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(continued)

1.1 Definitions

LEAKAGE (continued)	c. Total LEAKAGE Sum of the identified and unidentified LEAKAGE; d. Pressure Boundary LEAKAGE LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
LINEAR HEAT GENERATION RATE	LINEAR HEAT GENERATION RATE (LHGR) shall be the power generation in an arbitrary length of fuel rod, usually six inches. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)*

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 24% RTP <u>AND</u> 24 hours thereafter

*This Specification is effective starting from Hatch 2/Cycle 19.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation,

OR

One recirculation loop shall be in operation with the following limits applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," single loop operation limits specified in the COLR; and

d. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitor Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

---NOTE---
A single submittal may be made for a multiple unit station. The submittal should combine sections 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 Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1) The Average Planar Linear Heat Generation Rate for Specification 3.2.1.

2) The Minimum Critical Power Ratio for Specification 3.2.2.

3) The Linear Heat Generation Rate for Specification 3.2.3.

(continued)

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating LOCA and normal operation that determine the APLHGR limits are presented in References 1, 3, 4, 6, 9, and 10.

APLHGR limits are developed as a function of exposure and operating states to ensure adherence to 10 CFR 50.46 during the limiting LOCA (Refs. 6, 7, 9, and 10).

LOCA analyses are performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 10. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by an assumed conservatively small local peaking factor.

Some off-rated operating states require the reduction or set down of the rated APLHGR limit through multiplier factors (MAPFACs). A flow dependent multiplier, $MAPFAC_f$, is necessary at core flows below 61% to provide protection for LOCA events (Ref. 12). For single recirculation loop operation, the $MAPFAC_f$ multiplier is limited to a maximum value specified in the Core Operating Limits Report (COLR). This maximum limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).
LCO	The APLHGR limits specified in the COLR are the result of the LOCA analyses. The limit is determined by multiplying the $MAPFAC_f$ factor times the exposure dependent APLHGR limits. For single recirculation loop operations, the $MAPFAC_f$ multiplier is limited to a maximum value specified in the Core Operating Limits Report (COLR).
APPLICABILITY	The APLHGR limits are primarily derived from fuel design evaluations and LOCA analyses that are assumed to occur at high power levels. Design calculations (Ref. 7) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 24% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.
ACTIONS	<u>A.1</u> If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the LOCA may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a LOCA occurring simultaneously with the APLHGR out of specification.

(continued)

BASES

ACTIONS (continued)

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply.

To achieve this status, THERMAL POWER must be reduced to < 24% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 24% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 24% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 24% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel," (revision specified in the COLR).
2. (Not used)
3. FSAR, Chapter 6.
4. FSAR, Chapter 15.
5. (Not used)
6. NEDC-32749P, "Extended Power Uprate Safety Analysis Report for Edwin I. Hatch Units 1 and 2," July 1997.
7. NEDC-30474-P "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program for E.I. Hatch Nuclear Plant, Units 1 and 2," December 1983.
8. (Not used)

(continued)

BASES

REFERENCES (continued)

9. NEDC-32720P, "Hatch Units 1 and 2 SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," March 1997.

10. GE-NE-0000-0000-9200-02P, "Hatch Units 1 and 2 ECCS-LOCA Evaluation for GE-14," March 2002.

11. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

12. Letter from Global Nuclear Fuel, M. E. Harding to E. B. Gibson, January 22, 2004, "Plant Hatch Technical Specification Modification to include LHGR."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel thermal-mechanical design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs), and to ensure that the peak clad temperature (PCT) during postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials into the reactor coolant. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating AOOs and normal operation that determine the LHGR limits are presented in Reference 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection systems) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations include:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet and cladding.

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit and certain other fuel design limits described in reference 1 are not exceeded during

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

continuous operation with LHGRs up to the operating limit specified in the Core Operating Limits Report (COLR). The analysis also includes allowances for short-term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

LHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Refs. 4 and 5). Off-rated operating states require the reduction or set down of the rated LHGR limit through multiplier factors (LHGRFACs) (Ref. 9).

Flow dependent multipliers, $LHGRFAC_f$, are determined (Ref. 5) using the three dimensional BWR simulator code (Ref. 6) to analyze slow flow runout transients. The flow dependent multiplier is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $LHGRFAC_p$, also are generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $LHGRFAC_p$ limits are provided for operation at power levels between 24% RTP and the previously mentioned bypass power level.

The exposure dependent LHGR limits are reduced by $LHGRFAC_p$ and $LHGRFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 7.

LOCA analyses are performed to ensure that the above determined LHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. See Section B 3.2.1 for more details.

For single recirculation loop operation, the LHGR operating limit is as specified in the COLR, and the LHGRFAC multiplier is limited to a maximum as specified in the COLR. The maximum limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

The LHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 8).

(continued)

BASES (continued)

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR limit calculated to cause a 1% fuel cladding plastic strain as well as the other design limits described in Ref. 1. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $LHGRFAC_f$ and $LHGRFAC_p$ factors times the exposure dependent LHGR limits. These values are specified in the COLR. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent LHGR limit by the smaller of either $LHGRFAC_f$, $LHGRFAC_p$, and a maximum value allowed during single loop operation as specified in the COLR.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 24% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the specification is only required when the reactor is operating at ≥ 24% RTP.

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or LOCA occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 24% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 24% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 24% RTP and every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slow changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 24% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."
2. FSAR, Chapter 15 (Unit 2).
3. NUREG-0800, Section II.A.2(g), Revision 2, July 1981.
4. NEDC-32749P, "Extended Power Uprate Safety Analysis Report for Edwin I. Hatch Units 1 and 2," July 1997.
5. NEDC-30474-P, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program for E. I. Hatch Nuclear Plant, Units 1 and 2," December 1983.
6. NRC approval of "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, "GESTAR II"—Implementing Improved GE Steady-State Methods (TAC No. MA6481)," November 10, 1999.
7. NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors," August 1986, and NEDE-24154-P-A, Supplement 1, Volume 4, Revision 1, February 2000.
8. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
9. Letter from Global Nuclear Fuel, M. E. Harding to E. B. Gibson, January 22, 2004, "Plant Hatch Technical Specification Modification to include LHGR."

BASES

BACKGROUND (continued)

effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 55 to 100% of RTP) without having to move control rods and disturb desirable flux patterns. In addition, core flow as a function of core thermal power, is usually maintained such that core thermal-hydraulic oscillations do not occur. These oscillations can occur during two-loop operation, as well as single-loop and no-loop operation. Plant procedures include requirements of this LCO as well as other vendor and NRC recommended requirements and actions to minimize the potential of core thermal-hydraulic oscillations.

Each recirculation loop is manually started from the control room. The MG set provides regulation of individual recirculation loop drive flows. The flow in each loop is manually controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Coolant Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgment. The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational occurrences (AOOs) (Ref. 2), which are analyzed in Chapter 15 of the FSAR.

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the LHGR and APLHGR requirements are modified accordingly (Refs. 1 and 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The MCPR setpoints for single loop operation are specified in the COLR. The APRM Simulated Thermal Power - High setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 4).

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied.

With only one recirculation loop in operation, modifications to the required APLHGR limits [(LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"], MCPR limits [LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"], LHGR limits [LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"], and APRM Simulated Thermal Power - High setpoint (LCO 3.3.1.1) must be applied to allow continued operation consistent with the assumptions of References 1 and 3.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

(continued)

BASES (continued)

ACTIONS

A.1

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA or AOO occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses or the AOO analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident or AOO sequence.

The 24 hour Completion Time is based on the low probability of an accident or AOO occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump.

B.1

With any Required Action and associated Completion Time of Condition A not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of Design Basis Accidents and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loops are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered not in operation. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

SR 3.4.1.2

(Not used.)

REFERENCES

1. NEDC-32720P, "E. I. Hatch Nuclear Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," March 1997.

2. FSAR, Section 5.5.1.4.

3. NEDO-24205, "E. I. Hatch Nuclear Plant Units 1 and 2 Single-Loop Operation," August 1979.

4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.