January 10, 19

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. 77144) (TS 290)

The Commission has issued the enclosed Amendment No. 187, to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant, Unit 2 (BFN2). This amendment is in response to your application dated July 13, 1990, as supplemented September 17, 1990. The amendment changes the BFN2 Technical Specifications (TS) to incorporate new allowable high temperature isolation setpoint values for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems.

The NRC staff has included in the enclosed copy of the safety evaluation, a recommendation regarding inoperable channels. The NRC staff suggests that TVA evaluate its practice of placing the inoperable channel in the tripped condition for extended periods, because of the potential effect on RCIC and HPCI system reliability. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by
Frederick J. Hebdon
for
Thierry M. Ross, Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 187 to
   License No. DPR-52
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:PDII-4/LA	PDII-4/PE	PDII-4/PM	OGC	PDII-4/DD	PDII-4/D	
NAME	:MKrebs MK	:BMozafari:as	TRoss	:	SBlack	FHebdon	
DATE	:12/17/90	:12/17/90	:12/18/90	:12/21/90	:12/26/90	:12/26/90	:

CP1

Mr. Oliver D. Kingsley, Jr.

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AMENDMENT NO. 187 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260
DATED: January 10, 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. 187
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 July 13, 1990 as supplemented September 17, 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. 187 , 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, NRR
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 10, 1991

ATTACHMENT TO LICENSE AMENDMENT NO.

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
3.2/4.2-18	3.2/4.2-18
3.2/4.2-19	3.2/4.2-19
3.2/4.2-22	3.2/4.2-22*
-	3.2/4.2-22a
3.2/4.2-23	3.2/4.2-23
3.2/4.2-24	3.2/4.2-24*
3.2/4.2-44	3.2/4.2-44*
3.2/4.2-45	3.2/4.2-45
3.2/4.2-46	3.2/4.2-46
3.2/4.2-47	3.2/4.2-47
3.2/4.2-67	3.2/4.2-67
3.2/4.2-68	3.2/4.2-68*
3.2/4.2-69	3.2/4.2-69*
3.2/4.2-70	3.2/4.2-70*
3.2/4.2-71	3.2/4.2-71*
3.2/4.2-72	3.2/4.2-72*
3.2/4.2-73	3.2/4.2-73*
-	3.2/4.2-73a*

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
1	HPCI Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1(2)	Instrument Channel - Condensate Header Low Level (LS-73-55A & B)	≥ Elev. 551'	A	1. Below trip setting will open HPCI suction valves to the suppression chamber.
1(2)	Instrument Channel - Suppression Chamber High Level	≤ 7" above instrument zero	A	1. Above trip setting will open HPCI suction valves to the suppression chamber.
2(2)	Instrument Channel - Reactor High Water Level (LIS-3-208A and LIS-3-208C)	≤ 583" above vessel zero	A	1. Above trip setting trips RCIC turbine.
1	Instrument Channel - RCIC Turbine Steam Line High Flow (PDIS-71-1A and 1B)	≤ 450" H_2O (7)	A	1. Above trip setting isolates RCIC system and trips RCIC turbine.
3(2)	Instrument Channel - RCIC Steam Supply Pressure - Low (PS 71-1A-D)	≥50 psig	A	1. Below trip setting isolates RCIC system and trips RCIC turbine.
3(2)	Instrument Channel - RCIC Turbine Exhaust Diaphragm Pressure - High (PS 71-11A-D)	≤20 psig	A	1. Above trip setting isolates RCIC system and trips RCIC turbine.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2(2)	Instrument Channel - Reactor High Water Level (LIS-3-208B and LIS-3-208D)	≤583" above vessel zero.	A	1. Above trip setting trips HPCI turbine.
1	Instrument Channel - HPCI Turbine Steam Line High Flow (PDIS-73-1A and 1B)	≤90 psi (7)	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
3(2)	Instrument Channel - HPCI Steam Supply Pressure - Low (PS 73-1A-D)	≥100 psig	A	1. Below trip setting isolates HPCI system and trips HPCI turbine.
3(2)	Instrument Channel - HPCI Turbine Exhaust Diaphragm (PS 73-20A-D)	≤20 psig	A	1. Above trip setting isolates HPCI system and trips HPCI turbine.
1	Core Spray System Logic	N/A	B	1. Includes testing auto initiation inhibit to Core Spray Systems in other units.
1	RCIC System (Initiating) Logic	N/A	B	1. Includes Group 7 valves. Refer to Table 3.7.A for list of valves.
1	RCIC System (Isolation) Logic	N/A	B	1. Includes Group 5 valves. Refer to Table 3.7.A for list of valves.
1 (16)	ADS Logic	N/A	A	
1	RHR (LPCI) System (Initiation)	N/A	B	

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
1(10)	Instrument Channel – Thermostat (Core Spray Area Cooler Fan)	≤ 100°F	A	1. Above trip setting starts Core Spray area cooler fans.
1(10)	RHR Area Cooler Fan Logic	N/A	A	
1(10)	Core Spray Area Cooler Fan Logic	N/A	A	
1(11)	Instrument Channel – Core Spray Motors A or D Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(11)	Instrument Channel – Core Spray Motor B or C Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(12)	Instrument Channel – Core Spray Loop 1 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(12)	Instrument Channel – Core Spray Loop 2 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(13)	RHRSW Initiate Logic	N/A	(14)	
1	RPT Logic	N/A	(17)	1. Trips recirculation pumps on turbine control valve fast closure or stop valve closure > 30% power.

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
1(16)	ADS Timer	t≤115 sec.	A	1. Above trip setting in conjunction with low reactor water level permissive, low reactor water level; high drywell pressure or ADS high drywell pressure bypass timer timed out, and RHR or CSS pumps running, initiates ADS.
1(16)	ADS High Drywell Pressure Bypass Timer	t≤322 sec.	A	1. Above trip setting, in conjuntion with low reactor water level permissive, low reactor water level, ADS timer timed out and RHR or CSS pumps running, initiates ADS.
2	RCIC Steam Line Space Torus Area High Temperature	≤155° F	E	1. Above trip setting isolates RCIC system and trips RCIC turbine.
2	RCIC Steam Line Space RCIC Pump Room Area High Temperature	≤180° F	E	1. Above trip setting isolates RCIC system and trips RCIC turbine.
2	HPCI Steam Line Space Torus Area High Temperature	≤180° F	E	1. Above trip setting isolates HPCI system and trips HPCI turbine.
2	HPCI Steam Line Space HPCI Pump Room Area High Temperature	≤200° F	E	1. Above trip setting isolates HPCI system and trips HPCI turbine.

NOTES FOR TABLE 3.2.B

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

   Action:

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

   B. Declare the system or component inoperable.

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

   D. No action required; indicators are considered redundant.

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

2. In only one trip system.

3. Not considered in a trip system.

4. Deleted.

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

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

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

8. Note 1 does not apply to this item.

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

NOTES FOR TABLE 3.2.B (Cont'd)

10. Only one trip system for each cooler fan.

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

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

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

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

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

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

17. Two RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds two consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if one RPT system is inoperable for more than 72 hours, an orderly power reduction shall be initiated and reactor power shall be less than 85 percent within four hours.

TABLE 4.2.B

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel Reactor Low Water Level (LIS-3-58A-D)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel Reactor Low Water Level (LIS-3-184 & 185)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel Reactor Low Water Level (LIS-3-52 & 62A)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel Drywell High Pressure (PIS-64-58E-H)	(1) (27)	Once/18 Months (28)	none
Instrument Channel Drywell High Pressure (PIS-64-58A-D)	(1) (27)	Once/18 Months (28)	none
Instrument Channel Drywell High Pressure (PIS-64-57A-D)	(1) (27)	Once/18 Months (28)	none
Instrument Channel Reactor Low Pressure (PIS-3-74A&B, PS-3-74A&B) (PIS-68-95, PS-68-95) (PIS-68-96, PS-68-96)	(1) (27)	Once/6 Months (28)	none

TABLE 4.2.B (Continued)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Core Spray Auto Sequencing Timers (Normal Power)	(4)	Once/operating cycle	none
Core Spray Auto Sequencing Timers (Diesel Power)	(4)	Once/operating cycle	none
LPCI Auto Sequencing Timers (Normal Power)	(4)	Once/operating cycle	none
LPCI Auto Sequencing Timers (Diesel Power)	(4)	Once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Normal Power)	(4)	Once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Diesel Power)	(4)	Once/operating cycle	none
ADS Timer	(4)	Once/operating cycle	none
ADS High Drywell Pressure Bypass Timer	(4)	Once/operating cycle	none
RCIC Steam Line Space Torus Area High Temperature	(1)	Once/3 months	none
RCIC Steam Line Space RCIC Pump Room Area High Temperature	(1)	Once/3 months	none

TABLE 4.2.B (Continued)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel - RHR Pump Discharge Pressure	(1)	Once/3 months	none
Instrument Channel - Core Spray Pump Discharge Pressure	(1)	Once/3 months	none
Core Spray Sparger to RPV d/p	(1)	Once/3 months	Once/day
Trip System Bus Power Monitor	Once/operating Cycle	N/A	none
Instrument Channel - Condensate Header Low Level (LS-73-56A, B)	(1)	Once/3 months	none
Instrument Channel - Suppression Chamber High Level	(1)	Once/3 months	none
Instrument Channel - Reactor High Water Level	(1)	Once/3 months	Once/day
Instrument Channel - RCIC Turbine Steam Line High Flow	(1)	Once/3 months	none
Instrument Channel - RCIC Steam Supply Low Pressure	Once/31 days	Once/18 months	none
Instrument Channel - RCIC Turbine Exhaust Diaphragm High Pressure	Once/31 days	Once/18 months	none
HPCI Steam Line Space Torus Area High Temperature	(1)	Once/3 months	none
HPCI Steam Line Space HPCI Pump Room Area High Temperature	(1)	Once/3 months	none

TABLE 4.2.B (Continued)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel - HPCI Turbine Steam Line High Flow	(1)	Once/3 months	none
Instrument Channel - HPCI Steam Supply Low Pressure	Once/31 days	Once/18 months	none
Instrument Channel - HPCI Turbine Exhaust Diaphragm High Pressure	Once/31 days	Once/18 months	none
Core Spray System Logic	Once/18 months	(6)	N/A
RCIC System (Initiating) Logic	Once/18 months	N/A	N/A
RCIC System (Isolation) Logic	Once/18 months	(6)	N/A
HPCI System (Initiating) Logic	Once/18 months	(6)	N/A
HPCI System (Isolation) Logic	Once/18 months	(6)	N/A
ADS Logic	Once/18 months	(6)	N/A
LPCI (Initiating) Logic	Once/18 months	(6)	N/A
LPCI (Containment Spray) Logic	Once/18 months	(6)	N/A
Core Spray System Auto Initiation Inhibit (Core Spray Auto Initiation)	Once/18 months (7)	N/A	N/A
LPCI Auto Initiation Inhibit (LPCI Auto Initiation)	Once/18 months (7)	N/A	N/A

flow instrumentation is a backup to the temperature instrumentation. In the event of a loss of the reactor building ventilation system, radiant heating in the vicinity of the main steam lines raises the ambient temperature above 200°F. The temperature increases can cause an unnecessary main steam line isolation and reactor scram. Permission is provided to bypass the temperature trip for four hours to avoid an unnecessary plant transient and allow performance of the secondary containment leak rate test or make repairs necessary to regain normal ventilation.

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 nominal setting of three 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 setpoint 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.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1-out-of-2 logic, and all sensors are required to be OPERABLE.

High temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system. Each trip system consists of two elements. Each channel contains one temperature switch located in the pump room and three temperature switches located in the torus area. The RCIC high flow and high area temperature sensing instrument channels are arranged in the same manner as the HPCI system.

The HPCI high steam flow trip setting of 90 psid and the RCIC high steam flow trip setting of 450" H_2O have been selected such that the trip setting is high enough to prevent spurious tripping during pump startup but low enough to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

The HPCI and RCIC steam line space temperature switch trip settings are high enough to prevent spurious isolation due to normal temperature excursions in the vicinity of the steam supply piping. Additionally, these trip settings ensure that the primary containment isolation steam supply valves isolate a break within an acceptable time period to prevent core uncovery and maintain fission product releases within 10 CFR 100 limits.

High temperature at the Reactor Water Cleanup (RWCU) System floor drain in the space near the RWCU system or in the space near the pipe trench containing RWCU piping could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two post treatment off-gas radiation monitors are provided and, when their trip point is reached, cause an isolation of the off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip or both have a downscale trip.

Both instruments are required for trip but the instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

Four radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the reactor zone ventilation exhaust ducts and in the refueling zone.

Trip setting of 100 mr/hr for the monitors in the refueling zone are based upon initiating normal ventilation isolation and SGTS operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the SGTS.

Flow integrators and sump fill rate and pump out rate timers are used to determine leakage in the drywell. A system whereby the time interval to fill a known volume will be utilized to provide a backup. An air sampling system is also provided to detect leakage inside the primary containment (See Table 3.2.E).

For each parameter monitored, as listed in Table 3.2.F, there are two channels of instrumentation except as noted. By comparing readings between the two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

Instrumentation is provided for isolating the control room and initiating a pressurizing system that processes outside air before supplying it to the control room. An accident signal that isolates primary containment will also automatically isolate the control room and initiate the emergency pressurization system. In addition, there are radiation monitors in the normal ventilation system that will isolate the control room and initiate the emergency pressurization system. Activity required to cause automatic actuation is about one mRem/hr.

Because of the constant surveillance and control exercised by TVA over the Tennessee Valley, flood levels of large magnitudes can be predicted in advance of their actual occurrence. In all cases, full advantage will be taken of advance warning to take appropriate action whenever reservoir levels above normal pool are predicted; however, the plant flood protection is always in place and does not depend in any way on advanced warning. Therefore, during flood conditions, the plant will be permitted to operate until water begins to run across the top of the pumping station at elevation 565. Seismically qualified, redundant level switches each powered from a separate division of power are provided at the pumping station to give main control room indication of this condition. At that time an orderly shutdown of the plant will be initiated, although surges even to a depth of several feet over the pumping station deck will not cause the loss of the main condenser circulating water pumps.

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the seismic response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for Browns Ferry Nuclear Plant and to determine whether the plant can continue to be operated safely. The instrumentation provided is consistent with specific portions of the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes."

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments will be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentration of potentially explosive gas mixtures in the off-gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with guidance provided in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20 Appendix B, Table II, Column 2. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3.2 BASES (Cont'd)

ATWS/RPT, Anticipated Transients without Scram/Recirculation Pump Trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an ATWS event. The response of the plant to this postulated event (ATWS/RPT) follows the BWR Owners Group Report by General Electric NEDE-31096-P-A and the accompanying NRC Staff Safety Evaluation Report.

ATWS/RPT utilizes the engineered safety feature (ESF) master/slave analog trip units (ATU) which consists of four level and four pressure channels total. The initiating logic consists of two independent trip systems each consisting of two reactor dome high pressure channels and two reactor vessel low level channels. A coincident trip of either two low levels or two high pressures in the same trip system causes initiation of ATWS/RPT. This signal from either trip system opens one of two EOC (end-of-cycle) breakers in series (the other system opens the other breaker) between the pump motor and the Motor Generator set driving each recirculation pump. Both systems are completely redundant such that only one trip system is necessary to perform the ATWS/RPT function. Power comes from the 250 VDC shutdown boards.

Setpoints for reactor dome high pressure and reactor vessel low level are such that a normal Reactor Protection System scram and accompanying recirculation pump trip would occur before or coincident with the trip by ATWS/RPT.

4.2 BASES

The instrumentation listed in Tables 4.2.A through 4.2.F will be functionally tested and calibrated at regularly scheduled intervals. The same design reliability goal as the Reactor Protection System of 0.99999 generally applies for all applications of (1-out-of-2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bistable trips associated with analog sensors and amplifiers are tested once/week.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1-out-of-n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = \sqrt{\frac{2t}{r}}$$

Where: i = the optimum interval between tests.

t = the time the trip contacts are disabled from performing their function while the test is in progress.

r = the expected failure rate of the relays.

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hour. Using this data and the above operation, the optimum test interval is:

$$i = \sqrt{\frac{2(0.5)}{10^{-6}}} = 1 \times 10^{3}$$

$$= 40 \text{ days}$$

<u>For additional margin a test interval of once per month will be used initially.</u>

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

The above calculated test interval optimizes each individual channel, considering it to be independent of all others. As an example, assume that there are two channels with an individual technician assigned to each. Each technician tests his channel at the optimum frequency, but the two technicians are not allowed to communicate so that one can advise the other that his channel is under test. Under these conditions, it is possible for both channels to be under test simultaneously. Now, assume that the technicians are required to communicate and that two channels are never tested at the same time.

(7) UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, page 10, Equation (24), Lawrence Radiation Laboratory.

Forbidding simultaneous testing improves the availability of the system over that which would be achieved by testing each channel independently. These one-out-of-n trip systems will be tested one at a time in order to take advantage of this inherent improvement in availability.

Optimizing each channel independently may not truly optimize the system considering the overall rules of system operation. However, true system optimization is a complex problem. The optimums are broad, not sharp, and optimizing the individual channels is generally adequate for the system.

The formula given above minimizes the unavailability of a single channel which must be bypassed during testing. The minimization of the unavailability is illustrated by Curve No. 1 of Figure 4.2-1 which assumes that a channel has a failure rate of 0.1×10^{-6}/hour and 0.5 hours is required to test it. The unavailability is a minimum at a test interval i, of 3.16×10^{3} hours.

If two similar channels are used in a 1-out-of-2 configuration, the test interval for minimum unavailability changes as a function of the rules for testing. The simplest case is to test each one independent of the other. In this case, there is assumed to be a finite probability that both may be bypassed at one time. This case is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

A more usual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested, and restored, and then immediately following, the second channel be bypassed, tested, and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1-out-of-n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the refueling area ventilation duct which initiate building isolation and standby gas treatment operation are arranged in two 1-out-of-2 logic systems. The bases given for the rod blocks apply here also and were used to arrive at the functional testing frequency. The off-gas post treatment monitors are connected in a 2-out-of-2 logic arrangement. Based on experience with instruments of similar design, a testing interval of once every three months has been found adequate.

The automatic pressure relief instrumentation can be considered to be a 1-out-of-2 logic system and the discussion above applies also.

4.2 BASES (Cont'd)

The criteria for ensuring the reliability and accuracy of the radioactive gaseous effluent instrumentation is listed in Table 4.2.K.

The criteria for ensuring the reliability and accuracy of the radioactive liquid effluent instrumentaiton is listed in Table 4.2.D.

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

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 187 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 INTRODUCTION

Tennessee Valley Authority (TVA or the licensee) submitted a request on July 13, 1990 to change the Browns Ferry Nuclear Plant (BFN) Technical Specifications (TS). The NRC staff reviewed the TS amendment request (TS 290) and concluded that additional information was necessary to complete our evaluation. By letter dated August 17, 1990, the NRC requested that TVA furnish the additional information. TVA responded to the NRC's request by letter dated September 17, 1990.

2.0 BACKGROUND

The Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) turbines use steam supplied from the reactor. Excessive steam flow and local area high temperature measuring instruments are used to detect a steam line break and ensure automatic closure of each system's primary containment isolation valves. This closure prevents excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system process barrier. The requested TS changes only affect the temperature measuring instrument channels, not the flow measuring instruments, used to detect a steam line break in the RCIC or HPCI systems. The licensee does not propose to change the existing surveillance intervals for either of the functional or calibration tests.

3.0 EVALUATION

The licensee has utilized computer modeling techniques to predict the temperature response of various reactor building zones during a steam line break. The licensee's study identified temperatures below the present TS setting of 200°F that could be present for various RCIC and HPCI line break scenarios. Therefore, the licensee proposes to change the temperature setpoint in the TS.

The RCIC and HPCI systems each has four sets of four bimetallic temperature measuring instruments located in the areas along the path of the steam supply piping of each system. The sixteen switches from the temperature instruments for each system are arranged into two divisional trip logic schemes with eight temperature switches in each division. The trip logic for each division is arranged in a "one-out-of-two taken twice" logic configuration for each of the four reactor building areas being monitored.

The following are the areas and new temperature setpoints:

			Number of Instruments
o	RCIC pump room	180°F or less	4
o	RCIC torus 3 areas	155°F or less	12 (4 per area)
o	HPCI pump room	200°F or less	4
o	HPCI torus 3 areas	180°F or less	12 (4 per area)

The licensee does not propose to change the existing surveillance frequency of functional test (which is once a month) or calibration test (which is once every three months). However, there is a proposed change regarding the action to be taken for an inoperable instrument/channel. The present action statement is:

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

B. Declare the system or component inoperable.

The new minimum number of operable channels required would be "two" for each of two trip systems.

TVA proposes to replace A and B (above) for HPCI and RCIC instrument temperature channels with a new note, 1.E, for Table 3.2.B which states: "Within 24 hours restore the inoperable channel(s) to OPERABLE status or place the inoperable channel(s) in the tripped condition." The NRC requested that TVA justify the impact on safety for not having a time limit on how long an inoperable channel is placed in the tripped condition. TVA's response to the NRC's concerns stated:

> Placing a channel in the tripped position restores the isolation function to a single failure tolerant condition. Although tripping the channel does result in a condition where a spurious actuation in another channel could prevent the function of the associated system, this condition is consistent with the design of the HPCI and RCIC systems since neither of these systems are themselves single failure proof.

The NRC staff agrees that neither the HPCI or RCIC systems have redundant counterparts; however, HPCI and RCIC systems can furnish high pressure coolant to the reactor vessel; furthermore, the HPCI system has a diverse counterpart, which is the Automatic Depression System (ADS) and Low Pressure Coolant Injection System (LPCI) or Core Spray System (CSS). Placing a safety system in a less reliable condition for an extended time period cannot be justified by the fact that the system is diverse rather than redundant.

When the channel is placed in the tripped condition, an inadvertent trip of any one of eight temperature detectors of two other channels will cause isolation of the steam supply to the respective turbine drive, thereby making the system inoperable. The probability of a steam line pipe rupture in which isolation would be required is far lower than of plant transient conditions in which it may be necessary to provide water to the reactor core. Although TVA's proposed TS change does not place a time limit on when or how long a channel is placed in the tripped condition, it is in agreement with the GE Standard Technical Specification (STS) for Boiling Water Reactors (BWR/4) and provides for reliable isolation. Extended time in trip, however, could reduce the reliability of HPCI or RCIC. We note that this situation is generic and recommend that TVA evaluate this issue further. We expect that should this situation develop, there would be sufficient time to evaluate plant conditions and determine whether alternative actions are necessary. However, we recognize that this observation is beyond the scope of this safety evaluation.

The licensee discussed spurious actuation as follows: "The setpoints are established above the maximum expected room temperatures to avoid spurious action due to ambient conditions and below the analytical limits to ensure timely pipe break detection and isolation." The licensee's analysis did not address the effect of normal cooling lost in the monitored areas, rate of temperature rise, or area temperature alarm setpoints to preclude spurious steam line supply line isolation. The staff requested that the licensee address this concern. TVA provided the following response:

> Substantial margin (at least 35°F) exists between the maximum abnormal temperature expected in each area and the minimum actuation temperature determined for each temperature switch. The maximum temperature expected could occur as a result of outside temperature excursions; temporary, greater than design heat loads; or degraded environmental control system operation. With the substantial margin between maximum abnormal temperatures for the areas and the minimum actuation temperature of the switches, the maximum abnormal temperatures cannot result in actuation of the switches.

TVA has changed the column of Table 3.2.B from "Trip Level Setting" to "Allowable Value" for the temperature values. This type of change is consistent with the future STS; however a definition for allowable value will also be in the future STS. TVA will establish trip settings in plant instructions to ensure that the allowable values are not exceeded. The trip settings will also reflect a margin for instrument drift and inaccuracies.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. 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. 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

The staff finds the lower isolation system temperature detector setpoints and allowable TS values for HPCI/RCIC turbine supply steam line rupture are acceptable. These settings were also selected to preclude inadvertent action from transient conditions not associated with a steam line rupture.

It is recommended that TVA evaluate its practice of placing an inoperable channel in the tripped condition for extended periods, because of the potential for a temperature detector failure to adversely affect RCIC/HPCI system reliability.

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 36352) on September 5, 1990 and consulted with the State of Alabama. No public comments were received and the State of Alabama did not have any comments.

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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Dated: January 10, 1991