Dockets Nos. 50-259/260/296

March 3, 1988

Mr. S. A. White
Manager of Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Mr. White:

SUBJECT: CLARIFICATION AND CORRECTION OF THE TECHNICAL SPECIFICATIONS (TS 228)
(TAC R00008, R00009, R00010)

Re: Browns Ferry Nuclear Plant, Units 1, 2, and 3

The Commission has issued the enclosed Amendments Nos. 147 , 143 , and 118 to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, respectively. These amendments are in response to your application dated April 3, 1987, as clarified by letter dated January 22, 1988.

The changes to the Technical Specifications clarify apparent conflicts and make various corrections.

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

Sincerely,

Original Signed by Gerald E. Gears for

Gary G. Zech, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Enclosures:
1. Amendment No. 147 to License No. DPR-33
2. Amendment No. 143 to License No. DPR-52
3. Amendment No. 118 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures:
See next page

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PDR ADOCK 05000259
P PDR

Mr. S. A. White
Tennessee Valley Authority

Browns Ferry Nuclear Plant
Units 1, 2, and 3

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
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 April 3, 1987, as clarified by letter dated January 22, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 147 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Gary G. Zech, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 147

FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

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

REMOVE	INSERT
i	i*
ii	ii
3.2/4.2-26	3.2/4.2-26
3.2/4.2-27	3.2/4.2-27*
3.2/4.2-40	3.2/4.2-40*
3.2/4.2-41	3.2/4.2-41
3.2/4.2-59	3.2/4.2-59*
3.2/4.2-60	3.2/4.2-60
3.5/4.5-20	3.5/4.5-20
3.5/4.5-21	3.5/4.5-21*
3.5/4.5-34	3.5/4.5-34
3.5/4.5-35	3.5/4.5-35*
3.6/4.6-15	3.6/4.6-15
3.6/4.6-16	3.6/4.6-16*

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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).

   A ratio of FRP/CMFLPD <1.0 is permitted at reduced power. See Specification 2.1 for APRM control rod block setpoint.

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 ≥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 ≤30 percent and when a peripheral 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 from the console. If the inoperable channel cannot be restored within 24 hours, the inoperable channel shall be placed in the tripped condition within one hour.

   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 ± 2 inches, placing the detector lower position 24 inches below the lower core plate.

12. This function may be bypassed in the SHUTDOWN or REFUEL mode. If this function is inoperable at a time when operability is required the channel shall be tripped or administrative controls shall be immediately imposed to prevent control rod withdrawal.

13. RBM upscale flow-biased setpoint clipped at 106 percent rated reactor power.

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

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Reactor Low Water Level (LIS-3-203A-D, SW 2-3)	(1)	(5)	once/day
Instrument Channel - Reactor High Pressure	(1)	once/3 months	None
Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	(1)	once/3 month	once/day
Instrument Channel - High Drywell Pressure (PS-64-56A-D)	(1)	(5)	N/A
Instrument Channel - High Radiation Main Steam Line Tunnel	once/3 months (29)	(5)	once/day
Instrument Channel - Low Pressure Main Steam Line (PT-1-72, -76, -82, -86)	once/3 months (27) (29)	once/operating cycle (28)	None
Instrument Channel - High Flow Main Steam Line (dPT-1-13A-D, -25A-D, -36A-D, -50A-D)	once/3 months (27) (29)	once/operating cycle (28)	once/day

BFN-Unit 1

TABLE 4.2.A (Cont'd)
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel – Main Steam Line Tunnel High Temperature	once/3 months (29)	once/operating cycle	None
Instrument Channel – Reactor Building Ventilation High Radiation – Reactor Zone	(1) (22)	once/3 months	once/day (8)
Instrument Channel – Reactor Building Ventilation High Radiation – Refueling Zone	(1) (22)	once/3 Months	once/day (8)
Instrument Channel – SGTS Train A Heater	(4)	(9)	N/A
Instrument Channel – SGTS Train B Heater	(4)	(9)	N/A
Instrument Channel – SGTS Train C Heater	(4)	(9)	N/A
Reactor Building Isolation Timer (refueling floor)	(4)	once/operating cycle	N/A
Reactor Building Isolation Timer (reactor zone)	(4)	once/operating cycle	N/A

BFN-Unit 1

NOTES FOR TABLES 4.2.A THROUGH 4.2.H except 4.2.D

1. Functional tests shall be performed once per month.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be operable or are tripped.

9. Calibration frequency shall be once/year.

10. Deleted

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

NOTES FOR TABLES 4.2.A THROUGH 4.2.H except 4.2.D (Continued)

14. (Deleted)

15. The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

20. Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. One channel of either the reactor zone or refueling zone Reactor Building Ventilation Radiation Monitoring System may be administratively bypassed for a period not to exceed 24 hours for functional testing and calibration.

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

LIMITING CONDITIONS FOR OPERATION

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

1. Whenever the core thermal power is ≥ 25% of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0, or the APRM scram and rod block setpoint equations listed in Sections 2.1.A and 2.1.B shall be multiplied by FRP/CMFLPD as follows:

$$S \leq (0.66W + 54\%) \frac{FRP}{CMFLPD}$$

$$S_{RB} \leq (0.66W + 42\%) \left(\frac{FRP}{CMFLPD}\right)$$

2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to ≤ 25% of rated thermal power within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is ≥ 25% of rated thermal power.

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DB274L

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.9
10,000	12.1
15,000	12.2
20,000	12.1
25,000	11.6
30,000	10.9
35,000	9.9
40,000	9.3

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: 8DB274H

Average Planar Exposure (Mwd/t)	MAPLHGR (kW/ft)
200	11.1
1,000	11.2
5,000	11.8
10,000	12.1
15,000	12.2
20,000	12.0
25,000	11.5
30,000	10.9
35,000	10.0
40,000	9.3

3.5 BASES (Cont'd)

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEIM-10735, August 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

5. Letter from R. H. Buchholz (GE) to P. S. Check (NRC), "Response to NRC Request For Information On ODYN Computer Model," September 5, 1980.

4.5 Core and Containme... Cooling Systems Surveillance ...equencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with monthly tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventive maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, cause the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period was caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Whenever a CSCS system or loop is made inoperable because of a required test or calibration, the other CSCS systems or loops that are required to be operable shall be considered operable if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

Maximum Average Planar LHGR, LHGR, and MCPR

The MAPLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

LIMITING CONDITIONS FOR OPERATION

3.6.H. Seismic Restraints, Supports, and Snubbers

During all modes of operation all seismic restraints, snubbers, and supports shall, be OPERABLE except as noted in 3.6.H.1. All safety-related snubbers are listed in Surveillance Instruction BF SI 4.6.H-1 and BF SI 4.6.H-2.

1. With one or more seismic restraint, support, or snubber INOPERABLE on a system that is required to be OPERABLE in the current plant condition, within 72 hours replace or restore the INOPERABLE seismic restraint(s), support(s), or snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system INOPERABLE and follow the appropriate Limiting Condition statement for that system.

SURVEILLANCE REQUIREMENTS

4.6.H. Seismic Restraints, Supports, and Snubbers

The surveillance requirements of paragraph 4.6.G are the only requirements that apply to any seismic restraint or support other than snubbers.

Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Surveillance Instruction BF SI 4.6.H-1 and BF SI 4.6.H-2.

1. Inspection Groups

   The snubbers may be categorized into two major groups based on whether the snubbers are accessible or inaccessible during reactor operation. These major groups may be further subdivided into groups based on design, environment, or other features which may be expected to affect the operability of the snubbers within the group. Each group may be inspected independently in accordance with 4.6.H.2 through 4.6.H.9.

2. Visual Inspection, Schedule, and Lot Size

   The first inservice visual inspection of snubbers not previously included in these technical specifications and whose visual inspection has

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.6.H. Seismic Restraints, Supports, and Snubbers

4.6.H.2. (Cont'd)

not been performed and documented previously, shall be performed within six months for accessible snubbers and before resuming power after the first refueling outage for inaccessible snubbers subsequent to being included in these specifications. The results of these inspections shall be used in the schedule table below to determine the subsequent visual inspection period. Snubbers previously included in these technical specifications shall continue on their previously earned inspection schedule without affect from adding snubbers not within their group.

No. Inoperable Snubbers per Inspection Period	*Subsequent Visual Inspection Period
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

*The inspection interval shall not be lengthened more than one step at a time.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.143
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 April 3, 1987, as clarified by letter dated January 22, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 143 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Gary G. Zech, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 143

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

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

REMOVE	INSERT
i	i*
ii	ii
1.1/2.1-1	1.1/2.1-1*
1.1/2.1-2	1.1/2.1-2
1.1/2.1-3	1.1/2.1-3
1.1/2.1-4	1.1/2.1-4
3.2/4.2-26	3.2/4.2-26
3.2/4.2-27	3.2/4.2-27*
3.2/4.2-40	3.2/4.2-40*
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3.6/4.6-16	3.6/4.6-16*

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SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specifications

A. Thermal Power Limits

1. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

   When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (RUN Mode) (Flow Biased)

   a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

2.1.A.1.a (Cont'd)

$S \leq (0.66W + 54\%)$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

b. For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

2.1.A.1.b. (Cont'd)

NOTE: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR $\leq$13.4 kW/ft and MCPR within limits of Specification 3.5.K. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within prescribed limits. Surveillance requirements for APRM scram setpoint are given in Specification 4.5.L.

c. The APRM Rod Block trip setting shall be:

$S_{RB} \leq (0.66W + 42\%)$

where:

S_{RB} = Rod Block setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

1.1.A Thermal Power Limits

2. Reactor Pressure $\leq$800 psia or Core Flow $\leq$10% of rated.

   When the reactor pressure is $\leq$800 psia or core flow is $\leq$10% of rated, the core thermal power shall not exceed 823 MWt ($\sim$25% of rated thermal power).

LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings (Cont'd)

d. Fixed High Neutron Flux Scram Trip Setting--When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

   S$\leq$120% power.

2. APRM and IRM Trip Settings (Startup and Hot Standby Modes).

   a. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

   b. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

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 and when a peripheral 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 from the console. If the INOPERABLE channel cannot be restored within 24 hours, the INOPERABLE channel shall be placed in the tripped condition within one hour.

   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 ± 2 inches, placing the detector lower position 24 inches below the lower core plate.

12. This function may be bypassed in the SHUTDOWN or REFUEL mode. If this function is INOPERABLE at a time when operability is required the channel shall be tripped or administrative controls shall be immediately imposed to prevent control rod withdrawal.

13. RBM upscale flow-biased setpoint clipped at 106 percent rated reactor power.

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

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Reactor Low Water Level (LIS-3-203A-D)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel - Reactor High Pressure	(1)	Once/3 months	None
Instrument Channel - Reactor Low Water Level (LIS-3-56A-D)	(1) (27)	Once/18 month (28)	Once/day
Instrument Channel - High Drywell Pressure (PIS-64-56A-D)	(1) (27)	Once/18 Months (28)	N/A
Instrument Channel - High Radiation Main Steam Line Tunnel	29	(5)	Once/day
Instrument Channel - Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	(29) (27)	Once/18 Months (28)	None
Instrument Channel - High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	(29) (27)	Once/18 Months (28)	Once/day

TABLE 4.2.A (Cont'd)
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel – Main Steam Line Tunnel High Temperature	(29)	Once/operating cycle	None
Instrument Channel – Reactor Building Ventilation High Radiation – Reactor Zone	(1) (22)	Once/3 months	Once/day (8)
Instrument Channel – Reactor Building Ventilation High Radiation – Refueling Zone	(1) (22)	Once/3 Months	Once/day (8)
Instrument Channel – SGTS Train A Heater	(4)	(9)	N/A
Instrument Channel – SGTS Train B Heater	(4)	(9)	N/A
Instrument Channel – SGTS Train C Heater	(4)	(9)	N/A
Reactor Building Isolation Timer (refueling floor)	(4)	Once/operating cycle	N/A
Reactor Building Isolation Timer (reactor zone)	(4)	Once/operating cycle	N/A

BFN–Unit 2

NOTES FOR TABLES 4.2.A ROUGH 4.2.H except 4.2. D

1. Functional tests shall be performed once per month.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.

9. Calibration frequency shall be once/year.

10. Deleted

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

14. (Deleted)

15. The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

20. Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. One channel of either the reactor zone or refueling zone Reactor Building Ventilation Radiation Monitoring System may be administratively bypassed for a period not to exceed 24 hours for functional testing and calibration.

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

LIMITING CONDITIONS FOR OPERATION

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

1. Whenever the core thermal power is $\geq$ 25% of rated, the ratio of FRP/CMFLPD shall be $\geq$ 1.0, or the APRM scram and rod block setpoint equations listed in Sections 2.1.A and 2.1.B shall be multiplied by FRP/CMFLPD as follows:

$$S \leq (0.66W + 54\%) \frac{FRP}{CMFLPD}$$

$$S_{RB} \leq (0.66W + 42\%) \left(\frac{FRP}{CMFLPD}\right)$$

2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq$ 25% of rated thermal power within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq$ 25% of rated thermal power.

Table 3.5.I-1

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: P8DRB284L QUAD+ and 8DRB284L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.2
1,000	11.3
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.2
30,000	10.8
35,000	10.0
40,000	9.4

Table 3.5.I-2

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Fuel Type: P8DRB265H

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	12.0
25,000	11.6
30,000	11.2
35,000	10.9
40,000	10.5
45,000	10.0

3.5.M. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2, NEDO - 24088-1 and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE - 24011-P-A and Addenda.

4.5 Core and Containm... Cooling Systems Surveillance ...equencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with monthly tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventive maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, cause the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period was caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Whenever a CSCS system or loop is made inoperable because of a required test or calibration, the other CSCS systems or loops that are required to be operable shall be considered operable if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Redundant operable components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

Maximum Average Planar LHGR, LHGR, and MCPR

The MAPLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

LIMITING CONDITIONS FOR OPERATION1

3.6.H. Seismic Restraints, Supports, and Snubbers

During all modes of operation all seismic restraints, snubbers, and supports shall be OPERABLE except as noted in 3.6.H.1. All safety-related snubbers are listed in Surveillance Instructions BF SI 4.6.H-1 and BF SI 4.6.H-2.

1. With one or more seismic restraint, support, or snubber INOPERABLE on a system that is required to be OPERABLE in the current plant condition, within 72 hours replace or restore the INOPERABLE seismic restraint(s), support(s), or snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system INOPERABLE and follow the appropriate Limiting Condition statement for that system.

SURVEILLANCE REQUIREMENTS

4.6.H. Seismic Restraints, Supports, and Snubbers

The surveillance requirements of paragraph 4.6.G are the only requirements that apply to any seismic restraint or support other than snubbers.

Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Surveillance Instructions BF SI 4.6.H-1 and BF SI 4.6.H-2.

1. Inspection Groups

The snubbers may be categorized into two major groups based on whether the snubbers are accessible or inaccessible during reactor operation. These major groups may be further subdivided into groups based on design, environment, or other features which may be expected to affect the operability of the snubbers within the group. Each group may be inspected independently in accordance with 4.6.H.2 through 4.6.H.9.

2. Visual Inspection, Schedule, and Lot Size

The first inservice visual inspection of snubbers not previously included in these technical specifications and whose visual inspection has

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4.6.H. Seismic Restraints, Supports, and Snubbers

4.6.H.2. (Cont'd)

not been performed and documented previously, shall be performed within six months for accessible snubbers and before resuming power after the first refueling outage for inaccessible snubbers subsequent to being included in these specifications. The results of these inspections shall be used in the schedule table below to determine the subsequent visual inspection period. Snubbers previously included in these technical specifications shall continue on their previously earned inspection schedule without affect from adding snubbers not within their group.

No. Inoperable Snubbers per Inspection Period	*Subsequent Visual Inspection Period
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

*The inspection interval shall not be lengthened more than one step at a time.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 118
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 April 3, 1987, as clarified by letter dated January 22, 1988 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 118, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Gerald E. Gears

Gary G. Zech, Assistant Director
for Projects
TVA Projects Division
Office of Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 118

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
1.1/2.1-1	1.1/2.1-1*
1.1/2.1-2	1.1/2.1-2
1.1/2.1-3	1.1/2.1-3
1.1/2.1-4	1.1/2.1-4
3.2/4.2-25	3.2/4.2-25
3.2/4.2-26	3.2/4.2-26*
3.2/4.2-39	3.2/4.2-39*
3.2/4.2-40	3.2/4.2-40
3.2/4.2-58	3.2/4.2-58*
3.2/4.2-59	3.2/4.2-59
3.5/4.5-20	3.5/4.5-20
3.5/4.5-35	3.5/4.5-35
3.5/4.5-36	3.5/4.5-36*
3.6/4.6-15	3.6/4.6-15
3.6/4.6-16	3.6/4.6-16*

TABLE OF CONTENTS

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specification

A. Thermal Power Limits

1. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

   When the reactor pressure is greater than 800 psia, the existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

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

   a. When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
Settings	2.1.A Neutron Flux Trip

2.1.A.1.a (Cont'd)

$S \leq (0.66W + 54\%)$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

b. For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

2.1.A.1.b (Cont'd)

NOTE: These settings assume operation within the basic thermal hydraulic design criteria. These criteria are LHGR $\leq$13.4 kW/ft and MCPR within limits of Specification 3.5.K. If it is determined that either of these design criteria is being violated during operation, action shall be initiated within 15 minutes to restore operation within the prescribed limits. Surveillance requirements for APRM scram setpoint are given in Specification 4.5.L.

c. The APRM Rod Block trip setting shall be:

$S_{RB} \leq (0.66W + 42\%)$

where:

S_{RB} = Rod Block setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

1.1.A Thermal Power Limits

2. Reactor Pressure ≤800 psia or Core Flow ≤10% of rated.

When the reactor pressure is ≤800 psia or core flow is ≤10% of rated, the core thermal power shall not exceed 823 MWt (~25% of rated thermal power).

LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

d. Fixed High Neutron Flux Scram Trip Setting--When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be:

$S \leq 120\%$ power.

2. APRM and IRM Trip Settings (Startup and Hot Standby Modes).

a. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

b. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

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).

   See Specification 2.1 for APRM control rod block setpoint.

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 counts per second 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 and when a peripheral 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 from the console. If the inoperable channel cannot be restored within 24 hours, the inoperable channel shall be placed in the tripped condition within one hour.

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

NOTES FOR TABLE 3.2.C (Cont'd)

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 ± 2 inches, placing the detector lower position 24 inches below the lower core plate.

12. This function may be bypassed in the SHUTDOWN or REFUEL mode. If this function is inoperable at a time when operability is required the channel shall be tripped or administrative controls shall be immediately imposed to prevent control rod withdrawal.

13. RBM upscale flow-biased setpoint clipped at 106-percent rated reactor power.

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

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel - Reactor Low Water Level (LIS-3-203A-D, SW 2-3)	(1)	(5)	once/day
Instrument Channel - Reactor High Pressure	(1)	once/3 months	None
Instrument Channel - Reactor Low Water Level (LIS-3-56A-D, SW #1)	(1)	once/3 month	once/day
Instrument Channel - High Drywell Pressure (PS-64-56A-D)	(1)	(5)	N/A
Instrument Channel - High Radiation Main Steam Line Tunnel	once/3 months (27)	(5)	once/day
Instrument Channel - Low Pressure Main Steam Line	once/3 months (27)	once/3 months	None
Instrument Channel - High Flow Main Steam Line	once/3 months (27)	once/3 months	once/day

TABLE 4.2.A (Cont'd)
SURVEILLANCE REQUIREMENTS FOR PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel – Main Steam Line Tunnel High Temperature	once/3 months (27)	once/operating cycle	None
Instrument Channel – Reactor Building Ventilation High Radiation – Reactor Zone	(1) (22)	once/3 months	once/day (8)
Instrument Channel – Reactor Building Ventilation High Radiation – Refueling Zone	(1) (22)	once/3 Months	once/day (8)
Instrument Channel – SGTS Train A Heater	(4)	(9)	N/A
Instrument Channel – SGTS Train B Heater	(4)	(9)	N/A
Instrument Channel – SGTS Train C Heater	(4)	(9)	N/A
Reactor Building Isolation Timer (refueling floor)	(4)	once/operating cycle	N/A
Reactor Building Isolation Timer (reactor zone)	(4)	once/operating cycle	N/A

NOTES FOR TABLES 4.2.A THROUGH 4.2.H except 4.2.D

1. Functional tests shall be performed once per month.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be operable or are tripped.

9. Calibration frequency shall be once/year.

10. (DELETED)

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

NOTES FOR TABLES 4.2.A THROUGH 4.2.H except 4.2.D (Continued)

14. (Deleted)

15. The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

20. Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. One channel of either the reactor zone or refueling zone Reactor Building Ventilation Radiation Monitoring System may be administratively bypassed for a period not to exceed 24 hours for functional testing and calibration.

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

LIMITING CONDITIONS FOR OPERATION

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

1. Whenever the core thermal power is $\geq$ 25% of rated, the ratio of FRP/CMFLPD shall be $\geq$ 1.0, or the APRM scram and rod block setpoint equations listed in Sections 2.1.A and 2.1.B shall be multiplied by FRP/CMFLPD as follows:

$$S \leq (0.66W + 54\%) \frac{FRP}{CMFLPD}$$

$$S_{RB} \leq (0.66W + 42\%) \left(\frac{FRP}{CMFLPD}\right)$$

2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.

3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to $\leq$ 25% of rated thermal power within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

FRP/CMFLPD shall be determined daily when the reactor is $\geq$ 25% of rated thermal power.

3.5.M References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A and Addenda.

2. "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01-A.

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested each month to assure their operability. A simulated automatic actuation test once each cycle combined with monthly tests of the pumps and injection valves is deemed to be adequate testing of these systems.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventive maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, cause the outage, then the demonstration of operability should be thorough enough to assure that a generic problem does not exist. For example, if an out-of-service period was caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

Whenever a CSCS system or loop is made INOPERABLE because of a required test or calibration, the other CSCS systems or loops that are required to be operable shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are INOPERABLE. If the function, system, or loop under test or calibration is found INOPERABLE or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Redundant OPERABLE components are subjected to increased testing during equipment out-of-service times. This adds further conservatism and increases assurance that adequate cooling is available should the need arise.

Maximum Average Planar LHGR, LHGR, and MCPR

The MAPLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

LIMITING CONDITIONS FOR OPERATION

3.6.H. Seismic Restraints, Supports, and Snubbers

During all modes of operation all seismic restraints, snubbers, and supports shall, be OPERABLE except as noted in 3.6.H.1. All safety-related snubbers are listed in Surveillance Instruction BF SI 4.6.H-1 and BF SI 4.6.H-2.

1. With one or more seismic restraint, support, or snubber INOPERABLE on a system that is required to be OPERABLE in the current plant condition, within 72 hours replace or restore the INOPERABLE seismic restraint(s), support(s), or snubber(s) to OPERABLE status and perform an engineering evaluation on the attached component or declare the attached system INOPERABLE and follow the appropriate Limiting Condition statement for that system.

SURVEILLANCE REQUIREMENTS

4.6.H. Seismic Restraints, Supports, and Snubbers

The surveillance requirements of paragraph 4.6.G are the only requirements that apply to any seismic restraint or support other than snubbers.

Each safety-related snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 3.6.H/4.6.H. These snubbers are listed in Surveillance Instruction BF SI 4.6.H-1 and BF SI 4.6.H-2.

1. Inspection Groups

   The snubbers may be categorized into two major groups based on whether the snubbers are accessible or inaccessible during reactor operation. These major groups may be further subdivided into groups based on design, environment, or other features which may be expected to affect the operability of the snubbers within the group. Each group may be inspected independently in accordance with 4.6.H.2 through 4.6.H.9.

2. Visual Inspection, Schedule, and Lot Size

   The first inservice visual inspection of snubbers not previously included in these technical specifications and whose visual inspection has

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4.6.H. Seismic Restraints, Supports, and Snubbers

4.6.H.2. (Cont'd)

not been performed and documented previously, shall be performed within six months for accessible snubbers and before resuming power after the first refueling outage for inaccessible snubbers subsequent to being included in these specifications. The results of these inspections shall be used in the schedule table below to determine the subsequent visual inspection period. Snubbers previously included in these technical specifications shall continue on their previously earned inspection schedule without affect from adding snubbers not within their group.

No. INOPERABLE Snubbers per Inspection Period	*Subsequent Visual Inspection Period
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

*The inspection interval shall not be lengthened more than one step at a time.

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

SAFETY EVALUATION BY THE OFFICE OF SPECIAL PROJECTS

SUPPORTING AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

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

1.0 INTRODUCTION

By letter dated April 3, 1987, as clarified January 22, 1988, (TS-228) the Tennessee Valley Authority (TVA or the licensee) requested amendments to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2 and 3 (BFN). The amendments would change the Technical Specifications (TS) as follows:

1.1 Deletion of Section 3.5.M, Reporting Requirements

The TS would be revised to delete Section 3.5.M, Reporting Requirements, the bases for it, and its reference in the index.

This section is redundant to the requirements in 10 CFR 50.73 and requirements in the Administrative Controls section of the TS.

1.2 Revision of Note 7.d for Table 3.2.C

Table 3.2.C, Instrumentation That Initiates Rod Blocks, requires both channels of the Rod Block Monitor (RBM) to be operable except for its reference to note 7 which has four parts, (7.a, b, c, d). The current note 7.d immediately prevents control rod movement if the conditions for the table are not met. However, note 7.c allows one channel to be bypassed and inoperable for 24 hours without having to prevent control rod movement. Note 7.d would be revised since it currently causes a conflict with note 7.c. The new wording would be taken from Standard Technical Specifications (STS). It would not produce any conflict and would furthermore address the possibility of both RBM channels being inoperable, a condition which is not specifically addressed in the TS at present.

1.3 Correcting Limiting Condition for Operating (LCO) 3.6.H.1

Recent amendments (Nos. 128, 123, and 99 for units 1, 2, and 3, respectively) revised Surveillance Requirement to reference the correct Surveillance Instruction (SI) containing snubber lists. Specifically, a reference to SI 4.6.H was

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changed to SI 4.6 H-1 and 4.6 H-2. However, the corresponding LCO which contained the same incorrect reference was not changed. These amendments would make that same correction to LCO 3.6.H.

1.4 Revision to Section 2.1.A.1.c for APRM Setpoint Reference

BFN units 2 and 3 TS 2.1.A.1.b for APRM setpoints contained an erroneous reference to TS 4.1.B. TS 4.1.B specifies testing for the Reactor Protection System power monitoring system. This reference would be changed to TS 4.5.L which specifies surveillance requirements for APRM setpoints. This same error was corrected in Unit 1 TS by Amendment No. 128. By letter dated January 22, 1988, the licensee provided additional information for clarification and corrected the references from Section 2.1.A.1.b to Section 2.1.A.1.c of the TS. This supplement did not change the application from that initially noticed on June 17, 1987 (52 FR 23107) and did not alter the staff's proposed no significant hazards determination.

1.5 Delete Note (14) Table 4.2.A

There is no relationship between the surveillance testing required by table 4.2.A for the reactor zone and refueling zone radiation monitor instrumentation channels and either of the surveillance requirements referenced in footnote 14 for table 4.2.A. This note is only referenced by these two instrument channels. Therefore, this footnote which ties performance of these surveillance requirements together should be deleted.

1.6 Change to Note 22 Table 4.2.A

Footnote 22 of table 4.2.A is being moved to table 3.2.A as footnote 9. The wording has not changed and would still be referenced by the same instrumentation. This note would be more appropriate in table 3.2.A since it provides a relief from the LCO. Furthermore, a phrase would be added to footnote 11 which acknowledges the applicability of footnote 9 to the specific radiation monitoring instrumentation to which it is intended to apply. This change would remove an apparent conflict between the provisions of footnote 11 to table 3.2.A and the old footnote 22 to table 4.2.A by acknowledging the provisions of footnote 22 to be an exception to footnote 11.

By letter dated January 22, 1988 the licensee decided to correct the the concern via a surveillance instruction rather than changing the TS. This supplement did not change the application from that initially noticed on June 17, 1987 (52 FR 23107) and did not alter the staff's proposed no significant hazards determination.

2.0 EVALUATION

2.1 Deletion of Section 3.5.M, Reporting Requirements

The proposed amendments would delete Section 3.5.M, Reporting Requirements which requires TVA to report any time the limiting values of Average Planar Linear Heat Generation Rate, Linear Heat Generation Rate, Minimum Critical Power Ratio and APRM Setpoints are exceeded, the bases for it, and its reference in the index. The reporting requirements are being deleted because they are redundant to the reporting requirements contained in 10 CFR 50.73 as referenced in Section 6.0 of the Administrative Controls Section of the TS. The TS would continue to require reporting any time any of the above TS limits are exceeded under the requirements of 10 CFR 50.73 Section(a)(2)(B) which is referenced in Section 6 of the TS. Therefore, this change will not result in any decrease in the requirements of the TS. Based on the above, this is an administrative change that eliminates redundant reporting requirements and does not alter the requirements of the TS and is therefore acceptable.

2.2 Revision of Note 7.d for Table 3.2.C

Table 3.2.C requires both channels of the RBM to be operable except for its reference to note 7 which has four parts. The current note 7.d immediately prevents control rod movement if the minimum operable channels per trip function are not available; however, note 7.c allows one channel to be bypassed and inoperable for 24 hours without having to prevent control rod movement. The proposed amendments would change the TS because of the conflict between notes 7.c and 7.d. The proposed amendments would use wording and logic used in the STS to prevent control rod movement when RBM's are out of service. The new wording would eliminate the conflict between note 7.c and 7.d, to be consistent with STS and would address the possibility of both RBM channels being inoperable which is not specifically addressed at present. These proposed amendments would remove the conflict between notes 7.c and 7.d, clarify required operator actions and be consistent with the requirements of the STS. Based on the above evaluation, the staff finds the proposed amendments eliminate any conflicts or confusion with the current TS note 7 for Table 3.2.C and will enhance safe operation of the plant. Therefore, the proposed change is acceptable.

2.3 Correcting LCO 3.6.H.1

In amendments (Nos. 128, 133 and 99 for units 1, 2, and 3, respectively) Surveillance Requirement 4.6.H was revised to reference the correct SI containing the safety-related snubber list. However, the corresponding LCO reference was not changed and therefore has the incorrect reference. The proposed change would correct the LCO reference. Correcting the reference to the SI that lists safety-related snubbers is an administrative change that in no way affects TS requirements or operations and will not have any effect on the operation of the plant. Therefore, the staff finds the change acceptable.

2.4 Revision to Section 2.1.A.1.b for APRM Setpoint Reference

The proposed amendments would correct an error in the units 2 and 3 TS. The current TS incorrectly reference section 4.1.B of the TS where surveillance requirements for APRM scram setpoints can be found. Correcting the reference describing where to find APRM setpoint requirements is an administrative correction of an error and will not change any TS requirement or operations. This change was previously approved for unit 1 by amendment no. 128. Therefore, the staff finds this change acceptable.

2.5 Delete Note (14) Table 4.2.A

The proposed amendments would correct an error in the units 1, 2, and 3 TS by deleting note 14 of table 4.2.A. That note applies to the refueling zone and reactor building zone radiation monitoring instrument channels. It refers to requirements that do not exist in two other TS sections. In fact, one of the sections referenced does not exist and the other section referenced does not contain any testing requirements for these instruments. Deleting this note will not remove any required testing and will not have any effect on the operation of the plant. It will preclude any confusion which could be caused by the error. Therefore, based on the above evaluation, the staff finds the change acceptable.

2.6 Changes To Notes 9 and 11 Table 3.2.A and Note 22 Table 4.2.A

In a meeting with the licensee on October 29, 1987, the requirements for this specific TS change were discussed. The licensee proposed the TS change to close an open item in Inspection Report 85-05-07. The open item dealt with the confusion in the SI concerning the testing of radiation monitors in the Reactor Building-Reactor Zone and Refuel Zone listed in Table 3.2.A of the TS. Based on the discussions between the staff and the licensee it was determined that the best course of action was to change the SI and make it more specific concerning the testing and operability of the radiation monitors. In further discussions with Mr. M. G. May (TVA) it was agreed upon that the SI would be changed to correct the open item rather than the TS. The staff finds this acceptable and no change to the TS will be made. By letter dated January 22, 1988 the licensee provided additional clarification. This supplement did not change the application from that initially noticed on June 17, 1987 (52 FR 23107) and did not alter the staff's proposed no significant hazards determination.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments relate to changes in recordkeeping, reporting, or administrative procedures and requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

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

Principal Contributor: John Stang

Dated: March 3, 1988