AUGUST 2 1978

Distribution
Docket
ORB #3
Local PDR
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Grimes
TIppolito
RClark
SSheppard
Attorney, OELD
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BJones (12)
BScharf (10)
JMcGough
DEisenhut
ACRS (16)
OPA (CMiles)
DRoss
TBAbernathy
JRBuchanan
RDiggs
PCheck

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

Tennessee Valley Authority
ATTN: Mr. N. B. Hughes
Manager of Power
830 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 40, 38 and 14 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your request of February 24, 1977 as supplemented by your letter of May 23, 1978.

The amendments change the Technical Specifications to lower the reactor low water level setpoint by 20 inches (i.e., from 490" to 470").

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 40 to DPR-33
2. Amendment No. 38 to DPR-52
3. Amendment No. 14 to DPR-68
4. Safety Evaluation
5. Notice

cc w/enclosures: See page 2

subject to changes in SE as noted

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SURNAME➤	SSheppard	RClark:mjf	PCheck	CUTCHIN	TIppolito
DATE➤	7/ /78	7/7/78	7/7/78	8/1/78	8/2/78

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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345 Courtland Street
Atlanta, Georgia 30308

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
License No. DPR-33

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated February 24, 1977, as supplemented by letter dated May 23, 1978, 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 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. 40, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

TAIppolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 2, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 40

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages:

   55/56
   61/62
   63/64
   111/112

2. Marginal lines indicate revised area. Overlead pages are provided for convenience.

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 470" above vessel zero.	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation

TABLE 3.2.A (Continued)

Minimum No. Operable Per Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Main Steam Line Tunnel High Temperature	$\leq$ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Reactor Water Cleanup System Floor Drain High Temperature	160 - 180°F	C	1. Above trip setting initiates Isolation of Reactor Water Cleanup Line from Reactor and Reactor Water Return Line.
2	Instrument Channel - Reactor Water Cleanup System Space High Temperature	160 - 180°F	C	1. Same as above
1	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	$\leq$ 100 mr/hr or downscale	G	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate reactor zone and refuleing floor. c. Close atmosphere control system.
1	Instrument Channel - Reactor Building Ventilation High Radiation Refuleing Zone	$\leq$ 100 mr/hr or downscale	F	1. 1 upscale or 2 downscale will a. Initiate SGTS. b. Isolate refueling floor. c. Close atmosphere control system.
2 (7)(8)	Instrument Channel SGTS Flow - Train A Heaters	Charcoal Heaters $\leq$ 2000 cfm R. H. Heaters $\leq$ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R. H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train B Heaters	Charcoal Heaters $\leq$ 2000 cfm R.H. Heaters $\leq$ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train C Heaters	Charcoal Heaters $\leq$ 2000 cfm R.H. Heaters $\leq$ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.

6. Channel shared by RP nd Primary Containment & Reactor ssel Isolation Control System. A channel failure may be a channel failure in each system.

7. A train is considered a trip system.

8. Two out of three SGTS trains required. A failure of more than one will require action A and F.

9. There is only one trip system with auto transfer to two power sources.

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	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 378" 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)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. del timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 & 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	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	1≤ p≤ 2 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.

Unit-1

Amendment No. 40

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	$\leq$ 2 psig	A	1. Above trip setting in conjunction wi low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal.(15).
2	Instrument Channel - Reactor Low Water Level (LS-3-56A, B, C, D)	$\geq$ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PS-3-204 A, B, C, D)	$\leq$ 1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	$\leq$ 2 psig	A	1. Above trip setting in conjunction wi low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	$\leq$ 2 psig	A	1. Above trip setting in conjunction wi low reactor water level, drywell hig pressure, 120 sec. delay timer and C or RHR pump running, initiates ADS.
2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B, SW #2) (PS-68-95, SW #2) (PS-68-96, SW #2)	450 psig $\pm$ 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A & B, SW #1) (PS-68-95, SW #1) (PS-68-96, SW #1)	230 psig $\pm$ 15	A	1. Recirculation discharge valve actuation.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig $\pm$ 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPCI) admission valves.
2	Core Spray Auto Sequencing Timers (5)	$6 \le t \le 8$ secs.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \le t \le 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A3, B1, C3, and D1 Timers	$13 \le t \le 15$ sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \le t \le 1$ sec. $6 \le t \le 8$ sec. $12 \le t \le 16$ sec. $18 \le t \le 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A3, B1, C3, and D1 Timers	$27 \le t \le 29$ sec.	A	1. With normal power 2. One per pump

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177.7" (538" above vessel zero) above the top of the active fuel closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and the Reactor Water Cleanup Lines (Group 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel closes the Main Steam Line Isolation Valves and Main Steam, RCIC, and HPCI Drain Valves (Group 1 and 7). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) also initiate the RCIC and HPCI, provides input to the

LPCI loop selection logic and trips the recirculation pumps. The low reactor water level instrumentation that is set to trip when reactor water level is 17.7" (378" above vessel zero) above the top of the active fuel (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus the results given above are applicable here also.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR. An alarm, with a nominal set point of 1.5 x normal full power background, is provided also.

Pressure instrumentation is provided to close the main steam isolation valves in Run Mode when the main steam line pressure drops below 825 psig.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. DPR-52

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated February 24, 1977, as supplemented by letter dated May 23, 1978, 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 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. 38, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

TAIppolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 2, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages:

   55/56
   61/62
   63/64
   111/112

2. Marginal lines indicate revised area. Overlead pages are provided for convenience.

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 470" above vessel zero.	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation

Amendment No. 38

TABLE 3.2.A (Continued)

Minimum No. Operable Per Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Main Steam Line Tunnel High Temperature	$\leq$ 200^{o}F	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Reactor Water Cleanup System Floor Drain High Temperature	160 - 180^{o}F	C	1. Above trip setting initiates Isolation of Reactor Water Cleanup Line from Reactor and Reactor Water Return Line.
2	Instrument Channel - Reactor Water Cleanup System Space High Temperature	160 - 180^{o}F	C	1. Same as above
1	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	$\leq$ 100 mr/hr or downscale	G	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate reactor zone and refuleing floor. c. Close atmosphere control system.
1	Instrument Channel - Reactor Building Ventilation High Radiation Refuleing Zone	$\leq$ 100 mr/hr or downscale	F	1. 1 upscale or 2 downscale will a. Initiate SGTS. b. Isolate refueling floor. c. Close atmosphere control system.
2 (7)(8)	Instrument Channel SGTS Flow - Train A Heaters	Charcoal Heaters $\leq$ 2000 cfm R. H. Heaters $\leq$ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R. H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train B Heaters	Charcoal Heaters $\leq$ 2000 cfm R.H. Heaters $\leq$ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.
2 (7)(8)	Instrument Channel SGTS Flow - Train C Heaters	Charcoal Heaters $\leq$ 2000 cfm R.H. Heaters $\leq$ 2000 cfm	H and (A or F)	1. Below 2000 cfm, trip setting charcoal heaters will turn on. 2. Below 2000 cfm, trip setting R.H. heaters will shut off.

6. Channel shared by RPS d Primary Containment & Reactor V el Isolation Control System. A channel failure may be a channel failure in each system.

7. A train is considered a trip system.

8. Two out of three SGTS trains required. A failure of more than one will require action A and F.

9. There is only one trip system with auto transfer to two power sources.

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	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 378" 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)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. del timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 & 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	Instrument Channel - Drywell High Pressure (PS-64-58 E-H)	1≤ p≤ 2 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.

Unit 2

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	≤ 2 psig	A	1. Above trip setting in conjunction wi low reactor pressure initiates CSS. Multiplier relays initiate HPCI. 2. Multiplier relay from CSS initiates accident signal.(15).
2	Instrument Channel - Reactor Low Water Level (LS-3-56A, B, C, D)	≥ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PS-3-204 A, B, C, D)	≤ 1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	≤ 2 psig	A	1. Above trip setting in conjunction wi low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤ 2 psig	A	1. Above trip setting in conjunction wi low reactor water level, drywell hig pressure, 120 sec. delay timer and C or RHR pump running, initiates ADS.
2	Instrument Channel - Reactor Low Pressure (PS-3-74 A & B, SW #2) (PS-68-95, SW #2) (PS-68-96, SW #2)	450 psig ± 15	A	1. Below trip setting permissive for openi CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A & B, SW #1) (PS-68-95, SW #1) (PS-68-96, SW #1)	230 psig ± 15	A	1. Recirculation discharge valve actuation.

Amendment No. 38

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
1	Instrument Channel - Reactor Low Pressure (PS-68-93 & 94, SW #1)	100 psig $\pm$ 15	A	1. Below trip setting in conjunction with containment isolation signal and both suction valves open will close RHR (LPC admission valves.
2	Core Spray Auto Sequencing Timers (5)	$6 \leq t \leq 8$ secs.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \leq t \leq 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A3, B1, C3, and D1 Timers	$13 \leq t \leq 15$ sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \leq t \leq 1$ sec. $6 \leq t \leq 8$ sec. $12 \leq t \leq 16$ sec. $18 \leq t \leq 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A3, B1, C3, and D1 Timers	$27 \leq t \leq 29$ sec.	A	1. With normal power 2. One per pump

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177.7" (538" above vessel zero) above the top of the active fuel closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Group 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel closes the Main Steam Line Isolation Valves and Main Steam, RCIC, and HPCI Drain Valves (Group 1 and 7). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) also initiate the RCIC and HPCI, provides input to the

LPCI loop selection logic and trips the recirculation pumps. The low reactor water level instrumentation that is set to trip when reactor water level is 17.7" (378" above vessel zero) above the top of the active fuel (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Groups 2 and 8 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low water level instrumentation; thus the results given above are applicable here also.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel cladding temperatures remain below 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Section 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 200°F for the main steam line tunnel detector is low enough to detect leaks of the order of 15 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.2 FSAR. An alarm, with a nominal set point of 1.5 x normal full power background, is provided also.

Pressure instrumentation is provided to close the main steam isolation valves in Run Mode when the main steam line pressure drops below 825 psig.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. DPR-68

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

   A. The application for amendments by Tennessee Valley Authority (the licensee) dated February 24, 1977, as supplemented by letter dated May 23, 1978, 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 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. 14, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 2, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 14

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages:

   57
   64
   65
   108

2. Marginal lines indicate revised area.

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level (6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 470" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
2	Instrument Channel - High Radiation Main Steam Line Tunnel (6)	≤ 3 times normal rated full power background	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Low Pressure Main Steam Line	≥ 850 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.

ENCLOSURE 1

Table 3.2.B

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimun No. Operable Per Trip Sys (1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Below trip setting, associated with LPCI loop selection. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 378" 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)	≥ 378" above vessel zero.	A	1. Below trip settings in conjunction with drywell high pressure, low water level permissive, 120 sec. del timer and CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 & 62, SW #1)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of of containment spray during accident condition.

Amendment No. 14

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 - Drywell High Pressure (PS-64-58 E-H)	1≤p≤ 2 psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2	Instrument Channel - Drywell High Pressure (PS-64-58 A-D, SW #2)	≤ 2 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	Instrument Channel - Reactor Low Water Level (LS-3-56A, B, C, D)	≥ 470" above vessel zero	A	1. Below trip setting trips recirculation pumps
2	Instrument Channel Reactor High Pressure (PS-3-204 A, B, C, D)	≤1120 psig	A	1. Above trip setting trips recirculation pumps
2	Instrument Channel - Drywell High Pressure (PS-64-58A-D, SW #1)	≤ 2 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2 (16)	Instrument Channel - Drywell High Pressure (PS-64-57A-D)	≤ 2 psig	A	1. Above trip setting in conjunction with low reactor water level, drywell high pressure, 120 sec, delay timer and CSS or RHR pump running, initiates ADS.

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 177.7" (538" above vessel zero) above the top of the active fuel closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Group 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel closes the Main Steam Line Isolation Valves and Main Steam RCIC, and HPCI Drain Valves (Group 1 and 7). Details of valve grouping and required closing times are given in Specification 3.7. These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

The low reactor water level instrumentation that is set to trip when reactor water level is 109.7" (470" above vessel zero) above the top of the active fuel (Table 3.2.B) also initiate the RCIC

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT NOS. 1, 2 AND 3

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

1.0 Introduction

By letters dated February 24, 1977 (Reference 1) and May 23, 1978 (Reference 2) the Tennessee Valley Authority (TVA) has requested approval for setting the Browns Ferry Unit Nos. 1, 2, and 3 (BF1, BF2, BF3) low water level setpoints at 470 inches above vessel zero. This represents a reduction of 20 inches below the current level. By reducing the low water level setpoint TVA hopes to avoid isolations due to low water level during turbine trips at high power.

The low water level setpoint, which is commonly called the L_2 setpoint, is that reactor water level below which the main steamline isolation valves close, HPCI and RCIC flows are initiated, and the recirculation pumps trip.

Lowering L_2 by 20 inches would mean that those system functions could be initiated later in time during any transient or accident involving reduction in water level. For the Browns Ferry reactors the most severe events involving water level reduction are LOCA, steamline breaks, feedwater pump trip, loss of offsite or auxiliary power, MSIV closure, turbine trip, load rejection, and pressure regulator failure. The worst case LOCA and the most severe of the anticipated transients, the loss of feedwater flow, have been analyzed to determine the effect of the proposed reduction in L_2 on plant safety (References 1 and 2). Our evaluation of these analyses is presented in the following discussion.

2.0 Discussion

2.1 ECCS Performance With the L_2 Setpoint at 470 Inches

To justify that the Browns Ferry ECCS performance will remain acceptable with the new L_2 setpoint, TVA has calculated and provided the maximum changes in LOCA peak clad temperature (PCT) expected to result from the setpoint reduction.

A wide spectrum of break sizes and break locations has been analyzed using approved calculational methods and input. For the large breaks analyzed, reduction of the L_2 setpoint resulted in increases in PCT which were in each case less than 20°F, and for the small breaks the largest increase in PCT was 15°F (Reference 2).

Peak clad temperatures associated with the worst small breaks (less than 1 ft^2) are below the large break values by much more than 15°F (Reference 3 for BF3 and Reference 6 for BF1 and BF2). This means that reduction of L_2, which would involve an increase in small break PCT no more than 15°F, could not cause any small break LOCA to become the worst case.

For each of the three plants the worst break with the current L_2 setpoint is a DBA size break in the suction side of the recirculation line, and the worst single failure is the failure of the LPCI injection valve. In Reference 7, TVA described the extent and result of the large break analyses which were performed to evaluate the possibility that the proposed change in L_2 might affect the nature of the worst break. Based on our review of that information, we concluded that sufficient analyses have been completed to demonstrate that reduction in L_2 by 20 inches would not change the size or location of the worst large break, and that the worst single failure would also remain the same. This conclusion is valid for BF1 and BF2 which are LPCI modified plants and for BF3 which has loop selection logic.

The staff has recently completed a reevaluation (after correction to errors in ECCS model and data) of peak clad temperatures for Browns Ferry Units Nos. 1 and 2 as part of our evaluation of the initial core refuelings (reloads) for these facilities (references 4 and 5). For BF1 and BF2, the maximum PCT is 2151°F.

For BF3 the PCT has been calculated to be 2030°F (Reference 3). Although errors have been identified in both the input data and ECCS evaluation model on which the BF3 PCT is based (Reference 10), we have concluded that these errors have opposite effects on the calculated PCT and that the corrected value would not be significantly above 2030°F. Since the March 10, 1977 Orders to all licensees with BWR facilities, we have completed our evaluation of 16 revised ECCS analyses submitted in response to those Orders. In all but one case, the revised PCT was decreased as a result of the correction in the ECCS model errors. (In the one case, the PCT increased by 3°F) Based on this information, we conclude that when the ECCS analysis for Browns Ferry Unit 3 is revised to account for the model errors, the PCT will probably decrease but certainly will not increase by more than 20°F. Thus, at the very worst, we can conclude that the present PCT for BF3 is no more than 2050°F for the worst break condition.

We conclude that addition of 20°F to the current PCT values adequately represents the effect of the proposed L_2 setpoint reductions. Therefore, the resulting PCT for each of the three Browns Ferry reactors would remain below the 2200°F safety limit. On this basis, we conclude that the proposed reduction in L_2 is acceptable in terms of its possible effect on ECCS performance.

2.2 Effect of Reduction in L_2 on Results of Anticipated Transients

MCPR reductions or LHGR increases during anticipated transients are affected by the L_2 setpoint only through the recirculation pump trip which would occur on low water signal during turbine trips or load rejection transients. However, the analyses performed to determine the MCPR reduction due to these events do not take credit (and not taking credit is conservative) for the recirculation pump trip (References 8 and 9). It has not been necessary, therefore, to determine the effect of the proposed L_2 reduction on MCPR or LHGR limits.

Because the MSIV closure-flux scram event, which demonstrated compliance with the ASME Code requirements on peak vessel pressure, does not take credit for any L_2 trips, this event has not been re-analyzed.

We have considered the possibility that the reduction in L_2 could involve an increase in the release of fission products associated with a break in the steamline outside containment. However, the isolation signal caused by water level below the L_2 setpoint would be preceeded in time by either two or three other independent isolation signals, depending on the break size. Even if a break too small to result in MSIV flows above the high MSIV flow isolation setpoint should occur, isolation would be initiated by either high temperature or high radiation levels in the steamline tunnel before the water level drops to the L_2 setpoint. On this basis we conclude that the potential consequences of postulated steamline breaks will not increase due to the reduction in L_2.

To provide assurance that no anticipated transient would result in uncovery of the top of the active fuel, those transients involving reduction in reactor water inventory have been reviewed. Of the transients of this type mentioned in Section 1 of this report, information in Section 14.54 of the FSAR shows that the most severe is the loss of feedwater flow due to a feedwater pump trip. The feedwater pump trip has been evaluated with the proposed lower L_2 setpoint. The minimum water level which would be reached should such a transient occur would be 60 inches above the top of the active fuel (Reference 2). This represents only a 10 inch decrease from the minimum water level without the change in L_2.

On the basis that MCPR reduction, LHGR, and MSIV-closure-Flux Scram are not sensitive to the L_2 setpoint, we conclude that new analyses of MCPR, LHGR or pressure limits are not necessary. We have also concluded that the consequences of steam line breaks will not increase. Furthermore, on the basis of the evaluation provided by TVA of the most severe water level reduction transient, showing that a feedwater pump trip would not result in uncovery of the top of the active fuel, we conclude that the margin between the minimum water level and the top of the active fuel with the proposed L_2 setpoint is acceptable. These conclusions are valid for all three of the Browns Ferry units which are identical with regard to the transients of interest.

3.0 Evaluation

Based on our review of TVA's analyses of the worst case LOCA and the most severe anticipated transient assuming the proposed L_2 set point, we conclude that the reduction of the low water level set point by 20 inches is acceptable. The change will not involve a significant decrease in safety margins or a significant increase in the probability or consequences of any accident or transient.

4.0 Environmental Considerations

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

5.0 Conclusion

We have concluded that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 2, 1978

References

1. Letter from H. G. Parris of TVA to B. C. Rusche of NRC, February 24, 1977.

2. Letter from J. E. Gilliland of TVA to G. Lear of NRC, May 23, 1978.

3. Supplement No. 8 to the Safety Evaluation Report by the Division of Project Management, Office of Nuclear Reactor Regulation, USNRC, in the Matter of TVA Browns Ferry Nuclear Plant, Units 1, 2 and 3, Docket Nos. 50-259, 50-260 and 50-296.

4. Amendment No. 35 to Facility License No. DPR-33 dated January 10, 1978 approving operation of Browns Ferry Unit No. 1 in cycle 2 and terminating the Commission's Order for Modification of License dated March 11, 1977 based on the acceptability of a revised ECCS analysis.

5. Amendment No. 35 to Facility License No. DPR-52 dated June 21, 1978 approving operation of Browns Ferry Unit No. 2 in cycle 2 and terminating the Order for Modification of License dated March 11, 1977 relating to ECCS reevaluation.

6. Safety Evaluation Report Approving Operation of Browns Ferry Units 1 and 2 with Four of the Six ADS Valves Operable, May 1978.

7. Letter from J. E. Gilliland of TVA to T. A. Ippolito of NRC, July 19, 1978.

8. General Electric "Generic Reload Fuel Application", NEDE 24011P, May 1977.

9. Safety Evaluation for the General Electric Topical Report, "Generic Reload Fuel Application" (NEDE-24011-P), April 1978.

10. Letter, A. Schwencer, NRC, to Godwin Williams, Jr., TVA dated March 11, 1977 transmitting "Order for Modification of License No DPR-68" for Browns Ferry Nuclear Plant, Unit No. 3.

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 40 to Facility Operating License No. DPR-33, Amendment No. 38 to Facility Operating License No. DPR-52 and Amendment No. 14 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

The amendments change the Technical Specifications to lower the reactor low water level setpoint from 490 inches to 470 inches.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated February 24, 1977, as supplemented by letter dated May 23, 1978, (2) Amendment No. 40 to License No. DPR-33, Amendment No. 38 to License No. DPR-52, and Amendment No. 14 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 2 day of August 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors