December 28, 1990

Docket No. 50-260

Mr. Oliver D. Kingsley, Jr.
Senior Vice President, Nuclear Power
Tennessee Valley Authority
6N 38A Lookout Place
1101 Market Street
Chattanooga, Tennessee 3702-2801

Dear Mr. Kingsley:

SUBJECT: REDISTRIBUTION OF TECHNICAL SPECIFICATION PAGES FOR AMENDMENT NOS. 180 AND 181 - BROWNS FERRY NUCLEAR PLANT, UNIT 2 (TAC NOS.76902 AND 76934)

On December 10 and 18, 1990, the NRC issued Amendment Nos. 180 and 181, respectively, to Facility Operating License DPR-52 for the Browns Ferry Nuclear Plant, Unit 2. After distribution was made, we discovered Technical Specification (TS) Page 3.2/4.2-61 was inadvertently omitted from the Amendment 180 TS copies. In addition, the incorrect Amendment No. appeared on the overleaf pages for Amendments 180 and 181.

The corrected TS pages in their entirety are being redistributed.

We apologize for any inconvenience this may have caused.

Sincerely,

Original signed by Brenda Mozafari for

Thierry M. Ross, Project Manager
Project Directorate II-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
As Stated

cc w/enclosures:
See next page

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ATTACHMENT TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

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

REMOVE	INSERT
3.2/4.2-16	3.2/4.2-16
3.2/4.2-17	3.2/4.2-17*
3.2/4.2-40	3.2/4.2-40
3.2/4.2-41	3.2/4.2-41*
3.2/4.2-44	3.2/4.2-44*
3.2/4.2-45	3.2/4.2-45
3.2/4.2-61	3.2/4.2-61
3.2/4.2-61a	3.2/4.2-61a

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TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
2	Instrument Channel – Reactor Low Pressure (PIS-3-74 A & B) (PIS-68-95, 96)	450 psig ± 15	A	1. Below trip setting permissive for opening CSS and LPCI admission valves.
2	Instrument Channel – Reactor Low Pressure (PS-3-74 A & B) (PS-68-95, 96)	230 psig ± 15	A	1. Recirculation discharge valve actuation.
2	Core Spray Auto Sequencing Timers (5)	$6 \le t \le 8$ sec.	B	1. With diesel power 2. One per motor
2	LPCI Auto Sequencing Timers (5)	$0 \le t \le 1$ sec.	B	1. With diesel power 2. One per motor
1	RHRSW A1, B3, C1, and D3 Timers	$13 \le t \le 15$ sec.	A	1. With diesel power 2. One per pump
2	Core Spray and LPCI Auto Sequencing Timers (6)	$0 \le t \le 1$ sec. $6 \le t \le 8$ sec. $12 \le t \le 16$ sec. $18 \le t \le 24$ sec.	B	1. With normal power 2. One per CSS motor 3. Two per RHR motor
1	RHRSW A1, B3, C1, and D3 Timers	$27 \le t \le 29$ sec.	A	1. With normal power 2. One per pump

TABLE 3.2.B (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Trip Level Setting	Action	Remarks
1(16)	ADS Timer	105 sec ± 7	A	1. Above trip setting in conjunction with low reactor water level permissive, low reactor water level, high drywell pressure or high drywell pressure bypass timer timed out, and RHR or CSS pumps running, initiates ADS.
1(16)	ADS Timer (12 1/2 min.) (High Drywell Pressure Bypass Timer)	12 1/2 min. ± 2	A	1. Above trip setting, in conjunction with low reactor water level permissive, low reactor water level, 105 sec. delay timer, and RHR or CSS pumps running, initiates ADS.
2	Instrument Channel - RHR Discharge Pressure	100 ±10 psig	A	1. Below trip setting defers ADS actuation.
2	Instrument Channel CSS Pump Discharge Pressure	185 ±10 psig	A	1. Below trip setting defers ADS actuation.
1(3)	Core Spray Sparger to Reactor Pressure Vessel d/p	2 psid ±0.4	A	1. Alarm to detect core sparger pipe break.
	RHR (LPCI) Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems and valves.

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

Function	Functional Test	Calibration Frequency	Instrument Check
Instrument Channel – Reactor Low Water Level (LIS-3-203A-D)	(1) (27)	Once/18 Months (28)	Once/day
Instrument Channel – Reactor High Pressure (PS-68-93 & 94)	(31)	Once/18 months	None
Instrument Channel – Reactor Low Water Level (LIS-3-56A-D)	(1) (27)	Once/18 months (28)	Once/day
Instrument Channel – High Drywell Pressure (PIS-64-56A-D)	(1) (27)	Once/18 Months (28)	N/A
Instrument Channel – High Radiation Main Steam Line Tunnel	29	(5)	Once/day
Instrument Channel – Low Pressure Main Steam Line (PIS-1-72, 76, 82, 86)	(29) (27)	Once/18 Months (28)	None
Instrument Channel – High Flow Main Steam Line (PdIS-1-13A-D, 25A-D, 36A-D, 50A-D)	(29) (27)	Once/18 Months (28)	Once/day

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

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

TABLE 4.2.B
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration		Instrument Check
Instrument Channel Reactor Low Water Level (LIS-3-58A-D)	(1) (27)	Once/18 Months	(28)	Once/day
Instrument Channel Reactor Low Water Level (LIS-3-184 & 185)	(1) (27)	Once/18 Months	(28)	Once/day
Instrument Channel Reactor Low Water Level (LIS-3-52 & 62A)	(1) (27)	Once/18 Months	(28)	Once/day
Instrument Channel Drywell High Pressure (PIS-64-58E-H)	(1) (27)	Once/18 Months	(28)	none
Instrument Channel Drywell High Pressure (PIS-64-58A-D)	(1) (27)	Once/18 Months	(28)	none
Instrument Channel Drywell High Pressure (PIS-64-57A-D)	(1) (27)	Once/18 Months	(28)	none
Instrument Channel Reactor Low Pressure (PIS-3-74A&B, PS-3-74A&B) (PIS-68-95, PS-68-95) (PIS-68-96, PS-68-96)	(1) (27)	Once/6 Months	(28)	none

TABLE 4.2.B (Continued)
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Core Spray Auto Sequencing Timers (Normal Power)	(4)	Once/operating cycle	none
Core Spray Auto Sequencing Timers (Diesel Power)	(4)	Once/operating cycle	none
LPCI Auto Sequencing Timers (Normal Power)	(4)	Once/operating cycle	none
LPCI Auto Sequencing Timers (Diesel Power)	(4)	Once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Normal Power)	(4)	Once/operating cycle	none
RHRSW A1, B3, C1, D3 Timers (Diesel Power)	(4)	Once/operating cycle	none
ADS Timer (105 sec.)	(4)	Once/operating cycle	none
ADS Timer (12 1/2 min.) (High Drywell Pressure Bypass Timer)	(4)	Once/operating cycle	none

26. This instrument check consists of comparing the background signal levels for all valves for consistency and for nominal expected values (not required during refueling outages).

27. Functional test consists of the injection of a simulated signal into the electronic trip circuitry in place of the sensor signal to verify operability of the trip and alarm functions.

28. Calibration consists of the adjustment of the primary sensor and associated components so that they correspond within acceptable range and accuracy to known values of the parameter which the channel monitors, including adjustment of the electronic trip circuitry, so that its output relay changes state at or more conservatively than the analog equivalent of the trip level setting.

29. The functional test frequency decreased to once/3 months to reduce challenges to relief valves per NUREG-0737, Item II.K.3.16.

30. Calibration shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one-point source check of the detector below 10 R/hr with an installed or portable gamma source.

31. Functional Tests shall be performed once/3 months.

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ATTACHMENT TO LICENSE AMENDMENT NO. 181

FACILITY OPERATING LICENSE NO. DPR-52

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

REMOVE	INSERT
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
1.1/2.1-12	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
--	1.1/2.1-16a
3.2/4.2-25	3.2/4.2-25
--	3.2/4.2-25a
3.5/4.5-20	3.5/4.5-20
3.5/4.5-20a	3.5/4.5-20a*

*Denotes overleaf or spillover page

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

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

Specifications

A. Thermal Power Limits

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

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

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

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

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

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 (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

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

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

2.1.A **Neutron Flux Trip Settings**

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

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

c. The APRM Rod Block trip setting shall be:

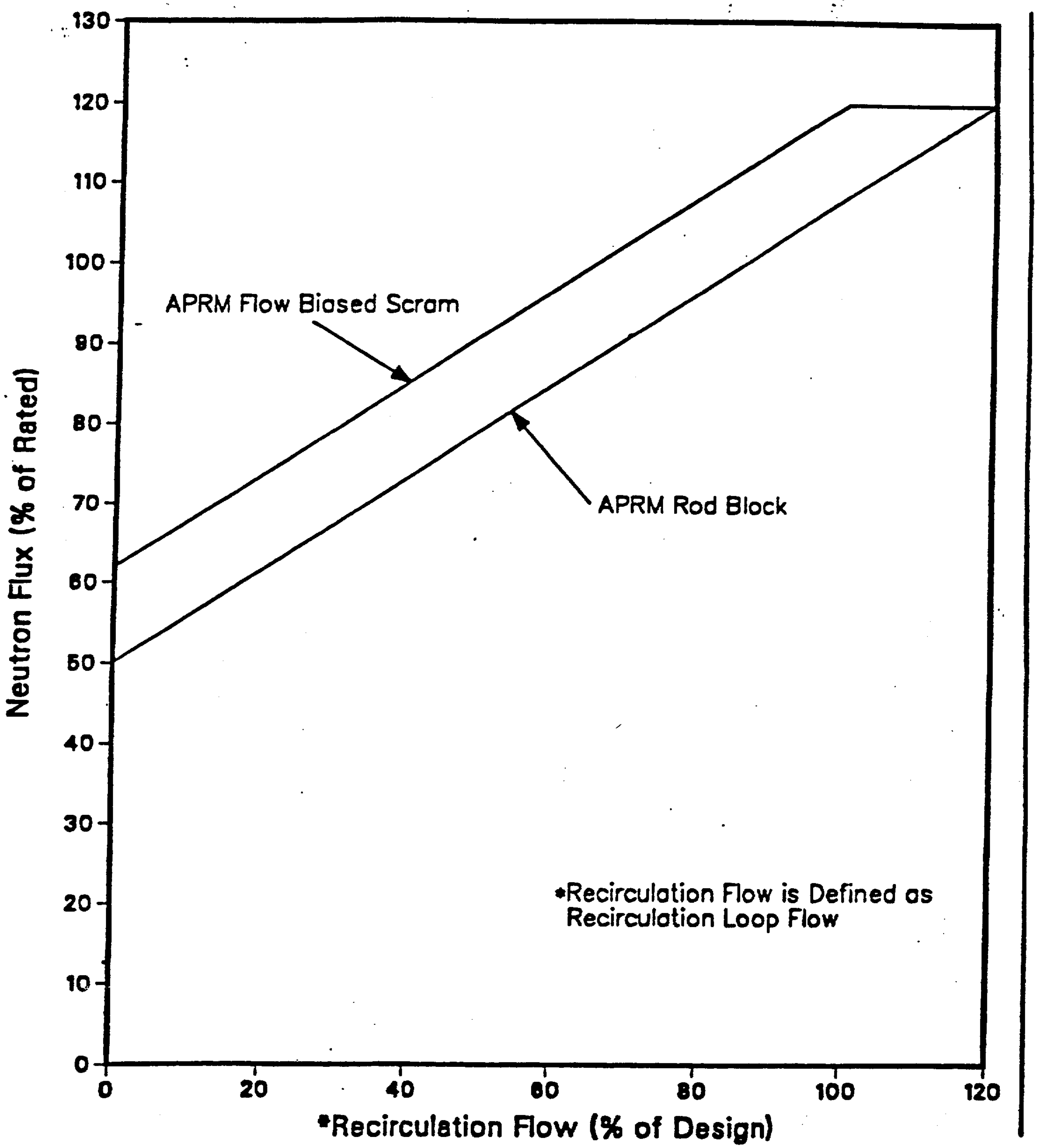
$$S_{RB} \leq (0.58W + 50\%)$$

where:

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

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

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.



APRM Flow Reference Scram and APRM Rod Block Settings
Fig. 2.1-1

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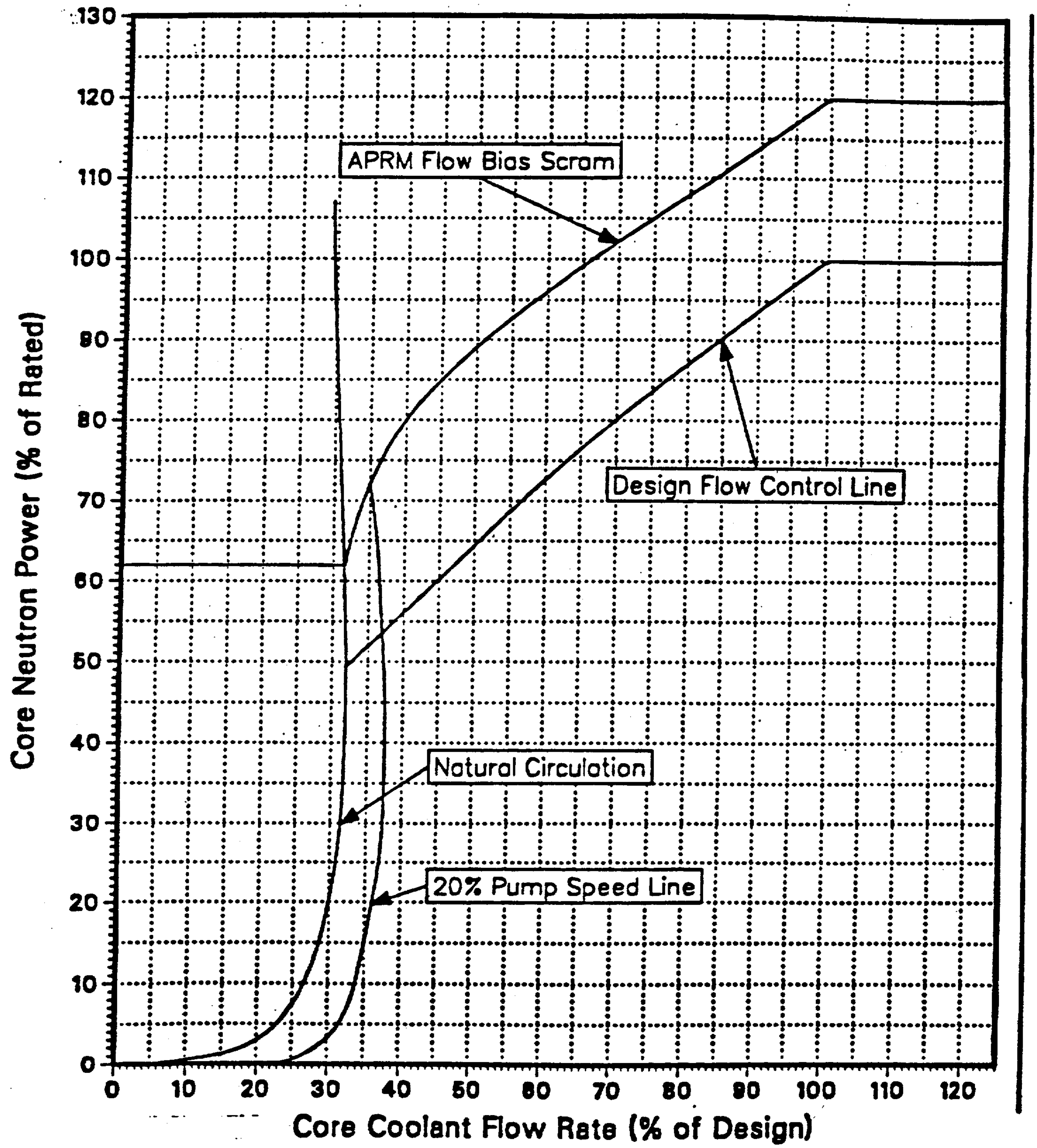

APRM Flow Bias Scram vs. Reactor Core Flow
Fig. 2.1-2

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In summary

1. The licensed maximum power level is 3,293 MWt.

2. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

3. The abnormal operational transients were analyzed to a power level of 3,440 MWt.

4. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

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 limits specified in Specification 3.5.k.

2. APRM Flux Scram Trip Setting (Refuel or Start & 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 and the Rod Sequence Control System. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

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

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 3). 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 108 percent of rated thermal power because of the APRM rod block trip setting. 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.

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

2. Generic Reload Fuel Application, Licensing Topical Report NEDE-20411-P-A, and Addenda.

3. Browns Ferry Nuclear Plant Unit 2, Cycle 6, Licensing Report, Extended Load Line Limit Analysis, TVA-BFE-052, April, 1990.

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TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
4(1)	APRM Upscale (Flow Bias)	≤0.58W + 50% (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)	≤0.66W + 40% (2)(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
2(1)	RCSC Restraint (PS85-61A,B)	147 psig turbine first stage pressure
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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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5 Core and Containment Cooling Systems

L. **APRM Setpoints**

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

$$S \leq (0.58W + 62\%) \left(\frac{FRP}{CMFLPD}\right)$$

$$S_{RB} \leq (0.58W + 50\%) \left(\frac{FRP}{CMFLPD}\right)$$

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

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

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:

4.5 Core and Containment Cooling Systems

L. **APRM Setpoints**

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

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.

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