ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-339
MARKED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
viii	viii	viii
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1.1/2.1-2	1.1/2.1-15	1.1/2.1-2
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II. MARKED PAGES

See attached.

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LIST OF ILLUSTRATIONS

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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.

NEW DEFINITION 1.U.5

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.

1.0 DEFINITIONS (Cont'd)

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.

Change from lower case to all upper case

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.

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

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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.

NEW DEFINITION 1.OO

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.

1.1/2.1 FUEL CLADDING INTEGRITY

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

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

0.58W + 62 %

S≤(~~0.66W + 64%~~)

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 34.2x10^6 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 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.

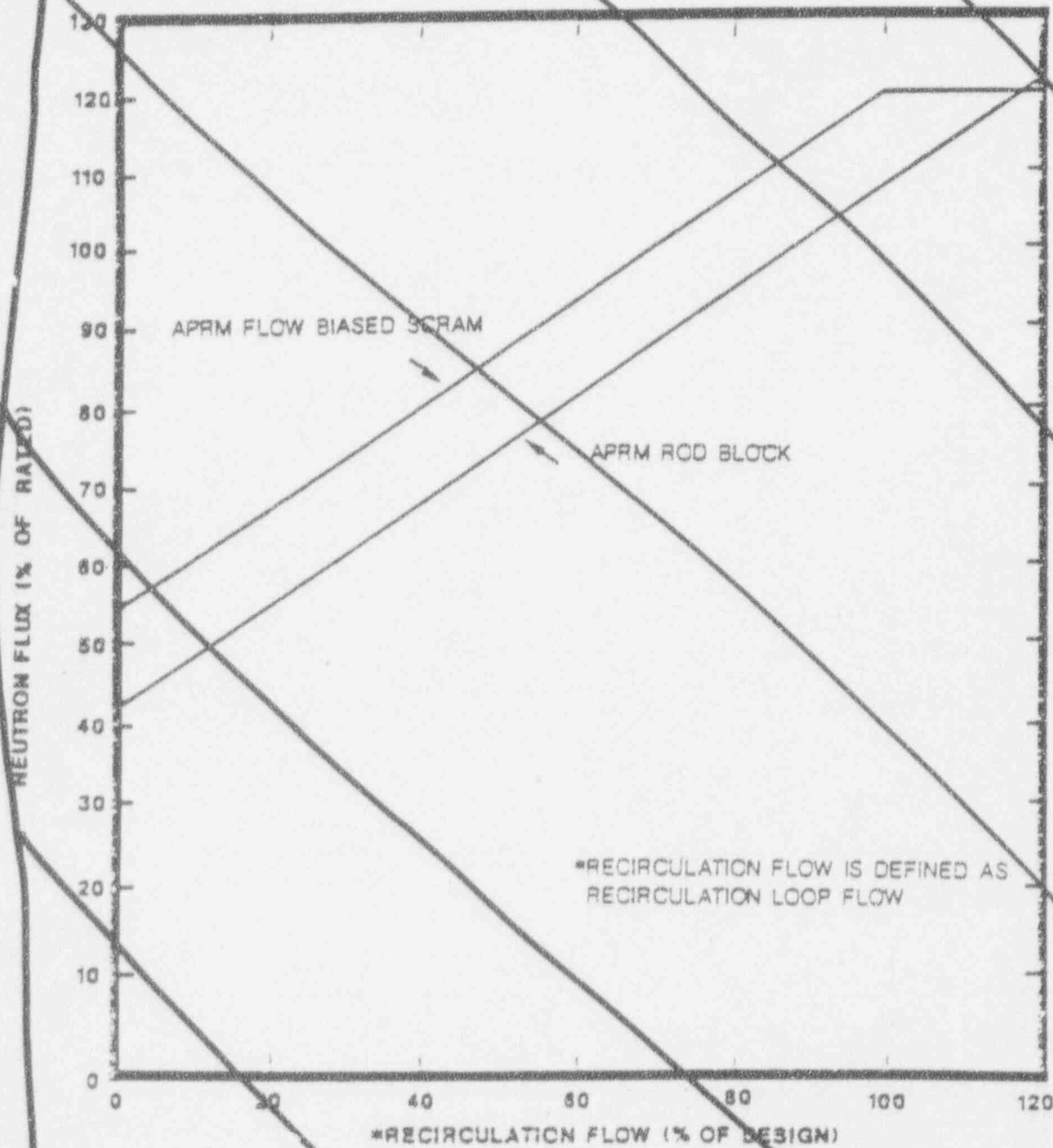
~~$S_{RB} \leq (0.66W + 42\%)$~~

~~where:~~

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

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

DELETE THIS FIGURE



APRM FLOW REFERENCE SCRAM AND APRM ROD BLOCK SETTINGS

FIGURE 2.1-1

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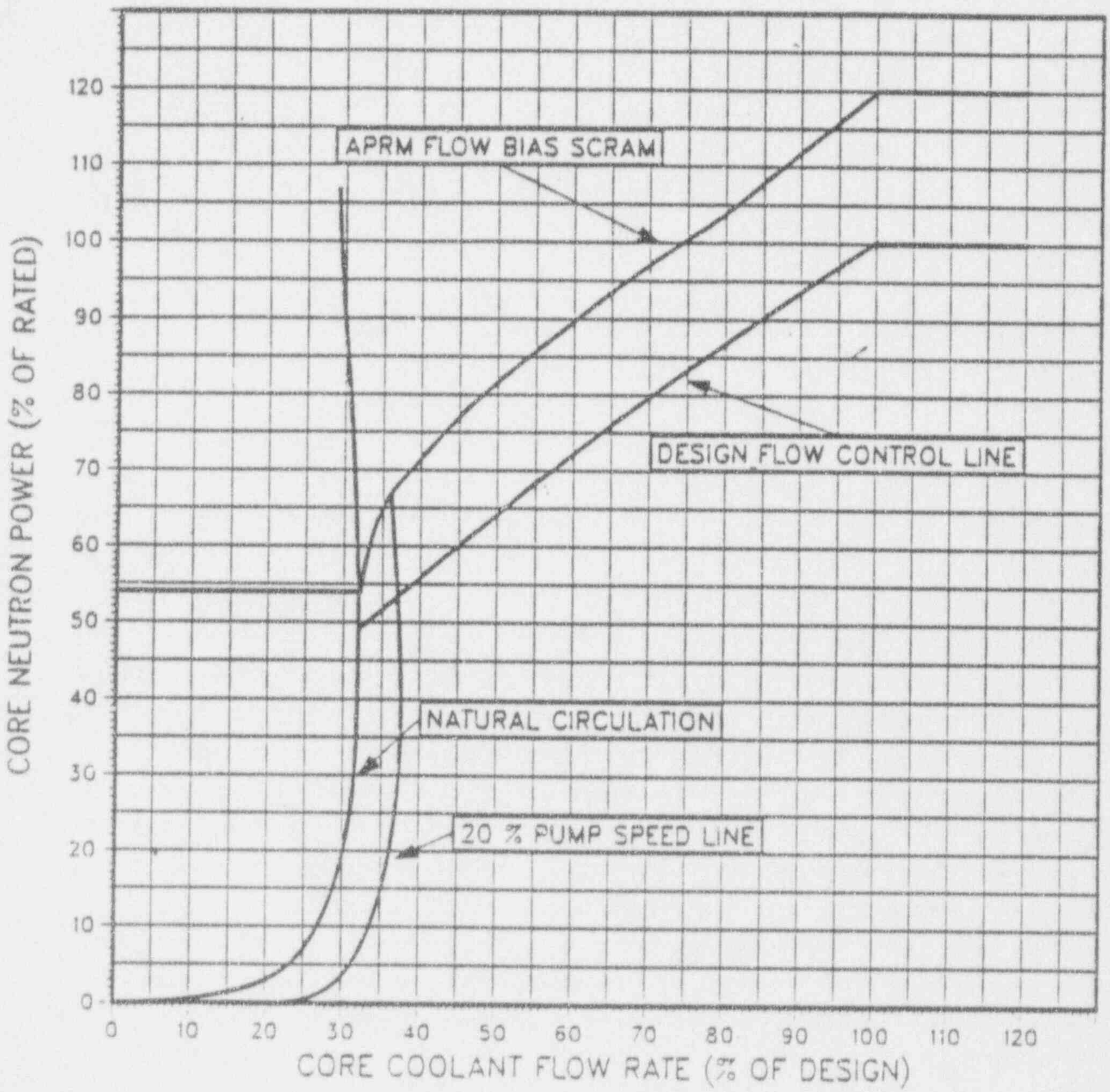
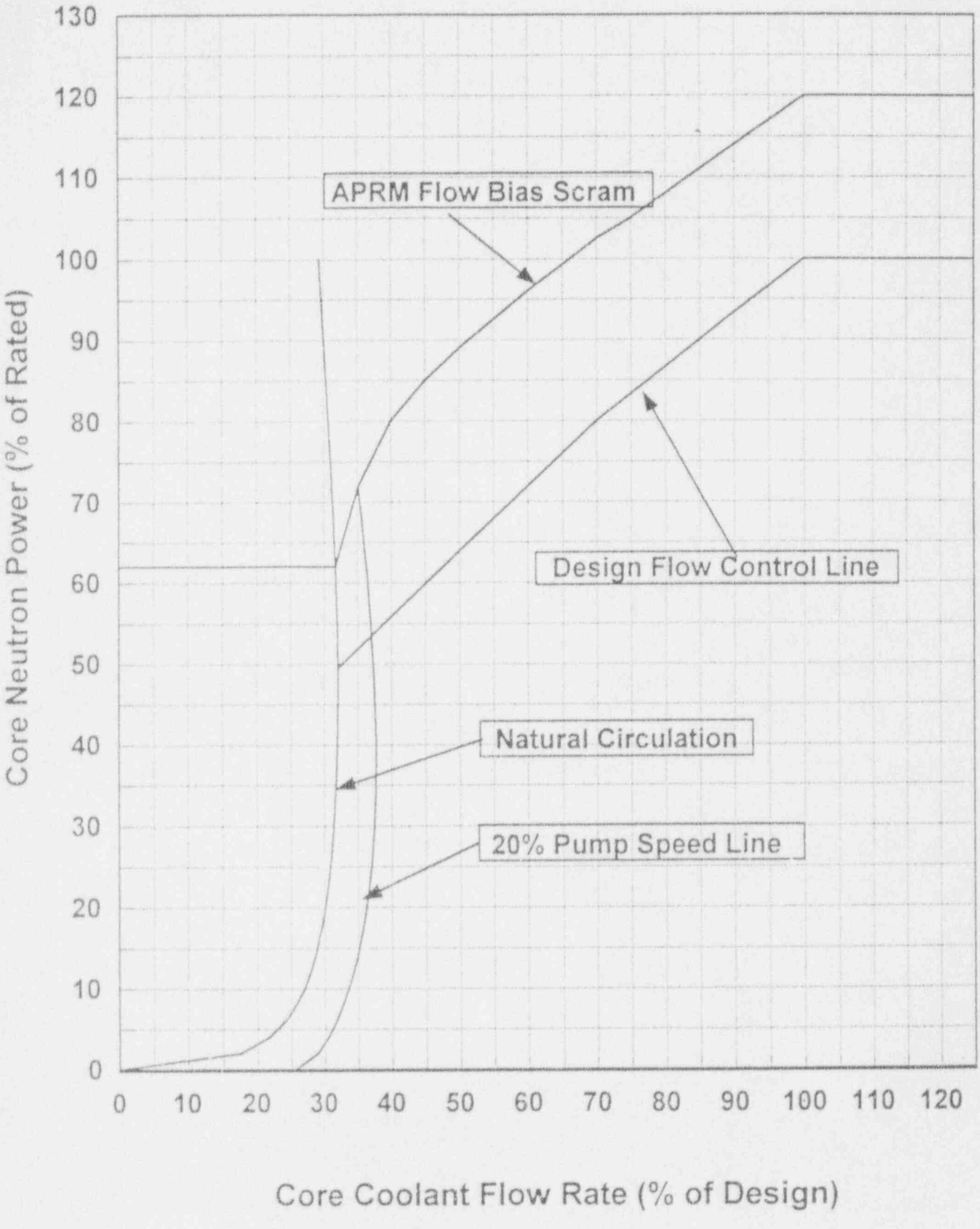


APRM FLOW BIAS SCRAM Vs. REACTOR CORE FLOW
Fig. 2.1-2



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

2.1 BASES (Cont'd)

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The bases for individual setpoints are discussed below:

MAKE ALL UPPER CASE

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.

power increase

During 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 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.

scram

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. ~~and~~ An ~~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.

MAKE ALL UPPERCASE

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 ~~recirculation flow range.~~ 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 ~~100 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 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.

the maximum thermal power level permitted by the APRM rod block trip setting, which is found in the CORE OPERATING LIMITS REPORT.

2.1 BASES (Cont'd)

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F. (Deleted)

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

The scram feature that occurs when the main steam line isolation valves close shuts down the reactor

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. ~~Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown~~ 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.

MAKE ALL LOWER CASE

MAKE ALL UPPER CASE

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.

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)	~~≤0.66W + 42%~~ (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
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. ~~W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt)~~.

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

   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 ~~and~~ 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 ± 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.

Change to all upper case

13. ~~RBM upscale flow biased setpoint clipped at 100 percent rated reactor power.~~

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
3.3.B. Control Rods	4.3.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.	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. 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.	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 ~~limiting control rod patterns, as determined by the designated qualified personnel~~ 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.	5. ~~When a limiting control rod pattern exists,~~ 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.

MAKE ALL UPPER CASE

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.

DELETE

~~A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit, (i.e., MCPR given by Specification 3.5.K or LHGR given by Specification 3.5.J). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its OPERABILITY will assure that improper withdrawal does not occur. It is normally the responsibility of the nuclear engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.~~

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

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.

4.5.J Linear Heat Generation Rate (LHGR)

MAKE ALL UPPER CASE

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at ≥ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern ~~as described in the bases for Specification 3.3~~.

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.

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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 ~~and rod block~~ setpoint equation~~s~~ listed in Section~~s~~ 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT ~~and 2.1.B~~ shall be multiplied by FRP/CMFLPD~~, as follows:~~

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.5.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.

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

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.

(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.

LIST OF ILLUSTRATIONS

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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 ~~[illegible]~~ 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.

AMENDMENT NO. 181

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:

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

~~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)~~

less than or equal to the limit specified in the CORE OPERATING LIMITS REPORT.

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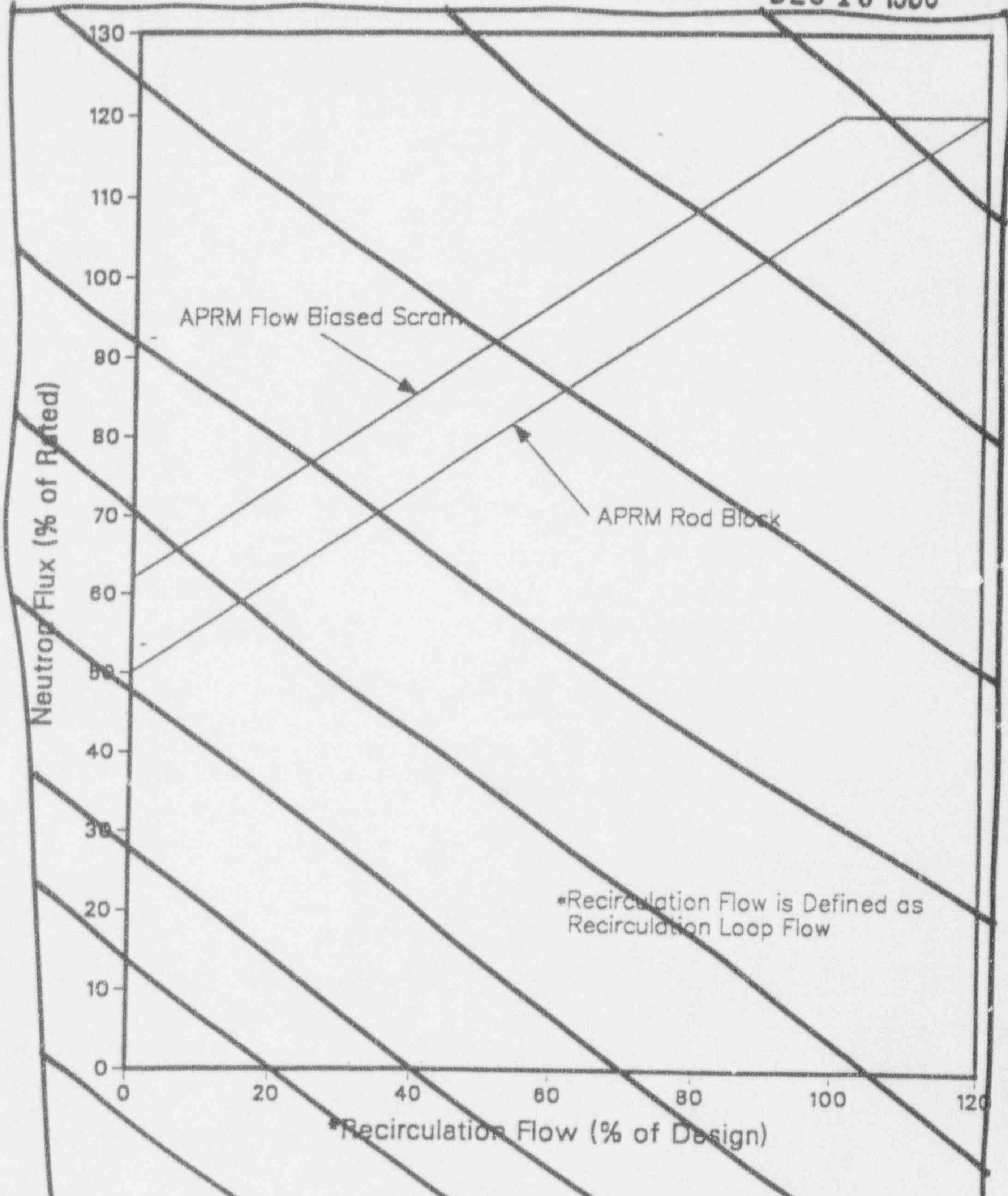

APRM Flow Reference Scram and APRM Rod Block Settings
Fig. 2.1-1

DELETE THIS FIGURE

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 ~~100 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.

the maximum thermal power level permitted by the APRM rod block trip setting, which is found in the CORE OPERATING LIMITS REPORT.

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
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.

NOTES FOR TABLE 3.2.C

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

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

2. ~~W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt).~~ 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.

X 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 (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. ~~The other channel may also be defeated only if the conditions of "e" or "f" are met.~~ If the inoperable channel cannot be restored within 24 hours, ~~and the conditions of "e" or "f" are not met,~~ the inoperable channel shall be placed in the tripped condition within one hour.

~~*The provisions of Note 7e and 7f are applicable during unit 2 cycle 6 only.~~

NOTES FOR TABLE 3.2.C (Cont'd)

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7. (Continued)

   d. With both RBM channels inoperable, ~~and the conditions of "e" or "f" not met~~, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

   ~~e. The RBM need not be OPERABLE when reactor power is ≥90 percent and MCPR is ≥1.40.~~

   ~~f. The RBM need not be OPERABLE when reactor power is <90 percent and MCPR is ≥1.70.~~

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

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

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

    a. SRM and IRM

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

       (2) Power supply voltage low.

       (3) Circuit boards not in circuit.

    b. APRM

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

       (2) Less than 14 LPRM inputs.

       (3) Circuit boards not in circuit.

    c. RBM

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

       (2) Circuit boards not in circuit.

       (3) RBM fails to null.

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

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

~~*The provisions of Note 7e and 7f are applicable during unit 2 cycle 6 only.~~

NOTES FOR TABLE 3.2.C (Cont'd)

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

13. ~~RBM upscale flow biased setpoint clipped at 100 percent rated reactor power.~~

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

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The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

~~A General Electric study, GE NE 770-06-0392 shows for the unit 2 cycle 6 core that if the initial MCPR is as specified in item 7e or 7f of Table 3.2.G, then no single rod withdrawal error can cause the MCPR to decrease below the MCPR safety limit. When core operating conditions have been verified to be within the limits of items 7e or 7f of Table 3.2.G, the RBM is not required.~~ ~~When the RBM is required, the minimum instrument channel requirements apply. These~~ The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are

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LIMITING CONDITIONS FOR OPERATION

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

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

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

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

and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT

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:

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

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

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.

6.9.1.6 SOURCE TESTS

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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.

LIST OF ILLUSTRATIONS

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1.0 DEFINITIONS (Cont'd)

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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.

NEW DEFINITION 1.U.5

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.

1.0 DEFINITIONS (Cont'd)

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.

change from lower case to all upper case

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.

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

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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.

NEW DEFINITION 1.00

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.

1.1/2.1 FUEL CLADDING INTEGRITY

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

2.1.A.1.a (Cont'd) 0.58W+62%

S≤(~~0.66W + 54%~~)

where:

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

W = Loop recirculation flow rate in percent of rated ~~(rated loop recirculation flow rate equals 34.2x10^6 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 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

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

~~where:~~

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

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

less than or equal to the limit specified in the CORE OPERATING LIMITS REPORT.

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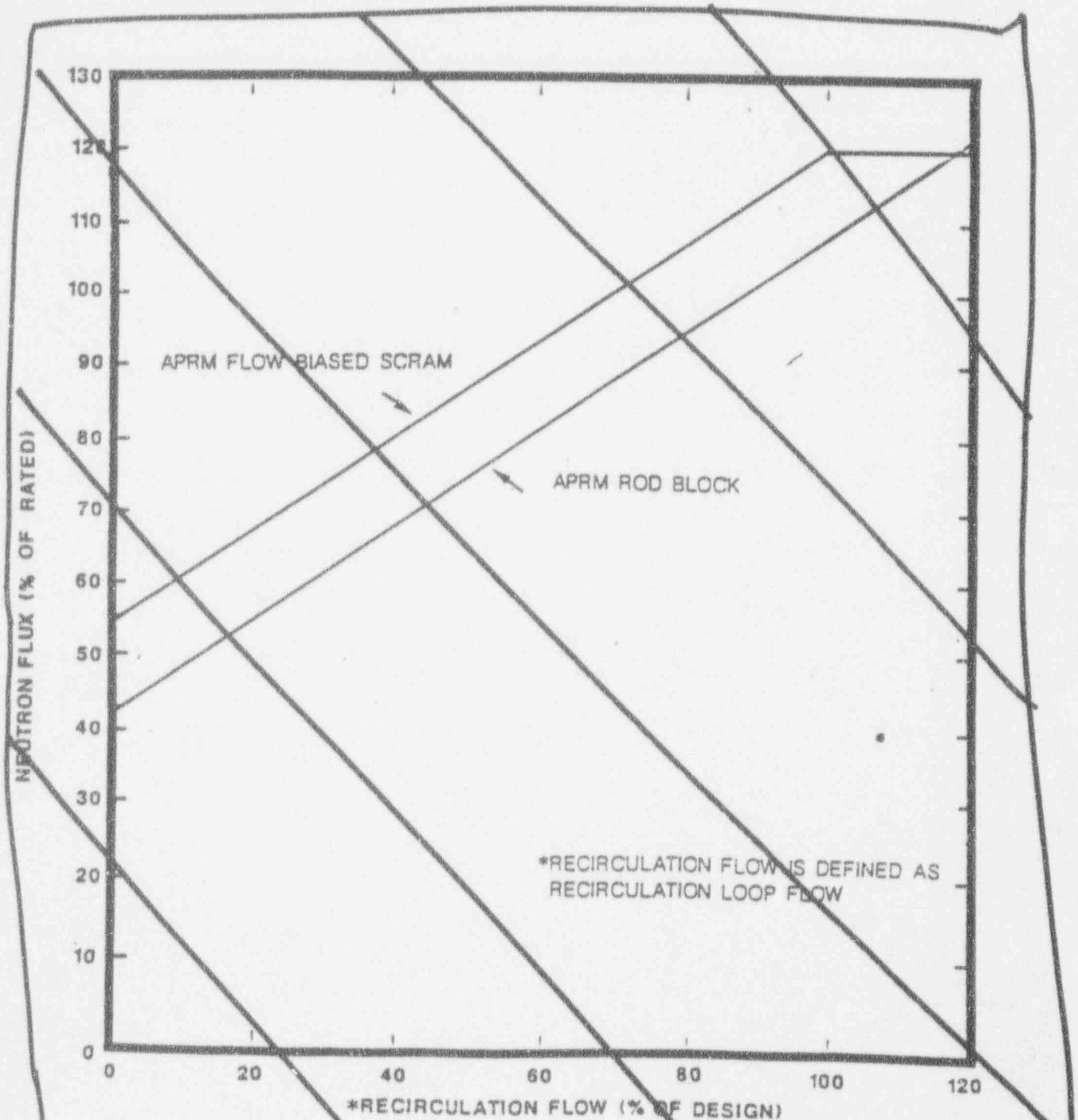
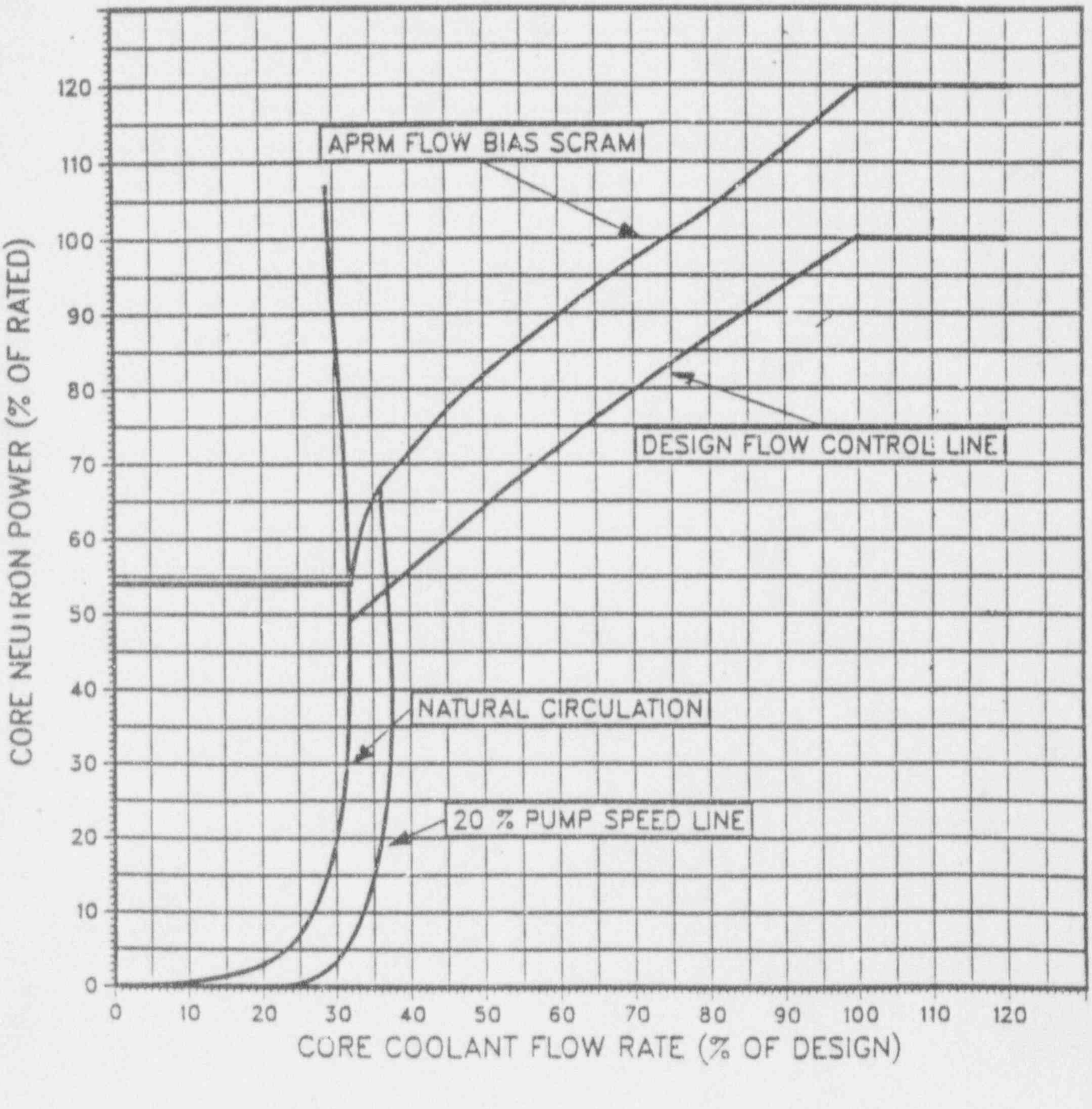


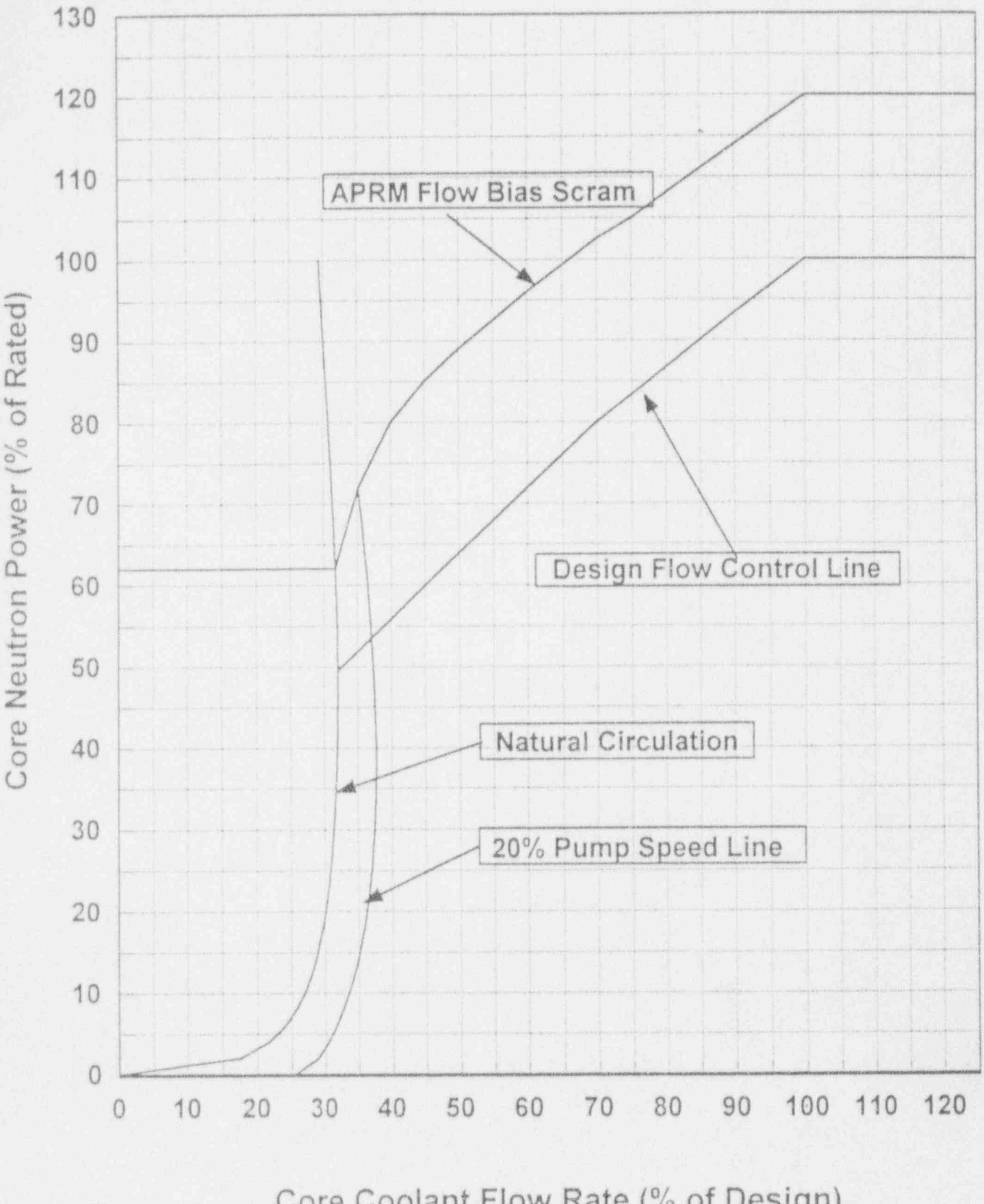
APRM FLOW REFERENCE SCRAM AND APRM ROD BLOCK SETTINGS

FIGURE 2.1-1

REPLACE THIS FIGURE WITH NEW FIGURE ON FOLLOWING PAGE



APRM FLOW BIAS SCRAM Vs. REACTOR CORE FLOW
Fig. 2.1–2



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

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The bases for individual setpoints are discussed below: MAKE ALL UPPER CASE

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.

power increase

During 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 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.

scram

2.1 BASES (Cont'd)

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. ~~and~~ An ~~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.

MAKE ALL UPPER CASE

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 ~~recirculation flow range.~~ power/flow domain including above the rated rod line (Reference 1). The margin to the Safety Limit increases as the flow

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The bases for individual setpoints are discussed below:

MAKE ALL UPPER CASE

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.

power increase

During 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 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.

scram

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~~, and~~. An ~~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.

MAKE ALL UPPER CASE

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 ~~recirculation flow range.~~ 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 ~~100 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 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.

the maximum thermal power level permitted by the APRM rod block trip setting, which is found in the CORE OPERATING LIMITS REPORT.

2.1 BASES (Cont'd)

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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. ~~Advantage is taken of the scram feature that occurs when the main steam line isolation valves are closed, to provide for reactor shutdown~~ 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.

The scram feature that occurs when the main steam line isolation valves close shuts down the reactor

MAKE ALL LOWER CASE

MAKE ALL UPPER CASE

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).

BFN
Unit 3

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)	~~≤0.66W + 42%~~ (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
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. ~~W is the recirculation loop flow in percent of design. Trip level setting is in percent of rated power (3293 MWt).~~ The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

   ~~See Specification 3.1 for APRM control rod block setpoint.~~

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

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

   SRM detector not in startup position is bypassed when the count rate is ≥100 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 ≤30 percent ~~and~~ 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.

Change to all upper case

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

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

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

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.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.

MAKE ALL UPPER CASE

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 ~~limiting control rod patterns, as determined by the designated qualified personnel,~~ 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. ~~When a limiting control rod pattern exists,~~ 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.

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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.

DELETE

~~A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR given by Specification 3.5.K or LHGR given by Specification 3.5.J). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its OPERABILITY will assure that improper withdrawal does not occur. It is normally the responsibility of the nuclear engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform these functions may be designated by the plant superintendent to perform these functions.~~

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

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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 ~~as described in the bases for Specification 3.3.~~

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.

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LIMITING CONDITIONS FOR OPERATION

3.5 Core and Containment Cooling Systems

L. APRM Setpoints

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.5 Core and Containment Cooling Systems

L. APRM Setpoints

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

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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

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.

(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.

AMENDMENT NO. 170

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-339
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
viii	viii	viii
1.0-8	1.1/2.1-2	1.0-8
1.0-9*	1.1/2.1-3	1.0-9*
1.0-12a	1.1/2.1-6	1.0-12a
1.1/2.1-2	1.1/2.1-15	1.1/2.1-2
1.1/2.1-3	3.2/4.2-25	1.1/2.1-3
1.1/2.1-6	3.2/4.2-26	1.1/2.1-6
1.1/2.1-7	3.2/4.2-27	1.1/2.1-7
1.1/2.1-12	3.2/4.2-68	1.1/2.1-12
1.1/2.1-14	3.5/4.5-20	1.1/2.1-14
1.1/2.1-15	6.0-26a	1.1/2.1-15
1.1/2.1-16	6.0-26b*	1.1/2.1-16
3.2/4.2-25		3.2/4.2-24
3.2/4.2-26		3.2/4.2-25
3.2/4.2-27		3.2/4.2-26
3.3/4.3-8		3.3/4.3-8
3.3/4.3-17		3.3/4.3-17
3.5/4.5-19		3.5/4.5-19
3.5/4.5-20		3.5/4.5-20
6.0-26a		6.0-26a

* Spillover pages

II. REVISED PAGES

See attached.

LIST OF ILLUSTRATIONS

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 function 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)

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.

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.

Figure 2.1-1

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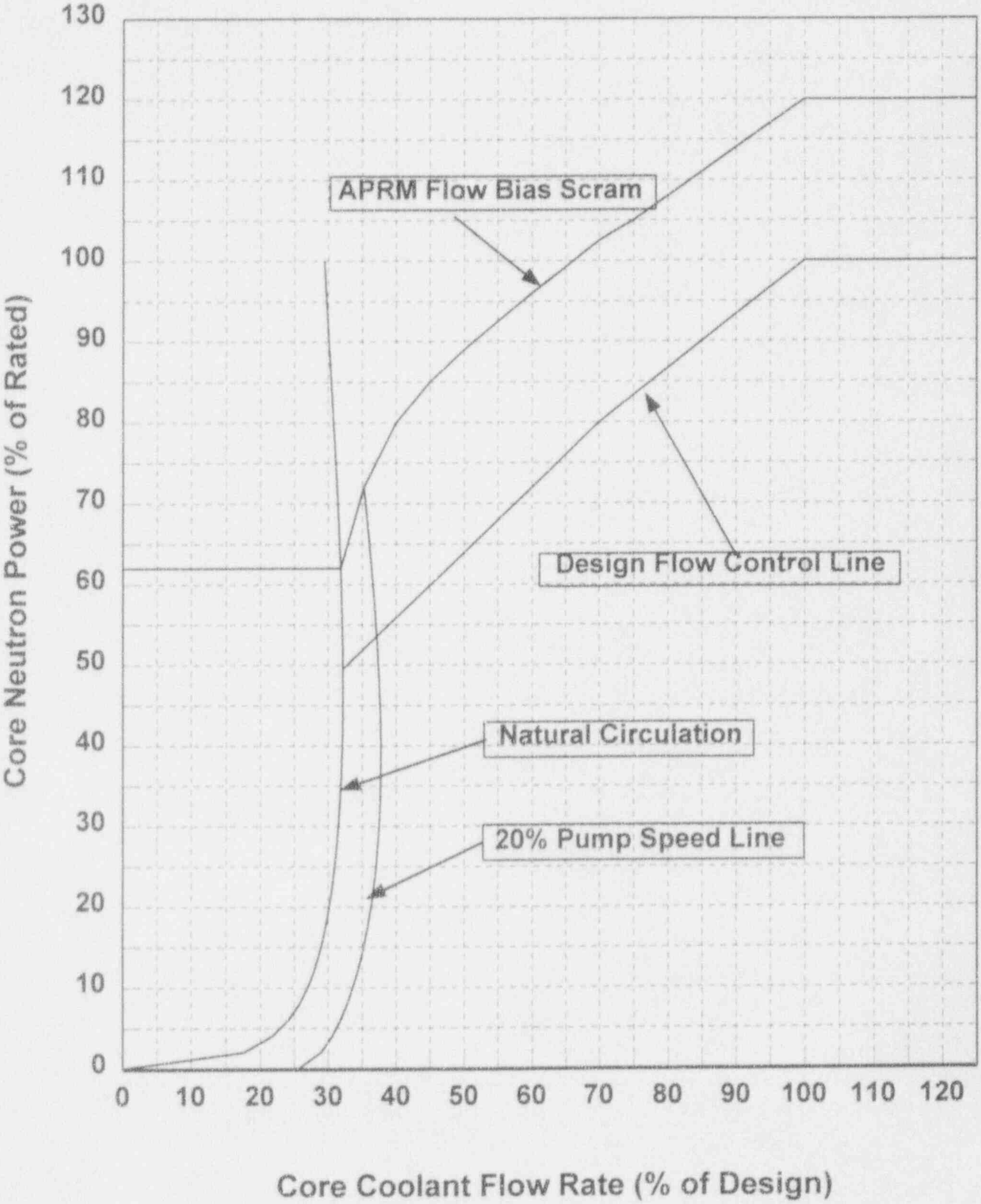

APRM Flow Bias Scram vs. Reactor Core Flow

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.

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

2.1 BASES (Cont'd)

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 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.

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.

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	SURVEILLANCE REQUIREMENTS
3.3.B. Control Rods	4.3.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.	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. 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.	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, either: a. Both RBM channels shall be OPERABLE: or b. Control rod withdrawal shall be blocked.	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

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 $\geq$ 25% of rated, the ratio of FRP/CMFLPD shall be $\geq$ 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 $\leq$ 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 $\geq$ 25% of rated thermal power.

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.

LIST OF ILLUSTRATIONS

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.

Figure 2.1-1

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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)	≤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.

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.

3.2 BASES (Cont'd)

The instrumentation which initiates CSCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.07. The trip logic for this function is 1-out-of-n: e.g., any trip on one of six APRMs, eight IRMs, or four SRMs will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.07.

The RBM rod block function provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern.

If the IRM channels are in the worst condition of allowed bypass, the sealing arrangement is such that for unbypassed IRM channels, a rod block signal is generated before the detected neutrons flux has increased by more than a factor of 10.

A downscale indication is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented.

The refueling interlocks also operate one logic channel, and are required for safety only when the mode switch is in the refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are

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.

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:

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.

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.

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.

6.9.1.7 CORE OPERATING LIMITS REPORT (Continued)

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.

LIST OF ILLUSTRATIONS

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)

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.

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.

Figure 2.1-1

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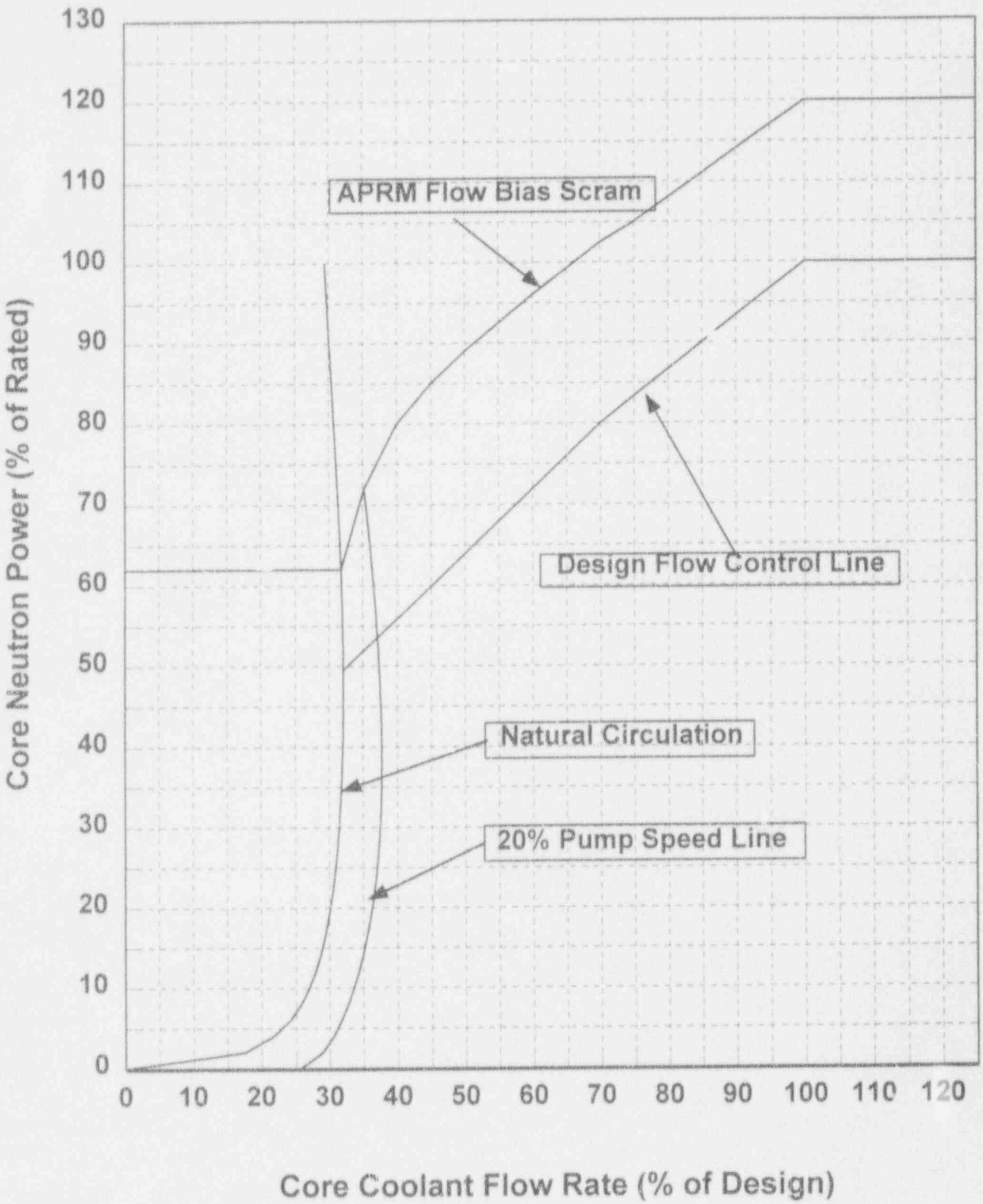

APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

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)

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 dis ribution 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).

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.

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 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 ≤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	SURVEILLANCE REQUIREMENTS
3.3.B. Control Rods	4.3.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.	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. 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.	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, either: a. Both RBM channels shall be OPERABLE: or b. Control rod withdrawal shall be blocked.	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

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.

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.