

May 31, 1994

Docket Nos. 50-259
and 50-296

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

Dear Mr. Kingsley:

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE BROWNS FERRY
NUCLEAR PLANT UNITS 1 AND 3 (TAC NOS. M82650 AND M82652) (TS-300)

The Commission has issued the enclosed Amendment Nos. 206 and 179 to Facility
Operating License Nos. DPR-33 and DPR-68 for the Browns Ferry Nuclear Plant
(BFN), Units 1 and 3, respectively. These amendments revise the Technical
Specifications (TS) in response to your application dated January 14, 1992.

The amendments add requirements to the BFN Units 1 and 3 TS intended to ensure
thermal-hydraulic stability. These changes are consistent with guidance
provided by NRC Bulletin 88-07 "Power Oscillations in Boiling Water Reactors"
and Supplement 1 to that Bulletin.

The TS Bases changes for all three BFN units, also documented in your
January 14, 1992 letter, were issued on July 31, 1992.

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

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PDR ADOCK 05000259
P PDR

Enclosures:

1. Amendment No. 206 to
License No. DPR-33
2. Amendment No. 179 to
License No. DPR-68
3. Safety Evaluation

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cc w/enclosures:

See next page

OFC	PDII-4/AA	PDII-4/PM	PDII-4/PM	OGC	PDII-4/D
NAME	Boston	JWilliams	Trimble	Retard	FHebdon
DATE	5/4/94	5/4/94	5/14/94	5/19/94	5/31/94

DOCUMENT NAME: G:\BFN\TS300.AMD

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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. 206
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 January 14, 1992 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

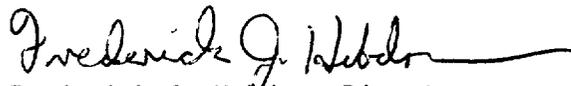
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-33 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 206, 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. Heddon, 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: May 31, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 206

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
ii
iii
iv
vii
viii
3.5/4.5-20
3.5/4.5-21
3.5/4.5-22
3.5/4.5-23
3.5/4.5-24
3.5/4.5-25
3.5/4.5-34
3.5/4.5-35
3.6/4.6-11
3.6/4.6-12
3.6/4.6-13
3.6/4.6-14
3.6/4.6-32
3.6/4.6-33
3.6/4.6-34
3.6/4.6-35

i*
ii
iii*
iv*
vii*
viii
3.5/4.5-20*
3.5/4.5-21
3.5/4.5-22
3.5/4.5-23
3.5/4.5-24
3.5/4.5-25
3.5/4.5-34
3.5/4.5-35*
3.6/4.6-11*
3.6/4.6-12
3.6/4.6-13
3.6/4.6-14**
3.6/4.6-32
3.6/4.6-33**
3.6/4.6-34
3.6/4,6-35*

TABLE OF CONTENTS

<u>Section</u>	<u>Page No.</u>
1.0	Definitions 1.0-1
<u>SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS</u>	
1.1/2.1	Fuel Cladding Integrity 1.1/2.1-1
1.2/2.2	Reactor Coolant System Integrity. 1.2/2.2-1
<u>LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS</u>	
3.1/4.1	Reactor Protection System.. . . . 3.1/4.1-1
3.2/4.2	Protective Instrumentation. 3.2/4.2-1
A.	Primary Containment and Reactor Building Isolation Functions. 3.2/4.2-1
B.	Core and Containment Cooling Systems - Initiation and Control 3.2/4.2-1
C.	Control Rod Block Actuation. 3.2/4.2-2
D.	(Deleted). 3.2/4.2-3
E.	Drywell Leak Detection 3.2/4.2-4
F.	Surveillance Instrumentation 3.2/4.2-4
G.	Control Room Isolation 3.2/4.2-4
H.	Flood Protection 3.2/4.2-4
I.	Meteorological Monitoring Instrumentation. . . 3.2/4.2-4
J.	Seismic Monitoring Instrumentation 3.2/4.2-5
K.	Explosive Gas Monitoring Instrumentation . . . 3.2/4.2-6
L.	ATWS Recirculation Pump Trip 3.2/4.2-6a
3.3/4.3	Reactivity Control. 3.3/4.3-1
A.	Reactivity Limitations 3.3/4.3-1
B.	Control Rods 3.3/4.3-5
C.	Scram Insertion Times. 3.3/4.3-9

<u>Section</u>	<u>Page No.</u>
D. Reactivity Anomalies	3.3/4.3-11
E. Reactivity Control	3.3/4.3-12
F. Scram Discharge Volume	3.3/4.3-12
3.4/4.4 Standby Liquid Control System	3.4/4.4-1
A. Normal System Availability	3.4/4.4-1
B. Operation with Inoperable Components	3.4/4.4-3
C. Sodium Pentaborate Solution.	3.4/4.4-3
3.5/4.5 Core and Containment Cooling Systems.	3.5/4.5-1
A. Core Spray System (CSS).	3.5/4.5-1
B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	3.5/4.5-4
C. RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS).	3.5/4.5-9
D. Equipment Area Coolers	3.5/4.5-13
E. High Pressure Coolant Injection System (HPCIS).	3.5/4.5-13
F. Reactor Core Isolation Cooling System (RCICS).	3.5/4.5-14
G. Automatic Depressurization System (ADS).	3.5/4.5-16
H. Maintenance of Filled Discharge Pipe	3.5/4.5-17
I. Average Planar Linear Heat Generation Rate	3.5/4.5-18
J. Linear Heat Generation Rate (LHGR)	3.5/4.5-18
K. Minimum Critical Power Ratio (MCPR).	3.5/4.5-19
L. APRM Setpoints	3.5/4.5-20
M. Core Thermal-Hydraulic Stability	3.5/4.5-21
3.6/4.6 Primary System Boundary	3.6/4.6-1
A. Thermal and Pressurization Limitations	3.6/4.6-1
B. Coolant Chemistry.	3.6/4.6-5
C. Coolant Leakage.	3.6/4.6-9

<u>Section</u>	<u>Page No.</u>
D. Relief Valves	3.6/4.6-10
E. Jet Pumps	3.6/4.6-11
F. Recirculation Pump Operation	3.6/4.6-12
G. Structural Integrity	3.6/4.6-13
H. Snubbers	3.6/4.6-15
3.7/4.7 Containment Systems	3.7/4.7-1
A. Primary Containment	3.7/4.7-1
B. Standby Gas Treatment System	3.7/4.7-13
C. Secondary Containment	3.7/4.7-16
D. Primary Containment Isolation Valves	3.7/4.7-17
E. Control Room Emergency Ventilation	3.7/4.7-19
F. Primary Containment Purge System	3.7/4.7-21
G. Containment Atmosphere Dilution System (CAD)	3.7/4.7-22
H. Containment Atmosphere Monitoring (CAM) System H ₂ Analyzer	3.7/4.7-24
3.8/4.8 Radioactive Materials	3.8/4.8-1
A. Liquid Effluents	3.8/4.8-1
B. Airborne Effluents	3.8/4.8-3
C. (Deleted).	3.8/4.8-4
D. Mechanical Vacuum Pump	3.8/4.8-4
E. Miscellaneous Radioactive Materials Sources.	3.8/4.8-5
F. (Deleted).	3.8/4.8-6
3.9/4.9 Auxiliary Electrical System	3.9/4.9-1
A. Auxiliary Electrical Equipment	3.9/4.9-1
B. Operation with Inoperable Equipment.	3.9/4.9-8
C. Operation in Cold Shutdown	3.9/4.9-15
D. Diesel Generators Required For Units 1, 2, and 3 Shared Systems	3.9/4.9-15a

<u>Section</u>	<u>Page No.</u>
3.10/4.10 Core Alterations	3.10/4.10-1
A. Refueling Interlocks	3.10/4.10-1
B. Core Monitoring.	3.10/4.10-5
C. Spent Fuel Pool Water.	3.10/4.10-7
D. Reactor Building Crane	3.10/4.10-8
E. Spent Fuel Cask.	3.10/4.10-9
F. Spent Fuel Cask Handling-Refueling Floor	3.10/4.10-10
3.11/4.11 Deleted.	3.11/4.11-1
5.0 Major Design Features	5.0-1
5.1 Site Features.	5.0-1
5.2 Reactor.	5.0-1
5.3 Reactor Vessel	5.0-1
5.4 Containment.	5.0-1
5.5 Fuel Storage	5.0-1
5.6 Seismic Design	5.0-2

LIST OF TABLES (Cont'd)

<u>Table</u>	<u>Title</u>	<u>Page No.</u>
4.2.E	Minimum Test and Calibration Frequency for Drywell Leak Detection Instrumentation	3.2/4.2-53
4.2.F	Minimum Test and Calibration Frequency for Surveillance Instrumentation	3.2/4.2-54
4.2.G	Surveillance Requirements for Control Room Isolation Instrumentation.	3.2/4.2-56
4.2.H	Minimum Test and Calibration Frequency for Flood Protection Instrumentation	3.2/4.2-57
4.2.J	Seismic Monitoring Instrument Surveillance Requirements	3.2/4.2-58
4.2.K	Explosive Gas Instrumentation Surveillance	3.2/4.2-62
4.2.L	ATWS-Recirculation Pump Trip (RPT) Instrumentation Surveillance	3.2/4.2-63a
3.5-1	Minimum RHRSW and EECW Pump Assignment	3.5/4.5-11
4.9.A	Auxiliary Electrical Systems	3.9/4.9-16
4.9.A.4.C	Voltage Relay Setpoints/Diesel Generator Start	3.9/4.9-18
6.2.A	Minimum Shift Crew Requirements.	6.0-4

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page No.</u>
2.1.1	APRM Flow Reference Scram and APRM Rod Block Settings	1.1/2.1-6
2.1-2	APRM Flow Bias Scram Vs. Reactor Core Flow	1.1/2.1-7
4.1-1	Graphical Aid in the Selection of an Adequate Interval Between Tests	3.1/4.1-13
4.2-1	System Unavailability.	3.2/4.2-64
3.5.M-1	BFN Power/Flow Stability Regions	3.5/4.5-22
3.6-1	Minimum Temperature °F Above Change in Transient Temperature.	3.6/4.6-24
4.8.1.a	Gaseous Release Points and Elevations	3.8/4.8-7
4.8.1.b	Land Site Boundary	3.8/4.8-8

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5.K Minimum Critical Power Ratio (MCPR)

4.5.K Minimum Critical Power Ratio (MCPR)

4.5.K.2 (Cont'd)

- b. T 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.

L. APRM Setpoints

1. Whenever the core thermal power is $\geq 25\%$ 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 $\leq 25\%$ of rated thermal power within 4 hours.

L. APRM Setpoints

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

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

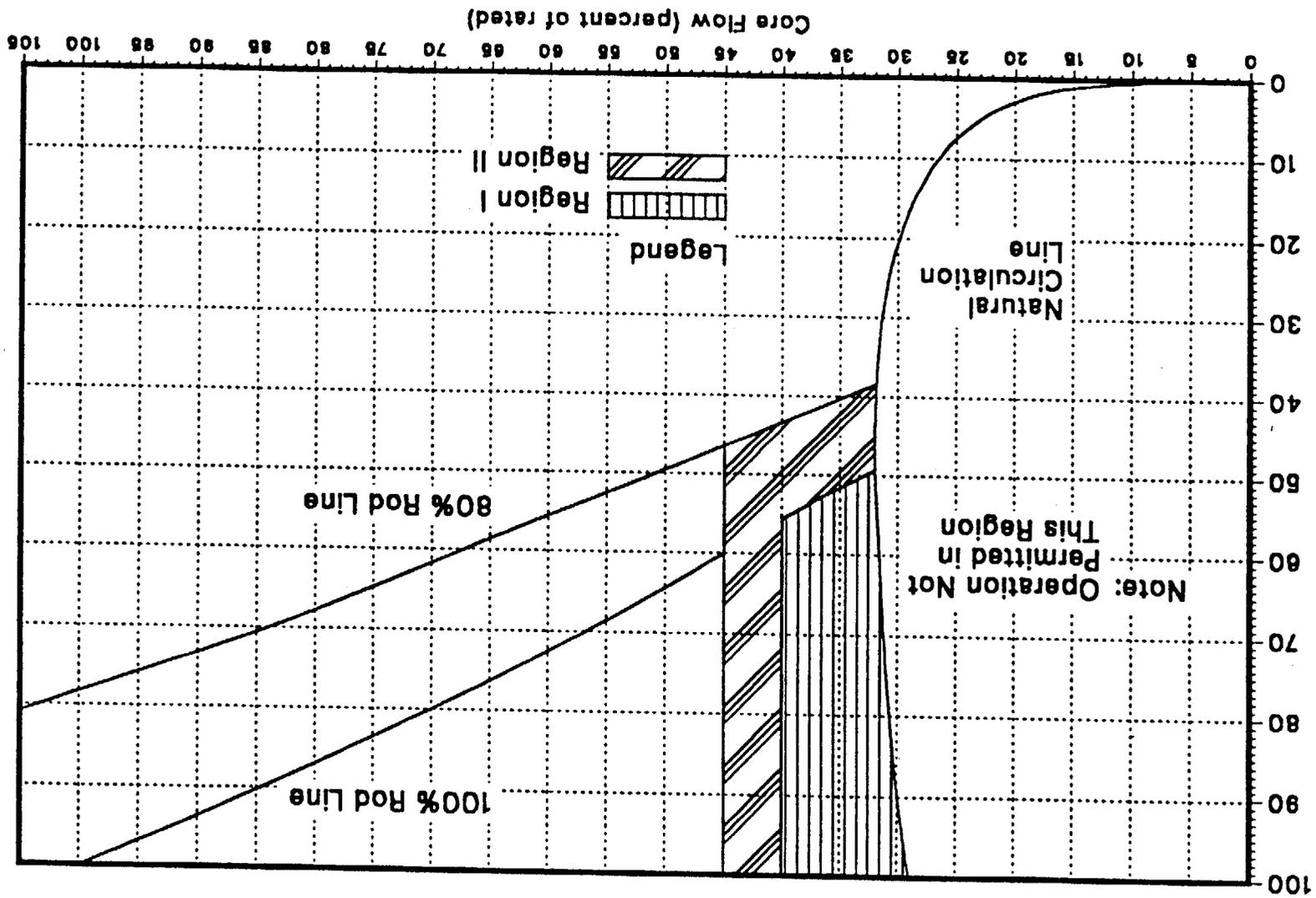
3.5.M Core Thermal-Hydraulic Stability

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.
2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.
3. If Region II of Figure 3.5.M-1 is entered:
 - a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and
 - b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

4.5.M Core Thermal-Hydraulic Stability

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:
 - a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and
 - b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

Figure 3.5.M-1
BFN Power/Flow Stability Regions



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3.5 BASES (Cont'd)

3.5.M. Core Thermal-Hydraulic Stability

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of Figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

During regional oscillations, the safety limit MCPR is not approached until APRM oscillations are 30 percent peak-to-peak or larger in magnitude. In addition, periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. 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 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 in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, 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.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, 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.

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.D. Relief Valves

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:
 - a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.
 - b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
 - c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
2. Following one pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation

SURVEILLANCE REQUIREMENTS

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating with the equalizer valve closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

4.6.F Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.
2. No additional surveillance required.
3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

temperature of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

SURVEILLANCE REQUIREMENTS

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

3.6.G Structural Integrity

3.6.G.1 (Cont'd)

- b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

SURVEILLANCE REQUIREMENTS

4.6.G. Structural Integrity

- 3. For Unit 1 an augmented inservice surveillance program shall be performed to monitor potential corrosive effects of chloride residue released during the March 22, 1975 fire. The augmented inservice surveillance program is specified as follows:
 - a. Browns Ferry Mechanical Maintenance Instruction 53, dated September 22, 1975, paragraph 4, defines the liquid penetrant examinations required during the first, second, third and fourth refueling outages following the fire restoration.
 - b. Browns Ferry Mechanical Maintenance Instruction 46, dated July 18, 1975, Appendix B, defines the liquid penetrant examinations required during the sixth refueling outage following the fire restoration.

3.6/4.6 BASES

3.6.E/4.6.E (Cont'd)

If they do differ by 10 percent or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10 percent or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the unit shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Operation without forced recirculation is permitted for up to 12 hours when the reactor is not in the RUN mode. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring at least one recirculation pump to be OPERABLE while in the RUN mode (i.e., requiring a manual scram if both recirculation pumps are tripped) provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50% of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6/4.6 BASES

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in plant procedures to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

An augmented inservice surveillance program is required to determine whether any stress corrosion has occurred in any stainless steel piping, stainless components, and highly-stressed alloy steel such as hanger springs, as a result of environmental conditions associated with the March 22, 1975 fire.

REFERENCES

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)
2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code
3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)
4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)
5. Mechanical Maintenance Instruction 46 (Mechanical Equipment, Concrete, and Structural Steel Cleaning Procedure for Residue From Plant Fire - Units 1 and 2)

3.6/4.6 BASES

REFERENCES (Cont'd)

6. Mechanical Maintenance Instruction 53 (Evaluation of Corrosion Damage of Piping Components Which Were Exposed to Residue From March 22, 1975 Fire)
7. Plant Safety Analysis (BFNP FSAR Subsection 4.12)

3.6.H/4.6.H Snubbers

Snubbers are designed to prevent unrestrained pipe or component motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping or components as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the protection is required only during relatively low probability events, a period of 72 hours is allowed to replace or restore the inoperable snubber to operable status and perform an engineering evaluation on the supported component or declare the supported system inoperable and follow the appropriate limiting condition for operation statement for that system. The engineering evaluation is performed to determine whether the mode of failure of the snubber has adversely affected any safety-related component or system.

To verify snubber operability FUNCTIONAL TESTS shall be performed during the refueling outages, at approximately 18-month intervals.

These tests will include stroking of the snubbers to verify proper movement, activation, and bleed or release. Ten percent represents an adequate sample for such tests. Observed failures on these samples will require an engineering analysis and testing of additional units. If the engineering analysis results in the determination that the failure of a snubber to activate or to stroke (i.e., seized components) is the result of manufacture or design deficiency, all snubbers subject to the same defect shall be functionally tested. A thorough inspection of the snubber threaded attachments to the pipe or components and the anchorage will be made in conjunction with all required FUNCTIONAL TESTS. The stroke setting of the snubbers selected for functional testing also will be verified.

All safety-related snubbers are also visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate fluid level if applicable, and proper attachment of the snubber to piping and structures. The removal of insulation or the verification of torque values for threaded fasteners is not required for visual inspections.

3.6/4.6 BASES (Continued)

The visual inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25 percent) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber in a visual inspection is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible and operability verified by inservice functional testing, if applicable, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. Inspection groups may be established based on design features, and installed conditions which may be expected to be generic. Each of these inspection groups is inspected and tested separately unless an engineering analysis indicates the inspection group is improperly constituted. All suspect snubbers are subject to inspection and testing regardless of inspection groupings.



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. 179
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 January 14, 1992 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.

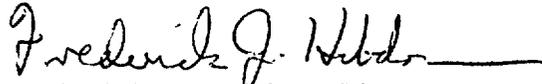
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-68 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 179, 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. Heddon, 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: **May 31, 1994**

ATTACHMENT TO LICENSE AMENDMENT NO. 179

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

i
ii
iii
iv
vii
viii
3.5/4.5-20
3.5/4.5-20a
3.5/4.5-21
3.5/4.5-22
3.5/4.5-23
3.5/4.5-24
3.5/4.5-25
3.5/4.5-26
3.5/4.5-35
3.5/4.5-36
3.6/4.6-11
3.6/4.6-12
3.6/4.6-13
3.6/4.6-14
3.6/4.6-32
3.6/4.6-33

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i*
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vii*
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3.5/4.5-20*
3.5/4.5-20a
3.5/4.5-21
3.5/4.5-22
3.5/4.5-23
3.5/4.5-24
3.5/4.5-25
3.5/4.5-26
3.5/4.5-35
3.5/4.5-36*
3.6/4.6-11*
3.6/4.6-12
3.6/4.6-13
3.6/4.6-14**
3.6/4.6-32
3.6/4.6-33**

TABLE OF CONTENTS

<u>Section</u>		<u>Page No.</u>
1.0	Definitions	1.0-1
<u>SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS</u>		
1.1/2.1	Fuel Cladding Integrity	1.1/2.1-1
1.2/2.2	Reactor Coolant System Integrity.	1.2/2.2-1
<u>LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS</u>		
3.1/4.1	Reactor Protection System	3.1/4.1-1
3.2/4.2	Protective Instrumentation.	3.2/4.2-1
A.	Primary Containment and Reactor Building Isolation Functions.	3.2/4.2-1
B.	Core and Containment Cooling Systems - Initiation and Control	3.2/4.2-1
C.	Control Rod Block Actuation.	3.2/4.2-2
D.	(Deleted).	3.2/4.2-3
E.	Drywell Leak Detection	3.2/4.2-4
F.	Surveillance Instrumentation	3.2/4.2-4
G.	Control Room Isolation	3.2/4.2-4
H.	Flood Protection	3.2/4.2-4
I.	Meteorological Monitoring Instrumentation.	3.2/4.2-4
J.	Seismic Monitoring Instrumentation	3.2/4.2-5
K.	Explosive Gas Monitoring Instrumentation	3.2/4.2-6
L.	ATWS-Recirculation Pump Trip	3.2/4.2-6a
3.3/4.3	Reactivity Control.	3.3/4.3-1
A.	Reactivity Limitations	3.3/4.3-1
B.	Control Rods	3.3/4.3-5
C.	Scram Insertion Times.	3.3/4.3-9

<u>Section</u>	<u>Page No.</u>
D. Reactivity Anomalies	3.3/4.3-11
E. Reactivity Control	3.3/4.3-12
F. Scram Discharge Volume	3.3/4.3-12
3.4/4.4 Standby Liquid Control System	3.4/4.4-1
A. Normal System Availability	3.4/4.4-1
B. Operation with Inoperable Components	3.4/4.4-3
C. Sodium Pentaborate Solution.	3.4/4.4-3
3.5/4.5 Core and Containment Cooling Systems.	3.5/4.5-1
A. Core Spray System (CSS).	3.5/4.5-1
B. Residual Heat Removal System (RHRS) (LPCI and Containment Cooling)	3.5/4.5-4
C. RHR Service Water and Emergency Equipment Cooling Water Systems (EECWS).	3.5/4.5-9
D. Equipment Area Coolers	3.5/4.5-13
E. High Pressure Coolant Injection System (HPCIS).	3.5/4.5-13
F. Reactor Core Isolation Cooling System (RCICS).	3.5/4.5-14
G. Automatic Depressurization System (ADS).	3.5/4.5-16
H. Maintenance of Filled Discharge Pipe	3.5/4.5-17
I. Average Planar Linear Heat Generation Rate	3.5/4.5-18
J. Linear Heat Generation Rate (LHGR)	3.5/4.5-18
K. Minimum Critical Power Ratio (MCPR).	3.5/4.5-19
L. APRM Setpoints	3.5/4.5-20
M. Core Thermal-Hydraulic Stability	3.5/4.5-20a
3.6/4.6 Primary System Boundary	3.6/4.6-1
A. Thermal and Pressurization Limitations	3.6/4.6-1
B. Coolant Chemistry.	3.6/4.6-5
C. Coolant Leakage.	3.6/4.6-9

Section

Page No.

D.	Relief Valves	3.6/4.6-10
E.	Jet Pumps	3.6/4.6-11
F.	Recirculation Pump Operation	3.6/4.6-12
G.	Structural Integrity	3.6/4.6-13
H.	Snubbers	3.6/4.6-15
3.7/4.7	Containment Systems	3.7/4.7-1
A.	Primary Containment.	3.7/4.7-1
B.	Standby Gas Treatment System	3.7/4.7-13
C.	Secondary Containment.	3.7/4.7-16
D.	Primary Containment Isolation Valves	3.7/4.7-17
E.	Control Room Emergency Ventilation	3.7/4.7-19
F.	Primary Containment Purge System	3.7/4.7-21
G.	Containment Atmosphere Dilution System (CAD)	3.7/4.7-22
H.	Containment Atmosphere Monitoring (CAM) System H ₂ Analyzer	3.7/4.7-23a
3.8/4.8	Radioactive Materials	3.8/4.8-1
A.	Liquid Effluents	3.8/4.8-1
B.	Airborne Effluents	3.8/4.8-3
C.	(Deleted).	3.8/4.8-4
D.	Mechanical Vacuum Pump	3.8/4.8-4
E.	Miscellaneous Radioactive Materials Sources	3.8/4.8-5
F.	(Deleted).	3.8/4.8-6
3.9/4.9	Auxiliary Electrical System	3.9/4.9-1
A.	Auxiliary Electrical Equipment	3.9/4.9-1
B.	Operation with Inoperable Equipment.	3.9/4.9-8
C.	Operation in Cold Shutdown Condition	3.9/4.9-14
D.	Diesel Generators Required for Units 1, 2, and 3 Shared Systems	3.9/4.9-14a

<u>Section</u>	<u>Page No.</u>
3.10/4.10 Core Alterations	3.10/4.10-1
A. Refueling Interlocks	3.10/4.10-1
B. Core Monitoring	3.10/4.10-5
C. Spent Fuel Pool Water	3.10/4.10-7
D. Reactor Building Crane	3.10/4.10-8
E. Spent Fuel Cask	3.10/4.10-9
F. Spent Fuel Cask Handling-Refueling Floor.	3.10/4.10-9
 3.11/4.11 Deleted.	 3.11/4.11-1
 5.0 Major Design Features	 5.0-1
5.1 Site Features	5.0-1
5.2 Reactor	5.0-1
5.3 Reactor Vessel	5.0-1
5.4 Containment	5.0-1
5.5 Fuel Storage	5.0-1
5.6 Seismic Design	5.0-2

LIST OF TABLES (Cont'd)

<u>Table</u>	<u>Title</u>	<u>Page No.</u>
4.2.E	Minimum Test and Calibration Frequency for Drywell Leak Detection Instrumentation.	3.2/4.2-52
4.2.F	Minimum Test and Calibration Frequency for Surveillance Instrumentation	3.2/4.2-53
4.2.G	Surveillance Requirements for Control Room Isolation Instrumentation.	3.2/4.2-55
4.2.H	Minimum Test and Calibration Frequency for Flood Protection Instrumentation	3.2/4.2-56
4.2.J	Seismic Monitoring Instrument Surveillance Requirements	3.2/4.2-57
4.2.K	Explosive Gas Instrumentation Surveillance	3.2/4.2-61
4.2.L	ATWS-Recirculation Pump Trip (RPT) Instrumentation Surveillance.	3.2/4.2-62a
3.5-1	Minimum RHRSW and EECW Pump Assignment.	3.5/4.5-11
4.9.A	Auxiliary Electrical System	3.9/4.9-15
4.9.A.4.C	Voltage Relay Setpoints/Diesel Generator Start. . .	3.9/4.9-17
6.2.A	Minimum Shift Crew Requirements	6.0-4

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page No.</u>
2.1.1	APRM Flow Reference Scram and APRM Rod Block Settings	1.1/2.1-6
2.1-2	APRM Flow Bias Scram Vs. Reactor Core Flow	1.1/2.1-7
4.1-1	Graphical Aid in the Selection of an Adequate Interval Between Tests	3.1/4.1-12
4.2-1	System Unavailability.	3.2/4.2-63
3.5.M-1	BFN Power/Flow Stability Regions	3.5/4.5-21
3.6-1	Minimum Temperature °F Above Change in Transient Temperature.	3.6/4.6-24
4.8.1.a	Gaseous Release Points and Elevation	3.8/4.8-7
4.8.1.b	Land Site Boundary	3.8/4.8-8

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.5 Core and Containment Cooling Systems

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

L. APRM Setpoints

1. Whenever the core thermal power is $\geq 25\%$ 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:

$$S_{\Delta} (0.66W + 54\%) \frac{FRP}{CMFLPD}$$

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

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.

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

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

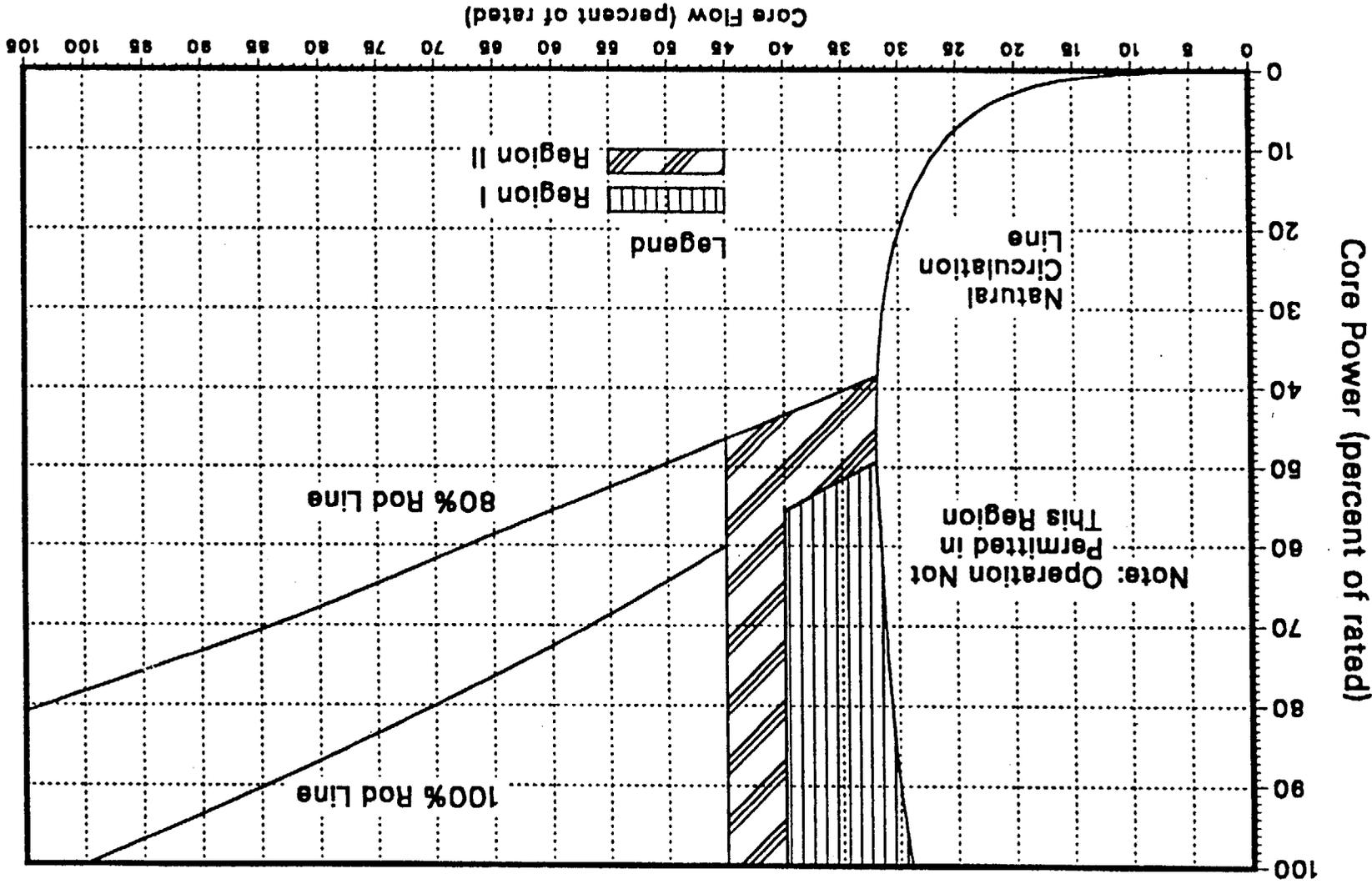
3.5.M Core Thermal-Hydraulic Stability

1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.
2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.
3. If Region II of Figure 3.5.M-1 is entered:
 - a. Immediately initiate action and exit the region within 2 hours by inserting control rods or by increasing core flow (starting a recirculation pump to exit the region is not an appropriate action), and
 - b. While exiting the region, immediately initiate a manual scram if thermal-hydraulic instability is observed, as evidenced by APRM oscillations which exceed 10 percent peak-to-peak of rated or LPRM oscillations which exceed 30 percent peak-to-peak of scale. If periodic LPRM upscale or downscale alarms occur, immediately check the APRM's and individual LPRM's for evidence of thermal-hydraulic instability.

4.5.M Core Thermal-Hydraulic Stability

1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:
 - a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and
 - b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

Figure 3.5.M-1
BFN Power/Flow Stability Regions



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3.5 BASES (Cont'd)

beyond that allowed by the one-percent plastic strain limit. A six-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

3.5.M. Core Thermal-Hydraulic Stability

The minimum margin to the onset of thermal-hydraulic instability occurs in Region I of Figure 3.5.M-1. A manually initiated scram upon entry into this region is sufficient to preclude core oscillations which could challenge the MCPR safety limit.

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of Figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

Clear indications of thermal-hydraulic instability are APRM oscillations which exceed 10 percent peak-to-peak or LPRM oscillations which exceed 30 percent peak-to-peak (approximately equivalent to APRM oscillations of 10 percent during regional oscillations). Periodic LPRM upscale or downscale alarms may also be indicators of thermal hydraulic instability and will be immediately investigated.

During regional oscillations, the safety limit MCPR is not approached until APRM oscillations are 30 percent peak-to-peak or larger in magnitude. In addition, periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

Normal operation of the reactor is restricted to thermal power and core flow conditions (i.e., outside Regions I and II) where thermal-hydraulic instabilities are very unlikely to occur.

3.5.N. 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 in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

When components and subsystems are out-of-service, overall core and containment cooling reliability is maintained by OPERABILITY of the remaining redundant equipment.

Whenever a CSCS system or loop is made inoperable, 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.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, 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.

3.6.E. Jet Pumps

1. Whenever the reactor is in the STARTUP or RUN modes, all jet pumps shall be OPERABLE. If it is determined that a jet pump is INOPERABLE, or if two or more jet pump flow instrument failures occur and cannot be corrected within 12 hours, an orderly shutdown shall be initiated and the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

4.6.D. Relief Valves

3. The integrity of the relief valve bellows shall be continuously monitored when valves incorporating the bellows design are installed.
4. At least one relief valve shall be disassembled and inspected each operating cycle.

E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the STARTUP or RUN modes with both recirculation pumps running, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:
 - a. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.
 - b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.
 - c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.F Recirculation Pump Operation

1. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service, the plant shall be placed in a HOT SHUTDOWN CONDITION within 24 hours unless the loop is sooner returned to service.
2. Following one-pump operation, the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.
3. When the reactor is not in the RUN mode, REACTOR POWER OPERATION with both recirculation pumps out-of-service for up to 12 hours is permitted. During such interval restart of the recirculation pumps is permitted, provided the loop discharge temperature is within 75°F of the saturation temperature

4.6.E. Jet Pumps

2. Whenever there is recirculation flow with the reactor in the STARTUP or RUN Mode and one recirculation pump is operating with the equalizer valve closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of an individual jet pump in a loop shall not vary from the mean of all jet pump differential pressures in that loop by more than 10%.

4.6.F Recirculation Pump Operation

1. Recirculation pump speeds shall be checked and logged at least once per day.
2. No additional surveillance required.
3. Before starting either recirculation pump during REACTOR POWER OPERATION, check and log the loop discharge temperature and dome saturation temperature.

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.F Recirculation Pump Operation

3.6.F.3 (Cont'd)

of the reactor vessel water as determined by dome pressure. The total elapsed time in natural circulation and one pump operation must be no greater than 24 hours.

4. The reactor shall not be operated with both recirculation pumps out-of-service while the reactor is in the RUN mode. Following a trip of both recirculation pumps while in the RUN mode, immediately initiate a manual reactor scram.

3.6.G Structural Integrity

1. The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained in accordance with Specification 4.6.G throughout the life of the plant.
 - a. With the structural integrity of any ASME Code Class 1 equivalent component, which is part of the primary system, not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or maintain the reactor coolant system in either a Cold Shutdown condition or less than 50°F above the minimum temperature required by NDT considerations, until each indication of a defect has been investigated and evaluated.

4.6.G Structural Integrity

1. Inservice inspection of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
2. Additional inspections shall be performed on certain circumferential pipe welds to provide additional protection against pipe whip, which could damage auxiliary and control systems.

3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

3.6.G Structural Integrity

4.6.G Structural Integrity

3.6.G.1 (Cont'd)

- b. With the structural integrity of any ASME Code Class 2 or 3 equivalent component not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from all OPERABLE systems.

3.6/4.6 BASES

3.6.E/4.6.E (Cont'd)

area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115 percent to 120 percent for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3 percent to 6 percent) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a jet pump diffuser body; however, the converse is not true. The lack of any substantial stress in the jet pump diffuser body makes failure impossible without an initial nozzle-riser system failure.

3.6.F/4.6.F Recirculation Pump Operation

Operation without forced recirculation is permitted up to 12 hours when the reactor is not in the RUN mode. And the start of a recirculation pump from the natural circulation condition will not be permitted unless the temperature difference between the loop to be started and the core coolant temperature is less than 75°F. This reduces the positive reactivity insertion to an acceptably low value.

Requiring at least one recirculation pump to be OPERABLE while in the RUN mode (i.e., requiring a manual scram if both recirculation pumps are tripped) provides protection against the potential occurrence of core thermal-hydraulic instabilities at low flow conditions.

Requiring the discharge valve of the lower speed loop to remain closed until the speed of the faster pump is below 50 percent of its rated speed provides assurance when going from one-to-two pump operation that excessive vibration of the jet pump risers will not occur.

3.6.G/4.6.G Structural Integrity

The requirements for the reactor coolant systems inservice inspection program have been identified by evaluating the need for a sampling examination of areas of high stress and highest probability of failure in the system and the need to meet as closely as possible the requirements of Section XI, of the ASME Boiler and Pressure Vessel Code.

3.6/4.6 BASES

3.6.G/4.6.G (Cont'd)

The program reflects the built-in limitations of access to the reactor coolant systems.

It is intended that the required examinations and inspection be completed during each 10-year interval. The periodic examinations are to be done during refueling outages or other extended plant shutdown periods.

Only proven nondestructive testing techniques will be used.

More frequent inspections shall be performed on certain circumferential pipe welds as listed in plant procedures to provide additional protection against pipe whip. These welds were selected in respect to their distance from hangers or supports wherein a failure of the weld would permit the unsupported segments of pipe to strike the drywell wall or nearby auxiliary systems or control systems. Selection was based on judgment from actual plant observation of hanger and support locations and review of drawings. Inspection of all these welds during each 10-year inspection interval will result in three additional examinations above the requirements of Section XI of ASME Code.

References

1. Inservice Inspection and Testing (BFNP FSAR Subsection 4.12)
2. Inservice Inspection of Nuclear Reactor Coolant Systems, Section XI, ASME Boiler and Pressure Vessel Code
3. ASME Boiler and Pressure Vessel Code, Section III (1968 Edition)
4. American Society for Nondestructive Testing No. SNT-TC-1A (1968 Edition)



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

ENCLOSURE 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1 AND 3

DOCKET NOS. 50-259 AND 50-296

1.0 INTRODUCTION

By letter dated January 14, 1992, the Tennessee Valley Authority (the licensee) requested Technical Specification (TS) changes for the Browns Ferry Nuclear Plant (BFN) Units 1 and 3. The proposed changes define regions on the operating power-flow map and operating restrictions on activities relating to those regions. These same changes had been previously approved for BFN Unit 2 on October 5, 1989. There were also proposed changes to the Bases for TS 4.2 for all three units.

The proposed regions and restrictions for BFN Units 1 and 3 are intended to avoid problems with thermal-hydraulic instability. Design requirements to avoid this instability are given in 10 CFR 50, Appendix A, General Design Criterion 12, "Suppression of reactor power oscillations."

Thermal-hydraulic instability concerns have been a focus of NRC attention following the LaSalle instability event of March 1988. This attention resulted in the issuance of NRC Bulletin (NRCB) 88-07 and NRCB 88-07, Supplement 1. NRCB 88-07 and NRCB 88-07, Supplement 1 requested utilities to provide operator training, instrumentation verification, and operating procedures intended to minimize instability potential or consequences. The requested operating procedures of NRCB 88-07, Supplement 1 are based on the General Electric (GE) Interim Recommendations for Stability Actions (IRSA), and are presented in an attachment to the supplement. These recommendations, along with other NRC staff requests presented in the supplement, constitute current NRC recommendations for BWR thermal-hydraulic stability (THS) operations. They were the result of calculations and reviews by the NRC, GE, the BWR Owner's Group (BWROG), and associated consultants.

NRCB 88-07, Supplement 1 requested that licensees implement the IRSA (and other associated requests) by modifying relevant procedures. Modification of the TS was not specifically requested. However, several licensees have modified their TS to correspond to the bulletin requests. Since BFN Unit 2 did not have stability-related TS when startup was requested in 1989, the NRC indicated that, in addition to procedural changes, the licensee should provide TS addressing NRCB 88-07, Supplement 1 requests before BFN Unit 2 restart. The licensee has now also requested the same TS modifications for BFN Units 1 and 3.

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The proposed changes to the BFN Units 1 and 3 TS are (1) addition of TS 3.5.M.1, 3.5.M.2, 3.5.M.3, 4.5.M.1, Figure 3.5.M.1 and the addition of the associated Bases 3.5.M, and (2) changes to TS 3.6.F.3 and 4.6.F.3, the addition of 3.6.F.4 and additions to the Bases for 3.6.F/4.6.F. There are also associated changes to the Table of Contents and List of Illustrations. In addition, the licensee's letter of January 14, 1992 proposed changes to Bases 4.2, unrelated to THS, for all three units which clarify testing requirements for high pressure coolant injection and reactor core isolation cooling. This change to Bases 4.2 was accepted by the staff in a letter dated July 31, 1992.

2.0 EVALUATION

The IRSA specify three regions (A, B, and C) on the power-flow map involving different degrees of allowed or prohibited operation. These are bounded by constant flow lines or control rod lines (lines of flow variation with all other reactor parameters, particularly control rod position, held constant). Region A is above the 100 percent rod line (intercepts 100 percent rated power at 100 percent rated flow) and below 40 percent flow. Region B is between the 80 and 100 percent rod lines and below 40 percent flow. Region C is above the 80 percent rod line and between 40 and 45 percent flow. Deliberate entry into regions A and B is not permitted. If it occurs, immediate exit is required. For a Group 2 plant (such as BFN Units 1 and 3), an immediate scram is required in region A, while for region B, control rod insertion or flow increase may be used to exit. Operations may be conducted in region C, with suitable surveillance, if required during startup to prevent fuel damage. If during operations in regions B or C, instability occurs, the reactor shall be immediately scrammed, with evidence for instability coming from Average Power Range Monitor (APRM) oscillation greater than 10 percent or Local Power Range Monitor (LPRM) upscale or downscale alarms.

The proposed BFN Units 1 and 3 TS conservatively implement these region designations and associated operation requirements by adding a new specification, TS 3/4.5.M, Core Thermal-Hydraulic Stability, and a power-flow map, Figure 3.5.M-1. The regions designated in Figure 3.5.M-1 are the same as in IRSA, except that regions B and C are combined into a single Region II, with region A designated Region I. The IRSA operating restrictions of Region B are conservatively applied throughout Region II. There is no allowed operation such as is permitted by IRSA, such as for startup in Region C. TS 3.5.M.1, 3.5.M.2, and 3.5.M.3 specify that operation is not permitted in Regions I and II. Upon inadvertent entry, a reactor scram is required if in Region I, and immediate action to depart by control rod insertion or flow increase is required for Region II. While exiting Region II, scram is required if there are indications of instability as evidenced by APRM oscillations above 10 percent peak-to-peak of rated power or LPRM oscillations above 30 percent, and LPRM upscale or downscale alarms require immediate checks of APRM and LPRM readings. These requirements all meet or exceed the IRSA specifications, and are acceptable for meeting the bulletin requests for implementing the interim recommendations. TS 4.5.M provides appropriate surveillance requirements for determining that operation is outside of Regions I and II when operating in the vicinity of these regions, and are also acceptable. The new Bases 3.5.M provides a reasonable discussion of the

background, regions, operations, and requirements for these specifications and is also acceptable.

NRCB 88-07, Supplement 1 also requested that plants which do not have effective automatic scram protection for regional oscillations (Group 2 plants in the IRSA), should initiate a manual reactor scram when two recirculation pumps trip (or with no pumps operating) with the reactor in the RUN mode. BFN Units 1 and 3 are Group 2 plants, and the proposed addition of TS 3.6.F.4 to recirculation pump requirements is intended to comply with this request. It specifies that the reactor shall not be operated in the RUN mode with both recirculation pumps out-of-service, and an immediate manual scram is required, in the RUN mode, following a trip of both recirculation pumps. This is an acceptable implementation of the NRCB 88-07 recommendation.

There are also modifications to TS 3/4.6.F.3 which currently permits operation for up to 12 hours with both recirculation pumps out-of-service. The modifications permit such operation at power only while not in the RUN mode (i.e., permitted only at low power). This change is consistent with the requirements for recirculation pump operation discussed above, and is acceptable.

The staff concludes that the proposed TS changes and the material submitted to support the changes are acceptable. It should be noted however, that the NRC staff, its consultants, the BWROG, GE, and others are continuing the review of THS concerns. The BWROG is developing several long-term solutions for this problem. In connection with the experience gained in that work, the BWROG, in a March 18, 1992 letter to BWROG representatives, provided further "Implementation Guidance for Stability Interim Corrective Actions." The NRC endorses this guidance, and recommends that it be considered as useful enhancement of current guidance.

3.0 SUMMARY

The NRC staff has reviewed the reports submitted by TVA for BFN Units 1 and 3, proposing TS changes relating to THS requirements for power-flow map operating constraints and surveillance. Based on this review, the staff concludes that appropriate documentation was submitted and the proposed power-flow action regions, surveillance and TS changes satisfy staff positions and requirements in these areas. Operation in the modes proposed is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (57 FR 13138). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Howard Richings

Dated: ~~May~~ 31, 1994