2.0 SAFETY LIMITS and LIMITING SAFETY SYSTEM SETTINGS

BASES

INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.06. MCPR greater than 1.06 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding integrity Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

for two recirculation loop operation and 1.07 for single-loop operation

2.1 SAFETY LIMITS

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/h, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/h. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

Bases Table B2.1.2-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT*

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow, Two Recirc. Loop operation	2.5
One Recirc. Loop operation	6.0
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings, Two Recirc. Loop Operation	6.3
One Recirc. Loop Operation	6.8
R Factor	1.5
Critical Power	3.6

* The uncertainty analysis used to establish the core wide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core. The values given apply to both one and two recirculation loop operation except as noted.

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3. The limits of Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 shall be reduced to a value of 0.84 times the two recirculation loop operation limit when in single recirculation loop operation.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, or 3.2.1-3, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits determined from Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 (appropriately adjusted per specification 3.2.1):

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

   a) Place the recirculation flow control system in the ~~Master~~ Local Manual (Position control) mode, and

   b) Reduce THERMAL POWER to ≤ 50% of RATED THERMAL POWER, and,

   c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.07 per Specification 2.1.2, and,

   ~~d) Increase the MCPR Limiting Condition for Operation by 0.01 per Specification 3.2.3, and,~~ delete

   d.) ~~e)~~ Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value of 0.84 times the two recirculation loop operation limit per Specification 3.2.1, and,

   e.) ~~f)~~ Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 2.2.1, 3.2.2, and 3.3.6.

   INSERT 1

~~2. At least once per 12 hours:~~

   ~~a) Verify that the APRM flux noise averaged over 30 minutes does not exceed 5% peak to peak; otherwise, reduce the recirculation loop flow until the APRM flux noise is less than the 5% peak to peak limit, and,~~

   ~~b) Verify that the core plate ΔP noise does not exceed 1 psi peak to peak; otherwise, reduce the recirculation loop flow until the ΔP noise is less than the 1 psi limit.~~

2. The provisions of Specification 3.0.4 are not applicable.

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INSERT 1

f.) Reduce the volumetric flow rate of the operating recirculation loop to $\leq$ 45,000** gpm

g.) Perform surveillance requirement 4.4.1.1.3 if thermal power is $\leq$ 30%*** of rated thermal power or the recirculation loop flow in the operating loop is $\leq$ 50%*** of rated loop flow.

h.) Reduce recirculation loop flow in the operating loop until the APRM flux noise does not deviate from the established flux noise patterns at 50% power by more than 50% when averaged over a 30 minute period.

i.) Reduce recirculation loop flow in the operating loop until the core plate Δ P noise does not deviate from the established core plate Δ P noise patterns at 50% power by more than 50%.

INSERT 2

** This value represents the design volumetric recirculation loop flow which produces 100% core flow at 100% thermal power. The actual value to be applied will be determined during the Startup Test Program.

*** Initial values. Final values to be determined during Startup Testing based upon the threshold thermal power and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least ~~HOT~~ STARTUP ~~SHUTDOWN~~ within ~~the next~~ 6 hours and in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure (at the hydraulic control unit), and

b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing.

Insert 3 →

*See Special Test Exception 3.10.4.

INSERT 2

Insert 3

4.4.1.1.2 With one reactor coolant system recirculation loop not in service, at least once per 024 hours verify:

a.) Reactor THERMAL POWER is ≤ 50% of RATED THERMAL POWER

b.) The recirculation flow control system is in the LOCAL manual (Position control) mode

c.) The volumetric flow rate of the operating loop is ≤ ~~21,900~~ 45,000 gpm.**

d.) The APRM flux noise when averaged over a 30 minute period is less than 150% ~~peak-to-peak~~ of the established APRM flux noise patterns.

e.) The core plate ΔP noise is less than ~~1 psi peak-to-peak~~ 150% of the established core plate ΔP noise patterns.

4.4.1.1.3 With one reactor coolant system recirculation loop not in operation ~~per ACTION 3.4.1.1.a.3~~, within no more than 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30% *** ~~of RATED THERMAL POWER or the recirculation~~ loop flow in the operating loop is ≤ 50% *** of rated loop flow:

a. < 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b. < 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. < 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements, 4.4.1.1.3.b and c, do not apply when the loop not in operation is isolated from the reactor pressure vessel.

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when ~~the~~ both recirculation loops are in service and operating at the same flow control valve position.

a. The indicated recirculation loop flow differs by more than 10% from the established flow control valve position-loop flow characteristics for 2 recirculation pump operation.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements with 2 recirculation pump operation.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established 2 recirculation pump patterns by more than 10%.

4.4.1.2.2 ~~In addition to Surveillance Requirement 4.4.1.2.1,~~ During single recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE at least once per 24 hours by verifying that ~~the pressure drop for one jet pump in a loop does not vary from the mean of all jet pumps in that loop by more than 5%.~~ no two of the following conditions occur:

INSERT #4

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INSERT 4

a.) The indicated recirculation loop flow in the active loop differs by more than 10% from the established single pump flow control valve position - loop flow characteristics.

b.) The indicated total core flow differs by more than 10% from the established total core flow value from single recirculation loop flow measurements.

c.) The indicated difference-to-lower plenum differential pressure of any individual jet pump differs from established single pump patterns by more than 10%.

REACTOR COOLANT SYSTEM

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* with two recirculation pumps in operation

ACTION:

With the recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Declare one of the recirculation loops ~~with the lower flow not in operation~~ inoperable and take the ACTION required by Specification 3.4.1.1.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the safety analyses, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than ~~1.06~~ the fuel cladding safety limit during the limiting power transient analyzed in Section 15.2 of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than ~~1.06~~ {fuel cladding safety limit}. The occurrence of scram times longer then those specified should be viewed as an indication of a ~~systemic~~ systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) 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 dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady-state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figures 3.2.1-1, 3.2.1-2 and 3.2.1-3 *for two recirculation loop operation.* INSERT 5

The calculational procedure used to establish the APLHGR shown on Figures 3.2.1-1, 3.2.1-2, and 3.2.1-3 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR Part 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

INSERT 5

These values are to be multiplied by a factor of 0.84 for single loop operation. The multiplier for single loop operation is determined from comparison of the limiting analysis between one loop and two loop operation.

The calculational procedures and significant input parameters are documented in FSAR Section 6A. The reduction factor derived for single-loop operation is justified in ~~Amendment ___, to~~ the FSAR.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than ~~1.06~~ the fuel cladding safety limit or that > 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

for both two loop and single-loop operation

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3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR ~~of 1.06~~, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR ~~of 1.06~~, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in Figure 3.2.3-1.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-1 that are input to a GE-core dynamic behavior transient computer program. The code used to evaluate pressurization events is described in NEDO-24154(1) and the program used in nonpressurization events is described in NEDO-10802(2). The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic TASC code described in NEDE-25149(3). The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor of Figure 3.2.3-1 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated. The K_f factors were derived using THERMAL POWER and core flow corresponding to 105% of rated steam flow.

The K_f factors were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

INSERT 6

~~Operation with one reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated, and determined to be acceptable.~~

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. INSERT 7

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 12 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

INSERT 6

An assessment has been performed on the impact of single recirculation loop operation upon plant safety. Results show that single-loop operation is permitted providing the Fuel Cladding Safety Limit is increased, as noted by Specification 2.1.2, APRM scram, rod block and RBM setpoints are adjusted as noted in Specification 3.2.2, Table 2.2.1-1 and Table 3.3.6.2; MAPLGHR limits are decreased by the factor given in Specification 3.2.1 and surveillance on the volumetric flow rate of the operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below (30%) Thermal Power or (50%) rated recirculation loop flow, is to mitigate the undue thermal stress on vessel nozzles, recirculation pump, and vessel bottom head.

INSERT 7

In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.