 5 of Inspection Report 89-06 states:

  > "The latest procedure used by the licensee for setpoint calculations is the Division of Nuclear Engineering (DNE), Electrical Engineering Branch (EEB), instruction EEB-TI-28, Revision 1, dated October 24, 1988. … Procedure EEB-TI-28 incorporates the guidance found in RG 1.105 and ISA Standard 67.04 and is acceptable for assuring that setpoints are

1 TVA letter, T.E. Abney to NRC, dated May 6, 2004, "Browns Ferry Nuclear Plant (BFN) - Unit 1 – Technical Specifications (TS) Change 437 – Response to Request for Additional Information Regarding Scram Discharge Volume Water Level Setpoint (TAC NOS. MC1427)

2 NRC letter, B.A. Wilson to O.D. Kingsley, dated May 8, 1989, "Notice of Violation (NRC Inspection Report Nos. 50-259/89-06, 50-260/89-06 and 50-296/89-06).

established and held within specified limits for nuclear safety-related instruments used in nuclear power plants. The guidance provided by this procedure was reflected in the setpoint calculations which were reviewed during this inspection and are identified in the scope paragraph. The methodology of determining instrument loop errors and using them in the accuracy calculation reviewed is acceptable."

- In order to support the restart of BFN Unit 2, TVA submitted[3] a request to revise the TS low water level setpoint. On January 2, 1991, NRC approved[4] the requested amendment.

  "The amendment changes the Technical Specifications (TS) to incorporate a revised trip setpoint for the Level 1 low reactor pressure vessel (RPV) water level based on new calculation methodology."

  As stated in the Safety Evaluation:

  "TVA performed a Setpoint and Scaling Calculation to determine the accuracy of the instruments and loops. This accuracy was compared to the required accuracies to assure that there is sufficient margin between the setpoints and the operating limits, and the safety limits. The calculations reviewed by the staff at TVA's Rockville offices were as follows *(several calculations listed)*. The staff's review of the calculations verified that TVA addressed instrument and loop errors for normal operation and accident conditions … The methodology for determination of instrument setpoints used by TVA was in accordance with Regulatory Guide (RG) 1.105 that endorses Instrument Society of America (ISA) Standard ISA-S67.04 – 1982 "Setpoint for Nuclear Safety Related Instrumentation Used in Nuclear Power

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3 TVA letter, E.G. Wallace to NRC, dated August 6, 1990, "Browns Ferry Nuclear Plant (BFN) - Unit 2 - TVA BFN Technical Specification (TS) No. 291 - Revision to Level 1 Low Reactor Pressure Vessel (RPV) Water Level."

4 NRC letter, T.M. Ross to O.D. Kingsley, dated January 2, 1991, "Issuance of Amendment (TAC No. 77279) (TS 291).

> Plants". … The proposed changes to the LSSS *(limiting safety system setting)* and SL *(safety limit)* settings are acceptable because they are based on a value derived by approved calculational means. This change ensures that trips occur within the analytical limit used to confirm the design bases of the plant."

This NRC approved setpoint methodology continues to be used and has formed the basis for subsequent NRC approval of Technical Specification changes. For example, the NRC approved[5] a change in the reactor vessel water level safety limit and limiting safety system setting for BFN Units 1 and 3 by Amendments 222 and 196, respectively. The Safety Evaluation states:

> "The methodology used by the licensee to determine the LSSS is in accordance with the Instrument Society of America Standard ISA-S67.04 – 1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants." This methodology is consistent with the guidance of Regulatory Guide 1.105. Therefore, the proposed LSSS is acceptable."

Per discussions with the Acting Chief of Project Directorate PD-II-2, the supporting calculations do not have to be submitted on the docket. These calculations are available on-site for review.

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5 NRC letter, J.F. Williams to O.D. Kingsley, dated July 17, 1995, "Issuance of Technical Specification Amendments for the Browns Ferry Nuclear Plant Units 1, 2, and 3 (TAC NOS. M89248, M89249 and M89250) (TS 318)."

NRC REQUEST

2. The proposed new allowable value of the Reactor Vessel Water Level - Low is about 10 inches below the original allowable value, that will delay the protective action to mitigate the consequence of an accident. Please discuss any impact to the automatic load sequencer initiation which will provide emergency power to the Emergency Core Cooling System (ECCS) related components.

TVA RESPONSE

The ECCS logic is automatically initiated by High Drywell Pressure with a coincident Low Reactor Pressure signal, or by Low Reactor Water Level – Level 1. The proposed change to the Level 3 allowable value has no impact on the pressure signals or the Level 1 allowable value and therefore, will not impact the analyzed initiation time or the sequencing of the ECCS pumps and components. The Emergency Diesel Generators are automatically started on High Drywell Pressure or on a loss of offsite power. Consequently, the time for Diesel Generator power availability is also not impacted by the change to the Level 3 allowable value.

NRC REQUEST

3. The original Level 3 allowable value in the Reactor Protection System (RPS) was 538 inches, and in the ECCS was 544 inches. The proposed new Level 3 allowable value in the RPS and in the ECCS are set at the same allowable value of 528 inches. Discuss any impact on system interaction by setting at the same value.

TVA RESPONSE

As discussed on page E1-11 of the submittal:

> "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…. For the limiting (0.08 ft2) 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."

As discussed in these excerpts, the LOCA analyses (large and small breaks) assume the Reactor Protection System (RPS) scram occurs immediately (time = 0) based on high drywell pressure rather than low reactor vessel water level. Therefore there is no adverse impact on the LOCA analyses assumed scram function since the scram will occur immediately based on high drywell pressure (rather than low reactor vessel level) well before the water level reaches Level 3. However, if the reactor water level drops without an increase in drywell pressure (such as a steam line break outside primary containment), the scram would occur on main steam isolation valve closure due to high steam flow signal. Thus, there would be no impact on the level 3 setpoint.

The ECCS function at Level 3 is a confirmatory signal for initiation of the Automatic Depressurization System (ADS). This Level 3 ECCS signal alone does not initiate ADS. With the Level 3 confirmatory signal present, the ADS system timer is initiated when the reactor water level reaches Level 1 (398 inches above vessel zero). Since ADS does not initiate until after reaching Level 1, ADS initiation will occur well after a scram based on low water level and hence there is no impact on system interaction by making the Level 3 RPS and ECCS functions at the same level. High drywell pressure and ECCS pumps running signals also provide inputs to the ADS logic, but the proposed reactor water level 3 change does not affect these inputs.

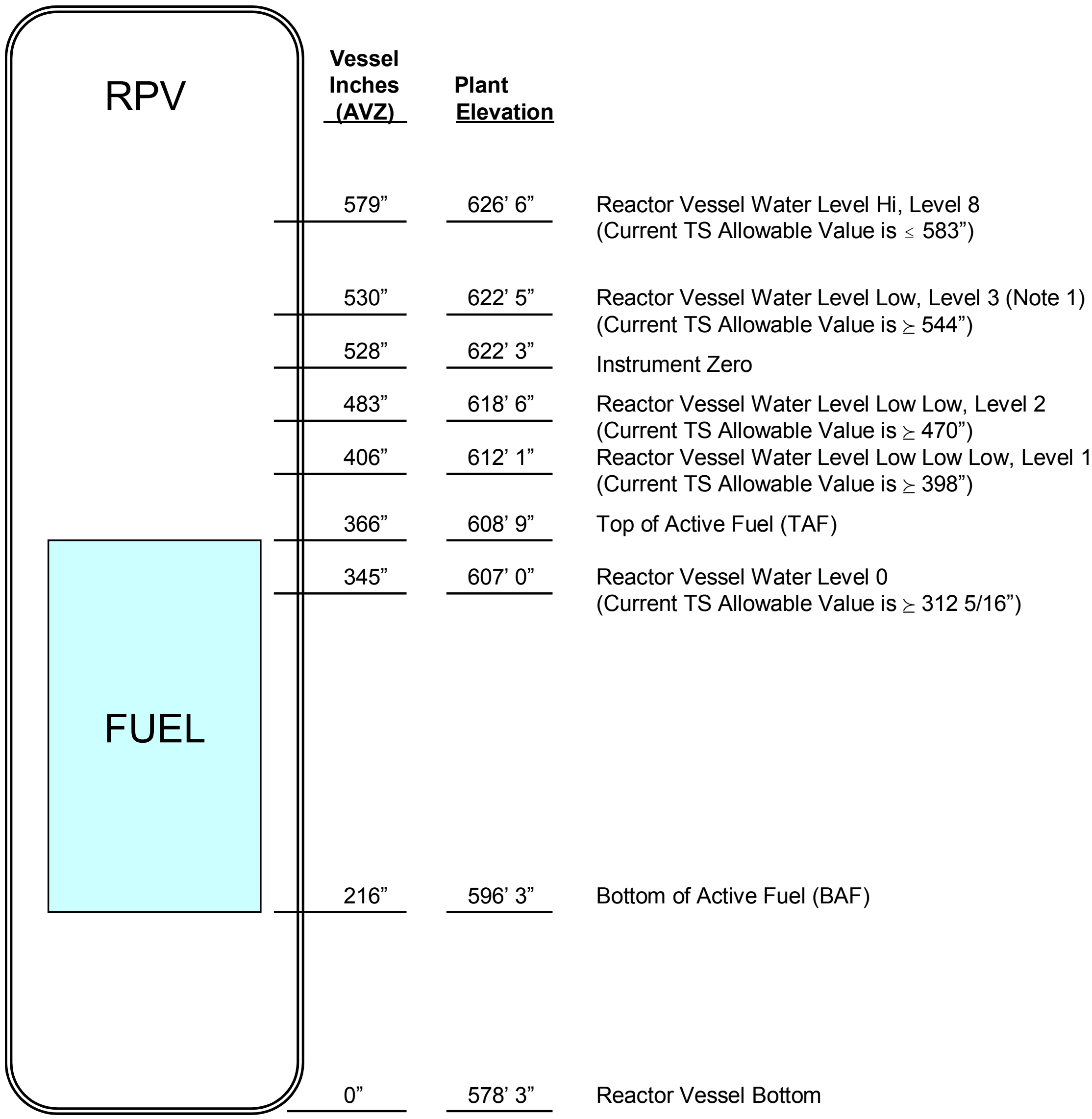
NRC REQUEST

4. Please provide a simplified figure that shows, as a minimum, the elevations of the water in the reactor vessel for Level 3 and the top of active fuel.

TVA RESPONSE

A copy of the simplified figure is attached.

REACTOR VESSEL WATER LEVEL SETPOINTS



Notes:
1. Proposed setpoint based on submitted Technical Specification Change Request TS 434.

NRC REQUEST

5. Please discuss the assumptions that resulted in a small reduction of the peak clad temperature for a small-break loss-of-coolant accident analysis.

TVA RESPONSE

The small decrease in predicted peak clad temperature (PCT) due to the propose Level 3 change for a small break loss of coolant accident is a result of the combination of an input assumption and a thermal-hydraulic modeling assumption. As discussed on page E1-8 of the submittal:

> "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."

The small break accident is modeled with an assumed input that the reactor water level starts at the Level 3 scram analytical limit (512 inches above vessel zero) as opposed to the normal water level (562.5 inches above vessel zero). Level 3 is still well above the top of active fuel (366.5 inches above vessel zero). Starting with a low level in the model is acceptable because the PCT is not affected during the time when the water level is dropping from its normal level to Level 3 because the fuel remains covered with water.

The artificial reduction in post LOCA peak clad temperature is due to combination of the assumed lower starting water level described above (530 inches versus 512 inches above vessel zero) along with a thermal-hydraulic modeling assumption in the computer code. The SAFER GESTR computer code assumes that the liquid in the reactor vessel below the feedwater spargers (498.5 inches above vessel zero) is sub-cooled liquid while the water above the top of active fuel is saturated liquid. Since the elevation of the feedwater spargers is not affected by lowering Level 3, the amount of sub-cooled liquid is not changed. However, by lowering Level 3 (and hence the initial water level assumed in the analysis as discussed above), the distance from Level 3 to the feedwater spargers is decreased. This changes the ratio of the mass of sub-cooled liquid versus saturated liquid and hence the total amount of energy in the reactor coolant. The relative decrease in saturated liquid reduces the total energy in the vessel which results in less energy to expel from the vessel. By having a lower amount of energy to expel via the ADS, the reactor pressure can be reduced to the low pressure ECCS shut off head earlier. The earlier initiation of the low pressure systems leads to a slight reduction in the predicted PCT. Thus, the combination of the lower water level input to the model coupled with the thermal-hydraulic modeling assumption leads to the calculated reduction in PCT. In reality, the PCT would not be significantly affected by the decrease in Level 3 since the thermal-hydraulic conditions in the vessel are controlled by the conditions that exist when the reactor water level reaches Level 1. The time to blow down to Level 1 from normal water level will not be affected by the reduction in Level 3. In this case the computer modeling of the complex interaction of many phenomena during a LOCA leads to a prediction of a reduced PCT.