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

August 16, 1999

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS REGARDING ALLOWABLE VALUE FOR REACTOR VESSEL WATER LEVEL (TAC NOS. MA5697 AND MA5698)

Dear Mr. Scalice:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 260 and 219 to Facility Operating License Nos. DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant Units 2 and 3, respectively. These amendments are in response to your application dated June 3, 1999 (Technical Specifications Change Request No. TS 397). The amendments reduce the Allowable Value used for Reactor Vessel Water Level - Low, Level 3 for several instrument functions.

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

Sincerely,

Original signed by:

William O. Long, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 and 50-296

Enclosures: 1. Amendment No. 260to License No. DPR-52
2. Amendment No. 219 to License No. DPR-68
3. Safety Evaluation

cc w/enclosures: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 16, 1999

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3 - ISSUANCE OF AMENDMENTS REGARDING ALLOWABLE VALUE FOR REACTOR VESSEL WATER LEVEL (TAC NOS. MA5697 AND MA5698)

Dear Mr. Scalice:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 260 and 219 to Facility Operating License Nos. DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant Units 2 and 3, respectively. These amendments are in response to your application dated June 3, 1999 (Technical Specifications Change Request No. TS 397). The amendments reduce the Allowable Value used for Reactor Vessel Water Level - Low, Level 3 for several instrument functions.

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

Sincerely,

William O. Long, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 and 50-296

Enclosures:
1. Amendment No. 260 to License No. DPR-52
2. Amendment No. 219 to License No. DPR-68
3. Safety Evaluation

cc w/enclosures: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 260
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 3, 1999, 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 Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 260 , 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

Sheri R. Peterson, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 16, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 260

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

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 |
|---|---|
| 3.3-8 | 3.3-8 |
| 3.3-47 | 3.3-47 |
| 3.3-59 | 3.3-59 |
| 3.3-61 | 3.3-61 |
| 3.3-65 | 3.3-65 |
| 3.3-70 | 3.3-70 |

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. Inop | 1,2 | 3(b) | G | SR 3.3.1.1.16 | NA |
| e. 2-Out-Of-4 Voter | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.14<br>SR 3.3.1.1.16 | NA |
| f. OPRM Upscale | 1 | 3(b) | I | SR 3.3.1.1.1<br>SR 3.3.1.1.7<br>SR 3.3.1.1.13<br>SR 3.3.1.1.16<br>SR 3.3.1.1.17 | NA |
| 3. Reactor Vessel Steam Dome Pressure - High | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.8<br>SR 3.3.1.1.10<br>SR 3.3.1.1.14 | ≤ 1090 psig |
| 4. Reactor Vessel Water Level - Low, Level 3 | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>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<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 10% closed |
| 6. Drywell Pressure - High | 1,2 | 2 | G | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>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<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 50 gallons |
| | 5(a) | 2 | H | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>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.

(b) Each APRM channel provides inputs to both trip systems.

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<br>SR 3.3.5.1.2<br>SR 3.3.5.1.5<br>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<br>SR 3.3.5.1.3<br>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<br>SR 3.3.5.1.3<br>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<br>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<br>SR 3.3.5.1.2<br>SR 3.3.5.1.5<br>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<br>SR 3.3.5.1.5<br>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<br>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<br>SR 3.3.5.1.2<br>SR 3.3.5.1.5<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.4<br>SR 3.3.6.1.6 | ≤ 188°F |
| b. Pipe Trench Area Temperature - High | 1,2,3 | 2 | F | SR 3.3.6.1.2<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.2.2<br>SR 3.3.6.2.3<br>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<br>SR 3.3.6.2.3<br>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<br>SR 3.3.6.2.2<br>SR 3.3.6.2.3<br>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<br>SR 3.3.6.2.2<br>SR 3.3.6.2.3<br>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<br>SR 3.3.7.1.2<br>SR 3.3.7.1.5<br>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<br>SR 3.3.7.1.5<br>SR 3.3.7.1.6 | ≤ 2.5 psig |
| 3. Reactor Zone Exhaust Radiation - High | 1,2,3<br>(a),(b) | 1 | C | SR 3.3.7.1.1<br>SR 3.3.7.1.2<br>SR 3.3.7.1.5<br>SR 3.3.7.1.6 | ≤ 100 mR/hr |
| 4. Refueling Floor Exhaust Radiation - High | 1,2,3,<br>(a),(b) | 1 | C | SR 3.3.7.1.1<br>SR 3.3.7.1.2<br>SR 3.3.7.1.5<br>SR 3.3.7.1.6 | ≤ 100 mR/hr |
| 5. Control Room Air Supply Duct Radiation - High | 1,2,3,<br>(a),(b) | 1 | D | SR 3.3.7.1.1<br>SR 3.3.7.1.2<br>SR 3.3.7.1.3<br>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.

UNITED STATES
## NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 219
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 3, 1999, 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 Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 219 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

Sheri R. Peterson, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 16, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 219

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

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 |
|---|---|
| 3.3-8 | 3.3-8 |
| 3.3-47 | 3.3-47 |
| 3.3-59 | 3.3-59 |
| 3.3-61 | 3.3-61 |
| 3.3-65 | 3.3-65 |
| 3.3-70 | 3.3-70 |

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. Inop | 1,2 | 3(b) | G | SR 3.3.1.1.16 | NA |
| e. 2-Out-Of-4 Voter | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.14<br>SR 3.3.1.1.16 | NA |
| 3. Reactor Vessel Steam Dome Pressure - High | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.8<br>SR 3.3.1.1.10<br>SR 3.3.1.1.14 | ≤ 1090 psig |
| 4. Reactor Vessel Water Level - Low, Level 3 | 1,2 | 2 | G | SR 3.3.1.1.1<br>SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>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<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 10% closed |
| 6. Drywell Pressure - High | 1,2 | 2 | G | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>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<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 50 gallons |
| | 5(a) | 2 | H | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 50 gallons |
| b. Float Switch | 1,2 | 2 | G | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>SR 3.3.1.1.14 | ≤ 50 gallons |
| | 5(a) | 2 | H | SR 3.3.1.1.8<br>SR 3.3.1.1.13<br>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.

(b) Each APRM channel provides inputs to both trip systems.

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<br>SR 3.3.5.1.2<br>SR 3.3.5.1.5<br>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<br>SR 3.3.5.1.3<br>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<br>SR 3.3.5.1.3<br>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<br>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<br>SR 3.3.5.1.2<br>SR 3.3.5.1.5<br>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<br>SR 3.3.5.1.5<br>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<br>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<br>SR 3.3.5.1.2<br>SR 3.3.5.1.5<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.4<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.2<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.1.5<br>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<br>SR 3.3.6.2.2<br>SR 3.3.6.2.3<br>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<br>SR 3.3.6.2.3<br>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<br>SR 3.3.6.2.2<br>SR 3.3.6.2.3<br>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<br>SR 3.3.6.2.2<br>SR 3.3.6.2.3<br>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<br>SR 3.3.7.1.2<br>SR 3.3.7.1.5<br>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<br>SR 3.3.7.1.5<br>SR 3.3.7.1.6 | ≤ 2.5 psig |
| 3. Reactor Zone Exhaust Radiation - High | 1,2,3<br>(a),(b) | 1 | C | SR 3.3.7.1.1<br>SR 3.3.7.1.2<br>SR 3.3.7.1.5<br>SR 3.3.7.1.6 | ≤ 100 mR/hr |
| 4. Refueling Floor Exhaust Radiation - High | 1,2,3,<br>(a),(b) | 1 | C | SR 3.3.7.1.1<br>SR 3.3.7.1.2<br>SR 3.3.7.1.5<br>SR 3.3.7.1.6 | ≤ 100 mR/hr |
| 5. Control Room Air Supply Duct Radiation - High | 1,2,3,<br>(a),(b) | 1 | D | SR 3.3.7.1.1<br>SR 3.3.7.1.2<br>SR 3.3.7.1.3<br>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.

UNITED STA"ES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 260 TO FACILITY OPERATING LICENSE NUMBER DPR-52

AND AMENDMENT NO. 219 TO FACILITY OPERATING LICENSE NUMBER DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS. 50-260 AND 50-296

## 1.0 INTRODUCTION

By application dated June 3, 1999, the Tennessee Valley Authority (the licensee) requested amendments to Facility Operating License Nos. DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant Units 2 and 3, respectively. The proposed amendments would reduce the Allowable Value (Av) specified for Reactor Vessel Water Level - Low, Level 3, for several instrument functions, in the Technical Specifications (TS).

The intent of the proposed TS changes is to reduce the likelihood of unnecessary reactor scrams and the associated engineered safety feature (ESF) 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 actuations of protective features.

## 2.0 DISCUSSION AND EVALUATION

### 2.1 Description of Functional Change

During operation, significant changes in 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 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 such that 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, Standby Gas Treatment System (SGTS) operation, Control Room Emergency Ventilation System (CREVS) operation and arming of the Automatic Depressurization System (ADS). These actuations are unneeded distraction factors for the operators in responding to the associated reactor trip.

The proposed change will provide an additional 10 inches of 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 will 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 will similarly reduce the likelihood of initiation of the other aforementioned system actuations, without increasing the consequences of events that rely upon these functions.

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

## 2.2 Safety Analysis

### 2.2.1 Scope

The licensee's amendment application presents a safety analysis to support the proposed change. The safety analysis identified the role each of the affected protective action plays in the mitigation of (a) abnormal operational occurrences, (b) loss of coolant accidents (LOCA), (c) anticipated operational occurrences (AOO), (d) anticipated transients without scram (ATWS), (e) Appendix R events (fires) and (f) other events involving a potential radiological release. The subsections below discuss the results of the licensee's analysis, for each category of event.

### 2.2.2 Abnormal Operational Occurrences

The licensee utilized a screening process to examine each design basis AOO to determine if a Level 3 reactor protection system (RPS) actuation is credited for mitigation of the event. The licensee found that a Loss of Feedwater event is the only AOO for which a Level 3 Low Water initiated scram occurs. For this event, a Reactor Core Isolation Cooling initiation subsequently occurs at a lower level and adequately maintains core coverage. Thus, no unacceptable safety consequences would occur for any AOO if the Level 3 setpoint is reduced.

### 2.2.3 Loss of Coolant Accident (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 vessel level decreases to Level 3. Due to this action, the licensee concluded that the change in vessel level trip would have no effect on large break LOCA consequences. For a small break LOCA, the licensee found that the effect of the reduced water level at the time of scram initiation is to decrease the peak cladding temperature (PCT) from 1367 degrees F to 1346 degrees F. The PCT is decreased due to the earlier initiation of the ADS.

The licensee's analyses 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 accident (DBA) 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 setpoint change would have no effect on these responses.

2.2.4 Fire

The licensee's analysis indicates that, for Appendix R (fire) events, the reactor is manually scrammed and, thus, the Level 3 setpoint has no effect on the consequences.

2.2.5 ATWS

In an ATWS scenario, no automatic or manual scram occurs. Thus, the change in RPS Level 3 initiation has no effect.

2.2.6 Main Steamline Break Outside Containment:

The licensee's analysis states that for the main steam line 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. For smaller breaks for which the high steam flow function is not sufficiently sensitive, the break would be sensed by other leakage detection systems that are not affected by the Level 3 setpoint.

2.2.7 Primary Containment Isolation Including Shutdown Cooling System Isolation

A protective feature of the Browns Ferry facilities is isolation of the primary containment penetrations if vessel level drops to Level 3. 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's analysis states 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 have no effect in the event of an accident during operation. The staff notes that the DBA-LOCA radiological dose consequences analysis is not affected by the scram delay, since the release is a bounding, standard source term, which is unchanged, and for which the release is assumed to begin at the time of a simultaneous break initiation and scram.

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. If the vessel level continues to drop to Level 1, the low pressure coolant injection function will be initiated and will restore water level. Because the safety injection function is not affected, the change in the Level 3 setpoint will have no adverse effect on a shutdown cooling event.

Another primary containment isolation function that occurs on a Low Level 3 signal is isolation of the Reactor Water Cleanup (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 licensee's analysis states that the reduced Level 3 setpoint would not impact the capability of the RWCU isolation valves to perform their intended function. The staff agrees, noting that, in the event of an RWCU line break the RWCU system would likely be isolated earlier as the result of other RWCU system leakage detection functions.

### 2.2.8 Secondary Containment Isolation and Standby Gas Treatment System

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.

Radiological dose calculations for the design basis LOCA assume that, in the event of a DBA-LOCA, primary containment isolation and secondary containment/SGTS initiation occur simultaneously with all primary containment leakage being released untreated at ground level until a negative pressure is established in the secondary containment by the SGTS. The system must be capable of establishing a specified negative pressure in secondary containment within a specified time from initiation, and with a specified secondary containment infiltration rate. Since secondary containment/SGTS initiation will continue to occur at the same time as primary containment isolation, and the design basis performance requirements are not changed, the radiological dose mitigation function of the secondary containment/SGTS would be unaffected.

### 2.2.9 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 DBA 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.

### 2.2.10 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 (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 Av will not affect the ability of ADS to perform its intended function.

## 3.0 STATE CONSULTATION

In accordance with the U.S. Nuclear Regulatory Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendments. The State official had no comments.

## 4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (64 FR 38037). Accordingly, the amendments meet 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 amendments.

## 5.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: August 16, 1999

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