December 7, 1994

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

Dear Mr. Kingsley:

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 (TAC NOS. M86007, M86008, AND M86009) (TS 331)

The Commission has issued the enclosed Amendment Nos. 213, 229, and 186 to Facility Operating Licenses 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 18, 1993, regarding miscellaneous administrative Technical Specification (TS) changes.

The amendments eliminate the requirements in the Technical Specifications (TS) regarding Unit 2 cycle 6, also, the amendment corrects administrative errors in previous technical specifications and resolves discrepancies between the specifications and the BFN Final Safety Analysis Report.

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-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 213 to License No. DPR-33
2. Amendment No. 229 to License No. DPR-52
3. Amendment No. 186 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures: See next page

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AMENDMENT NO. 213 FOR BROWNS FERRY UNIT 1 - DOCKET NO. 50-259
AMENDMENT NO. 229 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260
AMENDMENT NO. 186 FOR BROWNS FERRY UNIT 3 - DOCKET NO. 50-296
DATED: December 7, 1994

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 213
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 18, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 213, 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: December 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 213

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* and spillover** pages are provided to maintain document completeness.

REMOVE	INSERT
i	i*
ii	ii
iii	iii**
iv	iv**
v	v
vi	vi*
1.1/2.1-12	1.1/2.1-12*
1.1/2.1-13	1.1/2.1-13
3.1/4.1-14	3.1/4.1-14*
3.1/4.1-15	3.1/4.1-15
3.3/4.3-11	3.3/4.3-11*
3.3/4.3-12	3.3/4.3-12
3.4/4.4-3	3.3/4.3-3*
3.4/4.4-4	3.4/4.4-4
3.4/4.6-5	3.4/4.6-5*
3.6/4.6-6	3.6/4.6-6
3.6/4.6-9	3.6/4.6-9*
3.6/4.6-10	3.6/4.6-10
3.7/4.7-3	3.7/4.7-3
3.7/4.7-4	3.7/4.7-4*
3.7/4.7-23	3.7/4.7-23
3.7/4.7-24	3.7/4.7-24*
3.9/4.9-19	3.9/4.9-19*
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6.0-17	6.0-17*
6.0-18	6.0-18

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2.1 BASES (Cont'd)

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased High Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to core average neutron flux.

During transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

2.1 BASES (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/HOT STANDBY MODE)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

The IRM System consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be at 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions on that range.

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.

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

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

LIMITING CONDITIONS FOR OPERATION

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in the SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

SURVEILLANCE REQUIREMENTS

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the operable control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.E **Reactivity Control** If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.	4.3.E **Reactivity Control** Surveillance requirements are as specified in 4.3.C and .D above.
3.3.F **Scram Discharge Volume (SDV)** 1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.	4.3.F **Scram Discharge Volume (SDV)** 1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation. 1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.	2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.	3. No additional surveillance required.

3.4/4.4 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITIONS FOR OPERATION

3.4.B. Operation with Inoperable Components

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

3.4.C Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be OPERABLE, the following conditions shall be met:

1. At least 180 pounds Boron-10 must be stored in the Standby Liquid Control Solution tank and be available for injection.

2. The sodium pentaborate solution concentration must be equal to or less than 9.2% by weight.

SURVEILLANCE REQUIREMENTS

4.4.B. Surveillance with Inoperable Components

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

4.4.C Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Sodium Pentaborate Concentration check at least once per month. Also check concentration within 24 hours anytime water or boron is added to the solution.

3. Boron-10 Quantity:

   At least once per month, calculate and record the quantity of Boron-10 stored in the Standby Liquid Control Solution Tank.

4. Boron-10 Enrichment: At least once per 18 months and following each addition of boron to the Standby Liquid Control Solution Tank:

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	a. Calculate the enrichment within 24 hours. b. Verify by analysis within 30 days.
3.4.D **Standby Liquid Control System Requirements** The Standby Liquid Control System conditions must satisfy the following equation. $$\frac{(C)(Q)(E)}{(13\ wt.\%)(86\ gpm)(19.8\ atom\%)} \geq 1$$ where, C = sodium pentaborate solution concentration (weight percent) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2. Q = pump flow rate (gpm) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b. E = Boron-10 enrichment (atom percent Boron-10) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.	4.4.D **Standby Liquid Control System Requirements** Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.
1. If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem OPERABLE within 8 hours or the reactor shall be placed in a SHUTDOWN CONDITION with all operable control rods fully inserted within the following 12 hours.	1. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.6.B. Coolant Chemistry

1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply.

   a. Conductivity, µmho/cm at 25°C 2.0

   b. Chloride, ppm 0.1

2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply.

   a. Conductivity, µmho/cm at 25°C 1.0

   b. Chloride, ppm 0.2

SURVEILLANCE REQUIREMENTS

4.6.B. Coolant Chemistry

1. Reactor coolant shall be continuously monitored for conductivity except when there is no fuel in the reactor vessel.

   a. Whenever the continuous conductivity monitor is inoperable, a sample of reactor coolant shall be analyzed for conductivity every 4 hours except as listed below. If the reactor is in COLD SHUTDOWN CONDITION, a sample of reactor coolant shall be analyzed for conductivity every 8 hours.

   b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is >1.0 µmho/cm at 25°C.

2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

LIMITING CONDITIONS FOR OPERATION

3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits or the following maximum quality limits shall be cause for placing the reactor in the COLD SHUTDOWN CONDITION.

   a. Conductivity time above 1 µmho/cm at 25°C - 2 weeks/year.
      Maximum Limit 10 µmho/cm at 25°C

   b. Chloride concentration time above 0.2 ppm - 2 weeks/year.
      Maximum Limit - 0.5 ppm.

   c. The reactor shall be placed in the SHUTDOWN CONDITION if pH <5.6 or >8.6 for a 24-hour period.

SURVEILLANCE REQUIREMENTS

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including HOT STANDBY CONDITION) measurements of reactor water quality shall be performed according to the following schedule:

   a. Chloride ion content and pH shall be measured at least once every 96 hours.

   b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 µmho/cm at 25°C.

   c. A sample of reactor coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 µmho/cm at 25°C.

LIMITING CONDITIONS FOR OPERATION

3.6.C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

   b. Anytime the reactor is in RUN MODE, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN MODE except as defined in 3.6.C.1.c below.

   c. During the first 24 hours in the RUN MODE following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

SURVEILLANCE REQUIREMENTS

4.6.C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

LIMITING CONDITIONS FOR OPERATION

3.6.C Coolant Leakage

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

SURVEILLANCE REQUIREMENTS

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a, does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a.

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in accordance with Appendix J to 10 CFR 50 as modified by approved exemptions.

a. Three type A tests (overall integrated containment leakage rate) shall be conducted at 40 ± 10-month intervals during shutdown at P_a, 49.6 psig, during each 10-year plant inservice inspection.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

b. If any periodic type A test fails to meet 0.75 L_a, the test schedule for subsequent type A tests shall be reviewed and approved by the Commission.

If two consecutive type A tests fail to meet 0.75L_a, a type A test shall be performed at least every 18 months until two consecutive type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. 1. Test duration shall be at least 8 hours.

2. A 4-hour stabilization period will be required and the containment atmosphere will be considered stabilized when the change in weighted average air temperature averaged over an hour does not deviate by more than 0.5°R/hour from the average rate of change of temperature measured from the previous 4 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.G. Containment Atmosphere Dilution System (CAD)	4.7.G. Containment Atmosphere Dilution System (CAD)
2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE.	2. When FCV 84-8B is inoperable, each solenoid operated air/nitrogen valve of System B shall be cycled through at least one complete cycle of full travel and each manual valve in the flow path of System B shall be verified open at least once per week.
3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.	
4. If Specifications 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.	
5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.	
6. System A may be considered OPERABLE with FCV 84-8B inoperable provided that all active components in System B and all other active components in System A are OPERABLE.	
7. Specifications 3.7.G.6 and 4.7.G.2 are in effect until the first Cold Shutdown of unit 1 after July 20, 1984 or until January 17, 1985 whichever occurs first.	

LIMITING CONDITIONS FOR OPERATION

3.7.H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Whenever the reactor is not in Cold Shutdown, two independent gas analyzer systems shall be OPERABLE for monitoring the drywell and the torus.

2. With one hydrogen analyzer inoperable, restore at least two hydrogen analyzers to OPERABLE status within 30 days or be in at least Hot Shutdown within the next 24 hours.

3. With no hydrogen analyzer OPERABLE the reactor shall be in Hot Shutdown within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Each hydrogen analyzer system shall be demonstrated OPERABLE at least once per quarter by performing a CHANNEL CALIBRATION using standard gas samples containing a nominal eight-volume percent hydrogen balance nitrogen.

2. Each hydrogen analyzer system shall be demonstrated OPERABLE by performing a CHANNEL FUNCTIONAL TEST monthly.

3.9 BASES

The objective of this specification is to assure an adequate source of electrical power to operate facilities to cool the plant during shutdown and to operate the engineered safeguards following an accident. There are three sources of alternating current electrical energy available, namely, the 161-kV transmission system, the 500-kV transmission system, and the diesel generators.

The unit station-service transformer B for unit 1 or the unit station-service transformer B for unit 2 provide noninterruptible sources of offsite power from the 500-kV transmission system to the units 1 and 2 shutdown boards. Auxiliary power can also be supplied from the 161-kV transmission system through the common station-service transformers or through the cooling tower transformers by way of the bus tie board. The 4-kV bus tie board may remain out of service indefinitely provided one of the required offsite power sources is not supplied from the 161-kV system through the bus tie board.

The minimum fuel oil requirement of 35,280 gallons for each diesel generator fuel tank assembly is sufficient for seven days of full load operation of each diesel and is conservatively based on availability of a replenishment supply. Each diesel generator has its own independent 7-day fuel oil storage tank assembly.

The degraded voltage sensing relays provide a start signal to the diesel generators in the event that a deteriorated voltage condition exists on a 4-kV shutdown board. This starting signal is independent of the starting signal generated by the complete loss of voltage relays and will continue to function and start the diesel generators on complete loss of voltage should the loss of voltage relays become inoperable. The 15-day inoperable time limit specified when one of the three phase-to-phase degraded voltage relays is inoperable is justified based on the two-out-of-three permissive logic scheme provided with these relays.

A 4-kV shutdown board is allowed to be out of operation for a brief period to allow for maintenance and testing, provided all remaining 4-kV shutdown boards and associated diesel generators, CS, RHR, (LPCI and containment cooling) systems supplied by the remaining 4-kV shutdown boards, and all emergency 480-V power boards are OPERABLE.

The 480-V diesel auxiliary board may be out of service for short periods for tests and maintenance.

There is a safety related 250-V dc unit battery located in each unit. Each 250-V dc unit battery system consists of a battery, a battery charger, and a distribution panel. There is also a backup charger which can be assigned to any one of the three unit batteries. The 250-V dc unit battery systems provide power for unit control functions, unit DC motor loads and alternate control power to the 4160 and 480-V ac shutdown boards. The primary control power supplies to the 3A, 3C and 3D 4160-V ac shutdown boards and the Unit 3 480-V ac shutdown boards are

3.9 BASES (Cont'd)

also provided by unit batteries. There are five safety related 250-V dc shutdown battery systems assigned as primary control power supplies to 4160-V ac shutdown boards A, B, C, D, and 3EB. Each of these shutdown battery systems has as 250-V dc battery, a charger, and a distribution panel. A portable spare charger can be used to supply any one of the five shutdown battery systems.

Each 250-V dc shutdown board control power supply can receive power from its own battery, battery charger, or from a spare charger. The chargers are powered from normal plant auxiliary power or from the standby diesel-driven generator system. Zero resistance short circuits between the control power supply and the shutdown board are cleared by fuses located in the respective control power supply. Each power supply is located in the reactor building near the shutdown board it supplies. Each battery is located in its own independently ventilated battery room.

The 250-V dc system is so arranged, and the batteries sized so that the loss of any one unit battery will not prevent the safe shutdown and cooldown of all three units in the event of the loss of offsite power and a design basis accident in any one unit. Loss of control power to any engineered safeguard control circuits is annunciated in the main control room of the unit affected. The loss of one 250-V shutdown board battery affects normal control power for the 480-V and 4,160-V shutdown board which it supplies.

There are two 480-Volt ac RMOV boards that contain MG sets in their feeder lines. These 480-Volt ac RMOV boards have an automatic transfer from their normal to alternate power source (480-Volt ac shutdown boards). The MG sets act as electrical isolators to prevent a fault from propagating between electrical divisions due to an automatic transfer. The 480-Volt ac RMOV boards involved provide motive power to valves associated with the LPCI mode of the RHR system. Having an MG set out of service reduces the assurance that full RHR (LPCI) capacity will be available when required. Since sufficient equipment is available to maintain the minimum complement required for RHR (LPCI) operation, a 7-day servicing period is justified. Having two MG sets out of service can considerably reduce equipment availability; therefore, the affected unit shall be placed in Cold Shutdown within 24 hours.

The offsite power source requirements are based on the capacity of the respective lines. The Trinity line is limited to supplying two operating units because of the load limitations of CSST's A and B. The Athens line is limited to supplying one operating unit because of the load limitations of the Athens line. The limiting conditions are intended to prevent the 161-kV system from supplying more than two units in the event of a single failure in the offsite power system.

6.5.3 TECHNICAL REVIEW AND APPROVAL OF PROCEDURES

ACTIVITIES

6.5.3.1 Procedures required by Technical Specification 6.8.1.1 and other procedures which affect plant nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared, reviewed and approved. Each procedure or procedure change shall be reviewed by an individual other than the preparer. The reviewer may be from the same organization or from a different organization. Procedures other than Site Director Standard Practices will be approved by the responsible Section Supervisor, or applicable Manager.

6.5.3.2 Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Plant Manager. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to plant nuclear safety-related structures, systems and components shall be approved by the Plant Manager, prior to implementation.

6.5.3.3 Individuals responsible for reviews performed in accordance with 6.5.3.1 shall be members of the site supervisory staff previously designated by the Plant Manager. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by review personnel of the appropriate discipline.

6.5.3.4 The Plant Manager shall approve all administrative procedures requiring PORC review prior to implementation.

6.6 (Deleted)

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. 229
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 18, 1993, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 229 , 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: December 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 229

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* and spillover pages are provided to maintain document completeness.

REMOVE	INSERT
i	i*
ii	ii
iii	iii**
iv	iv**
v	v
v	vi*
1.1/2.1-12	1.1/2.1-12*
1.1/2.1-13	1.1/2.1-13
3.2/4.2-26	3.2/4.2-26
3.2/4.2-27	3.2/4.2-27
3.2/4.2-67	3.2/4.2-67*
3.2/4.2-68	3.2/4.2-68
3.3/4.3-11	3.3/4.3-11*
3.3/4.3-12	3.3/4.3-12
3.4/4.4-3	3.4/4.4-3*
3.4/4.4-4	3.4/4.4-4
3.5/4.5-18	3.5/4.5-18*
3.5/4.5-19	3.5/4.5-19
3.5/4.5-26	3.5/4.5-26*
3.5/4.5-27	3.5/4.5-27
3.6/4.6-5	3.6/4.6-5*
3.6/4.6-6	3.6/4.6-6
3.6/4.6-9	3.6/4.6-9*
3.6/4.6-10	3.6/4.6-10
3.7/4.7-3	3.7/4.7-3
3.7/4.7-4	3.7/4.7-4*
3.7/4.7-19	3.7/4.7-19
3.7/4.7-20	3.7/4.7-20*
3.7/4.7-23	3.7/4.7-23
3.7/4.7-24	3.7/4.7-24*
6.0-17	6.0-17*
6.0-18	6.0-18

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ADMINISTRATIVE CONTROLS

LIST OF TABLES

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

The bases for individual setpoints are discussed below:

A. <u>Neutron Flux Scram</u>

1. <u>APRM Flow-Biased High Flux Scram Trip Setting (RUN Mode)</u>

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to core average neutron flux.

During power increase transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased scram provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

2.1 BASES (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/HOT STANDBY MODE)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

The IRM System consists of eight chambers, four in each of the reactor protection system logic channels. The IRM is a five-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The five decades are covered by the IRM by means of a range switch and the five decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be at 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions on that range.

NOTES FOR TABLE 3.2.C

1. The minimum number of OPERABLE channels for each trip function is detailed for the STARTUP and RUN positions of the reactor mode selector switch. The SRM, IRM, and APRM (STARTUP mode), blocks need not be OPERABLE in "RUN" mode, and the APRM (flow biased) rod blocks need not be OPERABLE in "STARTUP" mode.

   With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt).

3. IRM downscale is bypassed when it is on its lowest range.

4. SRMs A and C downscale functions are bypassed when IRMs A, C, E, and G are above range 2. SRMs B and D downscale function is bypassed when IRMs B, D, F, and H are above range 2.

   SRM detector not in startup position is bypassed when the count rate is $\geq$100 CPS or the above condition is satisfied.

5. During repair or calibration of equipment, not more than one SRM or RBM channel nor more than two APRM or IRM channels may be bypassed. Bypassed channels are not counted as OPERABLE channels to meet the minimum OPERABLE channel requirements. Refer to section 3.10.B for SRM requirements during core alterations.

6. IRM channels A, E, C, G all in range 8 or above bypasses SRM channels A and C functions.

   IRM channels B, F, D, H all in range 8 or above bypasses SRM channels B and D functions.

7. The following operational restraints apply to the RBM only.

   a. Both RBM channels are bypassed when reactor power is $\leq$30 percent or when a peripheral (edge) control rod is selected.

   b. The RBM need not be OPERABLE in the "startup" position of the reactor mode selector switch.

   c. Two RBM channels are provided and only one of these may be bypassed with the console selector. If the inoperable channel cannot be restored within 24 hours, the inoperable channel shall be placed in the tripped condition within one hour.

NOTES FOR TABLE 3.2.C (Cont'd)

7. (Continued)

   d. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

8. This function is bypassed when the mode switch is placed in RUN.

9. This function is only active when the mode switch is in RUN. This function is automatically bypassed when the IRM instrumentation is OPERABLE and not high.

10. The inoperative trips are produced by the following functions:

    a. SRM and IRM

       (1) Local "operate-calibrate" switch not in operate.

       (2) Power supply voltage low.

       (3) Circuit boards not in circuit.

    b. APRM

       (1) Local "operate-calibrate" switch not in operate.

       (2) Less than 14 LPRM inputs.

       (3) Circuit boards not in circuit.

    c. RBM

       (1) Local "operate-calibrate" switch not in operate.

       (2) Circuit boards not in circuit.

       (3) RBM fails to null.

       (4) Less than required number of LPRM inputs for rod selected.

11. Detector traverse is adjusted to 114 $\pm$ 2 inches, placing the detector lower position 24 inches below the lower core plate.

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

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

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

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

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

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

High temperature at the Reactor Water Cleanup (RWCU) System in the main steam valve vault, RWCU pump room 2A, RWCU pump room 2B, RWCU heat exchanger room or in the space near the pipe trench containing RWCU piping could indicate a break in the cleanup system. When high temperature occurs, the cleanup system is isolated.

3.2 BASES (Cont'd)

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

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

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

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

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

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

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

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

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are

LIMITING CONDITIONS FOR OPERATION

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

SURVEILLANCE REQUIREMENTS

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

D. Reactivity Anomalies

During the STARTUP test program and STARTUP following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3.E. Reactivity Control	4.3.E. Reactivity Control
If Specifications 3.3.C and .D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.	Surveillance requirements are as specified in 4.3.C and .D above.
3.3.F. Scram Discharge Volume (SDV)	4.3.F. Scram Discharge Volume (SDV)
1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.	1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation. 1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.
2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.	2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.
3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.	3. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.4.B. Operation with Inoperable Components

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

3.4.C Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be OPERABLE, the following conditions shall be met:

1. At least 180 pounds Boron-10 must be stored in the Standby Liquid Control Solution tank and be available for injection.

2. The sodium pentaborate solution concentration must be equal to or less than 9.2% by weight.

SURVEILLANCE REQUIREMENTS

4.4.B. Surveillance with Inoperable Components

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

4.4.C Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Sodium Pentaborate Concentration check at least once per month. Also check concentration within 24 hours anytime water or boron is added to the solution.

3. Boron-10 Quantity:

   At least once per month, calculate and record the quantity of Boron-10 stored in the Standby Liquid Control Solution Tank.

4. Boron-10 Enrichment: At least once per 18 months and following each addition of boron to the Standby Liquid Control Solution Tank:

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	a. Calculate the enrichment within 24 hours. b. Verify by analysis within 30 days.
3.4.D **Standby Liquid Control System Requirements** The Standby Liquid Control System conditions must satisfy the following equation. $\frac{(C)(Q)(E)}{(13\ \text{wt.\%})(86\ \text{gpm})(19.8\ \text{atom\%})} \geq 1$ where, C = sodium pentaborate solution concentration (weight percent) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2. Q = pump flow rate (gpm) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b. E = Boron-10 enrichment (atom percent Boron-10) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.	4.4.D **Standby Liquid Control System Requirements** Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.
1. If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem OPERABLE within 8 hours or the reactor shall be placed in a SHUTDOWN CONDITION with all OPERABLE control rods fully inserted within the following 12 hours.	1. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5.I **Average Planar Linear Heat Generation Rate**	4.5.I **Average Planar Linear Heat Generation Rate (APLHGR)**
During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.	The APLHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.
J. **Linear Heat Generation Rate (LHGR)**	J. **Linear Heat Generation Rate (LHGR)**
During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and	The LHGR shall be checked daily during reactor fuel operation at ≥ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

J. Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

J. Linear Heat Generation Rate (LHGR)

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a LIMITING CONTROL ROD PATTERN.

2. The MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:

   a. τ as defined in the CORE OPERATING LIMITS REPORT prior to initial scram time measurements for the cycle, performed in accordance with Specification 4.3.C.1.

   b. τ as defined in the CORE OPERATING LIMITS REPORT following the conclusion of each scram-time surveillance test required by Specifications 4.3.C.1 and 4.3.C.2.

   The determination of the limit must be completed within 72 hours of each scram-time surveillance required by Specification 4.3.C.

3.5 Bases (Cont'd)

The suppression chamber can be drained when the reactor vessel pressure is atmospheric, irradiated fuel is in the reactor vessel, and work is not in progress which has the potential to drain the vessel. By requiring the fuel pool gate to be open with the vessel head removed, the combined water inventory in the fuel pool, the reactor cavity, and the separator/dryer pool, between the fuel pool low level alarm and the reactor vessel flange, is about 65,800 cubic feet (492,000 gallons). This will provide adequate low-pressure cooling in lieu of CSS and RHR (LPCI and containment cooling mode) as currently required in Specifications 3.5.A.4 and 3.5.B.9. The additional requirements for providing standby coolant supply available will ensure a redundant supply of coolant supply. Control rod drive maintenance may continue during this period provided no more than one drive is removed at a time unless blind flanges are installed during the period of time CRDs are not in place.

Should the capability for providing flow through the cross-connect lines be lost, a 10-day repair time is allowed before shutdown is required. This repair time is justified based on the very small probability for ever needing RHR pumps and heat exchangers to supply an adjacent unit.

REFERENCES

1. Residual Heat Removal System (BFNP FSAR subsection 4.8)

2. Core Standby Cooling Systems (BFNP FSAR Section 6)

3.5.C. RHR Service Water System and Emergency Equipment Cooling Water System (EECWS)

The EECW has two completely redundant and independent headers (north and south) in a loop arrangement inside and outside the Reactor Building. Four RHRSW pumps, two per header, (A3, B3, C3 and D3) are dedicated to automatically supplying the EECW system needs. Four additional pumps (A1, B1, C1 and D1) can serve as RHRSW system pumps or be manually valved into the EECW system headers and serve as backup for the RHRSW pumps dedicated to supplying the EECW system. Those components requiring EECW, except the control air compressors which are not safety related, are able to be fed from both headers thus assuring continuity of operation if either header becomes inoperable. The control air compressors only use the EECW north header as an emergency backup supply.

There are four RHR heat exchanger headers (A, B, C, & D) with one RHR heat exchanger from each unit on each header. There are two RHRSW pumps on each header; one normally assigned to each header (A2, B2, C2, or D2) and one on alternate assignment (A1, B1, C1, or D1). One RHR heat exchanger header can adequately deliver the flow supplied by both RHRSW pumps to any two of the three RHR heat exchangers on the header. One RHRSW pump can supply the full flow requirement of one RHR heat exchanger. Two RHR heat exchangers can more than adequately handle the cooling requirements of one unit in any abnormal or postaccident situation.

3.5 BASES (Cont'd)

The RHR Service Water System was designed as a shared system for three units. The specification, as written, is conservative when consideration is given to particular pumps being out of service and to possible valving arrangements. If unusual operating conditions arise such that more pumps are out of service than allowed by this specification, a special case request may be made to the NRC to allow continued operation if the actual system cooling requirements can be assured.

Should one of the two RHRSW pumps normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection become inoperable, an equal capability for long-term fluid makeup to the unit reactor and for cooling of the unit containment remains OPERABLE. Because of the availability of an equal makeup and cooling capability, a 30-day repair period is justified. Should the capability to provide standby coolant supply be lost, a 10-day repair time is justified based on the low probability for ever needing the standby coolant supply. Verification that the LPCI subsystem cross-tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem.

With only one unit fueled, four RHRSW pumps are required to be OPERABLE for indefinite operation to meet the requirements of Specification 3.5.B.1 (RHR system). If only three RHRSW pumps are OPERABLE, a 30-day LCO exists because of the requirement of Specification 3.5.B.5 (RHR system).

3.5.D Equipment Area Coolers

There is an equipment area cooler for each RHR pump and an equipment area cooler for each set (two pumps, either the A and C or B and D pumps) of core spray pumps. The equipment area coolers take suction near the cooling air discharge of the motor of the pump(s) served and discharge air near the cooling air suction of the motor of the pump(s) served. This ensures that cool air is supplied for cooling the pump motors.

The equipment area coolers also remove the pump, and equipment waste heat from the basement rooms housing the engineered safeguard equipment. The various conditions under which the operation of the equipment air coolers is required have been identified by evaluating the normal and abnormal operating transients and accidents over the full range of planned operations. The surveillance and testing of the equipment area coolers in each of their various modes is accomplished during the testing of the equipment served by these coolers. This testing is adequate to assure the OPERABILITY of the equipment area coolers.

REFERENCES

1. Residual Heat Removal System (BFN FSAR Section 4.8)

2. Core Standby Cooling System (BFN FSAR Section 6)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.B. Coolant Chemistry	**4.6.B. Coolant Chemistry**
1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply. a. Conductivity, µmho/cm at 25°C 2.0 b. Chloride, ppm 0.1	1. Reactor coolant shall be continuously monitored for conductivity except when there is no fuel in the reactor vessel. a. Whenever the continuous conductivity monitor is inoperable, a sample of reactor coolant shall be analyzed for conductivity every 4 hours except as listed below. If the reactor is in COLD SHUTDOWN CONDITION, a sample of reactor coolant shall be analyzed for conductivity every 8 hours. b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is >1.0 µmho/cm at 25°C.
2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply. a. Conductivity, µmho/cm at 25°C 1.0 b. Chloride, ppm 0.2	2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

LIMITING CONDITIONS FOR OPERATION

3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits or the following maximum quality limits shall be cause for placing the reactor in the COLD SHUTDOWN CONDITION.

   a. Conductivity time above 1 μmho/cm at 25°C - 2 weeks/year. Maximum Limit 10 μmho/cm at 25°C

   b. Chloride concentration time above 0.2 ppm - 2 weeks/year. Maximum Limit - 0.5 ppm.

   c. The reactor shall be placed in the SHUTDOWN CONDITION if pH <5.6 or >8.6 for a 24-hour period.

SURVEILLANCE REQUIREMENTS

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including HOT STANDBY CONDITION) measurements of reactor water quality shall be performed according to the following schedule:

   a. Chloride ion content and pH shall be measured at least once every 96 hours.

   b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 μmho/cm at 25°C.

   c. A sample of reactor coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 μmho/cm at 25°C.

LIMITING CONDITIONS FOR OPERATION

3.6.C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

   b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.

   c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

SURVEILLANCE REQUIREMENTS

4.6.C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

LIMITING CONDITIONS FOR OPERATION

3.6.C Coolant Leakage

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

SURVEILLANCE REQUIREMENTS

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

4.6.D Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a, does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a.

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.C shall be taken.

The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in accordance with Appendix J to 10 CFR 50 as modified by approved exemptions.

a. Three type A tests (overall integrated containment leakage rate) shall be conducted at 40 ± 10-month intervals during shutdown at P_a, 49.6 psig, during each 10-year plant inservice inspection.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

b. If any periodic type A test fails to meet 0.75 L_a, the test schedule for subsequent type A tests shall be reviewed and approved by the Commission.

If two consecutive type A tests fail to meet 0.75L_a, a type A test shall be performed at least every 18 months until two consecutive type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. 1. Test duration shall be at least 8 hours.

2. A 4-hour stabilization period will be required and the containment atmosphere will be considered stabilized when the change in weighted average air temperature averaged over an hour does not deviate by more than 0.5°R/hour from the average rate of change of temperature measured from the previous 4 hours.

LIMITING CONDITIONS FOR OPERATION

3.7.E. Control Room Emergency Ventilation

1. Except as specified in Specification 3.7.E.3 below, both control room emergency pressurization systems shall be OPERABLE at all times when any reactor vessel contains irradiated fuel.

2. a. The results of the inplace cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.

   b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ASTM D3803 (130°C, 95% R.H.).

SURVEILLANCE REQUIREMENTS

4.7.E Control Room Emergency Ventilation

1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%).

2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.

   b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

LIMITING CONDITIONS FOR OPERATION

3.7.E. Control Room Emergency Ventilation

c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.

3. From and after the date that one of the control room emergency pressurization systems is made or found to be inoperable for any reason, REACTOR POWER OPERATIONS or refueling operations are permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.

4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in COLD SHUTDOWN within 24 hours for REACTOR POWER OPERATIONS and refueling operations shall be terminated within 2 hours.

SURVEILLANCE REQUIREMENTS

4.7.E. Control Room Emergency Ventilation

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

d. Each circuit shall be operated at least 10 hours every month.

3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.

4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the necessary dampers operate as required.

LIMITING CONDITIONS FOR OPERATION

3.7.G. Containment Atmosphere Dilution System (CAD)

2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE.

3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.

4. If Specifications 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.

5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.

SURVEILLANCE REQUIREMENTS

4.7.G. Containment Atmosphere Dilution System (CAD)

LIMITING CONDITIONS FOR OPERATION

3.7.H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Whenever the reactor is not in Cold Shutdown, two independent gas analyzer systems shall be OPERABLE for monitoring the drywell and the torus.

2. With one hydrogen analyzer inoperable, restore at least two hydrogen analyzers to OPERABLE status within 30 days or be in at least Hot Shutdown within the next 24 hours.

3. With no hydrogen analyzer OPERABLE the reactor shall be in Hot Shutdown within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Each hydrogen analyzer system shall be demonstrated OPERABLE at least once per quarter by performing a CHANNEL CALIBRATION using standard gas samples containing a nominal eight-volume percent hydrogen balance nitrogen.

2. Each hydrogen analyzer system shall be demonstrated OPERABLE by performing a CHANNEL FUNCTIONAL TEST monthly.

6.5.3 TECHNICAL REVIEW AND APPROVAL OF PROCEDURES

ACTIVITIES

6.5.3.1 Procedures required by Technical Specification 6.8.1.1 and other procedures which affect plant nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared, reviewed and approved. Each procedure or procedure change shall be reviewed by an individual other than the preparer. The reviewer may be from the same organization or from a different organization. Procedures other than Site Director Standard Practices will be approved by the responsible Section Supervisor, or applicable Manager.

6.5.3.2 Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Plant Manager. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to plant nuclear safety-related structures, systems and components shall be approved by the Plant Manager, prior to implementation.

6.5.3.3 Individuals responsible for reviews performed in accordance with 6.5.3.1 shall be members of the site supervisory staff previously designated by the Plant Manager. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by review personnel of the appropriate discipline.

6.5.3.4 The Plant Manager shall approve all administrative procedures requiring PORC review prior to implementation.

6.6 (Deleted)

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. 186
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 18, 1993, 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. 186 , 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: December 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 186

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
i	i*
ii	ii
iii	iii**
iv	iv*
v	v
vi	vi*
1.1/2.1-12	1.1/2.1-12*
1.1/2.1-13	1.1/2.1-13
3.1/4.1-13	3.1/4.1-13*
3.1/4.1-14	3.1/4.1-14
3.3/4.3-11	3.3/4.3-11*
3.3/4.3-12	3.3/4.3-12
3.4/4.4-3	3.4/4.4-3*
3.4/4.4-4	3.4/4.4-4
3.5/4.5-29	3.5/4.5-29*
3.5/4.5-30	3.5/4.5-30
3.6/4.6-5	3.6/4.6-5*
3.6/4.6-6	3.6/4.6-6
3.6/4.6-9	3.6/4.6-9*
3.6/4.6-10	3.6/4.6-10
3.7/4.7-3	3.7/4.7-3
3.7/4.7-4	3.7/4.7-4*
3.7/4.7-19	3.7/4.7-19
3.7/4.7-20	3.7/4.7-20*
3.7/4.7-23	3.7/4.7-23
3.7/4.7-23a	3.7/4.7-23a*
6.0-17	6.0-17*
6.0-18	6.0-18

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2.1 BASES (Cont'd)

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

1. APRM Flow-Biased High Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to core average neutron flux.

During transients, the instantaneous fuel surface heat flux is less than the instantaneous neutron flux by an amount depending upon the duration of the transient and the fuel time constant. For this reason, the flow-biased scram APRM flux signal is passed through a filtering network with a time constant which is representative of the fuel time constant. As a result of this filtering, APRM flow-biased scram will occur only if the neutron flux signal is in excess of the setpoint and of sufficient time duration to overcome the fuel time constant and result in an average fuel surface heat flux which is equivalent to the neutron flux trip setpoint. This setpoint is variable up to 120 percent of rated power based on recirculation drive flow according to the equations given in Section 2.1.A.1 and the graph in Figure 2.1-2. For the purpose of licensing transient analysis, neutron flux scram is assumed to occur at 120 percent of rated power. Therefore, the flow biased provides additional margin to the thermal limits for slow transients such as loss of feedwater heating. No safety credit is taken for flow-biased scrams.

2.1 BASES (Cont'd)

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR >***.

2. APRM Flux Scram Trip Setting (REFUEL or STARTUP/HOT STANDBY MODE)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

The IRM System consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram setting of 120 divisions is active in each range of the IRM. For example, if the instrument was on range 1, the scram setting would be 120 divisions for that range; likewise if the instrument was on range 5, the scram setting would be 120 divisions on that range.

***See Section 3.5.K

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 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.

3.1 BASES (Cont'd)

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

LIMITING CONDITIONS FOR OPERATION

3.3.C. Scram Insertion Times

2. The average of the scram insertion times for the three fastest OPERABLE control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

3. The maximum scram insertion time for 90% insertion of any OPERABLE control rod shall not exceed 7.00 seconds.

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be placed in the SHUTDOWN CONDITION until the cause has been determined and corrective actions have been taken as appropriate.

SURVEILLANCE REQUIREMENTS

4.3.C. Scram Insertion Times

2. At 16-week intervals, 10% of the OPERABLE control rod drives shall be scram-timed above 800 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

D. Reactivity Anomalies

During the startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LIMITING CONDITIONS FOR OPERATION

3.3.E. Reactivity Control

If Specifications 3.3.C and 3.3.D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the SHUTDOWN CONDITION within 24 hours.

3.3.F. Scram Discharge Volume (SDV)

1. The scram discharge volume drain and vent valves shall be OPERABLE any time that the reactor protection system is required to be OPERABLE except as specified in 3.3.F.2.

2. In the event any SDV drain or vent valve becomes inoperable, REACTOR POWER OPERATION may continue provided the redundant drain or vent valve is OPERABLE.

3. If redundant drain or vent valves become inoperable, the reactor shall be in HOT STANDBY CONDITION within 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.E. Reactivity Control

Surveillance requirements are as specified in 4.3.C and 4.3.D above.

4.3.F. Scram Discharge Volume (SDV)

1.a. The scram discharge volume drain and vent valves shall be verified open PRIOR TO STARTUP and monthly thereafter. The valves may be closed intermittently for testing not to exceed 1 hour in any 24-hour period during operation.

1.b. The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with Specification 1.0.MM.

2. When it is determined that any SDV drain or vent valve is inoperable, the redundant drain or vent valve shall be demonstrated OPERABLE immediately and weekly thereafter.

3. No additional surveillance required.

LIMITING CONDITIONS FOR OPERATION

3.4.B. Operation with Inoperable Components

1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued operation permitted provided that the component is returned to an operable condition within seven days.

3.4.C Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be OPERABLE, the following conditions shall be met:

1. At least 180 pounds Boron-10 must be stored in the Standby Liquid Control Solution tank and be available for injection.

2. The sodium pentaborate solution concentration must be equal to or less than 9.2% by weight.

SURVEILLANCE REQUIREMENTS

4.4.B. Surveillance with Inoperable Components

1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

4.4.C Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Sodium Pentaborate Concentration check at least once per month. Also check concentration within 24 hours anytime water or boron is added to the solution.

3. Boron-10 Quantity:

   At least once per month, calculate and record the quantity of Boron-10 stored in the Standby Liquid Control Solution Tank.

4. Boron-10 Enrichment: At least once per 18 months and following each addition of boron to the Standby Liquid Control Solution Tank:

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	a. Calculate the enrichment within 24 hours. b. Verify by analysis within 30 days.
3.4.D Standby Liquid Control System Requirements The Standby Liquid Control System conditions must satisfy the following equation. $\frac{(C)(Q)(E)}{(13\ \text{wt.\%})(86\ \text{gpm})(19.8\ \text{atom\%})} \geq 1$ where, C = sodium pentaborate solution concentration (weight percent) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.2. Q = pump flow rate (gpm) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.A.2.b. E = Boron-10 enrichment (atom percent Boron-10) Determined by the most recent performance of the surveillance instruction required by Specification 4.4.C.4.	**4.4.D Standby Liquid Control System Requirements** Verify that the equation given in Specification 3.4.D is satisfied at least once per month and within 24 hours anytime water or boron is added to the solution.
1. If Specification 3.4.A through 3.4.D cannot be met, make at least one subsystem OPERABLE within 8 hours or the reactor shall be placed in a SHUTDOWN CONDITION with all OPERABLE control rods fully inserted within the following 12 hours.	1. No additional surveillance required.

3.5 Bases (Cont'd)

The suppression chamber can be drained when the reactor vessel pressure is atmospheric, irradiated fuel is in the reactor vessel, and work is not in progress which has the potential to drain the vessel. By requiring the fuel pool gate to be open with the vessel head removed, the combined water inventory in the fuel pool, the reactor cavity, and the separator/dryer pool, between the fuel pool low level alarm and the reactor vessel flange, is about 65,800 cubic feet (492,000 gallons). This will provide adequate low-pressure cooling in lieu of CSS and RHR (LPCI and containment cooling mode) as currently required in Specifications 3.5.A.4 and 3.5.B.9. The additional requirements for providing standby coolant supply available will ensure a redundant supply of coolant supply. Control rod drive maintenance may continue during this period provided no more than one drive is removed at a time unless blind flanges are installed during the period of time CRDs are not in place.

Should the capability for providing flow through the cross-connect lines be lost, a 10-day repair time is allowed before shutdown is required. This repair time is justified based on the very small probability for ever needing RHR pumps and heat exchangers to supply an adjacent unit.

REFERENCES

1. Residual Heat Removal System (BFNP FSAR subsection 4.8)
2. Core Standby Cooling Systems (BFNP FSAR Section 6)

3.5.C. RHR Service Water System and Emergency Equipment Cooling Water System (EECWS)

The EECW has two completely redundant and independent headers (north and south) in a loop arrangement inside and outside the Reactor Building. Four RHRSW pumps, two per header, (A3, B3, C3 and D3) are dedicated to automatically supplying the EECW system needs. Four additional pumps (A1, B1, C1 and D1) can serve as RHRSW system pumps or be manually valved into the EECW system headers and serve as backup for the RHRSW pumps dedicated to supplying the EECW system. Those components requiring EECW, except the control air compressors which are not safety related, are able to be fed from both headers thus assuring continuity of operation if either header becomes inoperable. The control air compressors only use the EECW north header as an emergency backup supply.

There are four RHR heat exchanger headers (A, B, C, & D) with one RHR heat exchanger from each unit on each header. There are two RHRSW pumps on each header; one normally assigned to each header (A2, B2, C2, or D2) and one on alternate assignment (A1, B1, C1, or D1). One RHR heat exchanger header can adequately deliver the flow supplied by both RHRSW pumps to any two of the three RHR heat exchangers on the header. One RHRSW pump can supply the full flow requirement of one RHR heat exchanger. Two RHR heat exchangers can more than adequately handle the cooling requirements of one unit in any abnormal or postaccident situation.

3.5 BASES (Cont'd)

The RHR Service Water System was designed as a shared system for three units. The specification, as written, is conservative when consideration is given to particular pumps being out of service and to possible valving arrangements. If unusual operating conditions arise such that more pumps are out of service than allowed by this specification, a special case request may be made to the NRC to allow continued operation if the actual system cooling requirements can be assured.

Should one of the two RHRSW pumps normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection become inoperable, an equal capability for long-term fluid makeup to the unit reactor and for cooling of the unit containment remains OPERABLE. Because of the availability of an equal makeup and cooling capability, a 30-day repair period is justified. Should the capability to provide standby coolant supply be lost, a 10-day repair time is justified based on the low probability for ever needing the standby coolant supply. Verification that the LPCI subsystem cross-tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem.

With only one unit fueled, four RHRSW pumps are required to be OPERABLE for indefinite operation to meet the requirements of Specification 3.5.B.1 (RHR system). If only three RHRSW pumps are OPERABLE, a 30-day LCO exists because of the requirement of Specification 3.5.B.5 (RHR system).

3.5.D Equipment Area Coolers

There is an equipment area cooler for each RHR pump and an equipment area cooler for each set (two pumps, either the A and C or B and D pumps) of core spray pumps. The equipment area coolers take suction near the cooling air discharge of the motor of the pump(s) served and discharge air near the cooling air suction of the motor of the pump(s) served. This ensures that cool air is supplied for cooling the pump motors.

The equipment area coolers also remove the pump, and equipment waste heat from the basement rooms housing the engineered safeguard equipment. The various conditions under which the operation of the equipment air coolers is required have been identified by evaluating the normal and abnormal operating transients and accidents over the full range of planned operations. The surveillance and testing of the equipment area coolers in each of their various modes is accomplished during the testing of the equipment served by these coolers. This testing is adequate to assure the OPERABILITY of the equipment area coolers.

REFERENCES

1. Residual Heat Removal System (BFN FSAR Section 4.8)

2. Core Standby Cooling System (BFN FSAR Section 6)

LIMITING CONDITIONS FOR OPERATION

3.6.B. Coolant Chemistry

1. PRIOR TO STARTUP and at steaming rates less than 100,000 lb/hr, the following limits shall apply.

   a. Conductivity, µmho/cm at 25°C 2.0

   b. Chloride, ppm 0.1

2. At steaming rates greater than 100,000 lb/hr, the following limits shall apply.

   a. Conductivity, µmho/cm at 25°C 1.0

   b. Chloride, ppm 0.2

SURVEILLANCE REQUIREMENTS

4.6.B. Coolant Chemistry

1. Reactor coolant shall be continuously monitored for conductivity except when there is no fuel in the reactor vessel.

   a. Whenever the continuous conductivity monitor is inoperable, a sample of reactor coolant shall be analyzed for conductivity every 4 hours except as listed below. If the reactor is in COLD SHUTDOWN CONDITION, a sample of reactor coolant shall be analyzed for conductivity every 8 hours.

   b. Once a week the continuous monitor shall be checked with an in-line flow cell. This in-line conductivity calibration shall be performed every 24 hours whenever the reactor coolant conductivity is >1.0 µmho/cm at 25°C.

2. During startup prior to pressurizing the reactor above atmospheric pressure, measurements of reactor water quality shall be performed to show conformance with 3.6.B.1 of limiting conditions.

LIMITING CONDITIONS FOR OPERATION

3.6.B. Coolant Chemistry

3. At steaming rates greater than 100,000 lb/hr, the reactor water quality may exceed Specification 3.6.B.2 only for the time limits specified below. Exceeding these time limits or the following maximum quality limits shall be cause for placing the reactor in the COLD SHUTDOWN CONDITION.

   a. Conductivity time above 1 μmho/cm at 25°C - 2 weeks/year. Maximum Limit 10 μmho/cm at 25°C

   b. Chloride concentration time above 0.2 ppm - 2 weeks/year. Maximum Limit - 0.5 ppm.

   c. The reactor shall be placed in the SHUTDOWN CONDITION if pH <5.6 or >8.6 for a 24-hour period.

SURVEILLANCE REQUIREMENTS

4.6.B. Coolant Chemistry

3. Whenever the reactor is operating (including HOT STANDBY CONDITION) measurements of reactor water quality shall be performed according to the following schedule:

   a. Chloride ion content and pH shall be measured at least once every 96 hours.

   b. Chloride ion content shall be measured at least every 8 hours whenever reactor conductivity is >1.0 μmho/cm at 25°C.

   c. A sample of reactor coolant shall be measured for pH at least once every 8 hours whenever the reactor coolant conductivity is >1.0 μmho/cm at 25°C.

LIMITING CONDITIONS FOR OPERATION

3.6.C Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

   b. Anytime the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24-hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.

   c. During the first 24 hours in the RUN mode following STARTUP, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met.

SURVEILLANCE REQUIREMENTS

4.6.C Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 4 hours.

LIMITING CONDITIONS FOR OPERATION

3.6.C Coolant Leakage

2. Anytime irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

   The air sampling system may be removed from service for a period of 4 hours for calibration, function testing, and maintenance without providing a temporary monitor.

3. If the condition in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

3.6.D. Relief Valves

1. When more than one relief valve is known to be failed, an orderly shutdown shall be initiated and the reactor depressurized to less than 105 psig within 24 hours. The relief valves are not required to be OPERABLE in the COLD SHUTDOWN CONDITION.

SURVEILLANCE REQUIREMENTS

4.6.C Coolant Leakage

2. With the air sampling system inoperable, grab samples shall be obtained and analyzed at least once every 24 hours.

4.6.D. Relief Valves

1. Approximately one-half of all relief valves shall be bench-checked or replaced with a bench-checked valve each operating cycle. All 13 valves will have been checked or replaced upon the completion of every second cycle.

2. In accordance with Specification 1.0.MM, each relief valve shall be manually opened until thermocouples and acoustic monitors downstream of the valve indicate steam is flowing from the valve.

LIMITING CONDITIONS FOR OPERATION

3.7.A. Primary Containment

2.a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. Primary containment integrity is confirmed if the maximum allowable integrated leakage rate, L_a, does not exceed the equivalent of 2 percent of the primary containment volume per 24 hours at the 49.6 psig design basis accident pressure, P_a.

c. If N_2 makeup to the primary containment averaged over 24 hours (corrected for pressure, temperature, and venting operations) exceeds 542 SCFH, it must be reduced to < 542 SCFH within 8 hours or the reactor shall be placed in Hot Shutdown within the next 16 hours.

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

2. Integrated Leak Rate Testing

Primary containment nitrogen consumption shall be monitored to determine the average daily nitrogen consumption for the last 24 hours. Excessive leakage is indicated by a N_2 consumption rate of > 2% of the primary containment free volume per 24 hours (corrected for drywell temperature, pressure, and venting operations) at 49.6 psig. Corrected to normal drywell operating pressure of 1.1 psig, this value is 542 SCFH. If this value is exceeded, the action specified in 3.7.A.2.c shall be taken.

The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in accordance with Appendix J to 10 CFR 50 as modified by approved exemptions.

a. Three type A tests (overall integrated containment leakage rate) shall be conducted at 40 ± 10-month intervals during shutdown at P_a, 49.6 psig, during each 10-year plant inservice inspection.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

b. If any periodic type A test fails to meet 0.75 L_a, the test schedule for subsequent type A tests shall be reviewed and approved by the Commission.

If two consecutive type A tests fail to meet 0.75 L_a, a type A test shall be performed at least every 18 months until two consecutive type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. 1. Test duration shall be at least 8 hours.

2. A 4-hour stabilization period will be required and the containment atmosphere will be considered stabilized when the change in weighted average air temperature averaged over an hour does not deviate by more than 0.5°R/hour from the average rate of change of temperature measured from the previous 4 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.E. Control Room Emergency Ventilation	4.7.E Control Room Emergency Ventilation
1. Except as specified in Specification 3.7.E.3 below, both control room emergency pressurization systems shall be OPERABLE at all times when any reactor vessel contains irradiated fuel.	1. At least once every 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (± 10%).
2. a. The results of the inplace cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal when tested in accordance with ANSI N510-1975.	2. a. The tests and sample analysis of Specification 3.7.E.2 shall be performed at least once per operating cycle or once every 18 months, whichever occurs first for standby service or after every 720 hours of system operation and following significant painting, fire, or chemical release in any ventilation zone communicating with the system.
b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide removal at a velocity when tested in accordance with ASTM D3803 (130°C, 95% R.H.).	b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.E. Control Room Emergency Ventilation	4.7.E. Control Room Emergency Ventilation
c. System flow rate shall be shown to be within ±10% design flow when tested in accordance with ANSI N510-1975.	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.
	d. Each circuit shall be operated at least 10 hours every month.
3. From and after the date that one of the control room emergency pressurization systems is made or found to be inoperable for any reason, REACTOR POWER OPERATIONS or refueling operations are permissible only during the succeeding 7 days unless such circuit is sooner made OPERABLE.	3. At least once every 18 months, automatic initiation of the control room emergency pressurization system shall be demonstrated.
4. If these conditions cannot be met, reactor shutdown shall be initiated and all reactors shall be in COLD SHUTDOWN within 24 hours for REACTOR POWER OPERATIONS and refueling operations shall be terminated within 2 hours.	4. During the simulated automatic actuation test of this system (see Table 4.2.G), it shall be verified that the necessary dampers operate as required.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.G. Containment Atmosphere Dilution System (CAD)	4.7.G. Containment Atmosphere Dilution System (CAD)
2. The Containment Atmosphere Dilution (CAD) System shall be OPERABLE whenever the reactor is in the RUN MODE.	
3. If one system is inoperable, the reactor may remain in operation for a period of 30 days provided all active components in the other system are OPERABLE.	
4. If Specifications 3.7.G.1 and 3.7.G.2, or 3.7.G.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN CONDITION within 24 hours.	
5. Primary containment pressure shall be limited to a maximum of 30 psig during repressurization following a loss of coolant accident.	

LIMITING CONDITIONS FOR OPERATION

3.7.H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Whenever the reactor is not in Cold Shutdown, two independent gas analyzer systems shall be OPERABLE for monitoring the drywell and the torus.

2. With one hydrogen analyzer inoperable, restore at least two hydrogen analyzers to OPERABLE status within 30 days or be in at least Hot Shutdown within the next 24 hours.

3. With no hydrogen analyzer OPERABLE the reactor shall be in Hot Shutdown within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.H. Containment Atmosphere Monitoring (CAM) System - H_2 Analyzer

1. Each hydrogen analyzer system shall be demonstrated OPERABLE at least once per quarter by performing a CHANNEL CALIBRATION using standard gas samples containing a nominal eight-volume percent hydrogen balance nitrogen.

2. Each hydrogen analyzer system shall be demonstrated OPERABLE by performing a CHANNEL FUNCTIONAL TEST monthly.

6.5.3 TECHNICAL REVIEW AND APPROVAL OF PROCEDURES

ACTIVITIES

6.5.3.1 Procedures required by Technical Specification 6.8.1.1 and other procedures which affect plant nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared, reviewed and approved. Each procedure or procedure change shall be reviewed by an individual other than the preparer. The reviewer may be from the same organization or from a different organization. Procedures other than Site Director Standard Practices will be approved by the responsible Section Supervisor, or applicable Manager.

6.5.3.2 Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Plant Manager. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to plant nuclear safety-related structures, systems and components shall be approved by the Plant Manager, prior to implementation.

6.5.3.3 Individuals responsible for reviews performed in accordance with 6.5.3.1 shall be members of the site supervisory staff previously designated by the Plant Manager. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by review personnel of the appropriate discipline.

6.5.3.4 The Plant Manager shall approve all administrative procedures requiring PORC review prior to implementation.

6.6 (Deleted)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 229 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 186 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 18, 1993, the Tennessee Valley Authority (the licensee) requested amendments of the technical specifications (TS) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3. The proposed revisions consist of changes deleting fuel cycle-specific requirements, corrections of administrative errors, correcting discrepancies between the technical specifications bases and the BFN Final Safety Analysis Report (FSAR), and clarification of certain requirements to ensure consistent application.

2.0 EVALUATION

The proposed changes consist of various, unrelated, administrative revisions to the BFN TS. Therefore, items are addressed individually.

2.1 BFN Units 1, 2, and 3 - Revision to TS 3.6.C.2

For BFN Units 1, 2, and 3 TS 3.6.C.2 reads:

> Both the sump and air sampling systems shall be OPERABLE during REACTOR POWER OPERATION. From and after the date that one of these systems is made or found to be inoperable for any reason, REACTOR POWER OPERATION is permissible only during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

The proposed specification reads:

> Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, both the sump and air sampling systems shall be OPERABLE. From and after the date that one of these systems is made or found to be inoperable for any reason, the reactor may remain in operation during the succeeding 24 hours for the sump system or 72 hours for the air sampling system.

ENCLOSURE

BFN TS 1.H defines REACTOR POWER OPERATION as "... any operation in the STARTUP/HOT STANDBY or RUN MODE with the reactor critical and above 1 percent rated power." The current specification applies at any temperature when the reactor is critical above 1 percent rated power. The proposed specification applies any time the reactor coolant temperature is above 212°F, regardless of whether the reactor is critical or not. The licensee proposes to revise the applicability of TS 3.6.C.2 to be consistent with TS 3.6.C.1. The proposed change is a clarification that requires the leak detection systems to be operable when the leakage rate limits are required to be met and is therefore acceptable.

2.2 BFN Unit 1 - Revision to TS Bases 3.9

The licensee proposes to revise the BFN Unit 1 TS Bases 3.9 to state that the loss of one 250 volt shutdown board battery affects normal control power for the associated 480 volt and 4160 volt shutdown board. This revision reflects the actual plant configuration and is therefore acceptable.

2.3 BFN Unit 2 - Deletion of Cycle-Specific Specifications

The licensee proposes changes to BFN Unit 2 TS Table 3.2.C, the Bases for section 3.2, and to TS 3.5.K and 4.5.K.2. The changes consist of items applicable only to operation during BFN Unit 2 Cycle 6, which was completed on January 29, 1993. These items are not relevant to future plant operations, and so their deletion is acceptable.

2.4 BFN Units 1, 2, and 3 - Deletion of TS 6.6

The licensee proposes to delete TS 6.6, "Reportable Event Action," and to revise the TS Table of Contents accordingly. TS 6.6 requires that reportable events (defined by TS definition 1.Z as any of the conditions specified by 10 CFR 50.73) shall be reported to the NRC and that each event will be reviewed by the Plant Operations Review Committee (PORC), with the results submitted to the Site Director and Nuclear Safety Review Board (NSRB). Requirements for PORC and NSRB review are redundant to other specifications. Also, the reference to the notification of the Site Director is redundant in TS 6.5.1.8 where all activities of the PORC shall be documented and provided to the Site Director. Therefore, deletion of the duplicate requirements under TS 6.6, and the corresponding Table of Contents revision is acceptable.

2.5 BFN Units 1, 2, and 3 - Revision of TS 4.7.A.2 Regarding Containment Leak Rates

The licensee proposes to delete a reference to the testing standard ANSI N45.4-1972 from TS 4.7.A.2. The proposed revision still requires the licensee to conform to 10 CFR Part 50 Appendix J which requires ANSI N45.4-1972 and ANSI/ANS 56.8-1987, and is therefore acceptable.

2.6 BFN Units 1, 2, and 3 - deletion of References to the Rod Sequence control System.

These changes were incorporated into Amendments 196, 212, and 169 for BFN Units 1, 2, and 3, which included other changes relevant to removal of Rod Sequence control System.

2.7 BFN Units 1 and 3 - Revision of TS Bases 3.1

The proposed change deletes a reference to a loss of condenser vacuum scram setting. This feature was deleted by Amendments 118, 113, and 89 for BFN Units 1, 2, and 3. Appropriate changes to the Bases have already been completed for BFN Unit 2. The proposed changes are consistent with the current plant design and are therefore acceptable.

2.8 BFN Units 2 and 3 - Revision of TS Bases 3.5

The proposed change revises a reference to the BFN Final Safety Analysis Report (FSAR) from sub-section 6.7 to section 6.0. This correction ensures the Bases reference the appropriate FSAR information, and is acceptable.

2.9 BFN Units 1, 2, and 3 - Correction of TS 3.6.B.3

The current specification includes the word "of" when the word "or" is correct. The current TS is not grammatical, and the proposed change corrects this problem, clarifying the specification requirements. Therefore, the proposed change is acceptable.

2.10 BFN Units 2 and 3 - Correction of TS 4.7.E.1

The current specification repeats the word "to," which is grammatically incorrect. The proposed change deletes the extra "to" which does not change the intent of the specification. Therefore, the proposed change is acceptable.

2.11 BFN Units 1, 2, and 3 - Revisions to Headings

The licensee proposes to change certain headings within the table of contents and for the corresponding specifications. The changes do not materially affect the current content or requirements of the specifications, and are therefore acceptable.

2.12 BFN Units 1, 2, and 3 - Capitalization

The licensee proposes to change certain words to all lowercase or all uppercase characters in various specifications. The proposed changes are:

- "INOPERABLE" is changed to lowercase on pages 1.0-4 and 3.3/4.3-12.

- "Operable" is changed to uppercase on pages 1.0-4 and 3.4/4.4-4. A similar change is made on page 3.3/4.3-12 for BFN Unit 1 only.

- "Operability" is changed to uppercase on page 1.0-4.

- "Refuel or StartUp/Hot Standby Mode" is changed to uppercase on page 1.1/2.1-13.

- "Shutdown condition" is changed to uppercase on page 3.4/4.4-4.

- "Cold shutdown condition" is changed to uppercase on page 3.5/4.5-19 for BFN Unit 2.

Words in all uppercase letters are typically terms which are defined by Specification 1.0. The licensee also proposes to capitalize "Specification" in one application on page 1.0-4. The proposed changes provide additional consistency and readability, and does not materially affect the current content or requirements of the specifications, and are therefore acceptable.

2.13 BFN Units 1, 2, and 3 - Revisions Regarding Core Operating Limits Report

These changes were incorporated into Amendments 197, 214, and 170 for BFN Units 1, 2, and 3, respectively, which include other changes relevant to incorporation of the Core Operating Limits report.

3.0 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.

4.0 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 (58 FR 17296). 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.

5.0 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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