January 2, 1991

Docket No. 50-260

Mr. Oliver D. Kingsley, Jr.
Senior Vice President, Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 77279) (TS 291)

The Commission has issued the enclosed Amendment No. 183, to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2. This amendment is in response to your application dated August 6, 1990, as supplemented October 9, 1990.

The amendment changes the Technical Specifications (TS) to incorporate a revised trip setpoint for the Level 1 low reactor pressure vessel (RPV) water level based on new calculational methodology. Specifically, TS Section 2.1.C. Tables 3.2.A, 3.2.B and 3.7.A and associated Bases are revised to specify a limiting safety system setting (LSSS), and the related trip setpoint of 398 inches above vessel zero (AVZ). In addition, the Safety Limit specified in TS Section 1.1.C and the associated Bases have been revised to reflect the new GE analytical limit of 372.5 AVZ.

The provisions of temporary TS Amendment No. 158 issued December 15, 1988, are also deleted by this amendment.

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

Sincerely,

Thierry M. Ross, Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 183 to License No. DPR-52
2. Safety Evaluation

cc w/enclosures:
See next page

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PDR ADOCK 05000260
P PDR

SNewberry
12/26/90
for S. Newberry

	:PDII-4/LA	PDII-4/PM	:PDII-1/PE	:OGC	PDII-4/DD	:PD II-4/D	:
NAME	:MKrebs MK	:TRoss	:DHMoran	:CBachmann	FOR SBlack	:FHebdon	:
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AMENDMENT NO. 183 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260
DATED: January 2, 1991

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 183
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 August 6, 1990, as supplemented October 9, 1990, 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. 183, 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-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 2, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 183

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 pages* are provided to maintain document completeness.

REMOVE	INSERT
1.1/2.1-5	1.1/2.1-5
--	1.1/2.1-5a
1.1/2.1-10	1.1/2.1-10
1.1/2.1-11	1.1/2.1-11*
3.2/4.2-7	3.2/4.2-7
3.2/4.2-8	3.2/4.2-8*
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
3.2/4.2-65	3.2/4.2-65
3.2/4.2-66	3.2/4.2-66*
3.7/4.7-25	3.7/4.7-25
3.7/4.7-26	3.7/4.7-26*
3.7/4.7-30	3.7/4.7-30
3.7/4.7-31	3.7/4.7-31*
3.7/4.7-49	3.7/4.7-49
3.7/4.7-50	3.7/4.7-50*

*Denotes overleaf or spillover page.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1.B. Power Transient

To ensure that the Safety Limits established in Specification 1.1.A are not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

C. Reactor Vessel Water Level

Whenever there is irradiated fuel in the reactor vessel, the water level shall be greater than or equal to 372.5 inches above vessel zero.

2.1.B. Power Transient Trip Settings

1.	Scram and isolation (PCIS groups 2,3,6) reactor low water level	≥ 538 in. above vessel zero
2.	Scram--turbine stop valve closure	≤ 10 percent valve closure
3.	Scram--turbine control valve fast closure or turbine trip	≥ 550 psig
4.	(Deleted)	
5.	Scram--main steam line isolation	≤ 10 percent valve closure
6.	Main steam isolation valve closure --nuclear system low pressure	≥ 825 psig

C. Water Level Trip Settings

1.	Core spray and LPCI actuation-- reactor low water level	≥ 398 in. above vessel zero
2.	HPCI and RCIC actuation-- reactor low water level	≥ 470 in. above vessel zero
3.	Main steam isolation valve closure-- reactor low water level	≥ 398 in. above vessel zero

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1.1 BASES (Cont'd)

The safety limit has been established at 372.5 inches above vessel zero to provide a point which can be monitored and also provide adequate margin to assure sufficient cooling.

REFERENCE

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10938.

2. General Electric Document No. EAS-65-0687, Setpoint Determination for Browns Ferry Nuclear Plant, Revision 2.

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed throughout the spectrum of planned operating conditions up to the design thermal power condition of 3,440 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 3,293 MWt is the licensed maximum power level of Browns Ferry Nuclear Plant, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The void reactivity coefficient and the scram worth are described in detail in Reference 1.

The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications as further described in Reference 1. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5 percent and 20 percent insertion. By the time the rods are 60 percent inserted, approximately four dollars of negative reactivity has been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50 percent and 90 percent insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients a MCPR > limits specified in Specification 3.5.k is conservatively assumed to exist prior to initiation of the transients. This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level(6) (LIS-3-203 A-D)	≥ 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 (PS-68-93 and -94)	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)	≥ 398" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PIS-64-56A-D)	≤ 2.5 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

TABLE 3.2.A (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
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 (PIS-1-72, 76, 82, 86)	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2(12)	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.
2(14)	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
2	Instrument Channel - Reactor Water Cleanup System Pipe Trench	≤ 150°F	C	1. Same as above
1	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	≤ 100 mr/hr or downscale	G	1. 1 upscale or 2 downscale will a. Initiate SGTS b. Isolate reactor zone and refueling floor. c. Close atmosphere control system.

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 initiated HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	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, 105 sec. delay timer and CSS or RHR pump running, initiates ADS. 2. Below trip settings, in conjunction with low reactor water level permissive, 105 sec. delay timer, 12 1/2 min. delay timer, 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.
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.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Drywell High Pressure (PIS-64-58 E-H)	$1 \leq p \leq 2.5$ psig	A	1. Below trip setting prevents inadvertent operation of containment spray during accident conditions.
2	Instrument Channel - Drywell High Pressure (PIS-64-58 A-D)	$\leq$ 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	Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	$\leq$ 2.5 psig	A	1. Above trip setting in conjunction with low reactor pressure initiates LPCI.
2(16)	Instrument Channel - Drywell High Pressure (PIS-64-57A-D)	$\leq$ 2.5 psig	A	1. Above trip setting, in conjunction with low reactor water level, low reactor water level permissive, 105 sec. delay timer 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.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location (there are 4 locations) in the steam line space.

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 (≥ 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 1 RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 85 percent within four hours.

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 setpoints 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 538 inches above vessel zero closes isolation valves in the RHR System, Drywell and Suppression Chamber exhausts and drains and Reactor Water Cleanup Lines (Groups 2 and 3 isolation valves). The low reactor water level instrumentation that is set to trip when reactor water level is 470 inches above vessel zero (Table 3.2.B) trips the recirculation pumps and initiates the RCIC and HPCI systems. The RCIC and HPCI system initiation opens the turbine steam supply valve which in turn initiates closure of the respective drain valves (Group 7).

The low water level instrumentation set to trip at $\geq$ 398 inches above vessel zero (Table 3.2.A) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). 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 $\geq$ 398 inches above vessel zero (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 postaccident 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.

ADS provides for automatic nuclear steam system depressurization, if needed, for small breaks in the nuclear system so that the LPCI and the CSS can operate to protect the fuel from overheating. ADS uses six of the 13 MSRVs to relieve the high pressure steam to the suppression pool. ADS initiates when the following conditions exist: low reactor water level permissive (level 3), low reactor water level (level 1), high drywell pressure or the high drywell pressure bypass timer timed out (12 1/2 min.), and a 105 second time delay. In addition, at least one RHR pump or two core spray pumps must be running.

The high pressure bypass timer is added to meet the requirements of NUREG 0737, Item II.K.3.18. This timer will bypass the high drywell pressure permissive after a sustained low water level. The worst case condition is a main steam line break outside primary containment with HPCI inoperable. With the bypass timer set at 15 minutes, a Peak Cladding Temperature (PCT) of 1424° F is reached for the worst case event. This temperature is well below the limiting PCT of 2200° F.

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 percent 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

TABLE 3.7.A
PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
1	Main steamline isolation valves (FCV-1-14, -26, -37, & -51; 1-15, -27, -38, & -52)	4	4	3 < T < 5	0	GC
1	Main steamline drain isolation valves (FCV-1-55 & 1-56)	1	1	15	0	GC
1 *	Reactor water sample line isolation valves	1	1	5	C	SC
2	RHRS shutdown cooling supply isolation valves (FCV-74-48 & -47)	1	1	40	C	SC
2	RHRS - LPCI to reactor (FCV-74-53 & -67)		2	40	C	SC
2	RHRS flush and drain vent to suppression chamber (FCV-74-102, -103, -119, & -120)		4	20	C	SC
2	Suppression chamber drain (FCV-75-57 & -58)		2	15	0**	GC
2	Drywell equipment drain discharge isolation valves (FCV-77-15A & -15B)		2	15	0	GC
2	Drywell floor drain discharge isolation valves (FCV-77-2A & -2B)		2	15	0	GC

*These valves isolate only on reactor vessel low low low water level (≥ 398") and main steam line high radiation of Group 1 isolations.

**These valves are normally open when the pressure suppression head tank is aligned to serve the RHR and CS discharge piping and closed when the condensate head tank is used to serve the RHR and CS discharge piping. (See Specification 3.5.H)

TABLE 3.7.A (Continued)

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
3	Reactor water cleanup system supply isolation valves (FCV-69-1 & -2)	1	1	30	0	GC
4	FCV-73-81 (Bypass around FCV-73-3)		1	10	0	GC
4	HPCIS steamline isolation valves (FCV-73-2 & -3)	1	1	20	0	GC
5	RCICS steamline isolation valves (FCV-71-2 & -3)	1	1	15	0	GC
6	Drywell nitrogen purge inlet isolation valves (FCV-76-18)		1	5	C	SC
6	Suppression chamber nitrogen purge inlet isolation valves (FCV-76-19)		1	5	C	SC
6	Drywell main exhaust isolation valves (FCV-64-29 & -30)		2	2.5	C	SC
6	Suppression chamber main exhaust isolation valves (FCV-64-32 & -33)		2	2.5	C	SC
6	Drywell/suppression chamber purge inlet (FCV-64-17)		1	2.5	C	SC
6	Drywell atmosphere purge inlet (FCV-64-18)		1	2.5	C	SC

NOTES FOR TABLE 3.7.A

Key: 0 = Open
C = Closed
SC = Stays Closed
GC = Goes Closed

Note: Isolation groupings are as follows:

Group 1: The valves in Group 1 are actuated by any one of the following conditions:

1. Reactor Vessel Low Low Low Water Level (≥ 398")
2. Main Steamline High Radiation
3. Main Steamline High Flow
4. Main Steamline Space High Temperature
5. Main Steamline Low Pressure

Group 2: The valves in Group 2 are actuated by any of the following conditions:

1. Reactor Vessel Low Water Level (538")
2. High Drywell Pressure

Group 3: The valves in Group 3 are actuated by any of the following conditions:

1. Reactor Low Water Level (538")
2. Reactor Water Cleanup System High Temperature
3. Reactor Water Cleanup System High Drain Temperature

Group 4: The valves in Group 4 are actuated by any of the following conditions:

1. HPCI Steamline Space High Temperature
2. HPCI Steamline High Flow
3. HPCI Steamline Low Pressure
4. HPCI Turbine Exhaust Diaphragm High Pressure

Group 5: The valves in Group 5 are actuated by any of the following conditions:

1. RCIC Steamline Space High Temperature
2. RCIC Steamline High Flow
3. RCIC Steamline Low Pressure
4. RCIC Turbine Exhaust Diaphragm High Pressure

Group 6: The valves in Group 6 are actuated by any of the following conditions:

1. Reactor Vessel Low Water Level (538")
2. High Drywell Pressure
3. Reactor Building Ventilation High Radiation

NOTES FOR TABLE 3.7.A (Continued)

Group 7: The valves in Group 7 are automatically actuated by only the following condition:

1. The respective turbine steam supply valve not fully closed.

Group 8: The valves in Group 8 are automatically actuated by only the following conditions:

1. High Drywell Pressure
2. Reactor Vessel Low Water Level (538")

follow ASTM D3803. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of each filter train for a minimum of 10 hours each month will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other systems must be tested daily. This substantiates the availability of the operable systems and thus reactor operation and refueling operation can continue for a limited period of time.

3.7.D/4.7.D Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a LOCA.

Group 1 - Process lines are isolated by reactor vessel low water level ($\geq$ 398") in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in Group 1, except the reactor water sample line valves, are also closed when process instrumentation detects excessive main steam line flow, high radiation, low pressure, or main steam space high temperature. The reactor water sample line valves isolate only on reactor low water level at $\geq$ 398" or main steam line high radiation.

Group 2 - Isolation valves are closed by reactor vessel low water level (538") or high drywell pressure. The Group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the Group 2 isolation signal by a transient or spurious signal.

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low-level isolation is provided.

Groups 4 and 5 - Process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of Groups 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

Group 6 - Lines are connected to the primary containment but not directly to the reactor vessel. These valves are isolated on reactor low water level (538"), high drywell pressure, or reactor building ventilation high radiation which would indicate a possible accident and necessitate primary containment isolation.

Group 7 - Process lines are closed only on the respective turbine steam supply valve not fully closed. This assures that the valves are not open when HPCI or RCIC action is required.

Group 8 - Line (traveling in-core probe) is isolated on high drywell pressure or reactor low water level (538"). This is to assure that this line does not provide a leakage path when containment pressure or reactor water level indicates a possible accident condition.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent, an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

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

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 183 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 INTRODUCTION

By letter dated August 6, 1990, the Tennessee Valley Authority (TVA), requested changes to the Technical Specifications (TS) for Browns Ferry Nuclear Plant, Unit 2. TVA also provided additional information regarding its TS amendment request by letter dated October 9, 1990. The change is to the trip setting for the Level 1 low reactor pressure vessel (RPV) water level. During the process of generating setpoint and accuracy calculations for plant parameters for which no calculational basis could be found, it was determined that the trip setting for the Level 1 low RPV water level was not conservatively based.

2.0 DISCUSSION

A summary of the proposed changes to the reactor pressure vessel (RPV) level instruments are as follows:

- o Trip Level Setting changed from 378 to 398 inches above vessel zero (IAVZ) - LIMITING SAFETY SYSTEM SETTING (LSSS).
- o Analytical limit changed from 378 to 372.5 IAVZ - SAFETY LIMIT (SL)
- o Revise the bases section.
- o Remove exception to operability requirements for certain reactor low water level instruments during the time the RPV water level modifications are being performed.

There are no proposed TS changes to the minimum number of operable instruments, action statement, surveillance requirements for frequency of functional or calibration testing.

The reactor vessel water level 1 instruments, 2-LS-3-58A-D and 2-LIS-3-56A-D, are used to measure reactor pressure vessel (RPV) water level. At the existing setpoint, 378 IAVZ, low water level causes the initiation of the following systems:

- o containment spray system (CSS)
- o low pressure coolant injection system (LPCI)
- o main steamline isolation
- o permissive inputs to the automatic depressurization system (ADS)

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3.0 EVALUATION

The original LSSS reactor vessel low water level value in TS Table 3.2.A and Table 3.2.B was equal to the SL of 378 inches above vessel zero (IAVZ). Section 50.36 "Technical Specifications" of 10 CFR 50 requires "Where a limiting safety system setting [LSSS] is specified for a variable on which a safety limit [SL] has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

This requirement of 10 CFR 50.36 cannot be achieved if the LSSS is equal to the SL. The LSSS must be set to actuate at a higher reactor vessel water level than the SL to account for instrument inaccuracies, loop inaccuracies, response time of; instrument channels, logic relays, isolation valves closing, or motor breaker closing, pump acceleration time, and injection water flow into the reactor vessel. Therefore, the licensee proposed to change the LSSS from 378 to 398 IAVZ to assure the SL is not exceeded, and still prevent inadvertant actuation from normal operating level transients. The LSSS is 37-11/16 inches above the top of the reactor core (which is at 360-5/16 IAVZ).

The Level 1 low RPV water level trip level setting of 398 IAVZ is the limiting value that instrument setpoint can have when tested periodically, beyond which the instrument channel is declared inoperable and corrective action must be taken.

TVA stated: "The analytical limit [SL] provided by GE [General Electric Company] was used as a design input to a scaling and setpoint calculation which determined the nominal trip setpoint and trip level setting [LSSS] based on inaccuracies associated with the instrument loops. The allowance for instrument inaccuracies in determining the actual trip setpoint provides conservative assurance that the trip function will be performed at or before reaching the analytical limit [SL]."

TVA performed a Setpoint and Scaling Calculation to determine the accuracy of the instruments and loops. This accuracy was compared to the required accuracies to assure that there is sufficient margin between the setpoints and the operating limits, and the safety limits. The calculations reviewed by the staff at TVA's Rockville office were as follows:

Instrument No.	Calculation No.	Revision No.
2-LT-3-56A	ED-Q2003-88122	3
2-LT-3-56B	ED-Q2003-88123	3
2-LT-3-56C	ED-Q2003-88124	3
2-LT-3-56D	ED-Q2003-88125	3
2-LT-3-58A	ED-Q2003-880126	4
2-LT-3-58B	ED-Q2003-880127	4
2-LT-3-58C	ED-Q2003-880128	4
2-LT-3-58D	ED-Q2003-880129	4

The staff's review of the calculations verified that TVA addressed instrument and loop errors for normal operation and accident conditions associated with the following sources:

- o temperature
- o pressure
- o zero
- o span
- o repeatability
- o drift
- o power supply
- o seismic
- o radiation
- o water leg
- o condensate pot location
- o vessel growth

The vendor's errors were extrapolated to 18 months plus 25%, which is 22-1/2 months. This is the maximum calibration interval.

The methodology for determination of instrument setpoints used by TVA was in accordance with Regulatory Guide (RG) 1.105 that endorses Instrument Society of America (ISA) Standard ISA-S67.04 - 1982 "Setpoint for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants". This standard provides guidance for ensuring that setpoints stay within TS limits.

The level instruments affected by this amendment, specified in the Setpoint and Scaling Calculations, are Rosemount models that have been identified in a 10 CFR Part 21 report, submitted by Rosemount, and NRC Bulletin 90-01 as being susceptible to failure under certain conditions. This failure is caused by leaking silicon oil from between the isolating diaphragm and sensing diaphragm of the instrument. The loss of silicon oil causes the transmitters to exhibit reduced performance (output shift, lack of response over their full range, and/or increase in response time) prior to detectable failure. The safety concern is a common cause failure since the redundant instruments are the same manufacture and model. Although there has not been a TS change to the surveillance frequency, TVA has committed to comply with the Bulletin and Rosemount Technical Bulletin Number 4. TVA's program included the development of procedures for increased surveillance. These procedures are Procedure Method PM89-02 R1 (EE) - "Handling of Rosemount Transmitters" and Procedure No. SII-2-XT-00-165, R0 - "Rosemount Transmitter Special Monitoring Program."

In the NRC letter of September 7, 1990, TVA was requested to provide documentation (surveying records) for the RPV zero elevation which is the reference for all water level instruments. TVA responded to this request by making available, at their Rockville office, a copy of a General Electric drawing 729E424 "Nuclear Boiler Vessel Instruments." The staff reviewed Revision 11 of this drawing and verified that the elevation listed on the drawing as 578 feet three inches above sea level was used in the calculations, and the drawing is referenced in the calculation in the "Source of Design Input Information (References)."

On November 15, 1990, TVA informed the staff "that documentation of the BFN Unit 2 vessel [reactor pressure vessel] zero elevation is on file in the BFN licensing TS record file." The pertinent information excerpted from the documentation dated September 14, 1971, and signed by H.L. Johnson was reviewed by the staff. The RPV zero elevation agreed with the elevation on drawing 729E424, Revision 11. The staff has no further concerns about the BFN Unit 2 RPV zero elevation.

Changing the LSSS will revise: Table 3.2.A instruments 2-LIS-3-56A-D; Table 3.2.B instruments 2-LS-3-58A-D; a Table 3.7.A note; and Bases Sections 3.2 and 3.7/4.7.

In this TS amendment, TVA is also deleting information which was added to the TS as a temporary amendment. The temporary amendment was requested in their application dated October 14, 1988, which the staff issued as Amendment No. 158 in a letter to TVA dated December 15, 1988. Amendment No. 158 modified the Limiting Condition for Operation which required specific conditions to be met when work involving the reactor vessel was being performed. The specific instruments involved in the change to the OPERABLE definition were level instruments 2-LIS-3-203A-D and 2-LIS-3-58A-D. The work on the reactor vessel involved installation of instrumentation for detection of inadequate core cooling in accordance with the NUREG-0737, Item II.F.2. Modifications were also made to the instrument sensing in accordance with TVA's response to Generic Letter 84-23. This work has been completed and Amendment 158 is no longer necessary.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (55 FR 36353). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

Based on our review of the material submitted by the licensee, we find the proposed changes acceptable. The proposed changes to the LSSS and SL settings are acceptable because they are based on a value derived by approved calculational means. This change ensures that trips occur within the analytical limit used to confirm the design bases of the plant. The deletion of a temporary amendment which modified the LCO requiring specific conditions be met when work involving the reactor vessel was performed is acceptable because the time for its need has passed.

TVA has a program to address the transmitter problems identified in Rosemount 10 CFR Part 21 report and NRC Bulletin 90-01. The adequacy of this program will be determined by the staff under separate correspondence and outside the scope of this safety evaluation.

The staff has no further concerns about the RPV zero elevation documentation since the survey records are now in the BFN licensing TS files.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

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