

Bases Continued:

temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

Operation with one recirculation pump and the equalizer line closed causes flow to bypass the core through idle jet pumps. This loss of core flow is accounted for in the specified setting adjustments for the one-pump, equalizer closed case. Reference the September 7, 1976 License Amendment Request from L O Mayer (NSP) to Victor Stello (USNRC).

The operator will set the APRM neutron flux trip setting no greater than that shown in Figure 2.3.1. However, the actual setpoint can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 18.

- B. APRM Control Rod Block Trips Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the Safety Limit (T.S.2.1.A). This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting,

Bases Continued:

over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. Operation with one recirculation pump and the equalizer line closed causes flow to bypass the core through the idle jet pumps. This loss of core flow is accounted for in the specified setting adjustments for the one-pump, equalizer-closed case. Reference the September 7, 1976 License Amendment Request from L O Mayer (NSP) to Victor Stello (USNRC). When the maximum total peaking factor exceeds the design value, the rod block setting is adjusted in accordance with the formula in Specification 2.3.B. If the APRM rod block setting should require a change due to an abnormal peaking condition, it will be done by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced rod block curve by the reciprocal of the APRM gain change.

The operator will set the APRM rod block trip settings no greater than that shown in Figure 2.3.1. However, the actual setpoint can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on Page 18.

- C. Reactor Low Water Level Scram The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained.

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual setpoint can be as much as 6 inches lower due to the deviations discussed on page 18.

- D. Reactor Low Low Water Level ECCS Initiation Trip Point The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. The design of the ECCS components to meet the above criterion was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could prevent the ECCS components from

Table 3.2.3
Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes in Which Function Must be Operable or Operating and Allowable Bypass Conditions**		Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (Notes 1,6)	Required Conditions*
		Refuel	Startup/Run			
1. <u>SIM</u> a. Upscale b. Detector not fully inserted	$\leq 5 \times 10^5$ cps	X	X(d)	2	1 (Note 3)	A or B or C
		X(a)	X(a)	2	1 (Note 3)	A or B or C
2. <u>IM</u> a. Downscale b. Upscale	$\geq 3/125$ full scale $\leq 108/125$ full scale	X(b)	X(b)	4	2 (Note 4)	A or B or C
		X	X	4	2 (Note 4)	A or B or C
3. <u>AFPM</u> a. Upscale (flow referenced) b. Downscale	Reference Specification 2.3.B $\geq 3/125$ full scale		X	3	1 (Note 7)	D or E
			X	3	1 (Note 7)	D or E

3.2/4.2

2.3 - Continued
 That Initiates Rod Block

Each Operable Low- Power Run	Total No. of Instrument Channels per Trip System	Min. No. of Oper- able or Operating Instrument Channels Per Trip System (Notes 1,6)	Required Conditions*
(c)	1	1 (Note 5)	D or E
(c)	1	1 (Note 5)	D or E

For each function. If the minimum number of
 for one of the two trip systems, this condition
 the operable system is functionally tested

recirculation driving flow in percent.

- (3) Only one of the four SHM channels may be bypassed.
- (4) There must be at least one operable or operating IIM channel monitoring each core quadrant.
- (5) One of the two RIMs may be bypassed for maintenance or testing for periods not in excess of 24 hours in any 30 day period. An RIM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

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Observations of W in Note 2, Page 58
 inconsistent with that provided in Page 11
 Page 11 shows W as % of Design recirculation
 Page 58 is not clear whether W
 is % of design or indicated.

Bases Continued:

- 3.2 The HPCI and/or RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 4 and/or Group 5 valves. The trip settings of 200°F and 150% of HPCI and 300% of RCIC design flows and valve closure times are such that the core will not be uncovered and fission product release will not exceed 10 CFR 100 guidelines.

The instrumentation which initiates ECCS 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.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR remains above the Safety Limit (T.S.2.1.A). The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, eight IRM's, or four SRM's will result in a rod block. The minimum instrument channel requirements for the IRM and RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. See Section 7.3 FSAR.

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

The RBM 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. The trip point is referenced to flow. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked at MCPR greater than the Safety Limit, thus allowing adequate margin. Below 60% power, MCPR remains above the Safety Limit for the worst case withdrawal of a single control rod without rod block action, thus below this level it is not required. This subject is discussed in General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application, NEDO-10958. Requiring at least half of the normal APRM inputs from each level to be operable assures that the RBM response will be adequate to prevent rod withdrawal errors. Operation with one recirculation pump and the equalizer line closed causes flow to bypass the core through idle jet pumps. This loss of core flow is accounted for in the specified setting adjustments for the one-pump, equalizer closed case. Reference the September 7, 1976 License Amendment Request from L O Mayer (NSP) to Victor Stello (USNRC).

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the Safety Limit (T.S.2.1.A).

A downscale indication of an APRM or IRM 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 downscale trips are set at 3/125 of full scale.

3.0 LIMITING CONDITIONS FOR OPERATION

I. Recirculation System

1. Except as specified in 3.5.1.2 below, whenever irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be operable.
2. The recirculation system cross tie valve interlocks may be inoperable if at least one cross tie valve is maintained fully closed.
3. Operation with one recirculation pump (equalizer valve closed) is permitted provided that within 24 hours from the time one pump operation commences the appropriate adjustments to limits specified in Specifications 2.3.A.1, 2.3.B, 3.2.C and 3.11.A are incorporated. If the settings cannot be adjusted or two-pump operation restored by the end of 24 hours, an orderly reactor shutdown shall be initiated.

3.5/4.5

4.0 SURVEILLANCE REQUIREMENTS

I. Recirculation System

1. Once per month, when irradiated fuel is in the reactor with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be demonstrated to be operable by verifying that the cross tie valves cannot be opened using the normal control switch.
2. When a recirculation system cross tie valve interlock is inoperable, the position of at least one fully closed cross tie valve shall be recorded daily.

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Bases Continued 3.5

G. Emergency Cooling Availability

The purpose of Specification G is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment cooling subsystems may be out of service. Specification 3.5.G.3 allows all core and containment cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.G.4 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.i. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

H. Deleted

I. Recirculation System

The capacity of the Emergency Core Coolant System is based on the potential consequences of a double ended recirculation line break. Such a break involves 3.9 sq. ft. when the cross tie valves are closed and 5.3 sq. ft. when the cross tie valves are open. Specification 3.11.A is based on an ECCS evaluation assuming a break area of 3.9 sq. ft.; the limitations of 3.11.A do not apply to the larger break area. Therefore, at least one cross tie valve must remain closed at all times to reduce the potential break area as required by Specifications 3.5.I.1 and 2.

An analysis of one-pump operation (equalizer valve closed) identifies certain limitations peculiar to that mode of operation. Reference the September 7, 1976 License Amendment Request from L G Mayer (NSP) to Victor Stello (USNRC). Operation with only one pump is not a normal mode; it will generally involve a forced outage of equipment. There may be insufficient time to make adjustments to the RBM and APRM flow referenced rod block and scram prior to commencing one-pump operation. The reduction in power with the reduced core flow will cause the APLHGR to reduce accordingly, naturally moving in the direction of the new limit. Specification 3.5.I.3 allows 24 hours before these new limits are required to be implemented.

3.0 LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During two-pump power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.11.1; during one-pump operation (equalizer valve closed) the respective limit shall be 0.86 times the value shown in these figures. If at any time during operation it is determined that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is 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.

4.0 SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLIES

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq 25\%$ rated thermal power.

Bases 3.11

A. 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 10CFR50, 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 cladding temperature by less than $\pm 20^{\circ}\text{F}$ 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 10CFR50 Appendix K limit. The limiting value for APLHGR is given by this specification.

Operation with one reactor recirculation pump and the equalizer line closed causes flow to bypass the core through the idle jet pumps. This loss of core flow results in an increased peak clad temperature during a LOCA. The limits on APLHGR are reduced correspondingly. Reference the September 7, 1976 License Amendment Request from L O Mayer (NSP) to Victor Stello (USNRC).

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding APLHGR limits in such cases need not be reported.

B. 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 power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation and axial gaps between core bottom and top and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding LHGR limits in such cases need not be reported.