February 24, 1995

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

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 (TAC NOS. M89251, M89252, AND M89253) (TS 339)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment Nos. 216, 232, and 190 to Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, respectively. These amendments are in response to your application dated March 31, 1994 regarding extended load line limit and revised rod block monitor operability requirements, deletion of specific values for rated reactor recirculation flow rate, relocation of rod block equations to the core operating limits report, and other miscellaneous changes.

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

Sincerely,

Original Signed By:

Joseph F. Williams, Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 216 to License No. DPR-33
2. Amendment No. 232 to License No. DPR-52
3. Amendment No. 190 to License No. DPR-68
4. Safety Evaluation

cc w/enclosures: See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 216
License No. DPR-33

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 31, 1994, 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. 216, 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: February 24, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 216

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
vii	vii*
viii	viii
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*
1.1/2.1-1	1.1/2.1-1*
1.1/2.1-2	1.1/2.1-2
1.1/2.1-3	1.1/2.1-3
1.1/2.1-4	1.1/2.1-4*
1.1/2.1-6	1.1/2.1-6
1.1/2.1-7	1.1/2.1-7
1.1/2.1-12	1.1/2.1-12
1.1/2.1-13	1.1/2.1-13*
1.1/2.1-14	1.1/2.1-14
1.1/2.1-15	1.1/2.1-15
1.1/2.1-16	1.1/2.1-16
1.1/2.1-17	1.1/2.1-17*
3.2/4.2-25	3.2/4.2-25
3.2/4.2-25a	3.2/4.2-25a*
3.2/4.2-26	3.2/4.2-26
3.2/4.2-27	3.2/4.2-27
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
3.5/4.5-20	3.5/4.5-20
3.5/4.5-21	3.5/4.5-21*
6.0-26	6.0-26*
6.0-26a	6.0-26a
6.0-26b	6.0-26b
6.0-26c	6.0-26c*

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

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. (Deleted)

1.0 DEFINITIONS (Cont'd)

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. (Deleted)

BB. Offsite Dose Calculation Manual (ODCM) - Shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.5 and 6.9.1.8.

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 current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61 and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

EE. (Deleted)

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.

1.0 DEFINITIONS (Cont'd)

NN. 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.

OO. 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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SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

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

Specifications

A. Thermal Power Limits

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

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

2.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

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

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

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

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

$S \leq (0.58W + 62\%)$

where:

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

W = Loop recirculation flow rate in percent of rated

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

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

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

c. The APRM Rod Block trip setting shall be less than or equal to the limit specified in the CORE OPERATING LIMITS REPORT.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1.A Thermal Power Limits	2.1.A Neutron Flux Trip Settings
	d. Fixed High Neutron Flux Scram Trip Setting--When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be: $S \leq 120\%$ power.
2. Reactor Pressure ≤ 800 psia or Core Flow $\leq 10\%$ of rated. When the reactor pressure is ≤ 800 psia or core flow is $\leq 10\%$ of rated, the core thermal power shall not exceed 823 MWt (.25% of rated thermal power).	2. APRM and IRM Trip Settings (Startup and Hot Standby Modes). a. APRM--When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power. b. IRM--The IRM scram shall be set at less than or equal to 120/125 of full scale.

Figure 2.1-1

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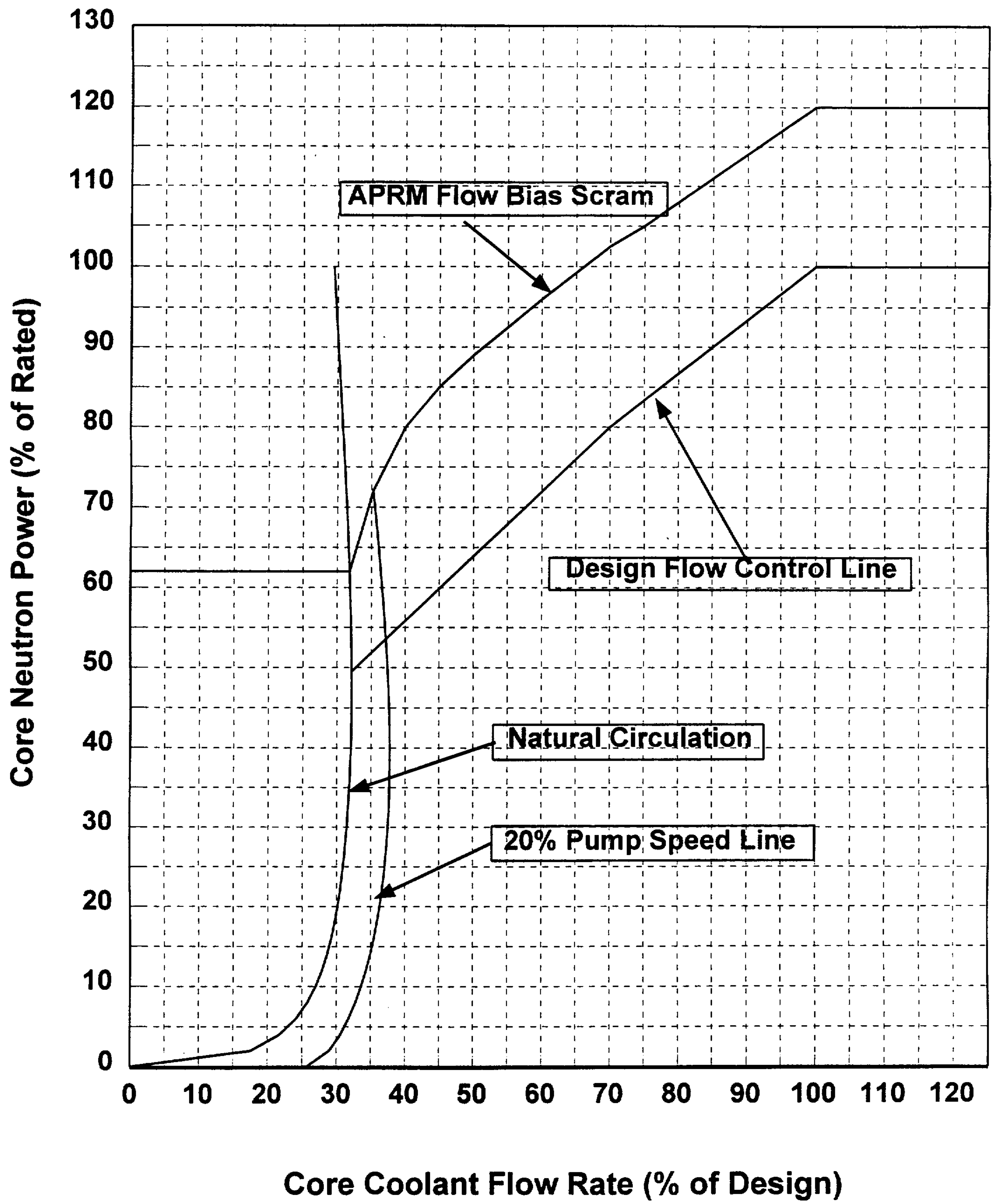

APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

2.1 BASES (Cont'd)

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

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

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

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

2.1 BASES (Cont'd)

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

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

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

3. IRM Flux Scram Trip Setting

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

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. In addition, the APRM 15 percent scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux. An IRM scram would result in a reactor shutdown well before any SAFETY LIMIT is exceeded. For the case of a single control rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in paragraph 7.5.5 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel SAFETY LIMIT and there is a substantial margin from fuel damage.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire power/flow domain including above the rated rod line (Reference 1). The margin to the SAFETY LIMIT increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at the maximum thermal power level permitted by the APRM rod block trip setting, which is found in the CORE OPERATING LIMITS REPORT. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system.

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

F. (Deleted)

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 825 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. The scram feature that occurs when the main steam line isolation valves close shuts down the reactor so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity SAFETY LIMIT. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the startup position, where protection of the fuel cladding integrity SAFETY LIMIT is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity SAFETY LIMIT. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

I.J.& K. Reactor Low Water Level Setpoint for Initiation of HPCI and RCIC Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps.

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. References

1. Supplemental Reload Licensing Report of Browns Ferry Nuclear Plant, Unit 1 (applicable cycle-specific document).

2. GE Standard Application for Reactor Fuel, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved version).

3. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactor," NEDO-24154-P, October 1978.

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

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AMENDMENT NO. 197

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)	(2)
4(1)	APRM Upscale (Startup Mode) (8)	≤12%
4(1)	APRM Downscale (9)	≥3%
4(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Flow Bias)	(13)
2(7)	RBM Downscale (9)	≥3%
2(7)	RBM Inoperative	(10c)
6(1)	IRM Upscale (8)	≤108/125 of full scale
6(1)	IRM Downscale (3)(8)	≥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)	≤ $1X10^5$ counts/sec.
3(1) (6)	SRM Downscale (4)(8)	≥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	≤10% difference in recirculation flows
2(1)	Flow Bias Upscale	≤115% recirculation flow
1	Rod Block Logic	N/A
1(12)	High Water Level in West Scram Discharge Tank (LS-85-45L)	≤25 gal.
1(12)	High Water Level in East Scram Discharge Tank (LS-85-45M)	≤25 gal.

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NOTES FOR TABLE 3.2.C

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

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

2. The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

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

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

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

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

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

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

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

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

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

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

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

NOTES FOR TABLE 3.2.C (Cont'd)

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

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

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

    a. SRM and IRM

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

        (2) Power supply voltage low.

        (3) Circuit boards not in circuit.

    b. APRM

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

        (2) Less than 14 LPRM inputs.

        (3) Circuit boards not in circuit.

    c. RBM

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

        (2) Circuit boards not in circuit.

        (3) RBM fails to null.

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

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

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

13. The trip level setting and clipped value for this setting shall be as specified in the CORE OPERATING LIMITS REPORT.

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 withdrawal of the designated rod(s) 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 steady state 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.

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 operation at $\geq$ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

3.5.J Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

If at any time during steady-state 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 corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

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

SURVEILLANCE REQUIREMENTS

4.5.J Linear Heat Generation Rate (LHGR)

4.5.K Minimum Critical Power Ratio (MCPR)

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

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

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

LIMITING CONDITIONS FOR OPERATION

3.5.K Minimum Critical Power Ratio (MCPR)

L. APRM Setpoints

1. Whenever the core thermal power is ≥ 25% of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0, or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.

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

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

SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

4.5.K.2 (Cont'd)

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.

L. APRM Setpoints

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

LIMITING CONDITIONS FOR OPERATION

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.

SURVEILLANCE REQUIREMENTS

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.

inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on measurements obtained with self reading dosimeter, TLD, or film badge. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent exposure received from external sources shall be assigned to specific major work functions.

b. Any mainsteam relief valve that opens in response to reaching its setpoint or due to operator action to control reactor pressure shall be reported.

6.9.1.3 MONTHLY OPERATING REPORT

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, to be submitted no later than the fifteenth of each month following the calendar month covered by the report. A narrative summary of operating experience shall be submitted in the above schedule.

6.9.1.4 REPORTABLE EVENTS

Reportable events, including corrective actions and measures to prevent re-occurrence, shall be reported to the NRC in accordance with Section 50.73 to 10 CFR 50.

6.9.1.5 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. A single submittal may be made for a multi-unit station. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

6.9.1.6 SOURCE TESTS

Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

6.9.1.7 CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established and shall be documented in the CORE OPERATING LIMITS REPORT prior to each operating cycle, or prior to any remaining portion of an operating cycle, for the following:

   (1) The APLHGR for Specification 3.5.I

   (2) The LHGR for Specification 3.5.J

   (3) The MCPR Operating Limit for Specification 3.5.K/4.5.K

   (4) The APRM Flow Biased Rod Block Trip Setting for Specification 2.1.A.1.c, Table 3.2.C, and Specification 3.5.L

(5) The RBM Upscale (Flow Bias) Trip Setting and clipped value for this setting for Table 3.2.C

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin limits, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.9.1.8 THE ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted by April 1, of each year. The report shall include summaries of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

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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. 232
License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 31, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 232, 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: February 24, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 232

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
vii	vii*
viii	viii
1.1/2.1-1	1.1/2.1-1*
1.1/2.1-2	1.1/2.1-2
1.1/2.1-3	1.1/2.1-3
1.1/2.1-4	1.1/2.1-4*
1.1/2.1-6	1.1/2.1-6
1.1/2.1-6a	---------
1.1/2.1-7	1.1/2.1-7*
1.1/2.1-7a	---------
1.1/2.1-14	1.1/2.1-14*
1.1/2.1-15	1.1/2.1-15
3.2/4.2-25	3.2/4.2-25
3.2/4.2-25a	3.2/4.2-25a*
3.2/4.2-26	3.2/4.2-26
3.2/4.2-27	3.2/4.2-27
3.2/4.2-27a	----------
3.2/4.2-27b	----------
3.5/4.5-20	3.5/4.5-20
3.5/4.5-20a	3.5/4.5-20a*
6.0-26	6.0-26*
6.0-26a	6.0-26a
6.0-26b	6.0-26b**
6.0-26c	6.0-26c*

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

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

Specifications

A. Thermal Power Limits

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

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

2.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

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

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

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

$S \leq (0.58W + 62\%)$

where:

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

W = Loop recirculation flow rate in percent of rated

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

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

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

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

c. The APRM Rod Block trip setting shall be less than or equal to the limit specified in the CORE OPERATING LIMITS REPORT.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
1.1.A Thermal Power Limits	2.1.A Neutron Flux Trip Settings (Cont'd)
	d. Fixed High Neutron Flux Scram Trip Setting—When the mode switch is in the RUN position, the APRM fixed high flux scram trip setting shall be: S≤120% power.
2. Reactor Pressure ≤800 psia or Core Flow ≤10% of rated. When the reactor pressure is ≤800 psia or core flow is ≤10% of rated, the core thermal power shall not exceed 823 MWt (25% of rated thermal power).	2. APRM and IRM Trip Settings (Startup and Hot Standby Modes). a. APRM—When the reactor mode switch is in the STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power. b. IRM—The IRM scram shall be set at less than or equal to 120/125 of full scale.

Figure 2.1-1

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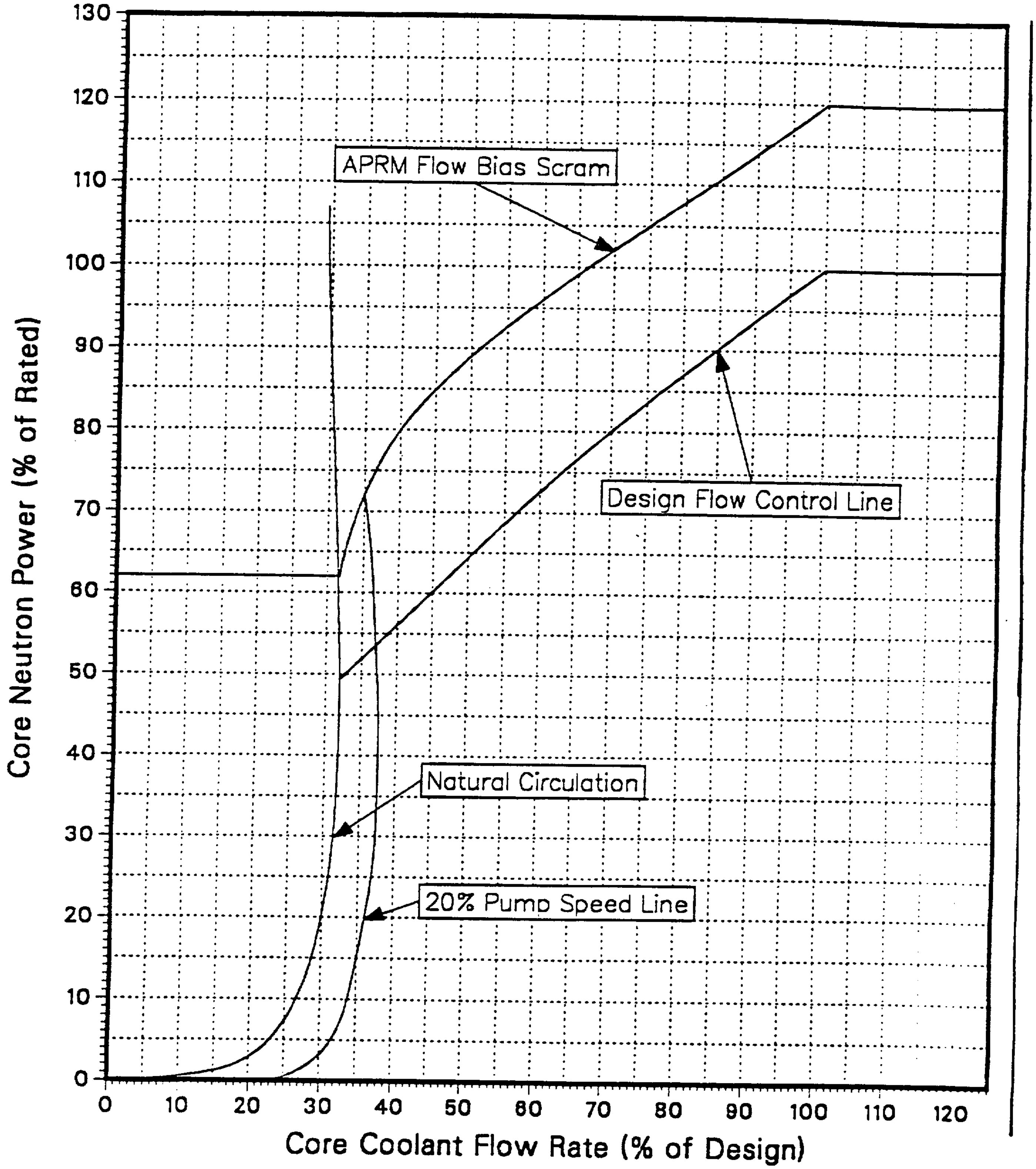

APRM Flow Bias Scram vs. Reactor Core Flow
Fig. 2.1-2

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. In addition, the APRM 15 percent scram prevents higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods that heat flux is in equilibrium with the neutron flux. An IRM scram would result in a reactor shutdown well before any safety limit is exceeded. For the case of a single control rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in paragraph 7.5.5 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel safety limit and there is a substantial margin from fuel damage.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting over the entire power/flow domain, including above the rated rod line (Reference 1). The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at the maximum thermal power level permitted by the APRM rod block trip setting, which is found in the CORE OPERATING LIMITS REPORT. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system.

C. Reactor Water Low Level Scram and Isolation (Except Main Steam lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

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)	(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)	(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

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

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

2. The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

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

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

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

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

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

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

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

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

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

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

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

NOTES FOR TABLE 3.2.C (Cont'd)

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

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

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

    a. SRM and IRM

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

        (2) Power supply voltage low.

        (3) Circuit boards not in circuit.

    b. APRM

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

        (2) Less than 14 LPRM inputs.

        (3) Circuit boards not in circuit.

    c. RBM

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

        (2) Circuit boards not in circuit.

        (3) RBM fails to null.

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

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

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

13. The trip level setting and clipped value for this setting shall be as specified in the CORE OPERATING LIMITS REPORT.

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 ≥ 25% of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0, or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.	FRP/CMFLPD shall be determined daily when the reactor is ≥ 25% of rated thermal power.
2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.	
3. If 3.5.L.1 and 3.5.L.2 cannot be met, the reactor power shall be reduced to ≤ 25% of rated thermal power within 4 hours.	
M. Core Thermal-Hydraulic Stability	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.	1. Verify that the reactor is outside of Region 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.	a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and
3. If Region II of Figure 3.5.M-1 is entered:	b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

LIMITING CONDITIONS FOR OPERATION

3.5 Core and Containment Cooling Systems

3.5.M.3. (Cont'd)

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.

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems total deep dose equivalent exposure received from external sources shall be assigned to specific major work functions.

b. Any mainsteam relief valve that opens in response to reaching its setpoint or due to operator action to control reactor pressure shall be reported.

6.9.1.3 MONTHLY OPERATING REPORT

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, to be submitted no later than the fifteenth of each month following the calendar month covered by the report. A narrative summary of operating experience shall be submitted in the above schedule.

6.9.1.4 REPORTABLE EVENTS

Reportable events, including corrective actions and measures to prevent re-occurrence, shall be reported to the NRC in accordance with Section 50.73 to 10 CFR 50.

6.9.1.5 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. A single submittal may be made for a multi-unit station. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

6.9.1.6 SOURCE TESTS

Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

6.9.1.7 CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established and shall be documented in the CORE OPERATING LIMITS REPORT prior to each operating cycle, or prior to any remaining portion of an operating cycle, for the following:

    (1) The APLHGR for Specification 3.5.I

    (2) The LHGR for Specification 3.5.J

    (3) The MCPR Operating Limit for Specification 3.5.K/4.5.K

    (4) The APRM Flow Biased Rod Block Trip Setting for Specification 2.1.A.1.c, Table 3.2.C, and Specification 3.5.L

    (5) The RBM Upscale (Flow Bias) Trip Setting and clipped value for this setting for Table 3.2.C

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

6.9.1.7 CORE OPERATING LIMITS REPORT (Continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin limits, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.9.1.8 THE ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted by April 1, of each year. The report shall include summaries of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated March 31, 1994, 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. 190, 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: February 24, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 190

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

REMOVE	INSERT
vii	vii*
viii	viii
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*
1.1/2.1-1	1.1/2.1-1*
1.1/2.1-2	1.1/2.1-2
1.1/2.1-3	1.1/2.1-3
1.1/2.1-4	1.1/2.1-4*
1.1/2.1-6	1.1/2.1-6
1.1/2.1-7	1.1/2.1-7
1.1/2.1-12	1.1/2.1-12
1.1/2.1-13	1.1/2.1-13*
1.1/2.1-14	1.1/2.1-14
1.1/2.1-15	1.1/2.1-15
1.1/2.1-16	1.1/2.1-16
1.1/2.1-17	1.1/2.1-17*
3.2/4.2-24	3.2/4.2-24
3.2/4.2-24a	3.2/4.2-24a*
3.2/4.2-25	3.2/4.2-25
3.2/4.2-26	3.2/4.2-26
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
3.5/4.5-20	3.5/4.5-20
3.5/4.5-20a	3.5/4.5-20a*
6.0-26	6.0-26*
6.0-26a	6.0-26a

LIST OF TABLES (Cont'd)

LIST OF ILLUSTRATIONS

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.

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. (Deleted)

W. Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or components 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. (Deleted)

BB. Offsite Dose Calculation Manual (ODCM) - Shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.5 and 6.9.1.8.

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 current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61 and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

EE. (Deleted)

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. 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.

OO. 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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SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

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

Specification

A. Thermal Power Limits

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

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

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

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

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

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

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

$S \leq (0.58W + 62\%)$

where:

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

W = Loop recirculation flow rate in percent of rated

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

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

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

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

c. The APRM Rod Block trip setting shall be less than or equal to the limit specified in the CORE OPERATING LIMITS REPORT.

SAFETY LIMIT

1.1.A Thermal Power Limits

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

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

LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

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

$S \leq 120\%$ power.

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

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

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

Figure 2.1-1

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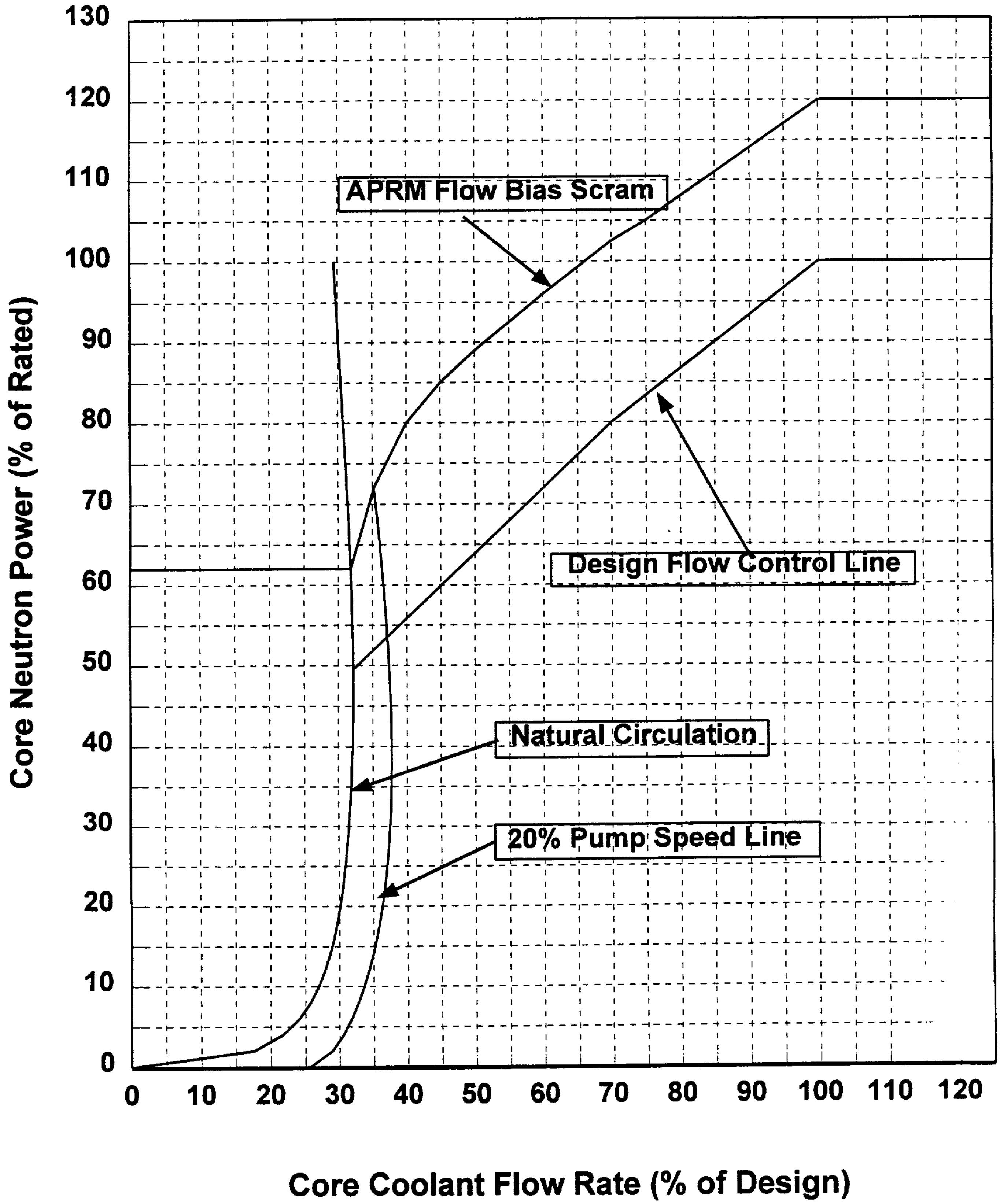

APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

2.1 BASES (Cont'd)

The bases for individual setpoints are discussed below:

A. Neutron Flux Scram

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

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

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

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

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

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

3. IRM Flux Scram Trip Setting

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

***See Section 3.5.K

IRM Flux Scram Trip Setting (Continued)

Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. A scram at 120 divisions on the IRM instruments remains in effect as long as the reactor is in the startup mode. The APRM 15 percent scram will prevent higher power operation without being in the RUN mode. The IRM scram provides protection for changes which occur both locally and over the entire core. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough, due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux. An IRM scram would result in a reactor shutdown well before any SAFETY LIMIT is exceeded. For the case of a single control rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in paragraph 7.5.5.4 of the FSAR. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

4. Fixed High Neutron Flux Scram Trip

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated power (3,293 MWt). The APRM system responds directly to neutron flux. Licensing analyses have demonstrated that with a neutron flux scram of 120 percent of rated power, none of the abnormal operational transients analyzed violate the fuel SAFETY LIMIT and there is a substantial margin from fuel damage.

B. APRM Control Rod Block

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.07. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excess values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire power/flow domain including above the rated rod line (Reference 1). The margin to the SAFETY LIMIT increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at the maximum thermal power level permitted by the APRM rod block trip setting, which is found in the CORE OPERATING LIMITS REPORT. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system.

C. Reactor Water Low Level Scram and Isolation (Except Main Steam Lines)

The setpoint for the low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR subsection 14.5 show that scram and isolation of all process lines (except main steam) at this level adequately protects the fuel and the pressure barrier, because MCPR is greater than 1.07 in all cases, and system pressure does not reach the safety valve settings. The scram setting is sufficiently below normal operating range to avoid spurious scrams.

D. Turbine Stop Valve Closure Scram

The turbine stop valve closure trip anticipates the pressure, neutron flux and heat flux increases that would result from closure of the stop valves. With a trip setting of 10 percent of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained even during the worst case transient that assumes the turbine bypass valves remain closed. (Reference 2)

E. Turbine Control Valve Fast Closure or Turbine Trip Scram

Turbine control valve fast closure or turbine trip scram anticipates the pressure, neutron flux, and heat flux increase that could result from control valve fast closure due to load rejection or control valve closure due to turbine trip; each without bypass valve capability. The reactor protection system initiates a scram in less than 30 milliseconds after the start of control valve fast closure due to load rejection or control valve closure due to turbine trip. This scram is achieved by rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50 percent greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar to that for the stop valve. No significant change in MCPR occurs. Relevant transient analyses are discussed in References 2 and 3 of the Final Safety Analysis Report. This scram is bypassed when turbine steam flow is below 30 percent of rated, as measured by turbine first state pressure.

2.1 BASES (Cont'd)

F. (Deleted)

G. & H. Main Steam Line Isolation on Low Pressure and Main Steam Line Isolation Scram

The low pressure isolation of the main steam lines at 850 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel. The scram feature that occurs when the main steam line isolation valves close shuts down the reactor so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity SAFETY LIMIT. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position, where protection of the fuel cladding integrity SAFETY LIMIT is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity SAFETY LIMIT. In addition, the isolation valve closure scram anticipates the pressure and flux transients that occur during normal or inadvertent isolation valve closure. With the scrams set at 10 percent of valve closure, neutron flux does not increase.

I.J.& K. Reactor Low Water Level Setpoint for Initiation of HPCI and RCIC Closing Main Steam Isolation Valves, and Starting LPCI and Core Spray Pumps.

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram setpoint and initiation setpoints. Transient analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. References

1. Supplemental Reload Licensing Report of Browns Ferry Nuclear Plant, Unit 3 (applicable cycle-specific document).

2. GE Standard Application for Reactor Fuel, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved version).

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AMENDMENT NO. 170

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)	(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)	(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

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

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

2. The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

    a. SRM and IRM

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

        (2) Power supply voltage low.

        (3) Circuit boards not in circuit.

    b. APRM

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

        (2) Less than 14 LPRM inputs.

        (3) Circuit boards not in circuit.

    c. RBM

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

        (2) Circuit boards not in circuit.

        (3) RBM fails to null.

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

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

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

13. The trip level setting and clipped value for this setting shall be as specified in the CORE OPERATING LIMITS REPORT.

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 withdrawal of the designated rod(s) 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 designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure N3.6-9) 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 corresponding action shall continue until reactor operation is within the prescribed limits.

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 operation at $\geq$ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

3.5.K Minimum Critical Power Ratio (MCPR)

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

SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

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

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

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

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

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

LIMITING CONDITIONS FOR OPERATION

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

1. Whenever the core thermal power is ≥ 25% of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0, or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.

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

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

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

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

LIMITING CONDITIONS FOR OPERATION

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.

SURVEILLANCE REQUIREMENTS

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.

may be estimates based on measurements obtained with self reading dosimeter, TLD, or film badge. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent exposure received from external sources shall be assigned to specific major work functions.

b. Any mainsteam relief valve that opens in response to reaching its setpoint or due to operator action to control reactor pressure shall be reported.

6.9.1.3 MONTHLY OPERATING REPORT

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, to be submitted no later than the fifteenth of each month following the calendar month covered by the report. A narrative summary of operating experience shall be submitted in the above schedule.

6.9.1.4 REPORTABLE EVENTS

Reportable events, including corrective actions and measures to prevent re-occurrence, shall be reported to the NRC in accordance with Section 50.73 to 10 CFR 50.

6.9.1.5 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. A single submittal may be made for a multi-unit station. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlines in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

6.9.1.6 SOURCE TESTS

Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

6.9.1.7 CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established and shall be documented in the CORE OPERATING LIMITS REPORT prior to each operating cycle, or prior to any remaining portion of an operating cycle, for the following:

   (1) The APLHGR for Specification 3.5.I

   (2) The LHGR for Specification 3.5.J

   (3) The MCPR Operating Limit for Specification 3.5.K/4.5.K

   (4) The APRM Flow Biased Rod Block Trip Setting for Specification 2.1.A.1.c, Table 3.2.C, and Specification 3.5.L

   (5) The RBM Upscale (Flow Bias) Trip Setting and clipped value for this setting for Table 3.2.C

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 232 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

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

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

1.0 INTRODUCTION

By letter dated March 31, 1994, the Tennessee Valley Authority (the licensee) requested amendments to the operating licenses for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3. These amendments consist of revisions to the BFN Unit 1, 2, and 3 Technical Specifications (TS). The licensee proposed to revise the BFN Unit 1 and 3 TS to incorporate changes to the extended load line limit (ELLL) and rod block monitor (RBM) operability requirements which were previously approved for BFN Unit 2. The licensee also proposed to delete a specific value for reactor recirculation flow rate, and to relocate rod block equations to the core operating limits report (COLR) for all three BFN units. The NRC staff evaluation of these revisions is given below.

2.0 EVALUATION

In its submittal of March 31, 1994, the licensee separated the components of its request into five sections, designated "A" through "E." This evaluation addresses each of these items in turn.

2.1 Part A: Extended Load Line Limit (BFN Units 1 and 3)

For BFN Units 1 and 3, the licensee has proposed to revise equations used for the flow-biased average power range monitor (APRM) flux reactor scram trip setpoint and the APRM rod block trip setting. Using the revised equations extends the allowable reactor operating envelope into the extended load line limit (ELLL) region. Similar changes have been previously approved for BFN Unit 2 on December 18, 1990. The changes proposed to support ELLL operation are given below.

To support ELLL operation, Limiting Safety System Setting (LSSS) 2.1.A.1.a will be revised from:

$$S \leq 0.66W + 54\% \quad \text{to:} \quad S \leq 0.58W + 62\%$$

where: S = flow biased APRM Flux Scram Trip setting, and
W = reactor core flow rate, % of rated.

ENCLOSURE 4

Figure 2.1-2, "APRM Flow Bias Scram Vs. Reactor Core Flow," will also be revised consistent with the change in the LSSS equation.

The licensee also proposes editorial changes consistent with the revised TS.

ELLL operation has been evaluated using the General Electric Standard Application for Reactor Fuel, or GESTAR-II, which is an NRC-approved methodology. GESTAR-II sets forth the requirements to demonstrate acceptable reactor operation within the ELLL. To support ELLL operation for BFN Unit 2, an ELLL analysis was performed which demonstrated design limits would not be exceeded for limiting anticipated operational occurences and loss of coolant accidents. The licensee has subsequently determined that this analysis is also applicable to BFN Units 1 and 3. The licensee states that BFN Units 1 and 3 reload analyses will be performed using the GESTAR-II methodology.

The proposed changes provide adequate assurance that reactor fuel design limits will be preserved for the appropriate range of accident conditions, consistent with methods accepted by the NRC staff. Therefore, the changes are acceptable.

2.2 Part B: Rod Block Monitor Operability Requirements (BFN Units 1 and 3)

The licensee proposes to revise rod block monitor (RBM) operability requirements to ensure two RBM channels are operable for low thermal margin conditions, and one RBM channel is operable for higher thermal margin conditions. This revision provides additional margin which permit implementation of a higher flow-biased RBM setpoint, improving operational flexibility. As discussed in Section 2.5 below, the flow-biased RBM setpoint will be located in the COLR. The licensee also proposes new and revised definitions for terms used to quantify thermal margin. These changes are similar to changes previously approved for BFN Unit 2 on October 21, 1993. The changes proposed by the licensee 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 Unit I and 3 TS 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 also proposes to add a new definition 1.00 describing a limiting control rod pattern. A limiting control rod pattern is an arrangement of control rods which results in the core operating at a thermal limit, such as for minimum CPR or linear heat generation rate. The proposed definition is consistent with standard BWR usage and is acceptable.

The licensee proposes revisions to TS 3.3.B.5 and 4.3.B.5 which apply these requirements for low thermal margin conditions. These revisions ensure thermal limits are not exceeded for limiting rod withdrawal events initiated from low thermal margin conditions. Therefore, the proposed changes are acceptable.

An additional change is proposed for TS 4.5.K.1 regarding surveillance requirements for minimum critical power ratio (MCPR). This change reflects the new definition of limiting control rod pattern, and is acceptable.

2.3 Part C: Miscellaneous Editorial Changes (BFN Units 1, 2, and 3)

The licensee proposes several miscellaneous editorial changes to the TS and the Bases. The submittal of March 31, 1994 describes a total of twelve items. Items 1 through 4, 7, and 9 through 11 pertain to revised Bases discussion. Items 5, 6, 8, and 12 affect TS requirements.

Item 1 adds wording to Bases 2.1.A.1 that clarifies the usage of the APRM flow-biased high flux scram trip setting for the "RUN" mode for power increase transients. The change appropriately describes this function and is acceptable.

Item 2 is an editorial change to add the word "scram" to Bases 2.1.A.1. This change corrects the previous wording, and is acceptable.

Item 3 is an editorial change to revise wording and punctuation in Bases 2.1.A.3. The meaning of this discussion is unchanged. Therefore, the revision is acceptable.

Item 4 revises Bases 2.1.G & H to rephrase the discussion of the scram on main steam isolation valve closure. The change appropriately describes this function and is acceptable.

Item 5 revises Note 7.a to Table 3.2.C to change the word "and" to "or" in the BFN Unit 1 and 3 TS. This change is similar to a change previously approved for BFN Unit 2 on July 2, 1992, and clarifies the operation of the RBM system. This change is acceptable.

Item 6 pertains to deletion of notes in the BFN Unit 2 TS which were relevant only during a previous operating cycle. These notes were previously deleted by an amendment dated December 7, 1994.

Item 7 proposed changes to the BFN Unit 2 TS Bases. These changes were incorporated into Amendment 229, which was issued on December 7, 1994.

Items 8 through 12 all refer to changing words in the TS and Bases from lower case to upper case, or vice versa, as appropriate to reflect the usage of upper case to designate terms defined in TS section 1, "Definitions." The changes appropriately reflect this standard usage, and is acceptable.

2.4 Part D: Deletion of Specific Value for Rated Loop Recirculation Flow Rate (BFN Units 1, 2, and 3)

The licensee proposes to delete a specific value for the rated loop recirculation flow rate from TS 2.1.A.1.a. This TS provides the APRM flow-biased flux scram trip setting, which is based on this rated flow value. This flow rate is the amount of recirculation system drive flow required to achieve 100% total core flow. The value given in the current TS was accurate early in the operating life of the facilities. However, as components have aged, the amount of loop flow required to yield 100% core flow has increased. The licensee believes that retaining an obsolescent value in the TS is unnecessary.

In response to questions from the reviewer, the licensee demonstrated why using the actual loop flow value measured each fuel cycle is more conservative than using the value specified in the TS. For BFN Unit 2, the APRM flow-biased scram setpoint is calculated by the equation:

$$S \leq 0.58W + 62\%$$

where: S = flow-biased APRM Flux Scram Trip setting, and
W = reactor core flow rate, % of rated.

Note that Section 2.1 above discusses implementing this same setpoint equation for BFN Units 1 and 3. Assuming a loop flow rate is 20×10^6 lbs/hr and the loop flow rate to yield 100% core flow of 34.2×10^6 lbs/hr (as given in the current TS), the flow-biased APRM flux scram trip setpoint would be less than or equal to 95.92% power. If one uses a loop flow rate for 100% core flow of 36.36×10^6 lbs/hr (as measured for BFN Unit 2 Cycle 8), the setpoint would be 93.90% power. Since a lower setpoint is more restrictive, using the cycle-specific flow rate is conservative. Therefore, the proposed change is acceptable.

2.5 Part E: Relocate APRM Rod Block and RBM Setpoint Equations to COLR (BFN Units 1, 2, and 3)

The licensee proposes to relocate the APRM rod block and RBM setpoint equations to the Core Operating Limits Report. The COLR was incorporated in the BFN Unit 1, 2, and 3 TS consistent with the guidance of Generic Letter (GL) 88-16 in amendments dated May 20, 1993.

To incorporate fuel cycle-specific parameters into the COLR, GL 88-16 requires that the parameters be established using an NRC-approved methodology consistent with all applicable limits of the plant safety analysis as described by the Final Safety Analysis Report (FSAR) for the facility. References to the cycle-specific parameters in the TS are modified to refer to the COLR, which is submitted to the NRC for each fuel cycle.

To transfer these equations from the TS to the COLR, several TS changes are required. The changes proposed by the licensee reflect deletion of the equations from the TS, and provide proper reference to the COLR as the source of the equations.

The licensee states that the APRM rod block and RBM setpoint equations are derived from fuel cycle-specific calculations based on NRC-approved methodology which preserves appropriate design limits. These equations are already part of the COLR for the current BFN Unit 2 operating cycle (see TVA letter dated October 31, 1994), and will be included in the COLR for future operating cycles of BFN Units 1 and 3. Transferring these equations from the TS to the COLR is consistent with the guidance of GL 88-16, and is acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 49437). The amendments also change recordkeeping or reporting requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

Principal Contributor: Joseph F. Williams

Dated: February 24, 1995