October 21, 1993

Docket No. 50-260

Dr. Mark O. Medford, Vice President
Technical Support
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

Dear Dr. Medford:

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS REGARDING FLOW-BIASED ROD BLOCK MONITOR UPSCALE SETPOINT (TAC NO. M84395) (TS 303)

The Commission has issued the enclosed Amendment No. 217 to Facility Operating License No. DPR-52 for the Browns Ferry Nuclear Plant (BFN), Unit 2. This amendment is in response to your application dated October 5, 1992, and supplemented September 15, 1993, requesting changes to the BFN Unit 2 Technical Specifications to change the flow-biased rod block monitor upscale setpoint, and to make other associated changes.

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

Sincerely,

ORIGINAL SIGNED BY:

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

Enclosures:
1. Amendment No. 217 to License No. DPR-52
2. Safety Evaluation

cc w/enclosures:
See next page

OGC S. Horn
10/8/93

OFC	PD II-4LA	PD II-4PM	PD II-4PM	PD II-4PM	SRXB	PD II-4D
NAME	BClayton	JWilliams	DTrimble	TRoss	R. Jones	FHebdon
DATE	9/28/93	9/28/93	9/30/93	9/30/93	9/30/93	10/21/93

DOCUMENT NAME: TS303.AMD

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Tennessee Valley Authority
ATTN: Dr. Mark O. Medford

BROWNS FERRY NUCLEAR PLANT

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 217
License No. DPR-52

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

A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 5, 1992, as supplemented September 15, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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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-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. 217, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

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

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: October 21, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 217

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

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

REMOVE	INSERT
1.0-7	1.0-7*
1.0-8	1.0-8
1.0-9	1.0-9**
1.0-10	1.0-10*
1.0-12a	1.0-12a
1.0-12b	1.0-12b*
3.2/4.2-25	3.2/4.2-25
3.2/4.2-25a	3.2/4.2-25a*
3.2/4.2-27a	3.2/4.2-27a
3.2/4.2-27b	3.2/4.2-27b*
3.3/4.3-7	3.3/4.3-7*
3.3/4.3-8	3.3/4.3-8
3.3/4.3-17	3.3/4.3-17
3.3/4.3-18	3.3/4.3-18*
3.5/4.5-18	3.5/4.5-18*
3.5/4.5-19	3.5/4.5-19

Q. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

R. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

S. CORE ALTERATION - CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, intermediate range monitors, traversing in-core probes, or special movable detectors (including undervessel replacement) is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe location.

T. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

U. Thermal Parameters

1. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

2. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

3. Core Maximum Fraction of Limiting Power Density (CMFLPD) - The highest ratio, for all fuel assemblies and all axial locations in the core, of the maximum fuel rod power density (kW/ft) for a given fuel assembly and axial location to the limiting fuel rod power density (kW/ft) at that location.

4. Average Planar Linear Heat Generation Rate (APLHGR) - The Average Planar Heat Generation Rate is applicable to a specific planar height and is equal to the sum of the linear heat generation rates for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

1.0 DEFINITIONS (Cont'd)

5. Core Maximum Fraction of Critical Power (CMFCP) - Core Maximum Fraction of Critical Power is the maximum value of the ratio of the flow-corrected CPR operating limit found in the CORE OPERATING LIMITS REPORT divided by the actual CPR for all fuel assemblies in the core.

V. Instrumentation

1. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

2. Channel - A channel is an arrangement of the sensor(s) and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

3. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

4. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

6. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

7. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

8. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

1.0 DEFINITIONS (Cont'd)

9. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

10. Logic - A logic is an arrangement of relays, contacts, and other components that produces a decision output.

    (a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

    (b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

11. Channel Calibration - Shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameters which the channel monitors. The channel calibration shall encompass the entire channel including alarm and/or trip functions and shall include the channel functional test. The channel calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated. Non-calibratable components shall be excluded from this requirement, but will be included in channel functional test and source check.

12. Channel Functional Test - Shall be:

    a. Analog/Digital Channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

    b. Bistable Channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

13. Source Check - Shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source or multiple of sources.

W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

X. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

Y. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

Z. Reportable Event - A reportable event shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

AA. Solidification - Shall be the conversion of radioactive wastes into a form that meets shipping and burial ground requirements.

BB. Offsite Dose Calculation Manual (ODCM) - Shall be a manual describing the environmental monitoring program and the methodology and parameters used in the calculation of release rate limits and offsite doses due to radioactive gaseous and liquid effluents. The ODCM will also provide the plant with guidance for establishing alarm/trip setpoints to ensure technical specifications sections 3.8.A.1 and 3.8.B.1 are not exceeded.

CC. Purge or purging - The controlled process of discharging air or gas from the primary containment to maintain temperature, pressure, humidity, concentration, or other operating condition in such a manner that replacement air or gas is required to purify the containment.

DD. Process Control Program - Shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

EE. Radiological Effluent Manual (REM) - Shall be a manual containing the site and environmental sampling and analysis programs for measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposure to individuals from station operation. It shall also specify operating guidelines for radioactive waste treatment systems and report content.

FF. Venting - The controlled process of discharging air or gas from the primary containment to maintain temperature, pressure, humidity, concentration, or other operating condition in such a manner that replacement air or gas is not provided or required. Vent, used in system names, does not imply a venting process.

NN. Appendix R Safe Shutdown Program

BFN has developed an Appendix R Safe Shutdown Program. This Program is to ensure that the equipment required by the Appendix R Safe Shutdown Analysis is maintained and demonstrated functional as follows:

1. The functional requirements of the Safe Shutdown systems and equipment, as well as appropriate compensatory measures should these systems/components be unable to perform their intended function are outlined in Section III of the Program.

2. Testing and monitoring of the Appendix R Safe Shutdown systems and equipment are defined in Section V of the Program.

3. Deleted

OO. CORE OPERATING LIMITS REPORT (COLR) - The COLR is the unit-specific document that provides the core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9.1.7. Plant operation within these limits is addressed in individual specifications.

PP. Limiting Control Rod Pattern - A limiting control rod pattern shall be a pattern which results in the core being on a thermal limit, i.e. operating on a limiting value for APLHGR, LHGR, or MCPR.

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TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
4(1)	APRM Upscale (Flow Bias)	$\leq$0.58W + 50% (2)
4(1)	APRM Upscale (Startup Mode) (8)	$\leq$12%
4(1)	APRM Downscale (9)	$\geq$3%
4(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Flow Bias)	$\leq$0.66W + 43% (2)(13)
2(7)	RBM Downscale (9)	$\geq$3%
2(7)	RBM Inoperative	(10c)
6(1)	IRM Upscale (8)	$\leq$108/125 of full scale
6(1)	IRM Downscale (3)(8)	$\geq$5/125 of full scale
6(1)	IRM Detector not in Startup Position (8)	(11)
6(1)	IRM Inoperative (8)	(10a)
3(1) (6)	SRM Upscale (8)	$\leq 1X10^5$ counts/sec.
3(1) (6)	SRM Downscale (4)(8)	$\geq$3 counts/sec.
3(1) (6)	SRM Detector not in Startup Position (4)(8)	(11)
3(1) (6)	SRM Inoperative (8)	(10a)
2(1)	Flow Bias Comparator	$\leq$10% difference in recirculation flows
2(1)	Flow Bias Upscale	$\leq$115% recirculation flow
1	Rod Block Logic	N/A
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	$\leq$25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	$\leq$25 gal.

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NOTES FOR TABLE 3.2.C (Cont'd)

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 109 percent rated reactor power.

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LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.b (Cont'd)

3. Should the RWM become inoperable on a shutdown, shutdown may continue provided that a second licensed operator or other technically qualified member of the plant staff is present at the console verifying compliance with the prescribed control rod program.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.b.2 The Rod Worth Minimizer (RWM) shall be demonstrated to be OPERABLE for a reactor shutdown by the following checks:

a. By demonstrating that the control rod patterns and Banked Position Withdrawal Sequence (or equivalent) input to the RWM computer are correctly loaded following any loading of the program into the computer.

b. Within 8 hours prior to RWM automatic initiation when reducing thermal power, verify proper annunciation of the selection error of at least one out-of-sequence control rod.

c. Within one hour after RWM automatic initation when reducing thermal power, the rod block function of the RWM shall be verified by moving an out-of-sequence control rod.

LIMITING CONDITIONS FOR OPERATION

3.3.B. Control Rods

3.c. If Specifications 3.3.B.3.b.1 through 3.3.B.3.b.3 cannot be met the reactor shall not be started, or if the reactor is in the run or startup modes at less than 10% rated power, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the shutdown position.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with CMFCP or CMFLPD equal to or greater than 0.95, either:

   a. Both RBM channels shall be OPERABLE:

   or

   b. Control rod withdrawal shall be blocked.

SURVEILLANCE REQUIREMENTS

4.3.B. Control Rods

3.b.3 When the RWM is not OPERABLE a second licensed operator or other technically qualified member of the plant staff shall verify that the correct rod program is followed.

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. During operation with CMFCP or CMFLPD equal to or greater than 0.95, an instrument functional test of the RBM shall be performed prior to control rod withdrawal and at least once per 24 hours thereafter.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

C. Scram Insertion Times

The control rod system is designated to bring the reactor subcritical at the rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transient is given in Reference 1. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification provide the required protection, and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occurred during plant STARTUP and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model 7RDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e., it can no longer interfere with scram performance, even if completely blocked.

The degraded performance of the original drive (CRD7RDB144A) under dirty operating conditions and the insensitivity of the redesigned drive (CRD7RDB144B) has been demonstrated by a series of engineering tests under simulated reactor operating conditions. The successful performance of the new drive under actual operating conditions has also been demonstrated by consistently good in-service test results for plants using the new drive and may be inferred from plants using the older model drive with a modified (larger screen size) internal filter which is less prone to plugging. Data has been documented by surveillance reports in various operating plants. These include Oyster Creek, Monticello, Dresden 2 and Dresden 3. Approximately 5000 drive tests have been recorded to date.

Following identification of the "plugged filter" problem, very frequent scram tests were necessary to ensure proper performance. However, the more frequent scram tests are now considered totally unnecessary and unwise for the following reasons:

1. Erratic scram performance has been identified as due to an obstructed drive filter in type "A" drives. The drives in BFNP are of the new "B" type design whose scram performance is unaffected by filter condition.

2. The dirt load is primarily released during STARTUP of the reactor when the reactor and its systems are first subjected to flows and pressure and thermal stresses. Special attention and measures are now being taken to assure cleaner systems. Reactors with drives identical or similar (shorter stroke, smaller piston areas) have operated through many refueling cycles with no sudden or erratic changes in scram performance. This preoperational and STARTUP testing is sufficient to detect anomalous drive performance.

3. The 72-hour outage limit which initiated the start of the frequent scram testing is arbitrary, having no logical basis other than quantifying a "major outage" which might reasonably be caused by an event so severe as to possibly affect drive performance. This requirement is unwise because it provides an incentive for shortcut actions to hasten returning "on line" to avoid the additional testing due a 72-hour outage.

LIMITING CONDITIONS FOR OPERATION

3.5.I **Average Planar Linear Heat Generation Rate**

During steady-state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of any fuel assembly at any axial location shall not exceed the appropriate APLHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

J. **Linear Heat Generation Rate (LHGR)**

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and

SURVEILLANCE REQUIREMENTS

4.5.I **Average Planar Linear Heat Generation Rate (APLHGR)**

The APLHGR shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

J. **Linear Heat Generation Rate (LHGR)**

The LHGR shall be checked daily during reactor fuel operation at $\geq$ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

J. Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

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

3.5.K Minimum Critical Power Ratio (MCPR)

Except when the provisions of Note 7 of Table 3.2.C are being employed due to the inoperability of the Rod Block Monitor, the minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR) as provided in the CORE OPERATING LIMITS REPORT. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

J. Linear Heat Generation Rate (LHGR)

4.5.K Minimum Critical Power Ratio (MCPR)

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

2. Except as provided by Note 7 of Table 3.2.C, the MCPR limit at rated flow and rated power shall be determined as provided in the CORE OPERATING LIMITS REPORT using:

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

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

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

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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 217 TO FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 INTRODUCTION

By letter dated October 5, 1992, as supplemented on September 15, 1993, the Tennessee Valley Authority (the licensee) proposed changes to the technical specifications for the Browns Ferry Nuclear Plant (BFN), Unit 2. The proposed changes revise the flow-biased upscale power level Rod Block Monitor (RBM) setpoint, add new definitions for the Core Maximum Fraction of Critical Power and Limiting Control Rod Pattern, revise RBM operability requirements to incorporate the new definitions, and ensure two RBM channels are available if core thermal margin is low. The licensee has requested these changes in order to improve operational flexibility. The licensee's supplement of September 15, 1993 provided clarifications which do not affect the staff's proposed finding of no significant hazards considerations.

2.0 EVALUATION

The Rod Block Monitor provides a signal to the Reactor Manual Control System to inhibit control rod withdrawal if local power range monitor signals exceed a calculated setpoint. Exceeding this setpoint is assumed to indicate local core conditions may be approaching a fuel safety limit. The existing BFN Unit 2 technical specification Limiting Condition for Operation (LCO) Table 3.2.C defines the flow-biased RBM upscale setpoint as:

$$\text{RBM, \% power} \leq 0.66W + 40$$

where W is the reactor recirculation coolant flow rate as a percentage of full flow. This equation is truncated to limit the maximum setpoint to 106% power. The setpoint is based on an analysis of a rod withdrawal error (RWE) from a thermally-limiting control rod pattern. Historically, this analysis included an assumption that one RBM channel is unavailable. This assumption minimizes the sensitivity of the system, and permits the greatest rod withdrawal before a block signal is generated, resulting in the largest decrease in thermal margin.

To improve operational flexibility, the licensee has proposed changes to the BFN Unit 2 technical specifications to increase the flow-biased RBM upscale setpoint. To support this setpoint change, the licensee also proposed new technical specifications which support the RWE analysis used to determine the new setpoint. These changes are discussed below.

The licensee has proposed a new definition 1.U.5, Core Maximum Fraction of Critical Power (CMFCP). This parameter is the maximum value of the flow-corrected critical power ratio (CPR), as defined by technical specifications, divided by the actual CPR for all fuel assemblies in the core. This definition of CMFCP is currently used in BFN procedures and is consistent with standard boiling water reactor (BWR) vocabulary. Application of the definition, as discussed below, provides an appropriate description of reactor thermal margin. Therefore, the proposed definition is acceptable.

The existing BFN technical specifications also include definition 1.U.3, Core Maximum Fraction of Limiting Power Density (CMFLPD). This parameter is defined as the ratio of the maximum fuel rod power density for a given fuel type to the limiting fuel rod power density for that fuel type.

CMFLPD and CMFCP are used to quantify core thermal margin. During normal operations, these values will be less than one, which indicates the core has margin to thermal operating limits. The closer the value of CMFLPD or CMFCP to one, the lower the core thermal margin. CMFLPD and CMFCP are calculated by the plant computer based upon current core thermal-hydraulic and power distribution characteristics, and are available to the plant operators. If the plant computer is unavailable, these parameters can be calculated off-line in accordance with existing plant procedures.

The licensee has performed a revised RWE analysis, assuming that two RBM channels are available if thermal margin, as defined by CMFCP and CMFLPD, is less than 5% (CMFCP or CMFLPD greater than or equal to 0.95). Otherwise, assumptions are consistent with previous RWE analysis. This revised analysis yields a new flow-biased RBM upscale setpoint,

$$\text{RBM, \% power} \leq 0.66W + 43.$$

The proposed revised setpoint is truncated to a maximum of 109% power, as required by the revised Note 13 to Table 3.2.C. The licensee has also proposed changes to LCO 3.3.B.5 to require two operable RBM channels when thermal margin is less than 5%, consistent with the revised RWE analysis assumptions. This RBM operability requirement provides additional RBM sensitivity, compensating for the higher flow-biased RBM upscale setpoint, and ensures thermal limits are not exceeded for RWE transients initiated from low thermal margin conditions.

The licensee's analysis also demonstrates that if thermal margin is greater than 5%, one RBM channel is sufficient to mitigate the consequences of the postulated RWE. Operation of the Reactor Manual Control System to withdraw a control rod requires input from a minimum of one RBM channel. The proposed setpoint change in Table 3.2.C, in combination with revised RBM operability requirements, ensures core thermal limits are not exceeded for a postulated RWE event. Therefore, the proposed changes to LCO Table 3.2.C and LCO 3.3.B.5 are acceptable.

A revision to surveillance requirement (SR) 4.3.B.5 is also proposed. This revision incorporates the CMFCP and CMFLPD thermal margin requirements with existing requirements for RBM instrument functional testing, and is acceptable.

The licensee has also proposed a new definition 1.PP for Limiting Control Rod Pattern, and to delete a discussion of limiting control rod patterns from the technical specification Bases for sections 3.3/4.3. The proposed definition is consistent with standard BWR definitions and is acceptable. Deletion of the Bases discussion is balanced by adding the new definition, and is also acceptable. Associated with this change, a revision to SR 4.5.K.1 is proposed to delete a reference to the Bases. This change is consistent with the addition of the new definition for limiting control rod pattern, and is acceptable.

The original licensee submittal of October 5, 1992 discusses the proposed changes only as they apply to BFN Unit 2 Cycle 6. Supplemental information provided in the September 15, 1993 letter states that the changes are applicable BFN Unit 2 Cycle 7 operation, and that the proposed changes will also be included in analyses of future fuel cycles to ensure their continued applicability.

3.0 SUMMARY

As discussed above, the staff has found that the proposed changes to the Browns Ferry technical specifications meet applicable requirements and provide reasonable assurance that the reactor core will not exceed thermal limits for postulated rod withdrawal error events. Therefore, the proposed changes are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes Surveillance Requirements and Bases. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 55593). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Williams

Date: October 21, 1993

AMENDMENT NO. 217 FOR BROWNS FERRY UNIT 2 - DOCKET NO. 50-260

DATED: October 21, 1993

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