ENCLOSURE 2
TENNESSEE VALLEY AUTHORITY (TVA)
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, and 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-353
MARKED PAGES

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1.1/2.1-3	1.1/2.1-3	1.1/2.1-3
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3.2/4.2-68	3.2/4.2-68	3.2/4.2-67
3.2/4.2-73	3.2/4.2-73a	3.2/4.2-72
3.3/4.3-8	3.3/4.3-8	3.3/4.3-8
3.3/4.3-17	3.3/4.3-17	3.3/4.3-17
3.5/4.5-18	3.5/4.5-18	3.5/4.5-18
3.5-4.5-19	3.5-4.5-19	3.5-4.5-19
3.5/4.5-20	3.5/4.5-20	3.5/4.5-20
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II. MARKED PAGES

See attached.

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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.

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.

(DELETED)

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

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

~~5. CORE MAXIMUM FRACTION OF CRITICAL POWER (CMFCP) - CORE MAXIMUM FRACTION OF CRITICAL POWER is the maximum value of the ratio of the flow-corrected CPR operating limit found in the CORE OPERATING LIMITS REPORT divided by the actual CPR for all fuel assemblies in the core.~~

V. Instrumentation

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

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

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

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

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

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

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

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

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

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

Specifications

A. Thermal Power Limits

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

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

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

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

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

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

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

(0.66W + 71%)

$S \leq \cancel{(0.58W + 62\%)}$

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.

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

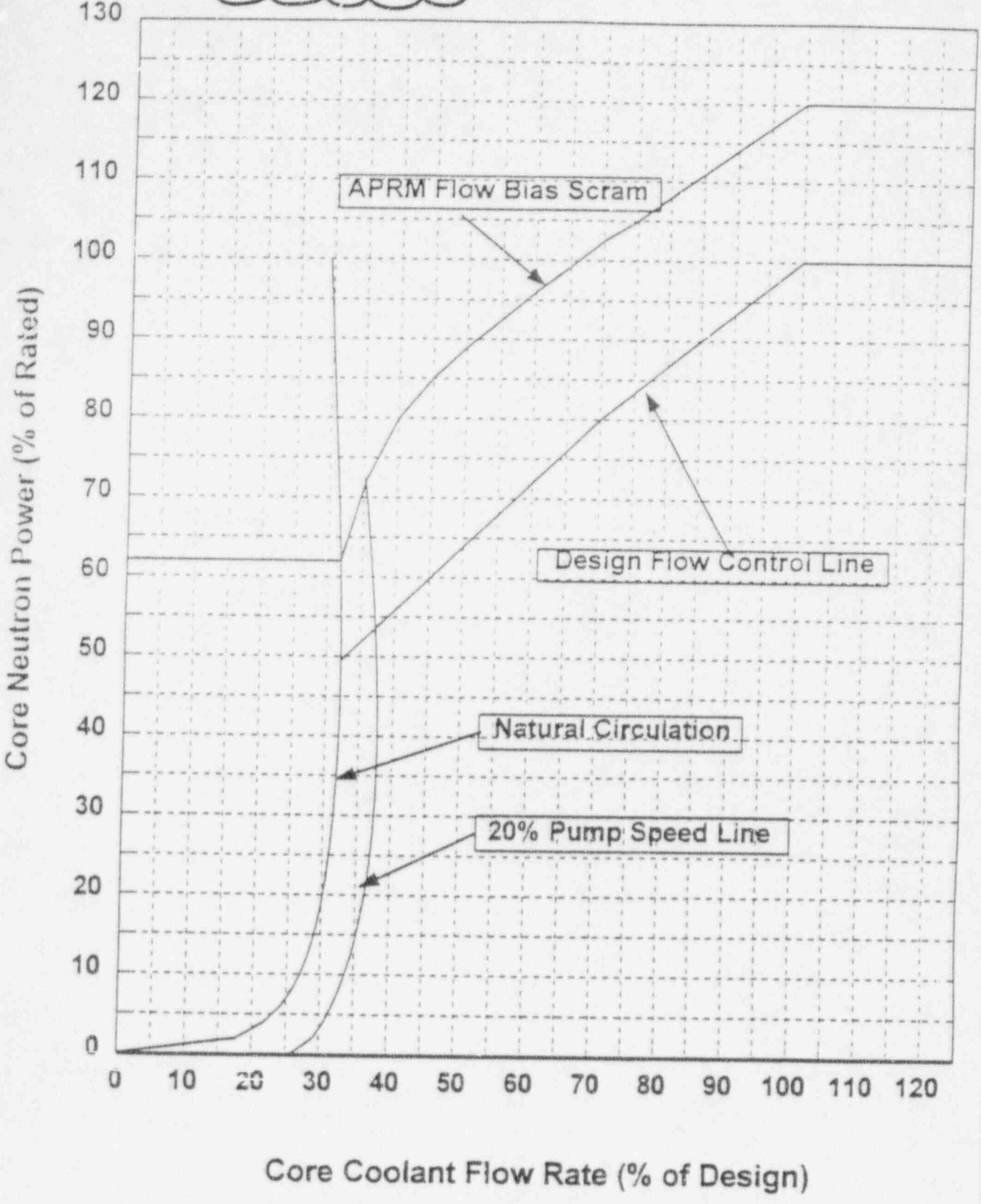
2.1.A Neutron Flux Trip Settings

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.

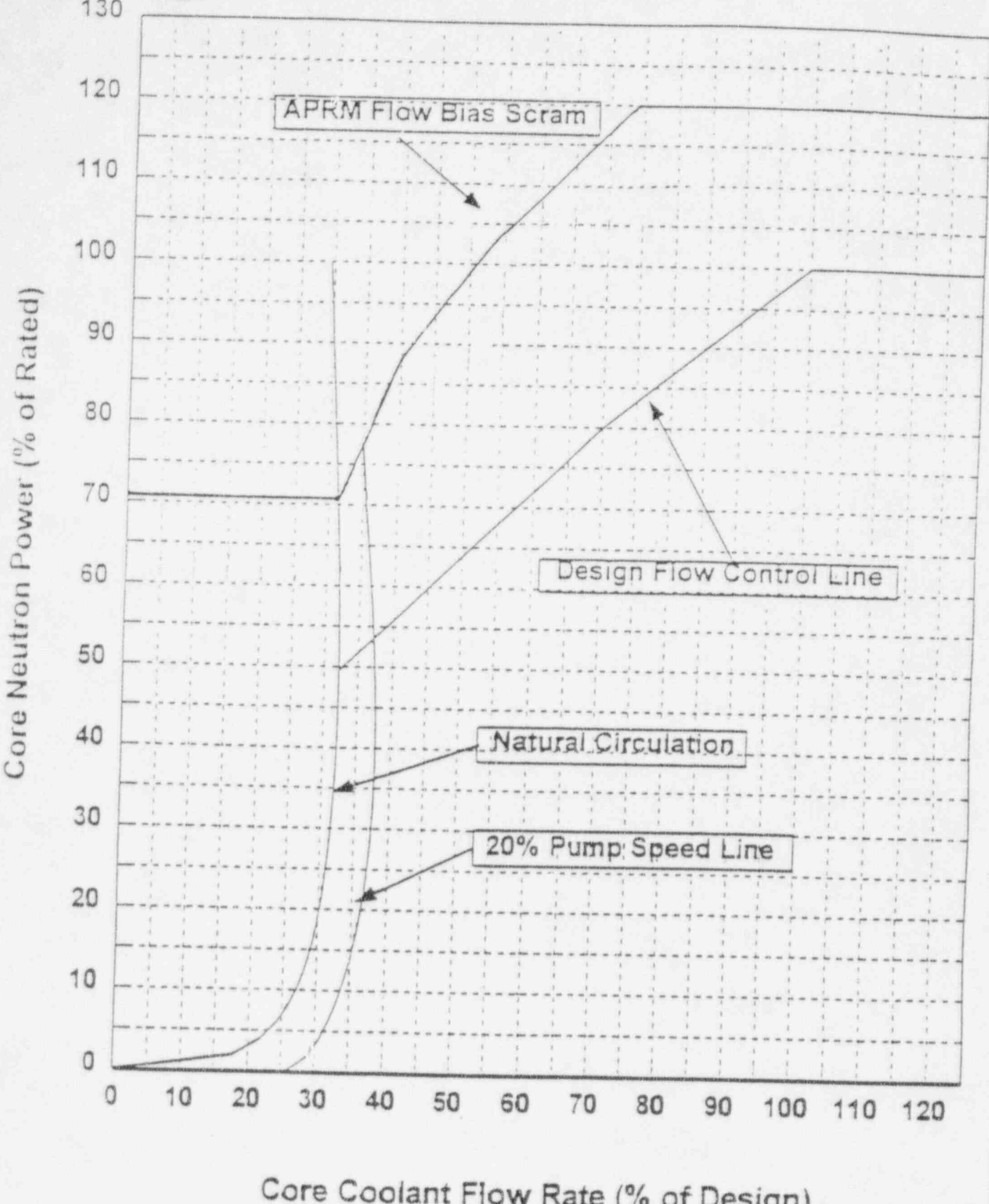
REPLACE WITH
NEW FIGURE



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

New



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

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.

5. MAXIMUM EXTENDED LOAD LINE LIMIT AND ARTS IMPROVEMENT PROGRAM ANALYSES FOR BROWNS FERRY NUCLEAR PLANT UNIT 1, 2 AND 3, NEDC-32433P.

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Shut-down	Refuel (7)	Startup/ Hot Standby	Run	Action (1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(22) (21)	X	(5)	1.A
3	Inoperable			X	X	(5)	1.A
	APRM (16)(24)(25)						
~~2~~ 3(11)	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B or 1.E
~~2~~ 3(11)	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B or 1.E
~~2~~ 3(11)	High Flux	≤ 15% rated power		~~X(21)~~	X(17)	(15)	1.A
~~2~~ 3(11)	Inoperative	(13)		~~X(21)~~	X(17)	X	1.A
2	~~Downscale~~ 2/4 Trip Voter	~~≥ 3 Indicated on Scale~~ (12)		~~(11)~~	~~(11)~~ X	X~~(11)~~	1.A ~~or 1.B~~ or 1.F
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A

Amendment No. 134
Corrected 8/24/87

JUL 17 1987

NOTES FOR TABLE 3.1.A

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for one trip system, trip the inoperable channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of operable instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An inoperable channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable channel shall be restored to operable status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all operable control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the STARTUP/HOT Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30 percent of rated.

INSERT B

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram with control rod block for reactor protection system reset.

3. Bypassed if reactor pressure is less than 1055 psig and mode switch not in RUN.

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the RUN position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. ~~APRM 15 percent scram~~ (deleted)

INSERT B:

E. For the APRM functions only, if only two APRM channels are OPERABLE, restore a third APRM channel to OPERABLE status or trip one of the inoperable APRM channels within 6 hours. If only one APRM channel is OPERABLE, trip one inoperable APRM channel immediately and restore an inoperable APRM channel to OPERABLE status or initiate alternative action within 2 hours.

F. For the APRM functions only, if one voter channel is inoperable in one trip system, restore the voter channel to OPERABLE status or trip the inoperable channel or the entire trip system within 12 hours. If one voter channel is inoperable in both trip systems, restore the inoperable voter channels to OPERABLE status or initiate alternative action within 6 hours.

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8. Not required to be OPERABLE when primary containment integrity is not required.

9. (Deleted)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

Replace with Insert C

11. ~~The APRM downscale trip function is only active when the reactor mode switch is in RUN.~~

Replace with Insert D

12. ~~The APRM downscale trip is automatically bypassed when the IRM instrumentation is OPERABLE and not high.~~

the required minimum number 7

13. Less than ~~14~~ 7 OPERABLE LPRMs will cause ~~a trip system trip~~. an instrument channel inoperative alarm.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first state pressure is greater than or equal to 154 psig.

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. (Deleted)

21. ~~The APRM High Flux and Inoperative Trips do not have to be OPERABLE in the REFUEL Mode if the Source Range Monitors are connected to give a noncoincidence, High Flux scram, at 5 x 10^5 cps.~~ The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the ~~Source Range Monitors~~. SRMs.

Replace with Insert E

INSERT C:

The same three (3) required APRM channels are shared by both RPS trip systems.

INSERT D:

Any combination of APRM upscale or inoperative trips from two different (non-bypassed) APRMs will trip all of the 2/4 voter units.

INSERT E:

In the REFUEL Mode unless adequate shutdown margin has been demonstrated per Specification 3.3.A.1, whenever any control rod is withdrawn from a core cell containing one or more fuel assemblies, shorting links shall be removed from the RPS circuitry to enable the Source Range Monitor (SRM) noncoincidence high-flux scram function.

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)	~~C~~	Trip Output Relays (4) (5) 2/4 VOTER LOGIC (10)	~~Before Each Startup and Weekly When Required to be Operable~~ EVERY 6 MONTHS (9) EACH REFUELING OUTAGE
High Flux (Flow Biased)	~~B~~	Trip Output Relays (4) (6) 2/4 VOTER LOGIC (10)	~~Once/Week~~ EVERY 6 MONTHS EACH REFUELING OUTAGE
High Flux (Fixed Trip)	~~B~~	Trip Output Relays (4) (5) 2/4 VOTER LOGIC (10)	~~Once/Week~~ EVERY 6 MONTHS EACH REFUELING OUTAGE
Inoperative	~~B~~	Trip Output Relays (4) (5) 2/4 VOTER LOGIC (10)	~~Once/Week~~ EVERY 6 MONTHS EACH REFUELING OUTAGE
~~Downscale~~	~~B~~	~~Trip Output Relays (4)~~	~~Once/Week~~
~~Flow Bias~~	~~B~~	~~(6)~~	~~(6)~~
2/4 Trip Voter		Trip Scram Contactors (11)	Once/Week
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level	A	Trip Channel and Alarm	Once/Month (1)

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be operable or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. ~~(Deleted)~~ Insert F

6. ~~The functional test of the flow bias network is performed in accordance with Table 4.2.C.~~ Replace w/ Insert G.

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

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

9. Not required to be performed when entering the STARTUP/HOT STANDBY Mode from RUN Mode until 12 hours after entering the STARTUP/HOT STANDBY Mode.

10. Functional test consists of simulating APRM trip conditions at the APRM channel outputs to check all combinations of two tripped inputs to the 2/4 voter logic in each voter channel.

11. Functional test consists of manually tripping the 2/4 voter trip output, one voter channel at a time, to demonstrate that each scram contactor for each RPS trip system channel (A1, A2, B1 and B2) operates and produces a half-scram.

INSERT F:

The channel functional test shall include both the APRM channels and the 2/4 voter channels.

INSERT G:

The channel functional test shall include both the APRM channels and the 2/4 voter channels plus the flow input function, excluding the flow transmitters.

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal	~~B~~	Heat Balance	Once/7 Days
Flow Bias Signal	~~B~~	Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal Indent	~~B~~	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume			
Electronic Lvl Switches (LS-85-45-A, B, G, H)	A	Calibrated Water Column (5)	Note (5)
Float Switches (LS-85-45C-F)	A	Calibrated Water Column (5)	Note (5)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive (PT-1-81A, B & PT-1-91A, B)	B	Standard Pressure Source	Once/Operating Cycle (9)
Turbine Control Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	Note (5)	Note (5)

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NOTES FOR TABLE 4.1.B

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1. A description of three groups is included in the bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

REPLACE WITH INSERT H

7. ~~The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operation during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.~~

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

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

INSERT H:

The flow bias signal calibration will consist of calibrating the analog differential pressure flow sensors once per operating cycle. Calibration of the flow bias processing system is done once per operating cycle as part of the overall APRM instrumentation calibration.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between nonclass 1E power supply and the class 1E RPS bus. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. ~~Three APRM instrument channels are provided for each protection trip system.~~

INSERT I

INSERT I:

The APRM system is divided into four APRM channels and four 2-out-of-4 trip voter channels. Each APRM channel provides input to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The APRM system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter units, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system resulting in a full scram.

Each APRM instrument channel receives input signals from forty-three (43) Local Power Range Monitors (LPRMs). A minimum of twenty (20) LPRM inputs with three (3) per axial level is required for the APRM instrument channel to be OPERABLE. Fewer than the required minimum number of LPRM inputs generates an instrument channel inoperative alarm and a control rod block but does not result in an automatic trip input to the 2-out-of-4 voters.

IRM

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Each protection trip system has one more ~~APRM~~ than is necessary to meet the minimum number required per channel. This allows the bypassing of

IRM

one ~~APRM~~ per protection trip system for maintenance, testing or calibration. ~~Additional IRM channels have also been provided to allow for bypassing of one such channel.~~ The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

3.1 BASES (Cont'd)

be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharge water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Reference Section 7.5.4 FSAR. Thus, the IRM is required in the REFUEL and STARTUP modes. In the power range the APRM system provides required protection. Reference Section 7.5.7 FSAR. Thus, the IRM System is not required in the RUN mode. The APRMs and the IRMs provide adequate coverage in the startup and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for STARTUP and RUN modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions as indicated in Table 3.1.1 operable in the REFUEL mode is to assure that shifting to the REFUEL mode during reactor power operation does not diminish the need for the reactor protection system.

Because of the APRM downscale rod block limit of ≥ 3 percent when in the RUN mode and high level Flux scram limit of ≤15 percent when in the STARTUP Mode, the transition between the STARTUP and RUN Modes must be made with the APRM instrumentation indicating between 3 percent and 15 percent of rated power ~~or a control rod/scram will occur~~. In addition, the IRM system must be indicating below the High Flux setting (120/125 of scale) or a scram will occur when in the STARTUP Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to shutdown). When power is being reduced, if a transfer to the STARTUP mode is made and the IRMs have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

Except for the APRMs which take credit for self-test capability,

4.1 BASES

The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (1). This concept was specifically adapted to the one out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failure such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1.A and 4.1.B are divided into three groups for functional testing. These are:

A. On-Off sensors that provide a scram trip function.

B. Analog devices coupled with bistable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as STARTUP or SHUTDOWN, or for which the only practical test is one that can be performed at shutdown.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.99999 probability of success (at the 50 percent confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95 percent confidence level is proposed. With the (1-out-of-2) X (2) logic, this requires that each sensor have an availability of 0.993 at the 95 percent confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history.[1]

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1. Reliability of Engineered Safety Features as a Function of Testing Frequency, I. M. Jacobs, "Nuclear Safety," Vol. 9, No. 4, July-August, 1968, pp. 310-312.

4.1 BASES (Cont'd)

~~The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. There are several instruments which must be calibrated and it will take several hours to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRMs resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.~~

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during STARTUP and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to SHUTDOWN or STARTUP: i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4 percent/month; i.e., in the period of a month a drift of 4 percent would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have been included in the latter table. These are: mode switch in SHUTDOWN and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

Insert J

INSERT J:

The APRM and 2-out-of-4 voter channel hardware is provided with a self-test capability which automatically checks most of the critical hardware at least once per 15 minute interval whenever the APRM channel is in the operate mode. This provides a virtually continuous monitoring of the essential APRM trip functions. In the event a critical fault is detected, an "inoperative" trip signal results. A fault detected in non-critical hardware results in an "inoperative" alarm. Following receipt of an "inoperative" trip or alarm signal, the operator can employ numerous diagnostic testing options to locate the problem.

The automatic self-test function is supplemented with a manual APRM trip functional test, including the 2-out-of-4 voter channels and the interface with the RPS trip systems. In combination with the virtually continuous self-testing, the manual APRM trip functional test provides adequate functional testing of the APRM trip function. Therefore, the six-month test frequency for the manual testing provides an acceptable level of availability of the APRM.

In addition to the above tests, the 2-out-of-4 voter is used to test the RPS scram contactors. The output of each voter channel is tripped to produce a scram signal into each of the RPS trip system channels (A1, A2, B1 and B2) to individually operate the respective scram contactors. The weekly test interval provides an acceptable level of availability of the scram contactors.

Each APRM receives the output signals from two analog differential pressure flow transducers, one associated with recirculation loop A and the other with recirculation loop B. These differential pressure signals are converted into representative digital loop flow signals within the same hardware that performs the APRM functions and are added to determine a total recirculation flow. The total recirculation flow value is used by the APRM to determine the flow biased setpoints. Each total recirculation flow signal developed by an APRM is compared in the hardware that performs the RBM functions to the signals from the remaining three APRMs. An alarm is given if a preset compare level setpoint is exceeded. The flow processing is integrated with the APRM processing and is covered by the same self-test and alarm functions described earlier. As a result of the virtually continuous monitoring of the equipment performing the flow processing, and the automatic comparison of redundant flow signals, it is acceptable to calibrate this equipment once per operating cycle.

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The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of ~~CMFLPD,~~ CPR~~,~~ and APLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
3 ~~4~~(1)	APRM Upscale (Flow Bias)	(2)
3 ~~4~~(1)	APRM Upscale (Startup Mode) (8)	≤12%
3 ~~4~~(1)	APRM Downscale (9)	≥3%
3 ~~4~~(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (~~Flow~~ POWER Bias)	~~[illegible]~~
2(7)	RBM Downscale (9) (13)	~~≥3%~~ (15)
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.
	Low Power Range (13)	(14)
	Intermediate Power Range (13)	(14)
	High Power Range (13)	(14)

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, OR APRM 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.

   INSERT K

   d. ~~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.

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

INSERT K:

7.c. The RBM need not be OPERABLE if either of the following two conditions is met:

(1) Reactor thermal power is $\geq$90 percent of rated and MCPR is $\geq$1.40, or

(2) Reactor thermal power is <90 percent of rated and MCPR is $\geq$1.70.

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

        THE REQUIRED MINIMUM NUMBER OF

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

        (2) Less than ~~14~~ LPRM inputs.

        (3) ~~Circuit boards not in circuit.~~ APRM MODULE UNPLUGGED

        (4) SELF-TEST DETECTED CRITICAL FAULT.

    c. RBM

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

        (2) ~~Circuit boards not in circuit.~~ RBM MODULE UNPLUGGED

        (3) RBM fails to null.

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

        (5) SELF-TEST DETECTED CRITICAL FAULT.

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

INSERT L

INSERT L:

13. The RBM rod block trip setpoints and applicable power ranges are specified in the CORE OPERATING LIMITS REPORT (COLR).

14. Less than or equal to the setpoint allowable value specified in the COLR.

15. Greater than or equal to the setpoint allowable value specified in the COLR.

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test		Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1)	(13)	once/~~3 months~~ OPERATING CYCLE	once/day (8)
APRM Upscale (Startup Mode)	(1)	(13)	once/~~3 months~~	once/day (8)
APRM Downscale	(1)	(13)	once/~~3 months~~	once/day (8)
APRM Inoperative	(1)	(13)	N/A	once/day (8)
RBM Upscale (~~Flow~~ POWER Bias)	(1)	(13)	once/~~6 months~~	~~once/day (8)~~ N/A
RBM Downscale	(1)	(13)	once/~~6 months~~	~~once/day (8)~~
RBM Inoperative	(1)	(13)	N/A	~~once/day (8)~~
IRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)		once/operating cycle (12)	N/A
IRM Inoperative	(1)(2)	(13)	N/A	N/A
SRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
SRM Inoperative	(1)(2)	(13)	N/A	N/A
~~Flow Bias Comparator~~	~~(1)(15)~~		~~once/operating cycle (20)~~	~~N/A~~
~~Flow Bias Upscale~~	~~(1)(15)~~		~~once/3 months~~	~~N/A~~
Rod Block Logic	(16)		N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter		once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter		once/operating cycle	N/A

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NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2. D AND 4.2.K

1. FOR IRMS AND SRMS Functional tests shall be performed once per month. FOR APRMS AND RBMS FUNCTIONAL TESTS SHALL BE PERFORMED ONCE PER 6 MONTHS.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.

9. Calibration frequency shall be once/year.

10. Deleted

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

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14. (Deleted)

DELETED

15. ~~The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.~~

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

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20. ~~Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.~~

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. (Deleted)

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

3.2 BASES (Cont'd)

The control rod block functions are provided to generate a trip signal to block rod withdrawal if the monitored power level exceeds a preset value. 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.

FOUR

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 provides a trip signal for blocking rod withdrawal when average reactor thermal power exceeds pre-established limits set to prevent scram actuation.

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 adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the reactor zone ventilation exhaust ducts and in the refueling zone.

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The conclusions to be drawn are these:

1. A 1-out-of-n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the reactor and refueling zones which initiate building isolation and standby gas treatment operation are arranged such that two sensors high (above the high level setpoint) in a single channel or one sensor downscale (below low level setpoint) or inoperable in two channels in the same zone will initiate a trip function. The functional testing frequencies for both the channel functional test and the high voltage power supply functional test are based on a Probabilistic Risk Assessment and system drift characteristics of the Reactor Building Ventilation Radiation Monitors. The calibration frequency is based upon the drift characteristics of the radiation monitors.

The automatic pressure relief instrumentation can be considered to be a 1-out-of-2 logic system and the discussion above applies also.

The RCIC and HPCI system logic tests required by Table 4.2.B contain provisions to demonstrate that these systems will automatically restart on a RPV low water level signal received subsequent to a RPV high water level trip.

INSERT M →

INSERT M:

The electronic instrumentation comprising the APRM rod block and Rod Block Monitor functions together with the recirculation flow instrumentation for flow bias purposes is monitored by the same self-test functions as applied to the APRM function for the RPS. The functional test frequency of every six months is based on this automatic self-test monitoring at 15 minute intervals and on the low expected equipment failure rates. Calibration frequency of once per operating cycle is based on the drift characteristics of the limited number of analog components, recognizing that most of the processing is performed digitally without drift of setpoint values.

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.~~ (Deleted)	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.~~ (Deleted)

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.I Average Planar Linear Heat Generation Rate

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

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

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

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

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

3.5.J Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the APPROPRIATE RATED, FLOW-DEPENDENT OR POWER-DEPENDENT 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 ≥ 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 OPERATING LIMIT ~~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	SURVEILLANCE REQUIREMENTS
3.5.K Minimum Critical Power Ratio (MCPR)	4.5.K Minimum Critical Power Ratio (MCPR)
	4.5.K.2 (Cont'd)
	b. 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	L. APRM Setpoints
(Deleted)	~~FRP/CMFLPD shall be determined daily when the reactor is ≥ 25% of rated thermal power.~~
~~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.~~	(Deleted)
~~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.~~	

3.5 BASES (Cont'd)

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

INSERT N

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

INSERT O

3.5.L. APRM Setpoints

~~The fuel cladding integrity safety limits of Section 2.1 were based on a total peaking factor within design limits (FRP/CMFLPD ≥ 1.0). The~~

(Deleted)

INSERT N:

At less than rated power conditions, the rated APLHGR limit is adjusted by a power dependent correction factor, MAPFAC(P). At less than rated flow conditions, the rated APLHGR limit is adjusted by a flow dependent correction factor, MAPFAC(F). The most limiting power-adjusted or flow-adjusted value is taken as the APLHGR operating limit for the off-rated condition.

The flow dependent correction factor, MAPFAC(F), applied to the rated APLHGR limit assures that (1) the 10 CFR 50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions. MAPFAC(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent correction factor, MAPFAC(P), applied to the rated APLHGR limit assures that the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated power conditions. MAPFAC(P) values are provided in the CORE OPERATING LIMITS REPORT.

INSERT O:

At less than rated power conditions, a power dependent MCPR operating limit, MCPR(P), is applicable. At less than rated flow conditions, a flow dependent MCPR operating limit, MCPR(F), is applicable. The most limiting power dependent or flow dependent value is taken as the MCPR operating limit for the off-rated condition.

The flow dependent limit, MCPR(F), provides the thermal margin required to protect the fuel from transients resulting from inadvertent core flow increases. MCPR(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent limit, MCPR(P), protects the fuel from the other limiting abnormal operating transients, including localized events such as a rod withdrawal error. MCPR(P) values are provided in the CORE OPERATING LIMITS REPORT.

~~APRM instruments must be adjusted to ensure that the core thermal limits are not exceeded in a degraded situation when entry conditions are less conservative than design assumptions.~~

3.5.M. Core Thermal-Hydraulic Stability

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

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

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

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

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

3.5.N. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEIM-10735, August 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

6.9.1.5 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

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

6.9.1.6 SOURCE TESTS

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

6.9.1.7 CORE OPERATING LIMITS REPORT

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

Revise per Insert P

(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, and Table 3.2.C~~,~~ ~~and Specification 3.5.L~~

Revise per Insert P

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

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

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

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

6.9.1.8 THE ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

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

INSERT P:

(1) The rated APLHGR limit; the Flow Dependent APLHGR Factor, MAPFAC(F); and the Power Dependent APLHGR Factor, MAPFAC(P) for Specification 3.5.I.

(2) The LGHR limit for Specification 3.5.J.

(3) The rated MCPR Operating Limit; the Flow Dependent MCPR Operating Limit, MCPR(F); and the Power Dependent MCPR Operating Limit, MCPR(P) for Specification 3.5.K/4.5.K.

(4) The APRM flow biased rod block trip setting for Specification 2.1.A.1.c and Table 3.2.C.

(5) The RBM downscale trip setpoint, high power trip setpoint, intermediate power trip setpoint, low power trip setpoint, and applicable reactor thermal power ranges for each of the setpoints for Table 3.2.C.

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.

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.

(Deleted)

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

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

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

V. Instrumentation

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

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

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

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

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

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

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

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

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

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

Specifications

A. Thermal Power Limits

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

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

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Specifications

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

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

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

1.1/2.1 FUEL CLADDING INTEGRITY

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

2.1.A Neutron Flux Trip Settings

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

(0.66W + 71%)

S ≤ ~~(0.58W + 62%)~~

where:

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

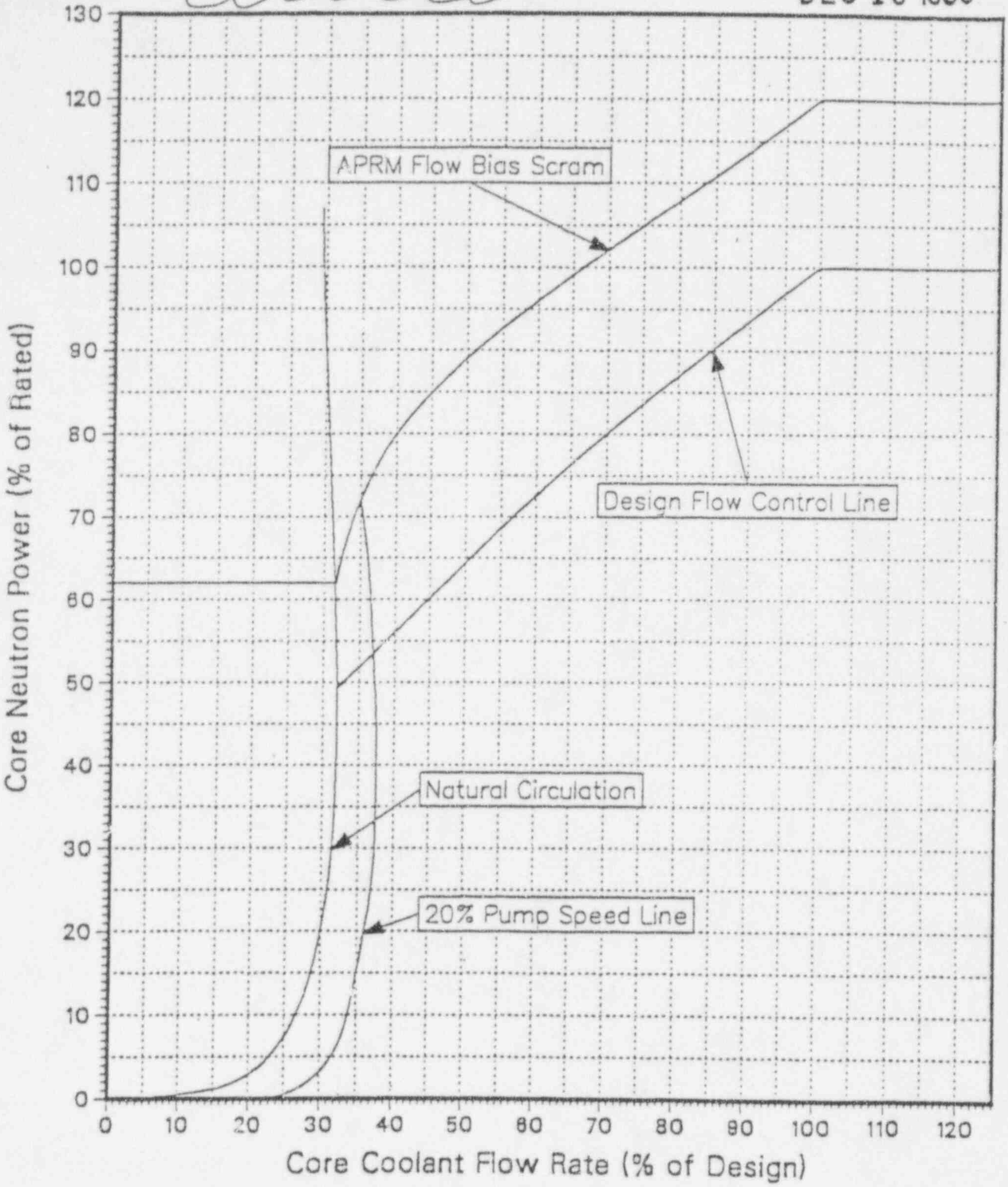
W = Loop recirculation flow rate in percent of rated

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.

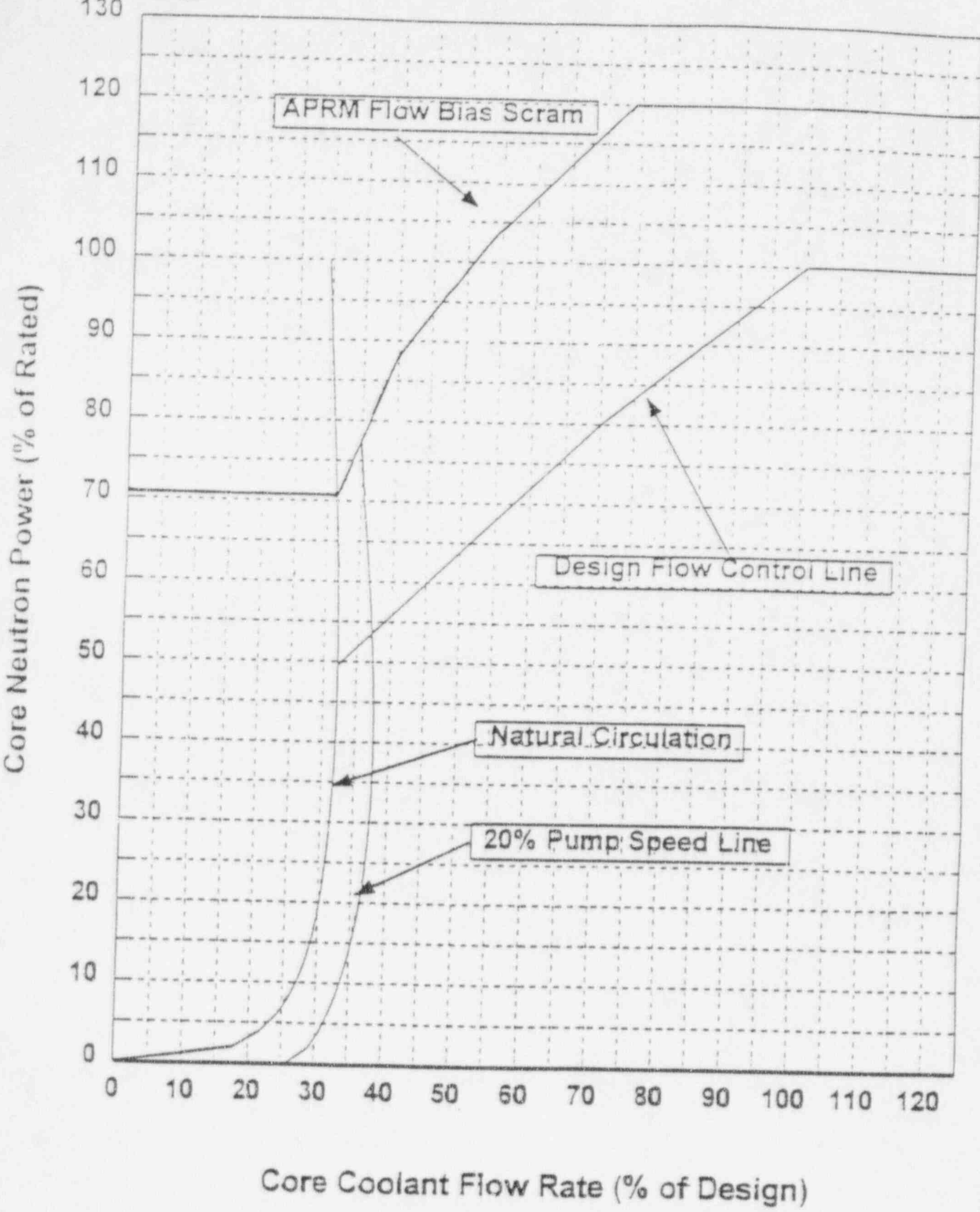
REPLACE WITH NEW Figure

DEC 18 1990



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

New



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

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

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

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

L. References

1. Supplemental Reload Licensing Report of Browns Ferry Nuclear Plant, Unit 2 (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. Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3, NEDC-32433P.

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Shut-down	Modes in which Function Must Be Operable Refuel (7)	Startup/ Hot Standby	Run	Action (1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	(21) X(22)	X	(5)	1.A
3	Inoperable			X	X	(5)	1.A
	APRM (16)(24)(25)						
~~2~~ 3(11)	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B or 1.E
~~2~~ 3(11)	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B or 1.E
~~2~~ 3(11)	High Flux	≤ 15% rated power		~~X(21)~~	X(17)	(15)	1.A
~~2~~ 3(11)	Inoperative	(13)		~~X(21)~~	X(17)	X	1.A
2	~~Downscale~~ 2/4 Trip Voter	~~≥3 Indicated on Scale~~ (12)		~~(11)~~	~~(11)~~ X	X~~(12)~~	1.A ~~or 1.B~~ or 1.F
2	High Reactor Pressure (PIS-3-22AA,BB,C,D)	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14) (PIS-64-56 A-D)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14) (LIS-3-203 A-D)	≥ 538" above vessel zero		X	X	X	1.A

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NOTES FOR TABLE 3.1.A

1. There shall be two OPERABLE or tripped trip systems for each function. If the minimum number of OPERABLE instrument channels per trip system cannot be met for one trip system, trip the INOPERABLE channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of OPERABLE instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An INOPERABLE channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the INOPERABLE channel shall be restored to OPERABLE status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of OPERABLE rods and complete insertion of all OPERABLE rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all OPERABLE control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the STARTUP/HOT Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

INSERT B

   D. Reduce power to less than 30 percent of rated.

2. Scram discharge volume high bypass may be used in SHUTDOWN or REFUEL to bypass scram discharge volume scram and scram pilot air header low pressure scram with control rod block for reactor protection system reset.

3. (Deleted)

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the RUN position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be OPERABLE:

   A. Mode switch in SHUTDOWN

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. ~~APRM 15 percent scram~~ (deleted)

   F. Scram pilot air header low pressure

INSERT B:

E. For the APRM functions only, if only two APRM channels are OPERABLE, restore a third APRM channel to OPERABLE status or trip one of the inoperable APRM channels within 6 hours. If only one APRM channel is OPERABLE, trip one inoperable APRM channel immediately and restore an inoperable APRM channel to OPERABLE status or initiate alternative action within 2 hours.

F. For the APRM functions only, if one voter channel is inoperable in one trip system, restore the voter channel to OPERABLE status or trip the inoperable channel or the entire trip system within 12 hours. If one voter channel is inoperable in both trip systems, restore the inoperable voter channels to OPERABLE status or initiate alternative action within 6 hours.

NOTES FOR TABLE 3.1.A (Cont'd) SEP 2 7 1994

8. Not required to be OPERABLE when primary containment integrity is not required.

9. (Deleted)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

11. ~~The APRM downscale trip function is only active when the reactor mode switch is in RUN.~~ Replace with Insert C

12. ~~The APRM downscale trip is automatically bypassed when the IRM instrumentation is OPERABLE and not high.~~ Replace with Insert D.

13. Less than ~~14~~ the required minimum number of OPERABLE LPRMs will cause ~~a trip system trip.~~ an instrument channel inoperative alarm.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first stage pressure is greater than or equal to 154 psig.

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. (Deleted)

21. ~~The APRM High Flux and Inoperative Trips do not have to be OPERABLE in the REFUEL Mode if the Source Range Monitors are connected to give a noncoincidence, High Flux scram,~~ at 5 x 10^5 cps. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the ~~Source Range Monitors~~ SRMs.

Replace with Insert E

INSERT C:

The same three (3) required APRM channels are shared by both RPS trip systems.

INSERT D:

Any combination of APRM upscale or inoperative trips from two different (non-bypassed) APRMs will trip all of the 2/4 voter units.

INSERT E:

In the REFUEL Mode unless adequate shutdown margin has been demonstrated per Specification 3.3.A.1, whenever any control rod is withdrawn from a core cell containing one or more fuel assemblies, shorting links shall be removed from the RPS circuitry to enable the Source Range Monitor (SRM) noncoincidence high-flux scram function.

BFN
Unit 2

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)	C	Trip Output Relays (4) (5) 2/4 Voter Logic (10)	~~Before Each Startup and Weekly When Required to be Operable~~ Every 6 months (4) Each Refueling Outage
High Flux (Flow Biased)	B	Trip Output Relays (4) (6) 2/4 Voter Logic (10)	~~Once/Week~~ Every 6 months Each Refueling Outage
High Flux (Fixed Trip)	B	Trip Output Relays (4) (5) 2/4 Voter Logic (10)	~~Once/Week~~ Every 6 months Each Refueling Outage
Inoperative	B	Trip Output Relays (4) (5) 2/4 Voter Logic (10)	~~Once/Week~~ Every 6 months Each Refueling Outage.
~~Downscale~~	~~B~~	~~Trip Output Relays (4)~~	~~Once/Week~~
~~Flow Bias~~	~~B~~	~~(6)~~	~~(6)~~
2/4 Trip Voter		Trip Scram Contactors (11)	Once/Week
High Reactor Pressure (PIS-3-22AA, BB, C, D)	B	Trip Channel and Alarm (7)	Once/Month
High Drywell Pressure (PIS-64-56 A-D)	B	Trip Channel and Alarm (7)	Once/Month
Reactor Low Water Level (LIS-3-203 A-D)	B	Trip Channel and Alarm (7)	Once/Month

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be OPERABLE or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an OPERABLE status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. ~~(Deleted)~~ Insert F

6. ~~The functional test of the flow bias network is performed in accordance with Table 4.2.C.~~ Replace with Insert G

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

8. The functional test frequency decreased to once every three months to reduce challenges to relief valves per NUREG 0737, Item II.K.3.16.

9. Not required to be performed when entering the STARTUP/HOT STANDBY Mode from RUN Mode until 12 hours after entering the STARTUP/HOT STANDBY Mode.

10. Functional test consists of simulating APRM trip conditions at the APRM channel outputs to check all combinations of two tripped inputs to the 2/4 voter logic in each voter channel.

11. Functional test consists of manually tripping the 2/4 voter trip output, one voter channel at a time, to demonstrate that each scram contactor for each RPS trip system channel (A1, A2, B1 and B2) operates and produces a half-scram.

INSERT F:

The channel functional test shall include both the APRM channels and the 2/4 voter channels.

INSERT G:

The channel functional test shall include both the APRM channels and the 2/4 voter channels plus the flow input function, excluding the flow transmitters.

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once/7 Days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure (PIS-3-22 AA, BB, C, D)	B	Standard Pressure Source	Once/6 Months (9)
High Drywell Pressure (PIS-64-56 A-D)	B	Standard Pressure Source	Once/18 Months (9)
Reactor Low Water Level (LIS-3-203 A-D)	B	Pressure Standard	Once/18 Months (9)
High Water Level in Scram Discharge Volume			
Float Switches (LS-85-45-C-F)	A	Calibrated Water Column	Once/18 Months
Electronic Level Switches (LS-85-45 A, B, G, H)	B	Calibrated Water Column	Once/18 Months (9)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive (PIS-1-81 A&B, PIS-1-91 A&B)	B	Standard Pressure Source	Once/18 Months (9)
Turbine Stop Valve Closure	A	Note (5)	Note (5)
Turbine Control Valve Fast Closure on Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Low Scram Pilot Air Header Pressure (PS 85-35 A1, A2, B1, & B2)	A	Standard Pressure Source	Once/18 Months

Indent

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NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. ~~The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operation during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.~~

Replace with Insert H

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

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

INSERT H:

The flow bias signal calibration will consist of calibrating the analog differential pressure flow sensors once per operating cycle. Calibration of the flow bias processing system is done once per operating cycle as part of the overall APRM instrumentation calibration.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made INOPERABLE for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between nonclass 1E power supply and the class 1E RPS bus. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of OPERABLE instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. ~~Three APRM instrument channels are provided for each protection trip system.~~

Insert I

INSERT I:

The APRM system is divided into four APRM channels and four 2-out-of-4 trip voter channels. Each APRM channel provides input to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The APRM system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter units, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system resulting in a full scram.

Each APRM instrument channel receives input signals from forty-three (43) Local Power Range Monitors (LPRMs). A minimum of twenty (20) LPRM inputs with three (3) per axial level is required for the APRM instrument channel to be OPERABLE. Fewer than the required minimum number of LPRM inputs generates an instrument channel inoperative alarm and a control rod block but does not result in an automatic trip input to the 2-out-of-4 voters.

IRM

Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. ~~Additional IRM channels have also been provided to allow for bypassing of one such channel.~~ The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharge water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Reference Section 7.5.4 FSAR. Thus, the IRM is required in the REFUEL and STARTUP modes. In the power range the APRM system provides required protection. Reference Section 7.5.7 FSAR. Thus, the IRM System is not required in the RUN mode. The APRMs and the IRMs provide adequate coverage in the STARTUP and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level, low scram pilot air header pressure and scram discharge volume high level scrams are required for STARTUP and RUN modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions as indicated in Table 3.1.A OPERABLE in the REFUEL mode is to assure that shifting to the REFUEL mode during reactor power operation does not diminish the need for the reactor protection system.

Because of the APRM downscale rod block limit of ≥ 3 percent when in the RUN mode and high level flux scram limit of ≤15 percent when in the STARTUP Mode, the transition between the STARTUP and RUN Modes must be made with the APRM instrumentation indicating between 3 percent and 15 percent of rated power ~~or a control rod scram will occur~~. In addition, the IRM system must be indicating below the High Flux setting (120/125 of scale) or a scram will occur when in the STARTUP Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to SHUTDOWN). When power is being reduced, if a transfer to the STARTUP mode is made and the IRMs have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

The low scram pilot air header pressure trip performs the same function as the high water level in the scram discharge instrument volume for fast fill events in which the high level instrument response time may be inadequate. A fast fill event is postulated for certain degraded control air events in which the scram outlet valves unseat enough to allow 5 gpm per drive leakage into the scram discharge volume but not enough to cause control rod insertion.

4.1 BASES

Except for the APRMs which take credit for self-test capability,

The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (1). This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failure such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1.A and 4.1.B are divided into three groups for functional testing. These are:

A. On-Off sensors that provide a scram trip function.

B. Analog devices coupled with bistable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as STARTUP or SHUTDOWN, or for which the only practical test is one that can be performed at SHUTDOWN.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.9999 probability of success (at the 50 percent confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95 percent confidence level is proposed. With the (1-out-of-2) X (2) logic, this requires that each sensor have an availability of 0.993 at the 95 percent confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history.[1]

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1. Reliability of Engineered Safety Features as a Function of Testing Frequency, I. M. Jacobs, "Nuclear Safety," Vol. 9, No. 4, July-August, 1968, pp. 310-312.

4.1 BASES (Cont'd)

~~The frequency of calibration of the APRM Flow Biasing Network has been established at each refueling outage. There are several instruments which must be calibrated and it will take several hours to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRMS resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.~~

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during STARTUP and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to SHUTDOWN or STARTUP: i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4 percent/month; i.e., in the period of a month a drift of 4 percent would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have been included in the latter table. These are: mode switch in SHUTDOWN and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

Insert J

INSERT J:

The APRM and 2-out-of-4 voter channel hardware is provided with a self-test capability which automatically checks most of the critical hardware at least once per 15 minute interval whenever the APRM channel is in the operate mode. This provides a virtually continuous monitoring of the essential APRM trip functions. In the event a critical fault is detected, an "inoperative" trip signal results. A fault detected in non-critical hardware results in an "inoperative" alarm. Following receipt of an "inoperative" trip or alarm signal, the operator can employ numerous diagnostic testing options to locate the problem.

The automatic self-test function is supplemented with a manual APRM trip functional test, including the 2-out-of-4 voter channels and the interface with the RPS trip systems. In combination with the virtually continuous self-testing, the manual APRM trip functional test provides adequate functional testing of the APRM trip function. Therefore, the six-month test frequency for the manual testing provides an acceptable level of availability of the APRM.

In addition to the above tests, the 2-out-of-4 voter is used to test the RPS scram contactors. The output of each voter channel is tripped to produce a scram signal into each of the RPS trip system channels (A1, A2, B1 and B2) to individually operate the respective scram contactors. The weekly test interval provides an acceptable level of availability of the scram contactors.

Each APRM receives the output signals from two analog differential pressure flow transducers, one associated with recirculation loop A and the other with recirculation loop B. These differential pressure signals are converted into representative digital loop flow signals within the same hardware that performs the APRM functions and are added to determine a total recirculation flow. The total recirculation flow value is used by the APRM to determine the flow biased setpoints. Each total recirculation flow signal developed by an APRM is compared in the hardware that performs the RBM functions to the signals from the remaining three APRMs. An alarm is given if a preset compare level setpoint is exceeded. The flow processing is integrated with the APRM processing and is covered by the same self-test and alarm functions described earlier. As a result of the virtually continuous monitoring of the equipment performing the flow processing, and the automatic comparison of redundant flow signals, it is acceptable to calibrate this equipment once per operating cycle.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of ~~CMFLPD,~~ CPR~~,~~ and APLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
3 ~~4~~(1)	APRM Upscale (Flow Bias)	(2)
3 ~~4~~(1)	APRM Upscale (Startup Mode) (8)	≤12%
3 ~~4~~(1)	APRM Downscale (9)	≥3%
3 ~~4~~(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (~~Flow~~ Power Bias)	[illegible]
2(7)	RBM Downscale (9) (13)	[illegible] (15) .
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.

Low Power Range (13)	(14)
Intermediate Power Range (13)	(14)
High Power Range (13)	(14)

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NOTES FOR TABLE 3.2.C

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

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

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

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

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

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

5. During repair or calibration of equipment, not more than one SRM or RBM or APRM 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.

   Insert k

   d ~~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.

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

INSERT K:

7.c. The RBM need not be OPERABLE if either of the following two conditions is met:

(1) Reactor thermal power is $\geq$90 percent of rated and MCPR is $\geq$1.40, or

(2) Reactor thermal power is <90 percent of rated and MCPR is $\geq$1.70.

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

        THE REQUIRED MINIMUM NUMBER OF

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

        (2) Less than ~~14~~ LPRM inputs.

        (3) ~~Circuit boards not in circuit.~~ APRM module unplugged.

        (4) Self test detected critical fault.

    c. RBM

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

        (2) ~~Circuit boards not in circuit.~~ RBM module unplugged.

        (3) RBM fails to null.

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

        (5) Self test detected critical fault.

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

Insert L

INSERT L:

13. The RBM rod block trip setpoints and applicable power ranges are specified in the CORE OPERATING LIMITS REPORT (COLR).

14. Less than or equal to the setpoint allowable value specified in the COLR.

15. Greater than or equal to the setpoint allowable value specified in the COLR.

BFN
Unit 2

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test		Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1)	(13)	once/3 months Operating cycle	once/day (8)
APRM Upscale (Startup Mode)	(1)	(13)	once/3 months	once/day (8)
APRM Downscale	(1)	(13)	once/3 months	once/day (8)
APRM Inoperative	(1)	(13)	N/A	once/day (8)
RBM Upscale (~~Flow~~ Power Bias)	(1)	(13)	once/6 months	~~once/day (8)~~ N/A
RBM Downscale	(1)	(13)	once/6 months	~~once/day (8)~~
RBM Inoperative	(1)	(13)	N/A	~~once/day (8)~~
IRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)		once/operating cycle (12)	N/A
IRM Inoperative	(1)(2)	(13)	N/A	N/A
SRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
SRM Inoperative	(1)(2)	(13)	N/A	N/A
~~Flow Bias Comparator~~	~~(1)(15)~~		~~once/operating cycle (20)~~	~~N/A~~
~~Flow Bias Upscale~~	~~(1)(15)~~		~~once/3 months~~	~~N/A~~
Rod Block Logic	(16)		N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter		once/18 months	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter		once/18 months	N/A

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NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.K JAN 26 1989

1. (For IRMs and SRMs) Functional tests shall be performed once per month. For APRMs and RBMs functional tests shall be performed once per 6 months.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.

9. Calibration frequency shall be once/year.

10. Deleted

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.K (Cont'd)

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14. (Deleted)

(Deleted)

15. ~~The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.~~

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

(Deleted)

20. ~~Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.~~

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. (Deleted)

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

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 generate a trip signal to block rod withdrawal if the monitored power level exceeds a preset value. 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.

four

When the RBM is required, the minimum instrument channel requirements apply. These 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 provides a trip signal for blocking rod withdrawal when average reactor thermal power exceeds pre-established limits set to prevent scram actuation.

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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The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1-out-of-n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the reactor and refueling zones which initiate building isolation and standby gas treatment operation are arranged such that two sensors high (above the high level setpoint) in a single channel or one sensor downscale (below low level setpoint) or inoperable in two channels in the same zone will initiate a trip function. The functional testing frequencies for both the channel functional test and the high voltage power supply functional test are based on a Probabilistic Risk Assessment and system drift characteristics of the Reactor Building Ventilation Radiation Monitors. The calibration frequency is based upon the drift characteristics of the radiation monitors.

The automatic pressure relief instrumentation can be considered to be a 1-out-of-2 logic system and the discussion above applies also.

The RCIC and HPCI system logic tests required by Table 4.2.B contain provisions to demonstrate that these systems will automatically restart on a RPV low water level signal received subsequent to a RPV high water level trip.

INSERT M →

INSERT M:

The electronic instrumentation comprising the APRM rod block and Rod Block Monitor functions together with the recirculation flow instrumentation for flow bias purposes is monitored by the same self-test functions as applied to the APRM function for the RPS. The functional test frequency of every six months is based on this automatic self-test monitoring at 15 minute intervals and on the low expected equipment failure rates. Calibration frequency of once per operating cycle is based on the drift characteristics of the limited number of analog components, recognizing that most of the processing is performed digitally without drift of setpoint values.

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

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.I Average Planar Linear Heat Generation Rate

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

rated, flow-dependent or power-dependent

J. Linear Heat Generation Rate (LHGR)

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

SURVEILLANCE REQUIREMENTS

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

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

J. Linear Heat Generation Rate (LHGR)

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

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LIMITING CONDITIONS FOR OPERATION

J. Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

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

3.5.K Minimum Critical Power Ratio (MCPR)

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

SURVEILLANCE REQUIREMENTS

J. Linear Heat Generation Rate (LHGR)

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at ≥ 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 operating limit 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	SURVEILLANCE REQUIREMENTS
3.5 Core and Containment Cooling Systems	4.5 Core and Containment Cooling Systems
L. APRM Setpoints	L. APRM Setpoints
~~1. Whenever the core thermal power is ≥ 25% of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0, or the APRM scram setpoint equation listed in Section 2.1.A and the APRM rod block setpoint equation listed in the CORE OPERATING LIMITS REPORT shall be multiplied by FRP/CMFLPD.~~	~~FRP/CMFLPD shall be determined daily when the reactor is ≥ 25% of rated thermal power.~~
~~2. When it is determined that 3.5.L.1 is not being met, 6 hours is allowed to correct the condition.~~	(Deleted)
~~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.~~	
(Deleted)	
M. Core Thermal-Hydraulic Stability	M. Core Thermal-Hydraulic Stability
1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.	1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:
2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.	a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and
3. If Region II of Figure 3.5.M-1 is entered:	b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

3.5 BASES (Cont'd)

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

Insert N

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at $\geq$ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

Insert O

3.5.L. APRM Setpoints

~~Operation is constrained to the LHGR limit of Specification 3.5.J. This limit is reached when core maximum fraction of limiting power~~

(Deleted)

INSERT N:

At less than rated power conditions, the rated APLHGR limit is adjusted by a power dependent correction factor, MAPFAC(P). At less than rated flow conditions, the rated APLHGR limit is adjusted by a flow dependent correction factor, MAPFAC(F). The most limiting power-adjusted or flow-adjusted value is taken as the APLHGR operating limit for the off-rated condition.

The flow dependent correction factor, MAPFAC(F), applied to the rated APLHGR limit assures that (1) the 10 CFR 50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions. MAPFAC(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent correction factor, MAPFAC(P), applied to the rated APLHGR limit assures that the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated power conditions. MAPFAC(P) values are provided in the CORE OPERATING LIMITS REPORT.

INSERT O:

At less than rated power conditions, a power dependent MCPR operating limit, MCPR(P), is applicable. At less than rated flow conditions, a flow dependent MCPR operating limit, MCPR(F), is applicable. The most limiting power dependent or flow dependent value is taken as the MCPR operating limit for the off-rated condition.

The flow dependent limit, MCPR(F), provides the thermal margin required to protect the fuel from transients resulting from inadvertent core flow increases. MCPR(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent limit, MCPR(P), protects the fuel from the other limiting abnormal operating transients, including localized events such as a rod withdrawal error. MCPR(P) values are provided in the CORE OPERATING LIMITS REPORT.

density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than 100-percent rated power and only with APRM scram settings as required by Specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the 1-percent plastic strain limit. A 6-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.

3.5 M. <u>Core Thermal-Hydraulic Stability</u>

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

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

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

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

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

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:

Revise per Insert P

(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~~,~~ and 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).

INSERT P:

(1) The rated APLHGR limit; the Flow Dependent APLHGR Factor, MAPFAC(F); and the Power Dependent APLHGR Factor, MAPFAC(P) for Specification 3.5.I.

(2) The LGHR limit for Specification 3.5.J.

(3) The rated MCPR Operating Limit; the Flow Dependent MCPR Operating Limit, MCPR(F); and the Power Dependent MCPR Operating Limit, MCPR(P) for Specification 3.5.K/4.5.K.

(4) The APRM flow biased rod block trip setting for Specification 2.1.A.1.c and Table 3.2.C.

(5) The RBM downscale trip setpoint, high power trip setpoint, intermediate power trip setpoint, low power trip setpoint, and applicable reactor thermal power ranges for each of the setpoints for Table 3.2.C.

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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.

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.

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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.

~~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.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

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

Specification

A. Thermal Power Limits

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

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

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

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

Objective

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

Specification

The limiting safety system settings shall be as specified below:

A. Neutron Flux Trip Settings

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

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

SAFETY LIMIT

LIMITING SAFETY SYSTEM SETTING

2.1.A Neutron Flux Trip Settings

2.1.A.1.a (Cont'd) (0.66W + 71%)

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

where:

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

W = Loop recirculation flow rate in percent of rated

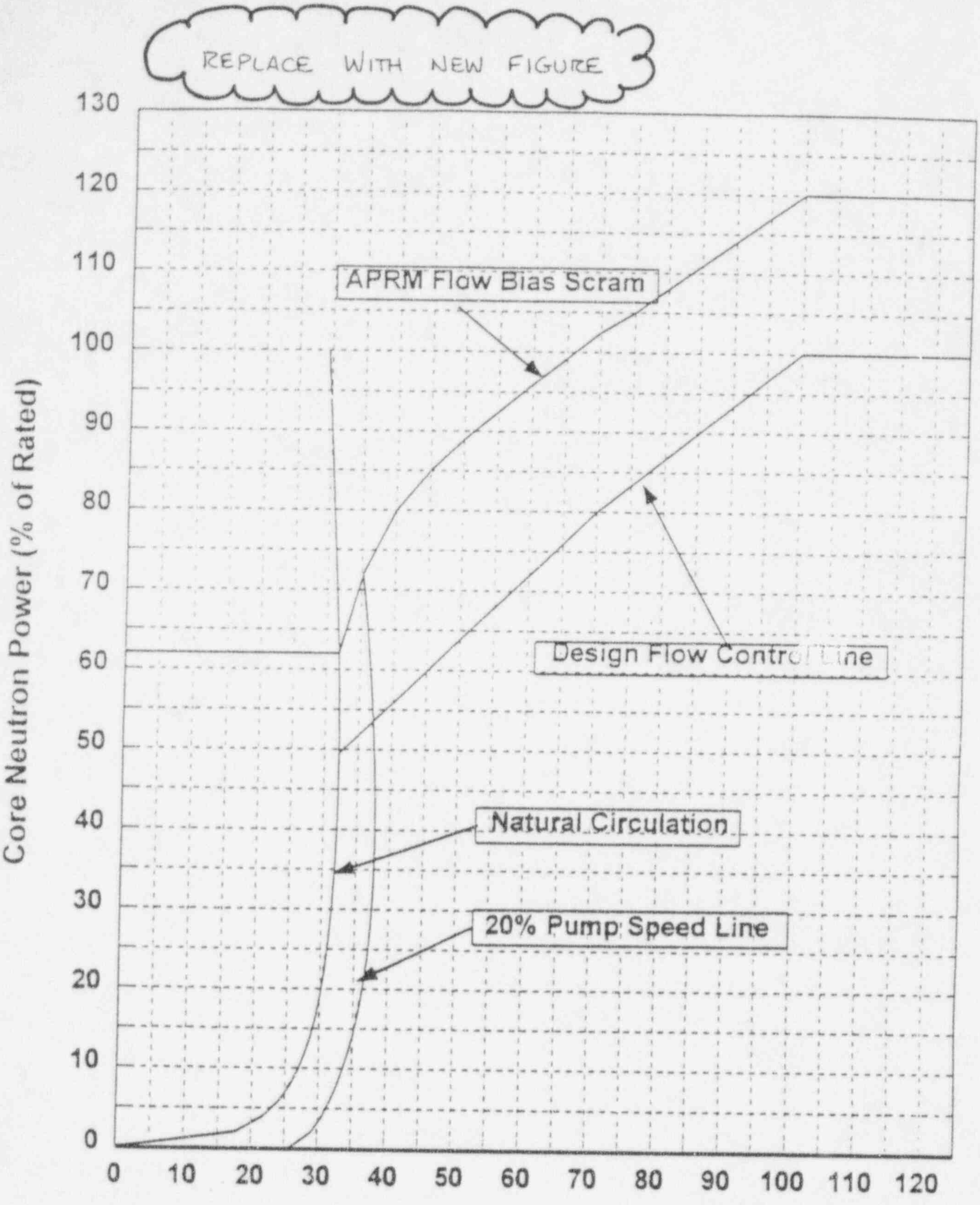
b. For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

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.~~

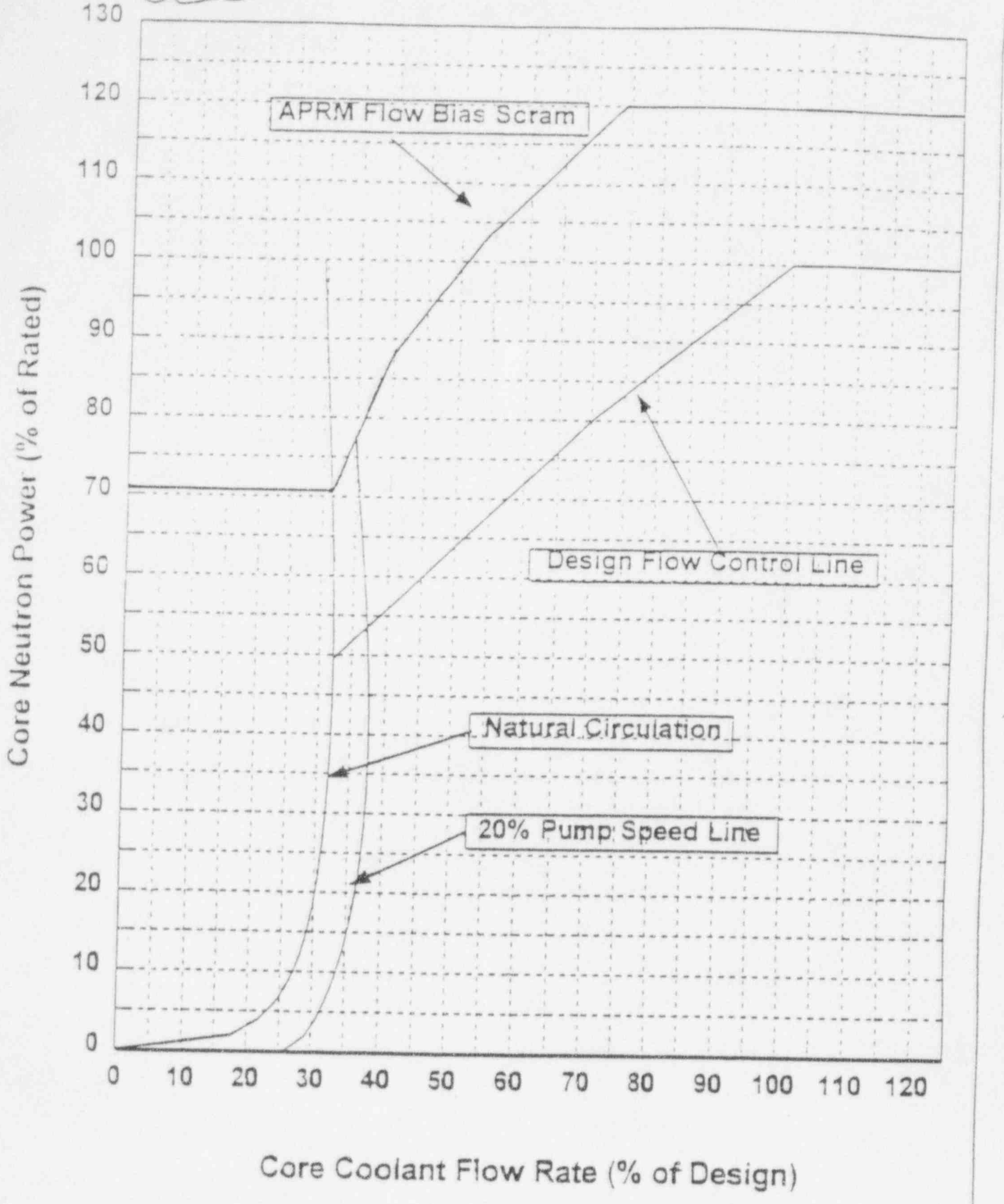
c. The APRM Rod Block trip setting shall be less than or equal to the limit specified in the CORE OPERATING LIMITS REPORT.



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

New



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

MAY 2 0 1993

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.

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).

3. MAXIMUM EXTENDED LOAD LINE LIMIT AND ARTS IMPROVEMENT PROGRAM ANALYSES FOR BROWNS FERRY NUCLEAR PLANT UNIT 1, 2 AND 3, NEDC-32433P.

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Shut-down	Refuel (7)	Startup/ Hot Standby	Run	Action (1)
				Modes in Which Function Must Be Operable			
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(22) (21)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)(24)(25)						
~~2~~ 3(11)	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B or 1.E.
~~2~~ 3(11)	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B or 1.E
~~2~~ 3(11)	High Flux	≤ 15% rated power		~~X(21)~~	X(17)	(15)	1.A
~~2~~ 3(11)	Inoperative	(13)		~~X(21)~~	X(17)	X	1.A
2	~~Downscale~~ 2/4 Trip Voter	~~≥ 3/Indicated on Scale~~ (12)		~~(11)~~	~~X(11)~~ X	X ~~(12)~~	1.A ~~or 1.B~~ or 1.F
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A

Amendment No. 105
Corrected 8/24/87

JUL 17 1987

NOTES FOR TABLE 3.1.A

1. There shall be two OPERABLE or tripped trip systems for each function. If the minimum number of OPERABLE instrument channels per trip system cannot be met for one trip system, trip the INOPERABLE channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of OPERABLE instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An INOPERABLE channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the INOPERABLE channel shall be restored to OPERABLE status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of OPERABLE rods and complete insertion of all OPERABLE rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all OPERABLE control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the STARTUP/HOT STANDBY position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30 percent of rated.

INSERT B

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram with control rod block for reactor protection system reset.

3. DELETED

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the RUN position.

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be OPERABLE:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. ~~APRM 15 percent scram~~ (deleted)

INSERT B:

E. For the APRM functions only, if only two APRM channels are OPERABLE, restore a third APRM channel to OPERABLE status or trip one of the inoperable APRM channels within 6 hours. If only one APRM channel is OPERABLE, trip one inoperable APRM channel immediately and restore an inoperable APRM channel to OPERABLE status or initiate alternative action within 2 hours.

F. For the APRM functions only, if one voter channel is inoperable in one trip system, restore the voter channel to OPERABLE status or trip the inoperable channel or the entire trip system within 12 hours. If one voter channel is inoperable in both trip systems, restore the inoperable voter channels to OPERABLE status or initiate alternative action within 6 hours.

NOTES FOR TABLE 3.1.A (Cont'd) SEP 27 1994

8. Not required to be OPERABLE when primary containment integrity is not required.

9. (Deleted)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

11. ~~The APRM downscale trip function is only active when the reactor mode switch is in RUN.~~ Replace with Insert C

12. ~~The APRM downscale trip is automatically bypassed when the IRM instrumentation is OPERABLE and not high.~~ Replace with Insert D

13. Less than ~~14~~ the required minimum number 7 OPERABLE LPRMs will cause ~~a trip system trip.~~ an instrument channel inoperative alarm.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first stage pressure is greater than or equal to 154 psig.

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. (Deleted)

21. ~~The APRM High Flux and Inoperative Trips do not have to be OPERABLE in the REFUEL Mode if the Source Range Monitors are connected to give a noncoincidence, High Flux scram, at 5×10^5 cps.~~ The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the ~~Source Range Monitors.~~

Replace with Insert E

INSERT C:

The same three (3) required APRM channels are shared by both RPS trip systems.

INSERT D:

Any combination of APRM upscale or inoperative trips from two different (non-bypassed) APRMs will trip all of the 2/4 voter units.

INSERT E:

In the REFUEL Mode unless adequate shutdown margin has been demonstrated per Specification 3.3.A.1, whenever any control rod is withdrawn from a core cell containing one or more fuel assemblies, shorting links shall be removed from the RPS circuitry to enable the Source Range Monitor (SRM) noncoincidence high-flux scram function.

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(1)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)	~~C~~	Trip Output Relays (4) (5) 2/4 VOTER LOGIC (10)	~~Before Each Startup and Weekly When Required to be Operable~~ EVERY 6 MONTHS (9) EACH REFUELING OUTAGE
High Flux (Flow Biased)	~~B~~	Trip Output Relays (4) (6) 2/4 VOTER LOGIC (10)	~~Once/Week~~ EVERY 6 MONTHS EACH REFUELING OUTAGE
High Flux (Fixed Trip)	~~B~~	Trip Output Relays (4) (5) 2/4 VOTER LOGIC (10)	~~Once/Week~~ EVERY 6 MONTHS EACH REFUELING OUTAGE
Inoperative	~~B~~	Trip Output Relays (4) (5) 2/4 VOTER LOGIC (10)	~~Once/Week~~ EVERY 6 MONTHS EACH REFUELING OUTAGE
~~Downscale~~	~~B~~	~~Trip Output Relays (5)~~	~~Once/Week~~
~~Flow Bias~~	~~B~~	~~(6)~~	~~(6)~~
2/4 Trip Voter		Trip Scram Contactors (11)	Once/Week
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level	A	Trip Channel and Alarm	Once/Month (1)

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be OPERABLE or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an OPERABLE status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. ~~(Deleted)~~ Insert F

Replace with Insert G

6. ~~The functional test of the flow bias network is performed in accordance with Table 4.2.C.~~

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

8. Functional test frequency decreased to once/3 months to reduce the challenges to relief valves per NUREG 0737, Item II.K.3.16.

9. Not required to be performed when entering the STARTUP/HOT STANDBY Mode from RUN Mode until 12 hours after entering the STARTUP/HOT STANDBY Mode.

10. Functional test consists of simulating APRM trip conditions at the APRM channel outputs to check all combinations of two tripped inputs to the 2/4 voter logic in each voter channel.

11. Functional test consists of manually tripping the 2/4 voter trip output, one voter channel at a time, to demonstrate that each scram contactor for each RPS trip system channel (A1, A2, B1 and B2) operates and produces a half-scram.

INSERT F:

The channel functional test shall include both the APRM channels and the 2/4 voter channels.

INSERT G:

The channel functional test shall include both the APRM channels and the 2/4 voter channels plus the flow input function, excluding the flow transmitters.

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal	B	Heat Balance	Once Every 7 Days
Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal Indent	B	TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume			
Float Switches (LS-85-45C-F)	A	Calibrated Water Column (5)	Note (5)
Electronic Lvl Switches (LS-85-45-A, B, G, H)	B	Calibrated Water Column	Once/Operating Cycle (9)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 Months
Turbine Control Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	Note (5)	Note (5)

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NOTES FOR TABLE 4.1.B

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1. A description of three groups is included in the Bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. ~~The Flow Bias Signal Calibration will consist of calibrating the sensors, flow converters, and signal offset networks during each operating cycle. The instrumentation is an analog type with redundant flow signals that can be compared. The flow comparator trip and upscale will be functionally tested according to Table 4.2.C to ensure the proper operation during the operating cycle. Refer to 4.1 Bases for further explanation of calibration frequency.~~

REPLACE WITH INSERT H

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

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

INSERT H:

The flow bias signal calibration will consist of calibrating the analog differential pressure flow sensors once per operating cycle. Calibration of the flow bias processing system is done once per operating cycle as part of the overall APRM instrumentation calibration.

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made INOPERABLE for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of OPERABLE instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor. ~~Three APRM instrument channels are provided for each protection trip system.~~

INSERT I

INSERT I:

The APRM system is divided into four APRM channels and four 2-out-of-4 trip voter channels. Each APRM channel provides input to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The APRM system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter units, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system resulting in a full scram.

Each APRM instrument channel receives input signals from forty-three (43) Local Power Range Monitors (LPRMs). A minimum of twenty (20) LPRM inputs with three (3) per axial level is required for the APRM instrument channel to be OPERABLE. Fewer than the required minimum number of LPRM inputs generates an instrument channel inoperative alarm and a control rod block but does not result in an automatic trip input to the 2-out-of-4 voters.

Each protection trip system has one more ~~APRM~~ IRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one ~~APRM~~ IRM per protection trip system for maintenance, testing or calibration. ~~Additional IRM channels have also been provided to allow for bypassing of one such channel.~~ The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

3.1 <u>BASES</u> (Cont'd)

be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharge water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Reference Section 7.5.4 FSAR. Thus, the IRM is required in the REFUEL and STARTUP modes. In the power range the APRM system provides required protection. Reference Section 7.5.7 FSAR. Thus, the IRM System is not required in the RUN mode. The APRMs and the IRMs provide adequate coverage in the STARTUP and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for STARTUP and RUN modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions as indicated in Table 3.1.1 OPERABLE in the REFUEL mode is to assure that shifting to the REFUEL mode during reactor power operation does not diminish the need for the reactor protection system.

Because of the APRM downscale rod block, limit of ≥ 3 percent when in the RUN mode and high level flux scram limit of ≤15 percent when in the STARTUP Mode, the transition between the STARTUP and RUN Modes must be made with the APRM instrumentation indicating between 3 percent and 15 percent of rated power ~~or a control rod scram will occur~~. In addition, the IRM system must be indicating below the High Flux setting (120/125 of scale) or a scram will occur when in the STARTUP Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to shutdown). When power is being reduced, if a transfer to the STARTUP mode is made and the IRMs have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

Except for the APRMs which take credit for self-test capability.

4.1 BASES

The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (1). This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failure such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

The channels listed in Tables 4.1.A and 4.1.B are divided into three groups for functional testing. These are:

A. On-Off sensors that provide a scram trip function.

B. Analog devices coupled with bistable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as STARTUP or SHUTDOWN, or for which the only practical test is one that can be performed at shutdown.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.99999 probability of success (at the 50 percent confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95-percent confidence level is proposed. With the (1-out-of-2) X (2) logic, this requires that each sensor have an availability of 0.993 at the 95 percent confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history.[1]

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1. Reliability of Engineered Safety Features as a Function of Testing Frequency, I. M. Jacobs, "Nuclear Safety," Vol. 9, No. 4, July-August, 1968, pp. 310-312.

4.1 BASES (Cont'd)

~~The frequency of calibration of the APRM Flow Biasing Network has been established at each refueling outage. There are several instruments which must be calibrated and it will take several hours to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRMs resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.~~

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during STARTUP and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to SHUTDOWN or STARTUP; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4 percent/month; i.e., in the period of a month a drift of .4-percent would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have been included in the latter table. These are: mode switch in SHUTDOWN and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

Insert J

INSERT J:

The APRM and 2-out-of-4 voter channel hardware is provided with a self-test capability which automatically checks most of the critical hardware at least once per 15 minute interval whenever the APRM channel is in the operate mode. This provides a virtually continuous monitoring of the essential APRM trip functions. In the event a critical fault is detected, an "inoperative" trip signal results. A fault detected in non-critical hardware results in an "inoperative" alarm. Following receipt of an "inoperative" trip or alarm signal, the operator can employ numerous diagnostic testing options to locate the problem.

The automatic self-test function is supplemented with a manual APRM trip functional test, including the 2-out-of-4 voter channels and the interface with the RPS trip systems. In combination with the virtually continuous self-testing, the manual APRM trip functional test provides adequate functional testing of the APRM trip function. Therefore, the six-month test frequency for the manual testing provides an acceptable level of availability of the APRM.

In addition to the above tests, the 2-out-of-4 voter is used to test the RPS scram contactors. The output of each voter channel is tripped to produce a scram signal into each of the RPS trip system channels (A1, A2, B1 and B2) to individually operate the respective scram contactors. The weekly test interval provides an acceptable level of availability of the scram contactors.

Each APRM receives the output signals from two analog differential pressure flow transducers, one associated with recirculation loop A and the other with recirculation loop B. These differential pressure signals are converted into representative digital loop flow signals within the same hardware that performs the APRM functions and are added to determine a total recirculation flow. The total recirculation flow value is used by the APRM to determine the flow biased setpoints. Each total recirculation flow signal developed by an APRM is compared in the hardware that performs the RBM functions to the signals from the remaining three APRMs. An alarm is given if a preset compare level setpoint is exceeded. The flow processing is integrated with the APRM processing and is covered by the same self-test and alarm functions described earlier. As a result of the virtually continuous monitoring of the equipment performing the flow processing, and the automatic comparison of redundant flow signals, it is acceptable to calibrate this equipment once per operating cycle.

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The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CMFLPD, CPR, and APLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
3(1)	APRM Upscale (Flow Bias)	(2)
3(1)	APRM Upscale (Startup Mode) (8)	≤12%
3(1)	APRM Downscale (9)	≥3%
3(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (~~Flow~~ POWER Bias)	~~(13)~~
2(7)	RBM Downscale (9) (13)	~~≥3%~~ (15)
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.
	Low Power Range (13)	(14)
	Intermediate Power Range (13)	(14)
	High Power Range (13)	(14)

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 OR APRM 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.

   INSERT K

   d. ~~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.

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

INSERT K:

7.c. The RBM need not be OPERABLE if either of the following two conditions is met:

(1) Reactor thermal power is $\geq$90 percent of rated and MCPR is $\geq$1.40, or

(2) Reactor thermal power is <90 percent of rated and MCPR is $\geq$1.70.

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

        THE REQUIRED MINIMUM NUMBER OF

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

        (2) Less than ~~14~~ LPRM inputs.

        (3) ~~Circuit boards not in circuit.~~ APRM MODULE UNPLUGGED

        (4) SELF-TEST DETECTED CRITICAL FAULT.

    c. RBM

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

        (2) ~~Circuit boards not in circuit.~~ RBM MODULE UNPLUGGED

        (3) RBM fails to null.

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

        (5) SELF-TEST DETECTED CRITICAL FAULT.

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

INSERT L

INSERT L:

13. The RBM rod block trip setpoints and applicable power ranges are specified in the CORE OPERATING LIMITS REPORT (COLR).

14. Less than or equal to the setpoint allowable value specified in the COLR.

15. Greater than or equal to the setpoint allowable value specified in the COLR.

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test	Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1) (13)	once/~~3 months~~ OPERATING CYCLE	once/day (8)
APRM Upscale (Startup Mode)	(1) (13)	once/~~3 months~~	once/day (8)
APRM Downscale	(1) (13)	once/~~3 months~~	once/day (8)
APRM Inoperative	(1) (13)	N/A	once/day (8)
RBM Upscale (~~Flow~~ POWER Bias)	(1) (13)	once/~~6 months~~	~~once/day (8)~~ N/A
RBM Downscale	(1) (13)	once/~~6 months~~	~~once/day (8)~~
RBM Inoperative	(1) (13)	N/A	~~once/day (8)~~
IRM Upscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)	once/operating cycle (12)	N/A
IRM Inoperative	(1)(2) (13)	N/A	N/A
SRM Upscale	(1)(2) (15)	once/3 months	once/day (8)
SRM Downscale	(1)(2) (13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)	once/operating cycle (12)	N/A
SRM Inoperative	(1)(2) (13)	N/A	N/A
~~Flow Bias Comparator~~	~~(1)(15)~~	~~once/operating cycle (20)~~	~~N/A~~
~~Flow Bias Upscale~~	~~(1)(15)~~	~~once/3 months~~	~~N/A~~
Rod Block Logic	(16)	N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter	once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter	once/operating cycle	N/A

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NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.K JAN 26 1989

1. (FOR IRMS AND SRMS) Functional tests shall be performed once per month. FOR APRMS AND RBMS FUNCTIONAL TESTS SHALL BE PERFORMED ONCE PER 6 MONTHS.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be operable or are tripped.

9. Calibration frequency shall be once/year.

10. (DELETED)

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

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14. (Deleted)

15. ~~The flow bias comparator will be tested by putting one flow unit in "Test" (producing 1/2 scram) and adjusting the test input to obtain comparator rod block. The flow bias upscale will be verified by observing a local upscale trip light during operation and verified that it will produce a rod block during the operating cycle.~~

DELETED

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

20. ~~Calibration of the comparator requires the inputs from both recirculation loops to be interrupted, thereby removing the flow bias signal to the APRM and RBM and scramming the reactor. This calibration can only be performed during an outage.~~

DELETED

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. (Deleted)

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

3.2 <u>BASES</u> (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 generate a trip signal to block rod withdrawal if the monitored power level exceeds a preset value. 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.

four

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 provides a trip signal for blocking rod withdrawal when average reactor thermal power exceeds pre-established limits set to prevent scram actuation.

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 adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

The conclusions to be drawn are these:

1. A 1-out-of-n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the reactor and refueling zones which initiate building isolation and standby gas treatment operation are arranged such that two sensors high (above the high level setpoint) in a single channel or one sensor downscale (below low level setpoint) or inoperable in two channels in the same zone will initiate a trip function. The functional testing frequencies for both the channel functional test and the high voltage power supply functional test are based on a Probabilistic Risk Assessment and system drift characteristics of the Reactor Building Ventilation Radiation Monitors. The calibration frequency is based upon the drift characteristics of the radiation monitors.

The automatic pressure relief instrumentation can be considered to be a 1-out-of-2 logic system and the discussion above applies also.

The RCIC and HPCI system logic tests required by Table 4.2.B contain provisions to demonstrate that these systems will automatically restart on a RPV low water level signal received subsequent to a RPV high water level trip.

INSERT M

INSERT M:

The electronic instrumentation comprising the APRM rod block and Rod Block Monitor functions together with the recirculation flow instrumentation for flow bias purposes is monitored by the same self-test functions as applied to the APRM function for the RPS. The functional test frequency of every six months is based on this automatic self-test monitoring at 15 minute intervals and on the low expected equipment failure rates. Calibration frequency of once per operating cycle is based on the drift characteristics of the limited number of analog components, recognizing that most of the processing is performed digitally without drift of setpoint values.

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.~~ (Deleted)	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.~~ (Deleted)

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.I Average Planar Linear Heat Generation Rate

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

RATED, FLOW-DEPENDENT OR POWER-DEPENDENT

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

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

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

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.5.K Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the APPROPRIATE RATED, FLOW-DEPENDENT OR POWER-DEPENDENT 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.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 OPERATING LIMIT 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.~~

(Deleted)

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.~~

(Deleted)

3.5 BASES (Cont'd)

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

Insert N

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

Insert O

3.5.L. APRM Setpoints

~~Operation is constrained to the LHGR limit of Specification 3.5.J. This limit is reached when core maximum fraction of limiting power density (CMFLPD) equals 1.0. For the case where CMFLPD exceeds the fraction of rated thermal power, operation is permitted only at less than~~

(Deleted)

INSERT N:

At less than rated power conditions, the rated APLHGR limit is adjusted by a power dependent correction factor, MAPFAC(P). At less than rated flow conditions, the rated APLHGR limit is adjusted by a flow dependent correction factor, MAPFAC(F). The most limiting power-adjusted or flow-adjusted value is taken as the APLHGR operating limit for the off-rated condition.

The flow dependent correction factor, MAPFAC(F), applied to the rated APLHGR limit assures that (1) the 10 CFR 50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions. MAPFAC(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent correction factor, MAPFAC(P), applied to the rated APLHGR limit assures that the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated power conditions. MAPFAC(P) values are provided in the CORE OPERATING LIMITS REPORT.

INSERT O:

At less than rated power conditions, a power dependent MCPR operating limit, MCPR(P), is applicable. At less than rated flow conditions, a flow dependent MCPR operating limit, MCPR(F), is applicable. The most limiting power dependent or flow dependent value is taken as the MCPR operating limit for the off-rated condition.

The flow dependent limit, MCPR(F), provides the thermal margin required to protect the fuel from transients resulting from inadvertent core flow increases. MCPR(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent limit, MCPR(P), protects the fuel from the other limiting abnormal operating transients, including localized events such as a rod withdrawal error. MCPR(P) values are provided in the CORE OPERATING LIMITS REPORT.

~~100 percent rated power and only with APRM scram settings as required by Specification 3.5.L.1. The scram trip setting and rod block trip setting are adjusted to ensure that no combination of CMFLPD and FRP will increase the LHGR transient peak beyond that allowed by the one-percent plastic strain limit. A six-hour time period to achieve this condition is justified since the additional margin gained by the setdown adjustment is above and beyond that ensured by the safety analysis.~~

3.5.M. Core Thermal-Hydraulic Stability

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

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

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

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

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

3.5.N. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A and Addenda.

of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlines in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

6.9.1.6 SOURCE TESTS

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

6.9.1.7 CORE OPERATING LIMITS REPORT

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

Revise per Insert P

(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 and 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).

INSERT P:

(1) The rated APLHGR limit; the Flow Dependent APLHGR Factor, MAPFAC(F); and the Power Dependent APLHGR Factor, MAPFAC(P) for Specification 3.5.I.

(2) The LGHR limit for Specification 3.5.J.

(3) The rated MCPR Operating Limit; the Flow Dependent MCPR Operating Limit, MCPR(F); and the Power Dependent MCPR Operating Limit, MCPR(P) for Specification 3.5.K/4.5.K.

(4) The APRM flow biased rod block trip setting for Specification 2.1.A.1.c and Table 3.2.C.

(5) The RBM downscale trip setpoint, high power trip setpoint, intermediate power trip setpoint, low power trip setpoint, and applicable reactor thermal power ranges for each of the setpoints for Table 3.2.C.

ENCLOSURE 3
TENNESSEE VALLEY AUTHORITY (TVA)
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, and 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-353
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
1.0-7	1.0-7	1.0-7
1.0-8	1.0-8	1.0-8
1.1/2.1-2	1.1/2.1-2	1.1/2.1-2
1.1/2.1-3	1.1/2.1-3	1.1/2.1-3
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1.1/2.1-17	1.1/2.1-16	1.1/2.1-16
3.1/4.1-3	3.1/4.1-3	3.1/4.1-2
3.1/4.1-5	3.1/4.1-5	3.1/4.1-4
3.1/4.1-6	3.1/4.1-6	3.1/4.1-5
3.1/4.1-7	3.1/4.1-7	3.1/4.1-6
3.1/4.1-8	3.1/4.1-8	3.1/4.1-7
3.1/4.1-10	3.1/4.1-10	3.1/4.1-9
3.1/4.1-11	3.1/4.1-11	3.1/4.1-10
3.1/4.1-12	3.1/4.1-12	3.1/4.1-11
3.1/4.1-14	3.1/4.1-14	3.1/4.1-13
3.1/4.1-15	3.1/4.1-15	3.1/4.1-14
3.1/4.1-16	3.1/4.1-16	3.1/4.1-15
3.1/4.1-17	3.1/4.1-17	3.1/4.1-16
3.1/4.1-19	3.1/4.1-19	3.1/4.1-18
3.1/4.1-20	3.1/4.1-20	3.1/4.1-19
3.2/4.2-25	3.2/4.2-25	3.2/4.2-24
3.2/4.2-26	3.2/4.2-26	3.2/4.2-25
3.2/4.2-27	3.2/4.2-27	3.2/4.2-26
3.2/4.2-27a	3.2/4.2-27a	3.2/4.2-26a
3.2/4.2-27b	3.2/4.2-27b	3.2/4.2-26b
3.2/4.2-50	3.2/4.2-50	3.2/4.2-49
3.2/4.2-59	3.2/4.2-59	3.2/4.2-58
3.2/4.2-60	3.2/4.2-60	3.2/4.2-59
3.2/4.2-68	3.2/4.2-68	3.2/4.2-67
3.2/4.2-73	3.2/4.2-73a	3.2/4.2-72
3.3/4.3-8	3.3/4.3-8	3.3/4.3-8
3.3/4.3-17	3.3/4.3-17	3.3/4.3-17
3.5/4.5-18	3.5/4.5-18	3.5/4.5-18
3.5-4.5-19	3.5-4.5-19	3.5-4.5-19
3.5/4.5-20	3.5/4.5-20	3.5/4.5-20
3.5/4.5-33	3.5/4.5-31	3.5/4.5-34
3.5/4.5-34	3.5/4.5-32	3.5/4.5-35
3.5/4.5-35	3.5/4.5-33	3.5/4.5-36
3.5/4.5-36	3.5/4.5-34	3.5/4.5-37
3.5/4.5-37	3/5/4.5-35	3.5/4.5-38
6.0-26a	6.0-26a	6.0-26a
6.0-26b	6.0-26b	6.0-26b
6.0-26c		

II. REVISED PAGES

See attached.

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

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

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

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

U. Thermal Parameters

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

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

3. (Deleted)

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

1.0 DEFINITIONS (Cont'd)

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.

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal		Heat Balance	Once/7 Days
Flow Bias Signal		Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal		TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume			
Electronic Lvl Switches (LS-85-45-A, B, G, H)	A	Calibrated Water Column (5)	Note (5)
Float Switches (LS-85-45C-F)	A	Calibrated Water Column (5)	Note (5)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive (PT-1-81A, B & PT-1-91A, B)	B	Standard Pressure Source	Once/Operating Cycle (9)
Turbine Control Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	Note (5)	Note (5)

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. The flow bias signal calibration will consist of calibrating the analog differential pressure flow sensors once per operating cycle. Calibration of the flow bias processing system is done once per operating cycle as part of the overall APRM instrumentation calibration.

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

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

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between nonclass 1E power supply and the class 1E RPS bus. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor.

The APRM system is divided into four APRM channels and four 2-out-of-4 trip voter channels. Each APRM channel provides input to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip

system. The APRM system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter units, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system resulting in a full scram.

Each APRM instrument channel receives input signals from forty-three (43) Local Power Range Monitors (LPRMs). A minimum of twenty (20) LPRM inputs with three (3) per axial level is required for the APRM instrument channel to be OPERABLE. Fewer than the required minimum number of LPRM inputs generates an instrument channel inoperative alarm and a control rod block but does not result in an automatic trip input to the 2-out-of-4 voters.

Each protection trip system has one more IRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one IRM per protection trip system for maintenance, testing or calibration. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

3.1 BASES (Cont'd)

be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharge water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Reference Section 7.5.4 FSAR. Thus, the IRM is required in the REFUEL and STARTUP modes. In the power range the APRM system provides required protection. Reference Section 7.5.7 FSAR. Thus, the IRM System is not required in the RUN mode. The APRMs and the IRMs provide adequate coverage in the startup and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for STARTUP and RUN modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions as indicated in Table 3.1.1 operable in the REFUEL mode is to assure that shifting to the REFUEL mode during reactor power operation does not diminish the need for the reactor protection system.

Because of the APRM downscale rod block limit of $\geq$ 3 percent when in the RUN mode and high level flux scram limit of $\leq$15 percent when in the STARTUP Mode, the transition between the STARTUP and RUN Modes must be made with the APRM instrumentation indicating between 3 percent and 15 percent of rated power. In addition, the IRM system must be indicating below the High Flux setting (120/125 of scale) or a scram will occur when in the STARTUP Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to shutdown). When power is being reduced, if a transfer to the STARTUP mode is made and the IRMs have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

4.1 BASES

The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (1). This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failure such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

Except for the APRMs which take credit for self-test capability, the channels listed in Tables 4.1.A and 4.1.B are divided into three groups for functional testing. These are:

A. On-Off sensors that provide a scram trip function.

B. Analog devices coupled with bistable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as STARTUP or SHUTDOWN, or for which the only practical test is one that can be performed at shutdown.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.99999 probability of success (at the 50 percent confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95 percent confidence level is proposed. With the (1-out-of-2) X (2) logic, this requires that each sensor have an availability of 0.993 at the 95 percent confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history.[1]

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1. Reliability of Engineered Safety Features as a Function of Testing Frequency, I. M. Jacobs, "Nuclear Safety," Vol. 9, No. 4, July-August, 1968, pp. 310-312.

4.1 BASES (Cont'd)

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during STARTUP and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to SHUTDOWN or STARTUP: i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4 percent/month; i.e., in the period of a month a drift of 4 percent would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have been included in the latter table. These are: mode switch in SHUTDOWN and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

The APRM and 2-out-of-4 voter channel hardware is provided with a self-test capability which automatically checks most of the critical hardware at least once per 15 minute interval whenever the APRM channel is in the operate mode. This provides a virtually continuous monitoring of the essential APRM trip functions. In the event a critical fault is detected, an "inoperative" trip signal results. A fault detected in non-critical hardware results in an "inoperative" alarm. Following receipt of an "inoperative" trip or alarm signal, the operator can employ numerous diagnostic testing options to locate the problem.

The automatic self-test function is supplemented with a manual APRM trip functional test, including the 2-out-of-4 voter channels and the interface with the RPS trip systems. In combination with the virtually continuous self-testing, the manual APRM trip functional test provides adequate functional testing of the APRM trip function. Therefore, the six-month test frequency for the manual testing provides an acceptable level of availability of the APRM.

In addition to the above tests, the 2-out-of-4 voter is used to test the RPS scram contactors. The output of each voter channel is tripped to produce a scram signal into each of the RPS trip system channels (A1, A2, B1 and B2) to individually operate the respective scram contactors. The weekly test interval provides an acceptable level of availability of the scram contactors.

Each APRM receives the output signals from two analog differential pressure flow transducers, one associated with recirculation loop A and the other with recirculation loop B. These differential pressure signals are converted into representative digital loop flow signals within the same hardware that performs the APRM functions and are added to determine a total recirculation flow. The total recirculation flow value is used by the APRM to determine the flow biased setpoints. Each total recirculation flow signal developed by an APRM is compared in the hardware that performs the RBM functions to the signals from the remaining three APRMs. An alarm is given if a preset compare level setpoint is exceeded. The flow processing is integrated with the APRM processing and is covered by the same self-test and alarm functions described earlier. As a result of the virtually continuous monitoring of the equipment performing the flow processing, and the automatic comparison of redundant flow signals, it is acceptable to calibrate this equipment once per operating cycle.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CPR and APLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	2.1.A Neutron Flux Trip Settings 2.1.A.1.a (Cont'd) $S \leq (0.66W + 71\%)$ 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.

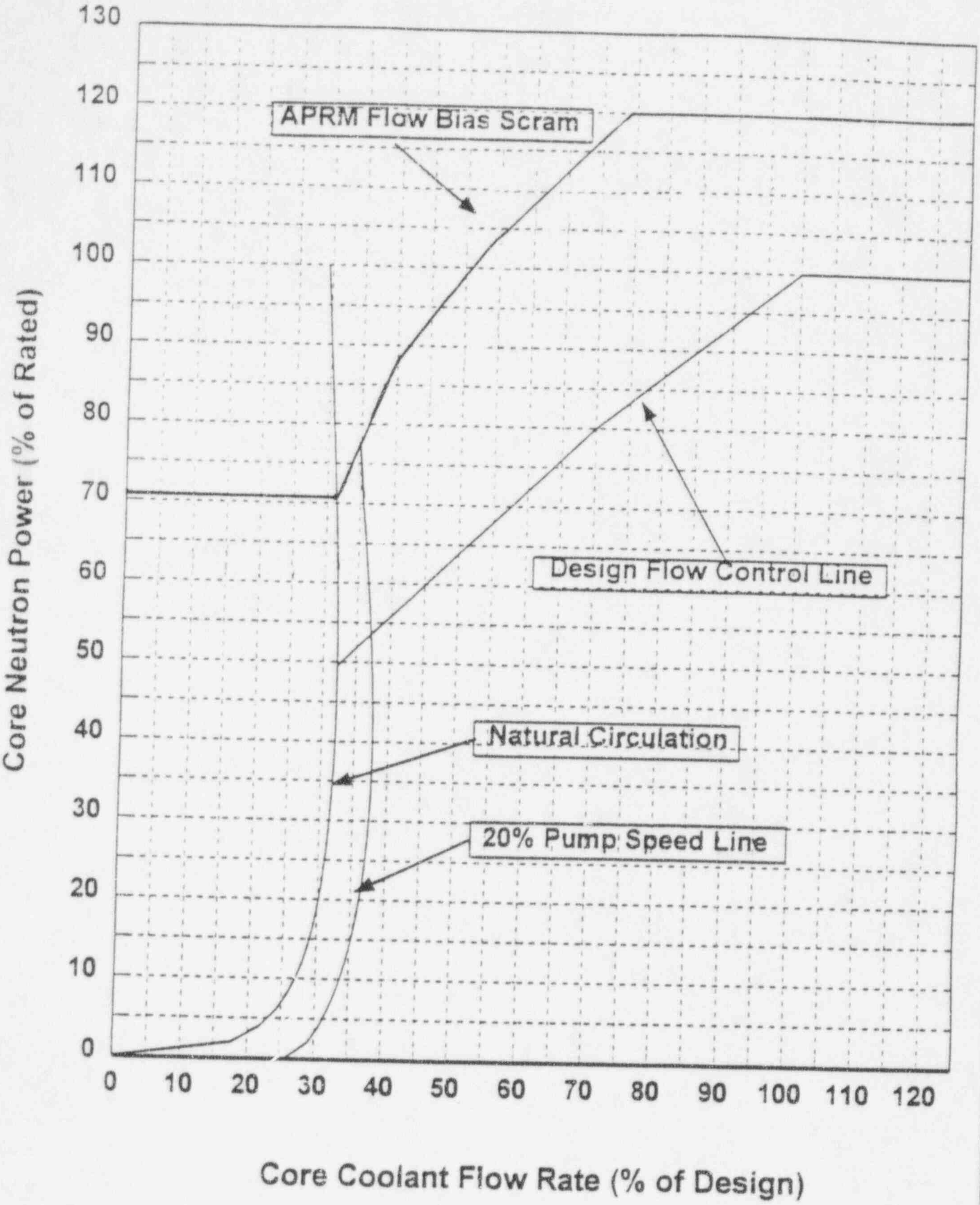
SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

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.

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



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

2.1 BASES (Cont'd)

5. Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3, NEDC-32433P.

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable: Shut-down	Refuel (7)	Startup/ Hot Standby	Run	Action (1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(21)(22)	X	(5)	1.A
3	Inoperable			X	X	(5)	1.A
	APRM (16)(24)(25)						
3(11)	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B or 1.E
3(11)	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B or 1.E
3(11)	High Flux	≤ 15% rated power			X(17)	(15)	1.A or 1.E
3(11)	Inoperative	(13)			X(17)	X	1.A or 1.E
2	2/4 Trip Voter	(12)			X	X	1.A or 1.F
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A

NOTES FOR TABLE 3.1.A

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels per trip system cannot be met for one trip system, trip the inoperable channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of operable instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An inoperable channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable channel shall be restored to operable status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all operable control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the STARTUP/HOT Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30 percent of rated.

   E. For the APRM functions only, if only two APRM channels are OPERABLE, restore a third APRM channel to OPERABLE status or trip one of the inoperable APRM channels within 6 hours. If only one APRM channel is OPERABLE, trip one inoperable APRM channel immediately and restore an inoperable APRM channel to OPERABLE status or initiate alternative action within 2 hours.

   F. For the APRM functions only, if one voter channel is inoperable in one trip system, restore the voter channel to OPERABLE status or trip the inoperable channel or the entire trip system within 12 hours. If one voter channel is inoperable in both trip systems, restore the inoperable voter channels to OPERABLE status or initiate alternative action within 6 hours.

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram with control rod block for reactor protection system reset.

3. Bypassed if reactor pressure is less than 1055 psig and mode switch not in RUN.

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the RUN position.

NOTES FOR TABLE 3.1.A (Cont'd)

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. (Deleted)

8. Not required to be OPERABLE when primary containment integrity is not required.

9. (Deleted)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

11. The same three (3) required APRM channels are shared by both RPS trip systems.

12. Any combination of APRM upscale or inoperative trips from two different (non-bypassed) APRMs will trip all of the 2/4 voter units.

13. Less than the required minimum number of OPERABLE LPRMs will cause an instrument channel inoperative alarm.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first state pressure is greater than or equal to 154 psig.

NOTES FOR TABLE 3.1.A (Cont'd)

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. (Deleted)

21. In the REFUEL Mode unless adequate shutdown margin has been demonstrated per Specification 3.3.A.1, whenever any control rod is withdrawn from a core cell containing one or more fuel assemblies, shorting links shall be removed from the RPS circuitry to enable the Source Range Monitor (SRM) noncoincidence high-flux scram function. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the SRMs.

22. The three required IRMs per trip channel is not required in the Shutdown or Refuel Modes if at least four IRMs (one in each core quadrant) are connected to give a noncoincidence, High Flux scram. The removal of four (4) shorting links is required to provide noncoincidence high-flux scram protection from the IRMs.

23. A channel may be placed in an inoperable status for up to 4 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

24. The Average Power Range Monitor scram function is varied (Reference Figure 2.1-1) as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with 2.1.A.

25. The APRM flow-biased neutron flux signal is fed through a time constant circuit of approximately 6 seconds. This time constant may be lowered or equivalently removed (no time delay) without affecting the operability of the flow-biased neutron flux trip channels. The APRM fixed high neutron flux signal does not incorporate the time constant but responds directly to instantaneous neutron flux.

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)		Trip Output Relays (4)(5) 2/4 Voter Logic (10)	Every 6 Months (9) Each Refueling Outage
High Flux (Flow Biased)		Trip Output Relays (4)(6) 2/4 Voter Logic (10)	Every 6 Months Each Refueling Outage
High Flux (Fixed Trip)		Trip Output Relays (4)(5) 2/4 Voter Logic (10)	Every 6 Months Each Refueling Outage
Inoperative		Trip Output Relays (4)(5) 2/4 Voter Logic (10)	Every 6 Months Each Refueling Outage
2/4 Trip Voter		Trip Scram Contactors (11)	Once/Week
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level	A	Trip Channel and Alarm	Once/Month (1)

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be operable or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an operable status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. The channel functional test shall include both the APRM channels and the 2/4 voter channels.

6. The channel functional test shall include both the APRM channels and the 2/4 voter channels plus the flow input function, excluding the flow transmitters.

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

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

9. Not required to be performed when entering the STARTUP/HOT STANDBY Mode from RUN Mode until 12 hours after entering the STARTUP/HOT STANDBY Mode.

10. Functional test consists of simulating APRM trip conditions at the APRM channel outputs to check all combinations of two tripped inputs to the 2/4 voter logic in each voter channel.

11. Functional test consists of manually tripping the 2/4 voter trip output, one voter channel at a time, to demonstrate that each scram contactor for each RPS trip system channel (A1, A2, B1 and B2) operates and produces a half-scram.

of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

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

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

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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 rated APLHGR limit; the Flow Dependent APLHGR Factor, MAPFAC(F); and the Power Dependent APLHGR Factor, MAPFAC(P) for Specification 3.5.I.

(2) The LHGR limit for Specification 3.5.J

(3) The rated MCPR Operating Limit; the Flow Dependent MCPR Operating Limit, MCPR(F); and the Power Dependent MCPR Operating Limit, MCPR(P) for Specification 3.5.K/4.5.K.

(4) The APRM flow biased rod block trip setting for Specification 2.1.A.1.c and Table 3.2.C.

(5) The RBM downscale trip setpoint, high power trip setpoint, intermediate power trip setpoint, low power trip setpoint, and applicable reactor thermal power ranges for each of the setpoints for Table 3.2.C.

6.9.1.7 CORE OPERATING LIMITS REPORT (Continued)

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

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

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

6.9.1.8 THE ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

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

1.0 DEFINITIONS (Cont'd)

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

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

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

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

U. Thermal Parameters

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

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

3. (Deleted)

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.

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.

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

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

$S \leq (0.66W + 71\%)$

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.

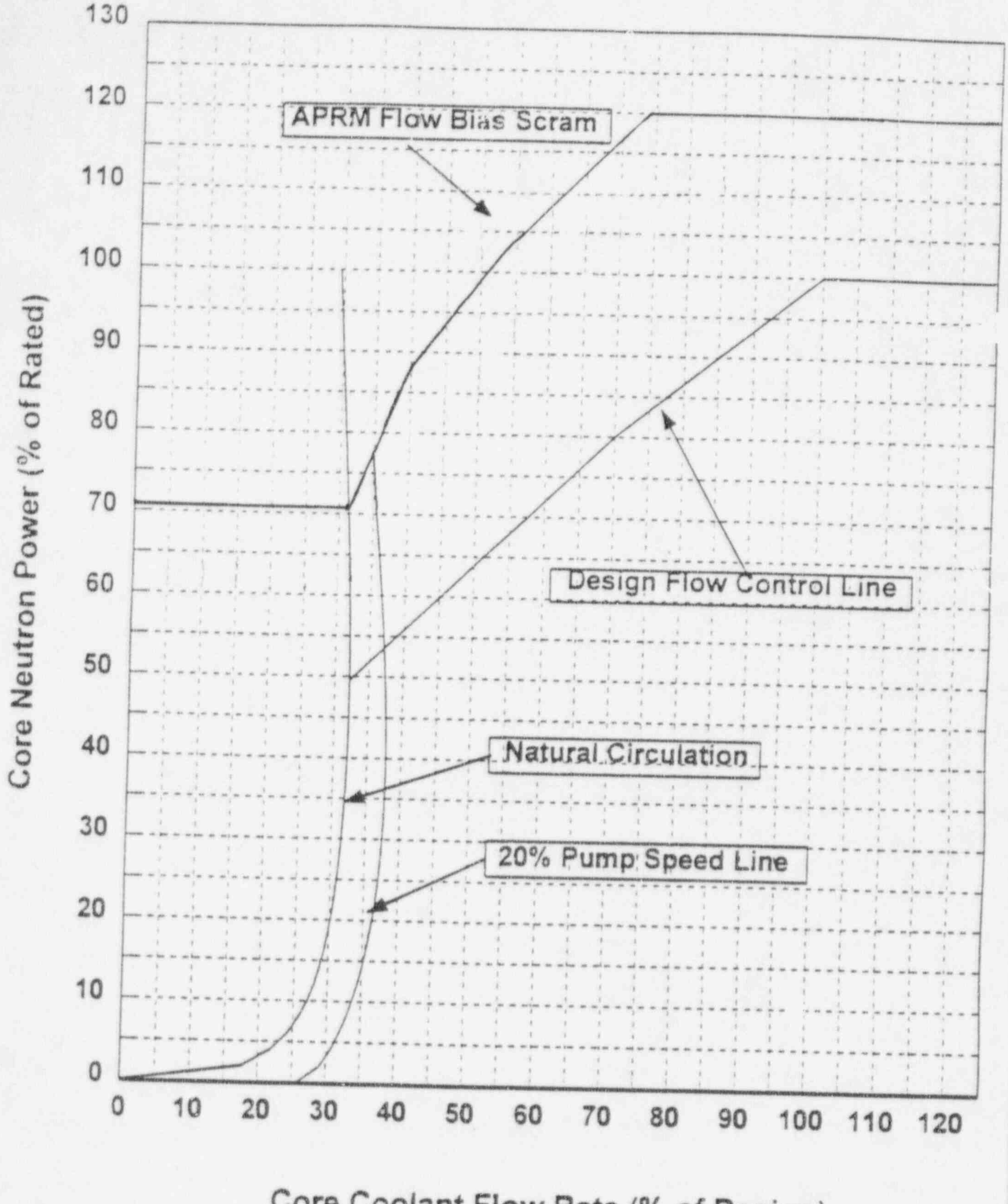
SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING

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.

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



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

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).

3. Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3, NEDC-32433P.

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Shut-down	Refuel (7)	Startup/ Hot Standby	Run	Action (1)
			Modes in Which Function Must Be Operable				
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(21)(22)	X	(5)	1.A
3	Inoperative			X	X	(5)	1.A
	APRM (16)(24)(25)						
3(11)	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B or 1.E
3(11)	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B or 1.E
3(11)	High Flux	≤ 15% rated power			X(17)	(15)	1.A or 1.E
3(11)	Inoperative	(13)			X(17)	X	1.A or 1.E
2	2/4 Trip Voter	(12)			X	X	1.A or 1.F
2	High Reactor Pressure	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14)	≥ 538" above vessel zero		X	X	X	1.A

NOTES FOR TABLE 3.1.A

1. There shall be two OPERABLE or tripped trip systems for each function. If the minimum number of OPERABLE instrument channels per trip system cannot be met for one trip system, trip the INOPERABLE channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of OPERABLE instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An INOPERABLE channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the INOPERABLE channel shall be restored to OPERABLE status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of OPERABLE rods and complete insertion of all OPERABLE rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all OPERABLE control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the STARTUP/HOT STANDBY position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30 percent of rated.

   E. For the APRM functions only, if only two APRM channels are OPERABLE, restore a third APRM channel to OPERABLE status or trip one of the inoperable APRM channels within 6 hours. If only one APRM channel is OPERABLE, trip one inoperable APRM channel immediately and restore an inoperable APRM channel to OPERABLE status or initiate alternative action within 2 hours.

   F. For the APRM functions only, if one voter channel is inoperable in one trip system, restore the voter channel to OPERABLE status or trip the inoperable channel or the entire trip system within 12 hours. If one voter channel is inoperable in both trip systems, restore the inoperable voter channels to OPERABLE status or initiate alternative action within 6 hours.

2. Scram discharge volume high bypass may be used in shutdown or refuel to bypass scram discharge volume scram with control rod block for reactor protection system reset.

3. DELETED

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the RUN position.

NOTES FOR TABLE 3.1.A (Cont'd)

6. The design permits closure of any two lines without a scram being initiated.

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be OPERABLE:

   A. Mode switch in shutdown

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. (Deleted)

8. Not required to be OPERABLE when primary containment integrity is not required.

9. (Deleted)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

11. The same three (3) required APRM channels are shared by both RPS trip systems.

12. Any combination of APRM upscale or inoperative trips from two different (non-bypassed) APRMs will trip all of the 2/4 voter units.

13. Less than the required minimum number of OPERABLE LPRMs will cause an instrument channel inoperative alarm.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

NOTES FOR TABLE 3.1.A (Cont'd)

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first stage pressure is greater than or equal to 154 psig.

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. (Deleted)

21. In the REFUEL Mode unless adequate shutdown margin has been demonstrated per Specification 3.3.A.1, whenever any control rod is withdrawn from a core cell containing one or more fuel assemblies, shorting links shall be removed from the RPS circuitry to enable the Source Range Monitor (SRM) noncoincidence high-flux scram function. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the SRMs.

22. The three required IRMs per trip channel is not required in the SHUTDOWN or REFUEL Modes if at least four IRMs (one in each core quadrant) are connected to give a noncoincidence, High Flux scram. The removal of four (4) shorting links is required to provide noncoincidence high-flux scram protection from the IRMs.

23. A channel may be placed in an INOPERABLE status for up to 4 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

24. The Average Power Range Monitor scram function is varied (Reference Figure 2.1-1) as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with 2.1.A.

25. The APRM flow-biased neutron flux signal is fed through a time constant circuit of approximately 6 seconds. This time constant may be lowered or equivalently removed (no time delay) without affecting the operability of the flow-biased neutron flux trip channels. The APRM fixed high neutron flux signal does not incorporate the time constant but responds directly to instantaneous neutron flux.

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once Per Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)		Trip Output Relays (4)(5) 2/4 Voter Logic (10)	Every 6 Months (9) Each Refueling Outage
High Flux (Flow Biased)		Trip Output Relays (4)(6) 2/4 Voter Logic (10)	Every 6 Months Each Refueling Outage
High Flux (Fixed Trip)		Trip Output Relays (4)(5) 2/4 Voter Logic (10)	Every 6 Months Each Refueling Outage
Inoperative		Trip Output Relays (4)(5) 2/4 Voter Logic (10)	Every 6 Months Each Refueling Outage
2/4 Trip Voter		Trip Scram Contactors (11)	Once/Week
High Reactor Pressure	A	Trip Channel and Alarm	Once/Month (1)
High Drywell Pressure	A	Trip Channel and Alarm	Once/Month (1)
Reactor Low Water Level	A	Trip Channel and Alarm	Once/Month (1)

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be OPERABLE or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an OPERABLE status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. The channel functional test shall include both the APRM channels and the 2/4 voter channels.

6. The channel functional test shall include both the APRM channels and the 2/4 voter channels plus the flow input function, excluding the flow transmitters.

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

8. Functional test frequency decreased to once/3 months to reduce the challenges to relief valves per NUREG 0737, Item II.K.3.16.

9. Not required to be performed when entering the STARTUP/HOT STANDBY Mode from RUN Mode until 12 hours after entering the STARTUP/HOT STANDBY Mode.

10. Functional test consists of simulating APRM trip conditions at the APRM channel outputs to check all combinations of two tripped inputs to the 2/4 voter logic in each voter channel.

11. Functional test consists of manually tripping the 2/4 voter trip output, one voter channel at a time, to demonstrate that each scram contactor for each RPS trip system channel (A1, A2, B1 and B2) operates and produces a half-scram.

probability of core instability following entry into Region II, the operator will take immediate action to exit the region. Although formal surveillances are not performed while exiting Region II (delaying exit for surveillances is undesirable), an immediate manual scram will be initiated if evidence of thermal-hydraulic instability is observed.

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

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

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

3.5.N. <u>References</u>

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 2, NEDO - 24088-1 and Addenda.

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

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE - 24011-P-A and Addenda.

4.5 <u>Core and Containment Cooling Systems Surveillance Frequencies</u>

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. The flow bias signal calibration will consist of calibrating the analog differential pressure flow sensors once per operating cycle. Calibration of the flow bias processing system is done once per operating cycle as part of the overall APRM instrumentation calibration.

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

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

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made INOPERABLE for brief intervals to conduct required functional tests and calibrations.

The reactor protection trip system is supplied, via a separate bus, by its own high inertia, ac motor-generator set. Alternate power is available to either Reactor Protection System bus from an electrical bus that can receive standby electrical power. The RPS monitoring system provides an isolation between nonclass 1E power supply and the class 1E RPS bus. This will ensure that failure of a nonclass 1E reactor protection power supply will not cause adverse interaction to the class 1E Reactor Protection System.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of OPERABLE instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor.

The APRM system is divided into four APRM channels and four 2-out-of-4 trip voter channels. Each APRM channel provides input to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip

3.1 BASES (Cont'd)

system. The APRM system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter units, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system resulting in a full scram.

Each APRM instrument channel receives input signals from forty-three (43) Local Power Range Monitors (LPRMs). A minimum of twenty (20) LPRM inputs with three (3) per axial level is required for the APRM instrument channel to be OPERABLE. Fewer than the required minimum number of LPRM inputs generates an instrument channel inoperative alarm and a control rod block but does not result in an automatic trip input to the 2-out-of-4 voters.

Each protection trip system has one more IRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one IRM per protection trip system for maintenance, testing or calibration. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharge water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Reference Section 7.5.4 FSAR. Thus, the IRM is required in the REFUEL and STARTUP modes. In the power range the APRM system provides required protection. Reference Section 7.5.7 FSAR. Thus, the IRM System is not required in the RUN mode. The APRMs and the IRMs provide adequate coverage in the STARTUP and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level, low scram pilot air header pressure and scram discharge volume high level scrams are required for STARTUP and RUN modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions as indicated in Table 3.1.A OPERABLE in the REFUEL mode is to assure that shifting to the REFUEL mode during reactor power operation does not diminish the need for the reactor protection system.

Because of the APRM downscale rod block limit of $\geq$ 3 percent when in the RUN mode and high level flux scram limit of $\leq$15 percent when in the STARTUP Mode, the transition between the STARTUP and RUN Modes must be made with the APRM instrumentation indicating between 3 percent and 15 percent of rated power. In addition, the IRM system must be indicating below the High Flux setting (120/125 of scale) or a scram will occur when in the STARTUP Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to SHUTDOWN). When power is being reduced, if a transfer to the STARTUP mode is made and the IRMs have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

The low scram pilot air header pressure trip performs the same function as the high water level in the scram discharge instrument volume for fast fill events in which the high level instrument response time may be inadequate. A fast fill event is postulated for certain degraded control air events in which the scram outlet valves unseat enough to allow 5 gpm per drive leakage into the scram discharge volume but not enough to cause control rod insertion.

4.1 BASES

The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (1). This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failure such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

Except for the APRMs which take credit for self-test capability, the channels listed in Tables 4.1.A and 4.1.B are divided into three groups for functional testing. These are:

A. On-Off sensors that provide a scram trip function.

B. Analog devices coupled with bistable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as STARTUP or SHUTDOWN, or for which the only practical test is one that can be performed at SHUTDOWN.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.9999 probability of success (at the 50 percent confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95 percent confidence level is proposed. With the (1-out-of-2) X (2) logic, this requires that each sensor have an availability of 0.993 at the 95 percent confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history.[1]

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1. Reliability of Engineered Safety Features as a Function of Testing Frequency, I. M. Jacobs, "Nuclear Safety," Vol. 9, No. 4, July-August, 1968, pp. 310-312.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during STARTUP and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to SHUTDOWN or STARTUP: i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4 percent/month; i.e., in the period of a month a drift of 4 percent would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have been included in the latter table. These are: mode switch in SHUTDOWN and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

The APRM and 2-out-of-4 voter channel hardware is provided with a self-test capability which automatically checks most of the critical hardware at least once per 15 minute interval whenever the APRM channel is in the operate mode. This provides a virtually continuous monitoring of the essential APRM trip functions. In the event a critical fault is detected, an "inoperative" trip signal results. A fault detected in non-critical hardware results in an "inoperative" alarm. Following receipt of an "inoperative" trip or alarm signal, the operator can employ numerous diagnostic testing options to locate the problem.

The automatic self-test function is supplemented with a manual APRM trip functional test, including the 2-out-of-4 voter channels and the interface with the RPS trip systems. In combination with the virtually continuous self-testing, the manual APRM trip functional test provides adequate functional testing of the APRM trip function. Therefore, the six-month test frequency for the manual testing provides an acceptable level of availability of the APRM.

In addition to the above tests, the 2-out-of-4 voter is used to test the RPS scram contactors. The output of each voter channel is tripped to produce a scram signal into each of the RPS trip system channels (A1, A2, B1 and B2) to individually operate the respective scram contactors. The weekly test interval provides an acceptable level of availability of the scram contactors.

Each APRM receives the output signals from two analog differential pressure flow transducers, one associated with recirculation loop A and the other with recirculation loop B. These differential pressure signals are converted into representative digital loop flow signals within the same hardware that performs the APRM functions and are added to determine a total recirculation flow. The total recirculation flow value is used by the APRM to determine the flow biased setpoints. Each total recirculation flow signal developed by an APRM is compared in the hardware that performs the RBM functions to the signals from the remaining three APRMs. An alarm is given if a preset compare level setpoint is exceeded. The flow processing is integrated with the APRM processing and is covered by the same self-test and alarm functions described earlier. As a result of the virtually continuous monitoring of the equipment performing the flow processing, and the automatic comparison of redundant flow signals, it is acceptable to calibrate this equipment once per operating cycle.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CPR and APLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal		Heat Balance	Once Every 7 Days
Flow Bias Signal		Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal		TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	A	Standard Pressure Source	Every 3 Months
Reactor Low Water Level	A	Pressure Standard	Every 3 Months
High Water Level in Scram Discharge Volume			
Float Switches (LS-85-45C-F)	A	Calibrated Water Column (5)	Note (5)
Electronic Lvl Switches (LS-85-45-A, B, G, H)	B	Calibrated Water Column	Once/Operating Cycle (9)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive	A	Standard Pressure Source	Every 6 Months
Turbine Control Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	Note (5)	Note (5)

NOTES FOR TABLE 4.1.B

1. A description of three groups is included in the Bases of this specification.

2. Calibrations are not required when the systems are not required to be OPERABLE or are tripped. If calibrations are missed, they shall be performed prior to returning the system to an OPERABLE status.

3. (Deleted)

4. Required frequency is initial startup following each refueling outage.

5. Physical inspection and actuation of these position switches will be performed once per operating cycle.

6. On controlled startups, overlap between the IRMs and APRMs will be verified.

7. The flow bias signal calibration will consist of calibrating the analog differential pressure flow sensors once per operating cycle. Calibration of the flow bias processing system is done once per operating cycle as part of the overall APRM instrumentation calibration.

8. A complete TIP system traverse calibrates the LPRM signals to the process computer. The individual LPRM meter readings will be adjusted as a minimum at the beginning of each operating cycle before reaching 100 percent power.

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

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the reactor coolant system.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevents criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made INOPERABLE for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is made up of two independent trip systems (refer to Section 7.2, FSAR). There are usually four channels provided to monitor each critical parameter, with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic such that either channel trip will trip that trip system. The simultaneous tripping of both trip systems will produce a reactor scram.

This system meets the intent of IEEE-279 for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2-out-of-3 system and somewhat less than that of a 1-out-of-2 system.

With the exception of the Average Power Range Monitor (APRM) channels, the Intermediate Range Monitor (IRM) channels, the Main Steam Isolation Valve closure and the Turbine Stop Valve closure, each trip system logic has one instrument channel. When the minimum condition for operation on the number of OPERABLE instrument channels per untripped protection trip system is met or if it cannot be met and the effected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor.

The APRM system is divided into four APRM channels and four 2-out-of-4 trip voter channels. Each APRM channel provides input to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip

3.1 BASES (Cont'd)

system. The APRM system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter units, but no trip inputs to either RPS trip system. A trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system resulting in a full scram.

Each APRM instrument channel receives input signals from forty-three (43) Local Power Range Monitors (LPRMs). A minimum of twenty (20) LPRM inputs with three (3) per axial level is required for the APRM instrument channel to be OPERABLE. Fewer than the required minimum number of LPRM inputs generates an instrument channel inoperative alarm and a control rod block but does not result in an automatic trip input to the 2-out-of-4 voters.

Each protection trip system has one more IRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one IRM per protection trip system for maintenance, testing or calibration. The bases for the scram setting for the IRM, APRM, high reactor pressure, reactor low water level, MSIV closure, turbine control valve fast closure, and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Instrumentation (pressure switches) for the drywell are provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting as the core cooling systems (CSCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent return to criticality. This instrumentation is a backup to the reactor vessel water level instrumentation.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.2.3.7 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system (120/125 scram) in conjunction with the APRM system (15 percent scram) provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The discharge volume tank accommodates in excess of 50 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not

be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which alarm and scram the reactor when the volume of water reaches 50 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharge water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Reference Section 7.5.4 FSAR. Thus, the IRM is required in the REFUEL and STARTUP modes. In the power range the APRM system provides required protection. Reference Section 7.5.7 FSAR. Thus, the IRM System is not required in the RUN mode. The APRMs and the IRMs provide adequate coverage in the STARTUP and intermediate range.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for STARTUP and RUN modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions as indicated in Table 3.1.1 OPERABLE in the REFUEL mode is to assure that shifting to the REFUEL mode during reactor power operation does not diminish the need for the reactor protection system.

Because of the APRM downscale rod block limit of $\geq$ 3 percent when in the RUN mode and high level flux scram limit of $\leq$15 percent when in the STARTUP Mode, the transition between the STARTUP and RUN Modes must be made with the APRM instrumentation indicating between 3 percent and 15 percent of rated power. In addition, the IRM system must be indicating below the High Flux setting (120/125 of scale) or a scram will occur when in the STARTUP Mode. For normal operating conditions, these limits provide assurance of overlap between the IRM system and APRM system so that there are no "gaps" in the power level indications (i.e., the power level is continuously monitored from beginning of startup to full power and from full power to shutdown). When power is being reduced, if a transfer to the STARTUP mode is made and the IRMs have not been fully inserted (a maloperational but not impossible condition) a control rod block immediately occurs so that reactivity insertion by control rod withdrawal cannot occur.

4.1 BASES

The minimum functional testing frequency used in this specification is based on a reliability analysis using the concepts developed in reference (1). This concept was specifically adapted to the one-out-of-two taken twice logic of the reactor protection system. The analysis shows that the sensors are primarily responsible for the reliability of the reactor protection system. This analysis makes use of "unsafe failure" rate experience at conventional and nuclear power plants in a reliability model for the system. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is functionally tested or attempts to respond to a real signal. Failure such as blown fuses, ruptured bourdon tubes, faulted amplifiers, faulted cables, etc., which result in "upscale" or "downscale" readings on the reactor instrumentation are "safe" and will be easily recognized by the operators during operation because they are revealed by an alarm or a scram.

Except for the APRMs which take credit for self-test capability, the channels listed in Tables 4.1.A and 4.1.B are divided into three groups for functional testing. These are:

A. On-Off sensors that provide a scram trip function.

B. Analog devices coupled with bistable trips that provide a scram function.

C. Devices which only serve a useful function during some restricted mode of operation, such as STARTUP or SHUTDOWN, or for which the only practical test is one that can be performed at shutdown.

The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation. During design, a goal of 0.99999 probability of success (at the 50 percent confidence level) was adopted to assure that a balanced and adequate design is achieved. The probability of success is primarily a function of the sensor failure rate and the test interval. A three-month test interval was planned for group (A) sensors. This is in keeping with good operating practices, and satisfies the design goal for the logic configuration utilized in the Reactor Protection System.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95-percent confidence level is proposed. With the (1-out-of-2) X (2) logic, this requires that each sensor have an availability of 0.993 at the 95 percent confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history.[1]

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1. Reliability of Engineered Safety Features as a Function of Testing Frequency, I. M. Jacobs, "Nuclear Safety," Vol. 9, No. 4, July-August, 1968, pp. 310-312.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during STARTUP and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to SHUTDOWN or STARTUP; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4 percent/month; i.e., in the period of a month a drift of .4-percent would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1.A and 4.1.B indicates that two instrument channels have been included in the latter table. These are: mode switch in SHUTDOWN and manual scram. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration during operation is not applicable, i.e., the switch is either on or off.

The APRM and 2-out-of-4 voter channel hardware is provided with a self-test capability which automatically checks most of the critical hardware at least once per 15 minute interval whenever the APRM channel is in the operate mode. This provides a virtually continuous monitoring of the essential APRM trip functions. In the event a critical fault is detected, an "inoperative" trip signal results. A fault detected in non-critical hardware results in an "inoperative" alarm. Following receipt of an "inoperative" trip or alarm signal, the operator can employ numerous diagnostic testing options to locate the problem.

The automatic self-test function is supplemented with a manual APRM trip functional test, including the 2-out-of-4 voter channels and the interface with the RPS trip systems. In combination with the virtually continuous self-testing, the manual APRM trip functional test provides adequate functional testing of the APRM trip function. Therefore, the six-month test frequency for the manual testing provides an acceptable level of availability of the APRM.

In addition to the above tests, the 2-out-of-4 voter is used to test the RPS scram contactors. The output of each voter channel is tripped to produce a scram signal into each of the RPS trip system channels (A1, A2, B1 and B2) to individually operate the respective scram contactors. The weekly test interval provides an acceptable level of availability of the scram contactors.

Each APRM receives the output signals from two analog differential pressure flow transducers, one associated with recirculation loop A and the other with recirculation loop B. These differential pressure signals are converted into representative digital loop flow signals within the same hardware that performs the APRM functions and are added to determine a total recirculation flow. The total recirculation flow value is used by the APRM to determine the flow biased setpoints. Each total recirculation flow signal developed by an APRM is compared in the hardware that performs the RBM functions to the signals from the remaining three APRMs. An alarm is given if a preset compare level setpoint is exceeded. The flow processing is integrated with the APRM processing and is covered by the same self-test and alarm functions described earlier. As a result of the virtually continuous monitoring of the equipment performing the flow processing, and the automatic comparison of redundant flow signals, it is acceptable to calibrate this equipment once per operating cycle.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. The APRM system, which uses the LPRM readings to detect a change in thermal power, will be calibrated every seven days using a heat balance to compensate for this change in sensitivity. The RBM system uses the LPRM reading to detect a localized change in thermal power. It applies a correction factor based on the APRM output signal to determine the percent thermal power and therefore any change in LPRM sensitivity is compensated for by the APRM calibration. The technical specification limits of CPR and APLHGR are determined by the use of the process computer or other backup methods. These methods use LPRM readings and TIP data to determine the power distribution.

Compensation in the process computer for changes in LPRM sensitivity will be made by performing a full core TIP traverse to update the computer calculated LPRM correction factors every 1000 effective full power hours.

As a minimum the individual LPRM meter readings will be adjusted at the beginning of each operating cycle before reaching 100 percent power.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
3(1)	APRM Upscale (Flow Bias)	(2)
3(1)	APRM Upscale (Startup Mode) (8)	≤12%
3(1)	APRM Downscale (9)	≥3%
3(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Power Bias)	
	Low Power Range (13)	(14)
	Intermediate Power Range (13)	(14)
	High Power Range (13)	(14)
2(7)	RBM Downscale (9) (13)	(15)
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)
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, RBM or APRM channel nor more than two 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. The RBM need not be OPERABLE if either of the following two conditions is met:

      (1) Reactor thermal power is $\geq$90 percent of rated and MCPR is $\geq$1.40, or

      (2) Reactor thermal power is $<$90 percent of rated and MCPR is $\geq$1.70.

d. 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.

e. 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 the required minimum number of LPRM inputs.

(3) APRM module unplugged.

(4) Self-tested detected critical fault.

c. RBM

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

(2) RBM module unplugged.

(3) RBM fails to null.

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

(5) Self-test detected critical fault.

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

NOTES FOR TABLE 3.2.C (Cont'd)

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

13. The RBM rod block trip setpoints and applicable power ranges are specified in the CORE OPERATING LIMITS REPORT (COLR).

14. Less than or equal to the setpoint allowable value specified in the COLR.

15. Greater than or equal to the setpoint allowable value specified in the COLR.

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TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test		Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1)	(13)	once/operating cycle	once/day (8)
APRM Upscale (Startup Mode)	(1)	(13)	once/operating cycle	once/day (8)
APRM Downscale	(1)	(13)	once/operating cycle	once/day (8)
APRM Inoperative	(1)	(13)	N/A	once/day (8)
RBM Upscale (Power Bias)	(1)	(13)	once/operating cycle	N/A
RBM Downscale	(1)	(13)	once/operating cycle	N/A
RBM Inoperative	(1)	(13)	N/A	N/A
IRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Detector not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
IRM Inoperative	(1)(2)	(13)	N/A	N/A
SRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Detector not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
SRM Inoperative	(1)(2)	(13)	N/A	N/A
Rod Block Logic	(16)		N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter		once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter		once/operating cycle	N/A

NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2. D AND 4.2.K

1. For IRMs and SRMs functional tests shall be performed once per month. For APRMs and RBMs functional tests shall be performed once per 6 months.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.

9. Calibration frequency shall be once/year.

10. Deleted

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.K (Cont'd)

14. (Deleted)

15. (Deleted)

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

20. (Deleted)

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. (Deleted)

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

3.2 BASES (Cont'd)

The control rod block functions are provided to generate a trip signal to block rod withdrawal if the monitored power level exceeds a preset value. The trip logic for this function is 1-out-of-n: e.g., any trip on one of four 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 provides a trip signal for blocking rod withdrawal when average reactor thermal power exceeds pre-established limits set to prevent scram actuation.

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 adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the reactor zone ventilation exhaust ducts and in the refueling zone.

4.2 BASES (Cont'd)

The conclusions to be drawn are these:

1. A 1-out-of-n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the reactor and refueling zones which initiate building isolation and standby gas treatment operation are arranged such that two sensors high (above the high level setpoint) in a single channel or one sensor downscale (below low level setpoint) or inoperable in two channels in the same zone will initiate a trip function. The functional testing frequencies for both the channel functional test and the high voltage power supply functional test are based on a Probabilistic Risk Assessment and system drift characteristics of the Reactor Building Ventilation Radiation Monitors. The calibration frequency is based upon the drift characteristics of the radiation monitors.

The automatic pressure relief instrumentation can be considered to be a 1-out-of-2 logic system and the discussion above applies also.

The RCIC and HPCI system logic tests required by Table 4.2.B contain provisions to demonstrate that these systems will automatically restart on a RPV low water level signal received subsequent to a RPV high water level trip.

The electronic instrumentation comprising the APRM rod block and Rod Block Monitor functions together with the recirculation flow instrumentation for flow bias purposes is monitored by the same self-test functions as applied to the APRM function for the RPS. The functional test frequency of every six months is based on this automatic self-test monitoring at 15 minute intervals and on the low expected equipment failure rates. Calibration frequency of once per operating cycle is based on the drift characteristics of the limited number of analog components, recognizing that most of the processing is performed digitally without drift of setpoint values.

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. (Deleted)	5. (Deleted)

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.I Average Planar Linear Heat Generation Rate

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

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

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

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

J. Linear Heat Generation Rate (LHGR)

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

LIMITING CONDITIONS FOR OPERATION

3.5.J Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the appropriate rated, flow-dependent or power-dependent 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 operating limit MCPR 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	SURVEILLANCE REQUIREMENTS
3.5.K Minimum Critical Power Ratio (MCPR)	4.5.K Minimum Critical Power Ratio (MCPR)
	4.5.K.2 (Cont'd)
	b. 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	L. APRM Setpoints
(Deleted)	(Deleted)

3.5 BASES (Cont'd)

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

At less than rated power conditions, the rated APLHGR limit is adjusted by a power dependent correction factor, MAPFAC(P). At less than rated flow conditions, the rated APLHGR limit is adjusted by a flow dependent correction factor, MAPFAC(F). The most limiting power-adjusted or flow-adjusted value is taken as the APLHGR operating limit for the off-rated condition.

The flow dependent correction factor, MAPFAC(F), applied to the rated APLHGR limit assures that (1) the 10 CFR 50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions. MAPFAC(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent correction factor, MAPFAC(P), applied to the rated APLHGR limit assures that the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated power conditions. MAPFAC(P) values are provided in the CORE OPERATING LIMITS REPORT.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

At less than rated power conditions, a power dependent MCPR operating limit, MCPR(P), is applicable. At less than rated flow conditions, a flow dependent MCPR operating limit, MCPR(F), is applicable. The most limiting power dependent or flow dependent value is taken as the MCPR operating limit for the off-rated condition.

The flow dependent limit, MCPR(F), provides the thermal margin required to protect the fuel from transients resulting from inadvertent core flow increases. MCPR(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent limit, MCPR(P), protects the fuel from the other limiting abnormal operating transients, including localized events such as a rod withdrawal error. MCPR(P) values are provided in the CORE OPERATING LIMITS REPORT.

3.5.L. APRM Setpoints

(Deleted)

3.5.M. Core Thermal-Hydraulic Stability

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

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

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

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

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

3.5.N. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEIM-10735, August 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

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

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

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

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

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6.9.1.5 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

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

6.9.1.6 SOURCE TESTS

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

6.9.1.7 CORE OPERATING LIMITS REPORT

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

   (1) The rated APLHGR limit; the Flow Dependent APLHGR Factor, MAPFAC(F); and the Power Dependent APLHGR Factor, MAPFAC(P) for Specification 3.5.I.

   (2) The LHGR limit for Specification 3.5.J

   (3) The rated MCPR Operating Limit; the Flow Dependent MCPR Operating Limit, MCPR(F); and the Power Dependent MCPR Operating Limit, MCPR(P) for Specification 3.5.K/4.5.K.

(4) The APRM flow biased rod block trip setting for Specification 2.1.A.1.c and Table 3.2.C.

(5) The RBM downscale trip setpoint, high power trip setpoint, intermediate power trip setpoint, low power trip setpoint, and applicable reactor thermal power ranges for each of the setpoints for Table 3.2.C.

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

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

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

6.9.1.8 THE ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted by April 1, of each year. The report shall include summaries of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. A single submittal may be made for a multi-unit station.

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
3(1)	APRM Upscale (Flow Bias)	(2)
3(1)	APRM Upscale (Startup Mode) (8)	≤12%
3(1)	APRM Downscale (9)	≥3%
3(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Power Bias)	
	Low Power Range (13)	(14)
	Intermediate Power Range (13)	(14)
	High Power Range (13)	(14)
2(7)	RBM Downscale (9) (13)	(15)
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)
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, RBM or APRM channel nor more than two 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. The RBM need not be OPERABLE if either of the following two conditions is met:

      (1) Reactor thermal power is $\geq$90 percent of rated and MCPR is $\geq$1.40, or

      (2) Reactor thermal power is <90 percent of rated and MCPR is $\geq$1.70.

NOTES FOR TABLE 3.2.C (Cont'd)

d. 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.

e. 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 the required minimum number of LPRM inputs.

(3) APRM module unplugged.

(4) Self-tested detected critical fault.

c. RBM

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

(2) RBM module unplugged.

(3) RBM fails to null.

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

(5) Self-test detected critical fault.

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

NOTES FOR TABLE 3.2.C (Cont'd)

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

13. The RBM rod block trip setpoints and applicable power ranges are specified in the CORE OPERATING LIMITS REPORT (COLR).

14. Less than or equal to the setpoint allowable value specified in the COLR.

15. Greater than or equal to the setpoint allowable value specified in the COLR.

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TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test		Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1)	(13)	once/operating cycle	once/day (8)
APRM Upscale (Startup Mode)	(1)	(13)	once/operating cycle	once/day (8)
APRM Downscale	(1)	(13)	once/operating cycle	once/day (8)
APRM Inoperative	(1)	(13)	N/A	once/day (8)
RBM Upscale (Power Bias)	(1)	(13)	once/operating cycle	N/A
RBM Downscale	(1)	(13)	once/operating cycle	N/A
RBM Inoperative	(1)	(13)	N/A	N/A
IRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Detector not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
IRM Inoperative	(1)(2)	(13)	N/A	N/A
SRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Detector not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
SRM Inoperative	(1)(2)	(13)	N/A	N/A
Rod Block Logic	(16)		N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter		once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter		once/operating cycle	N/A

NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2. D AND 4.2.K

1. For IRMs and SRMs functional tests shall be performed once per month. For APRMs and RBMs functional tests shall be performed once per 6 months.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.

9. Calibration frequency shall be once/year.

10. Deleted

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

<u>NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.K</u> (Cont'd)

14. (Deleted)

15. (Deleted)

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

20. (Deleted)

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. (Deleted)

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

The control rod block functions are provided to generate a trip signal to block rod withdrawal if the monitored power level exceeds a preset value. The trip logic for this function is 1-out-of-n: e.g., any trip on one of four 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 provides a trip signal for blocking rod withdrawal when average reactor thermal power exceeds pre-established limits set to prevent scram actuation.

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 adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two radiation monitors are provided for each unit which initiate Primary Containment Isolation (Group 6 isolation valves) Reactor Building Isolation and operation of the Standby Gas Treatment System. These instrument channels monitor the radiation in the reactor zone ventilation exhaust ducts and in the refueling zone.

4.2 BASES (Cont'd)

The conclusions to be drawn are these:

1. A 1-out-of-n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the reactor and refueling zones which initiate building isolation and standby gas treatment operation are arranged such that two sensors high (above the high level setpoint) in a single channel or one sensor downscale (below low level setpoint) or inoperable in two channels in the same zone will initiate a trip function. The functional testing frequencies for both the channel functional test and the high voltage power supply functional test are based on a Probabilistic Risk Assessment and system drift characteristics of the Reactor Building Ventilation Radiation Monitors. The calibration frequency is based upon the drift characteristics of the radiation monitors.

The automatic pressure relief instrumentation can be considered to be a 1-out-of-2 logic system and the discussion above applies also.

The RCIC and HPCI system logic tests required by Table 4.2.B contain provisions to demonstrate that these systems will automatically restart on a RPV low water level signal received subsequent to a RPV high water level trip.

The electronic instrumentation comprising the APRM rod block and Rod Block Monitor functions together with the recirculation flow instrumentation for flow bias purposes is monitored by the same self-test functions as applied to the APRM function for the RPS. The functional test frequency of every six months is based on this automatic self-test monitoring at 15 minute intervals and on the low expected equipment failure rates. Calibration frequency of once per operating cycle is based on the drift characteristics of the limited number of analog components, recognizing that most of the processing is performed digitally without drift of setpoint values.

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. (Deleted)	5. (Deleted)

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.I Average Planar Linear Heat Generation Rate

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

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

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

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

J. Linear Heat Generation Rate (LHGR)

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

LIMITING CONDITIONS FOR OPERATION

3.5.J Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

If at any time during steady-state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the COLD SHUTDOWN CONDITION within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.K Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the appropriate rated, flow-dependent or power-dependent 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 operating limit MCPR 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

(Deleted)

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

(Deleted)

3.5 BASES (Cont'd)

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

At less than rated power conditions, the rated APLHGR limit is adjusted by a power dependent correction factor, MAPFAC(P). At less than rated flow conditions, the rated APLHGR limit is adjusted by a flow dependent correction factor, MAPFAC(F). The most limiting power-adjusted or flow-adjusted value is taken as the APLHGR operating limit for the off-rated condition.

The flow dependent correction factor, MAPFAC(F), applied to the rated APLHGR limit assures that (1) the 10 CFR 50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions. MAPFAC(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent correction factor, MAPFAC(P), applied to the rated APLHGR limit assures that the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated power conditions. MAPFAC(P) values are provided in the CORE OPERATING LIMITS REPORT.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at $\geq$ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

At less than rated power conditions, a power dependent MCPR operating limit, MCPR(P), is applicable. At less than rated flow conditions, a flow dependent MCPR operating limit, MCPR(F), is applicable. The most limiting power dependent or flow dependent value is taken as the MCPR operating limit for the off-rated condition.

The flow dependent limit, MCPR(F), provides the thermal margin required to protect the fuel from transients resulting from inadvertent core flow increases. MCPR(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent limit, MCPR(P), protects the fuel from the other limiting abnormal operating transients, including localized events such as a rod withdrawal error. MCPR(P) values are provided in the CORE OPERATING LIMITS REPORT.

3.5.L. APRM Setpoints

(Deleted)

3.5.M. Core Thermal-Hydraulic Stability

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

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

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

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

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

3.5.N. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEIM-10735, August 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

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

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valve in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

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

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

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6.9.1.5 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

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

6.9.1.6 SOURCE TESTS

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

6.9.1.7 CORE OPERATING LIMITS REPORT

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

   (1) The rated APLHGR limit; the Flow Dependent APLHGR Factor, MAPFAC(F); and the Power Dependent APLHGR Factor, MAPFAC(P) for Specification 3.5.I.

   (2) The LHGR limit for Specification 3.5.J

   (3) The rated MCPR Operating Limit; the Flow Dependent MCPR Operating Limit, MCPR(F); and the Power Dependent MCPR Operating Limit, MCPR(P) for Specification 3.5.K/4.5.K.

(4) The APRM flow biased rod block trip setting for Specification 2.1.A.1.c and Table 3.2.C.

(5) The RBM downscale trip setpoint, high power trip setpoint, intermediate power trip setpoint, low power trip setpoint, and applicable reactor thermal power ranges for each of the setpoints for Table 3.2.C.

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

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

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

6.9.1.8 THE ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year of operation shall be submitted by April 1, of each year. The report shall include summaries of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. A single submittal may be made for a multi-unit station.

The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

1.0 DEFINITIONS (Cont'd)

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

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

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

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

U. Thermal Parameters

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

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

3. (Deleted)

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

1.0 DEFINITIONS (Cont'd)

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.

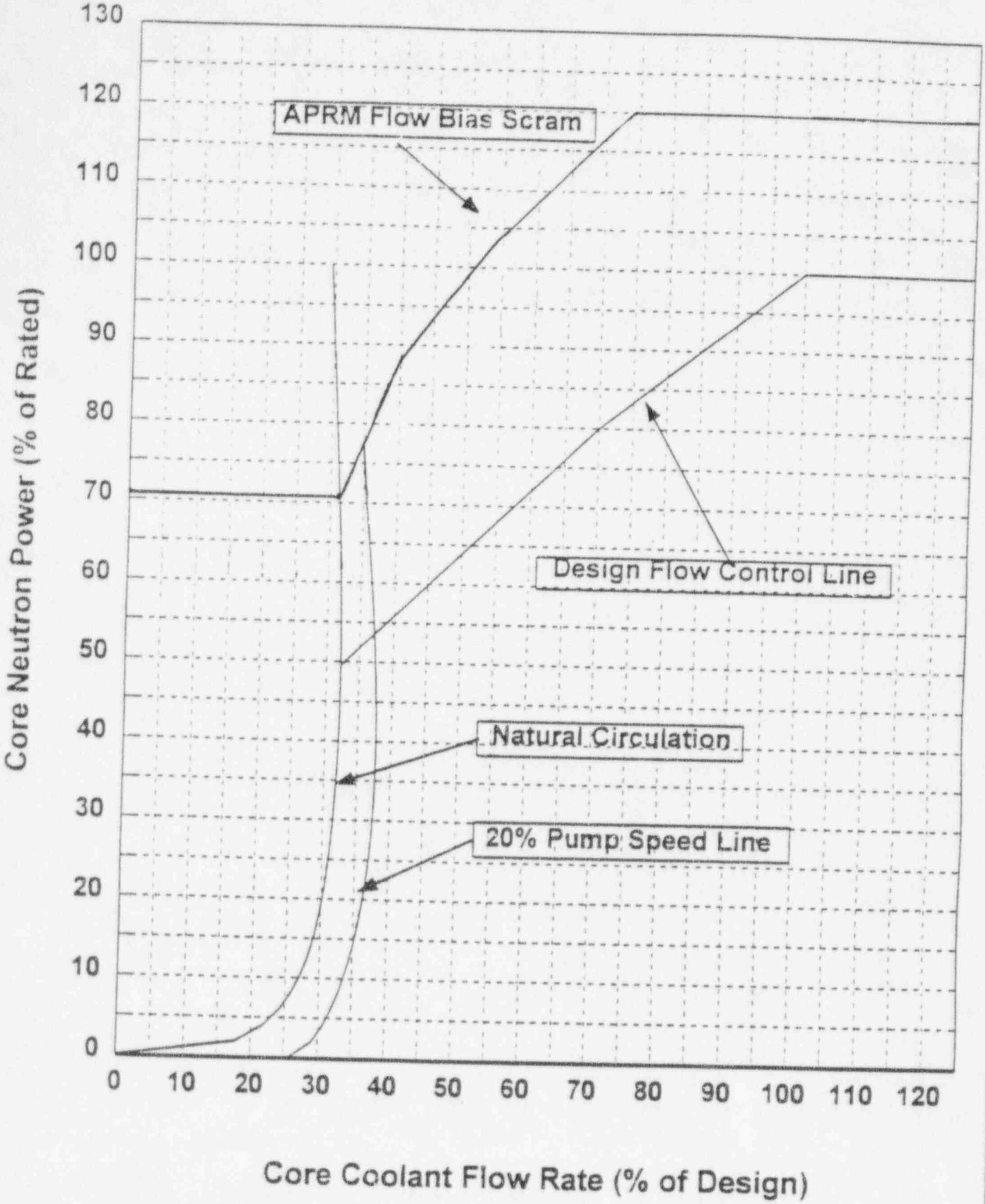
SAFETY LIMIT	LIMITING SAFETY SYSTEM SETTING
	2.1.A Neutron Flux Trip Settings 2.1.A.1.a (Cont'd) $S \leq (0.66W + 71\%)$ where: S = Setting in percent of rated thermal power (3293 MWt) W = Loop recirculation flow rate in percent of rated b. For no combination of loop recirculation flow rate and core 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.

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



APRM Flow Bias Scram vs. Reactor Core Flow

Fig. 2.1-2

F. (Deleted)

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

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

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

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

L. References

1. Supplemental Reload Licensing Report of Browns Ferry Nuclear Plant, Unit 2 (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. Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3, NEDC-32433P.

TABLE 3.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Min. No. of Operable Instr. Channels Per Trip System (1)(23)	Trip Function	Trip Level Setting	Modes in which Function Must Be Operable: Shut-down	Refuel (7)	Startup/ Hot Standby	Run	Action (1)
1	Mode Switch in Shutdown		X	X	X	X	1.A
1	Manual Scram		X	X	X	X	1.A
	IRM (16)						
3	High Flux	≤120/125 Indicated on scale	X(22)	X(21)(22)	X	(5)	1.A
3	Inoperable			X	X	(5)	1.A
	APRM (16)(24)(25)						
3(11)	High Flux (Flow Biased)	See Spec. 2.1.A.1				X	1.A or 1.B or 1.E
3(11)	High Flux (Fixed Trip)	≤ 120%				X	1.A or 1.B or 1.E
3(11)	High Flux	≤ 15% rated power			X(17)	(15)	1.A or 1.E
3(11)	Inoperative	(13)			X(17)	X	1.A or 1.E
2	2/4 Trip Voter	(12)			X	X	1.A or 1.F
2	High Reactor Pressure (PIS-3-22AA,BB,C,D)	≤ 1055 psig		X(10)	X	X	1.A
2	High Drywell Pressure (14) (PIS-64-56 A-D)	≤ 2.5 psig		X(8)	X(8)	X	1.A
2	Reactor Low Water Level (14) (LIS-3-203 A-D)	≥ 538" above vessel zero		X	X	X	1.A

NOTES FOR TABLE 3.1.A

1. There shall be two OPERABLE or tripped trip systems for each function. If the minimum number of OPERABLE instrument channels per trip system cannot be met for one trip system, trip the INOPERABLE channels or entire trip system within one hour, or, alternatively, take the below listed action for that trip function. If the minimum number of OPERABLE instrument channels cannot be met by either trip system, the appropriate action listed below (refer to right-hand column of Table) shall be taken. An INOPERABLE channel need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the INOPERABLE channel shall be restored to OPERABLE status within two hours, or take the action listed below for that trip function.

   A. Initiate insertion of OPERABLE rods and complete insertion of all OPERABLE rods within four hours. In refueling mode, suspend all operations involving core alterations and fully insert all OPERABLE control rods within one hour.

   B. Reduce power level to IRM range and place mode switch in the STARTUP/HOT Standby position within 8 hours.

   C. Reduce turbine load and close main steam line isolation valves within 8 hours.

   D. Reduce power to less than 30 percent of rated.

   E. For the APRM functions only, if only two APRM channels are OPERABLE, restore a third APRM channel to OPERABLE status or trip one of the inoperable APRM channels within 6 hours. If only one APRM channel is OPERABLE, trip one inoperable APRM channel immediately and restore an inoperable APRM channel to OPERABLE status or initiate alternative action within 2 hours.

   F. For the APRM functions only, if one voter channel is inoperable in one trip system, restore the voter channel to OPERABLE status or trip the inoperable channel or the entire trip system within 12 hours. If one voter channel is inoperable in both trip systems, restore the inoperable voter channels to OPERABLE status or initiate alternative action within 6 hours.

2. Scram discharge volume high bypass may be used in SHUTDOWN or REFUEL to bypass scram discharge volume scram and scram pilot air header low pressure scram with control rod block for reactor protection system reset.

3. (Deleted)

4. Bypassed when turbine first stage pressure is less than 154 psig.

5. IRMs are bypassed when APRMs are onscale and the reactor mode switch is in the RUN position.

6. The design permits closure of any two lines without a scram being initiated.

NOTES FOR TABLE 3.1.A (Cont'd)

7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be OPERABLE:

   A. Mode switch in SHUTDOWN

   B. Manual scram

   C. High flux IRM

   D. Scram discharge volume high level

   E. (Deleted)

   F. Scram pilot air header low pressure

8. Not required to be OPERABLE when primary containment integrity is not required.

9. (Deleted)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

11. The same three (3) required APRM channels are shared by both RPS trip systems.

12. Any combination of APRM upscale or inoperative trips from two different (non-bypassed) APRMs will trip all of the 2/4 voter units.

13. Less than the required minimum number of OPERABLE LPRMs will cause an instrument channel inoperative alarm.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first stage pressure is greater than or equal to 154 psig.

NOTES FOR TABLE 3.1.A (Cont'd)

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

20. (Deleted)

21. In the REFUEL Mode unless adequate shutdown margin has been demonstrated per Specification 3.3.A.1, whenever any control rod is withdrawn from a core cell containing one or more fuel assemblies, shorting links shall be removed from the RPS circuitry to enable the Source Range Monitor (SRM) noncoincidence high-flux scram function. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the SRMs.

22. The three required IRMs per trip channel is not required in the Shutdown or REFUEL Modes if at least four IRMs (one in each core quadrant) are connected to give a noncoincidence, High Flux scram. The removal of four (4) shorting links is required to provide noncoincidence high-flux scram protection from the IRMs.

23. A channel may be placed in an INOPERABLE status for up to 4 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

24. The Average Power Range Monitor scram function is varied (Reference Figure 2.1-1) as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with 2.1.A.

25. The APRM flow-biased neutron flux signal is fed through a time constant circuit of approximately 6 seconds. This time constant may be lowered or equivalently removed (no time delay) without affecting the operability of the flow-biased neutron flux trip channels. The APRM fixed high neutron flux signal does not incorporate the time constant but responds directly to instantaneous neutron flux.

BFN
Unit 2

TABLE 4.1.A
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

	Group (2)	Functional Test	Minimum Frequency(3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
Inoperative	C	Trip Channel and Alarm (4)	Once/Week During Refueling and Before Each Startup
APRM			
High Flux (15% Scram)		Trip Output Relays (4)(5) 2/4 Voter Logic (10)	Every 6 Months (9) Each Refueling Outage
High Flux (Flow Biased)		Trip Output Relays (4)(6) 2/4 Voter Logic (10)	Every 6 Months Each Refueling Outage
High Flux (Fixed Trip)		Trip Output Relays (4)(5) 2/4 Voter Logic (10)	Every 6 Months Each Refueling Outage
Inoperative		Trip Output Relays (4)(5) 2/4 Voter Logic (10)	Every 6 Months Each Refueling Outage
2/4 Trip Voter		Trip Scram Contactors (11)	Once/Week
High Reactor Pressure (PIS-3-22AA, BB, C, D)	B	Trip Channel and Alarm (7)	Once/Month
High Drywell Pressure (PIS-64-56 A-D)	B	Trip Channel and Alarm (7)	Once/Month
Reactor Low Water Level (LIS-3-203 A-D)	B	Trip Channel and Alarm (7)	Once/Month

NOTES FOR TABLE 4.1.A

1. Initially the minimum frequency for the indicated tests shall be once per month.

2. A description of the three groups is included in the Bases of this specification.

3. Functional tests are not required when the systems are not required to be OPERABLE or are operating (i.e., already tripped). If tests are missed, they shall be performed prior to returning the systems to an OPERABLE status.

4. This instrumentation is exempted from the instrument channel test definition. This instrument channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

5. The channel functional test shall include both the APRM channels and the 2/4 voter channels.

6. The channel functional test shall include both the APRM channels and the 2/4 voter channels plus the flow input function, excluding the flow transmitters.

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

8. The functional test frequency decreased to once every three months to reduce challenges to relief valves per NUREG 0737, Item II.K.3.16.

9. Not required to be performed when entering the STARTUP/HOT STANDBY Mode from RUN Mode until 12 hours after entering the STARTUP/HOT STANDBY Mode.

10. Functional test consists of simulating APRM trip conditions at the APRM channel outputs to check all combinations of two tripped inputs to the 2/4 voter logic in each voter channel.

11. Functional test consists of manually tripping the 2/4 voter trip output, one voter channel at a time, to demonstrate that each scram contactor for each RPS trip system channel (A1, A2, B1 and B2) operates and produces a half-scram.

TABLE 4.1.B
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency(2)
IRM High Flux	C	Comparison to APRM on Controlled Startups (6)	Note (4)
APRM High Flux			
Output Signal		Heat Balance	Once/7 Days
Flow Bias Signal		Calibrate Flow Bias Signal (7)	Once/Operating Cycle
LPRM Signal		TIP System Traverse (8)	Every 1000 Effective Full Power Hours
High Reactor Pressure (PIS-3-22 AA, BB, C, D)	B	Standard Pressure Source	Once/6 Months (9)
High Drywell Pressure (PIS-64-56 A-D)	B	Standard Pressure Source	Once/18 Months (9)
Reactor Low Water Level (LIS-3-203 A-D)	B	Pressure Standard	Once/18 Months (9)
High Water Level in Scram Discharge Volume			
Float Switches (LS-85-45-C-F)	A	Calibrated Water Column	Once/18 Months
Electronic Level Switches (LS-85-45 A, B, G, H)	B	Calibrated Water Column	Once/18 Months (9)
Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
Turbine First Stage Pressure Permissive (PIS-1-81 A&B, PIS-1-91 A&B)	B	Standard Pressure Source	Once/18 Months (9)
Turbine Stop Valve Closure	A	Note (5)	Note (5)
Turbine Control Valve Fast Closure on Turbine Trip	A	Standard Pressure Source	Once/Operating Cycle
Low Scram Pilot Air Header Pressure (PS 85-35 A1, A2, B1, & B2)	A	Standard Pressure Source	Once/18 Months

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
3(1)	APRM Upscale (Flow Bias)	(2)
3(1)	APRM Upscale (Startup Mode) (8)	≤12%
3(1)	APRM Downscale (9)	≥3%
3(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Power Bias) Low Power Range (13) Intermediate Power Range (13) High Power Range (13)	 (14) (14) (14)
2(7)	RBM Downscale (9)(13)	(15)
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)
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, RBM or APRM channel nor more than two 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. The RBM need not be OPERABLE if either of the following two conditions is met:

      (1) Reactor thermal power is ≥90 percent of rated and MCPR is ≥1.40, or

      (2) Reactor thermal power is <90 percent of rated and MCPR is ≥1.70.

NOTES FOR TABLE 3.2.C (Cont'd)

d. Two RBM channels are provided and only one of these may be bypassed from 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.

e. 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 the required minimum number of LPRM inputs.

        (3) APRM module unplugged.

        (4) Self-test detected critical fault.

    c. RBM

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

        (2) RBM module unplugged.

        (3) RBM fails to null.

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

        (5) Self-test detected critical fault.

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

NOTES FOR TABLE 3.2.C (Cont'd)

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

13. The RBM rod block trip setpoints and applicable power ranges are specified in the CORE OPERATING LIMITS REPORT (COLR).

14. Less than or equal to the setpoint allowable value specified in the COLR.

15. Greater than or equal to the setpoint allowable value specified in the COLR.

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TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test		Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1)	(13)	once/operating cycle	once/day (8)
APRM Upscale (Startup Mode)	(1)	(13)	once/operating cycle	once/day (8)
APRM Downscale	(1)	(13)	once/operating cycle	once/day (8)
APRM Inoperative	(1)	(13)	N/A	once/day (8)
RBM Upscale (Power Bias)	(1)	(13)	once/operating cycle	N/A
RBM Downscale	(1)	(13)	once/operating cycle	N/A
RBM Inoperative	(1)	(13)	N/A	N/A
IRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)		once/operating cycle (12)	N/A
IRM Inoperative	(1)(2)	(13)	N/A	N/A
SRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
SRM Inoperative	(1)(2)	(13)	N/A	N/A
Rod Block Logic	(16)		N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter		once/18 months	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter		once/18 months	N/A

NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.K

1. For IRMs and SRMs functional tests shall be performed once per month. For APRMs and RBMs functional tests shall be performed once per 6 months.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be OPERABLE or are tripped.

9. Calibration frequency shall be once/year.

10. Deleted

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.K (Cont'd)

14. (Deleted)

15. (Deleted)

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

20. (Deleted)

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. (Deleted)

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

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 generate a trip signal to block rod withdrawal if the monitored power level exceeds a preset value. The trip logic for this function is 1-out-of-n: e.g., any trip on one of four APRMs, eight IRMs, or four SRMs will result in a rod block.

When the RBM is required, the minimum instrument channel requirements apply. These 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 provides a trip signal for blocking rod withdrawal when average reactor thermal power exceeds pre-established limits set to prevent scram actuation.

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

4.2 BASES (Cont'd)

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1-out-of-n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the reactor and refueling zones which initiate building isolation and standby gas treatment operation are arranged such that two sensors high (above the high level setpoint) in a single channel or one sensor downscale (below low level setpoint) or inoperable in two channels in the same zone will initiate a trip function. The functional testing frequencies for both the channel functional test and the high voltage power supply functional test are based on a Probabilistic Risk Assessment and system drift characteristics of the Reactor Building Ventilation Radiation Monitors. The calibration frequency is based upon the drift characteristics of the radiation monitors.

The automatic pressure relief instrumentation can be considered to be a 1-out-of-2 logic system and the discussion above applies also.

The RCIC and HPCI system logic tests required by Table 4.2.B contain provisions to demonstrate that these systems will automatically restart on a RPV low water level signal received subsequent to a RPV high water level trip.

The electronic instrumentation comprising the APRM rod block and Rod Block Monitor functions together with the recirculation flow instrumentation for flow bias purposes is monitored by the same self-test functions as applied to the APRM function for the RPS. The functional test frequency of every six months is based on this automatic self-test monitoring at 15 minute intervals and on the low expected equipment failure rates. Calibration frequency of once per operating cycle is based on the drift characteristics of the limited number of analog components, recognizing that most of the processing is performed digitally without drift of setpoint values.

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. (Deleted)	5. (Deleted)

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.I Average Planar Linear Heat Generation Rate

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

J. Linear Heat Generation Rate (LHGR)

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

SURVEILLANCE REQUIREMENTS

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

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

J. Linear Heat Generation Rate (LHGR)

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

LIMITING CONDITIONS FOR OPERATION

J. Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

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

SURVEILLANCE REQUIREMENTS

J. Linear Heat Generation Rate (LHGR)

4.5.K Minimum Critical Power Ratio (MCPR)

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

2. The operating limit MCPR 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	SURVEILLANCE REQUIREMENTS
3.5 Core and Containment Cooling Systems	4.5 Core and Containment Cooling Systems
L. APRM Setpoints	L. APRM Setpoints
(Deleted)	(Deleted)
M. Core Thermal-Hydraulic Stability	M. Core Thermal-Hydraulic Stability
1. The reactor shall not be operated at a thermal power and core flow inside of Regions I and II of Figure 3.5.M-1.	1. Verify that the reactor is outside of Region I and II of Figure 3.5.M-1:
2. If Region I of Figure 3.5.M-1 is entered, immediately initiate a manual scram.	a. Following any increase of more than 5% rated thermal power while initial core flow is less than 45% of rated, and
3. If Region II of Figure 3.5.M-1 is entered:	b. Following any decrease of more than 10% rated core flow while initial thermal power is greater than 40% of rated.

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

At less than rated power conditions, the rated APLHGR limit is adjusted by a power dependent correction factor, MAPFAC(P). At less than rated flow conditions, the rated APLHGR limit is adjusted by a flow dependent correction factor, MAPFAC(F). The most limiting power-adjusted or flow-adjusted value is taken as the APLHGR operating limit for the off-rated condition.

The flow dependent correction factor, MAPFAC(F), applied to the rated APLHGR limit assures that (1) the 10 CFR 50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions. MAPFAC(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent correction factor, MAPFAC(P), applied to the rated APLHGR limit assures that the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated power conditions. MAPFAC(P) values are provided in the CORE OPERATING LIMITS REPORT.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at $\geq$ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5 BASES (Cont'd)

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

At less than rated power conditions, a power dependent MCPR operating limit, MCPR(P), is applicable. At less than rated flow conditions, a flow dependent MCPR operating limit, MCPR(F), is applicable. The most limiting power dependent or flow dependent value is taken as the MCPR operating limit for the off-rated condition.

The flow dependent limit, MCPR(F), provides the thermal margin required to protect the fuel from transients resulting from inadvertent core flow increases. MCPR(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent limit, MCPR(P), protects the fuel from the other limiting abnormal operating transients, including localized events such as a rod withdrawal error. MCPR(P) values are provided in the CORE OPERATING LIMITS REPORT.

3.5.L. APRM Setpoints

(Deleted)

3.5.M. Core Thermal-Hydraulic Stability

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

Because the probability of thermal-hydraulic oscillations is lower and the margin to the MCPR safety limit is greater in Region II than in Region I of figure 3.5.M-1, an immediate scram upon entry into the region is not necessary. However, in order to minimize the

TABLE 3.2.C
INSTRUMENTATION THAT INITIATES ROD BLOCKS

Minimum Operable Channels Per Trip Function (5)	Function	Trip Level Setting
3(1)	APRM Upscale (Flow Bias)	(2)
3(1)	APRM Upscale (Startup Mode) (8)	≤12%
3(1)	APRM Downscale (9)	≥3%
3(1)	APRM Inoperative	(10b)
2(7)	RBM Upscale (Power Bias)	
	Low Power Range (13)	(14)
	Intermediate Power Range (13)	(14)
	High Power Range (13)	(14)
2(7)	RBM Downscale (9)(13)	(15)
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)
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 counts per second or the above condition is satisfied.

5. During repair or calibration of equipment, not more than one SRM, RBM, or APRM channel nor more than two 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. The RBM need not be OPERABLE if either of the following two conditions is met:

      (1) Reactor thermal power is $\geq$90 percent of rated and MCPR is $\geq$1.40, or

      (2) Reactor thermal power is <90 percent of rated and MCPR is $\geq$1.70.

NOTES FOR TABLE 3.2.C (Cont'd)

d. 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.

e. 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 rot in circuit.

b. APRM

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

(2) Less than the required minimum number of LPRM inputs.

(3) APRM module unplugged.

(4) Self-test detected critical fault.

c. RBM

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

(2) RBM module unplugged.

(3) RBM fails to null.

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

(5) Self-test detected critical fault.

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

NOTES FOR TABLE 3.2.C (Cont'd)

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

13. The RBM rod block trip setpoints and applicable power ranges are specified in the CORE OPERATING LIMITS REPORT (COLR).

14. Less than or equal to the setpoint allowable value specified in the COLR.

15. Greater than or equal to the setpoint allowable value specified in the COLR.

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TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test		Calibration (17)	Instrument Check
APRM Upscale (Flow Bias)	(1)	(13)	once/operating cycle	once/day (8)
APRM Upscale (Startup Mode)	(1)	(13)	once/operating cycle	once/day (8)
APRM Downscale	(1)	(13)	once/operating cycle	once/day (8)
APRM Inoperative	(1)	(13)	N/A	once/day (8)
RBM Upscale (Power Bias)	(1)	(13)	once/operating cycle	N/A
RBM Downscale	(1)	(13)	once/operating cycle	N/A
RBM Inoperative	(1)	(13)	N/A	N/A
IRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)		once/operating cycle (12)	N/A
IRM Inoperative	(1)(2)	(13)	N/A	N/A
SRM Upscale	(1)(2)	(15)	once/3 months	once/day (8)
SRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
SRM Inoperative	(1)(2)	(13)	N/A	N/A
Rod Block Logic	(16)		N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter		once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter		once/operating cycle	N/A

NOTES FOR TABLES 4.2.A THROUGH 4.2.L except 4.2.D AND 4.2.K

1. For IRMs and SRMs functional tests shall be performed once per month. For APRMs and RBMs functional tests shall be performed once per 6 months.

2. Functional tests shall be performed before each startup with a required frequency not to exceed once per week.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

4. Tested during logic system functional tests.

5. Refer to Table 4.1.B.

6. The logic system functional tests shall include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the trip systems.

7. The functional test will consist of verifying continuity across the inhibit with a volt-ohmmeter.

8. Instrument checks shall be performed in accordance with the definition of instrument check (see Section 1.0, Definitions). An instrument check is not applicable to a particular setpoint, such as Upscale, but is a qualitative check that the instrument is behaving and/or indicating in an acceptable manner for the particular plant condition. Instrument check is included in this table for convenience and to indicate that an instrument check will be performed on the instrument. Instrument checks are not required when these instruments are not required to be operable or are tripped.

9. Calibration frequency shall be once/year.

10. (DELETED)

11. Portion of the logic is functionally tested during outage only.

12. The detector will be inserted during each operating cycle and the proper amount of travel into the core verified.

13. Functional test will consist of applying simulated inputs (see note 3). Local alarm lights representing upscale and downscale trips will be verified, but no rod block will be produced at this time. The inoperative trip will be initiated to produce a rod block (SRM and IRM inoperative also bypassed with the mode switch in RUN). The functions that cannot be verified to produce a rod block directly will be verified during the operating cycle.

14. (Deleted)

15. (Deleted)

16. Performed during operating cycle. Portions of the logic is checked more frequently during functional tests of the functions that produce a rod block.

17. This calibration consists of removing the function from service and performing an electronic calibration of the channel.

18. Functional test is limited to the condition where secondary containment integrity is not required as specified in Sections 3.7.C.2 and 3.7.C.3.

19. Functional test is limited to the time where the SGTS is required to meet the requirements of Section 4.7.C.1.a.

20. (Deleted)

21. Logic test is limited to the time where actual operation of the equipment is permissible.

22. (Deleted)

23. (Deleted)

24. This instrument check consists of comparing the thermocouple readings for all valves for consistence and for nominal expected values (not required during refueling outages).

25. During each refueling outage, all acoustic monitoring channels shall be calibrated. This calibration includes verification of accelerometer response due to mechanical excitation in the vicinity of the sensor.

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 generate a trip signal to block rod withdrawal if the monitored power level exceeds a preset value. The trip logic for this function is 1-out-of-n: e.g., any trip on one of four 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 provides a trip signal for blocking rod withdrawal when average reactor thermal power exceeds pre-established limits set to prevent scram actuation.

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 adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

4.2 BASES (Cont'd)

The conclusions to be drawn are these:

1. A 1-out-of-n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the reactor and refueling zones which initiate building isolation and standby gas treatment operation are arranged such that two sensors high (above the high level setpoint) in a single channel or one sensor downscale (below low level setpoint) or inoperable in two channels in the same zone will initiate a trip function. The functional testing frequencies for both the channel functional test and the high voltage power supply functional test are based on a Probabilistic Risk Assessment and system drift characteristics of the Reactor Building Ventilation Radiation Monitors. The calibration frequency is based upon the drift characteristics of the radiation monitors.

The automatic pressure relief instrumentation can be considered to be a 1-out-of-2 logic system and the discussion above applies also.

The RCIC and HPCI system logic tests required by Table 4.2.B contain provisions to demonstrate that these systems will automatically restart on a RPV low water level signal received subsequent to a RPV high water level trip.

The electronic instrumentation comprising the APRM rod block and Rod Block Monitor functions together with the recirculation flow instrumentation for flow bias purposes is monitored by the same self-test functions as applied to the APRM function for the RPS. The functional test frequency of every six months is based on this automatic self-test monitoring at 15 minute intervals and on the low expected equipment failure rates. Calibration frequency of once per operating cycle is based on the drift characteristics of the limited number of analog components, recognizing that most of the processing is performed digitally without drift of setpoint values.

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. (Deleted)	5. (Deleted)

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.I Average Planar Linear Heat Generation Rate

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

J. Linear Heat Generation Rate (LHGR)

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the appropriate LHGR limit provided in the CORE OPERATING LIMITS REPORT.

If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the

SURVEILLANCE REQUIREMENTS

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

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

J. Linear Heat Generation Rate (LHGR)

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

LIMITING CONDITIONS FOR OPERATION

J. Linear Heat Generation Rate (LHGR)

3.5.J (Cont'd)

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

SURVEILLANCE REQUIREMENTS

J. Linear Heat Generation Rate (LHGR)

4.5.K Minimum Critical Power Ratio (MCPR)

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

2. The operating limit MCPR 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	SURVEILLANCE REQUIREMENTS
3.5 Core and Containment Cooling Systems	4.5 Core and Containment Cooling Systems
L. APRM Setpoints	L. APRM Setpoints
(Deleted)	(Deleted)

3.5 BASES (Cont'd)

3.5.I. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit.

At less than rated power conditions, the rated APLHGR limit is adjusted by a power dependent correction factor, MAPFAC(P). At less than rated flow conditions, the rated APLHGR limit is adjusted by a flow dependent correction factor, MAPFAC(F). The most limiting power-adjusted or flow-adjusted value is taken as the APLHGR operating limit for the off-rated condition.

The flow dependent correction factor, MAPFAC(F), applied to the rated APLHGR limit assures that (1) the 10 CFR 50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions. MAPFAC(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent correction factor, MAPFAC(P), applied to the rated APLHGR limit assures that the fuel thermal mechanical design criteria would be met during abnormal operating transients initiated from less than rated power conditions. MAPFAC(P) values are provided in the CORE OPERATING LIMITS REPORT.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated.

The LHGR shall be checked daily during reactor operation at ≥ 25 percent power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25 percent of rated thermal power, the largest total peaking would have to be greater than approximately 9.7 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 percent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

At less than rated power conditions, a power dependent MCPR operating limit, MCPR(P), is applicable. At less than rated flow conditions, a flow dependent MCPR operating limit, MCPR(F), is applicable. The most limiting power dependent or flow dependent value is taken as the MCPR operating limit for the off-rated condition.

The flow dependent limit, MCPR(F), provides the thermal margin required to protect the fuel from transients resulting from inadvertent core flow increases. MCPR(F) values are provided in the CORE OPERATING LIMITS REPORT.

The power dependent limit, MCPR(P), protects the fuel from the other limiting abnormal operating transients, including localized events such as a rod withdrawal error. MCPR(P) values are provided in the CORE OPERATING LIMITS REPORT.

3.5.L. APRM Setpoints

(Deleted)

3.5.M. Core Thermal-Hydraulic Stability

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

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

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

Periodic upscale or downscale LPRM alarms will occur before regional oscillations are large enough to threaten the MCPR safety limit. Therefore, the criteria for initiating a manual scram described in the preceding paragraph are sufficient to ensure that the MCPR safety limit will not be violated in the event that core oscillations initiate while exiting Region II.

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

3.5.N. References

1. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24194A and Addenda.

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

3. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

4.5 Core and Containment Cooling Systems Surveillance Frequencies

The testing interval for the core and containment cooling systems is based on industry practice, quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, in the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable. Complete ADS testing during power operation causes an undesirable loss-of-coolant inventory. To increase the availability of the core and containment cooling system, the components which make up the system, i.e., instrumentation, pumps, valves, etc., are tested frequently. The pumps and motor operated injection valves are also tested in accordance with Specification 1.0.MM to assure their OPERABILITY. A simulated automatic actuation test once each cycle combined with testing of the pumps and injection valves in accordance with Specification 1.0.MM is deemed to be adequate testing of these systems. Monthly alignment checks of valves that are not locked or sealed in position which affect the ability of the systems to perform their intended safety function are also verified to be in the proper position. Valves which automatically reposition themselves on an initiation signal are permitted to be in a position other than normal to facilitate other operational modes of the system.

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

Whenever a CSCS system or loop is made inoperable, the other CSCS systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect they are inoperable. If the function, system, or loop under test or calibration is found inoperable or exceeds the trip level setting, the LCO and the required surveillance testing for the system or loop shall apply.

Average Planar LHGR, LHGR, and MCPR

The APLHGR, LHGR, and MCPR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

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of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlines in (1) t[illegible] DCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I [illegible] 10 CFR Part 50.

6.9.1.6 SOURCE TESTS

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

6.9.1.7 CORE OPERATING LIMITS REPORT

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

(1) The rated APLHGR limit; the Flow Dependent APLHGR Factor, MAPFAC(F); and the Power Dependent APLHGR Factor, MAPFAC(P) for Specification 3.5.I.

(2) The LHGR limit for Specification 3.5.J

(3) The rated MCPR Operating Limit; the Flow Dependent MCPR Operating Limit, MCPR(F); and the Power Dependent MCPR Operating Limit, MCPR(P) for Specification 3.5.K/4.5.K.

(4) The APRM flow biased rod block trip setting for Specification 2.1.A.1.c and Table 3.2.C.

(5) The RBM downscale trip setpoint, high power trip setpoint, intermediate power trip setpoint, low power trip setpoint, and applicable reactor thermal power ranges for each of the setpoints for Table 3.2.C.

6.9.1.7 CORE OPERATING LIMITS REPORT (Cont'd)

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

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

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

6.9.1.8 THE ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

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

Enclosure 1

ATTACHMENT 1

Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3, NEDC-32433P