April 16, 1996

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: ISSUANCE OF EMERGENCY TECHNICAL SPECIFICATION AMENDMENTS FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 (TAC NOS. M95165, M95166, AND M95167) (TS 375)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment Nos.229 ,244 , and 204 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3, respectively. These amendments are in response to your application dated April 14, 1996, requesting an emergency amendment to clarify reactor water level instrumentation requirements when testing excess flow check valves pursuant to technical specification 4.7.D.1.d.

A copy of the NRC's Safety Evaluation is enclosed. A Notice of Issuance of Amendment to Facility Operating License and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by
Joseph F. Williams, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260 and 50-296

Enclosures:
1. Amendment No.229 to License No. DPR-33
2. Amendment No.244 to License No. DPR-52
3. Amendment No.204 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures: See next page

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DATE	4/15/96		4/15/96		4/15/96		4/16/96		4/16/96	

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.229
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 April 14, 1996, 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-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. 229 , 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 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 16, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 229

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. *Overleaf and **spillover pages are included to maintain document completeness.

REMOVE	INSERT
3.2/4.2-14	3.2/4.2-14
3.2/4.2-15	3.2/4.2-15**
3.2/4.2-23	3.2/4.2-23*
3.2/4.2-24	3.2/4.2-24

TABLE 3.2.B
INSTRUMENTATION THAT INITATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	$\geq$ 470" above vessel zero	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level	$\geq$ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2(19)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	$\geq$ 398" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2)	$\geq$ 398" above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, ADS timer timed out and CSS or RHR pump running, initiates ADS. 2. Below trip settings, in conjunction with low reactor water level permissive, ADS timer timed out, ADS high drywell pressure bypass timer timed out, CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	$\geq$ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 and 62, SW #1)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.
2(18)	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	1≤ p≤2.5 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2(18)	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
2(18)	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)(18)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤ 2.5 psig	A	1. Above trip setting, in conjunction with low reactor water level, low reactor water level permissive, ADS timer timed out, and CSS or RHR pump running, initiates ADS.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by Section 3.5 to be OPERABLE, there shall be two OPERABLE trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not OPERABLE in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required; indicators are considered redundant.

   E. Within 24 hours restore the inoperable channel(s) to OPERABLE status or place the inoperable channel(s) in the tripped condition.

2. In only one trip system.

3. Not considered in a trip system.

4. Deleted

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 sec. later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec. with similar pumps starting after about 14 sec. and 21 sec., at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150 percent above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150 percent above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2.B (Cont'd)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160-V shutdown boards. See note 13.

12. In only one of the four 4160-V shutdown boards. See note 13.

13. An emergency 4160-V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 398" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.

18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

19. Only one trip system will be required to be OPERABLE during testing of the reactor coolant system instrument line flow check valves in accordance with TS Section 4.7.D.1.d, provided the reactor is in COLD SHUTDOWN. Manual and automatic initiating capability of CSS and LPCI will be available but with a reduced number of instrument channels.

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. 244
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 April 14, 1996, 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. 244, 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 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 16, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 244

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. *Overleaf and **spillover pages are included to maintain document completeness.

REMOVE	INSERT
3.2/4.2-14	3.2/4.2-14
3.2/4.2-15	3.2/4.2-15**
3.2/4.2-23	3.2/4.2-23*
3.2/4.2-24	3.2/4.2-24

TABLE 3.2.B

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
*2(19)	Instrument Channel - Reactor Low Water Level (LS-3-58A-D)	≥ 398" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LS-3-58A-D)	≥ 398" above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, ADS timer timed out and CSS or RHR pump running, initiates ADS. 2. Below trip settings, in conjunction with low reactor water level permissive, ADS timer timed out, ADS high drywell pressure bypass timer timed out, CSS or RHR pump running, initiates ADS.

*Only one trip system will be required to be OPERABLE during the period that the Reactor Vessel water level instrumentation modification requested by NRC Bulletin 93-03 is being performed, provided that the reactor is in the COLD SHUTDOWN CONDITION. Manual and automatic initiating capability of CSS and LPCI will be available, but with a reduced number of instrument channels.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LIS-3-52 and LIS-3-62A)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.
2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58 E-H)	1≤ p≤2.5 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58 A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)(18)	Instrument Channel - Drywell High Pressure (PIS-64-57A-D)	≤ 2.5 psig	A	1. Above trip setting, in conjunction with low reactor water level, low reactor water level permissive, ADS timer timed out, and CSS or RHR pump running, initiates ADS.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by Section 3.5 to be OPERABLE, there shall be two OPERABLE trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not OPERABLE in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required; indicators are considered redundant.

   E. Within 24 hours restore the inoperable channel(s) to OPERABLE status or place the inoperable channel(s) in the tripped condition.

2. In only one trip system.

3. Not considered in a trip system.

4. Deleted.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 seconds later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 sec. with similar pumps starting after about 14 sec. and 21 sec., at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150 percent above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150 percent above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.

NOTES FOR TABLE 3.2.B (Cont'd)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160-V shutdown boards. See note 13.

12. In only one of the four 4160-V shutdown boards. See note 13.

13. An emergency 4160-V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 398" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.

18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

19. Only one trip system will be required to be OPERABLE during testing of the reactor coolant system instrument line flow check valves in accordance with TS Section 4.7.D.1.d, provided the reactor is in COLD SHUTDOWN. Manual and automatic initiating capability of CSS and LPCI will be available but with a reduced number of instrument channels.

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. 204
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 April 14, 1996, 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. 204 , 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 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 16, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 204

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. *Overleaf and **spillover pages are included to maintain document completeness.

REMOVE	INSERT
3.2/4.2-14	3.2/4.2-14
3.2/4.2-15	3.2/4.2-15**
3.2/4.2-22	3.2/4.2-22*
3.2/4.2-23	3.2/4.2-23

TABLE 3.2.B
INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2(19)	Instrument Channel - Reactor Low Water Level (LS-3-58A-D)	≥ 398" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LS-3-58A-D)	≥ 398" above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, ADS timer timed out and CSS or RHR pump running, initiates ADS. 2. Below trip settings, in conjunction with low reactor water level permissive, ADS timer timed out, ADS high drywell pressure bypass timer timed out, CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
1	Instrument Channel - Reactor Low Water Level (LIS-3-52 and LIS-3-62A)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.
2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58 E-H)	1≤ p≤2.5 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58 A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal. (15)
2(18)	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	≤ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)(18)	Instrument Channel - Drywell High Pressure (PIS-64-57A-D)	≤ 2.5 psig	A	1. Above trip setting, in conjunction with low reactor water level, low reactor water level permissive, ADS timer timed out, and CSS or RHR pump running, initiates ADS.

NOTES FOR TABLE 3.2.B

1. Whenever any CSCS System is required by Section 3.5 to be OPERABLE, there shall be two OPERABLE trip systems except as noted. If a requirement of the first column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the first column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not OPERABLE in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. No action required; indicators are considered redundant.

   E. Within 24 hours restore the inoperable channel(s) to OPERABLE status or place the inoperable channel(s) in the tripped condition.

2. In only one trip system.

3. Not considered in a trip system.

4. Deleted.

5. With diesel power, each RHRS pump is scheduled to start immediately and each CSS pump is sequenced to start about 7 seconds later.

6. With normal power, one CSS and one RHRS pump is scheduled to start instantaneously, one CSS and one RHRS pump is sequenced to start after about 7 seconds with similar pumps starting after about 14 seconds and 21 seconds, at which time the full complement of CSS and RHRS pumps would be operating.

7. The RCIC and HPCI steam line high flow trip level settings are given in terms of differential pressure. The RCICS setting of 450" of water corresponds to at least 150 percent above maximum steady state steam flow to assure that spurious isolation does not occur while ensuring the initiation of isolation following a postulated steam line break. Similarly, the HPCIS setting of 90 psi corresponds to at least 150 percent above maximum steady state flow while also ensuring the initiation of isolation following a postulated break.

8. Note 1 does not apply to this item.

9. The head tank is designed to assure that the discharge piping from the CS and RHR pumps are full. The pressure shall be maintained at or above the values listed in 3.5.H, which ensures water in the discharge piping and up to the head tank.

<u>NOTES FOR TABLE 3.2.B</u> (Continued)

10. Only one trip system for each cooler fan.

11. In only two of the four 4160-V shutdown boards. See note 13.

12. In only one of the four 4160-V shutdown boards. See note 13.

13. An emergency 4160-V shutdown board is considered a trip system.

14. RHRSW pump would be inoperable. Refer to Section 4.5.C for the requirements of a RHRSW pump being inoperable.

15. The accident signal is the satisfactory completion of a one-out-of-two taken twice logic of the drywell high pressure plus low reactor pressure or the vessel low water level ($\geq$ 398" above vessel zero) originating in the core spray system trip system.

16. The ADS circuitry is capable of accomplishing its protective action with one OPERABLE trip system. Therefore, one trip system may be taken out of service for functional testing and calibration for a period not to exceed eight hours.

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 30 percent within four hours.

18. Not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

19. Only one trip system will be required to be OPERABLE during testing of the reactor coolant system instrument line flow check valves in accordance with TS Section 4.7.D.1.d, provided the reactor is in COLD SHUTDOWN. Manual and automatic initiating capability of CSS and LPCI will be available but with a reduced number of instrument channels.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 229 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 244 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 204 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

By letter dated April 14, 1996, the Tennessee Valley Authority (the licensee) submitted proposed changes to the technical specifications (TS) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3. The changes clarify operability requirements for reactor vessel water level instrumentation in TS Table 3.2.B to clearly permit surveillance testing of instrument line excess flow check valves required by TS 4.7.D.1.d.

This amendment was submitted under the emergency provisions of 10 CFR 50.91(a)(5). The licensee states that failure to act in a timely way would prevent resumption of power operations of Browns Ferry Unit 2. The NRC staff evaluation of this request is given below.

2.0 DESCRIPTION OF PROPOSED TECHNICAL SPECIFICATIONS CHANGES

TS Table 3.2.B currently requires a minimum of two operable channels per trip system for reactor water level instrumentation. These instruments provide signals which actuate engineered safety features required to mitigate accidents. Testing of instrument line excess flow check valves pursuant to TS 4.7.D.1.d disables one instrument in each of the two trip systems. Therefore, the licensee is unable to comply with the minimum instrumentation requirements while performing other testing required by TS.

The licensee proposes to add a note to TS Table 3.2.B to resolve this problem. The proposed note reads as follows:

> Only one trip system will be required to be OPERABLE during testing of the reactor coolant system instrument line flow check valves in accordance with TS section 4.7.D.1.d, provided the reactor is in COLD SHUTDOWN. Manual and automatic initiating

Enclosure 4

capability of CSS [core spray system] and LPCI [low pressure coolant injection] will be available, but with a reduced number of instrument channels.

3.0 EVALUATION

The reactor vessel water level instrumentation affected by the proposed change consists of four instruments: LIS-3-58A, LIS-3-58B, LIS-3-58C, and LIS-3-58D. The actuation logic is a one-out-of-two-taken-twice scheme. The "A" and "C" instruments input to one trip system; the "B" and "D" instruments input to the other. Testing of the excess flow check valves pursuant to TS 4.7.D.1.d affects one reactor vessel water level instrument variable leg at a time. One variable leg is used by the "A" and "B," or the "C" and "D" instruments simultaneously. The other variable leg remains functional during testing on the other leg.

TS Table 3.2.B requires a minimum of two instruments operable per trip system. Disabling the "A" and "B," or "C" and "D" instruments for the excess flow check valve testing violates this requirement. Therefore, a contradiction is created where the surveillance testing requirements cannot be fulfilled without violating the minimum equipment configuration requirements. The licensee states that the variable leg not being tested would be expected to function as designed, but points out that this configuration does not meet single failure criteria.

The licensee states the following factors provide reasonable assurance of safe operation:

1. The automatic initiating capability of the remaining reactor vessel instrumentation.

2. The low primary system temperature. The proposed change states the excess flow check valve testing is permissible when the reactor is in a cold shutdown condition, with primary system temperature less than 212°F.

3. The low probability of an event that would result in the drain down of the reactor vessel. Piping failures are extremely improbable for these conditions, given the low temperature and margin inherent in the reactor system piping design.

4. The other reactor level instrumentation and equipment that is available for manual operator intervention in the event of a plant transient or accident. There is independent water level instrumentation which would provide adequate indication of reactor vessel inventory for operators to take action as directed by emergency procedures.

The staff agrees that the probability of a loss of coolant requiring automatic CSS and LPCI initiation is very remote during the time that the excess flow check valve test is being performed. If a loss of coolant does occur, the instrumentation unaffected by the testing would be expected to function as designed. If automatic initiation fails, operators have sufficient independent instrumentation to manually initiate equipment required to mitigate loss of inventory. Therefore, the proposed change is acceptable.

4.0 EMERGENCY CIRCUMSTANCES

BFN Unit 2 is currently in a refueling outage. Testing of the excess flow check valves pursuant to TS 4.7.D.1.d will be performed prior to resuming power operations, and is currently scheduled for April 18, 1996. The licensee initially identified the conflict between TS Table 3.2.B and TS 4.7.D.1.d on April 13, 1996. The Manager of Site Licensing briefed the NRR Project Manager on the problem that afternoon. The licensee submitted a license amendment request to resolve the problem on April 14, 1996.

Since failure to issue the amendment in a timely way would prevent resumption of operation, the staff finds that an emergency situation exists. The staff also finds that the licensee acted promptly upon identification of the conflicting requirements, promptly notified the staff of the problem, and promptly proposed an amendment to remedy the situation. The staff concludes that the licensee has not abused the emergency provisions by failure to make a timely application for the amendment. Thus, conditions needed to satisfy 10 CFR 50.91(a)(5) exist, and this amendment is being processed on an emergency basis.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed change would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or,

2. Create the possibility of a new or different kind of accident from any previously evaluated; or,

3. Involve a significant reduction in a margin of safety.

The staff finds that the proposed changes do not involve a significant hazards consideration, because operation of the Browns Ferry Nuclear Plant, Units 1, 2, and 3 in accordance with the proposed change would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change in the applicability of the minimum number of reactor low level instrument channels required to be operable does not increase the frequency of precursors to design basis events or operational transients analyzed in the BFN Final Safety Analysis Report. Therefore, the probability of an accident previously evaluated is not significantly increased.

If a loss of coolant inventory occurs during excess flow check valve testing, the remaining reactor vessel water level instrumentation would be in service, and would be capable of initiating required safety functions. In addition, other independent instrumentation will remain in service which provide reactor operators with sufficient information to manually initiate required equipment in the event of a single failure of the variable leg not being tested. Therefore, the proposed change does not significantly increase the consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated.

   The change resolves a conflict between existing technical specification requirements, clarifying circumstances where testing which reduces instrumentation capability is permissible. The change does not modify the existing plant configuration, and does not create a new pathway for radioactive materials to reach the environment. Therefore, the possibility of a new or different kind of accident is not created.

3. Involve a significant reduction in a margin of safety.

   The proposed change does not change licensing or design basis limits for initiation of protective actions. The probability of a significant loss of inventory during the excess flow check valve testing is low. If a single failure prevents remaining instrumentation from performing its intended function, operator action based on independent instrumentation will ensure initiation of the core spray and LPCI systems, if required. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

6.0 STATE CONSULTATION

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

7.0 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 changes in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22.(c)(9). The Commission has made a final no significant hazards finding with respect to this amendment. 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.

8.0 CONCLUSION

The Commission has concluded, based upon the considerations discussed above, that: (1) the amendment does not (a) significantly increase the probability or consequences of an accident previously evaluated, (b) create the possibility of a new or different kind of accident from any previously evaluated, or (c) significantly reduce a margin of safety, and therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (3) such activities will be conducted in compliance with the Commission's regulations; and (4) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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