July 17, 1995

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

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 (TAC NOS. M89248, M89249, AND M89250) (TS 318)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment Nos. 222, 237, and 196 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively. These amendments are in response to your application dated March 30, 1994, and supplemented on November 21, 1994 and March 9, 1995. The amendments implement an analog transmitter/trip system on BFN Unit 3, revise the reactor vessel water level safety limit and limiting safety system setting for BFN Units 1 and 3, add instrument identifiers and revise calibration frequencies and functional test requirements for BFN Unit 2, revise the calibration frequency for instrumentation actuating the suppression chamber-reactor building vacuum breakers, and provide editorial changes.

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

Sincerely,

ORIGINAL SIGNED BY:

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

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

Enclosures:
1. Amendment No. 222 to License No. DPR-33
2. Amendment No. 237 to License No. DPR-52
3. Amendment No. 196 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures: See next page

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 222
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 March 30, 1994, and supplemented on November 21, 1994 and March 9, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 222, 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, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 17, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 222

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
vii	vii
viii	viii*
1.1/2.1-5	1.1/2.1-5
---	1.1/2.1-5a
1.1/2.1-10	1.1/2.1-10
1.1/2.1-11	1.1/2.1-11*
3.2/4.2-7	3.2/4.2-7
3.2/4.2-8	3.2/4.2-8*
3.2/4.2-14	3.2/4.2-14
3.2/4.2-15	3.2/4.2-15*
3.2/4.2-23	3.2/4.2-23*
3.2/4.2-24	3.2/4.2-24
3.2/4.2-65	3.2/4.2-65
3.2/4.2-66	3.2/4.2-66
3.7/4.7-9	3.7/4.7-9*
3.7/4.7-10	3.7/4.7-10
---	3.7/4.7-24a
---	3.7/4.7-24b
3.7/4.7-33	3.7/4.7-33*
3.7/4.7-34	3.7/4.7-34

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

FEB 24 1995

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1.B. Power Transient

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

2.1.B. Power Transient Trip Settings

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

C. Reactor Vessel Water Level

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

C. Water Level Trip Settings

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

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1.1 BASES (Cont'd)

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

REFERENCE

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

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

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1.

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level(6)	≥ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation (Groups 2, 3, and 6) c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure (PS-68-93 and 94)	100 ± 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	≥ 398" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PS-64-56A-D)	≤ 2.5 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS

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

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Low Pressure Main Steam Line	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2(12)	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.
2(14)	Instrument Channel - Reactor Water Cleanup System Floor Drain High Temperature	160 - 180°F	C	1. Above trip setting initiates Isolation of Reactor Water Cleanup Line from Reactor and Reactor Water Return Line.
2	Instrument Channel - Reactor Water Cleanup System Space High Temperature	160 - 180°F	C	1. Same as above
1(15)	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	≤ 100 mr/hr or downscale	G	1. 1 upscale channel or 2 downscale channels will a. Initiate SGTS b. Isolate reactor zone and refueling floor. c. Close atmosphere control system.

AMENDMENT NO. 212

SEP 27 1994

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

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #1)	≥ 398" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D, SW #2)	≥ 398" above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, ADS timer timed out and CSS or RHR pump running, initiates ADS. 2. Below trip settings, in conjunction with low reactor water level permissive, ADS timer timed out, ADS high drywell pressure bypass timer timed out, CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184 & 185, SW #1)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LITS-3-52 and 62, SW #1)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

TABLE 3.2.B (Continued)

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

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

   Action:

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

   B. Declare the system or component inoperable.

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

   D. No action required; indicators are considered redundant.

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

2. In only one trip system.

3. Not considered in a trip system.

4. Deleted

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

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

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

8. Note 1 does not apply to this item.

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

NOTES FOR TABLE 3.2.B (Cont'd)

10. Only one trip system for each cooler fan.

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

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

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

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

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

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

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

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

3.2 BASES

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

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

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

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

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

The low water level instrumentation set to trip at $\geq$ 398 inches above vessel zero (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). These trip settings are adequate to prevent core uncovery in the case of a break in the largest line assuming the maximum closing time.

3.2 BASES (Cont'd)

The low reactor water level instrumentation that is set to trip when reactor water level is $\geq$ 398 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

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

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

The ADS high drywell 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 ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded 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 high steam flow 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.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

j. Continuous Leak Rate Monitor

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. Drywell and Torus Surfaces

The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

LIMITING CONDITIONS FOR OPERATION

3.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c., below.

b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

SURVEILLANCE REQUIREMENTS

4.7.A PRIMARY CONTAINMENT

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

Minimum No. Operable Per Trip System	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	0.5 psid	(1)	Actuates the pressure suppression chamber-reactor building vacuum breakers.

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel-Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	Once/month(1)	Once/18 months(2)	None

Footnotes:

(1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.

(2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803-89. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

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

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

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

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

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contains the list of all the Primary Containment Isolation Valves for which the Technical Specification requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

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

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

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

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
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 March 30, 1994, and supplemented on November 21, 1994 and March 9, 1995, 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. 237, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: **July 17, 1995**

ATTACHMENT TO LICENSE AMENDMENT NO. 237

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
vii	vii
viii	viii*
3.2/4.2-39	3.2/4.2-39*
3.2/4.2-39a	3.2/4.2-39a
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
3.2/4.2-54	3.2/4.2-54
3.2/4.2-55	3.2/4.2-55*
3.7/4.7-9	3.7/4.7-9*
3.7/4.7-10	3.7/4.7-10
---	3.7/4.7-24a
---	3.7/4.7-24b

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

NOTES FOR TABLE 3.2.K

*(Deleted)
**(Deleted)
***During main condenser offgas treatment system operation

ACTION A

(Deleted)

ACTION B

(Deleted)

ACTION C

(Deleted)

ACTION D

(Deleted)

ACTION E

With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue provided that a temporary monitor is installed or grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION F

(Deleted)

Table 3.2.L
Anticipated Transient Without Scram (ATWS) -
Recirculation Pump Test (RPT) Surveillance Instrumentation

Minimum No. Channels Operable per Trip Sys (1)	Function	Trip Setting	Allowable Value	Action	Remarks
2	ATWS/RPT Logic Reactor Dome Pressure High (PIS-3-204A-D)	1118 psig	≤ 1146.5 psig	(2)	Two out of two of the high reactor dome pressure channels or the low reactor vessel level channels in either trip system trips both reactor recirculation pumps.
2	Reactor Vessel Level Low (LS-3-58 A1-D1)	483" above vessel zero	≥ 471.52" above vessel zero		

(1) One channel in only one trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided the other channels in that trip system are OPERABLE.

(2) Two trip systems exist, either of which will trip both recirculation pumps. Perform Surveillance/maintenance/calibration on one channel in only one trip system at a time. If a channel is found to be inoperable or if the surveillance/maintenance/calibration period for one channel exceeds 6 consecutive hours, the trip system will be declared inoperable or the channel will be placed in a tripped condition. If in RUN mode and one trip system is inoperable for 72 hours or both trip systems are inoperable, the reactor shall be in at least the HOT STANDBY CONDITION within 6 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, LS-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

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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 (LIS-3-208A-D)	(1)(27)	Once/18 months (28)	Once/day
Instrument Channel - RCIC Turbine Steam Line High Flow	(1)(27)	Once/18 months (28)	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)(27)	Once/18 months (28)	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

TABLE 4.2.F

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Water Level (LI-3-58A&B)	Once/18 months	Each Shift
2) Reactor Pressure (PI-3-74A&B)	Once/6 months	Each Shift
3) Drywell Pressure (PI-64-67B) and XR-64-50	Once/6 months	Each Shift
4) Drywell Temperature (TI-64-52AB) and XR-64-50	Once/6 months	Each Shift
5) Suppression Chamber Air Temperature (XR-64-52)	Once/6 months	Each Shift
8) Control Rod Position	N/A	Each Shift
9) Neutron Monitoring	(2)	Each Shift
10) Drywell Pressure (PS-64-67B)	Once/6 months	N/A
11) Drywell Pressure (PIS-64-58A)	Once/18 months	N/A
12) Drywell Temperature (TS-64-52A)	Once/6 months	N/A
13) Timer (IS-64-67A)	Once/6 months	N/A
14) CAD Tank Level	Once/6 months	Once/day
15) Containment Atmosphere Monitors	Once/6 months	Once/day

TABLE 4.2.F (Continued)
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
16) Drywell to Suppression Chamber Differential Pressure	Once/6 months	Each Shift
17) Relief Valve Tailpipe Thermocouple Temperature	N/A	Once/month (24)
18) Acoustic Monitor on Relief Valve Tailpipe	Once/cycle (25)	Once/month (26)
19) High Range Primary Containment Radiation Monitors and Recorders (RR-90-272, RR-90-273, RM-90-272C, and RM-90-273C)	Once/18 Months (30)	Once/month
20) Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159)	Once/18 Months	Once/shift
21) Drywell Pressure - Wide Range (PI-64-160A) (XR-64-159)	Once/18 Months	Once/shift
22) Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162) (TR-64-162)	Once/18 Months	Once/shift
23) Wide Range Gaseous Effluent Radiation Monitor and Recorder (RM-90-306 and RR-90-360)	Once/18 Months	Once/shift

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

j. Continuous Leak Rate Monitor

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. Drywell and Torus Surfaces

The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

LIMITING CONDITIONS FOR OPERATION

3.7.A Primary Containment

3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

   a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.

   b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

   a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c., below.

   b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.

SURVEILLANCE REQUIREMENTS

4.7.A Primary Containment

3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers

   a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.

   b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.

4. Drywell-Pressure Suppression Chamber Vacuum Breakers

   a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.

   b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

Minimum No. Operable Per Trip System	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	0.5 psid	(1)	Actuates the pressure suppression chamber-reactor building vacuum breakers.

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel- Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	Once/month[(1)]	Once/18 months[(2)]	None

Footnotes:

(1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.

(2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 196
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 March 30, 1994, and supplemented on November 21, 1994 and March 9, 1995, 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. **196** , 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, Director
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: **July 17, 1995**

ATTACHMENT TO LICENSE AMENDMENT NO. 196

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
vii	vii
viii	viii*
1.1/2.1-5	1.1/2.1-5
---	1.1/2.1-5a
1.1/2.1-10	1.1/2.1-10
1.1/2.1-11	1.1/2.1-11*
3.1/4.1-2	3.1/4.1-2
3.1/4.1-3	3.1/4.1-3
3.1/4.1-7	3.1/4.1-7
3.1/4.1-8	3.1/4.1-8
3.1/4.1-10	3.1/4.1-10
3.1/4.1-11	3.1/4.1-11*
3.1/4.1-13	3.1/4.1-13
3.1/4.1-14	3.1/4.1-14*
3.2/4.2-7	3.2/4.2-7
3.2/4.2-8	3.2/4.2-8
3.2/4.2-14	3.2/4.2-14
3.2/4.2-15	3.2/4.2-15
3.2/4.2-16	3.2/4.2-16
3.2/4.2-17	3.2/4.2-17*
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*
3.2/4.2-23	3.2/4.2-23
3.2/4.2-30	3.2/4.2-30
3.2/4.2-31	3.2/4.2-31*
3.2/4.2-38	3.2/4.2-38*
3.2/4.2-38a	3.2/4.2-38a
3.2/4.2-39	3.2/4.2-39
3.2/4.2-40	3.2/4.2-40*
3.2/4.2-43	3.2/4.2-43
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
3.2/4.2-53	3.2/4.2-53
3.2/4.2-54	3.2/4.2-54*
3.2/4.2-64	3.2/4.2-64
3.2/4.2-65	3.2/4.2-65
3.7/4.2-9	3.7/4.2-9*
3.7/4.2-10	3.7/4.2-10
---	3.7/4.2-23b
---	3.7/4.2-23c
3.7/4.7-32	3.7/4.7-32*
3.7/4.7-33	3.7/4.7-33

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

SAFETY LIMIT

1.1.B. Power Transient

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

C. Reactor Vessel Water Level

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

LIMITING SAFETY SYSTEM SETTING

2.1.B. Power Transient Trip Settings

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

C. Water Level Trip Settings

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

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1.1 BASES (Cont'd)

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

REFERENCE

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

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

2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Browns Ferry Nuclear Plant have been analyzed in support of planned operating conditions up to the maximum thermal power of 3293 MWt. The analyses were based upon plant operation in accordance with Reference 1.

The transient analyses performed for each reload are described in Reference 2. Models and model conservatisms are also described in this reference.

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Shut-down	Refuel (7)	Startup/ Hot Standby	Run	Action (1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)(24)(25)						
2	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B
2	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B
2	High Flux	≤ 15% rated power		X(21)	X(17)	(15)	1.A
2	Inoperative	(13)		X(21)	X(17)	X	1.A
2	Downscale	≥ 3 Indicated on Scale		(11)	(11)	X(12)	1.A or 1.B
2	High Reactor Pressure (PIS-3-22AA,BB,C,D)	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14) (PIS-64-56 A-D)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14) (LIS-3-203 A-D)	≥ 538" above vessel zero		X	X	X	1.A

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Shut-down	Modes in Which Function Must Be Operable Refuel (7)	Startup/ Hot Standby	Run	Action (1)
2	High Water Level in West Scram Discharge Tank (LS-85-45A-D)	≤ 50 Gallons	X(2)	X(2)	X	X	1.A
2	High Water Level in East Scram Discharge Tank (LS-85-45E-H)	≤ 50 Gallons	X(2)	X(2)	X	X	1.A
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure				X(6)	1.A or 1.C
2	Turbine Control Valve Fast Closure or Turbine Trip	≥550 psig				X(4)	1.A or 1.D
4	Turbine Stop Valve Closure	≤10% Valve Closure				X(4)	1.A or 1.D
2	Turbine First Stage Pressure Permissive (PIS-1-81A&B) (PIS-1-91A&B)	not ≥154 psig		X(18)	X(18)	X(18)	1.A or 1.D (19)

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)	C	Trip Output Relays (4)	Before Each Startup and Weekly When Required to be Operable
High Flux (Flow Biased)	B	Trip Output Relays (4)	Once/Week
High Flux (Fixed Trip)	B	Trip Output Relays (4)	Once/Week
Inoperative	B	Trip Output Relays (4)	Once/Week
Downscale	B	Trip Output Relays (4)	Once/Week
Flow Bias	B	(6)	(6)
High Reactor Pressure (PIS-3-22AA,BB,C,D)	B	Trip Channel and Alarm (7)	Once/Month
High Drywell Pressure (PIS-64-56 A-D)	B	Trip Channel and Alarm (7)	Once/Month
Reactor Low Water Level (LIS-3-203 A-D)	B	Trip Channel and Alarm (7)	Once/Month

TABLE 4.1.A (Continued)

	Group (2)	Functional Test	Minimum Frequency(3)
High Water Level in Scram Discharge Tank Float Switches (LS-85-45C-F)	A	Trip Channel and Alarm	Once/Month
Electronic Level Switches (LS-85-45A, B, G, H)	B	Trip Channel and Alarm (7)	Once/Month
Main Steam Line Isolation Valve Closure	A	Trip Channel and Alarm	Once/3 Months (8)
Turbine Control Valve Fast Closure or turbine trip	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive (PIS-1-81A and B, PIS-1-91A and B)	B	Trip Channel and Alarm (7)	Every three months
Turbine Stop Valve Closure	A	Trip Channel and Alarm	Once/Month (1)

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux Output Signal	B	Heat Balance	Once Every 7 Days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure (PIS-3-22AA,BB,C,D)	B	Standard Pressure Source	Once/6 Months(9)
High Drywell Pressure (PIS-64-56 A-D)	B	Standard Pressure Source	Once/18 Months(9)
Reactor Low Water Level (LIS-3-203 A-D)	B	Pressure Standard	Once/18 Months(9)
High Water Level in Scram Discharge Volume			
Float Switches (LS-85-45C-F)	A	Calibrated Water Column (5)	Note (5)
Electronic Lvl Switches (LS-85-45-A, B, G, H)	B	Calibrated Water Column	Once/Operating Cycle (9)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive (PIS-1-81A&B, PIS-1-91A&B)	B	Standard Pressure Source	Once/18 Months(9)
Turbine Control Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	Note (5)	Note (5)

NOTES FOR TABLE 4.1.B

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1. A description of three groups is included in the Bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operation during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

9. Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the trip level setting.

3.1 BASES

The Reactor Protection System automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.
2. Preserve the integrity of the reactor coolant system.
3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the LIMITING CONDITIONS FOR OPERATION necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between nonclass 1E power supply and the class 1E RPS bus. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The Reactor Protection System is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of OPERABLE instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. Three APRM instrument channels are provided for each protection trip system.

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

TABLE 3.2.A
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Reactor Low Water Level(6) (LIS-3-203A-D)	$\geq$ 538" above vessel zero	A or (B and E)	1. Below trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS
1	Instrument Channel - Reactor High Pressure (PS-68-93 and 94)	100 $\pm$ 15 psig	D	1. Above trip setting isolates the shutdown cooling suction valves of the RHR system.
2	Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	$\geq$ 398" above vessel zero	A	1. Below trip setting initiates Main Steam Line Isolation
2	Instrument Channel - High Drywell Pressure (6) (PIS-64-56A-D)	$\leq$ 2.5 psig	A or (B and E)	1. Above trip setting does the following: a. Initiates Reactor Building Isolation b. Initiates Primary Containment Isolation c. Initiates SGTS

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

Minimum No. Instrument Channels Operable Per Trip Sys(1)(11)	Function	Trip Level Setting	Action (1)	Remarks
2	Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	≥ 825 psig (4)	B	1. Below trip setting initiates Main Steam Line Isolation
2(3)	Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	≤ 140% of rated steam flow	B	1. Above trip setting initiates Main Steam Line Isolation
2(12)	Instrument Channel - Main Steam Line Tunnel High Temperature	≤ 200°F	B	1. Above trip setting initiates Main Steam Line Isolation.
1(15)	Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	≤ 100 mr/hr or downscale	G	1. 1 upscale channel or 2 downscale channels will a. Initiate SGTS b. Isolate reactor zone and refueling floor. c. Close atmosphere control system.
2	Instrument Channel Reactor Water Cleanup System Main Steam Valve Vault (TIS-069-834A-D)	≤ 201.0°F	C	Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor
2	Instrument Channel Reactor Water Cleanup System Pipe Trench (TIS-069-835A-D)	≤ 135.0°F	C	Above Trip Setting initiates Isolation of Reactor Water Cleanup Lines to and from the Reactor

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

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Below trip setting initiates HPCI.
2	Instrument Channel - Reactor Low Water Level (LIS-3-58A-D)	≥ 470" above vessel zero.	A	1. Multiplier relays initiate RCIC.
2	Instrument Channel - Reactor Low Water Level (LS-3-58A-D)	≥ 398" above vessel zero.	A	1. Below trip setting initiates CSS. Multiplier relays initiate LPCI. 2. Multiplier relay from CSS initiates accident signal (15).
2(16)	Instrument Channel - Reactor Low Water Level (LS-3-58A-D)	≥ 398" above vessel zero.	A	1. Below trip settings, in conjunction with drywell high pressure, low water level permissive, ADS timer timed out and CSS or RHR pump running, initiates ADS. 2. Below trip settings, in conjunction with low reactor water level permissive, ADS timer timed out, ADS high drywell pressure bypass timer timed out, CSS or RHR pump running, initiates ADS.
1(16)	Instrument Channel - Reactor Low Water Level Permissive (LIS-3-184, 185)	≥ 544" above vessel zero.	A	1. Below trip setting permissive for initiating signals on ADS.
1	Instrument Channel - Reactor Low Water Level (LIS-3-52 and LIS-3-62A)	≥ 312 5/16" above vessel zero. (2/3 core height)	A	1. Below trip setting prevents inadvertent operation of containment spray during accident condition.

TABLE 3.2.B (Continued)

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

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Reactor Low Pressure (PIS-3-74A&B) (PIS-68-95, 96)	450 psig $\pm$ 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel - Reactor Low Pressure (PS-3-74A&B) (PS-68-95, 96)	230 psig $\pm$ 15	A	1. Recirculation discharge valve actuation.
2	Core Spray Auto Sequencing Timers (5)	$6 \leq t \leq 8$ sec.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \leq t \leq 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A3, B1, C3, and D1 Timers	$13 \leq t \leq 15$ sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \leq t \leq 1$ sec. $6 \leq t \leq 8$ sec. $12 \leq t \leq 16$ sec. $18 \leq t \leq 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A3, B1, C3, and D1 Timers	$27 \leq t \leq 29$ sec.	A	1. With normal power 2. One per pump

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - RHR Discharge Pressure	100 ±10 psig	A	1. Below trip setting defers ADS actuation.
2	Instrument Channel CSS Pump Discharge Pressure	185 ±10 psig	A	1. Below trip setting defers ADS actuation.
1(3)	Core Spray Sparger to Reactor Pressure Vessel d/p	2 psid ±0.4	A	1. Alarm to detect core spray sparger pipe break.
1	RHR (LPCI) Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems and valves.

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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-73-56A & B)	$\geq$ Elev. 551'	A	1. Below trip setting will open HPCI suction valves to the suppression chamber.
2(2)	Instrument Channel - Suppression Chamber High Level	$\leq$ 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)	$\leq$ 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)	$\leq$ 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)	$\geq$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)	$\leq$ 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. 2. Group 7: A Group 7 isolation is automatically actuated by only the following condition: 1. The respective turbine steam supply valve not fully closed.
1	RCIC System (Isolation) Logic	N/A	B	1. Includes Group 5 valves. 2. Group 5: A Group 5 isolation is actuated by any of the following conditions: a. RCIC Steamline Space High Temperature b. RCIC Steamline High Flow c. RCIC Steamline Low Pressure d. RCIC Turbine Exhaust Diaphragm High Pressure
1 (16)	ADS Logic	N/A	A	

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

   Action:

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

   B. Declare the system or component inoperable.

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

   D. No action required; indicators are considered redundant.

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

2. In only one trip system.

3. Not considered in a trip system.

4. Deleted.

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

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

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

8. Note 1 does not apply to this item.

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

NOTES FOR TABLE 3.2.B (Continued)

10. Only one trip system for each cooler fan.

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

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

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

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

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

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

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

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

TABLE 3.2.F

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	LI-3-58A LI-3-58B	Reactor Water Level	Indicator - 155" to +60"	(1) (2) (3)
2	PI-3-74A PI-3-74B	Reactor Pressure	Indicator 0-1200 psig	(1) (2) (3)
2	XR-64-50 PI-64-67	Drywell Pressure	Recorder -15 to +65 psig Indicator -15 to +65 psig	(1) (2) (3)
2	TI-64-52 XR-64-50	Drywell Temperature	Recorder, Indicator 0-400°F	(1) (2) (3)
1	XR-64-52	Suppression Chamber Air Temperature	Recorder 0-400°F	(1) (2) (3)
1	N/A	Control Rod Position	6V Indicating Lights)	(1) (2) (3) (4)
1	N/A	Neutron Monitoring	SRM, IRM, LPRM 0 to 100% power)	
1	PS-64-67	Drywell Pressure	Alarm at 35 psig)	(1) (2) (3) (4)
1	XR-64-50 and PS-64-58 B and IS-64-67	Drywell Temperature and Pressure and Timer	Alarm if temp. > 281°F and pressure >2.5 psig after 30 minute delay)	
1	LI-84-2A	CAD Tank "A" Level	Indicator 0 to 100%	(1)
1	LI-84-13A	CAD Tank "B" Level	Indicator 0 to 100%	(1)

TABLE 3.2.F (cont'd)

Surveillance Instrumentation

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Notes
2	H_2M - 76 - 94 H_2M - 76 - 104	Drywell and Torus Hydrogen Concentration	0.1 - 20%	(1)
2	PdI-64-137 PdI-64-138	Drywell to Suppression Chamber Differential Pressure	Indicator 0 to 2 psid	(1) (2) (3)
1/Valve		Relief Valve Tailpipe Thermocouple Temperature or Acoustic Monitor on Relief Valve Tailpipe		(5)
2	LI-64-159A XR-64-159	Suppression Chamber Water Level-Wide Range	Indicator, Recorder 0-240"	(1) (2) (3)
2	PI-64-160A XR-64-159	Drywell Pressure Wide Range	Indicator, Recorder 0-300 psig	(1) (2) (3)
2	TI-64-161 TR-64-161 TI-64-162 TR-64-162	Suppression Pool Bulk Temperature	Indicator, Recorder 30° - 230° F	(1) (2) (3) (4) (6)
1	RR-90-272 RR-90-273 RM-90-272A RM-90-273A	High Range Primary Containment Radiation Monitors and Recorders	Monitor, Recorder $1 - 10^7$ R/Hr	(7) (8)
1	RM-90-306 RR-90-360	Wide Range Gaseous Effluent Radiation Monitor and Recorder	Monitor, Recorder (Noble Gas $10^{-7} - 10^{+5}$ μCi/cc)	(7)(8)(9)

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NOTES FOR TABLE 3.2.K

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*(Deleted)
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***During main condenser offgas treatment system operation

ACTION A

(Deleted)

ACTION B

(Deleted)

ACTION C

(Deleted)

ACTION D

(Deleted)

ACTION E

With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue provided that a temporary monitor is installed or grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION F

(Deleted)

Table 3.2.L
Anticipated Transient Without Scram (ATWS) -
Recirculation Pump Test (RPT) Surveillance Instrumentation

Minimum No. Channels operable per Trip Sys (1)	Function	Trip Setting	Allowable Value	Action	Remarks
2	ATWS/RPT Logic Reactor Dome Pressure High (PIS-3-204A-D)	1118 psig	≤ 1146.5 psig	(2)	Two out of two of the high reactor dome pressure channels or the low reactor vessel level channels in either trip system trips both reactor recirculation pumps.
2	Reactor Vessel Level Low (LS-3-58 A1-D1)	483" above vessel zero	≥ 471.52" above vessel zero		

(1) One channel in only one trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided the other channels in that trip system are OPERABLE.

(2) Two trip systems exist, either of which will trip both recirculation pumps. Perform Surveillance/maintenance/calibration on one channel in only one trip system at a time. If a channel is found to be inoperable or if the surveillance/maintenance/calibration period for one channel exceeds 6 consecutive hours, the trip system will be declared inoperable or the channel will be placed in a tripped condition. If in RUN mode and one trip system is inoperable for 72 hours or both trip systems are inoperable, the reactor shall be in at least the HOT STANDBY CONDITION within 6 hours.

TABLE 4.2.A
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Reactor Low Water Level (LIS-3-203A-D)	(1)(28)	once/18 months (29)	once/day
Instrument Channel - Reactor High Pressure (PS-68-93 & -94)	(31)	once/18 months	None
Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	(1)(28)	once/18 months (29)	once/day
Instrument Channel - High Drywell Pressure (PIS-64-56A-D)	(1)(28)	once/18 months (29)	N/A
Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	(28)(27)	once/18 months (29)	None
Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	(28)(27)	once/18 months (29)	once/day

TABLE 4.2.A
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Main Steam Line Tunnel High Temperature	once/3 months (27)	once/operating cycle	None
Instrument Channel - Reactor Building Ventilation High Radiation - Reactor Zone	(1) (30)	once/18 months	once/day (8)
Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone	(1) (30)	once/18 Months	once/day (8)
Instrument Channel - SGTS Train A Heaters	(4)	(9)	N/A
Instrument Channel - SGTS Train B Heaters	(4)	(9)	N/A
Instrument Channel - SGTS Train C Heaters	(4)	(9)	N/A
Reactor Building Isolation Timer (refueling floor)	(4)	once/operating cycle	N/A
Reactor Building Isolation Timer (reactor zone)	(4)	once/operating cycle	N/A

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 (LS-3-58A-D, LIS-3-58A-D)	(1)(28)	once/18 months(29)	once/day
Instrument Channel - Reactor Low Water Level (LIS-3-184 & 185)	(1)(28)	once/18 months(29)	once/day
Instrument Channel - Reactor Low Water Level (LIS-3-52 & 62A)	(1)(28)	once/18 months(29)	once/day
Instrument Channel - Drywell High Pressure (PIS-64-58E-H)	(1)(28)	once/18 months(29)	none
Instrument Channel - Drywell High Pressure (PIS-64-58A-D)	(1)(28)	once/18 months(29)	none
Instrument Channel - Drywell High Pressure (PIS-64-57A-D)	(1)(28)	once/18 months(29)	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)(28)	once/6 months(29)	none

TABLE 4.2.B (Cont'd)
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 A3, B1, C3, D1 Timers (Normal Power)	(4)	once/operating cycle	none
RHRSW A3, B1, C3, D1 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

TABLE 4.2.B (Cont'd)
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 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 (LIS-3-208A-D)	(1)(28)	once/18 months (29)	once/day
Instrument Channel - RCIC Turbine Steam Line High Flow	(1)(28)	once/18 months (29)	none
Instrument Channel - RCIC Steam Supply Low Pressure	once/31 days	once/18 months	once/day
Instrument Channel - RCIC Turbine Exhaust Diaphragm High Pressure	once/31 days	once/18 months	once/day
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
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 (Cont'd)
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)(28)	once/18 months (29)	none
Instrument Channel - HPCI Steam Supply Low Pressure	once/31 days	once/18 months	once/day
Instrument Channel - HPCI Turbine Exhaust Diaphragm High Pressure	once/31 days	once/18 months	once/day
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

TABLE 4.2.F
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
1) Reactor Water Level (LI-3-58A & B)	Once/18 months	Each Shift
2) Reactor Pressure (PI-3-74A & B)	Once/6 months	Each Shift
3) Drywell Pressure	Once/6 months	Each Shift
4) Drywell Temperature	Once/6 months	Each Shift
5) Suppression Chamber Air Temperature	Once/6 months	Each Shift
8) Control Rod Position	N/A	Each Shift
9) Neutron Monitoring	(2)	Each Shift
10) Drywell Pressure (PS-64-67)	Once/6 months	N/A
11) Drywell Pressure (PIS-64-58A)	Once/18 months	N/A
12) Drywell Temperature (TR-64-52)	Once/6 months	N/A
13) Timer (IS-64-67)	Once/6 months	N/A
14) CAD Tank Level	Once/6 months	Once/day
15) Containment Atmosphere Monitors	Once/6 months	Once/day

TABLE 4.2.F (Cont'd)
MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION

Instrument Channel	Calibration Frequency	Instrument Check
16) Drywell to Suppression Chamber Differential Pressure	Once/6 months	Each Shift
17) Relief Valve Tailpipe Thermocouple Temperature	N/A	Once/month (24)
18) Acoustic Monitor on Relief Valve Tailpipe	Once/cycle (25)	Once/month (26)
19) Suppression Chamber Water Level-Wide Range (LI-64-159A) (XR-64-159)	Once/cycle	Once/month
20) Drywell Pressure - Wide Range (PI-64-160A) (XR-64-159)	Once/cycle	Once/shift
21) Suppression Pool Bulk Temperature (TI-64-161) (TR-64-161) (TI-64-162) (TR-64-162)	Once/cycle	Once shift
22) High Range Primary Containment Radiation Monitors and Recorders (RR-90-272, RR-90-273, RM-90-272A, RM-90-273A)	Once/18 months (32)	Once/month
23) Wide Range Gaseous Effluent Radiation Monitor and Recorder (RM-90-306 and RR-90-360)	Once/18 months	Once/shift

3.2 BASES

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

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

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

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

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

The low water level instrumentation set to trip at $\geq$398 inches above vessel zero (Table 3.2.B) closes the Main Steam Isolation Valves, the Main Steam Line Drain Valves, and the Reactor Water Sample Valves (Group 1). 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 ≥398 inches above vessel zero (Table 3.2.B) initiates the LPCI, Core Spray Pumps, contributes to ADS initiation, and starts the diesel generators. These trip setting levels were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation so that postaccident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation is initiated in time to meet the above criteria.

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

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

The ADS high drywell 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 ADS high drywell pressure bypass timer analytical limit of 360 seconds, a Peak Cladding Temperature (PCT) of 1500°F will not be exceeded 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 high steam flow 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.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

j. Continuous Leak Rate Monitoring

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

k. The interior surfaces of the drywell and torus above the level one foot below the normal water line and outside surfaces of the torus below the water line shall be visually inspected each operating cycle for deterioration and any signs of structural damage with particular attention to piping connections and supports and for signs of distress or displacement.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A PRIMARY CONTAINMENT	**4.7.A PRIMARY CONTAINMENT**
3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers	3. Pressure Suppression Chamber-Reactor Building Vacuum Breakers
a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be per Table 3.7.A.	a. The pressure suppression chamber-reactor building vacuum breakers shall be exercised in accordance with Specification 1.0.MM, and the associated instrumentation including setpoint shall be functionally tested for proper operation per Table 4.7.A.
b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.	b. A visual examination and determination that the force required to open each vacuum breaker (check valve) does not exceed 0.5 psid will be made each refueling outage.
4. Drywell-Pressure Suppression Chamber Vacuum Breakers	4. Drywell-Pressure Suppression Chamber Vacuum Breakers
a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in 3.7.A.4.b and 3.7.A.4.c below.	a. Each drywell-suppression chamber vacuum breaker shall be tested in accordance with Specification 1.0.MM.
b. One drywell-suppression chamber vacuum breaker may be nonfully closed so long as it is determined to be not more than 3° open as indicated by the position lights.	b. When it is determined that two vacuum breakers are inoperable for opening at a time when OPERABILITY is required, all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

TABLE 3.7.A

INSTRUMENTATION FOR CONTAINMENT SYSTEMS

Minimum No. Operable Per Trip System	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel - Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	0.5 psid	(1)	Actuates the pressure suppression chamber-reactor building vacuum breakers.

Footnote:

(1) - Repair in 24 hours. If the function is not OPERABLE in 24 hours, declare the system or component inoperable.

TABLE 4.7.A

CONTAINMENT SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Function	Functional Test	Calibration	Instrument Check
Instrument Channel- Pressure suppression chamber-reactor building vacuum breakers (PdIS-64-20, 21)	Once/month[(1)]	Once/18 months[(2)]	None

Footnotes:

(1) - Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify OPERABILITY of the trip and alarm functions.

(2) - Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the level setting.

containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

The test frequencies are adequate to detect equipment deterioration prior to significant defects, but the tests are not frequent enough to load the filters, thus reducing their reserve capacity too quickly. That the testing frequency is adequate to detect deterioration was demonstrated by the tests which showed no loss of filter efficiency after two years of operation in the rugged shipboard environment on the US Savannah (ORNL 3726). Pressure drop across the combined HEPA filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow ASTM D3803-89. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1975. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

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

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

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

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

3.7.D/4.7.D Primary Containment Isolation Valves

The Browns Ferry Containment Leak Rate Program and Procedures contains the list of all the Primary Containment Isolation Valves for which the Technical Specification requirements apply. The procedures are subject to the change control provisions for plant procedures in the administrative controls section of the Technical Specifications. The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

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

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

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

Group 3 - Process lines are normally in use, and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from nonsafety related causes. To protect the reactor from a possible pipe break

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 237 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 196 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 INTRODUCTION

By letter dated March 30, 1994, the Tennessee Valley Authority (the licensee) requested amendments to the Technical Specifications (TS) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3. This request had the following major components:

- Replace the BFN Unit 3 reactor protection system (RPS) and emergency core cooling system (ECCS) mechanical and differential pressure switches with an analog transmitter/trip system (ATTS).
- Change the BFN Units 1 and 3 reactor vessel water level safety limit and Level 1 low reactor water level setpoint.
- Change the BFN Units 1, 2, and 3 suppression chamber-reactor building vacuum breaker calibration frequency.
- Change BFN Unit 1, 2, and 3 calibration frequencies and functional test descriptions, bases, and add instrument identifiers.

The licensee also proposes some editorial changes to the specifications.

The NRC staff requested additional information from the licensee on September 21, 1994 and January 19, 1995. The Licensee responded to these requests on November 18, 1994 and March 9, 1995, respectively. The staff's proposed finding of no significant hazards considerations was not affected by the additional information provided by the licensee. The lead staff reviewer for this evaluation also reviewed licensee records during a site visit from November 28 through December 2, 1994.

ENCLOSURE

2.0 EVALUATION

The staff evaluation for each of the components described above is given below.

2.1 BFN UNIT 3 ANALOG TRANSMITTER TRIP SYSTEM

TVA proposes to replace the Unit 3 RPS and ECCS mechanical and differential pressure switches with the ATTS. The ATTS modification includes the replacement of power supplies and associated electrical cabling, breakers and fuses. As a result, instrument identifiers, functional test description, group designator, minimum test frequency notes, minimum calibration frequencies and indicator range are being changed in the TS tables to reflect the new equipment. With the exception of some calibration frequency changes, Unit 3 changes make the TS consistent with changes previously approved for Unit 2.

The ATTS was proposed by General Electric Company (GE) in 1978 to replace the original mechanical and differential pressure switches which were not as reliable. The principal objective of the ATTS was to improve sensor intelligence and reliability while enhancing testing capability.

GE submitted the ATTS Topical Report, NEDO-21617-A, for review by the NRC and reference in license applications. The topical report was reviewed by the NRC staff and found acceptable, as documented in a letter dated June 27, 1978.

The ATTS provides the following system improvements:

- Continuous monitoring of parameters.
- Reduced functional tests and calibration frequency for the primary sensors.
- Decreased duration and complexity of required testing and calibration of inputs for safety related parameters.
- Reduced testing and maintenance related scrams.
- Reduced number of reportable events related to setpoint drift.
- Diversity associated with the Anticipated Transient Without Scram (ATWS) mitigation system required by 10 CFR 50.62.

The trip unit (with analog transmitters) and trip relays provide the input intelligence to the system logic for the RPS, ECCS, and the nuclear steam safety supply systems (NSSSS).

2.1.1 Information to Support ATTS Installation

In the staff's evaluation of NEDO-21617-A, the staff requested that each individual referencing application furnish the following information:

Specific Instrument Loops

Variable name, part number of device being deleted, system involved, divisional separation assignment, model number and vendor of the transmitters or RTDs.

Trip Unit Cabinet

Cabinet layout showing location areas of power supplies, trip relays, and trip units, divisional separation assignment, and layout of each card file in the trip unit cabinet showing the trip variable for each card file slot.

Environmental and Seismic Qualification

Demonstration of qualification of the ATTS system to the normal operating and post-accident environment temperature and humidity. Also, a comparison of the floor seismic spectra of the cabinet mounting location for the specific plant to the seismic test envelope in NEDO-21617-A for the ATTS cabinet. If the trip unit cabinets are not located in the preferred location as specified in NEDO-21617-A, provide justification for the alternate selected location.

Interconnection Diagram

An interconnection diagram which shows the interconnection between the existing logic cabinets.

During the staff review of the BFN Unit 2 ATTS, the licensee provided the above information in their letters dated May 8, 1985 and November 20, 1985. For the BFN Unit 3 ATTS, the above information and information regarding the use of Agastat relays in the ATTS was submitted by the licensee on March 30, 1994. The licensee's discussion of each of the criteria requested by NEDO-21617-A is summarized below.

2.1.1.2 Specific Instrument Loops

The licensee's March 30, 1994 letter indicated the variables and devices proposed for replacement by the ATTS. In a request for additional information (RAI) dated September 21, 1994, the staff noted that transmitter PDT-1-25D for Main Steam Line High Flow was assigned to RPS Division IB, which appeared to be in error. In its RAI response of November 18, 1994, the licensee agreed that the divisional assignment for this transmitter was in error, and it was changed to Division IIB.

The staff also noted in the September 21, 1994 RAI that the maximum qualified temperature for a number of Rosemount transmitters for BFN, Unit 2 was 350°F and identical transmitters for Unit 3 are listed as having a maximum qualified temperature of 415°F. The licensee was asked to clarify the difference in qualification temperature. The licensee clarified that the maximum qualified temperature of 350°F, specified for Rosemount 1153 transmitters was the intended qualification value specified in the Rosemount Test Report. However, during testing, the 350°F limit was exceeded with no damage to the transmitters. Therefore, a new qualification temperature of 415°F was specified for the BFN Unit 3 Rosemount transmitters, reflecting the actual as-qualified temperature. The staff finds the qualification temperature acceptable.

2.1.1.2 Trip Unit Cabinet

The licensee described a Trip Cabinet Assembly, discussing cabinet layout, location areas of power supplies, trip relays, and trip units. A Trip Cabinet Assembly showed divisional separation assignment, layout of each card file in the trip unit cabinet, and the trip variable for each card file slot. The staff finds the information supplied on the Trip Unit Cabinet acceptable.

2.1.1.3 Environmental and Seismic Qualification

The licensee provided an Environmental Interface Temperature and Humidity Table which identified the transmitters, maximum normal temperature and humidity, maximum post-accident temperature and humidity, and the maximum qualified temperature and humidity.

The following instruments were identified by the licensee as not being within the scope of the 10 CFR 50.49 environmental qualification program:

- Main steam low pressure - input to primary containment isolation system (PCIS), PT-1-72, 76, 82, and 86.

- Turbine first stage pressure permissive - input to RPS/RPT, PT-1-81A, 81B, 91A, and 91B.

- Reactor high pressure - input to RPS, PT 3-22AA, 22BB, 22C, AND 22D.

- Reactor high pressure - input to ATWS (ARI/RPT), PT-3-204A, 204B, 204C, and 204D.

In the staff RAI of September 21, 1994, the licensee was requested to justify why these instruments were outside the scope of 10 CFR 50.49. On November 18, 1994, the licensee responded that only PT 3-22A, 22B, 22C and 22D have a safety-related function, and that none of the above instruments provide a safety-related function in a post-accident harsh environment as specified in the requirements of 10 CFR 50.49. The staff finds this justification consistent with 10 CFR 50.49 and, therefore, acceptable.

The licensee supplied a seismic response spectra of the ATTS cabinet mounting location and a comparison to the seismic test envelope that the cabinet was tested to as documented in NEDO-21617-A. This comparison indicated that the mounting location seismic response spectra was within the cabinet seismic test envelop. The staff finds this acceptable.

The BFN Unit 3 ATTS cabinets are located in the auxiliary room and control room which is the preferred location recommended by NEDO-21617-A. The ATTS instrumentation for BFN Unit 3 will be installed the same way as for Unit 2 with the following exception. The transmitters for the Reactor High Water Level instrument channels (LT-3-208 A, B, C, and D and LIS-3-208 A, B, C, and D) that are identified in the TS are being replaced with qualified Rosemount 1153 transmitters instead of Gould transmitters. Gould transmitters were installed in Unit 2 because Rosemount transmitters were not available at the time of the Unit 2 modification. TVA committed in their March 5, 1993, response to NRC Bulletin 90-01, Supplement 1 "Loss of Fill-Oil in Transmitters Manufactured by Rosemount" to replace or refurbish the Rosemount Model 1153 Series B and D and Model 1154 transmitters in safety related or ATWS applications prior to restart of BFN Unit 3. The staff finds that the Rosemount instruments are capable of fulfilling the design requirements for this installation. Therefore, this exception and commitment are acceptable.

2.1.1.4 Interconnection Diagram

The BFN plant-specific interconnection diagram is represented by NEDO-21617-A, Figure 5-5. This Topical Report was reviewed by the NRC staff and found acceptable. The BFN installation is within the scope of the staff review of NEDO-21617-A, and is also acceptable.

2.1.1.5 Revised Indication Range

The reactor coolant system (RCS) pressure indicator range is changed from 0-to-1500 psig to 0-to-1200 psig due to use of a new indicator. This newly-installed equipment includes the full range of pressure for which operator actions would be required during accident conditions. RCS pressure is recorded over a range of 0-to-1500 psig. Although Regulatory Guide 1.97 "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident" recommends Category 1 RCS pressure instrumentation with a range of 0-to-1500 psig, the RCS indicator range of 0-to-1200 psig was accepted for the BFN reactors by the staff in the letter dated May 10, 1991. The revised TS accurately reflects the installation of new equipment, and is consistent with staff expectations, and is, therefore, acceptable.

2.1.2 ATTS Agastat Relays

The Agastat trip unit output relay is used to provide an output from the ATTS to the existing protection system. This relay has a specified qualified life for both the energized and non-energized states. On September 21, 1994, the licensee was requested to identify this qualified life and identify a maintenance program which would assure replacement of the relay prior to the end of its qualified life for both Unit 2 and Unit 3.

On November 18, 1994, the licensee responded to the staff's request for additional information as follows:

1. Agastat Model ETR and EGP relays have been qualified for 20 years of service in their expected service environment.

2. Functional testing either monthly or quarterly ensures a high degree of reliability.

3. Administrative controls ensure failures are identified and evaluated for adverse trends.

4. TVA will either replace the Agastat relays in the ATTS after 20 years of service or document the acceptability of a longer service life.

Further staff investigation into Agastat relay reliabilty identified industry experience which may not be consistent with TVA's position. Therefore, on January 19, 1995, the staff requested additional information to complete its evaluation of TVA's amendment request.

The licensee response to this request of March 9, 1995 is summarized as follows:

- The ATTS output relays are in a mild environment; therefore, determination of a qualified lifetime is not required.

- The current quality assurance requirements are sufficient to ensure adequate performance of this equipment.

- TVA has not identified any instances of Agastat relay failures in the ATTS during its review of BFN equipment failure data.

- Incipient age related failures of Agastat relays would be detected by the current trending program prior to the occurrence of concurrent failures that could defeat redundancy.

The staff concludes that current TS functional testing being performed monthly along with the current trending program will permit TVA to detect Agastat relay failures, properly evaluate these failures and take appropriate corrective action. Therefore, the staff's concerns in this regard are resolved.

2.2 REACTOR VESSEL WATER LEVEL

2.2.1 Safety Limit (SL)

For BFN Units 1 and 3, SL for reactor vessel level is being changed from 378 inches above vessel zero (IAVZ) to 372.5 IAVZ. The licensee states that the revised SL corresponds to the level which is used in design analyses. This level has been established by General Electric to provide a point which can be monitored and provides adequate margin to assure sufficient cooling. The revised limit makes the BFN Units 1 and 3 SL consistent with the BFN Unit 2

SL. The staff finds that these criteria are appropriate. Therefore, the revised safety limit is acceptable.

2.2.2 Limiting Safety System Setting (LSSS)

The original LSSS for reactor vessel low water level in TS Table 3.2.A and Table 3.2.B was equal to the SL of 378 IAVZ. NRC regulations (10 CFR 50.36) state:

> Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

This requirement cannot be achieved if the LSSS is equal to the SL. In this case, the LSSS must be set to actuate at a higher reactor water level than the SL to compensate for instrument and loop inaccuracies, and response time of instrumentation and components that actuate to mitigate an event.

For BFN Units 1 and 3, the reactor vessel low water level 1 LSSS is changed from 378 IAVZ to 398 IAVZ. Instruments with this LSSS initiate the following systems:

- Containment spray system (CSS) (TS Table 3.2.B).
- Low pressure coolant injection system (LPCI) (TS Table 3.2.B).
- Main steam isolation system (MSIS) (TS Table 3.2.A).
  main steam isolation valves (MSIV)
  main steam line drain valves
  reactor water sample valves
- Permissive inputs to the automatic depressurization system (ADS) (TS Table 3.2.B).

The reactor vessel low water level 1 LSSS trip setting was chosen to be high enough to prevent spurious actuation but low enough to initiate post-accident cooling while providing margin to the SL.

The methodology used by the licensee to determine the LSSS is in accordance with the Instrument Society of America Standard ISA-S67.04 - 1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants." This methodology is consistent with the guidance of Regulatory Guide 1.105. Therefore, the proposed LSSS is acceptable.

2.3 PRESSURE SUPPRESSION-REACTOR BUILDING VACUUM BREAKERS

For BFN Units 1, 2, and 3, the differential pressure instrumentation (which actuates the pressure suppression-reactor building vacuum breakers) calibration frequency is being revised to reflect current Units 2 and 3 calculations. The calibration frequency of the transmitters, as shown on Table 4.7.A, has been changed from every 3 months to 18 months. A Unit 1 specific calculation will be performed to confirm the calibration frequency prior to Unit 1 restart. The pressure differential setpoint which actuates the vacuum breakers has been changed from 1.1 psid to 0.5 psid as shown on Table 3.7.A. The calibration frequency scaling and setpoint calculations which reflect the above changes are in accordance with the guidance contained in Regulatory Guide 1.105, and are, therefore, acceptable.

2.4 INSTRUMENT IDENTIFIERS

The licensee proposes to revise the TS to add instrument identifiers for the Unit 2 equipment previously installed and Unit 3 equipment installed as part of the ATTS modification. These identifiers provide additional detail which enhances the usability of the TS, and are acceptable.

2.5 TEST DESCRIPTIONS AND CALIBRATION FREQUENCIES

The licensee proposes to revise the instrument calibration frequencies and functional test descriptions for the BFN Unit 2 Reactor High Water Level, reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) turbine steam line high flow, and drywell pressure instruments. The licensee states that it has performed scaling and setpoint calculations consistent with the guidance of Regulatory Guide 1.105.

The staff's acceptance of the reduced functional test frequencies and calibration frequencies of instruments and components associated with the ATTS is based upon recommendation by General Electric Company (GE) in their Topical Report NEDO-21617-A and the NRC acceptance of this report. These changes are consistent with GE's Technical Specification Improvement Analysis for Boiling Water Reactor Protection System, NEDC-30851P-A, that was reviewed and approved by the NRC generic safety evaluation report dated July 15, 1987 and the NRC Standard Technical Specifications for BWRs. The NRC has approved operating license amendments, regarding the Analog Transmitter Trip System, for BWRs as follows:

- Vermont Yankee Nuclear Power Station, November 3, 1980.
- Pilgrim Nuclear Power Station, August 6, 1986.
- Browns Ferry Unit 2, August 19, 1986.
- Brunswick Steam Electric Plant Units 1 and 2, March 16, 1989.

The staff finds that the licensee's proposed TS are consistent with regulatory guidance and overall industry practice, and are acceptable.

2.6 SUMMARY OF TS CHANGES

ATTS Implementing Technical Specifications

The following TS tables are to be revised to incorporate equipment installed as part of the BFN Unit 3 ATTS modifications, and to provide appropriate test requirements for those components:

Table 3.1.A, Reactor Protection System (Scram) Instrumentation Requirements,

Table 4.1.A, Reactor Protection System (Scram) Instrumentation Functional Tests Minimum Functional Test Frequencies for Safety Instrumentation and Control Circuits,

Table 4.1.B, Reactor Protection System (Scram) Instrumentation Calibration Minimum Calibration Frequencies for Reactor Protection Instrument Channels,

Table 3.2.A, Primary Containment and Reactor Building Isolation Instrumentation,

Table 3.2.B, Instrumentation that Initiates or Controls the Core and Containment Cooling Systems,

Table 3.2.F, Surveillance Instruction,

Table 3.2.L, Anticipated Transient Without Scram (ATWS) - Recirculation Pump Test (RPT) Surveillance Instrumentation,

Table 4.2.A, Surveillance Requirements for Primary Containment and Reactor Building Isolation Instrumentation,

Table 4.2.B, Surveillance Requirements for Instrumentation that Initiate or Control the CSCS,

Table 4.2.F, Minimum Test and Calibration Frequency for Surveillance Instrumentation.

The licensee has also proposed changes to calibration frequency and functional test requirements for BFN Unit 2. The staff has reviewed the proposed changes, and finds they are consistent with the evaluation provided in sections 2.1, 2.4, and 2.5 above. Therefore, these proposed changes are acceptable.

<u>BFN Units 1 and 3: Reactor Vessel Water Level Safety Limit and Limiting Safety System Setting</u>

The staff has reviewed the proposed changes to Tables 3.2.A and 3.2.B, and finds they are consistent with the evaluation provided in section 2.2 above. Therefore, these proposed changes are acceptable.

<u>BFN Units 1, 2, and 3: Suppression Chamber-Reactor Building Vacuum Breakers Calibration Frequency</u>

The staff has reviewed the proposed changes to TS 3.7.A.3.a, 3.7.A.3.b, 4.7.A.3.a, and new tables 3.7.A and 4.7.A, and finds they are consistent with the evaluation provided in section 2.3 above. Therefore, these proposed changes are acceptable.

BFN Units 1, 2, and 3: Editorial Changes

The licensee has proposed editorial changes to revise capitalization of terms used in the TS affected by the items discussed above. The staff has reviewed these editorial changes, and finds that the changes are consistent with routine practice, where terms defined by Section 1.0 of the TS are capitalized for ease of identification. Therefore, the proposed editorial changes are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. The NRC 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 49435). 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 or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based upon 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 (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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