September 18, 2006

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING ALLOWABLE VALUE FOR REACTOR VESSEL WATER LEVEL (TAC NO. MC2305) (TS-434)

Dear Mr. Singer:

The Commission has issued the enclosed Amendment No. 258 to Renewed Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. This amendment is in response to your application dated March 9, 2004, as supplemented by letters dated November 15, 2004, and March 7, 2006. The amendment reduces the Allowable Value used for Reactor Vessel Water Level - Low, Level 3, for several instrument functions.

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

Sincerely,

/RA/

Margaret H. Chernoff, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosures:
1. Amendment No. 258 to DPR-33
2. Safety Evaluation

cc w/encls: See next page

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 258
Renewed License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 9, 2004, as supplemented by letters dated November 15, 2004, and March 7, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 258, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L. Raghavan, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: September 18, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 258

RENEWED FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace Page 3 of Renewed Operating License DPR-33 with the attached Page 3.

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.3-7	3.3-7
3.3-46	3.3-46
3.3-47	3.3-47
3.3-58	3.3-58
3.3-60	3.3-60
3.3-64	3.3-64
3.3-69	3.3-69

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 258

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

1.0 INTRODUCTION

By letter dated March 9, 2004, as supplemented by letters dated November 15, 2004, and March 7, 2006, the Tennessee Valley Authority (the licensee) requested an amendment to Renewed Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant (BFN), Unit 1. The proposed amendment would reduce the Allowable Value (AV) specified for Reactor Vessel Water Level - Low, Level 3, for several instrument functions, in the Technical Specifications (TSs).

The supplements dated November 15, 2004, and March 7, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 13, 2004 (69 FR 19575).

2.0 REGULATORY EVALUATION

The proposed TS changes will lower the current Reactor Vessel Water Level - Low, Level 3 AV for several instrumentation functions. The affected functions are reactor protection system (RPS) actuation, Emergency Core Cooling Systems (ECCS) (including automatic depressurization system reactor vessel level confirmatory signal), primary containment isolation (including reactor water cleanup (RWCU) system and shutdown cooling system isolation), secondary containment isolation, and control room emergency ventilation system (CREVS) initiation. The regulations applicable to these instrumentation functions include Title 10 to the *Code of Federal Regulations* (10 CFR), Section 50.36, Technical Specifications; Section 50.46, Acceptance Criteria for ECCSs for Light-Water Nuclear Power Reactors, and Appendix K, ECCS Evaluation Models; Appendix A, General Design Criterion 19; Part 20, Standards for protection against radiation; and Part 100, Reactor site criteria.

3.0 TECHNICAL EVALUATION

3.1 Background

During operation, significant changes in reactor vessel water level can occur due to pressure transients that cause shrinking or swelling of the steam within the coolant system, or due to excessive rates of addition or removal of coolant from the reactor vessel, such as might result from a feedwater pump trip. There is a 23-inch difference in elevation between the normal reactor water level (561 inches) and the current reactor trip (scram) initiation level (Level 3, 538 inches). Process control systems are designed so the reactor can automatically recover from many transients, such as a trip of a feedwater system pump, which might cause a significant change in the water level. However, in some cases, with this narrow 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 to tripping the reactor, a drop in vessel level to Level 3 initiates primary and secondary containment isolations, CREVS operation and arming of the Automatic Depressurization System (ADS).

The proposed change will provide additional operating range between the normal reactor vessel water level and the level used as the setpoint for initiation of the above functions. The increased range should provide additional time for operators or plant control systems to automatically respond to recoverable transients such as feedwater system malfunctions and, thus, may avert unnecessary reactor scrams. This change should similarly reduce the likelihood of initiation of the other aforementioned system actuations, without increasing the consequences of events that rely upon these functions.

3.2 Description of Change

The proposed TS changes will lower the current Reactor Vessel Water Level - Low, Level 3 AV from $538 inches above vessel zero to $528 inches above vessel zero for the following functions:

- Table 3.3.1.1-1, Reactor Protection System Instrumentation, Function 4, Primary Containment Isolation - Reactor Vessel Water Level - Low, Level 3;

- Table 3.3.6.1-1, Primary Containment Isolation Instrumentation, Function 2.a, Primary Containment Isolation Instrumentation - Reactor Vessel Water Level - Low, Level 3,

- Table 3.3.6.1-1, Primary Containment Isolation Instrumentation, Function 5.h, Reactor Water Cleanup (RWCU) System Isolation - Reactor Vessel Water Level - Low, Level 3,

- Table 3.3.6.1-1, Primary Containment Isolation Instrumentation, Function 6.b, Shutdown Cooling System Isolation, Reactor Vessel Water Level - Low, Level 3,

- Table 3.3.6.2-1, Secondary Containment Isolation Instrumentation, Function 1, Reactor Vessel Water Level - Low, Level 3,

- Table 3.3.7.1-1, Control Room Emergency Ventilation System Instrumentation, Function 1, Reactor Vessel Water Level - Low, Level 3.

Additionally, the proposed TS changes will lower the current Reactor Vessel Water Level - Low, Level 3 AV from $544 inches above vessel zero to $528 inches above vessel zero for the following functions:

- Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation, Function 4.d, ADS Trip System A - Reactor Vessel Water Level - Low, Level 3 (Confirmatory),

- Table 3.3.5.1-1, Emergency Core Cooling System Instrumentation, Function 5.d, ADS Trip System B - Reactor Vessel Water Level - Low, Level 3 (Confirmatory).

The proposed changes to Reactor vessel water level - Low, Level 3 instrumentation setpoint AVs were calculated for each function using staff approved setpoint methodology. The staff's review of this methodology and the instrument operability determination was documented in a Safety Evaluation dated September 14, 2006.

For each analyzed accident/event, the effect of the change in initiation of these protective safety functions is discussed below:

3.3 Evaluation of Changes

3.3.1 RPS Actuation

To determine the effects of the change in reactor vessel water level - low, Level 3 scram AV, the licensee evaluated (a) abnormal operational occurrences (AOO), (b) loss-of-coolant accidents (LOCA), (c) anticipated transients without scram (ATWS), (d) Appendix R events (fires), and (e) events involving potential radiological releases. The results of the licensee's analysis and U.S. Nuclear Regulatory Commission (NRC) staff review are summarized below.

3.3.1.2 Abnormal Operational Occurrences

The licensee utilized a screening process to examine the AOO in the Updated Final Safety Analysis Report (UFSAR) for BFN Unit 1 to determine if a Level 3 RPS actuation is credited for mitigation of the event. The licensee found that a Total Loss of Feedwater event is the only AOO for which a Level 3 water level initiated scram occurs. For this event, a Reactor Core Isolation Cooling and High Pressure Core Injection systems initiation subsequently occurs at a lower level (Level 2) and adequately maintains core coverage. Thus, the licensee concluded that no unacceptable safety consequences would occur for any AOO if the Level 3 AV setpoint is reduced.

However, in a letter dated August 16, 2004, General Electric Company (GE) submitted a letter containing a Part 21 60-Day Interim Notification: Narrow Range Water Level Instrument Level 3 Trip (ADAMS Accession No. ML042720291). In the letter, GE stated that a conservative evaluation by GE Nuclear Energy has determined that water level instruments may indicate high by as much as 8 inches should the reactor water level drop below the dryer seal skirt. As part of a review of a separate license amendment request by the licensee, the NRC staff requested TVA to describe TVA's evaluation of the safety concern and explain, in detail, how the issue was resolved for BFN Unit 1.

In a letter dated March 7, 2006 (ADAMS Accession No. ML060720248), TVA responded to the NRC staff's request. TVA confirmed that during a loss-of-feedwater transient event, the dryer skirt will be exposed. TVA calculated the combined potential error to be 8.11 inches. TVA confirmed that the instrument would continue to function with sufficient margin from the analytical limit and the trip setting, without affecting safety margin.

Based on the review of the information provided by licensee, the staff determined that the licensee has adequately addressed the concerns identified in the GE letter described above and the change in the Level 3 AV setpoint for RPS will not result in significant impact on the AOO for BFN Unit 1.

3.3.1.3 LOCA

The licensee's analysis determined that for a large break LOCA, a reactor scram is initiated by high drywell pressure prior to the time that reactor vessel level decreases to Level 3 setpoint. Due to this action, the licensee concluded that the change in reactor vessel Level 3 trip would have no effect on large break LOCA consequences. For a small break LOCA, the licensee concluded that the reduced Analytical Water Level limit at the time of scram initiation slightly decreases the calculated peak cladding temperature (PCT). The reduction in the PCT is related to the earlier initiation of ADS on low level 1 signal due to the lower initial water level.

The licensee's analysis also encompassed a review of the potential effects on containment dynamic loads, safety/relief valve discharge loads, and suppression pool response for a design-basis LOCA. The analysis indicates that because a scram would be initiated as a result of high drywell pressure, prior to Level 3, the Level 3 AV setpoint change would have no effect on these responses.

Based on the review of the information provided by the licensee, the NRC staff agrees with the licensee's conclusion above.

3.3.1.4 ATWS

The licensee stated that in an ATWS scenario, no automatic or manual scram occurs. Thus, the change in Reactor Protection System Level 3 initiation has no effect. The NRC staff agrees with the licensee's assessment of ATWS event.

3.3.1.5 Fire

The licensee stated that for Appendix R (fire) events the reactor is manually scrammed and, thus, the Level 3 setpoint has no effect on the consequences. The NRC staff agrees.

3.3.1.6 Radiological Release

The licensee stated that the limiting pipe break for radiological releases outside containment is the design-basis main steamline break outside the containment. The licensee's analysis indicates that for the main steamline break event, a scram occurs due to the high steamline flow protective function and, thus, the change in the low water level function will not affect the consequences.

Also, the licensee indicated that the limiting pipe break for radiological releases inside the containment is the design-basis LOCA. The design-basis 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 AV has no impact on the radiological release analyses inside the containment for the design-basis LOCA analyses.

Based on the review of the information provided by the licensee, the NRC staff agrees with the licensee's conclusion about the impact of the reduced Level 3 AV on potential radioactive release.

3.3.1.7 Conclusion for RPS Level 3 reduction

Based on the discussion in previous sections, the NRC staff concludes that the reduced Level 3 RPS AV will not have any significant impact on the plant operation. The NRC staff also concludes that BFN Unit 1 continues to meet the requirements of 10 CFR 50.36, 10 CFR 50.46, and Appendix K. Therefore, the proposed change is acceptable.

3.3.2 Primary Containment Isolation Including Shutdown Cooling and RWCU System Isolation

A protective feature of the BFN Unit 1 is isolation of the primary containment penetrations if reactor vessel level drops to Level 3 setpoint. This function assures that onsite and offsite dose limits established by 10 CFR Part 20 and 10 CFR Part 100 are not exceeded.

The licensee stated that significant radiation releases cannot occur until after the core is uncovered and, with the reduced Level 3 setpoint, containment isolation will still occur well before core uncovery; thus, the small delay in primary containment isolation will not affect the ability of the containment isolation valves to perform their intended functions. Also, the licensee stated, that for LOCA events inside containment, a high drywell pressure signal will also initiate primary containment isolation for all systems affected by Level 3 signal (except RWCU) prior to Level 3 water level trip.

The residual heat removal system (RHRS) Primary Containment Isolation function is also required to be operable during shutdown cooling operations. During shutdown cooling operations a Level 3 condition will initiate closure of the shutdown cooling isolation valves. This prevents any further loss of coolant inventory via the RHRS if RHRS leakage is the reason for the reduction in vessel level. The licensee stated that the reduction of Level 3 AV will not affect the intended function of isolation valves since the system will still isolate at a water level far above the top of the core. Therefore, reducing the Level 3 AV has no impact on the ability of the shutdown cooling mode isolation to perform its intended functions.

Another primary containment isolation function that occurs on a Low Level 3 signal is isolation of the RWCU System. This signal is one of several that initiate an RWCU isolation in the event of loss of reactor coolant due to an RWCU 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 (Design-basis LOCA and main steamline break outside the containment). The licensee stated that the reduced Level 3 setpoint would not impact the capability of the RWCU isolation valves to perform their intended function. Also, in the event of an RWCU line break, the RWCU system may be isolated earlier as the result of other RWCU system leakage detection functions.

Based on the review of the information provided by licensee, the NRC staff agrees that the primary containment isolation function, including the isolation function for RHRS cold shutdown mode and RWCU isolation function will not have adverse impact from reduced Level 3 AV setpoint and associated Part 20 and Part 100 limits will not be exceeded. Therefore, the proposed change for reduced Level 3 AV setpoints for these systems is acceptable.

3.3.3 Secondary Containment Isolation and Standby Gas Treatment System (SGTS)

The primary containment system is enclosed by a secondary containment system which, in the event of an accident, confines gaseous primary containment leakage. This leakage is exhausted from the secondary containment enclosure by an SGTS and discharged to an elevated release point. Like primary containment isolation, operation of the secondary containment system is also initiated upon a vessel Low Level 3 condition.

The LOCA provides the most severe potential radiological release to the primary and secondary containment and, thus, serves as the bounding design-basis accident in determining the post-accident offsite dose. For LOCA events, the secondary containment and SGTS will actuate on high drywell pressure prior to reaching the Level 3 water level trip. Therefore, a reduced Level 3 AV would have no effect on the LOCA event analysis. For other loss of inventory events, the Level 3 actuation will occur well before any core uncovery, and potential radiological release. Therefore, small change in Level 3 actuation will not affect the secondary containment and SGTS performance.

Based on the above discussion, the proposed change in Level 3 AV is acceptable.

3.3.4 CREVS Actuation

The CREVS 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 CREVS is automatically initiated to pressurize the control room with filtered air to minimize the radiological doses to control room personnel. The LOCA provides the most severe potential radiological release to the primary and secondary containment and, thus, serves as the bounding design-basis accident in determining the control room dose, which must not exceed the criteria of General Design Criterion 19. For LOCA events, the CREVS will actuate on high drywell pressure prior to reaching the Level 3 water level trip. Therefore, a reduced Level 3 AV would have no effect on the LOCA event analysis. For other loss of inventory events, the Level 3 actuation will occur well before any core uncovery, which could result in potential radiological release. Therefore, a small change in Level 3 actuation will not affect the ability of the CREV system to perform its intended function.

Based on the above discussion, the proposed change in Level 3 AV is acceptable.

3.3.5 Automatic Depressurization System

The proposed TS change lowers ADS confirmatory signal Level 3 AV 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 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 AV will not affect the ability of ADS to perform its intended function and the proposed change is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (69 FR 19575). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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 to the health and safety of the public.

Principal Contributor: Chandu Patel

Date: September 18, 2006

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cc:

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