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SOUTHERN COMPANY
Energy to Serve Your World℠

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NL-03-1842

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Technical Specifications Revision to Include Monitoring of Linear Heat Generation Rate

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90(c)(1), Southern Nuclear Operating Company hereby proposes changes to the Hatch Units 1 and 2 Technical Specifications, Appendix A to Operating Licenses DPR- 57 and NPF – 5, respectively. This submittal adds a Limiting Condition for Operation (LCO) for the Linear Heat Generation Rate (LHGR). The new LCO will be included in Section 3.2, Power Distribution Limits. Before the proposed change, the LHGR was included as part of the Average Planar Linear Heat Generation Rate (APLHGR) monitoring, LCO 3.2.1. Supporting changes are also proposed for the recirculation loop LCO, Section 5.6.5, and to appropriate Bases.

Enclosure 1 provides a description and justification of the proposed change. Enclosure 2 contains the 10 CFR 50.92 evaluation and the justification for the categorical exclusion from performing an environmental assessment. Enclosure 3 provides the marked-up TS and Bases pages. Enclosure 4 provides the clean typed TS and Bases pages.

In accordance with the requirements of 10 CFR 50.91, a copy of this letter and all applicable enclosures will be sent to the designated state official of the Environmental Protection Division of the Georgia Department of Natural Resources.

To support start-up following the Unit 1 Spring 2004 refueling outage, SNC requests that NRC review and approve this amendment no later than March 10, 2004. SNC further requests that the amendments be issued with implementation at the start of Hatch 1/Cycle 22 for Unit 1 and at the start of Hatch 2/Cycle 19 for Unit 2.

A001

Mr. H. L. Sumner, Jr. states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

Lewis Sumner

H. L. Sumner, Jr.

Sworn to and subscribed before me this 3 *day of* October, *2003.*

Valerie A. O'Brien

Notary Public

My commission expires: 4-28-07

HLS/OCV/daj

Enclosures: Enclosure 1 – Description and Justification
Enclosure 2 – No Significant Hazards Evaluation and Environmental Assessment
Enclosure 3 – Marked-up Technical Specifications and Bases Pages
Enclosure 4 – Clean Typed Technical Specifications and Bases Pages

cc: Southern Nuclear Operating Company
Mr. J. D. Woodard, Executive Vice President
Mr. G. R. Frederick, General Manager – Plant Hatch
Document Services RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. S. D. Bloom, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

State of Georgia
Mr. L. C. Barrett, Commissioner – Department of Natural Resources

Enclosure 1
Edwin I. Hatch Nuclear Plant
Technical Specifications Revision to Include Monitoring of Linear Heat Generation Rate

Description and Justification

This proposed Technical Specifications (TS) amendment adds a limiting condition for operation (LCO) which will address plant monitoring and limitation requirements for the Linear Heat Generation Rate (LHGR). SNC proposes to add the new LCO to the power distribution section of the TS as LCO 3.2.3. The LCO will contain Required Action Statements (RAS) as well as Surveillance Requirements (SR). Like the existing Minimum Critical Power Ratio (MCPR) and Average Planar Linear heat Generation Rate (APLHGR) LCOs, the proposed LHGR limit will be applicable at a reactor power greater than 24%. Consequently, inability to meet the limit within the allotted Completion Time will require reducing power to below 24%. (Note: The LHGR specification was made applicable above 24% since the Measurement Uncertainty Power Uprate changes the thermal power applicability for thermal limits from 25% to 24% power.)

Additionally, a line item is being added to LCO 3.4.1, Recirculation Loops Operating, noting that the LHGR limit may need to be adjusted when operating with only one recirculation loop in service. Finally, a requirement is being added to Section 5.6.5 to require LHGR limits to be included in the Core Operating Limits Report (COLR).

The LHGR is defined as the power generated in an arbitrary length of fuel rod. The process computer monitors LHGR on a six-inch segment (node) basis for each fuel rod, with values reported in units of kilowatts per foot (kw/ft). The LHGR is monitored and limited to ensure that fuel thermal-mechanical design limits (e.g., 1% plastic strain on the cladding) which prevent fuel cladding failure are not exceeded during normal operation or Anticipated Operational Occurrences (AOOs). LHGR is also limited such that the peak clad temperature and other limits specified by 10 CFR 50.46 are not exceeded during a LOCA event.

The APLHGR is the average of the LHGRs for all the fuel rods in a fuel assembly at a specific elevation. APLHGR is monitored and limited to prevent exceeding a peak cladding temperature of 2200 °F and other limits specified by 10 CFR 50.46 following a LOCA. A temperature of 2200 °F could result in a brittle fracture of the clad upon subsequent rewetting by coolant from an emergency core cooling system.

From a historical perspective, LHGR was once included in the Hatch TS. Amendment 19 to GESTAR II allowed LHGR to be monitored as part of the APLHGR TS. Accordingly, both Hatch Units removed LHGR from the TS at the time of our conversion to the STS.

Currently, the LHGR is included in LCO 3.2.1 as part of the APLHGR. Consequently, monitoring and adhering to the APLHGR limits ensures compliance to the LHGR limit.

Recently, Global Nuclear Fuel (GNF) has made improvements in nuclear methods, identified as PANAC11 and TGBLA06. The new methods were approved by NRC in amendment 26 to GESTAR II, and included in GESTAR II revision 14, June, 2000.

Using these new methods, improved accuracy in core monitoring calculations can be achieved when separately monitoring LHGR and APLHGR. As a result, SNC proposes to add the LHGR back to the TS.

Adding the LHGR to the TS will not require changes to the APLHGR specification 3.2.1. However, as previously mentioned, the LHGR provides protection against fuel cladding damage for normal operations, AOOs, and LOCA; the APLHGR provides protection only for LOCA events. Therefore the APLHGR Bases section is being revised to remove discussions on AOOs. Other minor changes are also being made to the Bases to include new references and eliminate other references made obsolete by these changes. The current version of the BWR/4 Standard Technical Specifications (STS) contains LHGR as an optional specification.

Enclosure 2
Edwin I. Hatch Nuclear Plant
Technical Specifications Revision to Include Monitoring of Linear Heat Generation Rate

No Significant Hazards and Environmental Assessment

Proposed Change

A new Technical Specification (TS) Limiting Condition for Operation (LCO) is being proposed to add Linear Heat Generation Rate (LHGR) to section 3.2, "Power Distribution Limits".

10 CFR 50.92 Evaluation

In 10 CFR 50.92, the Nuclear Regulatory Commission (NRC) provides the following standards to be followed in determining the existence of a significant hazards consideration:

...a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22, or for a test facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in the margin of safety.

Southern Nuclear Operating Company (SNC) has reviewed the proposed amendment request and determined that its adoption does not involve a significant hazards consideration based on the following discussion:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed addition of LCO 3.2.3 and supporting Bases are being made to support new modeling improvements in core monitoring. This change is administrative in nature in that it does not involve, require, or result from any physical change to the plant, including the reactor core or its fuel. The addition of LCO 3.2.3 and Bases B 3.2.3 is consistent with Revision 2 of Volumes 1 and 2 of NUREG-1433. Changes being proposed for Bases section B 3.2.1 and TS Section 5.6.5 are simply supportive in nature to the relocation of LHGR from the APLHGR Section Bases B 3.2.1 to the new section LHGR B 3.2.3.

Also, no changes are being proposed to any plant system, structure, or component designed to prevent or mitigate the consequences of a previously evaluated event.

Therefore, because the physical characteristics and performance requirements of the plant systems, structures, and components (including the reactor core and fuel) will not be

altered, the proposed license amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

No plant systems, structures, or components (including the reactor core and fuel) will be altered by the proposed change to the LCO or supporting Bases.

Additionally, this TS change request does not propose changes in the operation of any plant system. Consequently, new and unanalyzed modes of operation are not introduced.

As a result, the possibility of a new or different kind of accident from any previously evaluated is not introduced.

3. The proposed change does not involve a significant reduction in the margin of safety.

Previously, the LHGR was included in the monitoring of the APLHGR. Now, SNC proposes to monitor LHGR on its own while continuing to monitor APLHGR. This proposed TS change adds an LCO for LHGR and a corresponding requirement for the COLR.

The margin of safety is not reduced since the LHGR and APLHGR will continue to be monitored.

Environmental Evaluation

10 CFR 51.22 (c)(9) provides criteria for the categorical exclusion from performing an environmental assessment. A proposed amendment to an operating facility requires no environmental assessment if operation of the facility in accordance with the proposed license amendment will not:

1. Involve a significant hazards consideration;

2. Result in a significant change in the types, or a significant increase in the amounts of any effluents that may be released off-site, or,

3. Result in a significant increase in individual or cumulative occupational radiation exposure.

Southern Nuclear has evaluated the proposed changes and determined that the changes do not involve (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or

(3) a significant increase in the individual or cumulative occupational exposure. Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), and an environmental assessment of the proposed changes is not required.

Enclosure 3
Edwin I. Hatch Nuclear Plant

Marked-up Technical Specifications and Bases Pages

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 AND 24 hours thereafter

[@] This Specification is effective starting from Hatch 1/Cycle 22.

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; ~~and~~

c. LCO 3.2.5, "LINEAR HEAT GENERATION RATE (LHGR)," single loop operation limits specified in the COLR; and

d. ~~c~~. 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.

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.

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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 AND 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; ~~and~~

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

d. ~~c.~~ 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)

format of the 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)

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.

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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)

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 ~~certain fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs) and 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 the fuel design limits are presented in References 1 and 2.~~ The analytical methods and assumptions used in evaluating ~~Design Basis Accidents (DBAs), anticipated operational transients,~~ LOCA and normal operation that determine the APLHGR limits are presented in References 1, ~~2~~, 3, 4, ~~5~~, 6, 9, and ~~7~~. 10

~~Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and certain other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly.~~ APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to 10 CFR 50.46 ~~to fuel design limits~~ during the limiting LOCA ~~AOOs~~ (Refs. ~~5~~, 6, and ~~7~~ 9 and 10). ~~Flow dependent APLHGR limits are determined (Ref. 7) using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, 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, $MAPFAC_p$, are also 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 $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The exposure dependent APLHGR limits are reduced by MAPFAC$_p$ and MAPFAC$_f$ 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 9.~~

LOCA analyses are ~~then~~ 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 ~~its~~ an assumed conservatively small local peaking factor. ~~A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.~~

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum ~~of 0.75 (Ref. 5)~~ 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.

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 ~~fuel design, DBA, and transient~~ LOCA analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the MAPFAC$_p$ and MAPFAC$_f$ factors times the exposure dependent APLHGR limits. 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 APLHGR limit by the smaller of either MAPFAC$_p$, MAPFAC$_f$, and ~~0.75, where 0.75 has been determined by a specific single recirculation loop analysis (Ref. 5)~~ a maximum value allowed during single loop operation as specified in the COLR.

(continued)

BASES (continued)

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA ~~and transient~~ 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 ≤ ~~25~~% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

24

ACTIONS

A.1

LOCA

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the ~~DBA and transient analyses~~ 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 ~~transient or~~ LOCA ~~DBA~~ occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to ~~in a~~ MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to 24 < ~~25~~% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~25~~% RTP in an orderly manner and without challenging plant systems.

24

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

24

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~25~~% 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

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 ≥ ~~25~~ 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. ~~FSAR, Chapter 3.~~ (NOT USED)

3. FSAR, Chapter 6.

4. FSAR, Chapter ~~14~~ 15, Unit 2.

5. ~~NEDO-24205, "E.I. Hatch Nuclear Plant Units 1 and 2 Single Loop Operation," August 1989.~~ (NOT USED)

6. ~~NEDO-24395, "Load Line Limit Analysis," October 1980.~~

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. ~~NEDO-30130-A, "Steady State Nuclear Methods," May 1985.~~ (NOT USED)

9. ~~NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.~~

10. ~~NEDO-31376, "E.I. Hatch Nuclear Plant SAFER/GESTAR-LOCA Analysis," December 1986.~~

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

NEDC-32749P, "Extended Power Uprate Safety Analysis Report for Edwin I. Hatch Units 1 and 2", July, 1997.

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

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

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 accidents (LOCA) do 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 pellet and expansion of the UO_2.

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 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.

Writers note: This Bases section is to be included in both Unit 1 and 2 Technical Specifications Bases

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.

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). Flow dependent LHGR limits are determined (Ref. 5) using the three dimensional BWR simulator code (Ref. 6) to analyze slow flow run out transients. The flow dependent multiplier, $LHGRFAC_f$, 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.

The LHGR satisfies criterion 2 of the NRC policy statement (Ref. 8).

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 other design limits described in Reference 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.

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.

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 judgement. The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational ~~transients~~ (Ref. 2), which are analyzed in Chapter 14 of the FSAR. occurrences (AOOs)

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 APLHGR requirements are modified accordingly (Ref. 3). s

1 and

(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 APLHGR and 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

LHGR limits, (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)")

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. Alternately, 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)"], and APRM Simulated Thermal Power - High setpoint (LCO 3.3.1.1) must be applied to allow continued operation consistent with the assumptions of Reference 3.

1 and

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

or AOO

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 occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

or the AOO analyses

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 sequence.

or AOO

or AOO

~~The 24~~ hour Completion Time is based on the low probability of an accident 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-~~31376~~32220P, "E.I. Hatch Nuclear Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," ~~December 1986~~. March, 1997

2. FSAR, Section 4.3.5.

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.

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 ~~certain fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs) and 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 the fuel design limits are presented in References 1 and 2.~~ The analytical methods and assumptions used in evaluating ~~Design Basis Accidents (DBAs), anticipated operational transients~~ LOCA, and normal operation that determine the APLHGR limits are presented in References 1, ~~2~~, 3, 4, ~~5~~, 6, 9, and ~~7~~. 10

~~Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and certain other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly.~~ APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to 10 CFR 50.46 ~~to fuel design limits~~ during the limiting LOCA ~~AOOs~~ (Refs. ~~2~~, 6, and ~~7~~ 9, 2 and 10). ~~Flow dependent APLHGR limits are determined (Ref. 7) using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, 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, $MAPFAC_p$, are also 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 $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The exposure dependent APLHGR limits are reduced by MAPFAC$_p$ and MAPFAC$_f$ 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 9.~~

LOCA analyses are ~~then~~ 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 ~~its~~ local peaking factor. ~~A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.~~

an assumed conservatively small

(COLR)

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum ~~of 0.75 (Ref. 5).~~ 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.

value specified in the Core Operating Limits Report

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 ~~fuel design, DBA, and transient~~ LOCA analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the MAPFAC$_p$ and MAPFAC$_f$ factors times the exposure dependent APLHGR limits. 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 APLHGR limit by the smaller of either MAPFAC$_p$, MAPFAC$_f$, and ~~0.75, where 0.75 has been determined by a specific single recirculation loop analysis (Ref. 5).~~

a maximum value allowed during single loop operation as specified in the COLR.

(continued)

BASES (continued)

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA ~~and transient~~ 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 ≤ ~~25~~% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required. 24

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the ~~DBA and transient analyses~~ 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 ~~transient or DBA~~ LOCA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to ~~in~~ a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~25~~ 24% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~25~~ 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 ≥ ~~25~~ 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

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 ≥ ~~25~~ 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. ~~FSAR, Chapter 4.~~ (NOT USED)

3. FSAR, Chapter 6.

4. FSAR, Chapter 15.

5. ~~NEDO-24205, "E.I. Hatch Nuclear Plant Units 1 and 2 Single Loop Operation," August 1989.~~ (NOT USED)

6. ~~NEDO-24395, "Load Line Limit Analysis," October 1980.~~

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. ~~NEDO-30130-A, "Steady-State Nuclear Methods," May 1985.~~ (NOT USED)

9. ~~NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.~~

10. ~~NEDO-31376, "E.I. Hatch Nuclear Plant SAFER/GESTAR-LOCA Analysis," December 1986.~~

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

NEDC-32749P, "Extended Power Uprate Safety Analysis Report for Edwin I. Hatch Units 1 and 2", July, 1997

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

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

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 ~~transients~~ (Ref. 2), which are analyzed in Chapter 15 of the FSAR. occurrences (AOOs)

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 APLHGR requirements are modified accordingly (Ref. 3). s

1 and

(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 APLHGR and 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

LHGR limits, (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)")

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. Alternately, 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)"), and APRM Simulated Thermal Power - High setpoint (LCO 3.3.1.1) must be applied to allow continued operation consistent with the assumptions of Reference 3.

1 and 2

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

or AOO

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 occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

or the AOO analyses

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 sequence.

or AOO

or AOO

The 24 hour Completion Time is based on the low probability of an accident 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-~~31376P~~ 32720, "E.I. Hatch Nuclear Plant Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," ~~December 1986~~ 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.

Enclosure 4
Edwin I. Hatch Nuclear Plant

Clean Typed Technical Specifications and Bases Pages

TABLE OF CONTENTS

(continued)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2

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 AND 24 hours thereafter

*This Specification is effective starting from Hatch 1/Cycle 22.

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.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	24 hours
B. Required Action and associated Completion Time of Condition A not met. OR No recirculation loops in operation.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	----------NOTE---------- Not required to be performed until 24 hours after both recirculation loops are in operation. ---------- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours
SR 3.4.1.2	(Not used.)	

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

format of the 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)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (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 NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (applicable amendment specified in the COLR).

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

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

TABLE OF CONTENTS

(continued)

(continued)

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 the various operating core flow and power 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.

For single recirculation loop operation, the MAPFAC 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.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).

(continued)

BASES (continued)

LCO

The APLHGR limits specified in the COLR are the result of the LOCA analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits. 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 APLHGR limit by the smaller of either $MAPFAC_p$, $MAPFAC_f$, and a maximum value allowed during single loop operation as specified in the 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

A.1

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.

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.

(continued)

BASES

ACTIONS

B.1 (continued)

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, Unit 2.

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 GE14," March 2002.

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

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 pellet and expansion of the UO_2.

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.

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.

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). Flow dependent LHGR limits are determined (Ref. 5) using the three dimensional BWR simulator code (Ref. 6) to analyze slow flow runout transients. The flow dependent multiplier, $LHGRFAC_f$, 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.

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

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

(continued)

BASES

LCO (continued)

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.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Coolant Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes more heat from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor Coolant Recirculation System consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one variable speed motor driven recirculation pump, a motor generator (MG) set to control pump speed and associated piping, jet pumps, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core. The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void

(continued)

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 judgement. 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 14 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 APLHGR requirements are modified accordingly (Refs. 1 and 3).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The transient analyses of Chapter 15 of the Unit 2 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 APLHGR and 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. Alternately, 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 4.3.5.

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.

TABLE OF CONTENTS

(continued)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2

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 AND 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.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	24 hours
B. Required Action and associated Completion Time of Condition A not met. OR No recirculation loops in operation.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	----------NOTE---------- Not required to be performed until 24 hours after both recirculation loops are in operation. ---------- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours
SR 3.4.1.2	(Not used.)	

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)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (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 NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (applicable amendment specified in the COLR).

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

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

TABLE OF CONTENTS

(continued)

(continued)

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 the various operating core flow and power 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.

For single recirculation loop operation, the MAPFAC 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.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).

(continued)

BASES (continued)

LCO

The APLHGR limits specified in the COLR are the result of the LOCA analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limits. 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 APLHGR limit by the smaller of either $MAPFAC_p$, $MAPFAC_f$, and a maximum value allowed during single loop operation as specified in the 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

A.1

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.

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.

(continued)

BASES

ACTIONS

B.1 (continued)

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.

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 pellet and expansion of the UO_2.

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.

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.

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). Flow dependent LHGR limits are determined (Ref. 5) using the three dimensional BWR simulator code (Ref. 6) to analyze slow flow runout transients. The flow dependent multiplier, $LHGRFAC_f$, 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.

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

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

(continued)

BASES

LCO (continued)

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.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Coolant Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes more heat from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor Coolant Recirculation System consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one variable speed motor driven recirculation pump, a motor generator (MG) set to control pump speed and associated piping, jet pumps, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core. The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void

(continued)

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 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 APLHGR and 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. Alternately, 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.