April 13, 1989

Docket Nos. 50-259/260/296

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: SEISMIC MONITORING TABLES 3.2.J AND 4.2.J OF THE TECHNICAL SPECIFICATIONS FOR BROWNS FERRY NUCLEAR PLANTS, UNITS 1, 2, AND 3 (TAC 00474, 00475, 00476) (TS 257)

The Commission has issued the enclosed Amendment Nos. 165 , 163 , and 136 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated September 29, 1988.

The amendments modify Tables 3.2.J and 4.2.J to reflect vendor's suggested testing frequencies for seismic monitoring equipment and to correct certain typographical errors found in these tables. The Bases for this seismic equipment is also changed. Table 3.2.B (pages 3.2/4.2-14 through 15), pages 3.2/4.2-65 through 69 and pages 3.2/4.2-71 through 73 of the Unit 2 TS changes are provided for continuity of the Bases Section. They do not change the substance of the application as published in the Federal Register on February 8, 1989 (54 FR 6211).

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

Sincerely,
Original signed by B. D. Liaw for

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 165 to License No. DPR-33
2. Amendment No. 163 to License No. DPR-52
3. Amendment No. 136 to License No. DPR-68
4. Safety Evaluation

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Docket Nos. 50-259/260/296

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6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. Kingsley:

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The Commission has issued the enclosed Amendment Nos. , , and to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated September 29, 1988.

The amendments modify Tables 3.2.J and 4.2.J to reflect vendor's suggested testing frequencies for seismic monitoring equipment and to correct certain typographical errors found in these tables.

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

Sincerely,

Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

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

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cc:
General Counsel
Tennessee Valley Authority
400 West Summit Hill Drive
E11 B33
Knoxville, Tennessee 37902

Mr. R. L. Gridley
Tennessee Valley Authority
5N 157B Lookout Place
Chattanooga, Tennessee 37402-2801

Mr. O. J. Zeringue
Tennessee Valley Authority
Browns Ferry Nuclear Plant
P.O. Box 2000
Decatur, Alabama 35602

Mr. P. Carier
Tennessee Valley Authority
Browns Ferry Nuclear Plant
P.O. Box 2000
Decatur, Alabama 35602

Mr. D. L. Williams
Tennessee Valley Authority
400 West Summit Hill Drive
W10 B85
Knoxville, Tennessee 37902

Chairman, Limestone County Commission
P.O. Box 188
Athens, Alabama 35611

Claude Earl Fox, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36130

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Atlanta, Georgia 30323

Resident Inspector/Browns Ferry NP
U.S. Nuclear Regulatory Commission
Route 12, Box 637
Athens, Alabama 35611

Dr. Henry Myers, Science Advisor
Committee on Interior
and Insular Affairs
U. S. House of Representatives
Washington, D.C. 20515

Tennessee Valley Authority
Rockville Office
11921 Rockville Pike
Suite 402
Rockville, Maryland 20852

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 165
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 29, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-33 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 165 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

For ~~Suzanne Black, Assistant~~ Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 165

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

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

REMOVE	INSERT
3.2/4.2-37	3.2/4.2-37
3.2/4.2-58	3.2/4.2-58
3.2/4.2-69	3.2/4.2-69*
3.2/4.2-70	3.2/4.2-70

Table 3.2.J

Seismic Monitoring Instrumentation

INSTRUMENT	MEASUREMENT RANGE	SETPOINT	MINIMUM OPERABLE
1. TRIAXIAL TIME HISTORY ACCELEROGRAPHS			
a. U-1 reactor bldg. base slab (El. 519.0)	0-1.0g	.01g	1
b. U-1 reactor bldg. floor slab (El.621.25)	0-1.0g	.01g	1
c. Diesel-gen. bldg. base slab (El.565.5)	0-1.0g	.01g	1
2. TRIAXIAL PEAK ACCELEROGRAPHS			
a. U-1 RBCCW, 10" pipe (El.625.75)	0-5.0g	N/A	1
b. U-1 RHRSW, 16" pipe (El.580.0)	0-5.0g	N/A	1
c. U-1 core spray system, 14" pipe (El.544.0)	0-5.0g	N/A	1
3. BIAXIAL SEISMIC SWITCHES			
a. U-1 reactor bldg. base slab (El. 519.0)	.025-.25g	0.1g	1*
b. U-1 reactor bldg. base slab (El. 519.0)	.025-.25g	0.1g	1*
c. U-1 reactor bldg. base slab (El. 519.0)	.025-.25g	0.1g	1*

*With control room indication

Table 4.2.J

SEISMIC MONITORING INSTRUMENT SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Channel Functional Test	Channel Calibration
1. TRIAXIAL TIME HISTORY ACCELEROGRAPHS			
a. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R
b. Unit 1 reactor bldg. floor slab (El. 621.25)	Monthly	SA	R
c. Diesel-generator bldg. base slab (El. 565.5)	Monthly	SA	R
2. TRIAXIAL PEAK ACCELEROGRAPHS			
a. U-1 RBCCW, 10" pipe (El. 625.75)	N/A	N/A	R
b. U-1 RHRSW, 16" pipe (El. 580.0)	N/A	N/A	R
c. U-1 core spray system, 14" pipe (El. 544.0)	N/A	N/A	R
3. BIAXIAL SEISMIC SWITCHES			
a. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R
b. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R
c. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R

3.2 BASES (Cont'd)

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.

3.2 BASES (Cont'd)

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 offgas 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.

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

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. 163
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 September 29, 1988, 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. 163 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

For Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 163

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. Pages with* are provided to maintain document completeness.

REMOVE	INSERT
3.2/4.2-14	3.2/4.2-14*
3.2/4.2-15	3.2/4.2-15*
3.2/4.2-37	3.2/4.2-37
3.2/4.2-58	3.2/4.2-58
3.2/4.2-65	3.2/4.2-65
3.2/4.2-66	3.2/4.2-66
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
3.2/4.2-70a	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	

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)	$\geq$ 470" above vessel zero.	A	1. Below trip setting initiated HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	$\geq$ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
* 2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	$\geq$ 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)	$\geq$ 378" 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)	$\geq$ 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 62)	$\geq$ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

* The automatic initiation capability of this instrument channel is not required to be OPERABLE while the Reactor Vessel water level monitoring modification is being performed. Manual initiation capability of the associated system will be available during that time the automatic initiation logic is out-of-service.

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≤ p≤2.5 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)	≤ 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)	≤ 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)	≤ 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.

Table 3.2.J

Seismic Monitoring Instrumentation

INSTRUMENT	MEASUREMENT RANGE	SETPOINT	MINIMUM OPERABLE
1. TRIAXIAL TIME HISTORY ACCELEROGRAPHS			
a. U-1 reactor bldg. base slab (El. 519.0)	0-1.0g	.01g	1
b. U-1 reactor bldg. floor slab (El.621.25)	0-1.0g	.01g	1
c. Diesel-gen. bldg. base slab (El.565.5)	0-1.0g	.01g	1
2. TRIAXIAL PEAK ACCELEROGRAPHS			
a. U-1 RBCCW, 10" pipe (El.625.75)	0-5.0g	N/A	1
b. U-1 RHRSW, 16" pipe (El.580.0)	0-5.0g	N/A	1
c. U-1 core spray system, 14" pipe (El.544.0)	0-5.0g	N/A	1
3. BIAXIAL SEISMIC SWITCHES			
a. U-1 reactor bldg. base slab (El. 519.0)	.025-.25g	0.1g	1*
b. U-1 reactor bldg. base slab (El. 519.0)	.025-.25g	0.1g	1*
c. U-1 reactor bldg. base slab (El. 519.0)	.025-.25g	0.1g	1*

*With control room indication

Table 4.2.J

SEISMIC MONITORING INSTRUMENT SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Channel Functional Test	Channel Calibration
1. TRIAXIAL TIME HISTORY ACCELEROGRAPHS			
a. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R
b. Unit 1 reactor bldg. floor slab (El. 621.25)	Monthly	SA	R
c. Diesel-generator bldg. base slab (El. 565.5)	Monthly	SA	R
2. TRIAXIAL PEAK ACCELEROGRAPHS			
a. U-1 RBCCW, 10" pipe (El. 625.75)	N/A	N/A	R
b. U-1 RHRSW, 16" pipe (El. 580.0)	N/A	N/A	R
c. U-1 core spray system, 14" pipe (El. 544.0)	N/A	N/A	R
3. BIAXIAL SEISMIC SWITCHES			
a. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R
b. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R
c. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R

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 378 inches above vessel zero (Table 3.2.B) 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 378 inches above vessel zero (Table 3.2.B)

3.2 **BASES** (Cont'd)

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 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.

The HPCI trip settings of 90 psi for high flow and 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of 450" H_2O for high flow and 200°F for temperature are based on the same criteria as the HPCI.

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.

3.2 <u>BASES</u> (Cont'd)

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.

3.2 BASES (Cont'd)

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.

3.2 **BASES** (Cont'd)

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 offgas 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.

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:

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

$$= 40 \text{ days}$$

4.2 <u>BASES</u> (Cont'd)

<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.

4.2 **BASES** (Cont'd)

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.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 136
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated September 29, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 136, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

For Suzanne Black, Assistant Director
for Projects
TVA Projects Division
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 136

FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

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

REMOVE	INSERT
3.2/4.2-37	3.2/4.2-37
3.2/4.2-57	3.2/4.2-57
3.2/4.2-68	3.2/4.2-68*
3.2/4.2-69	3.2/4.2-69

Table 3.2.J

Seismic Monitoring Instrumentation

INSTRUMENT	MEASUREMENT RANGE	SETPOINT	MINIMUM OPERABLE
1. TRIAXIAL TIME HISTORY ACCELEROGRAPHS			
a. U-1 reactor bldg. base slab (El. 519.0)	0-1.0g	.01g	1
b. U-1 reactor bldg. floor slab (El.621.25)	0-1.0g	.01g	1
c. Diesel-gen. bldg. base slab (El.565.5)	0-1.0g	.01g	1
2. TRIAXIAL PEAK ACCELEROGRAPHS			
a. U-1 RBCCW, 10" pipe (El.625.75)	0-5.0g	N/A	1
b. U-1 RHRSW, 16" pipe (El.580.0)	0-5.0g	N/A	1
c. U-1 core spray system, 14" pipe (El.544.0)	0-5.0g	N/A	1
3. BIAXIAL SEISMIC SWITCHES			
a. U-1 reactor bldg. base slab (El. 519.0)	.025-.25g	0.1g	1*
b. U-1 reactor bldg. base slab (El. 519.0)	.025-.25g	0.1g	1*
c. U-1 reactor bldg. base slab (El. 519.0)	.025-.25g	0.1g	1*

*With control room indication

Table 4.2.J

SEISMIC MONITORING INSTRUMENT SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Channel Functional Test	Channel Calibration
1. TRIAXIAL TIME HISTORY ACCELEROGRAPHS			
a. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R
b. Unit 1 reactor bldg. floor slab (El. 621.25)	Monthly	SA	R
c. Diesel-generator bldg. base slab (El. 565.5)	Monthly	SA	R
2. TRIAXIAL PEAK ACCELEROGRAPHS			
a. U-1 RBCCW, 10" pipe (El. 625.75)	N/A	N/A	R
b. U-1 RHRSW, 16" pipe (El. 580.0)	N/A	N/A	R
c. U-1 core spray system, 14" pipe (El. 544.0)	N/A	N/A	R
3. BIAXIAL SEISMIC SWITCHES			
a. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R
b. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R
c. Unit 1 reactor bldg. base slab (El. 519.0)	Monthly	SA	R

3.2 BASES (Cont'd)

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 offgas 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.

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

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

ENCLOSURE 4

SAFETY EVALUATION BY THE OFFICE OF SPECIAL PROJECTS

SUPPORTING AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 163 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 136 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

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

1.0 INTRODUCTION

The Tennessee Valley Authority (TVA), by submittal dated September 29, 1988, proposed to revise Tables 3.2.J and 4.2.J of the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3 Technical Specifications (TS) for seismic monitoring. The proposed revisions would reflect the manufacturer's suggested testing frequency for triaxial peak accelerographs and to correct typographical errors in these tables. Changes to the TS bases for the seismic instrumentation have also been proposed. TVA states that the new seismic instruments and the proposed TS changes will improve instrument efficiency and dependability.

2.0 EVALUATION

TVA is upgrading the seismic monitoring instrumentation for BFN, Units 1, 2, and 3. BFN TS Tables 3.2.J and 4.2.J, Seismic Monitoring Instrumentation, are being revised to reflect the suggested manufacturer's testing requirements. Specifically, the channel calibration frequency for the triaxial time history and triaxial peak accelerographs are proposed to be changed from "N/A" to "R" (once per refueling outage). The channel functional test frequency for the triaxial peak accelerographs is proposed to be changed from "12 months" to "N/A". The channel functional test frequency for the triaxial time history accelerographs and the biaxial seismic switches are proposed to be changed from "six months" to "SA" (semiannually). The calibration frequency for the biaxial seismic switches is proposed to be changed from once/operating cycle to "R". TVA also proposed to delete the note "except seismic switches" as referenced by the channel check requirements for the triaxial time history accelerographs and biaxial seismic switches. The proposed administrative TS changes include instruments listed in the seismic monitoring TS Tables, the numbering of table entries for each instrument, typographical corrections, and the addition of a reference elevation (E.R.) of 519.0 feet for each biaxial seismic switch.

The proposed administrative changes listed above provide consistency for the seismic monitoring TS Tables, correct spelling, and standardize the abbreviations for the surveillance requirement frequencies. These changes are strictly administrative in nature, do not affect safety, and are, therefore, found to be acceptable. A new requirement for performance of a channel calibration of the triaxial time history accelerographs of at least once per refueling is consistent with the General Electric Standard Technical Specifications (GE STS) and ANSI/ANS - 2.2-1978, "Earthquake Instrumentation Criteria for Nuclear Power Plants." TVA has proposed to delete the once per 12 month channel functional test requirement for the triaxial peak accelerographs and replace it with a new requirement to perform a channel calibration at least once per refueling. TVA states that the channel calibrations proposed are a more comprehensive operability verification and better serve the intent of the TS. The staff agrees that instrumentation operability would be better verified and, since the proposed frequencies are consistent with the GE STS, the NRC staff finds these proposed changes to be acceptable. The staff also notes that TVA has proposed to revise the TS bases to reflect the wording of Regulatory Guide 1.12. The staff finds these changes to be acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

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

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 6211) on February 8, 1989 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: T. Rotella

Dated: April 13, 1989